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Nuclear Engineering and Design 236 (2006) 723–729
Nuclear Engineering and Design
www.elsevier.com/locate/nucengdes

# Spent fuel reprocessing: A vital link in Indian nuclear power program

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Received 11 March 2004; received in revised form 26 September 2005; accepted 28 September 2005

**Abstract**

The success of the three stage Indian nuclear energy program is inter-linked with the establishment of an efficient closed fuel cycle approach with recycling of both fissile and fertile components of the spent fuel to appropriate reactor systems. The Indian reprocessing journey was started way back in 1964 with the commissioning of a plant based on PUREX technology to reprocess aluminum clad natural uranium spent fuel from the research reactor CIRUS. After achieving the basic skills, a power reactor reprocessing facility was built to reprocess spent fuel from power reactors. Adequate design and operating experience was gained from these two plants for mastering the reprocessing technology. The first plant, being the maiden venture, based on indigenous technology had to undergo many modifications during its operation and finally needed refurbishment for continued operation. Decommissioning and decontamination of this plant was carried out meticulously to allow unrestricted access to the cells for fresh installation. A third plant was built for power reactor spent fuel reprocessing to serve as a design standard for future plants with the involvement of industry. Over the years, spent fuel reprocessing based on PUREX technology has reached a matured status and can be safely deployed to meet the additional reprocessing requirements to cater to the expanding nuclear energy program. Side by side with the developments in the spent natural uranium fuel reprocessing, irradiated thoria reprocessing is also perused to develop THOREX into a robust process. The additional challenges in this domain are being addressed to evolve appropriate technological solutions. Advancements in the field of science and technology are being absorbed to meet the challenges of higher recovery combined with reduced exposure and environmental discharges.


## 1. Introduction

There are two fuel cycle options for long-term nuclear power production that are of relevance and under consideration at the present juncture, i.e., the once through cycle with disposal of spent fuel as such and the closed fuel cycle with reprocessing and recycle of uranium and plutonium. Both the options require efficient and safe waste management strategies.

The known Indian sources of uranium are insufficient to support the long-term Indian nuclear energy program. Closing the nuclear fuel cycle by reprocessing the spent fuel and recycle of uranium and plutonium back into reactor systems helps in exploiting the full potential of nuclear power and maximizes the resource utilization (Kakodkar, 2002).

The aim of reprocessing is to recover uranium and plutonium which form the bulk of the spent fuel for recycle in reactors and to concentrate the fission products wastes in a form suitable for long term storage. The reprocessing waste would contain mainly fission products and small amounts of transuranium elements which are immobilized in specially formulated glass matrix, sealed in metal canisters designed for long-term storage.

Reprocessing started in India in the early days of the nuclear energy program based on indigenous efforts. At present, India has three reprocessing plants to extract plutonium from spent fuel, the first Plutonium Plant at Trombay has a capacity to reprocess 60 tonne of research reactor spent fuel per year, the second and third plants at Tarapur and Kalpakkam each having a capacity to reprocess 100 tonne of power reactor spent fuel per year. The reprocessing capacity is being augmented in a phased manner to cater to the needs of the different reactor systems as per the transition of the stages.

*Abbreviations:* AERB, Atomic Energy Regulatory Board; AHWR, advanced heavy water reactor; BARC, Bhabha Atomic Research Centre; CALMIX, combined air lift mixer; CIRUS, Canada India Reactor Utility Services; FBR, fast breeder reactor; HM, heavy metal; HM/d, heavy metal per day; IAEA, International Atomic Energy Agency; IMPUREX, improved PUREX; KARP, Kalpakkam Reprocessing Plant; MWd/t, mega Watt days per tonne; MWe, mega Watt electrical; $MW_t$, mega Watt thermal; mpy, mils per year; NAG, nitric acid grade; PHWR, pressurised heavy water reactor; PREFRE, power reactor fuel reprocessing; PUREX, plutonium uranium reduction extraction; TBP, tri-*n*-butyl phosphate; THOREX, thorium uranium extraction

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doi:10.1016/j.nucengdes.2005.09.029

## 2. PUREX process

By the time India entered into the domain of reprocessing, the natural technology choice 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, totally amenable for automation and remote handling. The basic steps involved in the process are:

1. Head-end treatment involving chemical or mechanical decladding followed by dissolution of fuel in nitric acid, feed clarification and adjustment of chemical conditions of the solution for solvent extraction.
2. Co-decontamination involving extraction of uranium (as U(VI)) and of plutonium (as Pu(IV)) leaving bulk of the fission products in aqueous phase which goes as high active waste.
3. Washing/scrubbing of organic stream with nitric acid, some times using two nitric acid scrubs of different $HNO_3$ concentration to backwash fission products co-extracted with uranium and plutonium.
4. Partitioning of uranium and plutonium by selective reduction of Pu(IV) in the organic phase to Pu(III) which goes to the aqueous phase, and back extraction of U(VI) with dilute nitric acid.
5. Further purification of uranium and plutonium streams to obtain U and Pu of desired purity.
6. Treating the used solvent for its recycle.
7. Waste management.

The PUREX process has been the workhorse of fuel reprocessing for the last few decades and no other process developed before or after can claim its versatility. Today the word "PUREX" is a generic term. The process has undergone

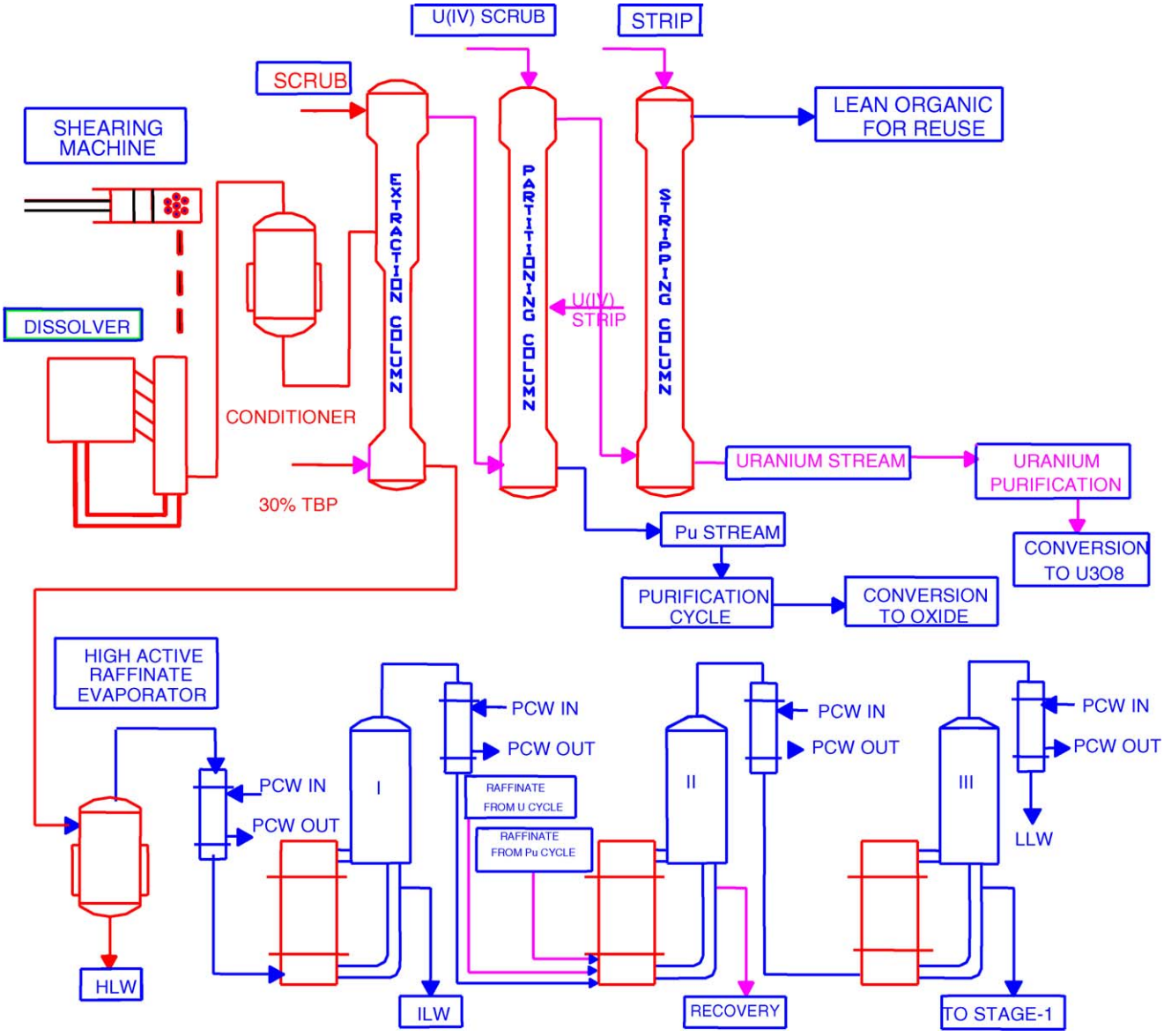


Fig. 1. PUREX process flow sheet.

Table 1
Radioactive source term of the spent fuel discharged from a 220 MWe PHWR

| Actinides g/t HM | | | Fission products Ci/t HM | | |
|---|---|---|---|---|---|
| Nuclide | 3 year cooling | 5 year cooling | Nuclide | 3 year cooling | 5 year cooling |
| $^{234}$U | 4.40E+01 | 4.40E+01 | $^{90}$Sr | 1.38E+04 | 1.32E+04 |
| $^{235}$U | 2.53E+03 | 2.53E+03 | $^{90}$Y | 1.38E+04 | 1.32E+04 |
| $^{236}$U | 6.91E+02 | 6.91E+02 | $^{95}$Zr | 5.99E+00 | 2.19E−03 |
| $^{238}$U | 9.86E+05 | 9.86E+05 | $^{95}$Nb | 1.38E+01 | 5.04E−03 |
| $^{237}$Np | 2.36E+01 | 2.37E+01 | $^{99}$Tc | 2.84E+00 | 2.84E+00 |
| $^{238}$Pu | 2.88E+00 | 2.84E+00 | $^{106}$Ru | 2.07E+04 | 5.24E+03 |
| $^{239}$Pu | 2.67E+03 | 2.67E+03 | $^{106}$Rh | 2.07E+04 | 5.24E+03 |
| $^{240}$Pu | 8.92E+02 | 8.91E+02 | $^{125}$Sb | 1.46E+03 | 8.86E+02 |
| $^{241}$Pu | 1.50E+02 | 1.36E+02 | $^{134}$Cs | 5.73E+03 | 2.92E+03 |
| $^{242}$Pu | 3.64E+01 | 3.64E+01 | $^{137}$Cs | 2.00E+04 | 1.91E+04 |
| $^{241}$Am | 2.52E+01 | 3.88E+01 | $^{144}$Ce | 2.92E+04 | 4.92E+03 |
| $^{243}$Am | 1.09E+00 | 1.09E+00 | $^{144}$Pr | 2.92E+04 | 4.92E+03 |
| $^{242}$Cm | 3.54E−03 | 1.84E−04 | $^{147}$Pm | 2.39E+04 | 1.41E+04 |
| $^{243}$Cm | 3.50E−03 | 3.34E−03 | $^{154}$Eu | 8.80E+02 | 7.49E+02 |
| $^{244}$Cm | 6.91E−02 | 6.40E−02 | $^{155}$Eu | 5.07E+02 | 3.83E+02 |

Basis: average burnup of 6600 MWd/t computed using Origen 2.

many modifications since its introduction and the flow sheets employed vary depending on the practitioners. Fig. 1. shows the typical flow sheet employed for the processing of spent fuels discharged from PHWRs with an average burnup between 6000 and 8000 MWd/t cooled for 3–5 years. The radioactive source term of spent fuel for a typical burnup of 6600 MWd/t discharged from a 220 MWe PHWR after a cooling of 3 and 5 years is given in Table 1.

Chemical aspects of various process steps are continuously under refinement to:

1. improve the recoveries of uranium and plutonium and the decontamination factors;
2. reduce the waste generated;
3. simplify the process e.g., reduced number of cycles;
4. remove the long-lived actinides and fission products from high level waste to minimize hazards of its long term storage;
5. develop alternate processes.

Side by side with changes in the chemical aspects, developments taking place elsewhere in the science and technology frontier had also its impact on the PUREX process. Equipment design, computerized data logging and control, robotics and automation, material development, etc. has taken this technology to further heights by reducing direct maintenance, providing automatic built-in safety features, reducing the radiation exposure to the working personnel, improving the project management with reduced construction time.

## 3. Evolution of reprocessing in India

The reprocessing program was launched with the design, construction and commissioning of the demonstration plant at Trombay. Preliminary design work was carried out during 1959–1961. During this period experiments with pulsed perforated columns were carried out to confirm design data. This was followed by finalization of the process and equipment design, and fabrication and installation of equipment and piping in the process cells and associated systems. The plant was commissioned in 1964 to reprocess the spent fuel from 40 $MW_t$ research reactor CIRUS. The metallic fuel elements 3.4 m long, were of natural uranium clad in aluminum. For the head-end treatment, chemical dejacketting was followed by dissolution of the fuel in concentrated nitric acid under reflux conditions. The PUREX process comprised a decontamination cycle, a partition cycle and two separate parallel cycles for the purification of uranium and plutonium. The reductant used in the partitioning stage was ferrous sulphamate solution in nitric acid medium. The final purification of plutonium nitrate solution was by ion exchange. The Trombay plant had a nominal capacity of 30 tonne HM/year. Being first of its kind, the design philosophy was completely based on direct maintenance concept. The experience in operating the plant and in assessing future requirements in reprocessing served as the basis for R&D program in the field. The successful operation of the plant also helped in providing plutonium for pursuing various programs of nuclear research and development (Prasad and Kumar, 1982).

### 3.1. Improved features in Tarapur plant

The Trombay plant gave sufficient impetus to continue R&D in the domain of reprocessing. In particular, these included solvent degradation studies, development of equipment and systems for higher plant throughput and bringing about improvement in performance, representative sampling and analysis, online instrumentation and data acquisition system for process control and operation safety. The results of these efforts were integrated into the design of the second plant constructed at Tarapur (commissioned in 1975) for reprocessing of Zircaloy clad oxide spent fuel from Tarapur and Rajasthan Atomic Power Stations. This plant uses a chop-leach technique for the head-end and uranous nitrate stabilized by hydrazine as the reductant for partitioning. Instead of the 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 needs of higher Pu throughput. Several innovations such as pneumatic pulsers in place of mechanical pulsing, air lift as a metering device for radioactive process solutions, thermosiphon evaporators, solvent wash systems for aqueous streams emanating from solvent extraction, etc. were introduced in the plant. The experience gained from the earlier plant gave enough insight into the material of construction used for critical equipment fabrication and qualification. Austenitic stainless steel variety 304 L was the choice available at that point of time. Except for the head-end, which had provision for remote maintenance of in-cell equipment, the concept used for the rest of the plant was that of direct maintenance. Several campaigns of reprocessing were carried out under international safeguards also at this facility, which provided valuable experience in material accounting practices to meet the international standards.

This plant also provided experience in the design of appropriate packages and safe in-land transportation of spent fuels, which is a vital input for locating the reprocessing facilities. All transport of radioactive materials in India is governed by guidelines issued by the statutory regulatory authority, the Atomic Energy Regulatory Board (AERB). These materials can be transported only in packages, which are designed in accordance with standards and guide lines prescribed by the AERB. These guidelines are based on the International Atomic Energy Agency (IAEA) advisory regulations for the safe transport of radioactive materials. Spent fuel assemblies from power reactors are shipped in special casks known as "Type B" packages. These casks have two functions: the containment of the nuclear materials, and protection of people and the environment from radiation. They are shielded with steel or a combination of steel and lead. These casks are tested to withstand the most severe accidents without breaking or leaking. Standards set by the AERB include dropping the cask, heating it in an intense fire, and immersing it in water. Since 1975 there have been safe shipments of spent fuel involving thousands of kilometers by rail and road.

### 3.2. Refurbishment of Trombay plant

The Trombay plant, being the first built in India for acquisition of skills required in the vital area of reprocessing, had to be subjected to decontamination from time to time in order to permit access into the process cells for trying out different concepts to optimize the process conditions. In view of the inherently corrosive environment to which process and piping were constantly exposed during the years of operation, it was considered desirable to decommission the plant for effecting the necessary replacements to extend its life. This opportunity was also utilized to enhance the capacity of the plant. The entire decommissioning program was meticulously planned to keep the personnel radiation exposures as low as possible, by training personnel in the type of operations involved and devising proper tools and equipment. Attention was paid to control and manage the wastes generated.

The decommissioning procedure specific to this plant comprised several sequential steps. The campaign of internal decontamination of equipment and piping followed multiple decontamination routes. The maximum number of equipments possible were covered in a single route to minimize the quantity of decontaminants used, so as to keep the resultant volume of radioactive liquid waste low. The range of equipment decontaminated included pulsed perforated plate columns, evaporators, condensers, ion-exchange columns, storage vessels with the associated piping, etc. After dismantling and disposal of equipment and piping, high pressure water jets, steam, chemicals, pneumatic chippers and concreting were used, as appropriate, to remove contamination or shield hot spots on cell interior surfaces. The success of the decommissioning operations could be gauged from the insignificantly low background levels of radiation fields ultimately achieved, the absence of transferable contamination on cell surfaces, and the fact that personnel exposures were well with in prescribed limits. This resulted in salvaging of most of the cells and permitted almost unrestricted access into them for carrying out fresh installation work. The feed back information and experience obtained during the execution of the above mentioned jobs once again emphasized the importance of making provisions for decommissioning to be incorporated at the design stage of reprocessing plants in future (Rao, 1978).

The major changes incorporated in the expanded plant at Trombay are introduction of metering airlifts in addition to metering pumps, steam ejectors with flooded suction, column interface controls using either airlifts or diaphragm control valves, introduction of thermo siphon evaporators, partitioning of plutonium in the very first cycle itself, use of uranous nitrate solution instead of ferrous sulphamate for partitioning, introduction of diluent wash system for aqueous streams emanating from the solvent extraction contactors, change of layout for reagent heads of metering pumps, and equipment off gas system, provision for continuous solvent treatment and also allowance for 100% redundancy for dissolver equipment. The decommissioned and augmented plant started operating from 1983 onwards.

### 3.3. Design standardization at Kalpakkam plant

After the successful operation of the power reactor reprocessing facility at Tarapur and the experience gained during the decommissioning operation of Trombay plant, the need arose to augment the reprocessing capacity to treat the spent fuel from the increased nuclear power generation. To cater to the needs of reprocessing Zircaloy clad natural uranium oxide spent fuel from Madras Atomic Power Station, a new plant was designed near the power station at Kalpakkam, with a 100 tonne HM/year capacity. The execution of this plant was carried out with the involvement of industry in the fabrication of equipment, their installation and piping work.

With the acceleration of power program and the increased spent fuel arisings, the need for fissile and fertile material recycling has to be augmented. It is essential that the projects in future are cost effective and the gestation periods will have to be shortened. Even though the engineering design of a reprocessing plant has many complex requirements with respect to

equipment, piping, ventilation, exhaust, remote handling, etc. by involving the industry in the detailed design, procurement and construction with the basic design support from the experts, the execution time can be reduced. The Kalpakkam plant design is to serve as a standard design for future plants. The design aims at availability of the plant capacity throughout the operation of the power station. The reprocessing capacity is augmented in a phased manner to cater to the fuel requirements of the reactor systems being introduced into the nuclear energy program in a "reprocess to recycle" mode.

The existing capacity of 200 tonne will be gradually enhanced to meet the fuel requirements of fast breeder reactors and advanced heavy water reactors.

## 4. Technological improvements in the reprocessing journey

Around 40 years of experience in the spent fuel reprocessing based on PUREX process has given the confidence that this technology can be successfully employed for the recovery of both U and Pu with yield exceeding 99.5%. It is reported that by fine tuning of the parameters, this can be improved to 99.8–99.9%. Even though Np inventory in the spent fuel discharged from the PHWR is low, Np recovery can also be achieved by minor changes in the flow-sheet. Feed clarification is one of the important step to obviate the interfacial activity accumulation and crud formation during solvent extraction. Substantial reduction in waste volume was achieved over the years by resorting to salt-free reagents. Replacement of ferrous sulphamate with hydrazine stabilized uranous in the partitioning cycle brought a major change in this field which reduced the corrosion problems to great extent and a much cleaner waste is being generated in this cycle. Evaporation followed by acid reduction by formaldehyde is used to reduce the high level waste volume. By resorting to these procedures the high level waste volume could be restricted to 600 L/tonne of HM processing. The overall decontamination factors for the Pu and U products from fission products exceed $10^6$ and the products are refabricated with minimum radiation protection.

Reprocessing plants are being built with high degree of remotization and automation. Elaborate safety features are incorporated in the design to obviate criticality incidents and to minimize radiation exposure. The gaseous and liquid effluents undergo thorough processing prior to the discharge to the environment. Over the years, improved practices has brought down the environmental discharges to much lower levels than the stipulated values.

In the technological domain many significant changes have taken place. The uranous production by electrolysis has been standardized to optimum current efficiency and conversion by resorting to new electrode material like titanium substrate insoluble anode and titanium cathode. Catalytic reduction of uranium(VI) to uranium(IV) over finely divided platinum dust is also being developed as another effective method of U(IV) production.

The in situ reduction of U(VI) to U(IV) for the partitioning of Pu using specially designed mixer settlers and pulsed columns have also been demonstrated on a pilot plant scale using titanium cathode and titanium substrate insoluble anode. The operating experience gained from these pilot studies will be used for firming up the design of such systems for future reprocessing plants.

Homogeneous sampling of key measurement points is vital to the nuclear material management strategy and needs careful engineering design to achieve the target value. Vacuum assisted airlift operated sampling design has been developed in house to meet the sampling requirement of the PUREX process with inbuilt safety features to control the vacuum and the flooding of the sample. This system was thoroughly tested during the IAEA safeguarded campaigns at Tarapur. Further efforts are on to automate the sampling system to minimize the exposure.

Head-end systems have undergone major engineering developments. An indigenous shearing machine was developed and installed at the Kalpakkam Reprocessing Plant (KARP) incorporating many design improvement. Changes have also been incorporated in the clapper door assembly to improve reliability and plant throughput. The design of shearing machine and clapper assembly is being standardized for the subsequent reprocessing plants. Laser assisted dismantling of the fuel bundle followed by single pin chopping is also being developed. Developments are under way to design and fabricate indexing casks and automated charging of the fuel bundles into the shearing machine magazine in an effort to increase the throughput with minimum radiation exposure to operator.

Another area which essentially determines the operating life of a reprocessing plant is the corrosion resistance of the material of construction. Some recent developments in corrosion and radiation resistant materials have resulted in reliable performance with reduced maintenance. The materials for reprocessing plant have to ensure resistance against intergranular corrosion in nitric acid at different concentrations and elevated temperatures. Earlier plants had worked on low carbon grade 304 L stainless steel. Factors other than sensitization that can cause intergranular corrosion in oxidizing nitric acid environments are the presence of active inclusions and segregation of Si and P to grain boundaries. With this understanding of the corrosion of austenitic stainless steel in nitric acid environments, tighter specifications have been formulated. These challenges have led to the development of nitric acid grade (NAG) type 304 L stainless steel in the country. The NAG stainless steel produced in India for reprocessing applications has already achieved a corrosion resistance rate as low as a value as 10 mpy.

## 5. Irradiated thorium processing

For India, the building up of fissile material inventory at a fast pace is a prerequisite for the early introduction of thorium in the fuel cycle as natural thorium does not contain any fissile component. In tune with the increase in fissile inventory with advancement in the nuclear energy program, various steps towards implementation of thorium fuel cycle were also initiated. To meet the challenges of thorium based fuel cycle, R&D

efforts are directed towards extractive metallurgy of thorium, fuel fabrication and utilization in reactors, reprocessing of irradiated thorium for $^{233}$U recovery and studies on $^{233}$U based reactor systems. Irradiation studies with thoria started in the blanket region of research reactor CIRUS. Reactor Physics studies led the use of thoria in the flux flattening during the initial startup of PHWR. With the introduction of AHWR into the power program in the beginning of the next decade, where a Th–Pu MOX is used as fuel, an altogether new dimension will be added to reprocessing requiring three component separation. In addition to this, the well known radiological aspects of thorium fuel cycle also need to be addressed.

### 5.1. THOREX process

The THOREX (thorium uranium extraction) process for treating irradiated thorium has several special features. The first and foremost challenge is the dissolution of the inert thoria fuel pellets. $^{233}$Pa formed by (n, γ) reactions of $^{232}$Th decays with a 27 day half life and its complete decay is to be ensured prior to reprocessing for $^{233}$U and thorium recovery. The (n, 2n) reactions encountered during the irradiation of Th give rise to long lived $^{231}$Pa and rather short lived $^{232}$U (68.9 years) with its hard beta gamma emitting daughter products. The yield of $^{232}$U depends on the burn up and the neutron spectrum in the reactor.

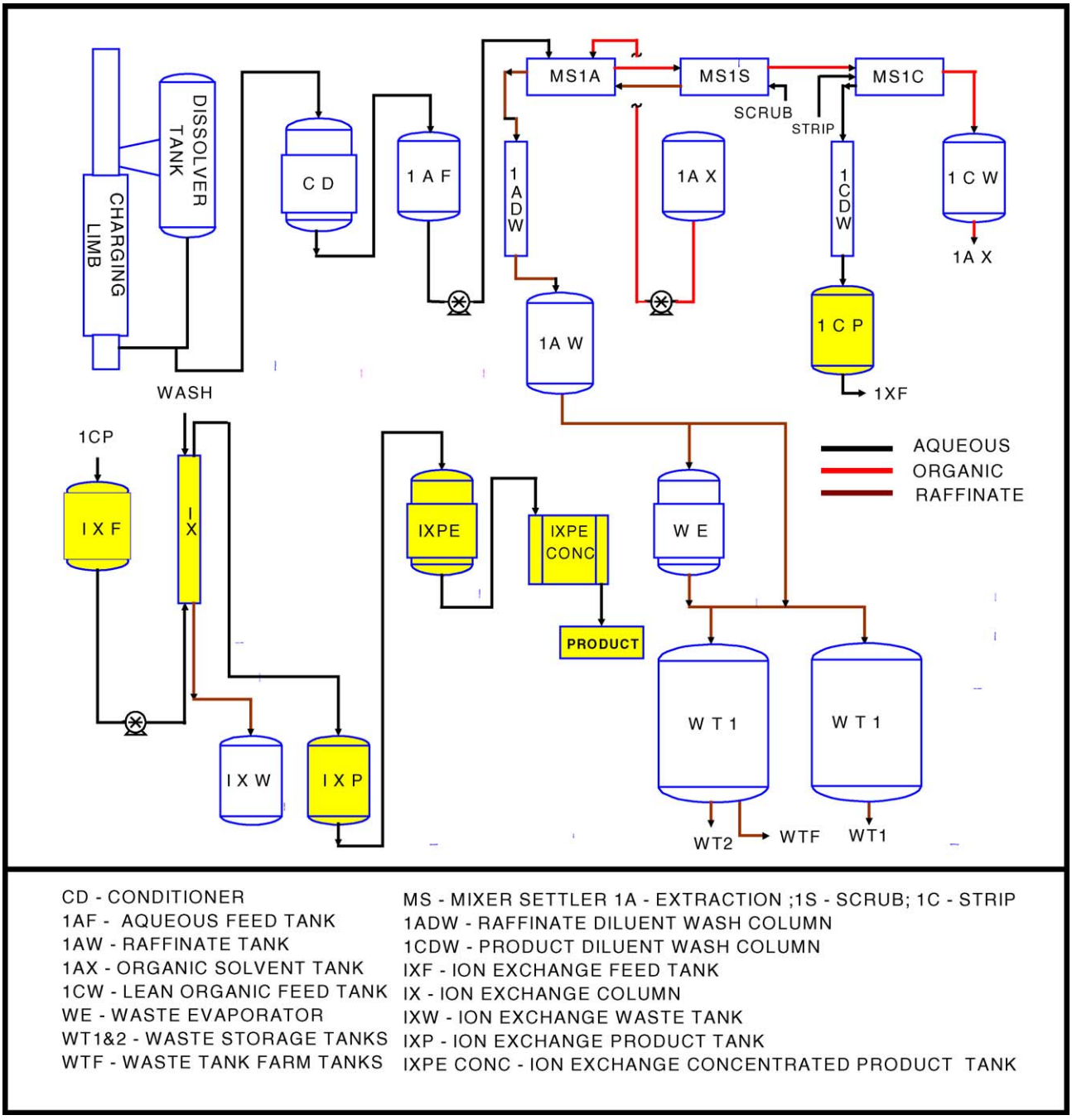


Fig. 2. THOREX process flow sheet.

The $^{232}U$ and its daughters in the separated $^{233}U$ and the $^{229}Th$ and $^{228}Th$ in thorium contribute to the radiation dose of these products. The Thorex HLW contains $^{231}Pa$ and to some extent $^{237}Np$ as the main long-lived actinides with long-term environmental impact. At the time of starting the THOREX process, the basic laboratory data were scarce. Most of the batch and counter-current extraction and stripping data were generated by in-house studies. Different flow-sheets were developed for the recovery of $^{233}U$ alone or for both $^{233}U$ and thorium using TBP-diluent systems. The chemical decladding and acid dissolution of aluminum clad fuel rods were adopted. The addition of fluoride ion was required in the dissolution of thorium oxide by nitric acid. This enhances the corrosion of stainless steel equipments. This problem is controlled by the addition of appropriate amounts of aluminum to complex the excess fluoride ion. If the thoria is clad in zirconium based alloys or in SS as expected in PHWRs and FBRs, then mechanical chop-leach process is adopted for dissolution.

The versatile extractant TBP in hydrocarbon diluent still remains the best choice for the extraction of both $^{233}U$ and thorium or for the selective extraction of $^{233}U$ alone. Depending on the requirement whether both thorium and $^{233}U$ are to be recovered or only $^{233}U$ is to be recovered, the TBP content in the diluent (usually Shell Sol-T, dodecane or n-paraffin) to be used as extractant varies. Generally, 3–7.5% has been used as extractant when $^{233}U$ alone is to be recovered. In this technique, $^{233}U$ is extracted preferentially with TBP leaving bulk of the thorium in the raffinate. A maximum of 8 g/L thorium is co-extracted along with $^{233}U$ in this step, when 5% TBP is used. The organic phase is scrubbed with 1–2 M nitric acid to remove the co-extracted thorium. Thus, the organic phase finally contains only $^{233}U$, and its final thorium contamination depends on the number of scrub stages provided and their efficiency in scrubbing out thorium. The $^{233}U$ from the organic phase is finally stripped with 0.01 M $HNO_3$ to recover the extracted uranium.

The final $^{233}U$ product was subjected to further purification by anion exchange process and the uranium product eluted was precipitated as diuranate and ignited to oxide and stored.

An engineering scale facility is in operation at Trombay for the processing and recovery of $^{233}U$ from CIRUS and Dhruva irradiated thorium fuel rods on a regular basis. The modifications felt necessary from the pilot plant have been incorporated in the design of equipment and in the choice of process flowsheets. Specially designed CALMIX mixer settlers have been chosen as contactor equipment. The scrub section has been extended sufficiently to provide for adequate scrubbing of thorium from the uranium loaded organic. The tail-end purification step by anion exchange has now been replaced by the cation exchange purification process developed to overcome the drawbacks felt in the earlier campaign. The thorium oxalate precipitation and separation route is being installed as laboratory back-up procedure for final purification of $^{233}U$. A schematic diagram of the THOREX flow sheet is given in Fig. 2.

## 6. Challenges in reprocessing

Even though PUREX process is well entrenched to meet the present and near future challenges, it is being constantly improved to achieve higher goals of recovery and reduced waste volumes. All round development in the field of Science and Technology has its implications in PUREX process too. Currently PUREX process is a multi cycle process and is being developed into a single cycle process on the lines of IMPUREX process, a technology demonstration process with advanced features. Some of the salient features of this process are computer control of extraction column with on-line monitoring and automated control of process parameters by sophisticated and sensitive instrumentation systems. Development of a single cycle process needs maintenance-free short residence contactors and annular pulse columns which are safe with respect to criticality with provisions for near real time accountancy of fissile and fertile actinides and in situ electrochemical, photolytic or other Pu and Np reduction and partitioning techniques for an ultimate salt-free process. Minimization of losses to waste, improvements in solvent quality, crystallization of uranyl nitrate, direct denitration of products to oxide, co-processing and co-conversion for fuel development, recovery of useful isotopes like $^{237}Np$, $^{99}Tc$, Pd, etc., separation of long lived actinides and fission products under partitioning and transmutation option with a general reduction of waste volumes are some of the challenging R&D tasks ahead. In processing high burn-up, short cooled fast reactor fuels, multiple recycle of Pu would pose problems in the process and in the handling of the final products due to alpha, gamma and neutron radiations arising from increasing amounts of $^{238}Pu$, $^{241}Am$ and Cm isotopes. The radiolytic degradation of TBP may also be severe. Under these circumstances pyroprocesses may become an alternative to aqueous reprocessing.

Irradiated thorium processing also needs developmental efforts for better head-end techniques for decladding and dissolution of Zircaloy and stainless steel clad Thoria, corrosion resistant materials of construction, modifications in process to deal with irradiated $ThO_2$–$PuO_2$ fuels, behaviour of $^{231}Pa$ in the process and its long term impact in waste, radiological safety in handling $^{233}U/^{232}U$. The state of the art remote handling technology and robotics will have their maximum application because of the associated activity of $^{232}U$ daughter products.

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