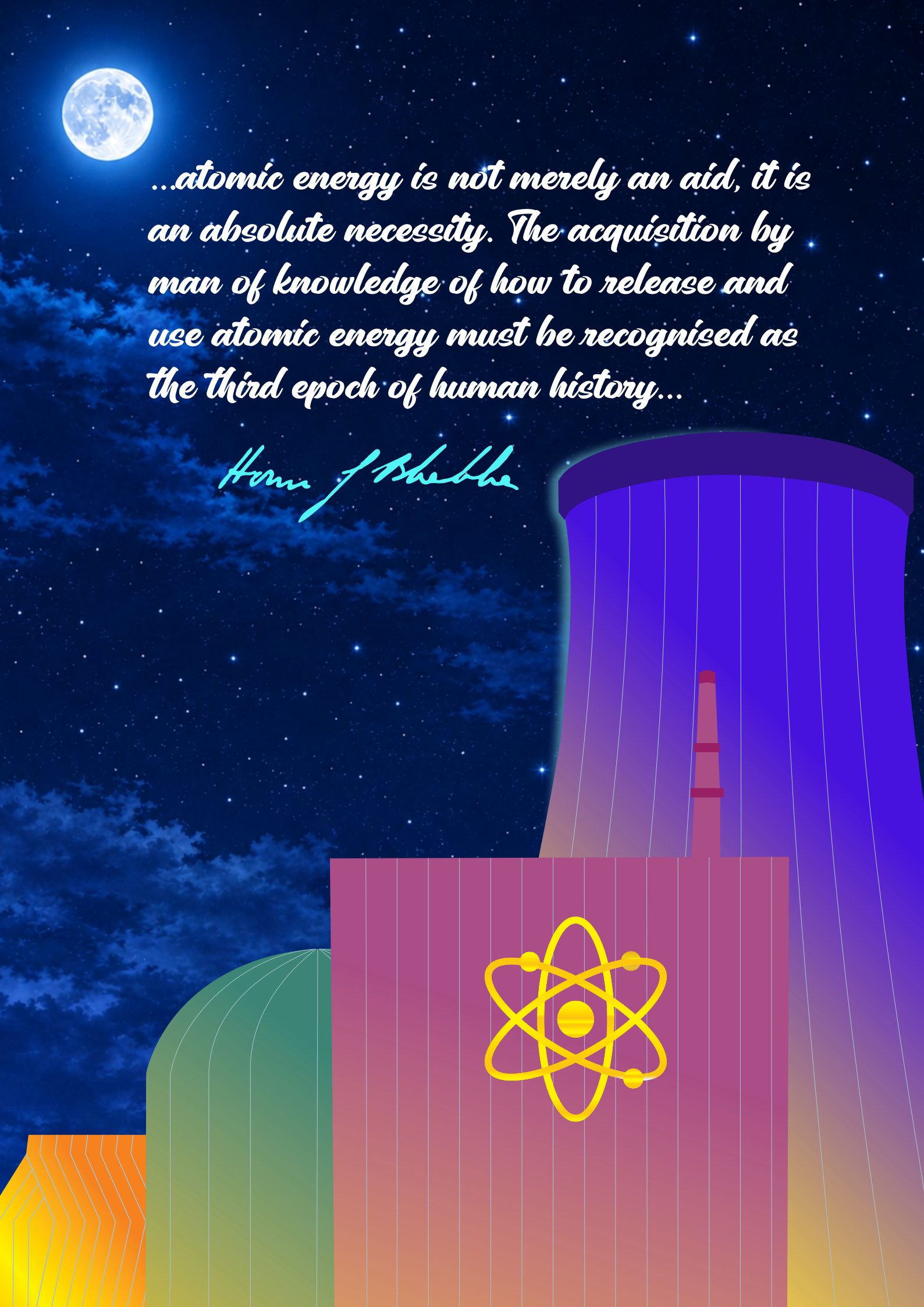

...atomic energy is not merely an aid, it is an absolute necessity. The acquisition by man of knowledge of how to release and use atomic energy must be recognised as the third epoch of human history...

EXPANDING Reprocessing Expertise in Nuclear Fuels

विस्तारशील पुनर्संसाधन: परमाणु ईंधन में विशेषज्ञता

India has established indigenous reprocessing capabilities for spent fuel from Pressurised Heavy Water Reactors (PHWRs), supported by decades of experience in spent-fuel handling, head-end operations, solvent-extraction based separation, and management of highly radioactive process streams. This experience provides a technological foundation for extending reprocessing expertise to advanced nuclear fuels. Research and development (R&D) are essential to address challenges in fuel dissolution, separation chemistry, material compatibility, and waste management, thereby demonstrating process feasibility, recovering valuable fissile materials, minimizing secondary waste, and ensuring safe, efficient, and economically viable reprocessing. This article presents experimental studies addressing challenges in processing uranium silicide fuel and spent uranium precipitate filters.

Reprocessing of Uranium Silicide Fuel: Uranium silicide fuel dispersed in an aluminium matrix and clad in aluminium alloy is used in the Apsara-U reactor. This plate-type fuel contains low-enriched uranium (LEU) and differs significantly from fuel types for which reprocessing technologies are established in India. Uranium silicide dispersion fuel is also proposed for the upcoming High Flux Research Reactor and Isotope Production Reactor at Visakhapatnam. Given the absence of prior reprocessing experience with this fuel, investigations into its dissolution and solvent extraction behaviour were necessary. Process optimisation also requires removal of undissolved silicon, selection of suitable solvent compositions and partitioning agents, and control of dissolved silica, which can cause crud formation during solvent extraction.

Extensive laboratory-scale experiments identified an

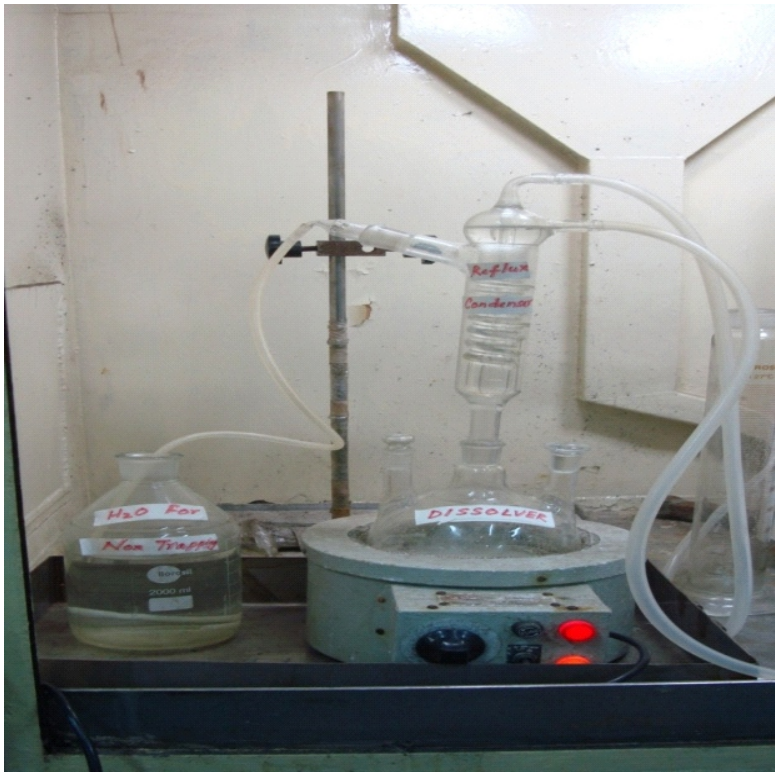


aqueous PUREX based route for silicide fuel reprocessing. Both acidic dissolution and alkali treatment followed by acid dissolution were investigated. The resulting process scheme comprises nitric acid dissolution, silica removal, feed conditioning, and PUREX-based solvent extraction.

Since dissolution is a major challenge, detailed studies investigated the fuel's behaviour in nitric acid. Aluminium constitutes more than 60% of the fuel, and its dissolution was identified as the rate-determining step. Kinetic studies of Al 6061 cladding and aluminium matrix dissolution examined the effects of nitric acid concentration and surface area to support engineering-scale process development. Kinetic modelling indicated zero-order dependence on aluminium concentration, characteristic of a surface-controlled reaction, and second-order dependence on nitric acid concentration. The dissolution rate decreased above 8 M nitric acid because of passive-layer interference. Dissolution below 2 M nitric acid should be avoided because of the potential generation of hydrogen gas. Further studies using a 5 L pilot-scale experimental setup are planned to validate laboratory findings.

Spent Uranium Precipitate Filters - Development of a Process Scheme: At the Fission Moly Plant (FMP), BRIT, irradiated uranium aluminide plates dispersed in an aluminium matrix and clad in aluminium are processed to recover Mo-99, which is used to generate Tc-99 for radiodiagnostic applications. This process generates spent uranium precipitate filters (SUPFs) containing unused uranium as sodium diuranate (SDU), alongside fission products such as ^{144}Ce, ^{144}Pr, and ^{90}Sr. Because these plates have relatively low irradiation levels, most feed uranium remains unused and reports to the SUPF, necessitating recovery and reuse. The process involves dissolution, uranium recovery, and purification. SDU dissolution in nitric acid is rapid. However, uranium recovery presents challenges owing to the considerable U-235 fraction and required product purity. The low uranium inventory per batch necessitates handling in appropriately shielded glove boxes (GBs).

Two approaches were investigated for uranium recovery and purification. The first



Dissolution setup for Al 6061.

employed solvent extraction using 15% TBP and simulated solutions. Laboratory-scale mixer-settler units were used to evaluate uranium recovery and fission-product decontamination. Decontamination factors achieved in a single cycle using eight extraction stages and four scrubbing stages are presented in the corresponding table.

Solvent extraction within GBs presents challenges in organic-solvent handling and maintaining the required leak tightness. Accordingly, membrane-based recovery studies were initiated through an intra-BARC collaboration to develop a functionalised-resin-based process. A styrene-divinylbenzene resin functionalised with phosphonate groups (PHP) showed promise. From 4 M HNO_3, the resin exhibited a uranium uptake capacity of 80 mg/g and very low fission-product uptake, with Kd values below 0.5 mL/g.



Laboratory scale mixer-settler for recovery and decontamination of U.



Pilot scale experimental setup for remote handling of unirradiated simulated SUPF.

Uranium was readily stripped using 1 M ammonium carbonate. The corresponding decontamination factors at different acidities are presented in the relevant table. The resin maintained its loading capacity and stripping behaviour over six cycles and showed radiation resistance, with no observed changes following irradiation to 10 kGy. Considering operational simplicity and improved decontamination performance, the resin-based process will be adopted for uranium recovery from dissolved SUPF.

Pilot-Scale Experimental Set-up: A pilot-scale experimental system has been established at Trombay to demonstrate remote handling of unirradiated simulated SUPF and dissolution of its uranium precipitate. The system comprises a 5 L batch dissolver, material-handling equipment, and an off-gas cleaning system. Uranium recovery from the dissolved solution will be investigated using resin-based methods. The setup will subsequently be extended to validate dissolution studies using unirradiated silicide fuel and uranium recovery from the resulting solution. It will support process-parameter optimisation for designing equipment to process silicide fuel and SUPFs.