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Chapter 3

The Tricolours of Nuclear India – U/Pu/Th Fuel Cycles

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Captions for Photo-Collage

1. *Bhabha on a visit to the Jaduguda mines.*
2. *Exploratory drilling of uranium ores in 1955, at Singbhum thrust belt area.*
3. *View of Lead Mini Cell (LMC) at Kalpakkam - a facility for reprocessing the spent mixed carbide fuel from Fast Breeder Test Reactor.*
4. *Nineteen element fuel bundles manufactured at NFC, Hyderabad for the 235 MWe Pressurised Heavy Water Reactors.*
5. *Underground mine at Jajawal in the state of Madhya Pradesh.*
6. *Reverse osmosis plant at waste immobilisation plant, Trombay*
7. *Sethna seen with the Nobel laureate Prof. Glenn T. Seaborg, celebrated for discovering plutonium. Also seen is the young Raja Ramanna.*

Fuel cycle

One of the most crucial facets of our nuclear power programme, apart from the design and construction of reactors, is the fuel cycle. The nuclear fuel cycle encompasses the entire gamut of activities starting from prospecting and mining of nuclear fuel materials to fabrication of fuel elements, reprocessing of the spent fuel and long term storage and disposal of radioactive wastes. Given the ambitious three-stage programme, developing the technologies necessary for closing the fuel cycle needed inter-disciplinary efforts as well as multi-institutional synergism. To understand the full implications of the closed fuel cycle and its importance, we have to consider our actual resource base and the eventual necessity for reprocessing the spent fuel. In general, the optimal use of all available fertile and fissile inventories, would guarantee the availability of nuclear power for centuries to come.

The responsibility for prospecting and establishing viability of new fuel reserves is vested with the Atomic Minerals Directorate (AMD). This Directorate, which started with just a handful of geologists in the early years of DAE, is now a mature organization with well over 3,000 employees. From its humble beginnings, where the geologists had to make do with simple tools and extremely arduous conditions, currently this organization is at the forefront and has developed advanced prospecting techniques. Using advanced technologies, many of which were developed in-house, this organization has identified many promising resources throughout the length and breadth of the country.

The Uranium Corporation of India Limited (UCIL), a public sector enterprise formed in October 1967 under the Department of Atomic Energy, does the commercial exploitation of the reserves that are identified by AMD. This organization is the only agency that is authorized to mine and to process uranium ores in the country. The processed ore has to be eventually fabricated into nuclear fuel elements, suitable for different types of reactors and the Nuclear Fuels Complex (NFC), Hyderabad is the organization which is responsible for this activity.

One of the unique distinctions of the Indian nuclear fuels programme has been the fabrication of a wide range of fuels. These are metallic alloys of uranium, uranium dioxide, mixed oxides and carbide fuels, which are used for fuelling research

and power reactors in the country. Currently, there are fifteen power reactors in operation, and NFC, which supplies fuel elements to these reactors, has grown many-fold in its scale of operations. In an international scenario of technological denial, the complete set of processes for fuel fabrication has been implemented indigenously. Over a period of time, many challenging problems of fuel fabrication like that of mixed carbide fuel for FBTR and the Al-²³³U plate fuel for the Kamini reactor have been successfully addressed. Thus, the country has demonstrable capability of meeting fuel fabrication needs of all the three stages of the nuclear power programme.

The limited natural uranium resources in India cannot sustain the envisaged growth of its nuclear power. Therefore, reprocessing of the spent fuel to extract the fissile plutonium and ²³³U is a compelling necessity. Such a strategy involves reprocessing of highly radioactive spent fuel using advanced remote handling technologies. Facilities for fuel reprocessing of spent fuel have been established in the country and are in a position to adequately meet the requirements of our nuclear programme.

One of the long-term concerns about nuclear technology has been in the area of satisfactory handling and safe disposal of radioactive waste. Currently, the latest methods of vitrification for the storage of waste are being followed in strict conformity with international guidelines. Research in the area of long term storage and disposal of radioactive waste is also being carried out, so as to pursue promising futuristic trends.

Atomic Minerals Exploration and Processing

Activities related to the prospecting and mining of uranium- and thorium-containing minerals and ores within the country were initiated right at the beginning of the atomic energy programme. The four member Atomic Energy Committee set up in 1945 under the chairmanship of Bhabha included Dr. D.N. Wadia, mineral advisor to the Central Government. Soon after independence, the Government of India took over the sole right to prospect for, mine and process naturally occurring minerals containing prescribed substances such as uranium and others listed in the Atomic Energy Act (1948) or such of those that may be notified from time to time. A rare



D. N. Wadia in active field work in formative years



Geophysical surveys in field

minerals survey unit was established in 1949, which was later brought under the Atomic Energy Commission (AEC) as Raw Materials Division (1954), and subsequently (1958) constituted as the Atomic Minerals Division (AMD). In the Golden Jubilee Year of AEC (1998), AMD was renamed as Atomic Minerals Directorate for exploration and research, reflecting the multifarious functions of the organization today. These activities include the survey, prospecting, and exploration of naturally occurring raw materials containing uranium, thorium, beryllium, niobium, tantalum, zirconium, lithium, titanium, rare earth elements, etc.; research & development on the design and development of instruments and analytical techniques for mineral survey and exploration; geological evaluation of sites for nuclear power plants and radioactive waste disposal repositories, and liaison and collaboration at national and international levels in the relevant areas.

The organization has grown in manpower from 17 earth scientists in 1950 to nearly 3000 scientific, technical and supporting personnel today. With its central head quarters at Hyderabad, the field activities of the Directorate are spread all over India, and these are coordinated from seven regional head quarters, viz., Central Zone (Nagpur), South Central (Hyderabad), North (New Delhi), South (Bangalore), East (Jamshedpur), North-East (Shillong), and West (Jaipur). Besides these, two sectional offices are located at Thiruvananthapuram and Visakhapatnam for the evaluation of heavy mineral resources occurring in the beach sands of the west and east coasts and inland river placers of the country.

Exploration Strategies

In the initial years, only bare minimum field survey equipment supported by radiation meters such as Geiger-Muller counters could be deployed for the exploration work.

New and innovative technologies were steadily inducted in due course. In 1955, airborne radiometric survey techniques, using helicopters with radiation equipment, were introduced. India thus became one of the first few countries to adopt this methodology for narrowing down target areas for uranium exploration. The necessary equipment for this purpose were conceptualized, designed and fabricated in-house. With increasing geological understanding and wider geographical spread in survey work, instruments such as radiation counters



Aircraft used for airborne surveys



Airborne gamma ray spectrometer on board



Ore recovery operation at Chhinjra



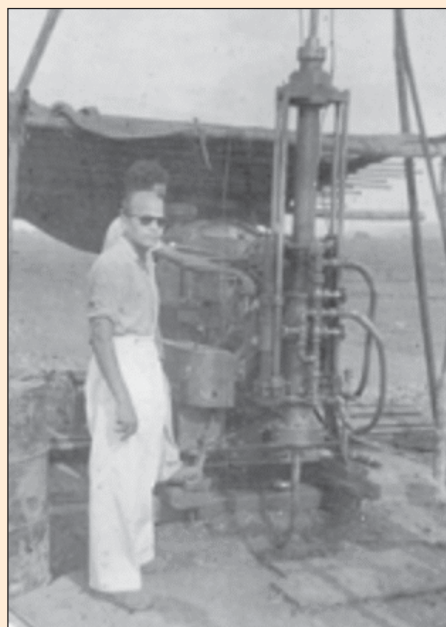
Exploratory drilling at Jajawal



Exploratory drilling in 1955, Singbhum thrust belt area



Underground survey in exploratory mine at Bodal



Bulk ore grade analyzer - scintillation arch at Jajawal



and shielded probe loggers were indigenously fabricated. In order to achieve better maneuverability, a fixed-wing aircraft was used in 1956 for airborne survey covering large areas along the Himalayan foothills.

Today, AMD adopts a spectrum of strategies in the exploration for uranium and other rare and strategic minerals. Regional reconnaissance is done by remote sensing using satellite and aircraft-based methods such as airborne gamma ray spectrometry, and geochemical and geophysical surveys,

for selecting geologically and structurally favourable target areas. Potential and promising areas of mineralization are delineated by detailed and semi-detailed surveys, geological mapping, sampling, trenching, pitting, and shielded probe logging. Reconnaissance, exploratory, and evaluation drilling are carried out to demarcate the ore body. Exploratory mining is taken up to confirm the sub-surface behaviour of the ore body as revealed by the drilling data. Bulk ore sampling is made for pilot scale leachability studies. Studies on rock mechanics are also performed for selecting suitable mining methods before commercial mining is eventually taken up.

“...World’s richest uranium deposits have so far been identified in Canada and Australia, with grades as high as 21 % and are known as the “Unconformity-type” which has specific age. Another significant type, established so far only in Australia, is known as “Breccia-Complex” which contains huge resources of copper and uranium (~ 1 million tonnes) with gold, silver and Rare Earth Elements (REE) association.

India is endowed with fourteen Middle-Proterozoic basins, which can host the “unconformity-type” and has the geological set-ups to host the “Breccia-Complex” type uranium deposits.

Both these types are concealed at great depths and can be deciphered by a combination of geophysical surveys followed by precise zero-deviation drilling.

Thus even-though we have identified highly potential geological set-ups in India to host both these major types of uranium deposits (which contribute to 50% of present world resources), harnessing them is constrained by non-availability of geophysical and precise drilling inputs within the country.

Our vision and dream would, therefore, be to develop these geophysical requirements indigenously and develop inputs for precision drilling and make India not only self-sufficient but a surplus country with regard to uranium.

The task is gigantic but can be achieved with pragmatic approach and dedicated team effort ...”

- R.M. Sinha
Director, AMD

Infrastructural Facilities

Mineral survey and exploration activities are supported by modern laboratory facilities for analyzing the large number of samples generated during various stages of exploration. The techniques that are routinely employed include x-ray diffraction, X-ray fluorescence spectrometry, atomic emission and absorption spectrometry, neutron activation analysis, mass spectrometry, etc. The eighties saw phenomenal growth in infrastructure and R&D. By then, full-fledged programmes covering geochronology, aerial survey, mineral technology, geophysics, and beach sand investigations had evolved. In the early years, field geologists used to carry geochemical kits for on-site investigation of metals and their chemical state in minerals. In the 1980s, this was improved by inducting geochemical vans fitted with equipment to delineate the percentage distribution of elements and their chemical states. These tools helped considerably in narrowing down the target areas for exploration, as well as in identifying pathfinder trace elements for locating concealed uranium mineralization. Measurements down to sub-ppm levels of concentration became possible. Geobotanical investigations yielded results with the identification of uranium-indicator plant species, especially so at Domiasiat in Meghalaya. Additional equipment were added to aid the understanding of petro-mineralogical



Geochemical mobile van operating in field



Calibration pad at Nagpur Airport

characteristics of rock species. In order to understand the paragenetic sequence of ore minerals and the temperature of ore-forming fluids, fluid-inclusion studies were initiated. A new facility was created at AMD, Nagpur, for the calibration of Airborne Gamma Ray Spectrometers (AGRS) used for the exploration of radioactive minerals and oil. This facility, which is of international standards, is the only such unit available in South East Asia. It is also being used extensively by national agencies like the Geological Survey of India, National Geophysical Research Institute and some universities. AMD has pioneered the AGRS technology, and undertaken airborne surveys for ONGC, identifying hydrocarbon deposits in the Krishna, Godavari and South Rewa basins.



Underground development at Jajawal mines



Uranium indicator plant (elastostemma rupsetre weed) in Khasi Hills in Meghalaya

Discovery of Uranium Deposits

Survey and exploration for atomic minerals had started in the copper belts of the Singbhum Shear Zone (SSZ), Jharkhand, the polysulphide mineral regions of Rajasthan, and the mica-pegmatite belts of Jharkhand, Rajasthan and Andhra Pradesh. The first uranium deposit was discovered in SSZ at Jaduguda in 1951. By the end of the seventies, more of uranium deposits were discovered along the 120 km SSZ. Among these, the ore deposits at Bhatin, Narwapahar, Turamdih, Banduhurang, Central Keruadungri, Bagjata, Kanyaluka, Mohuldih and Nandup have been found to be viable for commercial exploitation. The tailings from the copper-bearing lodes of Mosabani, Surda and Roam-Rakha in this belt were found to contain additional uranium resources as a by-product. The ore deposits at Jaduguda and Narwapahar were handed over to the Uranium Corporation of India Ltd., (UCIL) for commercial exploitation in 1966 and 1970, respectively. Similarly, Bhatin



and Turamdih (East) were handed over in 1979. The Banduhurang and Bagjata deposits will be opened up soon for commercial mining. Intensive exploration activities at the Bodal uranium deposits (Madhya Pradesh) by drilling and developmental mining established the existence of substantial ore body there, and this was handed over to UCIL in 1985.

Sustained investigations on the basis of conceptual geological modelling using integrated multi-disciplinary techniques have resulted in the discovery of many uranium deposits in different parts of the country. Uranium deposits have been identified in Karnataka, Andhra Pradesh, Meghalaya and Himachal Pradesh. The thrust areas currently under exploration include the Bhima basin, Karnataka; the Cuddapah basin, Andhra Pradesh; the Upper Cretaceous Mahadek formation, Meghalaya; and the Albitite Zones, Rajasthan. The potential basins include the Chhattisgarh basin, and the Chhattisgarh and Gwalior basin, Madhya Pradesh. Over the last five decades, a total of 94,000 tonnes of U_3O_8 have been identified. By 1993, the Domisiat (Meghalaya) uranium deposit was established. Exploratory mining was carried out there to excavate bulk ore for mineral beneficiation, and yellow cake (magnesium

diuranare) was recovered on a pilot plant scale following AERB guidelines. Over 600 kg of yellow cake was produced as a part of technology demonstration. As the country is bestowed with many low-grade and medium-tonnage uranium deposits, it was considered appropriate to undertake technology demonstration of uranium leaching from large-quantity ore bodies. Accordingly, heap leaching was carried out on a heap of 1000 tonnes of ore by exploratory mining at Jajawal in the Surguja District of Chhattisgarh. The experiment, initiated in 1996, successfully demonstrated the capability to undertake such operations to recover yellow cake on a large scale.

Exploration for other strategic minerals containing niobium, tantalum, beryllium, lithium and rare earth metals (all of them important in the nuclear energy programme) has also been carried out in various parts of India, and substantial deposits identified.

Beach Sand and Offshore Investigations for Strategic Minerals

Beach sands of India contain rich deposits of monazite (thorium ore), ilmenite, zircon, etc. AMD has surveyed 2088 km of the 6000 km of coastal length and established a number of valuable mineral deposits along the coastal areas of Maharashtra, Kerala, Tamil Nadu, Andhra Pradesh, Orissa and West Bengal. The total minerals established so far include about 8 million tonnes of monazite.



A Dredge and Wet Concentrator in operation at Orissa Sands Complex (OSCOM)

Mining and Processing of Uranium Ores

The Uranium Corporation of India Limited (UCIL), a public sector enterprise, was formed in October 1967, as the sole agency to mine and process uranium ore. Today it is at the forefront of the nuclear fuel cycle, fulfilling the requirement of uranium for the Pressurised Heavy Water Reactors. With four underground uranium mines, a 2090 tonnes per day (TPD) processing plant and some new mines on the anvil, UCIL plays a very significant role in India's nuclear power generation programme. Headquartered at the Jaduguda Mines in East Singhbhum in the Singhbhum Thrust Belt, UCIL has adopted state-of-the-art technology for its mines and process plant. This, coupled with the in-house expertise of a team of dedicated professionals, makes UCIL a company with a mission.

All the four uranium deposits under active mining, i.e. Jaduguda, Bhatin, Narwapahar and Turamdih, are located in the mineral rich Singhbhum Thrust Belt in Jharkhand. Uranium here occurs with copper. It exists as finely disseminated particles within the schistose host rocks. Uranium mines are put up where the dissemination is sufficiently concentrated for viable extraction of the ore. Visually there is no distinction between the ore and the surrounding host rock. Identification of the ore underground, and its grade control are carried out by physical methods. Geiger-Muller counters are employed to guide the excavations underground, and scintillation probes are used for grade control.

Uranium Mines in India Jaduguda Mines

The Jaduguda mine has the distinction of being the first uranium mine in the country. Mining operations began in 1967, with the commissioning of a shaft with tower mounted friction winder, a technical milestone for the mining industry in India. The mine is accessed by a 5 m diameter vertical shaft with a total depth of 640 m. The shaft is concrete-lined throughout, and has a cage and a skip with counterweights. The cage accommodates 50 persons and the skip has a capacity of hoisting 5 tonnes of ore at a time. This shaft is also the main ventilation intake, besides providing service lines such as compressed air and water pipe lines, communication and power cables, etc. The mine is well ventilated by boundary ventilation layout. Horizontal cut-and-fill method is followed for stoping. The ore is transferred to the adjacent process plant by

“...To appreciate the work done at Jaduguda, one has to remember what was the state in the early sixties - when we had done the designing. Here one must appreciate that not only the technology was new to us and had to be developed from scratch, but the size of the mill (to treat 1000 ton/day) was so large that even with established technology it would have been a challenging task...No one had yet done anything on such a scale on their own. Here in Jaduguda, it was not a question of physical beneficiation with acid leaching. One had to reckon with corrosion problems. The entire equipment had to be rubber lined – the tanks and the piping. The rubber lining was done at the site. It was being done on such a scale for the first time...”

- S. Fareeduddin,
Project Director, HWB (1969-80)



Uranium mine at Jaduguda

a conveyor and the mill tailings are used as fill. The main shaft caters up to a depth of 555 m, and an auxiliary shaft going up to 905 m depth was sunk to mine ore from deeper levels. Jaduguda mine has many firsts in terms of technology development and absorption. It has created a large skill base for the mining industry in general and uranium mining in particular.

Bhatin Mines

The Bhatin mine is at a distance of 5 km from Jaduguda and shares much of the infrastructure of the latter. Mining of this small deposit illustrates the country's commitment to optimally utilize its scarce uranium resources. This mine was commissioned in 1986 and it has been planned to operate it up to a depth of 250 m.

Narwapahar Mines

Narwapahar mine is another of UCIL's operating mines and was commissioned in April 1995. This is the most modern mine in the country, with a decline access to the underground and ramp accesses to the stopes. This permits the use of large diesel-powered equipment underground, giving a good working



Narwapahar uranium mine

Bulk uranium ore analyzer installed at Narwapahar uranium mines



environment that eliminates worker fatigue, thereby promoting high productivity.

Turamdih Mine

The Turamdih mine is about 23 km west of Jaduguda; the ore bodies here are lenticular in shape, occurring as several discrete lenses. This project was inaugurated in November 2002 and the work is proceeding ahead of schedule.

Diesel traction and electro-hydraulic drill jumbos are used for drilling in the mines. This ensures mobility of equipment and efficiency of electric power. The introduction of drill jumbos underground has reduced the physical strain of drilling, long considered to be the hardest work of miners. The longer depth



Mechanised ore drilling at Jaduguda

of round holes, and capability to drill parallel holes have given rise to a drilling productivity unparalleled in the Indian mining industry.

All mucking of blasted ore and waste is done by diesel-powered load-haul-dump units and low-profile-dump-trucks. In order to utilize the high productive capacities of these machines, adequate backup in terms of service vehicles such as passenger carriers, explosive vans, service trucks and crane trucks have been deployed, making the uranium mines unique in India.

Processing of Uranium Ores

The ore mined from the Narwapahar, Jaduguda, Bhatin and Turamdih mines is processed in the centralised processing plant

located close to the Jaduguda mines. Uranium is extracted from the ore in the Jaduguda mill by a hydro-metallurgical process. This mill has been expanded twice since its commissioning in 1968 to process additional ore from Bhatin and Narwapahar mines. The original installed capacity of the mill was 1000 ton/day, subsequently enhanced to 1300 ton/day; and further expanded to 2090 ton/day in 1996. Planning and implementation of the capacity expansion was carried out with in-house expertise. Specific emphasis was laid on adoption of new technology, automation and instrumentation, for better environmental control.

In the extraction process, after three stages of crushing, the crushed ore undergoes a two-stage wet grinding. The slurry thus obtained is pumped to leaching pachucas for dissolution of uranium in the liquor. The leached slurry is filtered to obtain a uranium-rich liquor. This is purified and concentrated by employing ion-exchange resins. Uranium is then precipitated from the concentrated liquor as magnesium diuranate, commercially known as the yellow cake. This is thickened, washed, filtered, dried in the spray drying plant and packed in drums; thereafter it is transported to the NFC for further processing and fabrication into fuel elements.

Two types of wastes are generated in the uranium ore processing. They are the leach liquor depleted in uranium after ion-exchange recovery of uranium, and the filtered solids depleted in uranium after filtration of the leached slurry. Both are neutralised with limestone and lime slurry to precipitate the remaining radionuclides along with heavy metals like manganese, iron, copper etc. The neutralised slurry is classified and the coarse fractions are pumped back to the mines for back filling of the voids. The fine particles are pumped to the tailing pond, where the slime settles, and the clear liquor is sent to the effluent treatment plant for further processing. The tailing pond is a well-engineered containment having an earthen dam on one side with the other three sides protected by hills.

A composite scheme has been implemented for the reclamation of water, and the treatment of effluents so as to make the final effluent discharges environmentally benign. Mine water from all mines is collected, clarified and reused in the ore processing plant. The tailing pond effluent is clarified, and a part of this is sent to ore processing plant for reuse, and the rest is re-treated with barium chloride and lime. This is clarified,

the settled precipitates are sent back to the tailing pond, and the clear effluent is monitored; it is generally clear enough to be discharged to the environment.

A full-fledged control, research & development laboratory with sound infrastructure, monitors the process parameters for the recovery of uranium and the byproducts from the ore. A technology demonstration pilot plant has been commissioned in November 2002 at the Jaduguda mill premises. This plant has a comprehensive facility for uranium ore processing - mechanized crusher house, grinding classification circuit, acid / alkali leaching tanks, ion-exchange / solvent extraction facilities, tailing neutralization and disposal system, and provision to study the recovery of by-products. The pilot plant studies are required for generation of process engineering parameters, which would help in the design and scale-up of industrial plants. With the commissioning of this pilot plant, it is possible to optimize various process parameters and establish techno-economic feasibility of the processes.

Mineral conservation has been the watchword in mining and processing activities. Sustained research and development efforts have gone in to recover small quantities of Cu, Ni and Mo sulphides present in the Jaduguda ore. In order to recover the magnetite contained in the ore, a separate plant was commissioned in 1974. The magnetite recovered is supplied to coal washeries for beneficiation of coal.

Radiological and Environmental Safety

A health physics unit and an environmental survey laboratory carry out the in-plant and environmental surveillance of all the UCIL units to evaluate and ensure overall safety in accordance with the standards prescribed by the national and international regulatory bodies like the Atomic Energy Regulatory Board (AERB) and the International Commission on Radiological Protection (ICRP). Monitoring of radioactivity and radiation levels in different matrices in the mine, mill and the surrounding environment is done at regular intervals. Based on the data obtained, corrective measures are recommended as and when required.

During its 36 years of operation, the UCIL has always shown utmost concern towards the safety of the people and the environment. The safety standards followed at all the operating units of the Corporation are the best amongst the comparable indus-



Uranium recovery pilot plant at Domiasiat, Meghalaya



Radioactive exposure along unconformity contact at Chitrial, Nalgonda district, A.P.

tries. The Corporation has successfully obtained ISO-9002 and ISO-14001 certification for its excellent work practices and environmental protection measures followed at its different operating units.

New Projects

Plans are afoot to expand the mining and processing activities in the Singhbhum East District, Jharkhand. The large uranium deposit at Banduhurang is located close to the Turamdih mine, and the ore body occurs near to the surface. A large opencast mine has been planned in this area. An ore processing plant at Turamdih is also on the anvil to meet requirements of the Turamdih and Banduhurang mines. An underground uranium mine has been planned at Bagjata. The Corporation has also taken up selected sites in other parts of the country for the



R. Chidambaram, then Chairman, AEC, examining yellow cake in Domiasiat, Meghalaya

“...The future of DAE shall be shaped and conditioned by the forces of transformation from the developmental aspirations of growing national population.

The vision of DAE to contribute substantially to India’s programme based on cleaner nuclear energy source will have a bearing on all round development fortified by the application of nuclear science and technology.

The department’s domination in the field of radiation technology shall place it as a unique contributor towards the rural economy. Application of this technology in the genetic improvement of crop plants with respect to yield, quality and tolerance to stress conditions shall facilitate upsurge in agricultural output. Value addition to agricultural products through irradiation shall help in preservation and export.

The Department, with pioneering research and innovative technology shall be the world leader for application of nuclear science towards societal development. Exploitation of uranium resources is, thus an inevitable imperative at the present juncture. ...”

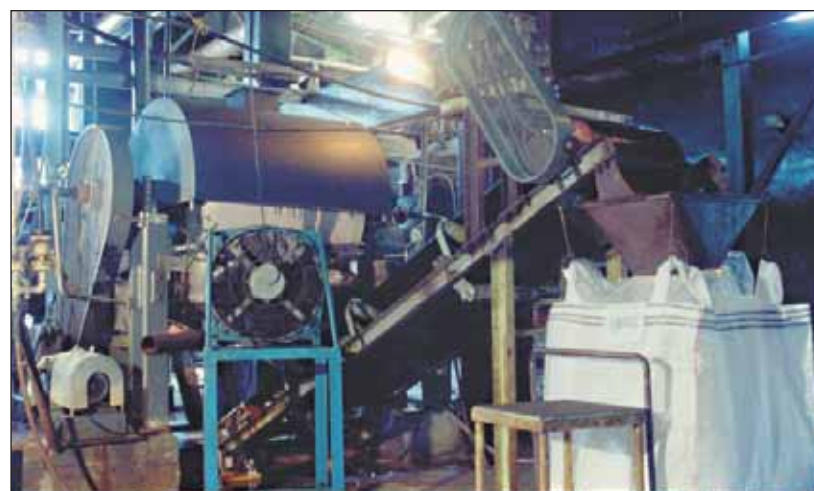
- R. Gupta,

*Chairman and Managing Director,
Uranium Corporation of India Ltd. (UCIL)*

construction of uranium mines and processing plants. These include an underground uranium mine and an ore processing plant at Lambapur-Peddagattu in Andhra Pradesh, an opencast mine and ore processing plant at Domiasiat in Meghalaya, an underground uranium mine at Gogi in Karnataka, and at Rohil-Ghateswar in Rajasthan.

Thorium Mining and Processing

One of the first acts of the Atomic Energy Commission after its formation in 1948 was to stop the free export of monazite, a rich source of thorium and rare earth elements. A committee was constituted to examine the possibility of setting up a facility to process monazite for the production of thorium and rare earths. In 1949, an agreement was signed with a French agency, Societe de Produits Chimiques des Terres Rares, for providing technical know-how and assistance in setting up a factory for



Rare Earths chloride production plant; crude thorium hydroxide with traces of uranium is the by-product

this purpose. Indian Rare Earths Ltd (IRE) was registered as a private limited company jointly owned by the Government of India and the then Government of Travancore-Cochin. Later IRE became a fullfledged Central Government undertaking under DAE. Construction of the plant was started near Alwaye in Kerala in April 1951, and was inaugurated by Prime Minister Jawaharlal Nehru in December 1952.

Monazite, which contains about 9% ThO_2 and 0.35% U_3O_8 , is a phosphate of thorium, uranium and rare earth elements. The raw material is ground and digested with caustic soda to convert it into the respective metal hydroxides and the phosphate as trisodium phosphate. The water-soluble solution

of trisodium phosphate is separated from the mixed thorium and rare earths by leaching. The trisodium phosphate is crystallized out of the solution, centrifuged and dried. The mixed thorium and rare earth hydroxides are then subjected to filtration and washing to remove traces of phosphate. The resulting cake is made into a thick slurry and commercial hydrochloric acid is added under controlled conditions to attain a stable pH of 3 at 333-343 K in rubber-lined steel tanks. Most of the rare earth elements react under this condition to form the corresponding chloride solution, which is then removed. The thorium hydroxide cake is left behind. Monazite produced at Chavara and Manavalakurichi is also processed at the Alwaye plant. The capacity of monazite processing was augmented in stages to 4200 tonnes per year. A thorium plant had been set up at Trombay to process the uranium-thorium concentrate for producing thorium nitrate and recovery of uranium as crude uranium tetrafluoride. IRE was operating the plant on behalf of DAE. This plant was closed during 1997-98. A new thorium plant has been established at the Orissa Sands Complex (OSCOM) at Chatrapur.

Recovery of Uranium from Secondary Resources

With 15 power reactors presently operating with high reliability in different parts of the country, and with many more at advanced stages of planning and execution, the urgent need of producing adequate quantities of uranium is obvious. While the UCIL is enlarging its operations to meet this requirement, several interim measures are also needed in the near term. This scenario has prompted the Department to start exploiting all secondary resources of uranium, through its oldest public sector unit, the Indian Rare Earths Ltd (IREL). IREL has taken up the task of recovering uranium present in monazite and the semi-finished products generated in monazite processing, such as crude thorium hydroxide and uranium fluoride concentrates. Also, it has taken on the responsibility of recovering uranium from phosphoric acid being produced / procured in large quantities by a number of fertilizer companies in India. The recovery process in all cases involves solvent extraction technology with novel combination of indigenous reagents.

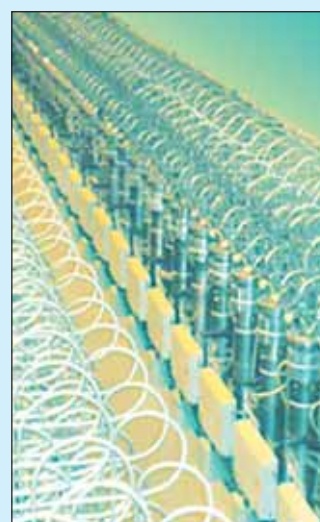
Technology for Uranium Enrichment

Initial feasibility studies were carried out on various

technologies like gaseous diffusion, separation nozzle and high speed rotor technology. Based on the development potential and low energy consumption, high speed rotor technology was selected as a technology demonstration unit. Successful deployment of high speed rotor technology depended upon the development of a host of difficult technologies like production and handling of elemental fluorine and hexafluoride gas, running of high speed sub-critical and super-critical rotors, special materials, development of advanced bearings, intricate cascading techniques and designing of complex vacuum systems. These tasks were made more challenging because of technology denial. A hundred percent indigenously developed high speed gas centrifuge technology was developed and the indigenous capabilities in this field were established.

Developmental efforts over the period of time have led to the creation of expertise in the entire spectrum of technologies including production and handling of cascade grade UF₆, conversion and safe storage of depleted material, electroless nickel plating and passivation for materials of construction, decontamination and recycling of components etc.

The experience gained in operating continuously the cascades of high speed machines both in series and parallel arrangements has demonstrated the maturity in this complex technology. Continuous development is pursued with a view to upgrading design for significant improvement in separative capacity.



Cascade hall of high speed rotor machines



Schematic of high speed rotor

FUEL FABRICATION

Fifty years ago while embarking on the journey to produce nuclear power, a very strong foundation was laid for indigenous development and manufacture of nuclear fuels and reactor components. The emphasis was on self-reliance in this advanced technological sector from the very beginning of the programme. That the lessons in this technology were learnt quickly was seen, when after producing the first nuclear pure ingot of uranium on 19th January 1959, over half the initial core of metallic uranium fuel for the Canada India Reactor (CIRUS) was fabricated in 1960. The performance of the fuel was as



Prime Minister Pandit Jawaharlal Nehru on a visit to Atomic Fuels Division, BARC with Bhabha and Brahm Prakash in the sixties

good as those supplied by Canada, our consultants at that time. Work on the manufacture of uranium dioxide as power reactor fuel began in the early sixties, about the same time when in other countries, like the USA and Canada, it became clear that uranium metal, with its poor thermal cycling and irradiation behaviour and high chemical reactivity, was not suitable as a fuel for power reactors. In fact, when ZERLINA needed a new core in 1965, two tonnes of uranium dioxide fuel elements in 19 rod cluster type elements were fabricated for this reactor. The confidence gained during this fabrication of nuclear fuel elements led to a decision of making half the initial core for the first Pressurized Heavy Water Reactor (PHWR) at Rajasthan. This technology development led to the creation and setting up of a full fledged facility for the production of uranium dioxide

based fuels as well as the reactor components at the Nuclear Fuel Complex (NFC), Hyderabad in the early seventies.

Simultaneously efforts were in progress, during the early sixties and seventies, for a technological leap, to develop and fabricate the more complex next generation fuel, required for the second stage of our nuclear program viz., plutonium based fuels. The facility to develop and fabricate plutonium based ceramic, alloy and dispersion fuels, was set up for the first time in 1972 for PURNIMA-1, which used pure plutonium oxide fuel. With this experience, a facility to fabricate uranium-plutonium mixed oxide for both thermal and fast reactors was set-up at Trombay. Technology for both low enriched as well as high enriched plutonium oxide fuels were developed in this facility. However, sometime around the year 1980, techno-economic problems related to compact fast reactor cores, dictated that we enter the highly complex and frontier technology arena in nuclear fuels – fabrication of plutonium-uranium mixed carbide fuel for the Fast Breeder Test Reactor (FBTR) at Kalpakkam. This unique fuel was successfully fabricated for the entire core of the FBTR and has successfully seen a burn-up of over 147 GWd/t, without any fuel failure. Another high-point has been the successful fabrication of uranium-plutonium mixed-oxide fuels as an alternate for the enriched uranium fuel for the BWR at Tarapur and as a resource conservation measure for the natural uranium fuelled PHWRs. In addition to these fuels various alloy fuels and systems were developed for research reactors and neutron source applications such as aluminum-plutonium, plutonium-beryllium etc., in these laboratories.

The advent of the Advanced Heavy Water Reactor (AHWR) signals an important beginning towards utilization of the vast thorium reserves in the country for harnessing nuclear energy. The technology for fabrication of fuels for reactors such as AHWR incorporating the fissile isotope ²³³U is complicated by the associated isotope ²³²U whose daughter products emit high energy gamma radiation on decay. However, a good amount of development has already been achieved in the manufacture of these fuels.

Nuclear fuel fabrication experience, expertise and technological skills, gathered over the last fifty years, has a strong footing in self-reliance and indigenous development, not only of physical, chemical and metallurgical processes, but also on those of equipment and engineering systems.

Production of Uranium Metal

The decision to put up Uranium Metal Plant (UMP) including a uranium refinery, to purify crude uranium fluoride obtained as a by-product of the thorium plant at Trombay and to produce nuclear grade uranium metal was taken at the end of 1956. The task was assigned to the "Project Fire-Wood" Group, with the main objective of producing metal fuels for CIRUS and ZERLINA reactors. Besides, it was also to function as a pilot plant to collect operational data and to train persons for larger plants to be set up in future. In early 1957 the design and layout work was initiated; equipment fabrication began in August 1957; details of the building were finalised by end of October 1957 and civil contract for building was awarded in the first week of December 1957. Thus, uranium metal plant, the first production unit of the DAE was set up. With the charge of crude uranium fluoride by-product from the Trombay thorium plant, the first metal ingot weighing 44 kg was produced on January 30, 1959.

The plant was set up at a total capital cost of Rs. 23 lakhs within a record period of 2 years after the conception of the idea. It is worth mentioning that along with various processes involved in the production flowsheet, this plant introduced the successful application of solvent extraction in the atomic energy programme in India. Tributyl phosphate diluted with purified kerosene to 30-35% was selected for the extraction process on the basis of optimization studies carried out in a pulsed packed column. Process modifications such as solvent scrubbing after extraction were introduced to avoid solvent entrainment problems. The stripping trials were conducted in a spray column with de-ionized water. Design data such as HTU values and flooding velocities were experimentally determined based on various process and equipment parameters.

UMP operated continuously from 1959 to 1980 and later underwent an expansion to meet fuel demands of DHRUVA. The expansion started in 1977, and was completed by 1984. While retaining the old process flowsheet, changes were implemented to improve the process technology and economy, such as utilization of mixer-settlers and later slurry extractors for solvent extraction as an alternative to pulsed perforated column, which led to a higher degree of purification and ease of slurry handling. Also the use of magnesium for the reduction of uranium tetrafluoride instead of calcium, rendered the whole process indigenous and economic. The process has been

scaled up with suitable instrumentation and control system for the production of 500 kg ingots.

There are several "first" credits to UMP. By extensive R&D, UMP developed a flowsheet for large scale leaching of the uranium ore (1000 tonne per day) and this facility was in operation for a few decades prior to UCIL operations. A process was developed to concentrate uranium from the "green cake" by-product obtained from the thorium processing plant. UMP was also the first to develop process conditions for the production of "ceramic grade UO_2 ", in collaboration with AFD. The oxide for the first half core of RAPS was supplied and this facility became the forerunner for the NFC uranium oxide plant. The enriched uranium oxide plant at NFC was started with the pilot plant trials at UMP, which proved to be a huge foreign exchange saving exercise.

Metallic Fuel Fabrication for Research Reactors CIRUS and DHRUVA

In 1964, when CIRUS became critical, half of its charge came through indigenous efforts. Over the last 40 years, pioneering efforts have been made in the field of fabrication of metallic fuel and related sub-assemblies. Besides, significant changes have been effected, not only to enhance the production rate but more importantly also to enhance the quality of the fuel being produced. In the seventies and eighties, fuel clusters of different designs were produced in conformity with the initial design verification trials. Thus it provided significant inputs for reaching the most optimum design for the fuel and core components of DHRUVA.

Metallic fuel fabrication flow-sheet includes vacuum induction melting and casting, high stage rolling / extrusion, β -heat-treatment and machining of uranium rods with subsequent cladding in aluminium finned tubes, seal welding and assembly. The quality control steps include ultrasonic testing of uranium billets; pressure testing of finned tubes; eddy current testing of fin tubes, machined uranium rods and fuel pins after cladding, radiography of welds and glycol leak testing of fuel pins.

For DHRUVA fuel production, in addition to rolling, a copper tube jacketed extrusion of the beta heat-treated uranium billet is also used. The jacketed extrusion route has resulted in an increase in the production rate and higher yield of uranium with significant reduction in the air activity. Rolling is still adopted

for CIRUS fuel elements but in two stages instead of three stages, used earlier. An on-line system for beta heat-treatment of uranium rods is on the anvil.

One of the major achievements in the fabrication of fuel for DHRUVA is the modification of the cumbersome steps for the assembly of the fuel pin. This was being carried out in the vertical position at an elevation of 4 meters from the ground, taking two shifts to make one assembly. The development of a horizontal fixture with split bushes and rollers facilitated assembly of 5 - 6 clusters in a single shift.

Blistering, a major cause of rejection of fuel pins, was avoided by eliminating the pickling operation, a major source of moisture and introducing in its place caustic soda cleaning with careful study of the growth rate of oxide over uranium rod and preheating the cladding tubes with dehumidification of the assembly area.

Indigenous suppliers have been developed for the boron controlled aluminium fin tubes for both CIRUS and DHRUVA. Die cast Tie plates for DHRUVA fuel assembly hitherto imported were economically produced indigenously using extrusion followed by spark erosion technique. Procedures were evolved for safe handling and storage of the pyrophoric uranium scrap such as chips, turnings and grinding waste powder.

Production of PHWR and BWR Fuels at NFC

During 1963-64, it was realized that the fuel requirement for the power programme required industrial scale of operation and the area in Trombay was not sufficient. A chain of events led to the shifting of the plant outside Trombay. A team, constituted to visit Bangalore, Madras and Hyderabad to search for a suitable site, recommended Hyderabad as an appropriate location. The Andhra Pradesh government came out first with a gift of land of 1000 acres along with water and power for setting up the industry in response to a letter from Dr. Bhabha. Dr. Bhabha came to Hyderabad and went around the places, Jeedimetla, Ramachandrapuram and Moula Ali, suggested by the local government. Dr. Bhabha finally selected Moula Ali. NFC was established to cater to the requirements of fuel and zirconium alloy hardware for all the power reactors in the country. Along with the fuel and zirconium alloy plants, a special materials plant was also set up for producing ultra high purity materials. Other facilities such as, quality control, tool room and workshop,



Handing over of 1,00,000th PHWR Fuel Bundle to CMD, NPCIL on 25-1-1994



Raja Ramanna presiding over the ceremony to mark completion of 2 lakh PHWR fuel bundles at NFC, Hyderabad



Handing over of 3,00,000th PHWR Fuel Bundle to CMD, NPCIL on 20.6.04

effluent management unit etc., were established as the production picked up. Later on, to utilise the expertise available and extra capacities of some of the equipment, production of seamless tubes of special materials was also taken up with augmentation of some more facilities. NFC was awarded ISO-9002 certification in December, 2000 and ISO-9001 in January, 2004. Currently NFC is setting up new expansion projects for increasing its capacity towards fuel and zirconium alloy hardware manufacture.

In the early seventies, it was quite an effort learning to make the highly complex nuclear fuel bundles on an industrial scale for the first time. From 1074 PHWR fuel bundles that were made during the year 1973-74, NFC has leaped to a figure of 30048 PHWR bundles in the year 2002-03. The amazing thirty-fold increase in production involved, not only drastic improvements in yields and recoveries but also lower consumption of raw materials and consumables with consequent reduction in the man-hours and kilowatt-hours per bundle.

It will be difficult to believe that what was once a completely isolated area and waste land now stands transformed into a symbol of the self-sufficiency that India has achieved in the production of nuclear fuel and various reactor components.

Technology denial by the advanced countries turned out to be a blessing in disguise for NFC, which developed the necessary technologies itself and also has gone a step ahead of the developed world. The innovations and improvements



Demonstration of indigenous technological capability - FBTR Fuel sub-assemblies with cut sections

carried out in the production of uranium dioxide to have consistent product quality and increased productivity include:

- Design and development of a slurry extractor in the place of pulsed column for uranium extraction, eliminating four re-pulping and three filtration steps.
- Batch precipitator in place of continuous precipitator-ADU route.
- Turbo drier and spray drier in place of band drier.
- Rotary tubular furnace in place of tunnel furnace.
- Elimination of blending due to achievement of perfect powder homogeneity.
- Integration of pre-compaction and granulation with powder transfer system.
- Use of admixed lubricant in place of die wall lubricant in pelletising.
- Incorporation of acetone dip testing to check compact integrity.
- Roll compaction in place of precompaction and granulation.
- Elimination of pickling of fuel elements and consequently the elimination of sample autoclaving of bundles.
- Graphite coating of fuel clad tubes to inhibit fuel-clad interaction during reactor power ramps.

The FBR component facility at NFC fabricated and supplied various types of sub-assemblies, such as the axial and radial blanket, nickel reflector and steel reflector, for FBTR at Kalpakkam. NFC also supplied the hardware, such as fuel tubes, plugs, springs etc needed for the fuel sub-assemblies.

A full fledged thorium oxide pelletising plant at NFC supplied the full requirement of thoria pellets for FBTR Blanket sub-assemblies, the LOTUS experiment, CIRUS, DHRUVA reactors and thoria assemblies in place of depleted uranium assemblies for PHWR start up. This Plant is now engaged in the fabrication development for PFBR subassemblies. Some of the assemblies required a band of chromium plating. Efforts to locate a suitable commercial electroplater over a year proved futile as the stringent specification of hardness, uniformity of thickness and microstructure could not be met. So, in-house electroplating facility was set up to meet the necessary technical specifications. Development work is also on for

Nuclear Fuel Complex — A multifaceted fuel fabrication facility

“ Nuclear Fuel Complex (NFC) is an unique facility created for manufacture of natural uranium fuel assemblies for PHWRs and enriched uranium fuel assemblies for BWRs, reactor core structurals in zirconium alloys for PHWRs and BWRs and fuel and blanket sub-assemblies for FBRs, thus catering to the requirements of all the power reactors in the country. NFC is unique as it is the only manufacturing facility to house all the production activities – right from ore concentrates to the finished products in both uranium and zirconium streams – under one roof, unlike any other facility in the world. Also installed in NFC are the excellent manufacturing facilities for seamless tubes in a variety of stainless steels, titanium alloys, high nickel alloys etc., catering to critical applications in strategic industries other than atomic energy like defence, aerospace, aeronautical etc. NFC has also to its credit, development and production of very high purity materials of tantalum, niobium, gold, indium, etc.

In NFC there is a continuous up-gradation of technologies for improving the operational efficiency and product quality. NFC carries out in-house, production oriented and product-oriented developmental activities. In addition, NFC developed indigenous capabilities for equipment building. In the context of restrictions of some foreign countries to supply equipment to DAE units, many critical equipment like vacuum annealing furnaces, high temperature sintering furnaces, appendage welding machines, tube rolling mills and electron beam welding units were developed and fabricated indigenously, with the help of Indian industry.

NFC is marching forward as the only facility in the country catering to all power reactors for their fuel and reactor core structurals, in an expansion mode to meet the increasing demand of the nuclear power programme. In order to make nuclear power competitive, the cost of nuclear fuel will have to be brought down. In fact, NFC’s endeavor will be to make it globally competitive such that even when the markets are opened for free trade of nuclear fuel, NPCIL would still be buying fuel from NFC. The focus therefore will be on technology upgradation, productivity improvement, mechanization and automation, energy conservation, environment protection and quality enhancement. These will be achieved through identifying special tasks and implementing them on mission mode. Non-nuclear activities will also be pursued - for example, promotion of the use of non-nuclear grade zirconium in chemical and refractory industries.

NFC will be taking up all activities connected with manufacture of fuel and blanket sub-assemblies for the PFBR and for the future FBRs. Stainless steel tubular products, hexagonal fuel channels and precision machined components required for these assemblies will be fabricated at NFC and the final assemblies will be made in Kalpakkam in a fuel cycle facility to be developed jointly with IGCAR. For the Advanced Heavy Water Reactor, all the fuel channels and other hardware in zirconium alloys will be manufactured in NFC. Developmental activity towards this is already on hand.

Thus NFC is involved in development and manufacture of products of all the three stages of nuclear power programme, namely, PHWRs, FBRs and AHWRs. NFC is also developing many products indigenously for defence and space programmes. NFC is looking ahead to widen its product range further to cater to critical applications in strategic industries in the country thus carving a niche for itself as a National Facility for Critical Components. NFC is looking forward to an exciting, challenging, interesting and hence rewarding future for itself playing a major role in the Indian nuclear power programme and other strategic applications.”

- R. Kalidas

*Chairman & Chief Executive
Nuclear Fuel Complex, Hyderabad*

electropolishing of stainless steel specified for some PFBR tubes and for zircaloy.

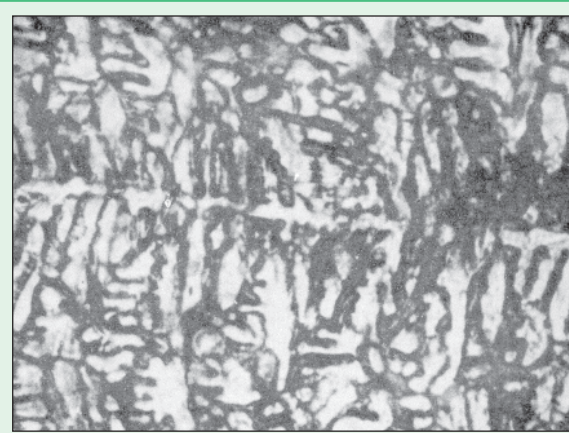
Zircaloy billets are clad with copper sheet before hot extrusion. The conventional process is mechanical cladding. To reduce the consumption of copper, some of the billets are being electroplated with copper by an alternate method. After extrusion, the copper is removed by dissolution in nitric acid. Alternatively, the removal of copper by electrolysis to enable recovery and reuse of metallic copper is also under development.

The three decades of experience on industrial scale manufacturing of zirconium alloy hardware (cladding tubes, end plugs, spacers, bearing pads, end plates, etc.,) and natural and enriched uranium oxide fuel for PHWRs and BWRs respectively has inspired confidence to take up, if needed, indigenous manufacturing of Zr-1%Nb clad enriched uranium oxide fuel for the forthcoming two VVER-1000 units at Kudankulam. India is blessed with a very large resource of the mineral zircon from which zirconium metal is extracted. India occupies a place of pride among the four countries in the world that have mastered the complex zirconium processing technology.

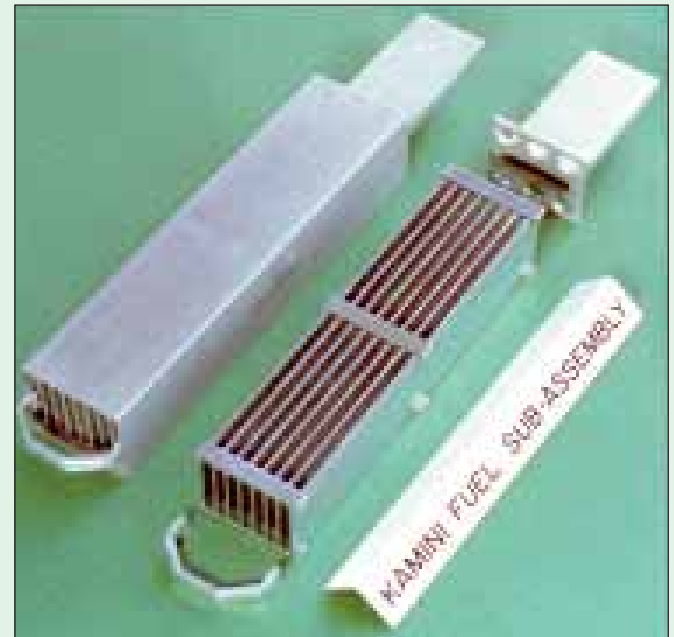
Today, DAE is poised not only to meet our domestic requirement of nuclear fuels, zirconium alloy and stainless steel core structurals but also to develop into a global player in the area of zirconium and tube technology.

Fabrication of ^{233}U -Al Alloy Fuel for the Neutron Source Reactor KAMINI

In the early sixties, along with the production of plutonium metal, preparation of aluminum-plutonium (Al-Pu) alloy which finds widespread applications as fuel in many research reactors was also taken up. A pilot scale facility was set-up in 1970 for the fabrication of Al-Pu alloy fuel elements. The facility was used to fabricate two 9-plate sub-assemblies of Al-10w/o Pu and Al-18w/o Pu plate elements by the picture frame technique for physics experiments in ZERLINA. Such Al-Pu plate fuel elements could be utilized as a substitute for the imported Al-13w/oU (85% enriched in ^{235}U) fuel used in the swimming pool reactor APSARA. The experience in this technology helped in preparing for the important step towards thorium utilization by developing and fabricating Al- ^{233}U alloy fuel for the neutron source reactor, KAMINI.



Microstructure of the first cast Al-Pu alloy prepared in 1961 (100X)



Fuel subassembly of KAMINI

The fuel for this reactor is a 2x62x260mm plate with a 1x55x250mm core of ^{233}U -Al alloy as meat and cover plates of 0.5mm thick aluminum mechanically bonded to the meat. The fabrication flow-sheet for this alloy fuel was developed and the fuel fabricated using the melting casting and picture-framing route. The major fabrication steps are the master alloy preparation, re-melting for dilution, ingot casting, hot rolling, picture-framing in aluminum clad components and roll-bonding.



9-plate fuel subassembly of Al clad Al-Pu alloy for ZERLINA

A small amount of zirconium is added during the alloy preparation stage to minimize the formation of the brittle UAl_4 inter-metallic phase to prevent cracking of the fuel alloy during hot-rolling. Zirconium stabilizes the relatively denser UAl_3 phase leading to improved ductility. A novel roll-swaging technique was used to mechanically lock the fuel plates in spacer grooves of short swaging strips. NDT techniques were used to ensure that the fuel plates meet the specification on chemical composition, bond-quality, fissile material homogeneity and dimensions of the fuel plates and the sub-assemblies. Besides the 9 sub-assemblies fabricated using this uranium fuel, 3 sub-assemblies using Al-23w/oPu fuel plates have also been fabricated to address the long-term reactivity requirements.

In the mid-eighties, when international efforts were directed towards development of high-volume fraction low-enrichment high uranium density fuel plates for research reactors, work in this facility was initiated on the development of a powder metallurgical route for Al alloy $Al-U_3Si_2$ (with LEU) dispersion plate fuel for the proposed 20MWt MPRR (Multi Purpose Research Reactor) and 100kWt LPRR (Low Power Research Reactor). The fabrication flowsheet for this fuel was successfully developed by fabricating a few of these plates. Neutron sources such as plutonium-beryllium, or americium-beryllium, have been fabricated for various applications like activation analysis, detector calibration, reactor start-ups etc.

Fabrication of Mixed Carbide Fuel for the Fast Breeder Test Reactor (FBTR)

The small core of FBTR necessitates a very high fissile atom concentration in the fuel to compensate for the high neutron escape probability. In the absence of a uranium enrichment facility at that stage, the use of high plutonium containing fuels was inevitable. The initial choice of fuel for FBTR was 76% PuO_2 -24% UO_2 (natural). Unfortunately, preliminary metallurgical investigations revealed that the plutonium rich mixed oxide was not chemically compatible with sodium coolant. The next logical step was to explore the possibility of developing plutonium rich mixed monocarbide or mononitride fuels. The initial experiments at Trombay established the feasibility of fabrication of $(U_{0.3}Pu_{0.7})C$ and $(U_{0.3}Pu_{0.7})N$ fuel pellets of controlled density and phase content on a laboratory scale starting from oxide powders, involving the powder pellet method.



Carbide fuel fabrication facility at BARC



MOX fabrication facility at AFFF, Tarapur

After detailed considerations, it was decided that the fuel for FBTR would be plutonium rich mixed carbide. Fabrication of this advanced fuel, called so due to its higher thermal conductivity, higher breeding ratio and high fissile atom density, is more difficult than the conventional oxide fuel mainly because of the highly reactive nature of the mixed-carbides. Production of the mixed carbide pellets is carried out by vacuum carbothermic reduction of mechanically mixed UO_2 , PuO_2 and graphite powders to prepare carbide clinkers which are crushed and milled to a fine powder before compaction and sintering in an inert gas atmosphere. Single step carbothermic reduction of oxide produced MC of relatively high oxygen content. In order to prepare nearly single phase MC with low oxygen content a two stage carbothermic reduction process involving preparation

of M_2C_3 , which has very low oxygen solubility, in first stage with excess carbon, followed by hydrogen reduction of M_2C_3 to MC in the second stage was developed. The sintered pellets are then inspected for physical and dimensional integrity, before encapsulating the pellets into a stainless-steel tube along with appropriate hardware by welding end-plugs.

On the basis of these encouraging results, a mixed carbide fuel fabrication facility of 200 kg capacity finished fuel per year was set at Radiometallurgy Division, BARC. Since large scale handling of highly radiotoxic plutonium and pyrophoric carbide was involved, several novel engineering and safety features were incorporated in the plant. These included a train of interconnected high integrity glove boxes with high purity inert gas for carbide pellet fabrication, a computer controlled pellet inspection unit, an automated TIG welding machine for fuel pin encapsulation and several analytical instruments for rapid analysis of process intermediates and final product. The indigenous fabrication of mixed carbide fuel for FBTR avoided the import of high enriched uranium and paved the way for gaining valuable experience of these advanced fuels over the entire fuel cycle. It may be noted that the fabrication of plutonium rich carbide fuel in this scale and its use as a driver fuel in fast reactors has been attempted for the first time in the world.

MOX Fuel Fabrication as an Alternate Fuel for BWRs

As part of our plutonium fuel development programme, (U,Pu) Mixed Oxide fuel has been developed and fabricated as an alternate for enriched UO_2 fuel for the BWRs. The 36 element BWR MOX assembly consists of fuel elements of three enrichments (0.9, 1.55 and 3.25 % PuO_2 by weight in natural UO_2) Fabrication of this fuel involves mixing of UO_2 and PuO_2 powders in the required proportion, homogeneous blending with an attritor, pre-compaction, granulation, compaction and sintering at high temperature. The sintered pellets after centerless grinding are inspected for density and dimensions, degassed and loaded into 4m long zircaloy tubes and sealed by welding end-plugs.

The co-milling of UO_2 and PuO_2 powder using an attritor instead of ball-mills, was done for the first time in India. The advantages of this method are that it reduces the time of milling

considerably, is safe to operate and does not involve an additional step of powder/ball separation. Further, the use of liquid/binder as a lubricant during milling improves the pellet yield considerably without requiring a separate de-waxing step.

Fabrication of MOX Fuel for PHWR

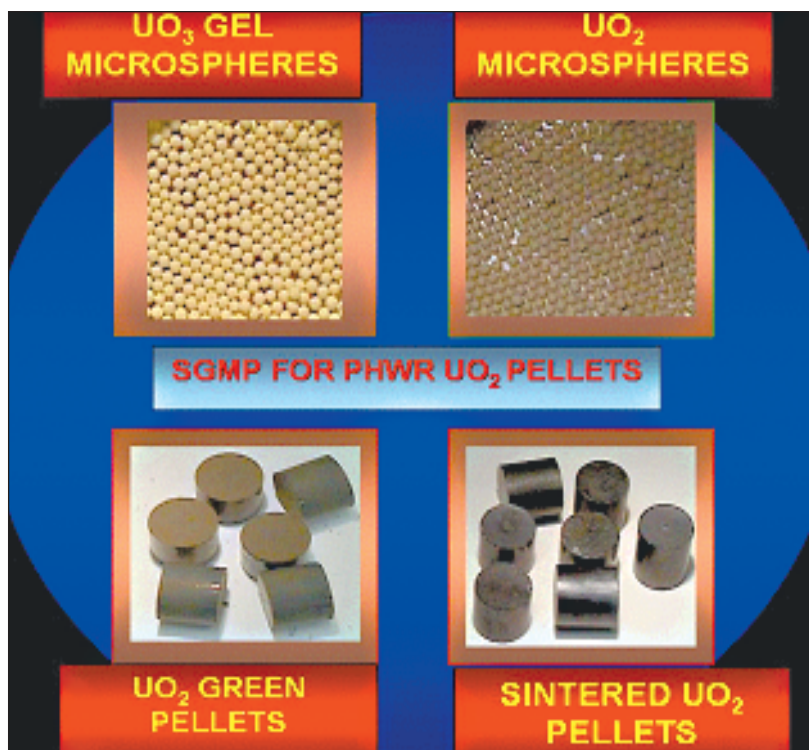
The use of plutonium based MOX in PHWRs not only improves fuel utilization but also reduces the volume of spent fuel discharged from present generation of PHWR reactors. Towards this, for experimental irradiation a few channels in Kakrapar Atomic Power Station have been loaded with 19 element fuel bundles of MOX-7 design. In the MOX fuel bundle of PHWR, the inner seven elements are replaced by MOX fuel elements using pellets having an enrichment of 0.4w/o PuO_2 , fabricated using the conventional powder metallurgy route. The end-plug welding of these fuel elements has been carried out using TIG welding technique. The end-plugs have been suitably modified to eliminate the requirement of machining the weld, as practiced in resistance welded PHWR fuel elements, and to make the welds amenable to evaluation by NDT.

Fabrication of Experimental MOX Fuel for Prototype Fast Breeder Reactor (PFBR)

Although plutonium-uranium mixed carbide fuel has been successfully used in FBTR, the first core of the fast breeder reactor PFBR-500 is based on mixed-oxide fuel, in view of the globally satisfactory experience with this fuel and also its suitability for large scale fabrication. Experimental short length fuel pins containing (U,Pu)MOX pellets of the typical composition and linear heat rating to be used in PFBR for irradiation testing in the FBTR have been fabricated. The composition of the fuel, including hardware, has been chosen to simulate the irradiation conditions in PFBR. This chemical composition is $(U_{0.71}Pu_{0.29})O_2$. The fissile content of natural UO_2 is enriched by $^{233}UO_2$ to the extent of around 50w/o. The annular shaped pellets have a nominal diameter of 5.6mm with a hole of about 1.8mm diameter. The pellets are encapsulated in D9 clad-tubes and the fuel elements are ensured for the quality using NDT techniques such as He-leak testing, X-ray radiography, gamma-scanning, visual inspection and metrology.

Fabrication of MOX Fuel for Advanced Heavy Water Reactor (AHWR)

The third stage of the nuclear power programme envisages utilization of the vast thorium resources in India. An advanced heavy water reactor, which uses (Th-Pu) and (Th-²³³U) MOX is a fore-runner to this stage. Fabrication of Thorium-plutonium mixed-oxide fuel can be carried out without much difficulty using technologies and facilities of uranium-plutonium mixed-oxide.



Sol gel microspheres and pellets for AHWR, ThO₂, (ThU)O₂ pellets

However fabrication of (Th,²³³U) MOX calls for advanced remote fabrication technologies as ²³³U is always associated with ²³²U impurity whose daughter products are hard gamma ray emitters. Among several approaches, sol-gel micro-sphere pelletisation (SGMP), impregnation and coated agglomerate pelletisation (CAP) are being developed as potential techniques for remote MOX fuel fabrication.

Sol-Gel Process

The sol-gel processes offer powderless route for the production of fuels, suitable for remotisation and total automation. The sol-gel processes are ideally suited for their adaptation at the back end of the fuel reprocessing plants, minimizing conversion steps. They also help in avoiding handling and transportation of fissile material. In this process,

droplets of the nitrate solution of the required heavy metal or heavy metals are converted to hydroxy gel microspheres and suitably heat-treated to obtain dry solid microspheres of desired density. Conversion of nitrate solution to hydroxy gel microspheres can be achieved by many methods, such as, dehydration, external gelation and internal gelation.

Development work on sol-gel technology was initiated in the late seventies. Detailed investigation of the chemistry of gelation and characteristics of gelled product indicated that internal gelation process is most amenable for the preparation of oxide, carbide and nitride of actinides such as U, Th and Pu. The chemical studies on the gelation behavior of nitrate solutions of uranium and thorium in internal gelation process, have resulted in a very good understanding of the morphology of the dry gel products minimizing the chemicals used for making gel microspheres. Hence internal gelation process was adopted and the properties of the gel microspheres of UO₂ and ThO₂ were optimised for their use as feed for the vibro-compacted fuel or for the pelletisation process. Initial experiments were carried out on gram scale with uranium using a manually operated set-up with a static gelation column. Production of UO₃ was demonstrated at 5kg/d scale on continuous basis using computer controlled auto-operated plant. The disadvantage of the production of large volume of low level liquid effluents in the internal gelation process has been overcome by a new effluent treatment scheme, based on ion-exchange removal of nitrate followed by controlled distillation to recycle ammonia and organics. The process automation has been worked out for pneumatic transfer of spherical solid particles.

Sol-Gel Micro Sphere Pelletisation (SGMP)

SGMP processes for the fabrication of UO₂ and (U,Pu)O₂ pellets have been developed at BARC. Originally gel spheres containing carbon pore former were used for the process which required very long and highly controlled heat treatment schemes. The SGMP process was modified by the use of gelation field diagram. Compositions of feed solutions suitable for obtaining crushable UO₂ microspheres were identified by observing the regular change of dry gel properties as a function of feed composition and heat treatment scheme. Low temperature oxidative sintering (LTS) procedure was used to make UO₂ pellets suitable for PHWR. Fuel channel using SGMP

pellets have been successfully irradiated at MAPS. A similar SGMP process was optimized for the preparation of ThO_2 and $(\text{Th}, 2\% \text{U})\text{O}_2$ pellets for their use in fabrication of fuel elements for AHWR. A glove box facility for the preparation of $(\text{U}, \text{Pu})\text{O}_2$ microspheres as well as a fore-runner sol-gel facility using a fully computerized control system were set up. Demonstration plants for sol-gel MOX microspheres are now being set up at Tarapur and another sol-gel facility at Kalpakkam.



Sol-Gel microsphere pellatisation for PHWR UO_2 pellets

POST-IRRADIATION EXAMINATION

Post-Irradiation examination (PIE) is a vital link in the nuclear fuel cycle. It gives valuable inputs towards development of technology for fuel and structural materials. PIE plays a significant role in determining the performance and residual life of the fabricated fuel and in-core structural materials and it gives valuable feedback to the fuel designers, fabricators, and reactor operating personnel. The PIE techniques in general can be broadly divided into two categories, viz- physical methods and chemical methods. In either case the operations are carried out in special facilities or enclosures such as fume-hoods, glove boxes, hot cells or poolside. The physical methods include visual examination, metrology, Non-Destructive Evaluation (NDE) techniques like X-radiography, neutron radiography, leak-testing, gamma scanning, eddy current testing, and destructive

techniques like mechanical testing, metallography, etc. The chemical methods include estimation of fission products, measurement of burn-up and distribution of fission products in the fuel, Fuel-Clad-Chemical-Interaction (FCCI), etc.

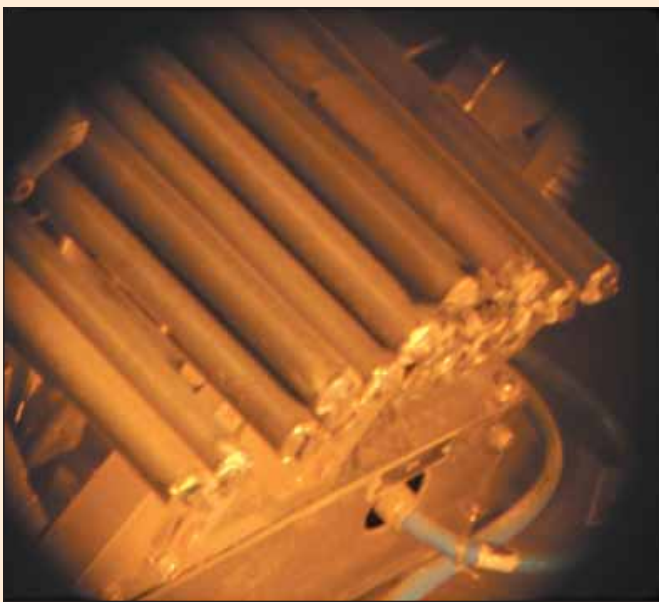
Various PIE techniques are used in logical sequence and have the common objective of obtaining unambiguous conclusions regarding the irradiation behavior of materials. Because of the complex nature of changes due to residence in the reactor coupled with the fact that all the tests are to be carried out remotely, it is normally considered prudent to plan for more than one technique for investigations on each aspect of the behavior.

Evolution of Hot Cell Technology

Recognizing the importance of PIE in our nuclear programme, extensive planning was undertaken as early as 1960 to establish indigenous PIE capabilities. The first beta-gamma hot-cells with air atmosphere were successfully built and commissioned at BARC in 1974, to carry out PIE on thermal reactor fuels and structural materials. The technology of PIE in India undertook a quantum leap, with the commissioning of



Operating area of RML Hot Cells, IGCAR



Laser dismantling of PHWR fuel bundle

alpha-beta-gamma hot-cells during 1994 at IGCAR to meet the needs of PIE of advanced fast reactor fuels.

The post-irradiation examination facility at PIED, BARC can handle gamma (1MeV) activity up to 3.7×10^6 GBq. It consists of six concrete cells with lead glass viewing windows and master slave manipulators for remote operations. These cells have been upgraded continuously and are now equipped with state of art testing facilities. A series of new hot cells is also under construction to augment the existing facility for handling larger components. The new hot cells facility consists of two concrete cells. One of them will be beta-gamma type and the other alpha-beta-gamma type, which will be used for examination of MOX fuel pins. A battery of lead cells is also planned as a low active laboratory.

At IGCAR, Radiometallurgy Laboratory (RML) has seven and Radiochemistry Laboratory (RCL) has five concrete shielded hot cells, each having a floor area of 5.5m x 2.1m. The RML cells are designed to handle 5×10^6 G Bq of gamma (1 MeV) activity. These cells were originally designed and constructed for handling mixed oxide fuel in air atmosphere maintained through once-through ventilation system. Subsequently based on the decision to switch over to mixed carbide fuel in FBTR, the hot cells were retrofitted with a high purity controlled nitrogen gas atmosphere maintained through recirculatory ventilation system.

Techniques utilized at PIED Hot cells facility at BARC for fuel examination include laser dismantling of bundle, visual examination, dimensional measurement, profilometry, gamma scanning, leak testing, fission gas release measurement, metallography and autoradiography. Techniques for pressure tube examination also include eddy current testing, ultrasonic testing, neutron radiography, mechanical testing, hydrogen/deuterium analysis, texture measurement and scanning electron microscopy.

In the RML hot cells at IGCAR, examination of fuel assemblies includes activities such as removal of sodium from fuel subassemblies, dimensional measurements, dismantling, non-destructive and destructive examinations such as leak testing, eddy current testing, x-radiography, neutron radiography, gamma scanning, fission gas analysis, metallography, density measurements and tensile testing. A separate mechanical testing facility has also been established, to carry out miniature

specimen testing, dynamic testing etc. of prefabricated specimens and samples taken out of irradiated structural materials. Kalpakkam Mini Reactor (KAMINI) located in the basement of the PIE facility is being used to carry out neutron radiography/ tomography of fuel subassemblies, control rods and fuel pins without any need for their transport outside the shielded and contained high purity nitrogen environment. The RCL hot cells have the facility to carry out fission gas analysis, burn-up estimation by radiochemical analysis and dissolution studies on irradiated fuel, etc.

PIE carried out on Fuel and Components of Various Reactors

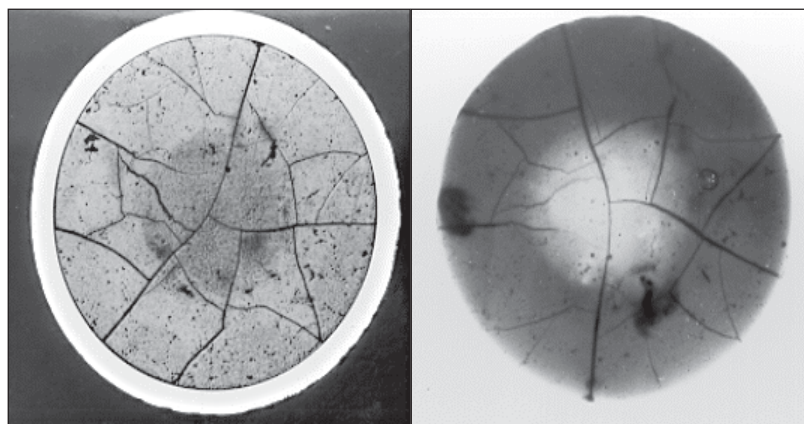
PIE has helped in understanding the in-reactor behavior of indigenously developed fuels and structural materials from various reactors such as DHRUVA, Tarapur Atomic Power Station (TAPS), Pressurized Heavy Water Reactors (PHWRs) and FBTR.

DHRUVA

PIE played a key role in evolution of fuel for DHRUVA. In the first phase, prototype fuel bundles which were fabricated using standard fuel fabrication route and test irradiated in CIRUS were examined to generate feed back information on their in-pile behavior and to qualify the fuel fabrication procedure. PIE of prototype bundles showed non-uniform elongation and shrinkage of fuel pins, fuel pin bowing and occurrence of primary and secondary hydride blisters. Based on the PIE findings the fuel fabrication and quality control was suitably modified. Experimental bundles fabricated with modified process showed excellent in-pile behavior and fuel assemblies were fabricated for the first core of the reactor. DHRUVA attained criticality in August 1985. After initial days of successful reactor operation abnormal increase in coolant activity indicated large-scale fuel failure. Post-irradiation examination of fuel assemblies from the reactor identified fretting wear of the aluminum cladding at the spacer grid location to be a global problem causing failure of fuel pins. PIE results helped in incorporating a number of modifications in the fuel assembly design to avoid failures. A few prototypes were fabricated and test irradiated in DHRUVA. PIE of modified DHRUVA bundles irradiated up to design burn-up of 1000MWD/T showed excellent performance.

PIE of TAPS Fuel

Under a program to assess the TAPS fuel performance, 18 full length fuel pins from eight fuel assemblies of TAPS-1 & 2 were selected for post-irradiation examination. The burn up of these fuel pins ranged from 5000 to 29,400 MWD/T. These fuel pins were examined in detail to identify causes of failure, extent of cladding corrosion, fission gas release, axial power and burn-up profile, pellet clad mechanical interaction, cladding properties, fuel restructuring and fuel clad chemical interaction.



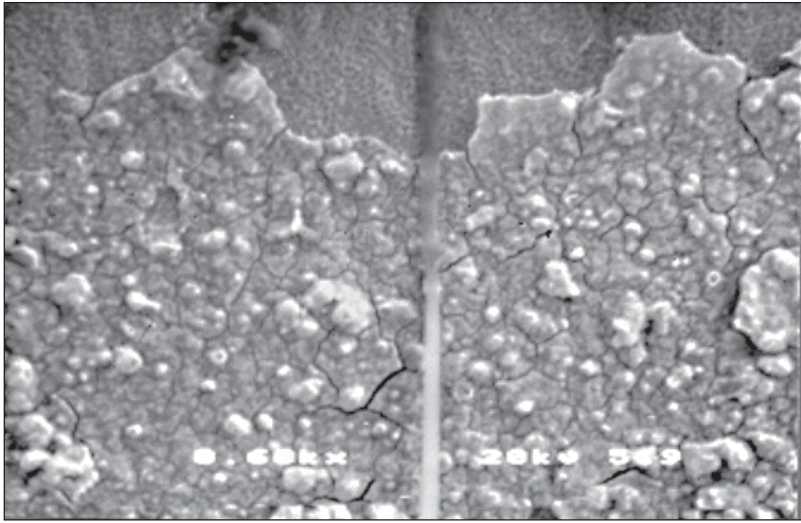
Photomicrograph and corresponding beta - gamma radiograph of a TAPS fuel section

PIE of MOX Fuel

The first experimental MOX fuel pin containing UO_2 -1.5% PuO_2 fuel in thin wall collapsible zircaloy-2 cladding was irradiated in PWL of CIRUS in 1975. The PIE of this fuel pin was carried out after it failed at a low burn-up. Its nature was identified as hydriding failure caused by moisture in the magnesia pellets used as an insulating pellet in the fuel column. In view of the use of MOX fuel in TAPS a number of fuel pin clusters containing UO_2 -4% PuO_2 fuel in a free standing cladding were irradiated up to a burn up of 16,000 MWD/T and detailed PIE of these fuel pins were carried out. Non-destructive examination of the fuel pins showed excellent behaviour. Based on the good performance of experimental fuel pins, MOX fuel assemblies have been introduced in TAPS.

Life Estimation of TAPS Pressure Vessel

Charpy impact test specimens representing base, weld and heat affected zone of the pressure vessel material were placed at the vessel wall and at shroud location during the construction



Typical fuel-clad chemical interaction site and a SEM photomicrograph of clad inner surface of TAPS fuel pin. Top picture shows the fission product deposit on the inside surface of cladding tube. Bottom picture shows the fuel cladding interaction

of the reactor to monitor the condition of the vessel as a part of the TAPS pressure vessel surveillance programme. These specimens were tested over a range of temperature to estimate irradiation induced decrease in the fracture toughness of the vessel material. The results were analyzed based on the latest revisions of regulatory guides, which confirmed the safety of the vessel not only for the design life of 40 years but also for an additional 20 years beyond the design life.

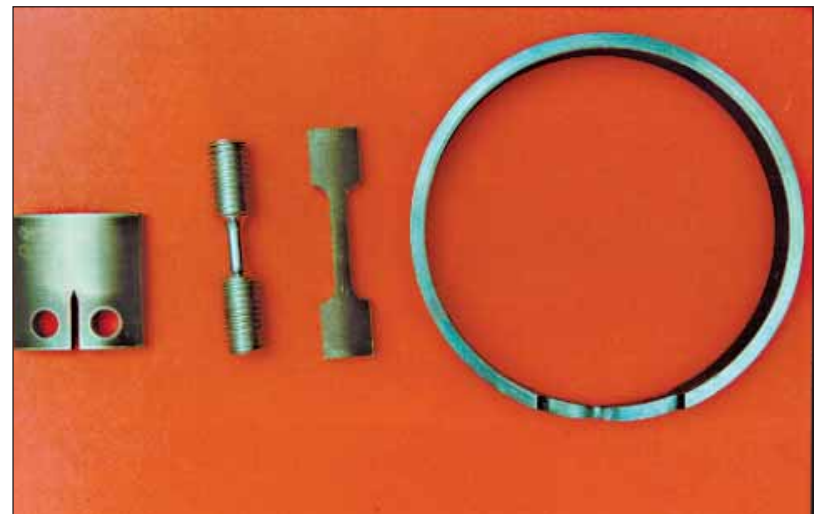
PIE of PHWR Fuel

A number of irradiated PHWR fuel bundles from Rajasthan Atomic Power Station (RAPS), Madras Atomic Power Station

(MAPS), Narora Atomic Power Station (NAPS) and Kakrapar Atomic Power Station (KAPS) have also been examined over the years at BARC. The burn-up of these fuels varied from 2,500 MWd/t to 14,000 MWd/t. PIE of these fuel bundles identified fretting and internal hydriding as the main causes of early life failures observed in some PHWR fuel bundles. Two ThO₂ fuel bundles from KAPS have recently been taken up for post-irradiation examination.

Life Management of PHWR Coolant Channel Components

During the last one decade PIE has played a vital role in providing valuable data on corrosion and hydrogen pick up behaviour and mechanical properties of irradiated coolant channel components such as pressure tubes, calandria tubes,



Specimens prepared from irradiated pressure tube for mechanical testing

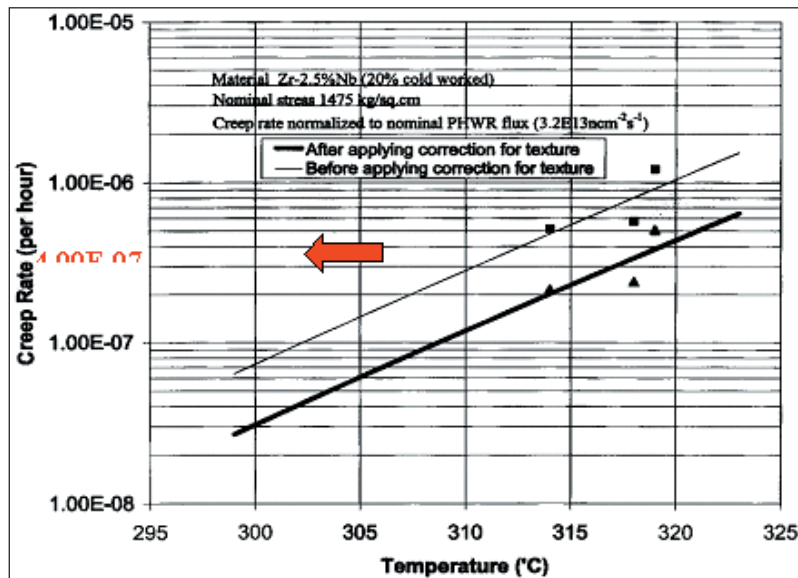
garter springs and end fittings from the operating PHWRs, for assessing the fitness for service of these components. So far, post-irradiation examination has been carried out on 15 full length pressure tubes, cut pieces from 17 pressure tubes and garter springs from 22 coolant channels. In addition scrape sliver samples from 22 pressure tubes from various reactors have been evaluated in the hot cells facility. The PIE data generated has enabled ensuring safe and continued operation of various PHWRs such as for RAPS-1&2, MAPS-1&2, NAPS-1&2 and KAPS-1.

Measurement of Irradiation Growth in Calandria Tube Samples

Irradiation growth specimens (100mm long, 12mm wide and 1.5 mm thick) made from seamless and seam welded zircaloy-2 calandria tube pieces were irradiated in DHRUVA reactor in G-9 pile position to estimated neutron fluence of about 1×10^{25} n/cm² at a temperature of 353 K. Growth measurement was made in PIED hot cells using LVDT. The irradiation growth rate in seamless calandria tube sample was of the same order as in the seam welded calandria tube samples. Hence, seamless calandria tubes have been chosen for use in RAPS-3 & 4.

PIE of Zircaloy Capsules for Evaluating Irradiation Creep Rate

PIE of irradiated pre-pressurized zircaloy capsule specimens has helped to evaluate the irradiation creep behavior of PHWR



Top graph shows temperature dependence of creep rate of Zr-2.5%Nb alloy under irradiation

Left photograph shows various parts of the pre-pressurized capsule

coolant tube materials such as Zircaloy-2 and Zr-2.5% Nb alloy. PIE was carried out at RML on irradiated pre-pressurized capsules made of Zircaloy-2 and Zr-2.5%Nb alloys, which are used for PHWR pressure tubes. The capsules numbering 130 were irradiated in FBTR for durations ranging from 35 to 80 effective full power days. This accelerated irradiation in the high flux core of FBTR is equivalent to about ten years of irradiation in the PHWR. The work involved development of special devices to remotely dismantle the carrier assembly, to extract the capsules from them through a series of precise machining work and to carry out diameter and weight measurements with very high accuracy. PIE helped in establishing the in-reactor creep rates of these indigenous PHWR materials.

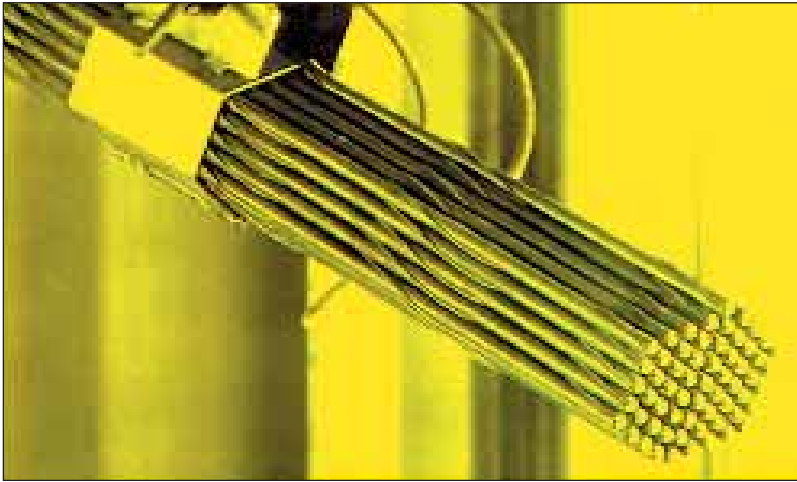
PIE of FBTR Fuel Sub-Assemblies

PIE on FBTR fuel sub-assemblies (FSAs) and fuel pins has helped in understanding the behavior of indigenously developed mixed carbide fuel as well as structural materials like the fuel-clad and hexagonal wrapper.

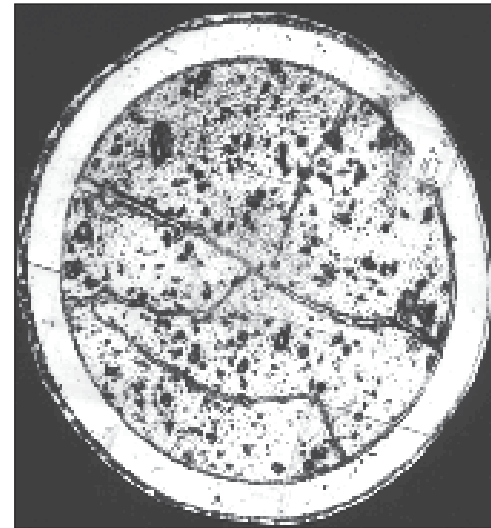
PIE on seven experimental fuel sub-assemblies irradiated in FBTR for short periods of 16 to 100 effective full power days helped in studying the evolution of the fuel structure and cracking behavior during the beginning of life of the fuel. Each sub-assembly contained one irradiation capsule housing a specially fabricated fuel pin. These fuel pins contained fuels with the Mark I core composition (70% PuC-30% UC) as well as the Mark II



FBTR fuel sub-assembly after burn-up 100 GWd/t

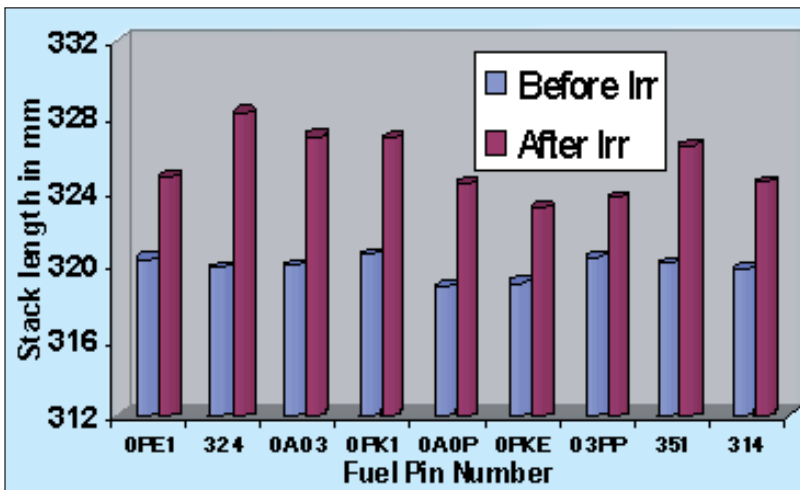


Fuel pin bundle extracted from fuel sub-assembly of FBTR

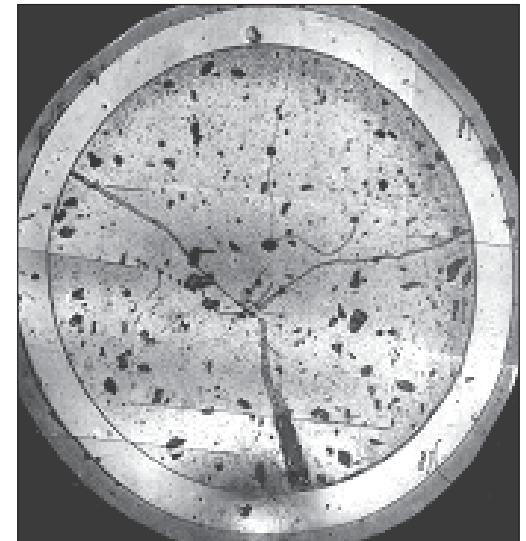


Middle of fuel column subjected to a burn-up of 25 GWd/t

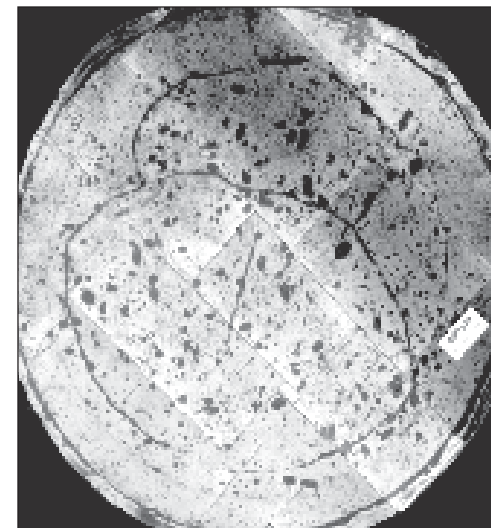
Variation of stack length estimated from X-radiography results



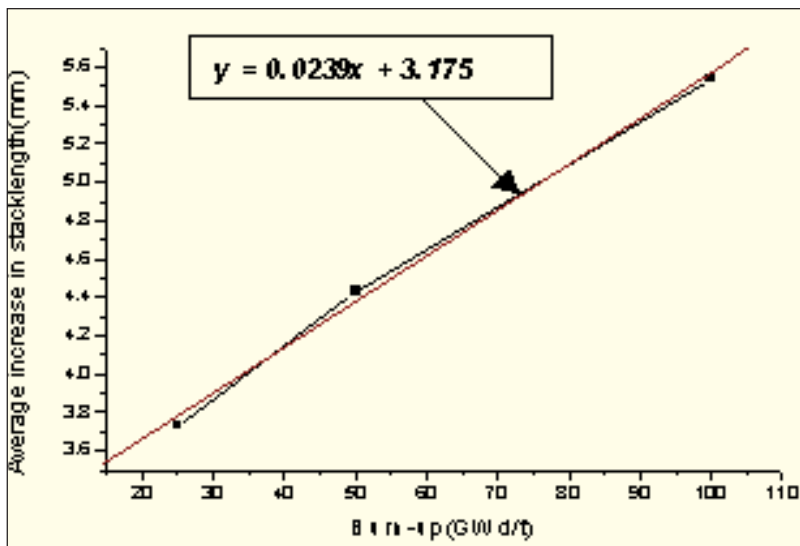
Histogram comparing variation in fuel stack length after 100 GWd/t burn-up on typical fuel pins



Middle of fuel column subjected to a burn-up of 50 GWd/t

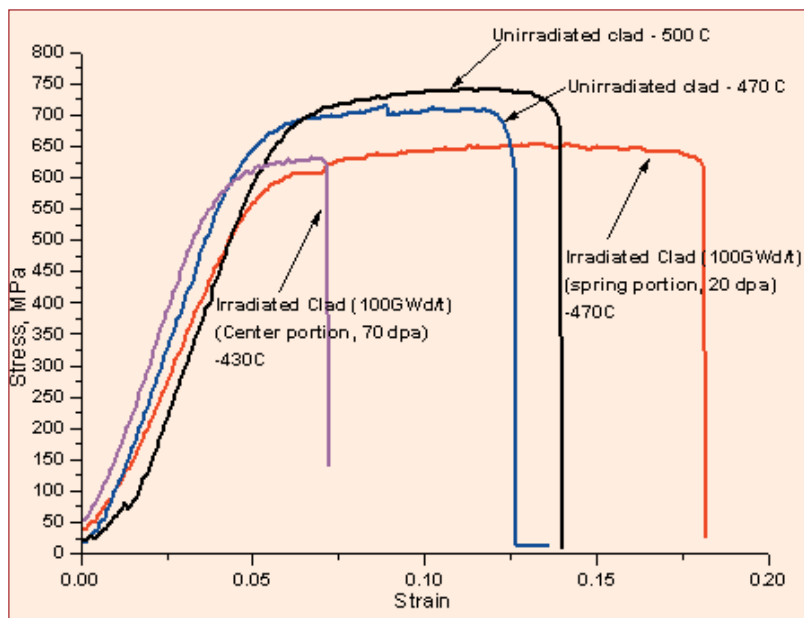


Middle of fuel column 100 GWd/t



Burn-up vs. increase in stack length

expanded core composition (55% PuC-45% UC). PIE conducted on the above experimental fuel pins revealed that cracking of fuel occurs at very low burn-up resulting in reduced fuel-clad gap and correspondingly low fuel center-line



Stress-strain plots obtained from tensile test data

temperature. PIE on two FSAs after 25,000 and 50,000 MWd/t burn-up has resulted in the progressive enhancement of burn-up of FBTR fuel from the initial limit of 25,000 MWd/t to 100,000 MWd/t. The liner heat rating of the fuel also could be enhanced progressively from the initial limit of 250 W/cm to 400 W/cm. PIE investigations were also carried out on the fuel sub-assembly after a burn-up of 100,000 MWd/t. Visual examination of the subassemblies indicated its overall good condition.

Fission gas analysis indicated that the percentage of fission gas release from the fuel to the plenum is of the order of 14%. Metallography revealed that the fuel-clad gap has closed at the center of the fuel column. Circumferential cracks are observed and porosities are found to be getting exhausted. Tensile testing of clad tube indicated that the ductility (uniform elongation) has come down and the residual ductility is of the order of 3%. Density measurements indicated that volumetric swelling is of the order of 4.4%. All the above results, including data on fission gas analysis, metallography, tensile testing and density measurements indicate that the fuel as well as the clad has performed very well and that the fuel can be taken to higher burn-ups of the order of 150 GWd/t.

Chemical Studies on FBTR Fuel

Comprehensive studies carried out on dissolution of irradiated carbide fuels have helped in standardizing dissolution parameters for the Lead Mini Cell plant. Electrolytically assisted



A set up for investigating carbide fuel dissolution conditions

dissolution and dissolution in boiling nitric acid have been successfully demonstrated on irradiated carbide fuel, which had seen burn-up of the order of 100,000 MWd/t. The facility for puncturing and extraction of fission gases inside the hot cells has been extensively used to arrive at the fission gas release of FBTR fuel pins. The burn-up estimation of FBTR fuel has been standardized with excellent results through dissolution and radiochemical analysis.

Development of New Techniques, Tools and Equipment for PIE

The technology for carrying out PIE of irradiated fuel and structural materials from PHWRs and FBTR has been mastered through consistent development efforts over the years. Various techniques, custom-built equipment, tools and gadgets have been developed indigenously and established in the hot cells of BARC and IGCAR. These equipments are designed to carry out precise measurement and inspections under high levels of radiation in controlled hot cell environments and are adapted for remote operation using master-slave-manipulators. Highlights and techniques developed and facilities set up at BARC and IGCAR are given below.

Development Work at BARC

Radiation resistant wall periscopes for viewing inside the hot cells of BARC

- Magnification up to 10X, capable of scanning the entire

area of the cell, with attachments for digital photography.

NDT techniques

For Pool-side inspection of PHWR fuel assemblies & pins

- Ultrasonic testing to identify leaking fuel pins, Eddy current testing for defect and oxide layer measurement, Profilometry to find out dimensional distortions, Gamma scanning for burn-up studies

For inspecting pressure tubes of PHWRs

- Ultrasonic testing technique based on dependence of ultrasonic velocity ratio on hydrogen concentration to detect presence of hydride blister in irradiated zircaloy-2 pressure tube at PT/CT contact location.
- Eddy current technique for measurement of oxide layer thickness on the inside surface of irradiated zircaloy-2 pressure tube of PHWR

Non-conventional testing techniques

- Techniques: Small punch test (SPT), Automatic Ball Indentation Tests (ABIT), Miniature disc bend & Shear punch test
- Features: Used to evaluate the mechanical properties and fracture toughness of irradiated structural materials using small size specimens. Most suited for irradiated components because of low man-rem exposure to operators.

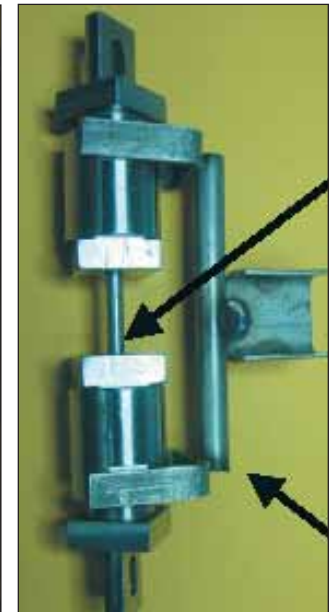
Development Work at IGCAR

- Sodium cleaning system to clean the sodium adhering to the sub-assemblies.
- 3-axis CNC machine for profilometry of the fuel sub-assemblies, dismantling and extraction of fuel pin bundle
- Profilometry–cum–eddy current test bench to scan the fuel pins along their length using LVDTs and eddy current probes and to plot the profile of diameter and surface defects of clad.
- X-radiography system for examination of fuel pins to detect defects
- Neutron radiography system for examination of FSA and fuel pins
- Fission gas extraction and gas chromatography for analyzing fission gas release and composition

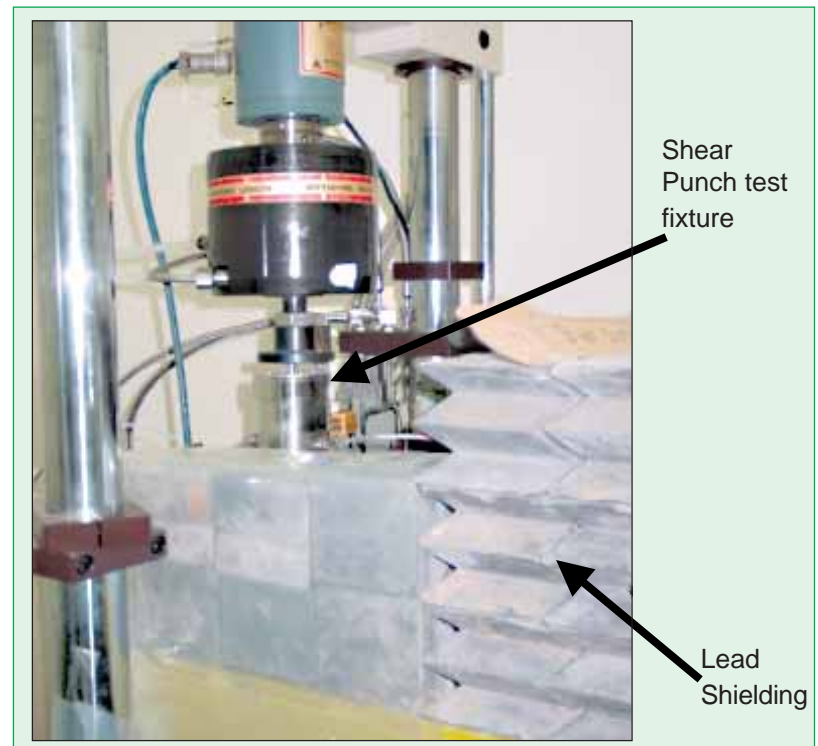
- High temperature remote tensile testing to evaluate mechanical properties of irradiated clad tubes at high temperature
- Specialized tools and gadgets for extracting fuel in bundle from hexcan, for unlocking capsules from carrier assembly, for remote repairs of carriage drive mechanism of neutron radiography etc. during PIE operations
- Miniaturized specimen testing techniques such as shear



Remote tensile test machine



Specimen gripping fixture



Miniature shear punch testing setup with shielding

punch test, ball indentation test and miniature disc bend test for estimation of post-irradiation mechanical properties of wrapper materials

Areas of Further Development

Important areas of development considered for near future include the following:

(i) Slit Burst Test Facility for measurement of fracture properties of irradiated pressure tube of PHWR.

(ii) Pool Side Inspection facility to generate statistically significant data on fuel performance by examination of a large number of fuel bundles discharged from the reactor. This facility will be used to identify leaky fuel pins /bundles for subsequent detailed PIE at hot cells. The facility consists of bundle dismantling, wet sipping and detailed NDT examination. The NDT examination will include visual examination, ultrasonic testing, eddy current defect and oxide layer measurement, profilometry and gamma scanning in the pool. The experience gained in the remote examination of fuels is being extended for developing under water inspection of FSAs of PFBR. A system has been designed and a prototype poolside inspection system is under development. The initial PIE of PFBR spent FSAs is planned in a poolside inspection facility consisting of an under-water bench for metrology and visual inspection.

(iii) In-cell experiment facility to study the behavior of fuel pins under off-normal and postulated accident conditions, in-cell experiments are planned to generate data on cladding oxidation, cladding deformation and ballooning and transient fission gas release behavior.

(iv) Laser dismantling of fuel assemblies. Another area of development has been the remote dismantling of FSAs. Laser dismantling is considered to have several advantages over the conventional slitting saw cutting such as ease of remote operation, minimum remote repair/replacement of parts and minimum generation of waste such as dust, chips, coolant etc. during cutting. Using a laser system, and a mechanical bench developed in-house, the dismantling of a dummy FSA has been demonstrated. This system is being upgraded and integrated for demonstration of remote dismantling inside hot cells.



Mechanical system used for the laser-dismantling of FBTR FSA

(v) Development of leak-tight manipulators for inert atmosphere hot cells is in progress. The indigenization of Power Manipulators also has been taken up and a prototype has been demonstrated.

(vi) A decontamination facility is being established with systems such as walk-in fume hood, glove boxes, ultrasonic and steam jet cleaning systems etc.

PIE Facilities established so far have met all the challenges and have succeeded in providing the required feedback to fuel designers, fabricators and reactor operating personnel regarding performance of critical components of operating reactors. The stage-wise assessment of performance and estimation of residual life of FBTR fuel has helped in progressively enhancing its burn-up and Linear Heat Rating to higher levels. PIE of fuel pins from TAPS, BWR, PHWRs and experimental loop has supplied valuable data on performance of natural UO_2 fuel, enriched UO_2 fuel and MOX fuel. PIE of irradiated zircaloy-2 pressure tubes of operating PHWR continues to provide valuable information on oxidation, hydrogen pick up and mechanical property changes in the pressure tubes, for life management of PHWRs. The technology of PIE has been mastered through long experience and is ready to meet newer challenges in the development of advanced reactors planned in our country.

EVOLUTION OF REPROCESSING IN INDIA

Thermal Reactor Fuel Reprocessing -The Initial Years

It is a tribute to the foresightedness of Dr.Homi Bhabha that India should embark on the reprocessing programme as one of its major activities. The closing of the fuel cycle by reprocessing and recycling of plutonium increases effective utilization of its meagre natural uranium resources and abundant natural thorium resources. While CIRUS was still under construction a formal order dated December 31, 1958 was issued to set up a plant to reprocess irradiated fuel discharged from it. For a country that had been independent only for a decade, the order conveyed the determination to embark on such path breaking, strategically important and scientifically challenging project. A team of young engineers and scientists from BARC training school were given the responsibility of commissioning the project. The only experience that all these team members had was the coursework on fundamentals of nuclear science and engineering during the training period. The preliminary design work was carried out on reprocessing during 1959-1961. During this period, experiments with pulsed perforated column were carried out to confirm the design data and also to get experience in the operation of the pulse column. This was followed by finalization of the process and equipment design, fabrication and installation of equipment and piping in the process cells and associated systems. The material of construction of the reprocessing plant at that time was SS 347. The plant was commissioned in 1964 to reprocess the spent fuel from CIRUS. The fuel elements were of natural uranium metal clad in aluminum.

The freedom and flexibility allowed to the project management ultimately made it possible to complete the project within sanctioned cost and committed time schedule. In the absence of much detailed information available except for the Brussels conference papers on the design and also the lack of experience in the industry for the specifications called for, the 'Trombay plant' was designed literally from first principles of chemistry and chemical engineering and the fabrication of the equipment was taken up in-house to ensure effective quality control. Ultimately the plant went 'hot' in August 1964 within 5 1/2 years from the date of order – a significant achievement and the first button of plutonium metal was produced in August 1965.

“...In the coming century nuclear energy will account for an increasing share of the electricity mix in India. Mature technologies for reprocessing, waste management and recycle of plutonium have been demonstrated and are available. Progress is under way on the Th - ²³³U cycle also. In this context, it is worth mentioning that because of our great interest in the closed nuclear fuel cycle, we have always considered spent fuel as a vital resource material. This was emphasized by us during the negotiations on the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive waste management. The closed fuel cycle, adopting a “reprocess to recycle Pu” approach after extended period of spent fuel storage, has several advantages. It renders reprocessing and nuclear waste management a more viable and safe technology, with reduced Man-Rem expenditures, since it minimizes the complication due to the presence of americium-241 in the recycled fuel fabrication process. The planning of reprocessing capacity should be such that the needs of the fast reactors/advanced PHWR, etc., which facilitate the utilization of plutonium and thorium, while reducing the input of natural uranium (in the process realizing the much higher energy potential of uranium) can be met on “just in time” basis, which is a very important concept in materials management...”

– R.Chidambaram,
43rd General Conference, IAEA, 1999

At the time of designing Trombay plant, and even today, the natural choice of technology was PUREX - a solvent extraction process using 30% TBP in an inert diluent mixture of paraffins with 12-14 carbon atoms or pure n-dodecane. The Trombay plant had a nominal capacity of 30 tonnes of HM/yr. The head-end operation consisted of chemical dejacketing followed by dissolution of the fuel in concentrated nitric acid under reflux conditions. The PUREX flowsheet used in Trombay plant comprised a single decontamination cycle, a partition cycle employing ferrous sulphamate as the reductant in nitric acid



Plutonium plant, Trombay

medium and two separate parallel cycles for the purification of uranium and plutonium. The final purification of plutonium was by anion exchange. The plant was based completely on direct maintenance concept. Though significant difficulties were encountered during the operation of this facility in the early days, none resulted in health hazard to personnel or environment, thanks to the commitment and devotion on the part of the personnel. Though there was no regulatory authority at that time, Dr. A.K. Ganguly who was the in-charge of safety used to discuss the design and operational aspects in detail, assess the situation and give clearance then and there. The experience in operating the plant and in assessing future requirements in reprocessing served as the basis for R&D programmes in the area of reprocessing. The long-term prospect of utilizing thorium had also been provided for when in 1968, ^{233}U was successfully separated from thorium irradiated in CIRUS.

The setting up of the reprocessing plant at Trombay had involved a major component of fabrication of stainless steel

“...Looking from the perspective of a large and growing economy like India, with its small hydrocarbon reserves and depleting coal reserves, development of nuclear energy based on a closed cycle approach enabling fuller use of uranium and thorium is the only way to meet development aspirations of over a billion people...”

- Anil Kakodkar, Chairman, AEC
47th IAEA General Conference, 2003.

and stainless steel welding technology and concrete technology for the radioactive cells. The construction of stack for this plant itself was a remarkable technology at that time. Some unique features of the plutonium plant are the underground waste tank farm with vertical tanks and a waste evaporation plant to concentrate the wastes.

An important programme in the development of a strong workforce of scientific and technical personnel was initiated in 1963 along with the construction of the plant. The persons recruited through the training programme turned out to be valuable assets in operating the plants constructed later.

With a few years of experience of reprocessing CIRUS fuel the next logical step was to set up a technology demonstration plant PREFRE at Tarapur to reprocess the Tarapur and Rajasthan power reactor fuels. This was not only a technologically important step, but also a strategy towards self-reliance in the face of embargo.

Improved Features in Tarapur Plant

The experience of operating Trombay plant gave sufficient impetus to continue developmental efforts in fuel reprocessing which include solvent degradation studies, sampling devices, analytical methods, equipment and systems for higher input capacity and on-line instrumentation and data acquisition systems. Mainly the material of construction has changed to SS 304L. All these resulted in the second reprocessing plant constructed at Tarapur with a design capacity of 100 t/yr for processing zircaloy clad spent fuel from Tarapur and Rajasthan atomic power stations which was commissioned in 1975. This plant uses chop leach process for the head end and uranous nitrate stabilized with hydrazine as the reductant for partitioning step. Instead of a separate co-decontamination and partitioning cycle, a combined co-decontamination cum partitioning cycle was introduced. The ion exchange purification of plutonium was replaced with a 20%TBP solvent extraction/stripping cycle to cater to the need of higher plutonium throughput. Several innovations such as pneumatic pulsing instead of mechanical pulsing, air-lifts as metering devices for the radioactive process solutions, thermo-syphon evaporators instead of pot type of evaporators and diluent wash for the raffinate solutions were introduced in the plant. The head end step is based on remote maintenance and the rest of the plant is based on direct



View of PREFRE plant at Tarapur

maintenance concept. Several campaigns of reprocessing were carried out under international safeguards and this plant gave rich experience in nuclear material accounting practices to meet the international standards.

This plant also provided experience in the design of appropriate transportation packages and safe in-land transportation of spent fuels adhering to the guidelines issued by the statutory regulatory authority. The safety guidelines at that time were based on the International Atomic Energy Agency (IAEA) advisory regulations for the transport of radioactive material. Spent fuel assemblies from reactor site are shipped in special casks called type B packages. These casks are made

of thick shielding with steel or combination of steel and lead and are intended to serve mainly two functions; containment of the radioactive material and protection of people and environment from radiation. Since 1975 there have been several shipments of spent fuel covering several thousand kilometers of road.

This plant has provided vital experience in reprocessing of power reactor fuel and during the initial period of operation some of the modifications to improve the plant availability were carried out which resulted in continuous operation for long stretches and provided the design basis for thermal reactor fuel reprocessing with higher throughputs. Parallel to this activity, the refurbishment of the old Trombay plant was undertaken.



Shipping of the cask containing spent fuel on trailer

Glove box train



Refurbishment of Trombay Plant

The original Trombay plant was meant to acquire the necessary skills required for mastering the technology of spent fuel reprocessing. Being the first ever facility, it had to be subjected to decontamination to permit access to the cells for trying out different new concepts to optimize the process conditions. In view of the inherently corrosive process environment to which the process vessels, equipment and piping were exposed during the years of operation, it was considered desirable to decommission the plant for effecting replacements to extend its life. This opportunity was also utilized to augment its capacity to 100 t/yr to meet the additional reprocessing requirement of fuel.

The entire decommissioning programme was meticulously planned to keep the personnel radiation exposures as low as possible. The personnel were trained in the type of operations involved and proper tools and equipment were devised for this specific purpose.

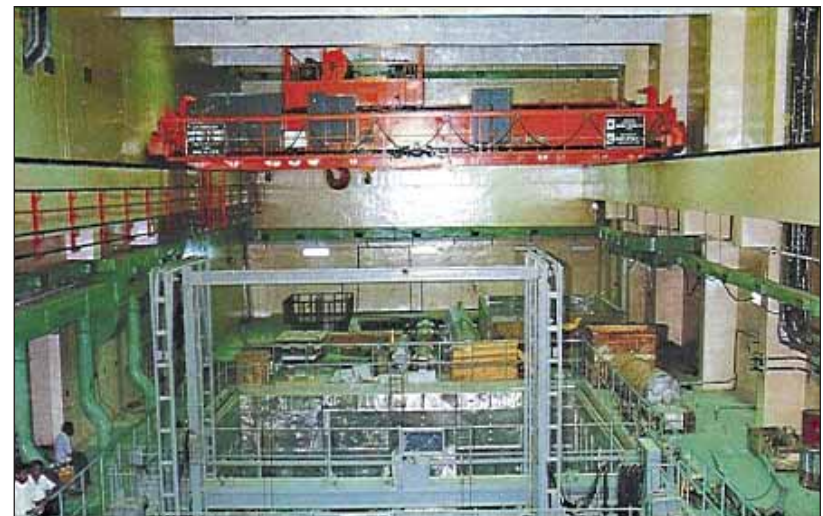
The campaign of internal decontamination of equipment and piping followed multiple decontamination routes. The maximum number of equipment was covered in a single route to minimize the decontaminants used so as to keep the resultant volumes of radioactive waste low. The range of equipment decontaminated includes liquid pulse columns, evaporators, condensers, ion-exchange columns, process vessels and associated piping etc. Following internal decontamination of the equipment, the task of decontamination of interior surfaces of the cells and the exterior surface of the equipment and piping was undertaken when radiation levels came down sufficiently to permit entry of personnel using protective gear. After dismantling and disposal of the equipment and piping, high-pressure water jets, steam, chemicals, pneumatic chippers and concreting were used as appropriate to remove contamination or shield the hot spots on the cell interior surfaces. Success of the decontamination resulted in salvaging of most of the cells and provided unrestricted access to the cells for installation of the refurbished plant. This decontamination operation gave valuable experience and provided vital information about the importance of making provisions for decommissioning during the design stage itself in the future reprocessing plants. Refurbishment of the plutonium plant was the first in the history of fuel reprocessing plants. The refurbished plant has the proven systems of Tarapur plant and has 100% redundant capacity for the dissolver system. The augmented plant started operating from 1983 and continues to be in operation.

Standard Design of Kalpakkam Plant

After successful operation of Power REactor Fuel REprocessing (PREFRE) plant at Tarapur and with the experience gained during decommissioning and refurbishment of Trombay plant, need arose to augment the reprocessing capacity to treat the fuel from the increased nuclear power generation from the PHWRs. To cater to the need of reprocessing zircaloy clad natural uranium oxide spent fuel from Madras atomic power station (MAPS), a third reprocessing plant



Typical cell piping in reprocessing plant



Fuel storage pond

was designed near the power station at Kalpakkam, with a capacity of 0.5 tonne heavy metal/day.

Unlike the earlier approach of constructing reprocessing plants totally by the departmental personnel, the execution of this plant was carried out by involving Indian industry in the fabrication of equipment, installation and piping work. The Kalpakkam plant design is to serve as a standard design for the future plants. The design aims at availability of the plant capacity for the entire life of the power station. With the acceleration of nuclear power programme with increased quantities of spent fuel, the facility for the fissile and fertile material recycling has to be augmented. For the economics of power generation the cost of the operations in the back end of the fuel cycle has to be minimized and new innovative project

management techniques like EPC have to be resorted to involve engineering industry within the country.

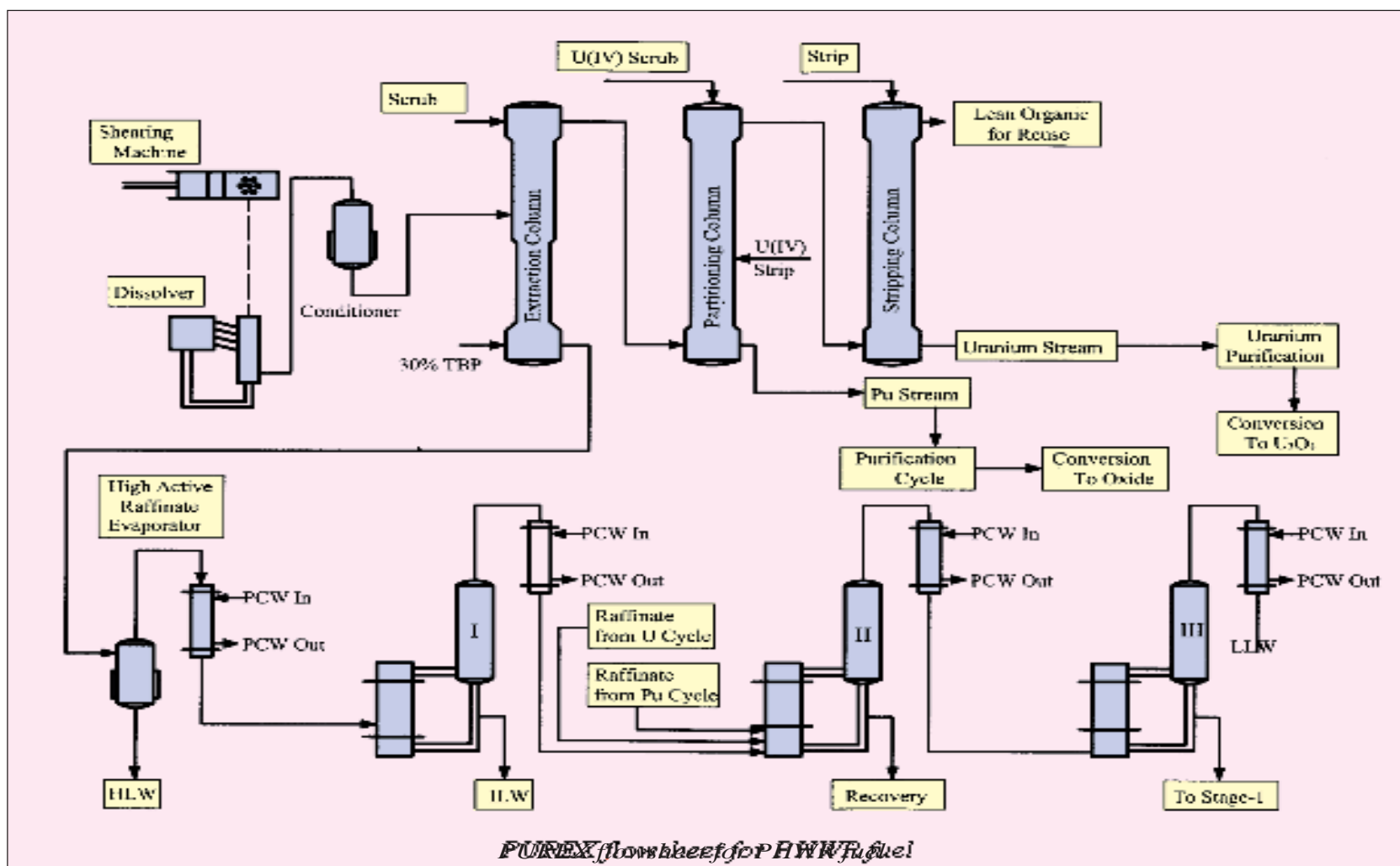
The existing capacity to reprocess power reactor spent fuel will be gradually enhanced to meet the fuel requirement of fast reactors and advanced heavy water reactors.



Remote precipitation column facility

The Process: PUREX

The basic steps of PUREX are the head end step involving chemical or mechanical decladding followed by the dissolution in nitric acid, feed clarification and adjustment of chemical conditions for the further solvent extraction step. The solvent extraction involves extracting both uranium and plutonium into organic solvent and back washing the small amount of fission products extracted along with uranium and plutonium into aqueous stream containing the majority of the fission products which is the high level waste. Again using solvent extraction, the organic is loaded with U and Pu and these are separated from each other using uranous solution (U(IV)) which reduces the Pu(IV) to Pu(III) thereby bringing the less extractable Pu(III) into the aqueous phase followed by solvent extraction purification cycles separately for uranium and plutonium.



Technological Improvements in Thermal Reprocessing Journey

About 40 years of experience in processing the spent fuel based on the PUREX process has given the confidence that this technology can be successfully implemented for the recovery of almost 99.5% of both U and Pu, which with careful improvements can be increased further. Some of the technological improvements over the years are:

- Partition with hydrazine stabilized uranous has reduced the corrosion problems in the plant to a great extent and resulted in a much cleaner waste being generated in the cycle.
- Substantial reduction in waste volumes was achieved over the years by resorting to salt free reagents and evaporation followed by acid reduction by formaldehyde.
- The uranous production by electrolysis has been standardized to optimum current efficiency and conversion by resorting to new electrodes such as titanium substrate insoluble anode and titanium cathode. Catalytic reduction of uranium (VI) to uranium (IV) by hydrogen over finely divided platinum dust has also been developed.
- The in-situ reduction of U(VI) to U(IV) for partitioning of Pu using mixer settlers and pulsed columns was demonstrated. The operating experience of these pilot devices will be used for designing systems for future reprocessing plants.
- Vacuum assisted airlift operated sampling design has been developed in-house to meet the requirements of PUREX process and further efforts are on to automate the system to minimize exposure.
- An indigenous shearing machine was developed and installed at KARP incorporating many design improvements. Changes have been incorporated in the clapper door assembly to improve reliability and plant throughput. Developments are underway to design and fabricate indexing casks and automated charging of the fuel bundles into the shearing machine magazine to increase the throughput and reduce the exposure to operator.
- To combat corrosion, low carbon grade 304L stainless steel is used in the reprocessing plants. Factors other than sensitization such as the presence of active inclusions and segregation of silica and phosphorous to grain boundaries, can cause inter granular corrosion in oxidizing nitric acid environment. With this understanding tighter specifications

were formulated which has resulted in nitric acid grade (NAG) steel being produced within the country with corrosion rates as low as 10 mpy.

R&D Issues in Thermal Fuel Reprocessing

Chemical aspects of the process are continuously refined to achieve the following:

- Improvements in the recoveries of Pu and U and decontamination factors.
- Reduction in the waste volume generation.
- Reduction in the number of cycles
- Recovery of the nuclides of commercial value.
- Removal of dissolved organic from aqueous streams

R&D efforts are focused to improve the head-end operations by incorporation of equipments of mechanically intense and trouble free performance for low down time of the plant. Development of robust automatic systems for bundle level charging and chopping will help in man-rem exposure control. Laser assisted bundle dismantling followed by pin level chopping is also being contemplated to increase the efficiency. Development of continuous rotary dissolver is also being undertaken. In-situ electro-partitioning will help in reducing the salt content and increase the throughput. To enable real time accounting for fissile and fertile actinides, efforts are underway to develop an on-line measurement technique using NDA methods. Active and passive neutron based techniques are also being explored for the assay of fissile material.

²³³U-Th Separation Activities at BARC and IGCAR

When BARC started working on THOREX process (acronym for the process used for the separation of ²³³U and Th from irradiated thoria) the basic laboratory data were scarce and most of the batch and counter-current extraction and stripping data had to be generated in-house. Different flow-sheets were developed for the recovery of ²³³U alone or for both ²³³U and thorium. As the recovery of ²³³U alone was contemplated during the initial phase of the DAE programme, more emphasis was given to the process using 5% TBP- odourless kerosene solvent.

Development of the laboratory scale flow sheet for the recovery of ²³³U from thorium was followed by setting up of a

pilot scale facility to reprocess aluminium clad thorium metal and oxide rods irradiated at J position in CIRUS for the recovery of ^{233}U . The head-end process employed chemical decladding of aluminium followed by dissolution in 8 M HNO_3 containing 0.05 M NaF and trace amounts of aluminum nitrate. ^{233}U was separated from Th and fission products by extraction in 5% TBP in Shell Sol-T followed by scrubbing with 1-2 M HNO_3 and stripped with acidified water. The contactor used was a mixer settler unit made of glass and the interconnections were made with PVC tubings! Further purification was carried out by anion exchange in chloride medium. The product solution obtained was then precipitated with ammonia solution and ignited at 1123K. During the late '80s, requirement of ^{233}U as fuel for KAMINI arose and a decision was taken to build the plant in the existing cells at Reprocessing Development Lab (RDL), IGCAR to reprocess Th and ThO_2 fuel elements irradiated at CIRUS and DHRUVA for recovering ^{233}U , based on the Interim-23 process. Alkali was used for chemical de-cladding and 5% tri-n- butyl phosphate was used as the solvent in a single cycle process. Final purification of ^{233}U was based on anion exchange. An 8 stage belt driven centrifugal extractor was employed for the extraction with the scrubbing and stripping carried out using a bank of air pulsed mixer settlers. The first campaign of J-Rods from CIRUS was carried out in 1988. The second campaign was carried out in 1999 for processing CIRUS and DHRUVA thorium rods. This plant processed a total of 78 Th/ ThO_2 rods in two separate campaigns. The fissile material recovered in the plant was used to make KAMINI fuel core and also for the fabrication of a fuel assembly of PFBR composition for test irradiation in FBTR. This facility is presently shut down. This experience was very useful in building the Lead Mini Cell (LMC) for the separation of U and Pu from the FBTR fuel. By 1985 a high speed small diameter feed clarification centrifuge was developed and this was used for the first time in J-Rod processing facility.

An engineering scale facility for ^{233}U Separation, UTSF (Uranium Thorium Separation Facility) was designed and commissioned at Trombay for the processing and recovery of ^{233}U from CIRUS and DHRUVA irradiated thoria rods on a regular basis. The modifications felt necessary from the pilot plant experiences of both Trombay and IGCAR facilities have been incorporated in the design of equipment and in the choice

of process flowsheets. Specially designed CALMIX mixer settlers have been chosen as contactors. The scrub section has been extended sufficiently to provide for adequate removal of thorium from the uranium loaded organic.

Power Reactor Thoria Reprocessing Facility (PRTRF)

The Power Reactor Thoria Reprocessing Facility (PRTRF) is being constructed at Trombay for processing the irradiated Thoria bundles from PHWR, to separate ^{233}U . Thoria bundles irradiated in power reactors such as, KAPS-1&2, MAPS-1, RAPS-3&4, KAIGA-1&2 will be reprocessed at PRTRF. This facility would provide rich experience as several new technologies are being adopted in the flowsheet.

The process flowsheet involves dismantling of the spent fuel bundle end plates for the first time by using a 220 Watt NdYAG LASER mounted on Computerised Numerical Controller(CNC). Individual pins from the bundle will be chopped into pieces in one stroke by a hydraulic cylinder and fed into dissolver for leaching with nitric acid in the presence of sodium fluoride-aluminium nitrate. ^{233}U will then be selectively extracted using 5% TBP/NPH in CALMIX contactors and further purified by ion exchange prior to conversion to oxide. The thorium stream containing bulk of the fission products will be treated as acidic waste and will be stored in the Waste Tank Form (WTF).

Minor Actinide Partitioning

Nuclear wastes especially the high level wastes from spent fuel reprocessing plants are proposed to be immobilised in suitable glass or synroc matrices and stored in deep geological repositories under surveillance. Removal of long lived minor actinides such as, ^{237}Np , ^{241}Am , ^{243}Am and ^{245}Cm from the heat generating isotopes such as ^{137}Cs and ^{90}Sr as well as the stable fission product lanthanides prior to immobilisation will reduce the overall cost of waste management. Several groups are involved in the synthesis of suitable extractants that can be used for the recovery of minor actinides and useful fission products. Extractants such as octyl phenyl N,N diisobutyl carbamoyl methyl phosphine oxide, malonamides such as dibutyl dimethyl tetradecyl malonamide(DBDMTDMA), diglycollamides such as tetraoctyl and tetra 2-ethylhexyl diglycol amide (TODGA, T2EHDGA) have been synthesised indigenously on large scales in high purity. Extensive batch

equilibration experiments as well as counter-current runs using mixer-settlers have been carried out with actual sulphate bearing high level waste solutions as well as simulated waste solutions. Selected extractants are also being evaluated to separate the large quantity of lanthanides from the actinides to reduce the vitrification load. Several ion specific crown ethers have also been synthesised in BARC and explored for the recovery of ^{137}Cs and ^{90}Sr .

Fast Reactor Fuel Reprocessing at IGCAR

At IGCAR reprocessing related activities were initiated in the year 1972 with an initial budget allocation of Rs.5 crores with a clear mandate of developing the technology of fuel reprocessing of short cooled (8 months) and high burn-up (> 100,000 MWd/t) fast reactor fuel with minimum Pu losses (0.1%) and recycling the recovered uranium and plutonium back into the reactor (closing the fuel cycle). The Reprocessing Development Laboratory (RDL) at IGCAR started in the year 1976 with 4 concrete cells and the fifth a lead shielded cell and an engineering laboratory and chemical laboratory. There was also a separate space allotted to build a lead shielded facility for active R&D work which was later upgraded to Lead Mini Cell (LMC) for processing of FBTR fuel.

Initial Years (1976-1986)

In the initial years the emphasis was on building facilities such as, workshop, chemical laboratory, and ventilation systems for the plant and the development of head end and solvent extraction equipment. Until around 1982, the initial reprocessing plans centered on mixed oxide fuel of U and Pu. Among the solvent extraction contactors, some operating experience on liquid pulse column was available at plutonium plant, BARC and air pulsed mixer settler was developed at BARC. Some of the glass mixer settler units were brought to IGCAR based on which a multi-stage unit was made in stainless steel for the experimental studies. The need for the development of short residence time contactors was realized from the very beginning because of the concern of radiation damage of solvent due to the high burn-up of the FBTR fuel for which no experience was available in DAE. A 50 mm centrifugal extractor (CE) with paddle mixing was developed in the year 1977 for the first time in DAE. Since the FBTR fuel reprocessing flow rates envisaged were

very small (3-10 L/hr) the need of the hour was to design smaller units, which demanded accurate calculation of weir level, posing difficulty in design and fabrication. Subsequently a three stage vertically stacked low hold up CE, with a capacity of 15 L/hr, was designed, fabricated and tested successfully in the year 1980 based on the Robatel multi-stage unit. By that time annular centrifugal extractor was developed by Argonne National Laboratory (ANL), which obviated the problems of maintenance of the paddle mixer type units. By 1982 an eight-stage annular CE, which is a forerunner of CE's presently used in LMC, was designed and used for the extraction cycle of the J-rod plant for the recovery of ^{233}U from thorium fuel irradiated at CIRUS. The CE banks helped in successful processing of J-rod fuel in two separate campaigns with a gap of 12 years in between, without any maintenance problem excepting the polyurethane drive belt replacement. During mid-eighties and early nineties larger capacity centrifugal extractors were fabricated and the effect of flooding capacities on the variations in internal geometries and outlet and inlet ports to the bowl were studied. Presently the technology exists to design centrifugal extractors from as low as 5 to 3000 L/hr.

Around 1980-81, it was felt that the fast reactor fuel reprocessing technology is manageable except for the chopper which used to be the subject of discussion often in the review meetings. Finally the group could visualize the general layout of a simple single pin shear unit with fuel pin feed mechanism, rotating magazine, and a holding and shearing gag mechanism. A simple tool holder design, commonly used in lathes was designed and fabricated. It was demonstrated by the end of 1983 with a dedicated electronic device to control the sequence of shearing. Later, the rotary motion of the magazine was improved and computer control was adopted for the various steps of operation.

By about 1981, it was clear that France would not supply the mixed oxide fuel for FBTR and India was forced to make its own mixed carbide fuel with natural uranium and plutonium for operating the FBTR. The immediate concern for fuel-reprocessing group in mid eighties was about the dissolution of carbide fuel as no previous experience existed and the second concern was the role of carbon compounds in the down stream solvent extraction cycles. Development of an Electro Oxidative Dissolution Technique (EODT) using silver as a redox



*EODT
dissolver
under
erection*

intermediate helped in destroying the carbon present in the solution and also aided in increasing the dissolution rate. Based on this experience a titanium dissolver was fabricated and installed in the present LMC.

Lead Mini Cell (LMC)

The LMC facility comprises of a lead shielded hot cell with 250 mm or 200 mm thick lead shielding (depending upon the



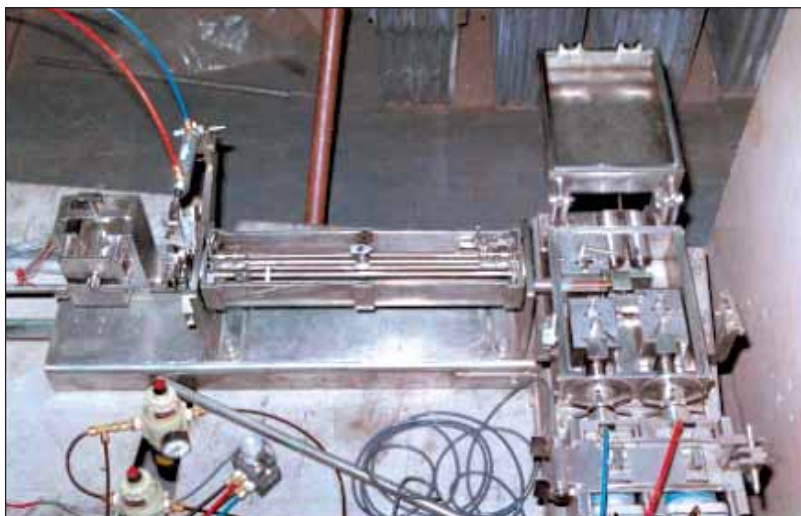
View of Lead Mini Cell (LMC)

β , γ radioactivity in different zones). The cell is a compact one and constructed on an area of 11m x 2m with a height of 3 m. An α -tight stainless steel containment box of 10m x 1.2m x 1.5m high is housed inside the lead shielding. The cell is provided with six radiation-shielding windows and six pairs of articulated arm type master slave manipulators to facilitate remote operation and maintenance of different equipment and systems. In addition, an α -tight blister box provides access for direct maintenance of small gadgets, which can be brought out of the cell. Specially designed annular and slab tanks have been installed to store solutions containing Pu to meet the safety requirements of criticality. About 2 km of intricate stainless steel piping involving 3000 bends and 2000 X-Radiography joints has been successfully completed within the limited area of the facility. The density of piping was a challenging task for the manipulation by welders. Around 35 process vessels and 30 different equipment are installed in the hot cell. A glove box, integrated with the containment box houses contactors for the partitioning and final purification cycle. Since large concentrations of plutonium are handled, the containment box and glove boxes are designed to be α -tight in addition to providing criticality control measures. The LMC facility is a unique facility housing a variety of equipment in a small space, simulating almost plant conditions meant for processing fuel with burn up of greater than 100,000 MWd/t.

FBTR fuel reprocessing is carried out in the LMC with a nominal capacity of 40 kg/yr. The objective is mainly to test the process flow sheet and the equipment developed. These equipment, after incorporation of design features based on the operational feed back, will be put up for plant level operation in the Demonstration Fast Reactor fuel Reprocessing Plant (DFRP), which will cater to the needs of closing the fuel cycle for FBTR.

Process Flow Sheet – Modified PUREX

Modified PUREX process based on 30% TBP in NPH diluent with two co-decontamination and one partitioning cycle is employed in LMC. The fuel sub-assemblies are dismantled and individual pins are loaded in the fuel magazines, which are transported to LMC in special α -tight containers, kept in shielded casks. The fuel is chopped in an improved single pin chopper and then dissolved in an electrolytic dissolver and after



Single pin-chopper for FBTR fuel



16-Stage centrifugal extractor bank

centrifuge clarification and three cycles of solvent extraction; pure Pu and U are separated. Third cycle has a facility for partitioning. The purified Pu solution is transferred to re-conversion area for subsequent conversion to oxide. Extensive modeling work has been carried out and is applied in the design of the plant.

Major Equipment Used in the Plant

All the process equipment are housed in the containment box and glove boxes. All the process tanks are located below the containment box within the lead shield. The fuel pins are chopped into small pieces using the single pin chopper and charged into the electrolytic titanium dissolver where the dissolution of the fuel is carried out in nitric acid with silver nitrate as catalyst. An expanded titanium wire mesh acts as the cathode, while platinum electroplated titanium wire mesh acts as the anode. The anode was developed in collaboration

with the Central Electrochemical Research Institute (CECRI), Karaikudi. A high-speed centrifuge is used for clarifying the feed solution.

The centrifugal extractors of improved design, based on feedback from ²³³U processing campaign, have been installed in the cell. For recovering pure Pu from a mixture of U and Pu, electrolytic partitioning mixer settlers are used after testing with natural uranium.

Apart from the process equipment, there are many hot cell systems, which have been tested in this facility. Some of these are mini-manipulators of improved design, remote sampling, in-cell crane and inter-cell trolley. LMC facility serves to simulate and trouble-shoot problems that are likely to arise during the operation of future fast reactor fuel reprocessing plants. This plant will also serve as a hot facility for testing novel solvents in future.

The LMC facility was commissioned in 2003. Mixed carbide fuel irradiated in FBTR was reprocessed for the first time in the world with progressively increasing burn-up. So far, two campaigns with low burn-up fuel and two more with 100 Gwd/t fuels have been completed. The difficulties in chopping due to the pyrophoric nature of the fuel and problems in solvent extraction due to the organic acids produced during the dissolution in nitric acid, have been overcome by optimizing the operation parameters. Single pin chopper, dissolver, feed clarification centrifuge and the centrifugal extractors which are developed for the fast reactor fuel reprocessing have been successfully operated and the feedback has been provided to the designers. Required decontamination factors from the fission products have been achieved and the plutonium oxide product has been separated.

Demonstration Fast Reactor Fuel Reprocessing Plant (DFRP)

DFRP will cater to the needs of closing the fuel cycle for FBTR. Based on the performance feedback of the equipment used in LMC, improvement in the design of chopper, centrifugal extractor and dissolver will be incorporated in this plant. In addition, it is planned to install equipment such as, laser dismantling machine, chopper etc., for PFBR spent fuel reprocessing so that the designs of these equipment could be improved for PFBR reprocessing plant coming up in the near

future. Presently the plant is under construction and about 70% of cell piping has been completed and all the process vessels (critically safe annular tanks) are in place. The plant is scheduled for operations in 2007.

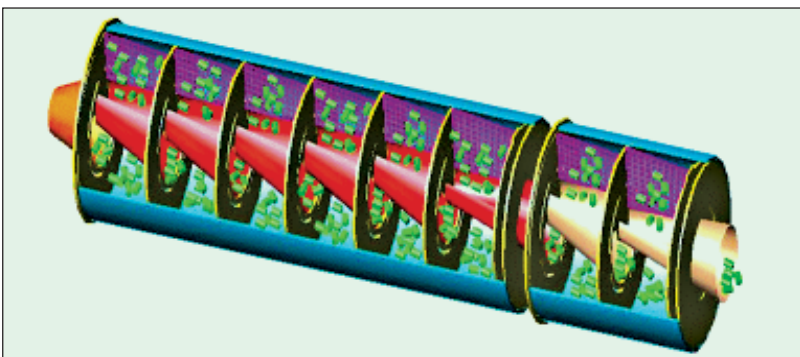
The objectives of DFRP include :

- Reprocessing of FBTR spent fuel for closing the fuel cycle and demonstration of the recovery and decontamination factors in line with the thermal reactor fuel reprocessing
- Waste volume reduction, acid recovery and reuse matching with the acceptance level for waste immobilization
- Demonstration of laser dismantling of fuel subassemblies
- Demonstration of improved construction materials for dissolver and evaporators
- Demonstration of techniques for on-line monitoring of Pu as well as selected fission products in process streams and monitoring of leached hulls
- Demonstration of robotic based sampling systems
- Demonstration of PFBR oxide fuel reprocessing
- Evolving in-service inspection techniques for fuel reprocessing plants

R&D Activities

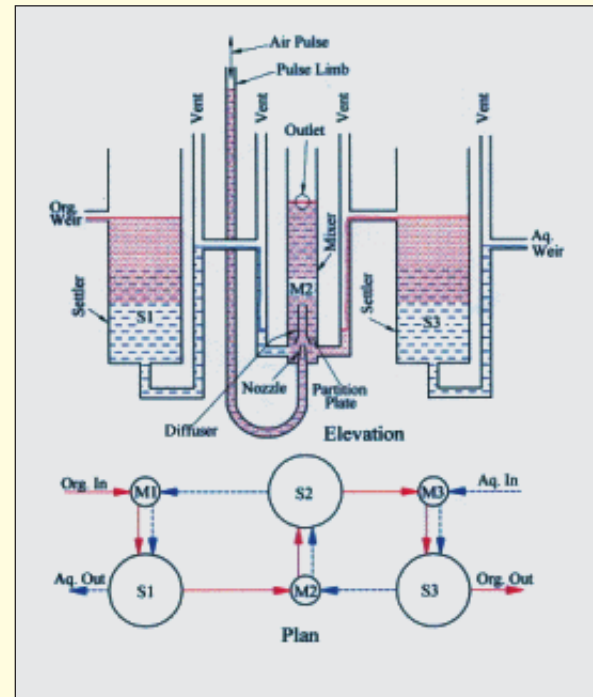
R&D for FBR fuel reprocessing requires development and interfacing of many complex technologies. The core areas of reprocessing R&D activities are; a) process development, b) equipment development and c) modeling. In addition major contribution from other groups of IGCAR are in the following areas; a) basic chemistry and NDA and on line monitoring, b) NDT techniques and remote handling, c) materials and d) instrumentation and control.

Process development includes development of process flow



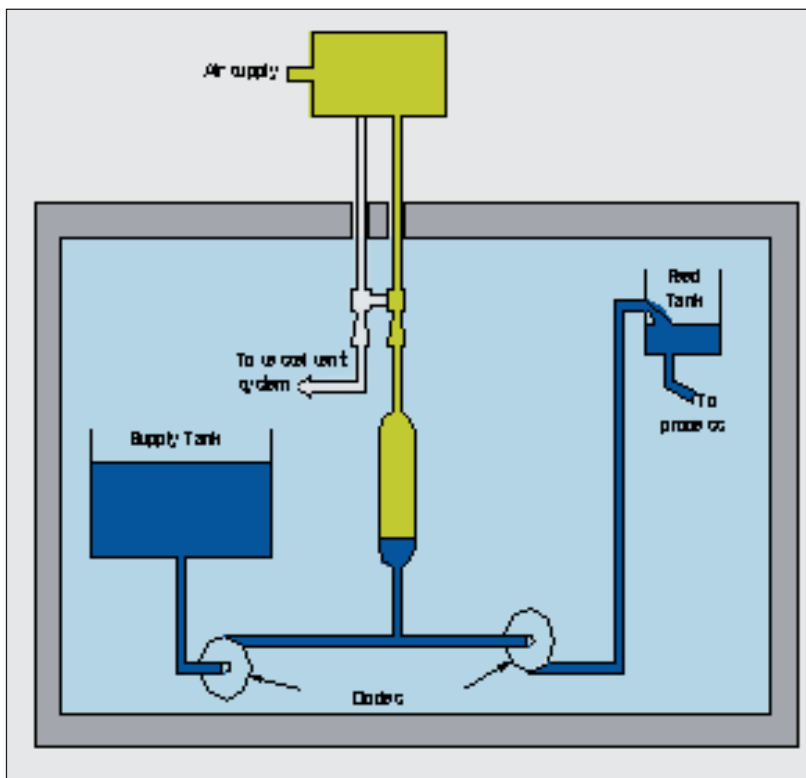
Rotary semi-continuous dissolver

sheets with conventional and novel extractants, electrochemical methods for fuel dissolution and conditioning, acid killing and recovery, partitioning studies, solvent wash with salt free reagents and recovery of solvent through evaporation and distillation and new analytical methods. Plant specific flowsheets will be tested in glove boxes with un-irradiated fuel using small ejector mixer settlers before the adoption of flow sheet for plant use. Electrolytic dissolver studies will continue for future oxide fuels and in-situ electrolytic conditioning of Pu will be developed.



High efficiency ejector mixer settler



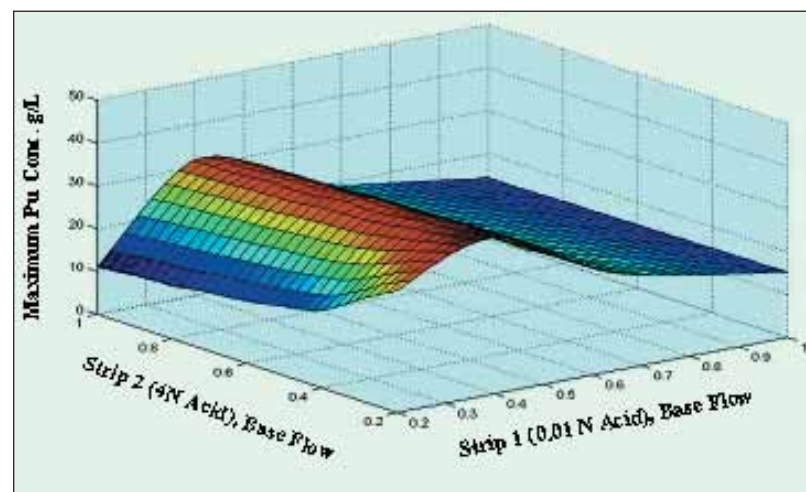


Fluidic diode pump
top: schematic, bottom : under evaluation in plant

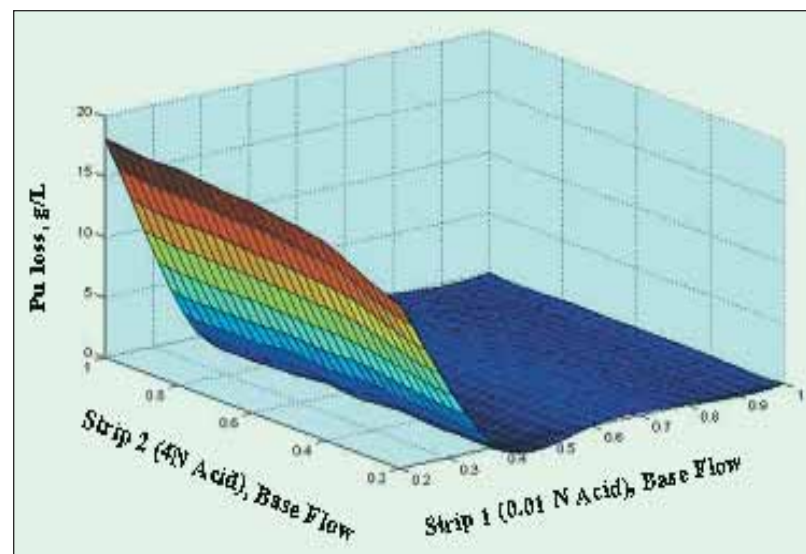


Electrochemical acid destruction has been already demonstrated and will be applied for plant scale. Process equipment include the development of laser based dismantling system, development of single pin chopper or multi pin chopper depending on the processing capacity, development of rotary semi continuous dissolver and advanced clarification systems. Most of these prototypes will be tested in the next two to three

years. Among the solvent extraction equipment centrifugal extractors and novel high efficiency air pulsed ejector mixer settlers (HEMS) are being used for various applications. Development of zero maintenance fluidic pumps for radioactive fluids based on fluidic diodes and reverse flow diverters (RFD) have reached an advance stage and will be deployed in the future plants.

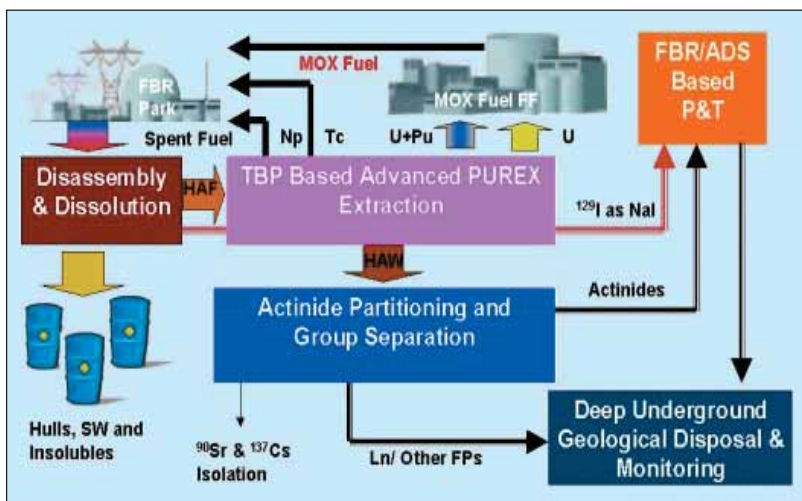


Variation of maximum Pu concentration inside the HC contactor: SIMPSEX results for 70%U+30%Pu flowsheet with feed concentration of 72 g.L⁻¹ (U+Pu)



Variation of Pu loss in the HC contactor: SIMPSEX results for 70%U+30%Pu flowsheet with feed concentration of 72 g.L⁻¹ (U+Pu)

The reprocessing group at IGCAR had developed considerable strength in modeling. These activities resulted in the in-house development of the codes SIMPSEX and PUTHEX for U-Pu separation with high Pu concentrations and U-Th separation. A separate analysis code was also developed for



Conceptual flowsheet for closing the FBR fuel cycle

handling in-situ electrolytic partitioning of U-Pu for high Pu loaded organic solutions. SIMPSEX code is extensively used in the simulation of solvent extraction behavior in LMC. Other major areas of modeling include physical properties of solutions of interest to reprocessing, solubility models that include the third phase limits for U, Pu and Th systems. The density equations developed for solutions containing uranium, plutonium and nitric acid have been extensively used by other groups internationally. This particular form of the density equation is likely to develop into an industrial standard.

Work is in progress for development of remote sampling and analytical systems, development of remote visual systems. The strength of IGCAR in NDT techniques is used for remote inspections in the reprocessing plants and development efforts are focused for remote repair techniques. Another important activity is the generation of basic distribution data with alternate solvents and resins and also laboratory scale flowsheet development runs with air pulsed ejector mixer settler banks. As a part of the solutions for long-term nuclear waste problem, work on minor actinide separations is initiated. Also pyrochemical methods are being pursued as alternate to aqueous reprocessing. Direct electrowinning of uranium metal and UO_2 has been demonstrated on a laboratory scale.

New advanced materials such as Ti-Ta-Nb alloys are pursued for fabricating equipment in the highly oxidizing and high temperature areas like dissolver and evaporators. IGCAR is also working on development of dissimilar metal joints and electrode coating for the electrolyte processes employed in reprocessing.

Non-Destructive Assay (NDA) Techniques

Non-destructive assay techniques play an important role in all the fuel cycle operations especially fuel fabrication, fuel reprocessing and waste management areas to comply with the strict regulations in force. The NDA methods do not demand elaborate sample preparation steps. The emphasis for developing NDA techniques was laid as early as late sixties.

The NDA techniques developed into mature technology include, 1) A neutron well coincidence counter (NWCC) for the assay of plutonium in nuclear waste packets, special nuclear material (SNM) storage containers, and fuel sub-assemblies as well as pins, 2) Passive and active neutron based assay methods for the detection of fissile elements in leached hulls during the reprocessing of PHWR and FBR fuels, 3) Passive gamma spectrometric assay systems for the quality assurance of fuel pins, 4) On-line measurement of high plutonium streams using neutron collar, 5) On-line alpha monitors to detect plutonium losses in re-conversion supernatant streams and 6) Passive gamma based assay systems including segmented gamma scanning with attenuation correction for both detection and characterisation of SNM in waste.

India, in the comity of nations that have a significant programme of nuclear power, has been consistent in declining the option of open-ended fuel cycle and in choosing the closed-fuel cycle. This choice is based on considerations of national energy security in the long run and is also predicated by concern for safety and security of the fissile material that is present in the spent fuel. Closing the fuel cycle is a prudent choice that ensures optimum return for our natural uranium investment in the national programme. While we approach the second stage of the nuclear power programme, the significance of this momentous decision on the part of our founding fathers assumes a larger significance. Recycling of nuclear materials, not merely as a principle but as a practice, is a contemporary reality.

NUCLEAR WASTE MANAGEMENT

Importance of safe and economic management of radioactive wastes, for the development of nuclear power and other beneficial uses of radioactive isotopes in medicine, industry and research in the country was realized from the inception of the research and development programme at the Atomic Energy Establishment in the country, at Trombay (AEET) in the mid-fifties. The development of processes and practices in radioactive waste treatment and disposal had closely followed the development in other areas of the nuclear fuel cycle, apart from operation of research reactors and application of isotopes in the country .

The Beginning – Processing of Low and Intermediate Level Wastes

The seeds for radioactive waste management in India were sown with the setting up and commissioning of an Effluent Treatment Plant (ETP) at Trombay in 1967. This demonstrated, for the first time in India, efficient treatment and management of radioactive waste from CIRUS. With the commissioning and

operation of this plant, technologies involving chemical treatment and ion-exchange as applicable to radioactive waste were established. Wastes from radiochemical laboratories in Trombay were collected in carbuoys and immobilized in cement matrices at ETP. This plant continues to be operational and has adapted itself to low level effluents from reprocessing and other laboratories at Trombay as well. The operational experience at Trombay has gone a long way in designing and setting up of effluent treatment facilities for power reactor wastes and low level effluents from reprocessing plants

While India's first Pressurised Heavy Water Reactor, RAPS-I was constructed and commissioned in active collaboration with Canadians at Rawatbhata near Kota, Rajasthan, liquid effluent management facility for this reactor was set up in parallel and commissioned solely as an indigenous effort. Though this facility continues to serve the intended purpose, it had to be augmented by a solar evaporation facility, which is a preferred mode of evaporation for larger volumes of waste with low activity at sites, which have favourable climatic conditions such as high ambient temperature, low humidity and high wind velocity.

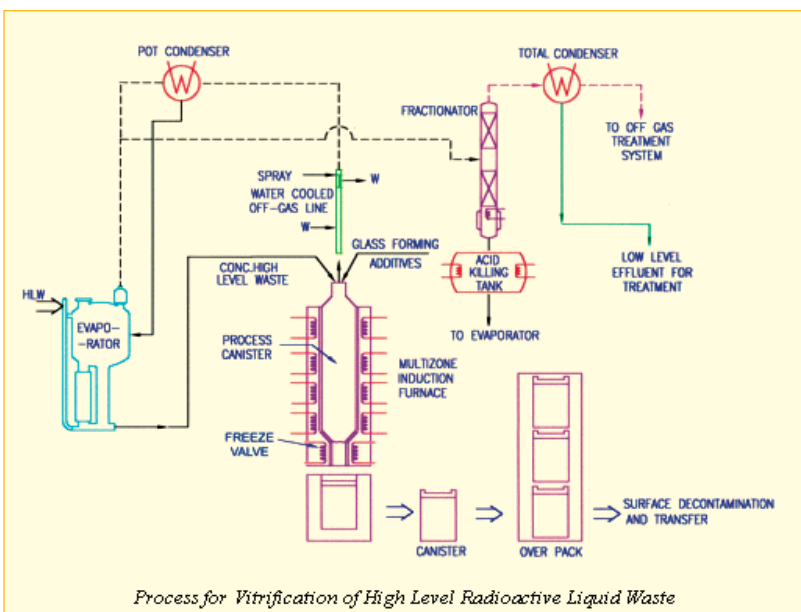
After gaining initial experience at ETP, and with the commissioning of waste management unit at RAPS I, the programme was expanded in the seventies to meet the challenge of waste management at new sites such as Kalpakkam and Narora as well as retrofitting at Tarapur. Augmentation of waste management facilities for TAPS I & II was carried out as the in-built radwaste system provided by GEC, USA was found inadequate with respect to decay of iodine and regenerant waste that required treatment. Accordingly Tarapur Radwaste Augmentation Plant (TRAP) was set up incorporating two tanks of 1000 Cubic Meters each serving as delay tanks for ^{131}I followed by chemical treatment and ion-exchange. This augmentation plant has resulted in substantial reduction of activity discharge to the environment. Based on the indigenous development of processes, technologies, equipment and assemblies, radioactive waste management facilities have been set up at various sites. Some of these facilities are in operation for more than 40 years. Valuable experience has been gained in the design, construction,



Cone mixer deployed at waste immobilisation plant, Trombay



Reverse osmosis plant at waste immobilisation plant, Trombay



Process for Vitrification of High Level Radioactive Liquid Waste

Process for Vitrification of High Level Radioactive Liquid Waste

operation and maintenance of such facilities.

Presently all the Pressurised Heavy Water Reactors (PHWR) at various sites like Rajasthan, Kalpakkam, Narora, Kakrapar and Kaiga have dedicated waste management systems consisting of liquid effluent segregation system, treatment and conditioning system and waste disposal system for discharge/disposal of waste. These plants employ treatment processes that include filtration, chemical treatment, ion exchange, steam evaporation, solar evaporation and membrane processes. This has resulted in discharge of effluents below limits set by regulatory authorities.

High Level Waste (HLW) Management

In view of the importance of fuel reprocessing in the Indian context, the challenges that could be imposed for high level liquid waste management were well recognized. Accordingly, in the early seventies, R&D activities on management of high level waste were directed to develop and characterise a number of alternative glass matrices suitable for immobilisation of HLW generated as first cycle raffinate in the PUREX process, from the Indian reprocessing units on the one hand and also develop, evaluate and perfect conditioning processes and techniques on the other.

In line with the international practice, a three-stage programme for the management of high-level waste was evolved.

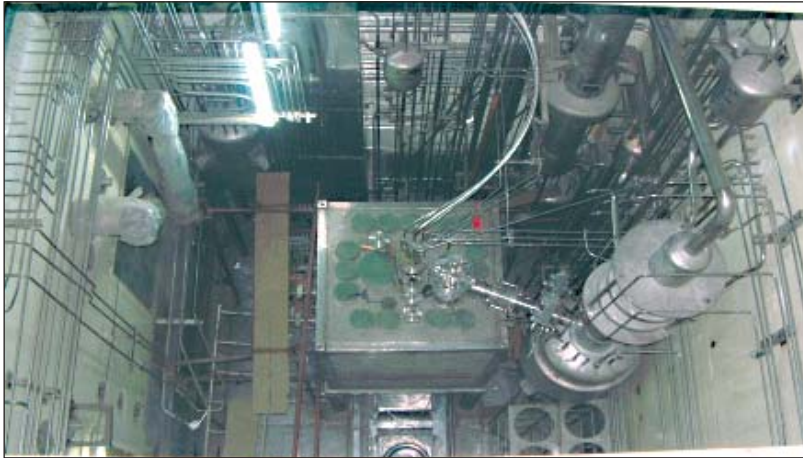
1. Conditioning of the highly radioactive liquid wastes wherein radionuclides present in the aqueous stream are immobilised in suitable matrices that are inert, thermodynamically stable, highly durable (resistant to chemical/aqueous attack), etc. and in turn contained in high integrity storage units which are subsequently overpacked.

2. Interim storage under surveillance and cooling of over packs containing conditioned wastes for periods ranging up to 30 years to allow reduction in decay heat to a level acceptable for geological disposal on the one hand and to ensure integrity of the waste form and its packaging on the other before a commitment is made for their irretrievable disposal.

3. Disposal in deep underground repository such that at no stage potentially hazardous radioactive materials are recycled back into human environment in concentrations that can subject man to a risk considered unacceptable.

Vitrification

Worldwide, borosilicate glass has been chosen as a suitable matrix for immobilizing HLW. The process that was initially chosen for development and evaluation of the process was based on Atomization Suspension Technique commonly known as AST. A pilot plant was developed and operated. Many difficulties were encountered during the course of operation of this pilot plant. Despite all the problems, evaluation of the overall performance of the plant and the basic process could be carried



Top view of process cell housing compact ceramic melter at Tarapur

out with a moderate degree of success thereby meeting the primary objective set forth. The operational experience gained did help in obtaining a better understanding of the problem and valuable information and data on many components and sub-systems which has gone a long way towards designing and setting up of subsequent plants.

The seventies witnessed a general shifting of emphasis, worldwide, from calcination and melting processes like AST to simpler concepts like pot-glass process. Like most other countries engaged in the field, attracted by the apparent simplicity of the pot glass process, India too initiated



Remote welding of high level vitrified waste canister

developmental work on similar lines during the early seventies. In fact, encouraged by early success in exploratory investigations carried out on a laboratory scale, a plan was drawn to set up a facility for further development of the process on a plant scale. The aim was to carry out a thorough evaluation of the process in conjunction with the needed auxiliaries and fine tune the same before committing the facility for processing of actual radioactive wastes from a nearby reprocessing unit on a routine basis. Thus the first Waste Immobilization Plant (WIP) at Tarapur took birth by the early eighties.

During the eighties, ceramic melters using joule heating emerged as an attractive alternative to metallic melters and work was initiated in India as well. Use of ceramic melter for vitrification of HLW has distinct advantage of higher throughput on account of continuous operation and better product durability due to higher achievable processing temperature. Based on these developmental efforts, an industrial scale ceramic melter based vitrification facility is under commissioning at Tarapur.

The vitrification process begins with metering of pre-concentrated waste and glass-forming additives in the form of a slurry into the process vessel located in a multi-zone furnace. The level of liquid waste is indicated by the temperatures sensed by the thermocouples located at different heights. The calcined mass is fused into glass at about 1223 K and is soaked at 1223 – 1273 K for eight hours to achieve homogenization. The molten mass is then drained into a stainless steel canister by operating the freeze valve. The canister filled with vitrified waste product (VWP) is allowed to cool slowly in an insulated assembly. This is then welded remotely by Pulse-TIG method. An elaborate off-gas cleaning system consisting of condenser, scrubber, chiller, demister and absolute High Efficiency Particulate Air (HEPA) filter is used to treat the gas before discharge through a 100 m tall stack to the atmosphere.

Multi-cell multi-compartment concept has been adopted at our vitrification plants so as to facilitate segregation of equipment and ease of maintenance. The process cells are equipped with state-of-the-art remote handling systems such as servo-manipulators, CCTV cameras and remote welding machine. Remote handling, robotics and automation are extremely essential for operation and routine maintenance of waste management plants handling high level of radioactivity. With the participation of Indian industry, all essential equipment and

assemblies for such applications have been developed and the country is self-sufficient at present in this regard. Development efforts are underway to improve the technology and to make available suitable technology for decommissioning of such plants whenever need will arise.

The second vitrification facility is also presently operational at BARC. The third Waste Immobilisation Plant being set up at Kalpakkam is designed for the treatment and conditioning of high level liquid waste generated during reprocessing of irradiated fuel from PHWR and FBR.

Interim Storage of Vitrified Waste

After setting up of WIP at Tarapur, the first Solid Storage and Surveillance Facility (SSSF) co-located with a vitrification plant was also conceived and constructed. Removal of decay heat from the overpack is achieved by natural convective ventilation induced by a 100 m high stack. Air-cooling system has been designed on the basis of storage unit geometry, array design, filling pattern and stack dimension. This is an inherently self-regulating system and takes care of the changes in decay heat. The cooling system ensures that the temperature within the vitrified waste product, under no circumstances, exceeds the softening point of the vitrified mass.

Geological Disposal

The role of an underground research laboratory for the development of technologies/methodologies leading to ultimately setting up a pilot repository was visualized by DAE in early eighties. The Indian programme on geological repository commenced in early eighties with underground experiments in an abandoned section of a gold mine at a depth of 1000 m at Kolar Gold Fields (KGF). The investigations were mainly directed towards development of methodology for in-situ assessment of thermo- mechanical behaviour of the host rock (amphibolite) and to develop and validate the mathematical models.

Based on such efforts, today, selection of a few suitable sites for development of site-specific Underground Research Laboratory (URL) possibly leading to setting up of a pilot repository is being pursued where extensive geo-scientific investigations and other state of the art methods and technologies will be tried and tested.



Thermo-mechanical experiment in underground chamber at Kolar Gold Fields

Solid Waste Disposal

The preliminary work on low-level solid waste disposal started in 1962 at south site of BARC where the first laboratories of radiochemistry and isotope production had started functioning. This was followed by setting up of near surface disposal facility at BARC near Gamma garden, now known as Radioactive Waste Storage and Management Site. Depending upon the various categories and levels of radioactivity, the solid and solidified wastes are emplaced in different disposal modules viz. Stone Lined Earth Trenches, Reinforced Concrete Trenches, Tile Holes etc. These wastes are normally from reactor operations, reprocessing plants and spent radiation sources. The safety, mainly achieved by isolation, is attained by placing barriers around the radioactive waste in order to restrict the release of radionuclides into the environment. The barriers can be either natural or engineered and an isolation system can consist of one or more barriers. A system of multiple barrier approach adopted has given greater assurance of isolation and helped in minimizing release of radionuclides to the environment.

Over the years, considerable experience and expertise has gone into refining and improving the design and construction of these disposal modules. Provisions for monitoring and surveillance are incorporated in the design of the disposal

facility. Bore holes are provided at appropriate locations and the groundwater samples are monitored periodically. Radiation survey of the entire site is carried out at predetermined intervals. The Reinforced Concrete trenches are provided with inspection pipes to monitor the inside condition after closure of the trench. Expertise has been developed for life assessment of Reinforced Concrete Trenches to ensure their integrity over a long period of time.

As a national policy, each nuclear facility in India has its own Near Surface Disposal Facility (NSDF) co-located. There are seven NSDFs currently operational within the country. These NSDFs have to address widely varied geological and climatological conditions.

Treatment of Gaseous Waste

Very efficient gas cleaning techniques have been developed indigenously, on the basis of R&D initiated at Trombay in the mid-sixties, and are being successfully employed in all the nuclear installations. Having successfully developed and deployed HEPA filters, BARC went on to produce combined particulate and iodine filters, charcoal impregnated sampling filters, and filter banks through sustained R&D efforts. These are currently in routine use in nuclear power plants. The techniques employ different types of wet scrubbers such as ventury, dust and packed bed scrubbers, cyclone separators, high-efficiency low-pressure drop demisters, chillers and HEPA filters to practically retain most of the particulate radionuclides.

Research and Development in Waste Processing Technology

Compaction of compressible waste was initiated in the late sixties using low capacity balers. These have now given way to compactors of 1000 tonnage capacity, resulting in very high volume reduction of metallic and other compressible wastes.

In India, decades of experience and expertise exist in the field of safe management of radioactive waste. It is recognized that the technologies currently adopted are adequate. However, there is a need to improve these technologies so as to enhance process performance and meet future challenges. Development and induction of cross-cutting technology have to be adopted not only to meet the challenges posed by approach to near-zero discharges but also to address recycle and recovery of



Compaction unit for radioactive waste drums and HEPA filters

valuable resource from these wastes leading to a positive impact on the environment. Besides, development in waste management areas have to keep pace to meet the requirements of radioactive waste likely to be generated from advanced fuel cycles for fast breeder reactors and advanced heavy water reactors.

Adoption of cesium-selective Resorcinol Formaldehyde Polycondensate Resin (RFPR) for treatment of alkaline intermediate level waste also enables the recovery of ^{137}Cs in kilocurie quantities. In a plant being set up at WIP, Trombay, the recovered ^{137}Cs will be further processed and immobilized in a vitreous matrix for use as the radiation source in blood irradiators.

The removal of nitrates present in reprocessing effluents is essential before such effluents can be discharged to the environment after treatment for removal of radioactivity. Efforts are presently underway to set up a flow-through bioreactor as a demonstration facility for this purpose.

Alternate vitrification technology based on the cold crucible induction melting is under study to address various requirements such as high temperature availability, compatibility with new matrices such as glass-ceramics, etc. A bench-scale cold crucible induction melter has been developed to demonstrate glass melting and pouring. Based on this feedback, a pilot scale melter is under design and development to establish melter start-up procedure, simulated liquid waste feeding and melter operational stability.

Studies on long-term evaluation of vitrified waste product under simulated conditions have been conducted in specially designed hot cells at Solid Storage and Surveillance Facility,



Glass pouring from cold crucible melter

Tarapur. The studies involve (a) core-drilling of high level waste product from statistically selected canisters, (b) sample preparation from core-drilled pencils, (c) studies for properties like homogeneity, thermal stability and chemical durability and (d) effect of components of repository like granite and back-fill material as well as corrosion products on leach rate of vitrified product. Some of the leach rate experiments have been continued for more than 700 days. The presence of secondary phases on the waste product and their effect on chemical durability has also been studied.

As a first step towards realization of the partitioning and transmutation option, a flow sheet has been developed for setting up of a pilot scale demonstration facility for partitioning of actinides from HLW. Laboratory runs carried out with actual high-level waste has led to partitioning of more than 99.99 % of actinides. A demonstration plant for partitioning of HLW is being setup .

Partitioning of HLW permits the use of ceramic waste forms as a special matrix for conditioning of selected waste streams in parallel with the established vitreous matrices. A stage-wise programme for ceramic matrix development is being pursued.

Synthetic Rock (Synroc)

Crystalline matrices such as synthetic rock (or Synroc) are considered as futuristic alternatives to borosilicate glass for the immobilization of HLW. Synroc is far superior to glass in chemical durability, and is considered especially suited for fast reactor HLW, which would contain very high levels of actinides and noble metals. It is typically an assemblage of four titanate minerals, viz., zirconolite ($\text{CaZrTi}_2\text{O}_7$), perovskite (CaTiO_3), hollandite ($\text{BaAl}_2\text{Ti}_6\text{O}_{16}$) and rutile (TiO_2). The waste stream is treated with calculated amounts of external additives (TiO_2 , ZrO_2 , Al_2O_3 , BaO and CaO in chemically active forms) and processed by hot pressing or hot isostatic pressing (HIP) at 1473 K to form a dense assemblage of the four mineral phases. These phases now contain the radwaste elements as dilute solid solutions in the minerals, much like the radioactive elements are found to exist in nature for millions of years. In the so-called Synroc-C waste form meant for HLW from commercial reactors, a waste loading of up to 20 weight% can be achieved.

In a concerted effort, Synroc waste forms with simulated HLW were fabricated by hot isostatic pressing of powders synthesized via alkoxide and citrate polymer routes. The calcined powder material was mixed with 2 weight% titanium metal powder, and subjected to Hot Isostatic Pressing (HIP) in stainless steel tubes, as packed powder or pre-compacted pellets. HIP was carried out at Defence Metallurgical Research Laboratory, Hyderabad. The hard ceramic monolith was found to have attained near-theoretical density (4.3 g/cc). X-ray powder diffraction revealed the presence of the synroc minerals in the ceramic compact. Examination of the monolith by SEM showed a predominantly uniform and dense microstructure. Efforts are now on to scale up these processes and also demonstrate the same using actual high level waste.

Remote Handling and Robotics

Remote handling and robotics play a very important role in fuel cycle activities, especially those related to the back end of the fuel cycle. The general philosophy of remote handling is to employ a “work station concept” consisting of a radiation shielding window (RSW), master slave manipulator (MSM) and a cell crane. All the equipment required for these duties are manufactured and available on a regular basis indigenously and has been standardized for such applications. Models of



Servo-manipulator workstation

MSMs of capacities ranging from 9-45 kg have been indigenized with electrical indexing capabilities to improve volume coverage. Articulated MSMs required for lead/steel hot cells have been developed. Indigenous capability has also been developed and tested for a 15 kg capacity bi-lateral force reflecting electrical servo manipulator for employment in high active cells with corrosive environment. Technology of power manipulator also has been developed and is in use. Bridge mounted power manipulator of 100 kg capacity with 7- degrees of freedom configuration and telescopic boom has been designed and manufactured for reprocessing and waste vitrification plants. The technology for design/manufacture of oil filled and dry type RSWs has been established. A six-axis gantry servo robot has been developed for decommissioning of glove boxes or similar equipment. A radio-controlled Remote Inspection Device (RID) has been developed for in-service surveillance of storage tanks for high and intermediate level waste tank farms with facility for visual video feed back and for collecting swab samples This is a mobile vehicle with remote traction and steering capability. This radio-controlled vehicle is being designed as a platform for many future remote intervention

operations.

A valuable human resource base has been built up, which consists of scientific and technical personnel well versed in design, construction, operation and maintenance aspects of the fuel cycle facilities and related R&D systems. Expertise in waste management has been provided to national and international institutions including International Atomic Energy Agency on advisory capacities as well as by imparting training to member countries as and when required.

Waste management facilities at various nuclear installation sites are operating safely and successfully for more than four decades. Indian experience in management of nuclear waste from power plants, and fuel reprocessing and allied installations, is rich and comparable with international practices. By suitable treatment and conditioning of waste it has been demonstrated that the prime objective of safety of environment is fully achieved. This has been possible due to indigenous development of technologies in the areas of process, auxiliary systems, remote handling, etc. related to management of various types of radioactive wastes.