IANCAS
Bulletin

An Overview of Fast Reactor Fuel Reprocessing

INDIAN ASSOCIATION OF NUCLEAR CHEMISTS AND ALLIED SCIENTISTS
C/o Radiochemistry Division, BHABHA ATOMIC RESEARCH CENTRE, MUMBAI – 400085
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An Overview of Fast Reactor Fuel Reprocessing

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Editor, IANCAS, sincerely thanks Guest Editors, Authors and Secretary, IANCAS, as well as IANCAS Executive Committee for their efforts/support in bringing out this thematic and informative bulletin.

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Every reasonable effort has been made to ensure accuracy. However, any errors, omissions, or typographical inaccuracies that may remain are inadvertent and are sincerely regretted.

Editor, IANCAS

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Dr. V. Jayaraman
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Focus

We are aware that the key to the success of three-stage nuclear power programme, envisaged by Dr.Homi Jahangir Bhabha, is dependent on the advancement of fast reactor programme and the associated fuel cycle. Indira Gandhi Centre for Atomic Research being the hub of fast reactor programmes, the Reprocessing Group at IGCAR is engaged in the development and establishment of technologies for reprocessing of all kinds of fast reactor fuels, since inception.

The Compact facility for Reprocessing Advanced fuels in Lead cells (CORAL) is being operated at IGCAR for the past two and half decades, which is processing the mixed carbide spent fuel from FBTR. Based on the knowledge and experience gained in CORAL operation, the Demonstration fast reactor Fuel Reprocessing Plant (DFRP) was constructed and dedicated to the nation by Honourable Prime Minister in January 2024. Subsequently, the facility was hot-commissioned in April 2024, which firmly established the indigenization of advanced technology and our technical capability of reprocessing fast reactor fuels, in-line with the Atmanirbhar Bharat. Furthermore, a commercial scale plant is being constructed at the Fast Reactor Fuel Cycle Facility (FRFCF), Kalpakkam for reprocessing the spent fuel discharged from PFBR, and the technology required for reprocessing is being transferred to FRFCF.

To commemorate the success of Indian fast reactor fuel reprocessing programme, the challenges met and solutions reached right from the establishment of a facility to cold-commissioning and hot-commissioning, it is proposed to divulge the science and technology of fast reactor fuel reprocessing in the thematic bulletin on “An Overview of Fast Reactor Fuel Reprocessing”. In this context, several experts from reprocessing group have authored and shared the experience on fast reactor fuel reprocessing in this bulletin.

I take this opportunity to commend the Guest Editors for compiling this important volume and all contributing authors for submission of articles in time. I also thank IANCAS and the present executive committee members especially Dr Ragunatha Acharya, Editor, IANCAS for the keen interest in publishing the bulletin on fast reactor fuel reprocessing. I am happy to note that this book will serve as a valuable reference material for researchers, engineers, and students engaged in fast reactor fuel reprocessing and chemistry of actinides.

V. Jayaraman

Director, Reprocessing Group

From Editor's Desk

Under Indian three stage nuclear power program, fast reactors using Pu-based fuel are very important towards achieving faster and higher MWe. India has successfully operated FBTR at IGCAR using (U, Pu) carbide fuel, and this expertise has led to go for mixed oxide fuel in PFBR by BHAVINI at Kalpakkam. In this way, Indian program on FBRs or fast reactors are expected to be successful in paving the way from all the future power reactors for achieving the target of 100 GWe by 2047 to help India become a developed nation i.e., "Viksit Bharat". At the same time, India follows the closed fuel cycle wherein the spent fuel can be reprocessed to get unspent fuel materials ready to use in existing/future FBR. Back-end fuel cycle is very important for utilization of all nuclear resources and sustainable growth of nuclear power.

The current issue of the IANCAS bulletin titled *"An overview of Fast Reactor Fuel Reprocessing"* gives overview of various developmental works towards successful reprocessing of fast reactor fuel mainly at IGCAR, Kalpakkam. I sincerely thank Dr. K. A. Venkatesan and his team helping us in timely publication of this very important bulletin by IANCAS. It is a very comprehensive as well as a complete bulletin with twelve chapters namely (i) Fast reactor reprocessing programme and the way forward, (ii) Operational experience of processing high burnup mixed carbide spent fuel at CORAL facility, (iii) Experiences in commissioning and operation of demonstration fast reactor fuel reprocessing plant, (iv) Process chemistry aspects in fast reactor fuel reprocessing, (v) Process design aspects of fast reactor fuel reprocessing plants, (vi) Design of robust solvent extraction flow-sheet for reprocessing of fast reactor fuels by indigenous computer code and simulation, (vii) Design and development of Head end equipment for fast reactor fuel reprocessing, (viii) Unique and special equipment for reprocessing fast reactor fuel, (ix) Concurrent Development and commissioning of indigenous customized hot cell equipment for accelerated plant construction, (x) Challenges faced and developments in electrical, electronics and instrumentation, (xi) Design and commissioning aspects of ventilation and off-gas systems in fast reactor fuel reprocessing plants, and (xii) Waste management aspects in fast reactor fuel reprocessing.

On behalf of IANCAS, I thank all the authors of the chapters for sharing their articles for making this bulletin an important one. I would like to sincerely thank the Guest Editors of this Issue Dr. K. A. Venkatesan, Associate Director, Reprocessing Group, IGCAR, and Shri Geo Mathews and Shri J. Kodandaraman, Reprocessing Group, IGCAR, for their untiring efforts and also for coordinating with all the contributors of articles in bringing out this bulletin in a very meticulous way. IANCAS is thankful to Dr. V. Jayaraman, Director, Reprocessing Group, IGCAR, for his support and also for his message under FOCUS. Sincere thanks to all the EC members and Secretary of IANCAS for the continued support and time to time suggestions to bring out such types of thematic bulletins relevance for DAE and society.

We sincerely thank Dr. A. K. Mohanty, Chairman, AEC & Secretary, DAE, and Shri Vivek Bhasin, Director, BARC, for their support towards IANCAS activities and IANCAS publications. On-behalf of IANCAS, sincere thanks and acknowledgement to Board of Research in Nuclear Sciences (BRNS), DAE; Chairman, BRNS; Scientific Secretary and Head, BRNS, for their support and granting funds towards IANCAS publications of such thematic bulletins.

Dr. Raghunath Acharya
Editor, IANCAS
OS & Head, IRAD, RC&IG, BARC

President's Message

Dear Friends,

Greetings for a Happy and Prosperous 2026!

I am happy to learn that the second volume of the IANCAS Bulletin for 2026 on the very important topic of spent nuclear fuel reprocessing involving fast reactors entitled "*An Overview of Fast Reactor Fuel Reprocessing*" is being brought out which will be of great interest to many IANCAS members as well as scientists / engineers working in the back end of nuclear fuel cycle.

The articles compiled in this thematic bulletin encompass a comprehensive range of subjects pertaining to fast reactor fuel reprocessing. The first article of the bulletin presents an overview of the fast reactor reprocessing programme, outlining its current status and future directions. Subsequent articles address critical domains such as process chemistry, process design, and modelling and simulation, which collectively constitute the scientific and engineering foundation of fast reactor reprocessing technology. A few articles are devoted to the design and development of head-end systems, advanced process equipment, and hot cell facilities, highlighting systematic efforts toward ensuring reliable remote handling, maintainability, and operational safety. Waste management, process and radiation instrumentation, as well as ventilation and off-gas treatment systems have also been discussed in details. The latter articles are focused on the processing of high burn-up mixed carbide spent nuclear fuel and commissioning and operation of the demonstration reprocessing plant. The contributions presented in the bulletin indicates the multidisciplinary character of fast reactor fuel reprocessing encompassing chemistry, chemical engineering, mechanical engineering, instrumentation, radiation protection, waste management, plant operations, etc.

The guest editors, Dr. K. A. Venkatesan, Shri Geo Mathews, and Shri J. Kodandaraman have made a marvellous job in compiling the articles, and IANCAS sincerely appreciates their contributions. I am thankful to Dr. V. Jayaraman, Director, Reprocessing Group, IGCAR for supporting the efforts made by the team contributing to this bulletin. A special word of appreciation goes to the members of the IANCAS Executive Committee, particularly the Vice-President (HQ), Secretary, and Editor, for their dedicated efforts. The team acknowledges the financial support of BRNS, DAE.

Dr. P. K. Mohapatra
Former Director, RC&IG, BARC
President, IANCAS

From Secretary's Desk

Indian Association of Nuclear Chemists and Allied Scientists (IANCAS) was founded in the year 1981 with an objective of popularizing nuclear and radiochemistry, applications of radioisotopes, and nuclear techniques among the scientific community in India. For this purpose, IANCAS is continuously organizing seminars, national workshops and publishing periodic thematic bulletins focused on fundamentals of nuclear and radiochemistry, and applications of radioisotopes in education, research, agriculture, medicine and industry. With active participations of the life-members, IANCAS has become one of the leading scientific associations for popularizing the subject of nuclear and radiochemistry across the country.

IANCAS through its various outreach programmes motivate the young researchers and scientists to apply nuclear and radiochemistry based methods in their respective research field. In addition, IANCAS life-members through IANCAS activities motivate students to pursue a career in the field of nuclear science. For the promotion of nuclear science among the researchers, IANCAS has instituted three Awards; (i) Dr. M. V. Ramaniah Memorial Award, (ii) Dr. Tarun Datta Memorial Award, and (iii) Prof. H. J. Arnikar Best Thesis Award. All these three awards are conferred annually. The details of these awards are available at the IANCAS website (www.iancas.org.in).

IANCAS conducts national workshops and outreach programmes at Indian universities and colleges. IANCAS also regularly publishes thematic bulletins on the topics directly related to the nuclear science and technology with the financial support from BRNS, DAE. These bulletins are made freely available at IANCAS website (www.iancas.org.in) for download.

In the series of IANCAS bulletins, the present bulletin titled *"An Overview of Fast Reactor Fuel Reprocessing*" aims at giving details of different aspects of fast reactor fuel reprocessing. IANCAS thanks to all contributors/authors of the articles for sparing their valuable time and in making such an important bulletin possible. I sincerely thank, Dr. R. Acharya, Editor, IANCAS, and the Guest Editors Dr. K. A. Venkatesan, Associate Director, Reprocessing Group, IGCAR and Shri Geo Mathews and Shri J. Kodandaraman, Reprocessing Group, IGCAR, for their efforts in bringing out this important thematic bulletin.

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Preface by Guest Editors

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Fast reactor fuel reprocessing is vital for closing the fast reactor fuel cycle. This step is crucial for utilization of all nuclear resources and sustainable growth of nuclear power in India. The present bulletin titled, "*An overview of Fast Reactor Fuel Reprocessing*", presents a comprehensive account of the experience and progress of fast reactor reprocessing since its inception, and the ongoing activities.

The articles included in this bulletin cover a broad spectrum of topics related to fast reactor fuel reprocessing. The opening contribution outlines the overall fast reactor reprocessing programme and future directions, setting the context for the subsequent technical discussions. This is followed by the contributions on process chemistry, process design, and modelling and simulation aspects of fast reactor reprocessing, which form the scientific and engineering foundation of the technology. Several articles focus on the design and development of head-end systems, advanced process equipment, and hot cell facilities, highlighting efforts performed towards reliable remote handling and maintenance.

Equally important areas such as waste management, process and radiation instrumentation, and ventilation and off-gas systems are also covered, emphasizing the integrated approach required for reprocessing plant design. The bulletin further includes articles detailing operational experience from the processing of high burn-up mixed carbide spent fuel, and new learnings from the commissioning and operation of demonstration reprocessing plant, for building the future reprocessing facilities.

These contributions reflect the multidisciplinary nature of fast reactor fuel reprocessing, encompassing chemistry, engineering, instrumentation, radiation protection, waste management, and operations. The collective knowledge presented here is expected to serve as a useful reference for ongoing and future activities in this field. Therefore, we express our sincere gratitude to all the authors for their valuable contributions. We also thank IANCAS and the present executive committee members especially Dr. Raghunath Acharya, Editor, IANCAS, who encouraged us to bring out a thematic bulletin covering all aspects of science and technology of fast reactor reprocessing.

We hope this bulletin would be useful to scientists, engineers, and students engaged in nuclear fuel cycle activities and to readers seeking an overview of fast reactor fuel reprocessing. The information presented in this bulletin is intended for academic and knowledge dissemination purposes only and has no commercial value.

Fast Reactor Reprocessing Programme and the Way Forward

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Abstract

India's nuclear power programme is structured around a long-term three-stage strategy, aimed at optimal utilization of limited indigenous uranium resources and the large thorium reserves available in the country. The first stage is based on pressurized heavy water reactors (PHWRs), while the second stage envisages fast breeder reactors (FBRs) to multiply fissile material inventory and bridge the transition from uranium to the thorium-based third stage reactors. Therefore, the fast reactors occupy a central position in India's nuclear roadmap that enables efficient plutonium utilization and sustained growth of nuclear power capacity. Since closing of fuel cycle is essential for meeting long-term strategy needs, reprocessing of fast reactors fuels is vital for the recovery & recycle of plutonium. Fuel reprocessing not only supports resource sustainability but also contributes to waste minimization and long-term radiotoxicity reduction.

India has adopted an indigenously developed, stage-wise fast reactor fuel reprocessing programme aligned with the progressive deployment of fast reactors. The programme commenced with pilot-scale facilities for carbide fuel, followed by demonstration-scale reprocessing plants to validate process technologies and hot cell equipment. These efforts have culminated in the development of commercial-scale fast reactor fuel reprocessing plants designed to support the Prototype Fast Breeder Reactor (PFBR) and future fast reactors. This article reviews the evolution, technological challenges, and strategic significance of India's fast reactor fuel reprocessing programme, and its way forward.

Keywords: *Fast reactor fuel reprocessing program, Plutonium Recycle, FBTR fuel reprocessing, PFBR fuel reprocessing*

1. Introduction

India's nuclear power programme is founded on a closed fuel cycle strategy aimed at maximizing energy extraction from limited indigenous uranium resources while progressively transitioning to thorium utilization. This objective is realized through the well-established three-stage nuclear power programme. The first stage employs pressurized heavy water reactors (PHWRs) using natural uranium oxide fuel, which generate plutonium in spent fuel. The second stage is based on fast breeder reactors (FBRs) utilizing plutonium-based fuels to breed additional fissile material, thereby expanding the fissile inventory and enabling sustained growth of nuclear capacity. The third stage envisages the deployment of thorium-based reactor systems, supported by fissile material bred in fast reactors. Within this framework, the fast reactors constitute the bridge between uranium-based and thorium-based fuel cycles.

Fast reactors operate with a fast neutron spectrum, offering superior neutron economy and enabling high breeding ratios as well as efficient transmutation of minor actinides such as neptunium, americium, and curium. These characteristics allow fast reactors to significantly improve fuel utilization and reduce the long-term radiotoxicity and heat load of high-level radioactive waste. However, fast reactor fuels are characterized by high fissile content, high burnup, and elevated decay heat and radiation fields at discharge. As a result, the deployment of fast reactors is intrinsically linked to the development of advanced fuel reprocessing technologies capable of handling short-cooled, high-activity spent fuel under fully remote conditions.

A closed fuel cycle is an essential pre-requisite for fast reactor systems, as the economic and strategic benefits of breeding can only be realized through timely recovery and recycle of plutonium and uranium. India has therefore adopted an integrated approach wherein reactor development, fuel fabrication, and reprocessing technologies are pursued in parallel. This approach has guided the evolution of India's fast reactor fuel reprocessing programme, which is based primarily on aqueous reprocessing using advanced variants of the PUREX (Plutonium Uranium Redox EXtraction) process, adapted to the specific challenges posed by fast reactor fuels.

The Indian fast reactor fuel reprocessing programme has evolved in a stage-wise manner, beginning with pilot-scale facility designed to establish a proof-of-concept for reprocessing of fast reactor fuels. This early facility enabled the development of specialized head-end processes such as chopping, and high-efficiency dissolution, as well as solvent extraction flowsheets capable of handling high plutonium concentrations. Significant emphasis was placed on the development of corrosion-resistant materials, criticality-safe equipment design, and high-reliability remote handling systems to address the demanding radiological and chemical environments. Building upon this experience, demonstration-scale reprocessing plants was established to validate process flowsheets, equipment performance, and plant operability under conditions representative of reactor-scale throughput. The facility also provided valuable operational data on solvent degradation, fission product behaviour, and waste stream management. The accumulated experience has culminated in the design and construction of commercial-scale fast reactor fuel reprocessing plants intended to support the Prototype Fast Breeder Reactor (PFBR) and future fast reactor deployments.

This article reviews the technical evolution of India's fast reactor fuel reprocessing programme, with particular emphasis on process development, engineering challenges, and operational experience. The role of reprocessing in enabling a sustainable closed fuel cycle for fast reactors and its strategic importance within India's nuclear energy roadmap are discussed.

2. Fast Reactor Fuels and Reprocessing Challenges

Reprocessing of fast reactor spent fuel presents a set of challenges that are considerably more demanding than those encountered in thermal reactor fuel reprocessing. These challenges arise primarily from the high fissile content, high burnup, compact fuel geometry, and short cooling times characteristic of fast reactor fuels. Addressing these issues requires specialized process flowsheets, equipment designs, and safety strategies.

High Fissile Content and Criticality Safety

Fast reactor fuels contain significantly higher plutonium concentrations compared to thermal reactor fuels. Typical mixed oxide (MOX) fuels for fast reactors contain plutonium in the range of 20–30 wt%. During reprocessing, such high fissile inventories lead to elevated risks of nuclear criticality, particularly in aqueous process streams where moderation effects can enhance reactivity. Criticality safety is therefore a dominant design consideration in fast reactor fuel reprocessing plants. This is addressed through a combination of geometric control using slender vessels, annular tanks, and critically safe contactors, as well as strict control of fissile concentration and neutron moderation. The high plutonium concentration in solvent extraction systems also increases the risk of third phase formation in the organic solvent, which can lead to phase instability, plutonium holdup, and potential criticality concerns. Careful control of solvent composition and plutonium loading is essential to mitigate third phase formation.

High Burnup and Solvent Degradation

Fast reactor fuels are irradiated to high burnup levels, resulting in elevated concentrations of fission products and transuranic elements in the spent fuel. The intense radiation fields associated with such fuels accelerate radiolytic degradation of the solvent used in extraction processes, namely tributyl phosphate (TBP) diluted in hydrocarbon diluents. Radiolysis and hydrolysis lead to the formation of degradation products such as dibutyl phosphate (DBP), which adversely affect phase separation, extraction efficiency, and product purity.

To minimize solvent damage, fast reactor reprocessing plants employ short-residence-time contactors such as Centrifugal extractors. This limits the radiation dose absorbed by the solvent while maintaining high mass transfer efficiency. In addition, solvent clean up systems and optimized operating conditions are essential to sustain long-term solvent performance under high radiation environments.

Short Cooling Time and High Heat Load

Fast reactor spent fuel is often reprocessed after relatively short cooling periods to reduce the doubling time. As a result, the spent fuel and resulting process streams exhibit high decay heat and intense radiation fields. This necessitates enhanced thermal management, robust shielding, and high-reliability cooling systems throughout the reprocessing plant. Hot cell equipment and instrumentation must be designed to operate reliably under high radiation conditions, further increasing engineering complexity.

Slender Fuel Pins

Fast reactor fuel assemblies typically consist of slender fuel pins with small diameters and tight lattice spacing. While advantageous from a reactor physics standpoint, this geometry poses significant challenges during head-end reprocessing operations. Mechanical chopping of slender pins requires high-precision equipment capable of maintaining minimum deformation of cladding material and without excessive generation of fines.

Requirement of Alpha-Tight Cells and Containment

Fast reactor spent fuel and reprocessing streams contain large quantities of plutonium and higher actinides, resulting in significant alpha activity. Alpha contamination poses a serious radiological hazard due to the long half-lives and high radiotoxicity of transuranic elements. Consequently, reprocessing of fast reactor fuel must be carried out in alpha-tight cells that provide robust containment against the release of alpha-active materials.

Alpha-tight cells are designed with leak-tight boundaries, typically employing stainless steel liners, welded joints, and controlled ventilation systems with multi-stage HEPA filtration. Operations within these cells are performed using remote handling systems such as master–slave manipulators, in-cell cranes, and shielded transfer systems. Maintaining negative pressure gradients ensures inward airflow and prevents contamination spread. The design and operation of alpha-tight cells significantly influence plant layout, maintenance philosophy, and operational reliability, and represent a critical requirement unique to fast reactor fuel reprocessing.

3. Evolution of Fast Reactor Fuel Reprocessing in India

Process and Equipment development

The initial phase of India's fast reactor fuel reprocessing programme focused on the development and optimization of reprocessing flowsheets and equipment suitable for high plutonium-content fuels. Early efforts addressed fundamental challenges such as dissolution of fast reactor fuels, solvent extraction behaviour at high plutonium concentrations, criticality-safe equipment design and contactors with low residence time.

Special attention was given to head-end processes, including mechanical chopping of slender fuel pins, development of high-efficiency dissolvers, and optimization of nitric acid dissolution chemistry for carbide and mixed oxide fuels. Parallel efforts were undertaken to adapt and refine aqueous reprocessing flowsheets based on modified PUREX processes, with emphasis on controlling third phase formation, minimizing solvent degradation, and ensuring effective separation of uranium and plutonium.

Equipment development was a critical component of this phase, encompassing the design of critically safe vessels, short-residence-time contactors, corrosion-resistant materials, and reliable remote handling systems. Extensive laboratory- and engineering-scale studies provided the data necessary for scaling up processes and formed the technical foundation for subsequent pilot-scale operations.

Figure 1. View of CORAL facility.

Pilot-Scale Reprocessing Facility - CORAL

The next stage involved the establishment of pilot-scale reprocessing facility – CORAL an acronym of Compact Reprocessing of Advanced Fuel in lead cell (Figure 1). The facility was aimed to validate the process and equipment that were developed for fast reactor fuel reprocessing.

The CORAL facility has reprocessed FBTR fuel with burnup as high as 155 GWd/t, and achieved very good recovery and decontamination factors. The major equipment developed namely the single pin chopper, titanium dissolver, high speed centrifuge and the centrifugal extractors were validated. Operational experience gained from CORAL provided valuable insights into the behaviour of the process and equipment. The lessons learned were instrumental in guiding design improvements for the subsequent larger-scale plants.

Demonstration Scale Reprocessing Plant- DFRP

The Demonstration Fast Reactor Fuel Reprocessing Plant (DFRP) (Figure 2) has been conceived with the objective of enabling regular reprocessing of spent fuel from the Fast Breeder Test Reactor (FBTR) and demonstrating the reprocessing of spent fuel from the Prototype Fast Breeder Reactor (PFBR). The plant can handle both the mixed carbide and mixed oxide fuels.

As a regularly operating reprocessing facility, DFRP incorporates several systems aimed at waste minimization and resource recovery. These include raffinate and waste evaporators for volume reduction of liquid waste streams, solvent treatment and regeneration systems, diluent washing systems, and dedicated recovery cycles for the retrieval of fissile materials from waste streams. These features contribute to improved material recovery and reduced waste.

Figure 2. External view of DFRP facility commissioned in the year 2024.

Commercial-Scale Reprocessing Plants for PFBR and Beyond

The Fast Reactor Fuel Cycle Facility (FRFCF) represents the commercial-scale realization of India's fast reactor fuel reprocessing programme and is intended to provide integrated fuel cycle support for the Prototype Fast Breeder Reactor (PFBR) and future fast reactor deployments. Conceived as a comprehensive and industrial-scale facility, FRFCF encompasses the reprocessing of spent fast reactor fuel, recovery and purification of fissile materials, and their subsequent conversion and refabrication, thereby enabling sustained closed fuel cycle operation. The reprocessing plant within FRFCF (Figure 3) is designed to handle high burnup, plutonium-rich mixed oxide (MOX) fuels discharged from fast reactors at short cooling times. The facility incorporates mature aqueous reprocessing flowsheets. FRFCF emphasizes high availability and maintainability through modular plant design, redundancy in critical systems, and advanced remote maintenance strategies. By integrating reprocessing with fuel fabrication and waste management within a single complex, FRFCF significantly reduces material transport requirements and enhances safeguards and material accountability. The facility constitutes a key enabling infrastructure for India's fast reactor programme and forms the backbone for the large-

Figure 3. Artistic view of Fuel Reprocessing Plant.

scale deployment of fast reactors and the eventual transition to the thorium-based third stage of India's nuclear power programme.

4. The way forward

It is realized that the growth and long-term sustainability of India's fast reactor programme is strongly dependent on fast reactor fuel reprocessing, which in turn depends on implementation of the processes that aims at minimization of waste at source, and improvement in the operational efficiency of the plant. These objectives can be marshalled by developing advanced methods, materials and technology with a focus on the recovery of actinides, recovery of all valuables from waste leaving negligible or zero-radioactivity to the waste, and automation of the plant, wherever possible, based on data analytics accumulated at CORAL and DFRP and control of the plant operation using data science, machine learning and AI protocols. The automation of the plant realized by exploring of computer codes for the process control and process simulation. The codes can be then employed for catering to process monitoring, operator training simulator, near real-time material accounting and real-time optimization and control, using data science and AI protocols.

An important direction is the integrated removal and management of minor actinides such as neptunium, americium, and curium. These nuclides contribute significantly to the long-term radiotoxicity and heat load of high-level waste. Their separation and recycle are therefore essential for waste minimization and repository optimization. Fast reactors, with their hard neutron spectrum and favourable transmutation characteristics, are particularly well suited for the transmutation of minor actinides. Integration of minor actinide recovery within fast reactor reprocessing flowsheets would enable their recycle as fuel or targets, thereby closing the actinide cycle more completely.

Future reprocessing plants may also emphasize the recovery of valuable fission products and strategic elements such as caesium-137, strontium-90 (yttrium-90), platinum Group Metals (PGMs), and neptunium. Selective recovery of these materials can contribute to heat load management, societal and industrial utilization, and improved waste stream conditioning. Development of suitable separation and purification schemes for these elements would be an important area of research, and implementation in the plant

From a process engineering perspective, a key objective is the transition from batch-type operations to continuous processing wherever feasible. Continuous head-end operations, including fuel chopping and dissolution, would improve throughput, reduce in-process inventory, and enhance criticality safety. Similarly, development of continuous reconversion processes for product streams would enable better integration with fuel fabrication and improve overall plant efficiency. While aqueous reprocessing remains the established and industrially mature technology for fast reactor fuels, parallel development of alternative fuel cycle options continues. In this context, aqueous reprocessing of metallic fast reactor fuels is being explored as a viable fall back option to pyrochemical processing. Such flexibility would provide resilience in fuel cycle strategy and ensure continuity of reprocessing capabilities as fuel forms and reactor technologies evolve.

In addition, the extensive body of operating data generated from pilot and demonstration scale fast reactor fuel reprocessing facilities represents a valuable asset for further technology advancement. Systematic digitization, validation, and analysis of this data enable the development of predictive models for process performance, equipment behavior, and solvent health under varying operating conditions. Data-driven modeling and simulation tools can support process optimization, early fault detection, and informed decision-making, thereby enhancing plant safety, availability, and efficiency. The integration of such digital frameworks with existing reprocessing infrastructure would play an increasingly important role in improving operational reliability and accelerating the transition toward more continuous and advanced fast reactor fuel cycle systems.

Overall, the way forward for fast reactor fuel reprocessing in India lies in progressive enhancement of separation capabilities, increased process integration and continuous operations, and sustained emphasis on safety, waste minimization, and resource sustainability. These developments will be central to realizing the full potential of fast reactors within India's long-term nuclear energy programme.

5. Conclusion

India's fast reactor fuel reprocessing programme has progressed steadily through a systematic and stage-wise approach, leading to the development of robust technologies and facilities capable of supporting fast reactor deployment. Reasonable technological maturity has been achieved in fast reactor fuel reprocessing, as evidenced by successful operation of pilot- and demonstration-scale facilities and the evolution toward commercial-scale plants.

Major challenges inherent to fast reactor fuel reprocessing—such as high fissile content, criticality safety, solvent degradation under high radiation fields, short cooling times, and the handling of slender fuel pins—have been effectively addressed through optimized process flowsheets, critically safe equipment design, short-residence-time contactors, and fully remote operation in alpha-tight cells. In particular, the capability to safely and reliably handle plutonium-rich fast reactor fuels has been firmly established.

The experience gained from facilities such as DFRP has significantly reduced technical uncertainties and provided confidence in the scalability and operability of reprocessing technologies for PFBR and future fast reactor fleets. These developments reinforce the central role of reprocessing in enabling a closed fuel cycle for fast reactors and underscore India's preparedness to support sustained fast reactor operation. Continued refinement and innovation, especially the automation of the plant operation, would further strengthen this capability, supporting long-term sustainability and resource utilization within India's nuclear power programme.

About the Authors

Dr. K. A. Venkatesan obtained MSc Degree from University of Madras in the year 1991, and joined BARC training school. After completion of the Training school, he joined BARC in 1992, and moved to IGCAR in 1998. At present he is Associate Director of Reprocessing Design and R&D Group, Reprocessing Group, IGCAR, Kalpakkam. He served as a secretary to IANCAS-SRC, and conducted several radiochemistry workshops in various schools and colleges in Chennai and Southern regional states.

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Operational Experience of Processing High Burnup Mixed Carbide Spent Fuel at CORAL Facility

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Abstract

The Compact Reprocessing of Advanced Fuels in Lead Cells (CORAL), located at the Reprocessing Development Laboratory at IGCAR, houses the equipment and systems installed for demonstrating the reprocessing of spent fuel from FBTR. The primary objective of the CORAL plant was to process the irradiated mixed carbide fuels with varying degrees of specific activities to gain the operating experience of reprocessing Pu-rich carbide fuels through an aqueous reprocessing route. It is a unique facility dedicated to innovative research, design, development, and deployment of closely guarded reprocessing technology for fast reactor fuels.

CORAL initially reprocessed the low-burnup spent fuel discharged from FBTR. Progressively, higher-burnup fuels were adopted after obtaining necessary regulatory approvals. The spent mixed carbide fuel from FBTR with 25, 50, 100, and 155 GWd/t burn-up have been reprocessed in CORAL. Currently, the mixed carbide fuel with 155 GWd/t burn-up is being reprocessed in campaign mode. During the reprocessing campaigns, CORAL implemented critical improvements in process, equipment, and operating procedure to ensure sustained safe operation of the facility. Fine-tuning of the process parameters/equipment was carried out primarily to minimize radioactive waste generation with minimum plant downtime. These operations gave vital design inputs and valuable operating experience for future plants. The CORAL plant continues to operate with an excellent recovery factor and radiation safety record.

Keywords: *Fast Reactor Fuel Reprocessing, Plant Operations*

1. Introduction

A facility known as CORAL (Compact Reprocessing of Advanced Fuel in CORAL), previously known as LMC, is set up in the Reprocessing Development Laboratory (RDL) in Indira Gandhi Centre for Atomic Research (IGCAR) for reprocessing of spent fuel discharged from Fast Breeder Test Reactor (FBTR). The CORAL facility is primarily a hot cell with lead shielding. The nominal processing capacity of CORAL will be one kg of irradiated FBTR fuel per month. Equipment such as dissolver, centrifuge, centrifugal extractors and transfer systems are designed to meet this objective.

The CORAL is capable of reprocessing the FBTR fuel on an experimental basis, and the facility can accommodate improvements in process and equipment. It provides operational experience and technical inputs for future plants. CORAL is the hot test facility for evaluating technology and equipment for fast-reactor fuel reprocessing. CORAL had initially reprocessed the low burn-up spent fuel discharged from FBTR. Progressively higher burn-up fuels were taken up by obtaining necessary regulatory approvals from AERB. The spent mixed carbide fuels from FBTR with 25, 50, 100, and 155 GWd/t burnup have been reprocessed as per approved technical specifications and license conditions. Currently, the mixed carbide of 155 GWd/t burn-up fuel is being reprocessed in campaign mode.

CORAL facility completed its initial design mandate of reprocessing 14 sub-assemblies of FBTR in February 2017. As per AERB norms, the re-licensing of the CORAL plant has to be carried out once in five years. After carrying out intensive refurbishment works, detailed in-service inspections of the waste vault, and regulatory inspections, the CORAL has been re-licensed for five years since September 2018 by the regulatory agency. During this period, CORAL successfully reprocessed spent fuels from FBTR, then carried out the prerequisites for relicensing activities, and submitted all required documents to the Regulatory Agency. Later, the CORAL facility was relicensed by AERB for a further period of 5 years from September 2023.

The objective of CORAL is to establish the reprocessing technology of irradiated carbide fuel discharged from FBTR with decay heat of not more than 70 Watts/kg. The cooling period increases with the burnup to meet this criterion. As an example, for a typical burnup of 75 Gwd/t the fuel shall be cooled for not less than 120 days and 150 Gwd/t it should be cooled for not less than 180 days. The facility is designed for remote operation and a combination of remote and contact maintenance. The process is based on the modified Purex process with 30% tri butyl phosphate in an alkane diluent. The fuel is dissolved in nitric acid. The recovered Pu and U will be converted into their respective oxides. The raffinate solution from the process is collected in storage tanks. The design basis for shielding ensures that the contact dose at the outer lead wall is no more than 0.1 mR/h. The criticality control is achieved by controlling the geometry, mass, and concentration of fissile materials.

The major steps involved in the process as shown in figure 1 are:

- **Chopping**: The chopper cuts the spent fuel pins into small bits.
- **Dissolution:** The cut pieces are dissolved in nitric acid solution in a titanium dissolver.
- **Feed clarification:** The dissolver solution is clarified using an air motor-driven high-speed centrifuge.
- **Conditioning and solvent extraction:** After conditioning with $NaNO_2$ for valency of Pu and acidity, the solution is sent to an extractor where the solvent 30% TBP extracts the valuable material from the dissolver solution. The loaded organic is scrubbed off the fission products and then stripped in the stripper. The raffinate containing fission products is transferred to the waste vault after analysis, without evaporation.
- **Partitioning:** In the third cycle, the loaded organic undergoes reductive stripping to separate Pu from U by electrolytic in-situ partitioning. Partitioning by the chemical precipitation method in the plutonium re-conversion lab is adopted as a process modification instead of in-situ partitioning in the third cycle.
- **Pu-Reconversion:** The Pu product solution is precipitated as Pu oxalate, filtered and calcined to Pu oxide.
- **U-Reconversion:** The uranium product solution is precipitated as ammonium diuranate, filtered and calcined to uranium oxide.
- **Waste handling:** The wastes generated from the reconversion lab and the off-gas scrubber circuits are monitored and sent for storage/disposal.

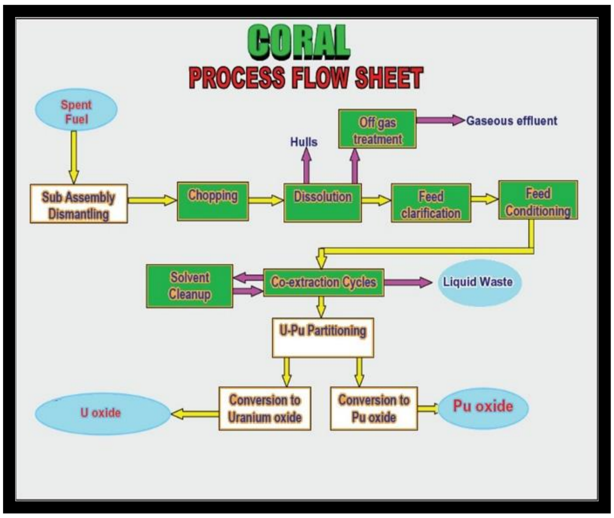

Figure 1. Flowsheet of the CORAL facility.

2. Major challenges in process and systems

2.1. Dissolution

In the initial stage, as per the design indent, the EODT had been applied for a complete destruction of carbon in the form of aqueous soluble organic compounds generated during the dissolution of carbide fuels in nitric acid. After EODT, it was observed that all plutonium was converted to its sixth valency state. It was difficult to reduce it to the desired fourth valency state from its sixth valency state during the feed conditioning step for liquid-liquid extraction. Hence, EODT was discontinued after the first campaign. Subsequently, it was found that approximately 90-94% carbon, which is in soluble organic form, was destroyed in concentrated nitric acid solution under total reflux conditions. In subsequent campaigns, dissolution and carbon destruction of the chopped pins were carried out in conc. 11.5 M nitric acid under total reflux conditions at three temperature settings for 72 hours.

2.2. Feed clarification

The dissolved solution is found to contain suspended solid particles containing noble alloys. It must be separated before proceeding with further process operations. The dissolved solution was clarified by passing it through a centrifuge, which is operated by an air-driven turbine. During reprocessing, the operation of the centrifuge is fine-tuned and stabilized to ensure it is available for each campaign. With a well-defined operating procedure, this system has consistently met the process requirements during reprocessing campaigns.

2.3. Extraction & stripping

As per the initial flow sheet, the feed solution is conditioned to 4N acidity before the extraction and stripping runs. Raffinate losses were high during the first cycle operation. This leads to recycle runs for complete processing of the solution. Organic volume also increased due to recycle runs. Due to frequent recycle runs, the product stream got diluted, which in turn increased the volume of solution processed on the reconversion step. The plutonium distribution coefficient increases with increasing feed acidity. Initially, the feed acidity was maintained at 4.0M. To increase plutonium recovery, the feed acidity of the conditioned solution was increased to 5.5 M. Plutonium recovery is increased, and the plutonium concentration in the raffinate is reduced to acceptable levels.

As FBTR fuel is short-cooled (> 2 yrs), ruthenium is a troublesome extractable fission product. It has a half-life of about 1 year. It is present in a considerable amount in the feed solution and is extracted into the organic stream. It contributes significantly to the total dose in the product stream. To improve the decontamination factor in the 1st cycle, the extraction flow sheet was standardized by changing the acidity of the scrub and aqueous feed to the extractor, which has resulted in achieving the required decontamination factor of the product solution in a single cycle. In this flow sheet standardization, CORAL has achieved a reduction in high-level liquid waste generation of up to 60% of the originally envisaged level.

Plutonium has a tendency to polymerize in aqueous solution at low acidity. As FBR reprocessing involves a high concentration of plutonium in the feed, a higher acidity of more than 0.3M must be maintained in the strip section to prevent plutonium polymerization. Hence, dual strips are provided in HC. One strip of 0.01M acid stream is given at the 16th stage, and another strip of 4M is given. So, the combined acidity of 0.3M is maintained in the plutonium stripping section.

Plutonium is retained in lean organic solvent after stripping due to solvent degradation. Primary degradation products are DBP, MBP, and Butyl alcohol. These degradation products form a strong complex with plutonium, and it is not strippable using a dilute nitric acid solution. For recovering retained plutonium, the lean organic stream is further processed using a suitable solvent wash procedure.

2.4. Improvements in the Centrifugal Extractors (CE)

The solvent extraction process in the CORAL consists of six cycles (three extraction cycles, followed by three stripping cycles). Each of these cycles is equipped with a centrifugal extractor (CE) bank consisting of 16 individual stages, namely HA, HC, 1A, 1C, 2A, and 2C (Fig. 2). Each centrifugal extractor is operated individually custom-built induction motor. The speed of each motor is within the range of 3300-3600 RPM. The motors have been designed to prevent damage by radiation and acid corrosion. The speed of each individual motor is continuously monitored, and if any deviation by 10% of the operating speed, the entire bank is automatically tripped off, and the inlet feed to the bank is also cut off to prevent solution overflow.

Figure 2. Centrifugal Extractor Bank.

CORAL deploys a 16-stage centrifugal extractor bank for the extraction and stripping operations. The centrifugal extractor bank comprises of an integrated set of 16 stationary bowls and their respective inlets and outlets. Each stationary bowl houses a rotating bowl. The shaft of the rotating bowl is connected to an electrical motor. In the initial stages of reprocessing campaigns, CORAL observed about higher motor failures in a year. This was either due to corrosion of the bearing or the winding of the CE motor. The problem of bearing failure was mitigated by the use of purge air to remove NOX vapour. Even after this modification, the failure rates remained high. The detailed analysis revealed that the failure was caused by the lower torque of the motor, which led to jamming or winding failure due to corrosion. Hence, CORAL had carried out an extensive search for MOC for the protection of the CE motor winding, which led to the selection of PEEK (Poly-Ether-Ether-Ketone) coating on the stator and rotor of the motor. Additionally, the installation of a variable frequency drive (VFD) at CORAL for controlling the CE motor's parameters, such as voltage, current, frequency, and power factor, has improved the motor's lifespan. These modifications to the motor increased torque, as well as the coating and control of motor parameters by the VFD, resulting in almost zero failure of the CE motor.

The individual stages are powered by three-phase induction motors due to requirements such as reversing, compact size, and reduced phase current. These motors are not available in the market and are custom-built. The design should ensure that the motors serve for a reasonably long time in a harsh environment. Corrosion resistance is necessary due to the nitric acid environment. The short-cooled fast reactor spent fuels with high burnups are highly radioactive, hence the motor has to tolerate an adequate radiation dose to reduce the frequency of maintenance.

To maintain the process performance, the (bowl) shaft speed of the motors is monitored. The speed signals from each motor group are displayed in the control room. The controller compares the speed setting and if a 10% deviation in the speed of any one of the motors enables the trip of VFD and stops the inlet-feed airlift.

2.4.1. Corrosion protection of stator and rotor

The rotor and stator stacks of the CE motor undergoes corrosion due to exposure to nitric acid vapor. Polyetheretherketone (PEEK) exhibits radiation resistance of 10^9 rads and also shows excellent chemical resistance. Therefore, a PEEK coating (ferrous grade) was developed for the rotor and inside the stator to provide corrosion protection (Fig. 3).

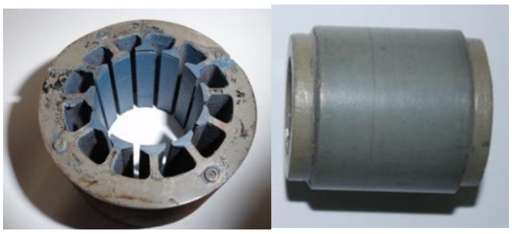

Figure 3. Photograph of PEEK coated rotor and stator.

2.4.2. Increase in height of the motor

To accommodate the PEEK coating, the air gap between the stator and the rotor was increased, resulting in a reduction of torque. To retain the torque, the height of the stator and rotor was increased.

2.4.3. Protection for cable

The cable was routed through the motor housing via a hole in the housing. Due to repeated flexing, the cable's sheath was damaged. To protect the cable from mechanical damage, a PEEK cable gland with an EPDM grommet was fabricated and fixed in the motor housing. The cable connecting the connector bank and the CE motor requires flexibility. To increase flexibility in the cable, polyimide material was selected for both insulation and sheath, as it offers radiation resistance of 10^{10} rads and excellent chemical resistance. Three core polyimide-insulated cables were used for the connection between the connector bank and the CE motor.

2.5. Lean organic treatment

Lean organic is the solvent that comes out of the stripper after the loaded heavy metals have been transferred to the aqueous phase or after reductive stripping. The TBP and dodecane used in the process are degraded by radiation damage and through hydrolytic and nitrolytic reactions with nitric and nitrous acids, forming various products. These include acidic degradation products from the TBP, as well as carboxylic acids, esters, and other compounds derived from the diluent. The degradation products affect the performance of the process both by changing the interfacial tension of the aqueous-organic systems (leading to a greater tendency to form stable emulsions) and by forming complexes with useful fissile metals, actinides, and fission products and retaining them. Heavy metals complexed with degraded products cannot be stripped back into the aqueous phase.

Lean organic waste generated from reprocessing campaigns has a marginally higher amount of Pu. The retention of Pu in lean organic is mainly due to the presence of solvent degradation products. Various chemicals, such as sodium carbonate, ammonium carbonate, hydrazine carbonate, oxalic acid, and sodium oxalate, were tested for the recovery of Pu from lean organic. Among them, oxalic acid and sodium carbonate are the best stripping agents for its recovery. CORAL has adopted this process and is continuing it for the recovery of Pu from the lean organic (Fig. 4).

The basic nature of these scrubbing solutions is that they allow removal of the short-chain organic acids which are formed by degradation of TBP, i.e., dibutylphosphoric acid (HDBP) and monobutylphosphoric acid (H2MBP), through formation of aqueous-soluble salts. Solvent clean up using sodium carbonate scrubbing is generally used in any reprocessing plant. However, its disadvantages include the generation of large quantities of sodium nitrate waste and phase separation problems with degraded organics containing long-chain organic acids. Oxalic acid will give better phase separations since it does not produce soaps from long-chain organic acids. The most important is the elimination of solid waste generated by the solvent clean up system.

The effectiveness of hydrazine carbonate increases with concentration. It was found that 15% hydrazine carbonate was suitable for removing di-butyl phosphate, uranium, and plutonium retained in the organic phase. The aqueous waste generated after recovery of U and Pu from hydrazine carbonate wash solution does not produce any salt during the waste volume reduction step.

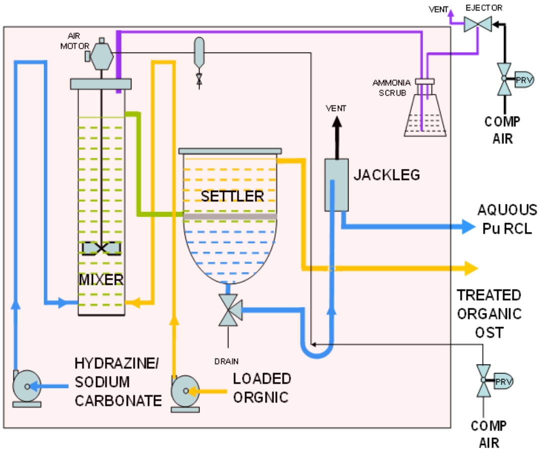

Figure 4. Schematic of solvent treatment System.

2.6. Aging management of CORAL

The service systems of the CORAL plant were refurbished as part of aging management. The fire alarm system, chilled water system, cooling tower, secondary process cooling water systems, air handling units, exhaust duct, boiler, diesel generator sets, and aviation warning lamps located in the stack were replaced successfully. Also, for the first time in CORAL history, the thickness measurement of the dissolver vessel, located inside the hot cell, has been carried out remotely with minimal man-rem expenditure. An indigenous ultrasonic transducer was used to measure the thickness of the dissolver in the CORAL hot cell, and the results confirmed that the dissolver's wall thickness has not decreased. No significant change in the thickness was observed, even after 20 years of operation. In 2023, as part of the ageing management plan, the condition monitoring of the civil structures of the CORAL plant was carried out, and the structures were qualified for further service.

2.7. Radiological challenges

The radiological status of the plant is consistently maintained as per technical specifications, and the plant discharges are well within the annual authorization limit. Various ALARA measures were taken to reduce the cumulative dose consumption of the plant. The various measures were undertaken, such as a remote sample aliquoting system, a capper-decapper system, and a remote CE dismantling system, to reduce the individual dose exposure during the reprocessing campaigns. Over the years, CORAL has effectively reduced cumulative dose consumption by almost 70%. Meticulous planning for the radiological works, following good radiological working practices and implementing strict administrative controls, helped in a significant reduction of the man-rem expenditure.

2.8. Waste management

Radioactive waste from the reprocessing plant is safely handled in accordance with standard operating procedures. The gases are discharged through the stack after multiple layers of particulate filters and are continuously monitored by the stack monitoring system. The gaseous discharges are well within the dose apportionment of the CORAL plant. Liquid and solid wastes are managed according to their categories. Raffinate waste is safely stored in a waste vault tank, and it will be transferred to the waste immobilization plant. Low-level liquid waste is safely transferred to CWMF for further management. Active solid waste is generated in the hot cell during reprocessing campaigns and is safely removed in alpha-tight containers to CWMF for interim storage. Also, low-active solid waste from the hot cell, glove boxes, and fume hoods was removed periodically through the bag-out technique and collected in a stainless drum. After the Pu assay using a helium-based neutron drum monitoring system, solid waste is transferred to CWMF. At any point in time, the cumulative discharge from the plant is maintained within the annual waste authorization limit.

3. Conclusion

The CORAL plant continues to operate with an excellent recovery factor and radiation safety record. During the reprocessing campaigns, CORAL implemented critical improvements in process, equipment, and operating procedure to ensure sustained safe operation of the facility. Process parameters and equipment were fine-tuned to minimize radioactive waste generation while maintaining minimal plant downtime. These operations gave vital design inputs and valuable operating experience for future plants.

About the Authors

Shri S. Somasundaram is Superintendent, Plant Operations, CORAL at Indira Gandhi Center for Atomic Research, Kalpakkam. He joined IGCAR in 2005 after completing the 4th batch at NFC-HWB Training School, Hyderabad. He has completed B. Tech in Chemical Engineering and M.Sc. (Engg.) from Homi Bhabha National Institute. His areas of expertise include the Operation of CORAL, a pilot-scale fast reactor reprocessing plant. He led several reprocessing campaigns involving high-burnup fast-reactor spent fuel. His areas of interest include troubleshooting of process systems, licensing requirements of the reprocessing plant, and dose budgeting for plant operations. He is the recipient of the DAE Group achievement award.

Shri Surajit Halder, a chemical Engineer from Jadavpur University, Calcutta, joined reprocessing group, IGCAR, in 1999. He has participated in cold and hot commissioning activities of CORAL and contributed significantly in the completion of reprocessing campaigns of fast reactor spent fuel with a burn-up of as high as 155 GWd/t with plutonium rich fuel for the first time in the world. As head, Reprocessing plant operation Division, he has led the team which has played important role for successfully completion of hot commissioning of Demonstration fast reactor fuel Reprocessing Plant (DFRP). Presently as Head, Coral Operation Thorium & Metallic Fuel Reprocessing Division, he is actively engaged in reprocessing of FBTR fuel at CORAL and various activities pertaining to process upgradation for Fast reactor fuel Reprocessing. He is the recipient of the DAE Group achievement award for “a decade of successful operation and challenges faced in CORAL paving way for closing of Fast Reactor Fuel Cycle- RpG”, “Remote replacement of radiation shielding window” and “Development of robotic Vehicle and remotely operated device for the visual inspection of fast reactor fuel reprocessing Facility” in 2013, 2015 and 2019 respectively.

Experience in Commissioning and Operation of Demonstration Fast Reactor Reprocessing Plant

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Abstract

In India, the Fast Breeder Test Reactor (FBTR) has been successfully operating for more than four decades, which was installed for testing the performance of different types of fuels such as mixed carbide, oxide and metallic fuels. For efficient utilization of all nuclear resources, India has adopted a closed fuel cycle strategy by way of reprocessing the spent nuclear fuels discharged from nuclear reactors. This strategy can give long term energy security to our nation from nuclear power. In contrast to thermal reactor reprocessing, however, there are few major challenges in reprocessing the FBTR fuel. They are the slender nature of fuel pins, pyrophoric carbide fuel, high plutonium content and very high specific activity. To overcome these issues, it is necessary to equip the fast reactor reprocessing plant with all necessary modules, for instance remotely operable hot cells with critically safe equipment, containment systems and other utilities etc. Reprocessing of carbide fuel at IGCAR, is being performed at the CORAL test bed facility, a COmpact Reprocessing facility for Advanced fuels in Lead shielded cell, for more than two decades. This facility played a major role in closing second stage of Indian Nuclear Power program. The Demonstration Fast Reactor Fuel Reprocessing Plant (DFRP) was established to reprocess spent fuel (carbide and oxide) discharged from Fast Breeder Test Reactor (FBTR) on regular basis and to demonstrate the reprocessing of Prototype Fast Breeder Reactor(PFBR) spent fuel. DFRP facility was dedicated to the nation by the Honorable Prime minister of India on 2nd January, 2024 and hot commissioned in April 2024.

Keywords: *FBTR, reprocessing plant, CORAL, DFRP*

1. Introduction

In general reprocessing plants are divided in to several facilities like (1) Head end facility (HEF), (2) Process plant facility (PPF), (3) Waste tank Farm (WTF), (4) Utility System, (5) Ventilation and Off-gas system etc. The design of Head End Facility depends on, nature of fuel, dimension of fuel pin, type of fuel sub assembly (FSA), burn-up and clad material. Different sets of head end equipment are required to process fuels from different reactor. Since dimensions and nature of fuel, handling criteria are different for FBTR and PFBR fuels, separate fuel handling system, chopper and dissolvers are available in DFRP at two different cells. Reprocessing of spent fuels of FBTR and PFBR can be done in separate campaigns following the standard operating procedure for every step of Head end facility and in process plant facility. In the design stage of reprocessing facility all the process and personnel safety provisions were considered to avoid and handle abnormal situations that may arise during operations. The schematic process flow sheet adopted at DFRP is given in the fig. 1 for reprocessing of spent fuels from FBTR and PFBR reactor.

<u>Fuel Handling</u>: Spent nuclear fuel pins are brought to DFRP PPF in cask on tractor trolley. After unloading from tractor trolley, cask is loaded onto a dolly kept on the railing, aligned and engaged with dissolver cell door with help of special lip seal door. With the help of push rod and manipulators SS can with spent fuel pins is transferred inside the cell. Fuel pins from can are loaded into fuel pin magazine and transferred to chopper unit with help of in-cell crane.

<u>Chopping</u>: To facilitate the dissolution of spent fuel, chop leach process is followed. Fuel pins are chopped into small pieces of 25-30mm length with the help of single pin chopper along with argon purge due to pyrophoric nature of carbide fuel. The chopped pieces fall through chopper chute line into thermo-siphon type dissolver. Prior to chopping, dissolver is filled with concentrated acid to avoid free fall of chopped pieces on to the dissolver floor. Chopper can be operated in manual or auto mode. The chopper tool is replaceable with manipulator and crane. After complete chopping, chopper chute is rinsed with dilute acid to remove fuel particles present in the chute, if any.

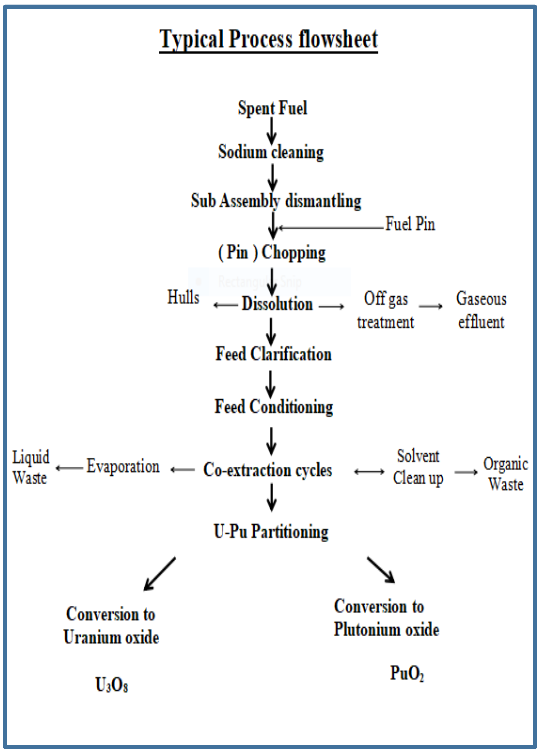

Figure 1. Schematic of the DFRP flow-sheet.

Dissolution: On completion of chopper chute rinse, dissolver basket is rotated to dissolution position and dissolver level is adjusted till required level for liquid circulation during heating. Dissolver vacuum is adjusted up to the required level with the help of Variable Frequency Drive (VFD) of Dissolver Off-gas Fan (DOG). Then Step wise progressive heating of dissolver solution is followed for dissolution and soluble organic destruction, with required acid makeup. Complete dissolution is ensured by sample analysis of dissolver solution.

Feed Clarification: Dissolver solution gets clarified in air-turbine driven high speed centrifuge to remove fines and undissolved solid particles. Dissolved solution is fed to centrifuge by vacuum aided airlifts and the clarified solution goes to a connected product pot and subsequently transferred to a tank.

Feed Conditioning: The aim of this stage is to condition the solution suitable for feeding to extractor to facilitate extraction process by using solvent. The clarified solution is transferred to conditioning tank for adjusting acidity and the valance of all plutonium. The conditioning of solution is carried with addition of required concentration of Nitric acid and NOx gas purging, respectively.

Co-Extraction cycles: Plutonium and Uranium from conditioner solution gets extracted into 30% Tri butyl Phosphate (TBP) in centrifugal extractor (CE) bank with dual scrub of HNO_3 to achieve required Decontamination Factor (DF). After that, loaded organic gets stripped in CE bank using Uranous (U+4), 0.01M and 4M HNO3 as strip solutions. Stripped product undergoes further conditioning. Lean organic undergoes solvent wash and then goes for disposal as organic waste. Raffinate of Co-decontamination cycle is processed in Evaporation cycle and then disposed as high level liquid waste (HLLW) after evaporation.

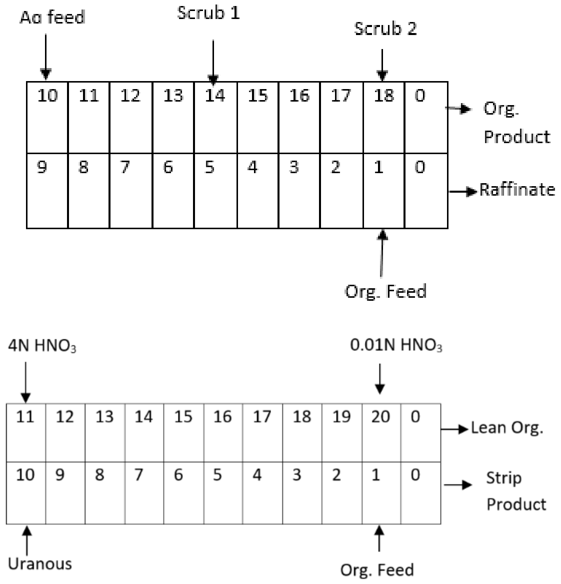

Figure 2. Flow-sheet of HA and HC.

U-Pu Partitioning: Stripped aqueous product from co-decontamination cycle (or) loaded organic product undergoes partitioning, in partition cycle. DFRP has the operational flexibility of aqueous and organic partitioning. In aqueous partitioning, the stripped aqueous product of co-decontaminated cycle is conditioned to Pu valency, acidity. This conditioned solution is partitioned in partitioning cycle with 30%TBP extractant, U^{4+} scrub. In organic partitioning, loaded organic product partitioned along with 30%TBP, U^{4+} scrubs. The aqueous product from this cycle undergoes Pu conditioning and then transferred to Pu reconversion laboratory for further processing. Organic product undergoes Uranium stripping; this stripped product is transferred to Uranium reconversion laboratory for further processing.

__Plutonium reconversion operations__: Purified plutonium nitrate solution generated in partitioning cycle is transferred to Reconversion Laboratory (Fig. 3). Solution feed acidity, valence is adjusted. Plutonium nitrate gets precipitated as plutonium oxalate by the addition of oxalic acid to the conditioned feed. The precipitation is carried out in a critically safe continuous precipitator column. Plutonium oxalate slurry collected from the column is washed, dried and calcined at specified temperature profile to get stable PuO_2 powder as a product. The PuO_2 product obtained is ground and collected in the SS container. After mixing and sampling, the product is weighed and stored in the birdcage for further use. The plutonium oxalate filtrate contains residual plutonium which has to be recovered by solvent extraction. Since, oxalate interferes in the solvent extraction process it is destroyed by the addition of $KMnO_4$ followed by H_2O_2 to remove the excess $KMnO_4$ and the treated solution is returned to the plant for further processing. With this approach Quantitative recovery of plutonium is achieved.

$$Pu(NO_3)_4 + 2H_2C_2O_4 \rightarrow Pu(C_2O_4)_2 + 4HNO_3$$

$$Pu(C_2O_4)_2 \xrightarrow{\Delta} PuO_2$$

Figure 3. Plutonium Reconversion Laboratory.

__Uranium reconversion operations__: Purified uranium nitrate solution generated is transferred to Reconversion Laboratory. Feed uranium solution and ammonium hydroxide are pumped into the precipitator column at pre-determined flow rates to get Ammonium di-uranate (ADU) precipitate. ADU slurry is filtered, washed and calcined to get U_3O_8 product. The obtained U_3O_8 product is ground and collected in the SS container. After mixing and sampling, the product is weighed and kept in the storage room for further use.

$$2UO_2(NO_3)_2 + 6NH_4OH \rightarrow (NH_4)_2U_2O_7 + 4NH_4NO_3 + 3H_2O$$

$$(NH_4)_2U_2O_7 \rightarrow 2NH_3 + 2UO_3 + H_2O$$

$$6UO_3 \xrightarrow{\Delta} 2U_3O_8 + O_2$$

In every stage, indigenously developed and designed equipments are used. All vital equipments are designed as a modular type, remotely operable, remotely maintainable without compromise of the process performance and safety.

Criticality control was ensured in design stage itself under different combinations of mass control, concentration control and geometry control, whichever applicable.

2. Commissioning

Commissioning and sustained operation of nuclear fuel Reprocessing plant is challenging due to the installation of first of its kind equipment, and systems, which requires remote operation and maintenance. Due to this, planning and sequential commissioning of process, equipments play a vital role. Hence, the DFRP (Fig. 4) is commissioned in three phases.

Figure 4. DFRP plant Building.

__Phase I__: Commissioning of process plant facility (PPF) for reprocessing of FBTR spent fuel; here fuel is received as pins in cask. The dismantling of sub-assembly and retrieval of fuel pins is performed at RML, IGCAR. This phase is focused on safe handling of fuel and functionality of process equipment as per design intent.

***Phase II*:** Receipt of spent fuel sub assembly from FBTR in Head End Facility (HEF) of DFRP, for dismantling and extraction of pins from SA, subsequent reprocessing in DFRP PPF.

***Phase III*:** Receipt of PFBR spent fuel sub assembly (FSA) in HEF of DFRP and subsequent reprocessing for demonstration purpose.

The commissioning strategy and sequence should follow as per regulatory guidelines; in case of reprocessing plant, following are the stages of commissioning: -

Stage I: *Pre operational check of different systems*

In this stage physical inspection of all systems, and equipment were completed to verify the continuity and connectivity of process lines, drains, nozzles of process tanks and equipment. Slope of process lines, instrument lines, off gas vent headers, drain lines from various off gas filters, headers were also checked. In this process, water flushing of equipment and piping system to remove dust and other foreign materials was carried.

***Pre-operational Tests*:** In this stage, following mentioned activities were completed at DFRP. These include, calibration of all process tanks, equipment, instrumentation systems, verifications of instrument operability, finding of tank and equipment holdup at various conditions, testing of process utility systems. Communication system, glove box leak testing, efficiency checking of off-gas and exhaust filters, testing of fire detection and firefighting system, radiometry of all concrete and lead cells was also completed. During preoperational check and test, several problems appeared during the test were addressed with necessary modification in design and in operational procedure.

Stage II: *Water and acid run to test individual systems*

In this stage, the main aim was to obtain the data for functional and performance testing of equipment, systems and in integrated manner. During stage II, functionality and performance testing were completed for all process equipment, such as dissolver, feed clarification system, conditioning, extraction, stripping, partitioning etc. The functionality and performance testing of evaporators, associated systems, instruments and control systems with interlock were tested. Performance test of Pu reconversion laboratory systems, equipment, vacuum transfer system, precipitator, filtration system, furnace for Calcination and blender were checked and verified with the design intent. All unit operations in Uranium reconversion systems were also tested in this commissioning stage.

Acid killing in evaporators was tested and finalized parameters for operations. In this stage, first of its kind equipment were tested and modified to get desired performance as per process requirements.

Stage III: *Acid-Solvent Run*

After completion of water-acid runs, ensured cells and containment boxes were free of mechanical repairs, combustible materials to remove ignition sources. During this stage solvent (30% TBP) was introduced into the process for testing of extraction and stripping cycles for their hydraulic and mass transfer performances with respect to Acid-TBP system. Calibration and performance tests were also carried for systems involving solvent, besides solvent wash, diluent wash equipment for hydraulic and mass transfer performance were tested. In this stage of commissioning all the process parameters were tested thoroughly, modified as per requirement and documented for future reference.

Stage IV: *Cold feed run*

For the cold feed run, Natural Uranium oxide fuel pins of FBTR dimension were used. Performance of single pin chopper system, dissolution process, feed clarification using high speed centrifuge were checked. In the cold feed run the mass transfer performance of extraction and stripping systems using centrifugal extractor were tested and compared with design intent. After this stage some of the systems were modified to get better result during plant operations.

Figure 5. View of Fuel handling cask and area.

Stage V: Limited hot runs

In the limited hot runs, actual irradiated spent fuel from reactor was introduced, abiding technical specifications. In this stage the plant becomes radioactive, marks the start of operation, hence from this stage onwards all radiological safety requirement criteria were followed. Increased the batch capacity progressively from 1/3rd to 1/2, then full batch operation to evaluate the performance of different process, equipment and instruments. During limited hot runs all radiological safety philosophy and procedures were followed and no abnormalities were observed in radiological status of DFRP plant.

3. Operation of DFRP

Major challenges for process operation of Fast Reactor Fuel Reprocessing plant are achieving of required decontamination factors, limiting the losses of heavy metals in raffinate, retention of heavy metals in lean organic, organic degradation due to radioactivity and high acid environment. The other challenges are remote operation, remote maintenance, and control radiation dose exposure to the plant people. In DFRP during the campaigns, several parameters were varied for extraction, stripping, partitioning steps to get improved throughput, reduce solvent degradation, reduced loss of heavy metals and for getting better product specifications, minimal waste generation and for safe operation.

4. Conclusion

The successful hot commissioning and operation of DFRP for reprocessing of spent carbide fuels from FBTR was achieved. This serves as a major milestone in closing the second stage nuclear fuel cycle of the Indian nuclear power program. The commissioning and operating experience gained at DFRP could serve as an invaluable source of knowledge in establishing the future fast reactor fuel reprocessing plant.

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Process Chemistry Aspects in Fast Reactor Fuel Reprocessing

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Abstract

An overview of the current chemical aspects of fast reactor nuclear fuel reprocessing is presented, highlighting the PUREX process, which has continued to be the indispensable reprocessing technology for the past seven decades. In this article, the major steps involved in process chemistry associated with reprocessing of spent fuel discharged from fast reactor including achieving higher decontamination from fission products; uranium-plutonium partitioning strategy and suitable analytical methodology for the determination of process parameters are discussed. Furthermore, the distributions of neptunium into various process stream and results of preliminary studies pertaining to recovery of neptunium, a precursor material for the production of ^{238}Pu for use in space program are presented. The methodology for separation of valuable radionuclides such as Sr-90 from high-level waste and its subsequent purification to enable for ^{90}Y milking are also described.

Keywords: *Fast reactors, Fuel reprocessing, PUREX process.*

1. Introduction

In the PUREX process, uranium and plutonium are selectively extracted into 30% Tri n-butyl phosphate diluted with n-dodecane (TBP/n-DD) leaving fission products and minor actinides in high level wastes (HLW). The first major step in the reprocessing of fast reactor (FR) spent fuel is the co-decontamination cycle, in which the organic solvent is contacted with an aqueous feed solution that has been pre-conditioned with respect to acidity and the oxidation state of uranium and plutonium. To ensure efficient extraction, plutonium is quantitatively adjusted to the Pu (IV) oxidation state using an appropriate conditioning agent. Subsequently, the extracted uranium and plutonium from the organic phase are stripped together using dilute nitric acid. In the second step, Pu and U in the aqueous strip product are partitioned by following an aqueous partitioning procedure. Each product stream is then subjected to precipitation process for obtaining respective oxides [1]. A simplified schematic of the overall process flow sheet is shown in Figure 1.

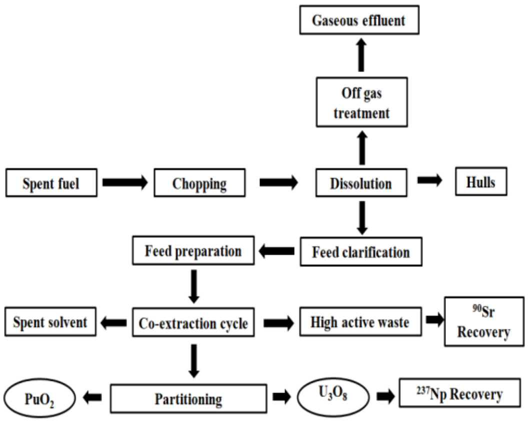

Figure 1. Process flow sheet of fast reactor fuel reprocessing including ^{90}Sr and ^{237}Np recovery.

2. Major Steps of the PUREX Process

2.1. Dissolution

The objective of the aqueous dissolution process is to transform solid nuclear fuel into a chemically soluble form that is suitable for subsequent downstream separation process.

The chop leach method has been established as the most effective technique for dissolving oxide and carbide fuel cladded with stainless steel. In this process, the mechanically sheared fuel segments are loaded into a perforated metal basket and immersed in hot, boiling nitric acid (11.5 M), enabling the dissolution of uranium, plutonium, minor actinides, and most of the fission products [2]. The probable reaction taking place during dissolution is described by following chemical equations (1), (2) & (3).

$$PuC + 8\ HNO_3 + O_2 \rightarrow Pu(NO_3)_4 + 4NO_2\uparrow + H_2O + H_2C_2O_4 + CO_2\uparrow \tag{1}$$

$$Pu_2C_3 + 10\ HNO_3 + O_2 \rightarrow 2Pu(NO_3)_4 + 5H_2O + N_2O\uparrow + CO_2\uparrow + \text{Mellitic acid (trace)} \tag{2}$$

$$UC + 4HNO_3 + O_2 \rightarrow UO_2(NO_3)_2 + 2NO + 2H_2O + H_2C_2O_4 + CO_2\uparrow \tag{3}$$

Soluble organic compounds have been reported as by products during the dissolution of carbide fuel [3,4]. Nevertheless, the applied reflux conditions result in decomposition of the organic moiety to CO_2 with efficiency greater than 98%. The small amount of residual soluble organic matter remaining in the dissolver does not interfere with downstream process steps.

The NO_2 gas evolved during the dissolution process stabilizes plutonium predominantly in the Pu (IV) oxidation state. During sparging, the NO_2 gas is removed, leading to change in redox condition of solution. As a result, a fraction of plutonium is oxidized to Pu(VI). The extent of Pu(VI) formation is governed by the reflux condition and duration of sparging, with approximately 10–15% of the Pu in the dissolver solution is being oxidized to Pu(VI). A representative UV–Visible spectrum of the diluted dissolver solution is shown in Fig. 2

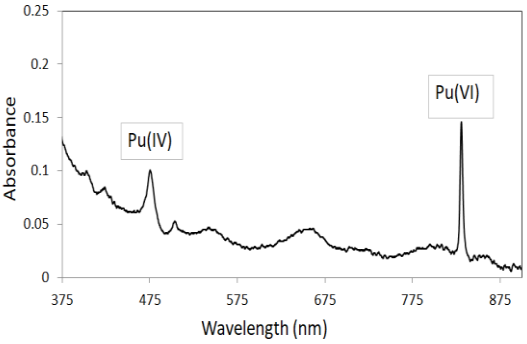

Figure 2. Typical UV-Visible spectrum of dissolver solution, DFRP.

During dissolution, fission gases are released and it is routed through series off gas scrubber systems consisting DM water and caustic scrubber for the removal of volatile fission products like iodine, cesium and ruthenium. Engineering scale dissolution study of MOX fuel with high Pu content demonstrates that electro oxidative dissolution is indispensable for achieving complete dissolution of such fuels. Accordingly, a dissolver unit integrated with an electrolyzer, enabling in situ generation of Ce (IV) is employed for the dissolution of test fuel subassemblies containing $(Pu,U)O_2$ with 44 % Pu [2,3].

During the dissolution of spent nuclear fuel in HNO_3, the cladding material (stainless steel) remains undissolved and is retained in the hull basket. Prior to disposal, it is mandatory to assess the Pu content in the hulls. For this purpose, following the transfer of the dissolver solution to downstream tanks, the entire hull is subjected to reflux with fresh nitric acid in the dissolver system for 24 hrs. The rinsed hull is subsequently subjected to hull monitoring.

Due to intense gamma radiation field, hull monitoring is done remotely inside concrete cell by gamma spectrometry technique. Since, the dissolution chemistry of Ce is similar to Pu, presence of cerium can serve as an indirect indicator (tag) of retained Pu in the hull [1]. The ^{144}Ce is a fission product formed in significant yield during reactor operation. In presence of intense gamma radiation field due to presence of stainless steel activation product (^{60}Co) present in the hull, it is possible to determine ^{144}Ce activity by analyzing its characteristic gamma photon (2.185 MeV, 0.7% yield) using high purity germanium detector. By determining the mole ratio of ^{144}Ce to Pu in the dissolver solution and the concentration of ^{144}Ce in the hull, the quantity of Pu retained in the hull can be estimated.

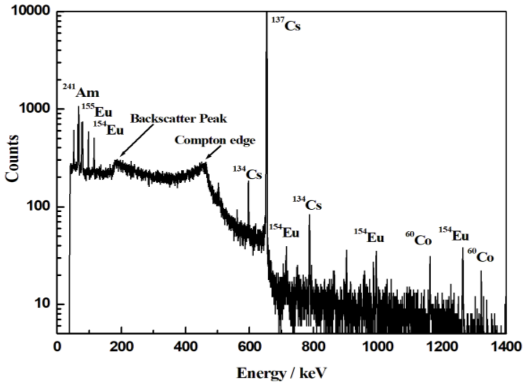

Figure 3. Gamma Spectrum of diluted dissolver sample before ^{137}Cs separation.

However, direct determination of ^{144}Ce activity in dissolver solution by gamma spectrometry becomes challenging for long cooled (> 5 years) spent fuel due to the short half-life of ^{144}Ce (285 days) and the presence of ^{137}Cs, which produces a high Compton continuum in the low energy (< 500 keV) region. A typical such gamma spectrum is given in figure 3. Consequently, extractive gamma spectrometry using N,N,N′,N′-tetraoctyldiglycolamide (TODGA) was employed to selectively separate lanthanides from the diluted dissolver solution. The resulting TODGA layer containing ^{144}Ce was subsequently quantified by analyzing its characteristic gamma photon at 133.5 keV (yield ≈ 10%). The gamma spectrum of ^{144} Ce in the TODGA layer is given in figure 4. However, this approach based on the quantification of ^{144}Ce is effective only for spent fuel with cooling periods shorter than five years.

In contrast, neutron interrogation techniques are expected to yield more reliable estimates, as this method provides lower detection limits and it does not depend on cooling period of spent fuel and high level of gamma field associated with hull. The preliminary investigation shows that the results are promising.

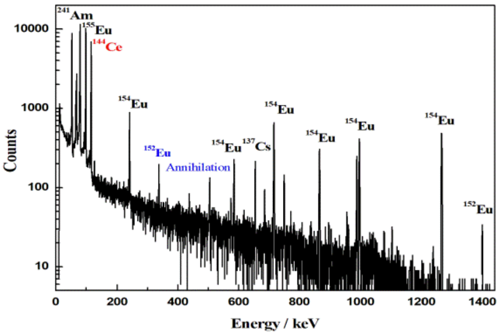

Figure 4. Gamma Spectrum of ^{144}Ce after extraction into TODGA.

2.2. Feed clarification

The dissolver solution obtained from high burn-up spent fuel contains extremely fine, sub-micron intermetallic alloy particles, the quantity of which increases with increase of burn-up. In thermal-reactor spent fuel processing, feed clarification is primarily required for the removal of cladding fines. However, in the case of fast-reactor spent fuel, both intermetallic alloy particles and cladding fines must be removed [5]. Since high level of radiation field associated with irradiated fuel solution and insoluble residue of sub-micron level, conventional filter media is not found to be suitable. Therefore, a high-speed centrifuge is employed for the effective separation of insoluble residue from the dissolver solution [2]. Characterization of the insoluble residues revealed the presence of Mo, Ru, Zr, Pd, and Rh, along with trace amounts of Pu. This residue was observed to be resistant to complete dissolution even after prolonged boiling in concentrated nitric acid and even in presence of aqua regia. However, these fines were separated completely by feed clarification procedure in fast reactor fuel reprocessing.

2.3. Feed conditioning

Prior to solvent extraction the clarified feed solution is chemically adjusted to the required acidity, oxidation state, and metal ion concentration. Under the strongly oxidizing conditions prevailing during dissolution, a significant quantity of plutonium is converted to Pu (VI). Although Pu (VI) is partially extractable, Pu (IV) exhibits a substantially higher distribution ratio in TBP/n-DD. Therefore, Pu (IV) is the preferred oxidation state for efficient extraction of Pu along with U. The gradual addition of sodium nitrite solution to the conditioner tank achieves complete conversion of all plutonium to Pu (IV), however sparging of nitrogen dioxide gas (NO_2) is preferred over $NaNO_2$ addition as it does not involve introduction of dissolved solids into the fuel solution. This NO_2 gas also enabled to oxidise Pu (III) formed if any by means of disproportionation (Fig. 5). The chemical reactions taking place during conditioning step is given in following equations (4), (5) and (6). In the case of fast-reactor spent fuels, which are characterized by high plutonium content, high burnup, and elevated fission-product activity, stringent control of feed acidity is critical to minimize plutonium loss to waste streams and to achieve improved decontamination from fission products in the first extraction cycle [1,6]. In the dissolver solution, uranium is present predominantly in the stable hexavalent oxidation state in nitric acid medium and exhibits high extractability.

$$2\,Pu^{3+} + NO_2 + 2\,H^+ \rightleftharpoons 2\,Pu^{4+} + NO + H_2O \qquad \textbf{(4)}$$

$$PuO_2^{2+} + 2NO_2 + 4\,H^+ \rightleftharpoons Pu^{4+} + 2\,NO_3^- + 2\,H_2O \qquad \textbf{(5)}$$

$$2\,NO_2 + H_2O \rightleftharpoons HNO_2 + HNO_3 \qquad \textbf{(6)}$$

During thermal reactor fuel reprocessing (TR), the dissolver solution comprises approximately 350 g/L uranium, 1 g/L plutonium, and an acidity of 2–2.5 M. The desired decontamination factor is achieved by loading metal ions to roughly 70–80% of the organic phase's theoretical capacity. However, attaining 70–80% theoretical loading in the case of fuels with fast reactor (FR) composition is not feasible due to the limited solubility of the $Pu(NO_3)_4{\cdot}2TBP$ complex in non-polar organic diluents which results in third phase formation. To overcome this limitation, the U/Pu concentration ratio in the feed solution during reprocessing of FBTR spent fuel in initial campaigns were changed from 0.4 to 2.3 by the external addition of uranium, with the objective of achieving the desired fission-product decontamination. Subsequent studies showed that comparable decontamination factors could be achieved under high-acid flowsheet conditions. This behaviour was attributed to a reduction in free TBP availability at higher nitric acid concentrations due to the formation of HNO_3–TBP complex which suppresses the extraction of fission products. In all cases, the plutonium concentration in organic phase is typically limited to below 25 g/L to prevent third-phase formation

[7], even under off-normal hydrodynamic conditions.

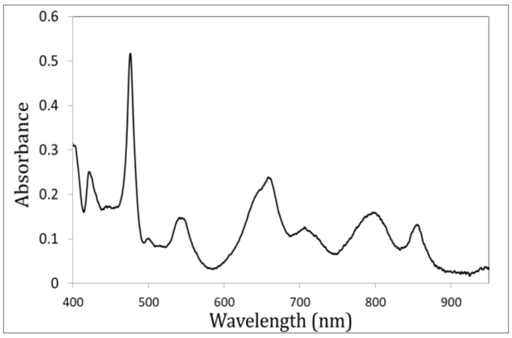

Figure 5. A UV visible spectrum of plutonium nitrate in conditioned feed solution.

The minor actinide namely neptunium (Np) can exhibit multiple oxidation states. Under highly acidic conditions, Np(V) undergoes disproportionation to form Np (IV), Np (VI) and this reaction is described by equation (7). Both the +4 and +6 oxidation states show relatively high distribution ratios and are readily extractable with TBP/n-DD system. Consequently, high-acid flowsheet particularly used for fast reactor fuel reprocessing enable routing of neptunium (>90%) to product stream during HA cycle.

$$2NpO_2^{+} + 4\,H^{+} \rightleftharpoons NpO_2^{2+} + Np^{4+} + 2H_2O \qquad \textbf{(7)}$$

2.4. Co-decontamination cycle

The conditioned feed solution is then subjected to contact with organic phase consisting 30 % TBP/n-DD. The solvent extraction process consists of two main stages: (i) purification of uranium and plutonium by removing fission products, and (ii) mutual separation of the purified uranium and plutonium. In both stages, the reprocessing of fast reactor fuel differs significantly from that of thermal reactor fuel, owing not only to its higher plutonium content but also to the substantially greater levels of radioactivity and extractable fission product species characteristic of fast reactor spent fuel. As spent fuel of higher burn-ups and short cooled is discharged from fast reactor, the inventory of fission products is higher. But the product purity required for fuel fabrication should be of the same purity as that of the thermal spectrum reactors, therefore, higher decontamination factor in the order of 10^5– 10^6 is required. Beyond process design considerations, the selection of suitable extraction equipment is critical. In view of the stringent decontamination requirements and criticality safety considerations, high-efficiency contactors such as centrifugal extractors are employed for solvent extraction in fast reactor fuel reprocessing plants. In contrast, PHWR reprocessing facilities employ simpler extraction devices, such as pulse columns, which typically exhibit residence time of several minutes, compared with only a few seconds in the case of centrifugal extractors.

In nitric acid media, actinide ions (Pu^{4+}& UO_2^{2+}) exhibit a strong affinity for nitrate ligands, leading to the formation of electrically neutral metal nitrate complexes. This speciation is a prerequisite for solvent extraction as tri-*n*-butyl phosphate, a neutral solvating extractant, preferentially transfers uncharged species into the organic phase, whereas ionic complexes remain largely confined to the aqueous phase. Coordination occurs through the phosphoryl (P=O) oxygen of TBP, which acts as a strong Lewis base and donates electron pair to vacant orbitals of metal, stabilizing the solvated complex [6,7]. The formation of metal-TBP complex is described by chemical equations (8) & (9). The resultant metal-solvate complexes are soluble in organic phase leaving aqueous phase consisting most of the fission products, corrosion products and trivalent minor actinides.

$$UO_2(NO_3)_2 + 2TBP \rightleftharpoons UO_2(NO_3)_2{\cdot}2TBP\ (org) \qquad \textbf{(8)}$$

$$Pu\ (NO_3)_4 + 2TBP \rightleftharpoons Pu(NO_3)_4{\cdot}2TBP\ (org) \qquad \textbf{(9)}$$

In addition to above, Np also extracted as described in chemical reaction (10) & (11)

$$NpO_2(NO_3)_2 + 2TBP \rightleftharpoons NpO_2(NO_3)_2{\cdot}2TBP\ (org) \qquad \textbf{(10)}$$

$$Np(NO_3)_4 + 2TBP \rightleftharpoons Np(NO_3)_4{\cdot}2TBP\ (org) \qquad \textbf{(11)}$$

This step provides required decontamination from fission products such as Cs-134, 137, Sr-90, Ru-103, 106, Sb-125 and lanthanides (Ce-144, Eu-154,155) and actinides (Am, Cm). The decontamination factors with respect to few major fission products encountered in FR are given in the table 1.

Table 1. Decontamination factors of different fission product nuclides.

Fission product nuclides	Observed decontamination factor
$Eu^{152\text{-}155}$	$3{\times}10^3$
Sb^{125}	$2{\times}10^3$
Ru^{106}	$4{\times}10^4$
Zr^{95}	$4{\times}10^4$
Ce^{144}	$6{\times}10^4$
Cs^{137}	$3{\times}10^5$

The extracted uranium and plutonium present in the organic phase are subsequently stripped using dilute nitric acid. Since the plutonium concentration in the loaded organic phase is relatively high, direct contact with dilute nitric acid in the stripper is avoided to minimize the risk of plutonium polymerization. To address this, the 20-stage centrifugal contactor stripper is divided into two sections as given in fig 6.

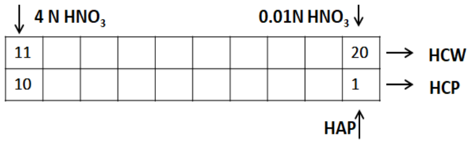

Figure 6. HC bank with dual acid strip profile.

In stages 1–11, the aqueous phase acidity is maintained below 1.0 M, enabling more than 99% of the plutonium to be stripped in this region, referred to as the plutonium strip section. For this purpose, dual strip acids (4M and 0.01M HNO_3) are fed into the stripper. The organic phase entering to stage 11 contains primarily uranium with residual plutonium. Upon subsequent contact with very dilute nitric acid, uranium is completely stripped into the aqueous phase. Furthermore, the difference in aqueous nitric acid concentration in stage 11 & 12 in the stripper bank leads to U reflux and this effect also significantly enhances the back extraction of Pu from organic phase at stage 11 even in the presence of significant concentration of HDBP.

As Pu is susceptible to disproportionation in dilute nitric acid as provided in equation (12), Pu in the dilute nitric section of stripper bank undergoes disproportionation; this behavior is confirmed by presence of Pu (VI) in the co-strip product which has characteristic absorbance peak at 830 nm.

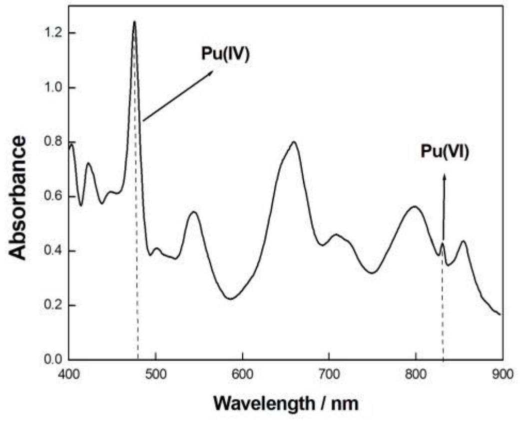

Figure 7. Typical UV-Visible absorbance spectrum of co-strip product.

The typical UV–Visible absorbance spectrum obtained for the strip product is given the fig 7.

$$3Pu^{4+} + 2H_2O \rightleftharpoons PuO_2^{2+} + 2\,Pu^{3+} + 4\,H^{+} \qquad \textbf{(12)}$$

Nevertheless, the stripped organic phase contains residual plutonium in the range of 50–100 mg/L and uranium below 50 mg/L. Because these metal ions are strongly complexed with di-*n*-butyl phosphate, they cannot be removed by dilute nitric acid stripping. Consequently, additional recovery steps employing CO_3^{2-}/$C_2O_4^{2-}$-based reagents for the removal of U/Pu along with solvent treatment operations to eliminate HDBP, are required prior to solvent recycle.

2.5. Partition Cycle (U/Pu Separation)

In the partitioning cycle, plutonium is separated from uranium by reducing extractable Pu (IV) to the in-extractable Pu (III) species in the organic phase. Though, various reducing agents such as hydrazine nitrate, ferrous sulphamate and hydrazine-stabilized uranous nitrate are available, hydrazine-stabilized uranous nitrate in nitric acid medium is found to a be a promising partitioning agent. In practice, the amount of uranous nitrate required for quantitative reduction of Pu is found to be 8–10 times the stoichiometric requirement in organic partitioning method generally followed in thermal reactor reprocessing plant. This is due to several factors, including the poor distribution of uranous ions especially in the presence of heavily loaded U in the organic phase, limited availability of hydrazine in the organic phase, and the consumption of uranous ions in many side reactions during partitioning. While this excess reagent requirement is not critical in thermal reactor fuel reprocessing because of the relatively low plutonium inventory, it becomes significant for fast reactor spent fuel, where plutonium concentration are much higher. Consequently, the flowsheet must be modified to optimize uranous nitrate consumption, and aqueous partitioning emerges as a promising alternative for effective plutonium–uranium separation [2]. In this method, the aqueous product solution containing co stripped U & Pu is treated with stoichiometric quantity of uranous nitrate after adjusting hydrazine nitrate concentration to 0.5 M and feed acidity of 1.5 M. All the Pu in this solution is converted to Pu (III) with a stoichiometric amount of uranous nitrate and chemical reaction involved in this process is given as per equations (13-15). The nitrous acid present in the solution is destroyed by hydrazine nitrate which prevents the re-oxidation of Pu (III) to Pu (IV) by nitrous acid.

$Pu^{4+}{}_{(aq)} + U^{4+}{}_{(aq)} \rightarrow Pu^{3+}{}_{(aq)} + UO_2^{2+}{}_{(aq)}$ **(13)**

$N_2H_5^+ + NO_2^- \rightarrow HN_3 + 2H_2O$ **(14)**

$HN_3 + HNO_2 \rightarrow N_2 + N_2O + H_2O$ **(15)**

Subsequently, the conditioned feed solution is contacted with an organic phase comprising TBP in n-dodecane for the selective extraction of uranium, while plutonium remains in the aqueous phase. The resultant uranium in organic phase is further stripped with dilute nitric acid (0.01 M). As this level of plutonium contamination in the uranium product is relatively high, additional purification step is mandatory to make it suitable for tabletop handling.

2.6. Reconversion operations

Plutonium nitrate product conversion to PuO_2

The plutonium recovered from the spent nuclear fuel by PUREX process is Pu $(NO_3)_4$, whereas the starting material for fabrication of plutonium containing fuel assemblies is plutonium oxide. Therefore, it is necessary to convert plutonium nitrate solution to PuO_2. The preparation of plutonium oxide generally involves two main steps: selective precipitation followed by calcination. Though several options such as peroxide, fluoride and oxalate precipitation methods are available, the oxalate precipitation route is preferred due to its operational simplicity and also the resulting product meets the specifications stipulated by fuel fabricators. Therefore, oxalate precipitation route is being followed in DFRP and CORAL plant. In reconversion laboratory, the Pu product after partitioning step is received. Then, the acidity of plutonium nitrate solution is increased to 3.0-3.5M and passed through precipitator column with a constant flow rate along with required oxalic acid at fixed flow calculated based on the stoichiometric requirement of oxalate ion for plutonium oxalate precipitation with an excess of 0.1 M oxalic acid in the precipitation medium. Under this condition, the typical Pu loss in oxalate filtrate is within acceptable limit. The Pu oxalate slurry is washed with wash acid consisting mixture of nitric acid (3M) and oxalic acid (0.1M) to get decontamination from U, Fe and FPs. The chemical reaction involved in precipitation and subsequent calcination is described in the equation (16 & 17).

$Pu^{4+} + 2C_2O_4^{2-} \rightarrow Pu(C_2O_4)_2 \downarrow$ **(16)**

$Pu(C_2O_4)_2 \rightarrow PuO_2 + 2CO + 2CO_2$ (550 °C) **(17)**

The final PuO_2 product is suitable for storage or MOX fuel fabrication.

Uranium nitrate product to U_3O_8

The uranium recovered during reprocessing of spent fuel by PUREX process is in the form of uranyl nitrate solution. This uranyl nitrate solution is subjected to hydroxide precipitation using NH_4OH solution or by ammonia gas. During this process, a yellow cake of ammonium di- uranate (ADU) is formed, which on calcination at 550 °C yields U_3O_8 product. The effluent generated in uranium reconversion operation is ADU filtrate. The alpha activity in ADU filtrate is found to be within acceptable range of category II effluent.

3. Solvent degradation and treatment

Solvent degradation poses a known challenge in PUREX process, but fast-reactor fuel reprocessing intensifies these effects remarkably. The severity arises from elevated plutonium concentration that enhances α-radiolytic degradation, the use of higher nitric acid concentration that promotes acid-catalyzed hydrolysis and the β,γ induced solvent degradation. Upon contact with nitric acid, tri-n-butyl phosphate (TBP) undergoes chemical degradation primarily through hydrolysis and de-alkylation reactions. The principal reactions involved are described below (18-20)

$(C_4H_9O)_3PO + H_2O \rightarrow (C_4H_9O)_2PO(OH) + C_4H_9OH$ **(18)**

$(C_4H_9O)_2PO(OH) + H_2O \rightarrow (C_4H_9O)PO(OH)_2 + C_4H_9OH$ **(19)**

$(C_4H_9O)PO(OH)_2 + H_2O \rightarrow H_3PO_4 + C_4H_9OH$ **(20)**

A comprehensive study was conducted on the alpha radiolysis behavior of tri-n-butyl phosphate in n-dodecane (TBP/n-DD) solvent system using plutonium nitrate solution as an intrinsic α-radiation source. Based on the result, it was observed that the presence of acid during irradiation in addition to cumulative alpha absorbed dose plays a vital role. The factors that are affected primarily upon alpha irradiation are the extent of TBP degradation and di-butyl phosphate (HDBP) formation which in turn causes plutonium and ruthenium retention in the organic phase during stripping and clean up procedures. The G-value of TBP under various alpha radiolytic experimental conditions were estimated and are compared in table 2 against literature reported values. Based on Table 2, the G value for TBP in present study resembles closely with reported literature values upon alpha or gamma irradiation condition [8-10].

Table 2. Comparison of G-values of TBP under various conditions of irradiation.

Experimental conditions	G^{-}_{TBP} / μmol . J^{-1}	G^{+}_{DBP} /μm olJ^{-1}	Ref.
1.1 M TBP with 10 g.L^{-1} Pu, 0.5 M HNO_3	0.431	0.047	This study
1.1 M TBP with 10 g.L^{-1} Pu, 0.1 M HNO_3	0.618	0.077	
1.1 M TBP with 20 g.L^{-1} Pu, 0.5 M HNO_3 (Low specific activity)	0.534	0.041	
1.1 M TBP with 20 g.L^{-1} Pu, 0.5 M HNO_3 (High specific activity)	0.613	0.082	
1.1 M TBP with Pu^{238} (Dose: 0-160 kGy)	0.622	0.104	10
1.1 M TBP upon γ irradiation at room temperature	0.561	NA	9

*NA: Not available

These reactions produce degradation products such as di-n-butyl phosphate (HDBP), mono-n-butyl phosphate (H_2MBP), and ortho phosphoric acid. H_2MBP and H_3PO_4 have minimal impact on the process due to very low concentration and high aqueous solubility. Hence majority of these degradation products are routed to the aqueous waste and filtrate stream.

Di-n-butyl phosphate (HDBP), the primary acidic degradation product of TBP, exhibits strong complexing behavior toward multivalent metal ions owing to its organophosphate structure and acidic proton. In dilute nitric acid, HDBP partially dissociates to form the mono anionic species DBP^{-}, which acts as a bidentate or bridging ligand through its phosphoryl (P=O) and deprotonated oxygen ($P-O^{-}$) atoms [11]. This chelating capability enables HDBP to form highly stable inner-sphere complexes with hard Lewis acid metal ions, particularly tetravalent and hexavalent actinides and results in increased plutonium retention in the organic phase during stripping operations [6,7].

For fission products like Zr and Ru which prefer to have high coordination numbers, HDBP competes effectively with nitrate ligands, displacing them to form neutral or weakly charged metal–DBP complexes. These species display enhanced affinity for the organic phase, leading to their co-extraction with uranium and plutonium and a consequent reduction in decontamination factor. Furthermore, HDBP's amphiphilic properties drive interfacial accumulation, stabilizing metal-HDBP complex emulsions that increase the phase disengagement time.

To mitigate the adverse effects of TBP degradation products, particularly HDBP, systematic solvent washing and regeneration strategies are employed. Alkaline carbonate wash are commonly used to selectively remove HDBP from the organic phase by converting them into water-soluble salts. This is typically followed by dilute nitric acid wash to recondition the solvent and to remove residual alkali entrained in to the organic phase.

4. Diluent degradation and treatment

In addition to extractant degradation, deterioration of the hydrocarbon diluent represents a significant chemical challenge in the PUREX process, particularly under high radiation and acidic operating conditions encountered during fast-reactor fuel reprocessing. Aliphatic diluents such as n-dodecane are susceptible to both radiolytic and chemical degradation, leading to the formation of a variety of reactive organic species that adversely affect solvent performance. Under intense γ- and α-radiation fields, diluent molecules undergo radiolysis, producing alkyl radicals, olefins, and low-molecular-weight hydrocarbons. These reactive intermediates further react with nitric acid and forming nitroalkanes, nitrites, and oxygenated compounds such as alcohols, ketones, and carboxylic acids. The degradation products of the diluent are often surface active and tend to accumulate at the aqueous–organic interface, impairing phase disengagement and increasing the propensity for emulsion formation [12]. Diluent degradation also alters key physical properties of the solvent like density, viscosity, and dielectric constant, thereby influencing extraction equilibria and mass-transfer kinetics.

In addition, solvent regeneration schemes incorporating adsorptive media (e.g., alumina or silica) are generally implemented to control the build up of degradation products and radiolysis by-products. Vacuum distillation is also practiced as a secondary solvent cleanup technique to remove diluent degradation products and other high-boiling impurities, thereby restoring solvent quality. Effective solvent clean up is essential to maintain phase separation characteristics, preserve extraction performance, and ensure long-term solvent recyclability under high radiation PUREX operating conditions.

5. Process chemistry of Np in fast reactor fuel reprocessing

The natural abundance of ^{237}Np is negligible, and it is generated almost exclusively in nuclear reactors through neutron induced reactions followed by radioactive decay. The production routes and resulting inventory of ^{237}Np are strongly dependent on the neutron energy spectrum. Thermal reactors produce ^{237}Np primarily via successive neutron-capture reactions on ^{235}U; however, this contribution remains limited owing to the low isotopic abundance of ^{235}U and competing fission reactions. In fast reactors, the harder neutron spectrum and higher fast neutron flux significantly enhance threshold reactions such as ^{238}U (n,2n) ^{237}U, followed by β^- decay to ^{237}Np, leading to formation of substantially higher ^{237}Np concentration than those encountered in thermal reactors. The enhanced generation of ^{237}Np in fast reactors has important implications for fuel cycle management and reprocessing strategies. Moreover, because ^{237}Np serves as a key target nuclide for the production of ^{238}Pu used in radioisotope power systems for space applications, its efficient separation and recovery from spent nuclear fuel have become a subject of considerable scientific and technological interest. In this context, suitable analytical method for the estimation of ^{237}Np in FBTR spent fuel was developed and the inventory to the plant was ascertained [13].

Under typical PUREX conditions, neptunium can exist in three oxidation states such as Np(IV), Np(V) and Np(VI) and its extractability by TBP is strongly oxidation state dependent, following the order Np(VI) >Np(IV) ≫ Np(V). The speciation of Np in nitric acid solution is governed primarily by the solution acidity and the concentration of nitrous acid (HNO_2). At low HNO_2 concentration, Np(V) is readily oxidized to the more extractable Np (VI), whereas at elevated HNO_2 levels, the reduction of Np (VI) to the poorly extractable Np(V) becomes dominant. Consequently, HNO_2 plays a dual role in Np redox chemistry, acting either as an oxidant or as a reductant depending on its concentration, thereby exerting a decisive influence on Np behaviour within the PUREX flow sheet.

The oxidation state of Np in the conditioned feed solution is crucial in determining whether Np enters the product stream or the HLW. In reprocessing of spent fuel from FBTR, to obtain improved decontamination with respect to ruthenium a troublesome FP & minimize the loss of Pu in HLW, high-acid flowsheet condition is employed in CORAL and DFRP. Under these conditions, major quantity of Np exist in +6 which favor the extraction of Np in organic phase consisting TBP/n-DD system. During aqueous phase partitioning process, plutonium is reduced to Pu (III) by external addition of hydrazine stabilized uranous nitrate rendering it in-extractable form. Under this reducing environment, oxidation state of all Np is changed to Np (IV) and uranium remains in U(VI) oxidation sate. When this conditioned process solution containing Pu (III), Np (IV) and U(VI) is contacted with 30 % TBP/n-DD system, Pu follows aqueous route whereas, Np and U are extracted into organic phase.

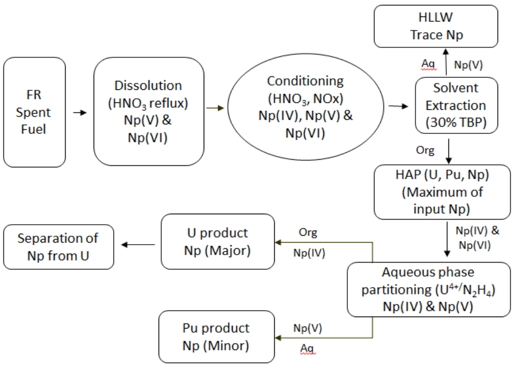

Figure 8. Routing of Np identified based on the determination of Np in various process streams.

A small portion of Np (IV) follows the Pu stream during aqueous phase partitioning as its distribution ratio is lower with respect to the feed acidity typically maintained for partitioning scheme. For the fast reactor spent fuel reprocessing employing a high acid flowsheet condition and aqueous partitioning, Np mass balance were performed and it is given in fig. 8 which indicates major portion of the neptunium reports to the uranium product stream [14].

Therefore, attempts for the laboratory scale separation of ^{237}Np from the U product were initiated and successfully isolated significant quantity of ^{237}Np from the U_3O_8 product. In the present work, the oxide product is converted to nitrate form and the Np is separated from both Pu & U by performing suitable solvent extraction. Stabilizer and reducing agent were maintained in appropriate concentration which helps to reduce the Np to Np(IV) and Pu to Pu(III). Subsequently, the separated Np after stripping with nitric acid was further purified by ion exchange process. Overall decontamination factor obtained with respect to Pu and U in the final Np product by following this separation scheme is 6.6×10^3 and 1.1×10^6 respectively. The neptunium nitrate solution thus obtained was precipitated and calcined at 550°C to prepare neptunium oxide. The purified Np $(NO_3)_4$ solution and NpO_2 are shown in figure 9.

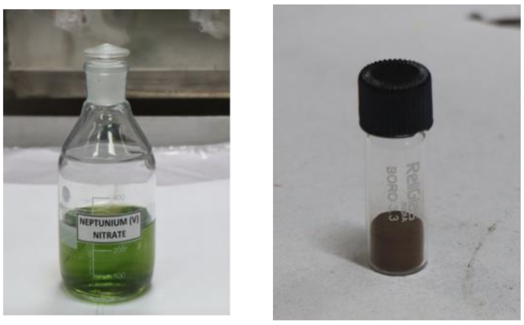

Figure 9. (Left) Purified $Np(NO_3)_4$ and (right) Purified NpO_2.

The $^{237}NpO_2$ thus obtained was characterized by various analytical techniques and found to be suitable as a target material in the production of ^{238}Pu in the reactor.

6. Recovery & purification of ^{90}Sr from FBTR spent fuel and production of carrier-free ^{90}Y

The utilization of radioisotopes for the diagnosis and therapy of various diseases is a key program of the Department of Atomic Energy. Several radionuclides are produced using research reactors and particle accelerators for applications in nuclear medicine. Another valuable source of radionuclides is high level waste (HLW), generated during the reprocessing of high burnup fast reactor spent fuel via the modified PUREX process [1]. This waste stream is rich in radionuclides such as ^{137}Cs, ^{90}Sr, ^{106}Ru, ^{144}Ce, ^{237}Np, ^{241}Am, ^{242}Cm etc making it a resource rather than a waste. One of the most significant radionuclides present in HAW is ^{90}Sr, which decays to ^{90}Y, a pure beta emitter (β_{max}= 2.28 MeV, $T_{1/2}$= 64.1 h) with strong therapeutic potential in nuclear medicine. With its availability in large quantities from HAW, ^{90}Sr serves as a long-lived parent for generating carrier-free ^{90}Y.

However, separation and purification of such radionuclides from HLW is highly challenging. To isolate ^{90}Sr from high level liquid waste (HLLW), several techniques have been developed, including precipitation, liquid–liquid extraction, extraction chromatography, and ion exchange chromatography. Among these, solvent extraction has gained considerable attention due to its established application in spent fuel reprocessing and its multiple advantages such as ease of handling, remote operability, flexibility in stage design and scalability [15,16].

The Reprocessing Group at IGCAR made a collective effort to selectively separate ^{90}Sr from high level waste (HLW) using a solvent extraction experiment inside a hot cell, DFRP (Fig. 10,11).

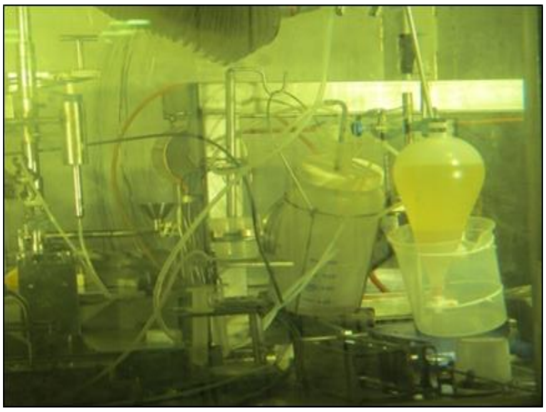

Figure 10. Hot cell operation for the separation of ^{90}Sr form HLW of fast reactor reprocessing.

The extractant used was 0.1 M 4, 4′, 5-di-(t-butylcyclohexano)-18-crown-6 in a mixture of 1:1 n-octanol and dodecane. Due to the presence of alpha-emitting impurities, fission products, and various metallic elements in the waste solution, a single separation technique was insufficient to achieve the desired radionuclide purity of ^{90}Sr. Therefore, a combination of separation techniques including solvent extraction and precipitation was employed for its purification. Upon separation and purification, ^{90}Sr can serve as a long lasting source for the generation of carrier-free ^{90}Y.

A solvent extraction-based separation method was developed in laboratory at RpG, IGCAR, for the separation of ^{90}Y from ^{90}Sr, and it has proven to be a convenient system for milking ^{90}Y. (Fig. 12)

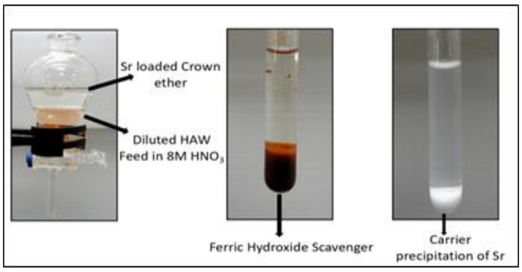

Figure 11. Purification of ^{90}Sr from alpha impurities.

Extensive studies have shown that the generator system is suitable for obtaining clinically acceptable ^{90}Y, with a breakthrough level of ^{90}Sr less than 10^{-6} Ci/Ci of ^{90}Y. The alpha contamination in purified ^{90}Y are found be $<10^{-9}$ Ci/Ci of ^{90}Y. The metallic impurity present in purified 90Y solution was analysed by ICP-OES was found within the acceptable limit.

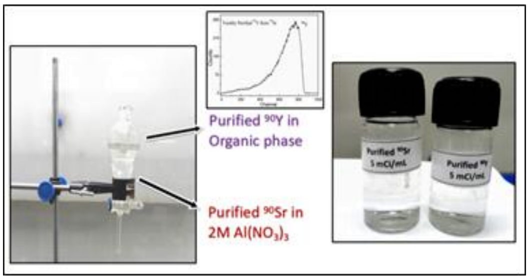

Figure 12. Milking of ^{90}Y from Purified ^{90}Sr.

7. Analytical method development for multi-component analysis in fast reactor reprocessing stream using chemometric approach

To meet design requirements, the PUREX process necessitates quantification of uranium, plutonium, and nitric acid in process streams, ensuring plutonium losses to HLW remain below 5 mg/L and a fission-product decontamination factor of ~10^6 is achieved [2]. Conventional analytical methods for U, Pu and HNO_3 are time consuming, generate large volumes of secondary radioactive analytical waste, and increase radiation exposure, especially when handling short-cooled, high burn-up fuels. Hence, real-time or near-real-time process control techniques are essential for fast reactor fuel reprocessing plants. UV–Visible spectrophotometry provides a rapid, non-destructive measurement of chemical composition, but spectra from complex mixtures often suffer from overlapping signals and noise. Applying chemometric methods such as partial least squares (PLS), principle component regression (PCR), or orthogonal projection to latent structure (OPLS) enables reliable deconvolution of these spectra and accurate determination of U, Pu, and HNO_3 concentrations. This integrated spectrophotometric–chemometric approach ensures efficient, low-waste, and safer process control [17].

<u>Simultaneous determination of uranium and free acidity</u>

Unlike lanthanides, the actinide elements exhibit multiple oxidation states (+3, +4, +5, +6 &+7) due to the poor shielding of 5f orbitals and exhibits difference in chemical properties. But, in case of lanthanides, 4f orbitals are more contracted and localized close to the nucleus, therefore, the electrons in 4f orbitals possesses more screening effect than that of 5f electrons. Hence, lanthanides exhibit +3 oxidation state. Moreover, the aqueous solution containing actinides, metal ion with oxidation state beyond +4 exists mainly as oxo-cation ($AnO_2^{(n-4)+}$).

In uranyl ion (UO_2^{2+}), uranium is in the +6-oxidation state ($5f^0$ configuration) bonded linearly to two oxygen atoms. Molecular orbital (MO) theory describes its bonding through combination of 5f/6d orbitals of U and 2s/2p orbitals of oxygen. The key bonding orbitals (σ_u, σ_g, $2\pi_u$, $2\pi_g$) accommodate 12 valence electrons, all from the 2p orbital of oxygen hence having mostly P character. The energy order of occupied orbitals is $\pi_g < \pi_u < \sigma_g << \sigma_u$, with σ_u as the HOMO, influenced by antibonding overlap between oxygen $2p_z$ and uranium $6d_{z^2}/5f_{z^3}$ orbitals. The LUMOs δ_u (f_{xyz}, $f_{z(x^2-y^2)}$) and φ_u ($f_{y(3x^2-y^2)}$, $f_{x(x^2-3y^2)}$) are nonbonding in uranium 5f orbital. Electronic transitions from $\sigma_u \rightarrow \delta_u$ and $\sigma_u \rightarrow \varphi_u$ cause characteristic UV–Vis absorptions. The excited state configuration in this transition are predicted as $^1\Phi_g$ ($\sigma_u\varphi_u$) or $^1\Delta_g$ ($\sigma_u\delta_u$). These LMCT transitions, $^1\Sigma^+_g \rightarrow {}^1\Phi_g$ and $^1\Sigma^+_g \rightarrow {}^1\Delta_g$ are spin allowed, but forbidden because of the transition between the energy states with same parity ($g \rightarrow g$, g: gerade) [18]. These transitions are parity-forbidden (g → g), but gain intensity through vibronic coupling or ligand-field distortion. These ligands to metal charge transfer (LMCT) transitions ($^1\Sigma_g^+ \rightarrow {}^1\Phi_g$ or $^1\Delta_g$) produce yellow coloration and weak absorptivity. The UV-Visible absorption spectra of U(VI) in 3 M HNO_3 (400–470 nm) show that absorbance at all fine structure maxima increases linearly with uranium concentration (1.1–20.3 g L^{-1}) as shown in fig. 13, demonstrating adherence to the Beer Lambert law under these conditions.

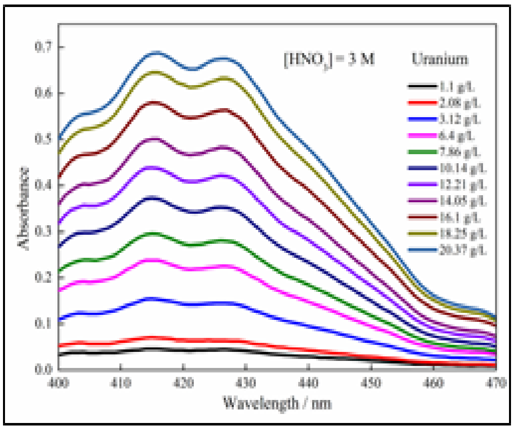

Figure 13. Uranyl nitrate UV-Visible absorption spectra as a function of U concentration.

However, when the nitric acid concentration is varied, significant spectral changes occur like the wavelength maxima shift, and band shapes evolve (sharp at low acidity, broadened at ≈6 M, then sharpening again at >6 M with further red-shifts) as illustrated in fig. 14.

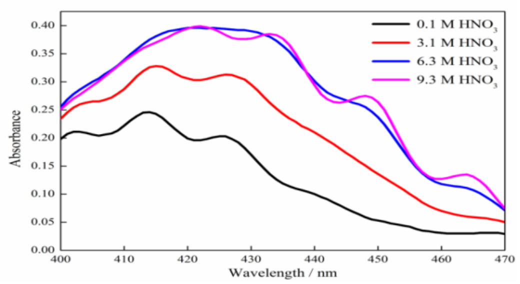

Figure 14. Uranyl nitrate absorption spectrum as a function of nitric acid concentration.

These shifts reflect changes in the speciation of uranyl nitrate complexes. Uranyl nitrate complexes show strong spectral sensitivity to solution conditions. Typically, UO_2^{2+} predominates in acidic media (pH 0.1–2), exhibiting characteristic peaks at 403, 414, and 426 nm. Increasing nitrate concentration alters uranyl speciation and shifts absorption maxima, complicating UV–Vis quantification of uranium in nitrate matrices. The uranyl nitrate complex usually exists as the mononitrate or dinitrate; however, the uranyl trinitrato complex can be prepared in nitric acid of higher concentration (>10 M). Therefore, finding the concentration of U is not feasible by means of direct UV–Visible spectra without knowing acidity. However, the acidity induced variation can be utilized for the determination of feed acidity and uranium concentration using the absorption spectra with chemometric modelling. Therefore, an orthogonal projection to latent structure (OPLS) assisted principal component regression (PCR) chemometric model was developed for simultaneous quantification of uranium (3–21 g L^{-1}) and nitric acid (2–12 M) from UV-Visible absorbance spectra recorded in the wavelength region 400–470 nm. Here OPLS enables in removing the variations not related to target variable. For test samples, the model yielded root-mean-square errors of prediction (RMSEP) of 0.7 g L^{-1} for uranium and 0.4 M for acidity.

This method was validated using samples generated during the cold run of DFRP. The results obtained were in good agreement with the conventional methods. Also, in reprocessing plant this method can be followed for process control analysis of uranium product stream samples.

Simultaneous determination of plutonium and free acidity

Plutonium exhibits multiple oxidation states in nitric acid media because the redox potentials of Pu(III)/Pu(IV)/Pu(VI) couples lie very close to each other. In solutions with acidity below 1 M HNO_3, Pu(IV) undergoes disproportionation to Pu (III) and Pu(VI), and it is more pronounced if acidity is below 0.5 M, whereas in reprocessing streams the plutonium valence is controlled and maintained in the +4 state due to maintaining feed acidity of samples beyond 1 M in view of reducing potential for hydrolysis. In nitric acid solution, Pu (IV) shows characteristic 5f–5f absorption bands in the UV–Visible region arising from intra-configurational $5f^4$ transitions, giving a brown–yellow colour at low acidity that change to green at HNO_3 concentrations above 6M [19]. The spectrum is structured and of low molar absorptivity (Laporte-forbidden), but the bands are sufficiently intense and well defined for quantitative spectrophotometric determination at fixed acidity as shown in fig 15.

The absorption band positions and intensities depend strongly on nitric acid concentration because of progressive complexation by nitrate ligands and changes in the Pu(IV) coordination environment. When the nitric acid concentration is increased from about 1.3 M to 12 M, the main Pu (IV) absorption maximum near 475 nm at low acid shifts to longer wavelengths, and the band shape and intensity change systematically as shown in figure 16. This behaviour reflects a speciation shift from predominantly hydrated Pu (IV) to a series of nitrato complexes, including mono-, di-, tetra- and ultimately hexa nitrato species, each with a distinct ligand field that perturbs the 5f energy levels exhibiting variation in wavelength of maximum absorbance (λ_{max}) and band intensity. As a result, direct spectrophotometric quantification of Pu (IV) across varying acidities is unreliable unless the nitric acid concentration (and therefore speciation) is controlled.

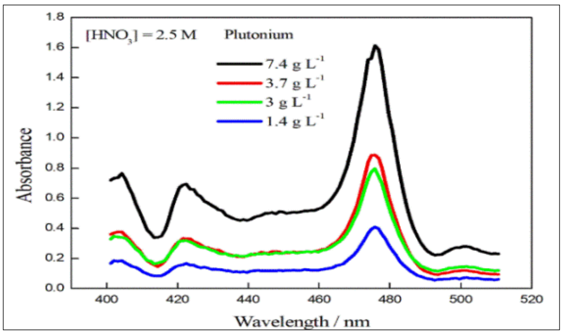

Figure 15. Plutonium nitrate absorption spectrum as a function of plutonium concentration.

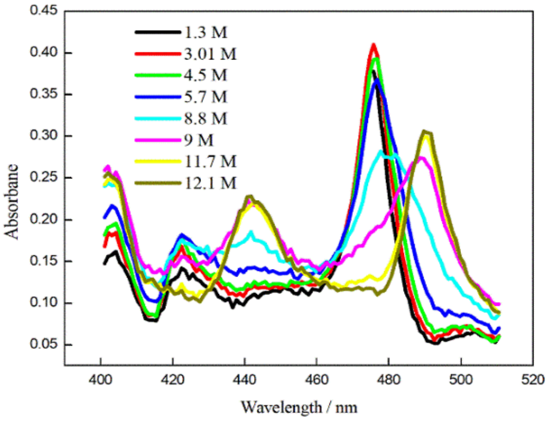

Figure 16. Plutonium nitrate absorption spectrum with varying concentration of HNO_3.

To overcome this limitation, a combined UV–Vis spectrophotometry and chemometric method was developed using an orthogonal-partial least squares (OPLS) enhanced Principal Component Regression (PCR) model. The model was built using Pu(IV) concentrations of 1.4–8.0 g·L^{-1} and nitric acid concentrations ranging from 1–12 M, thereby capturing the full range of spectral variation due to both analyte concentration and acidity. Validation on test samples delivered root-mean-square errors of prediction (RMSEP) of 0.25 g·L^{-1} for Pu and 0.25 M for nitric acid, indicating high accuracy and precision [20]

Spent fuel with 155 GWD/ tonne burn-up from Fast breeder test reactor (FBTR) being reprocessed at CORAL facility and this method was validated for quantification of plutonium and acidity simultaneously in the process samples received after HC run.

The interference from the uranium can be cut-off by doing the calibration in the 460-520 nm range as in this region uranium interference is minimum. Further, work is in progress for the simultaneous determination of uranium, plutonium and free acidity in samples in presence of high levels of fission product activities.

8. Conclusion and future Outlook

The solvent extraction behaviour of plutonium and uranium present in the plutonium rich fuel solution is quite different from thermal reactor fuel reprocessing, which allows to develop advanced process flow sheet for reprocessing plutonium rich fuels. In fast reactor fuel reprocessing, the requirement of lower plutonium inventory into the organic phase due to third phase formation constrains necessitated the adoption of a high acid flow sheet to achieve the required decontamination factor from fission products. The co-stripping of uranium and plutonium from the loaded organic phase using dual nitric acid is another unique feature adopted in FR fuel reprocessing, enabling the aqueous partitioning of procedure with stoichiometric quantity of U(IV) in FR reprocessing.

The ^{237}Np inventory in FR spent fuel was higher by an order than that observed TR spent fuel. The high acid flow sheet employed for FR fuel reprocessing and aqueous partitioning procedure with stoichiometric routed maximum ^{237}Np to the uranium product stream. The experience gained in the recovery of ^{237}Np from the U product stream of FR spent fuel gave confidence in the recovery of this minor actinide in large scale FR reprocessing plants. This strategy also paves way to the separation of significant amount ^{237}Np required for the production of ^{238}Pu which in turn required for the application in space missions.

Similarly, the FR fuel undergoes burning up to 15 atom%, which produces higher quantity of ^{90}Sr/kg of the spent fuel. The preliminary attempts for the recovery of ^{90}Sr from the HLW stream generated from the FR spent fuel showed that, the recovery and purification was feasible. Milking of alpha impurity free ^{90}Y from the ^{90}Sr was demonstrated for meeting the high demand of the radiopharmaceutical industry. The feasibility of recovery of other valuable isotopes like ^{107}Pd, ^{137}Cs, ^{241}Am etc. from the reprocessing waste are also being explored in the reprocessing group.

The remote analytical methods and facilities amenable for the monitoring of Pu, U and free acidity were vital for reprocessing plant. In this context, the application of multivariable chemometric method developed for FR reprocessing showed significant promise in reducing the man rem exposure. In conclusion the advancements reported in this article paves a way for efficient, sustainable and economically viable reprocessing strategies not only for FR reprocessing but also thermal fuel reprocessing.

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Process Design Aspects of Fast Reactor Fuel Reprocessing Plants

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Abstract

The spent nuclear fuel discharged from fast reactor is processed by an aqueous based PUREX process for the recovery of uranium and plutonium. The complexity of fast reactor fuel reprocessing in this case is mainly due to the processing of short cooled, high burn up and plutonium rich spent fuel as compared to thermal reactor fuel reprocessing. In view of this, the processing of the spent fuel from fast reactors require heavily shielded process cells equipped with remote operation and maintenance technology, alpha tight transfer systems, leak tight containment systems/cells, employment of critically safe tanks, prevention of undesirable third phase formation during solvent extraction and achieve the required plant performance. Based on the experience gained in the design of fast reactor fuel reprocessing facility at IGCAR, this paper highlights the general process design aspects of fast reactor fuel reprocessing plant.

Keywords: *Fast reactor, Fuel reprocessing, Process design, Uranium, Plutonium.*

1. Introduction

Fast reactor fuel reprocessing plant handles and processes the spent fuels with higher level of plutonium content (max 70% in FBTR & 28% in PFBR) and burn-ups (max 155 GWd/Te in FBTR & 100 GWd/Te in PFBR). This necessitates processing of the spent fuel inside the thick shielded process cells (concrete and lead cells) equipped with remote operation and maintenance technology, alpha tight transfer systems, employment of critically safe tanks for handling plutonium solution, systems to prevent undesirable third phase formation during solvent extraction, and achieving the required decontamination factor and maximum recovery. A simplified and proven PUREX flow-sheet is adopted for achieving the required overall decontamination factor (10^7) and maximum recovery (99.9%).

A typical reprocessing plant consists of head end section, process section, reconversion section, evaporation and waste storage facilities, ventilation systems and other process support systems, etc. For a given throughput, the overall size of the plant is determined by the number of solvent extraction (SX) cycles adopted, type of SX equipment employed, mode of operation in head end and reconversion sections (batch or continuous), tank capacities etc. A smaller plant size with lower capital, operation & maintenance (O&M) and decommissioning costs is always desirable, without compromising the required plant capacity, safety, DF and recovery. This has a direct impact on the economics of fuel cycle and ultimately on the cost of power. In addition, the overall fissile inventory stored in the plant (for lesser doubling time) would also be minimum.

Based on the experience gained in the design of fast reactor fuel reprocessing facility at IGCAR, the general process design features of the fast reactor fuel reprocessing plant and its process systems are discussed in the paper.

2. Processing steps

The typical processing steps involved in the fast reactor fuel reprocessing plant is given in figure 1.

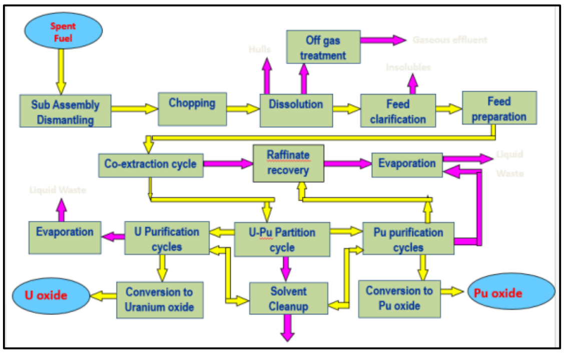

Figure 1. Process flow diagram for a typical fast reactor fuel reprocessing plant.

In the chop-leach PUREX method employed widely in the MOX spent fuel reprocessing plants, irradiated fuel subassembly received from the fast reactor is dismantled and the fuel pin bundle is retrieved from the wrapper. After segregation, the fuel pins are transferred to a container and transported for loading into chopper magazine. From the magazine, the fuel pins are fed to a multi pin chopper where the fuel pins are chopped into pieces to expose the fuel element. The chopped fuel pins fall into a dissolver. Concentrated nitric acid near boiling conditions is used for dissolution of MOX fuel (<30%Pu), whereas Electro-Oxidative Dissolution Technique (EODT) is used for dissolution of MOX fuel (Pu >30%). For mixed carbide fuel (70%Pu), dissolution with concentrated nitric acid near boiling condition, followed by refluxing is being carried out to destroy organics. After dissolution, the hulls are removed from the dissolver and monitored for disposal. The solution from the dissolver is clarified and conditioned. The presence of platinum group metal fission products is significant in fast breeder reactors, which are insoluble in dissolver. These insoluble residues are clarified in solid bowl centrifuge and the clarified solution is conditioned for acidity and valency adjustment of U & Pu by NO_2 gas. The conditioned solution is subjected to extraction and stripping. After stripping both U and Pu, a partitioning step was adopted to separate U & Pu using U(IV) solution. U & Pu nitrate solution after partitioning cycle is sent to respective purification cycles for further decontamination if required and sent to respective reconversion section for conversion to PuO_2 & U_3O_8 powder. The liquid wastes generated are evaporated for volume reduction before storage. Diluent wash is carried out to remove the TBP from the aqueous wastes before evaporation.

3. Material of construction

The tanks, equipment and pipes, handling highly radioactive and acidic solution, are designed and constructed for a minimum service of 40 years of plant life. This is achieved by incorporating appropriate choice of material, equipment/component design, best fabrication practices, thorough inspection techniques and remote maintenance. The process medium being nitric acid, austenitic stainless steel is the material of construction, as it is the best commercially available materials for meeting the conditions in fast reactor reprocessing plants. Austenitic stainless steels are susceptible for intergranular corrosion (IGC) in nitric acid medium and hence, SS304L with reduced carbon content of maximum 0.025% is used as a material of construction for process tanks, equipment and pipes except dissolver and associated piping system. For dissolving plutonium rich fuels, dissolution of chopped fuel pins is carried out in thermo-syphon dissolver in concentration nitric acid (11.5 M) near boiling condition. For these highly corrosive conditions, titanium grade 2 is selected as a material of construction for dissolver and its associated piping, and is connected with other plant piping (SS 304L) through dissimilar Ti-SS304L joint.

4. Tanks, Equipment and Cell layout

<u>Tanks and equipment layout considerations</u>

Storage tanks and equipment such as evaporators, diluent wash columns etc. are housed in shielded concrete cells without the requirement of remote maintenance. The Pu process solution is stored in the annular tanks based on criticality safety consideration. Pu lean solution are stored in cylindrical tanks. Because of high Pu content in fast reactor fuel reprocessing plant, annular tanks are used from dissolution onwards unlike thermal reprocessing plants, which occupies much space in the shielded concrete cells. The tanks with cooling jackets are provided for storage of solution with high decay heat, such as dissolver solution storage tanks, co-decontamination conditioner tanks and raffinate storage tanks. The minimum separation distance between tanks is governed by criticality safety requirements which cannot be compromised. In addition, adequate space is provided for ease of erection and possible future maintenance. Access platform is provided around the tanks and equipment for ease of erection, inspection and maintenance. Tanks and equipment are located along the periphery of the cells to facilitate man and material access for maintenance and decommissioning. The 3D view of tanks and its supports in the typical process cell is shown in figure 2.

Tributyl phosphate (TBP) in n- dodecane is used as the primary solvent for solvent extraction in reprocessing plants, which undergoes degradation due to radiation. Hence the tanks holding solvent are isolated from high active tanks and kept in separate cells. The provision for fissile material tracking are provided for proper nuclear material accounting. Optimum capacity of the storage tanks is provided considering the operation flexibility. An optimized number of equipment, vessels and transfers are provided to ensure plant availability and operability. The overall cell layout is optimized by meeting the requirements of criticality safety, erection feasibility, ease of maintenance, minimal shielding and thereby resulting in optimal capital cost.

Figure 2. 3D view of tanks and supports in the typical process cell of DFRP.

Criticality aspects

One of the major design objectives of the plant is to prevent the occurrence of the nuclear criticality accident. Therefore, the individual tanks and equipment are designed to be critically safe. However, the number of closely located sub-critical systems may become critical due to neutron interaction between adjacent systems. In view of this, criticality analysis is carried out considering the maximum fissile material inventory under off normal conditions to arrive at the layout of tanks and equipment. The criticality alarm systems are also provided in areas handling fissile material.

Evaporator cells

Evaporators are located in dedicated & isolated shielded cubicles for ease of access. Entry into the shielded cubicle is provided via shielded door at the base and roof of the cell for the contact maintenance and replacement of evaporator components during operating life of the plant.

Dissolution with soluble neutron poison

In large-capacity plants, criticality control by controlling only the fissile concentration and dissolver geometry would be impractical. Hence a dissolution process employing a soluble neutron poison for criticality control has been adopted in high throughput fast reactor reprocessing plants. The impact of the soluble neutron poison on solvent extraction performance has been evaluated and found to be satisfactory.

Hot cells with remote operation and maintenance

Due to handling of high radioactivity, short contact time centrifugal extractors are employed as solvent extraction equipment in the fast reactor reprocessing plants in order to minimize the solvent degradation due to radiolysis. Centrifugal contactors have rotating parts which requires remote O&M. For sampling of radioactive solutions, vacuum aided airlift remote sampling system is adopted which require remote O&M. Hence centrifugal extractors and sampling systems are housed in specially designed containment box hot cells. These hot cells are provided with radiation shielding viewing windows (RSW), master slave manipulators (MSM) and special in-cell cranes for carrying out remote O&M. All extractors and sampling system equipment are housed inside SS304L containment boxes surrounded by lead walls for shielding. Equipment layout for these cells has to be designed considering the reach of remote handling gadgets in addition to the criticality safety and process requirements. Since open handling of Pu bearing solutions is done in these cells, they are qualified for leak tightness and provided with alpha tight transfer system to prevent plutonium contamination of external surfaces during O&M. Hot cells with MSM, RSW are shown in figure 3. The view of centrifugal extractor bank is shown in figure 4.

Figure 3. Operating area outside containment box showing MSM and RSW provision.

Figure 4. Centrifugal extractor bank.

Waste tank farm (WTF)

The Waste Tank Farm (WTF) houses storage tanks for high-level waste (HLW), intermediate-level waste (ILW), organic liquid waste (OLW), and analytical liquid wastes. The material of construction for all tanks is SS 304L. The WTF is located underground to take advantage of radiation shielding provided by the surrounding soil and thereby minimize additional radiation shielding requirements.

The tanks are constructed in multiple units, collectively referred to as a tank farm, so that instrumentation and servicing facilities can be shared. Adequate numbers of storage tanks are provided for Category V waste to allow decay of Ru-106 activity prior to transferring the liquid waste for vitrification. These Category V waste storage tanks are provided with provisions for process cooling water inlet and outlet, air inlet for ballast tank operation to keep particulate matter, if any, in suspension, and for sweeping of radiolytic hydrogen.

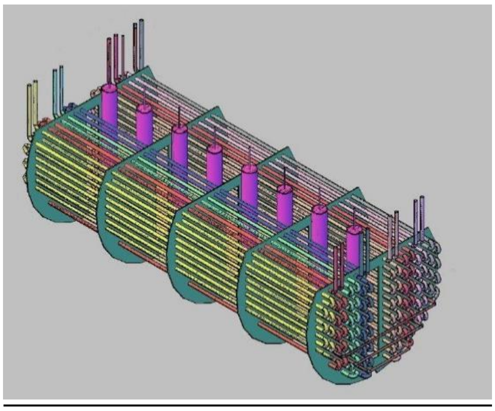

Figure 5. 3D model showing the internals of the HLW tank of FRP—process cooling water coils, support baffles, and ballast tanks.

Figure 6. HLW tank of FRP under fabrication.

All tanks are housed within SS-lined concrete dykes designed to contain the full volume of the tank in the event of leakage. Each dyke is provided with a sump for leak detection. A roof of high-density concrete is provided over the WTF tanks to serve as radiation shielding. Figure 5 & 6 gives the details about HLW tank of FRP.

Provisions are incorporated in the layout for lowering of in-service inspection (ISI) vehicle and for their movement along fixed rails. ISI helps in regular monitoring of healthiness of process tanks and equipment in service without affecting the regular plant operation.

Infiltration gallery

For process cells located at underground elevations, such as the Waste Tank Farm and LLW HUT areas, an enclosed concrete structure referred to as the "*infiltration gallery*" is provided as a protective measure. Any groundwater ingress is collected in the infiltration gallery. Sumps are provided within the infiltration gallery to detect groundwater ingress and, after sampling, to pump the collected water to the storm water drain. The internal surface of process cells are lined with SS sheets for the ease of decontamination and additional containment.

5. Liquid transfer systems

The radioactive liquid transfer is carried out by simple and reliable systems that have no moving parts and therefore, does not require maintenance. Airlift liquid transfer and Steam jet transfer are generally employed for these reasons. Steam jet transfers are used for solution containing nil or very low plutonium content, whereas airlift liquid transfers are used for all solutions, including plutonium bearing solutions. Airlifts are the primary mode of liquid transfer. In this mode, about 30% submergence is ensured by locating the foot piece for air entry below the tank bottom at appropriate elevation. Two stage airlifts are used to develop higher discharge head. Metering air lifts are two stage transfer air lifts used to feed solution at a constant rate to process equipment. It consists of a primary airlift and a secondary airlift. The constant submergence of the metering airlift is achieved by the primary airlift (by maintaining the liquid level in the primary airlift vent pot via an overflow). The secondary airlift pumps from the primary airlift vent pot to the respective process equipment. The airflow in secondary airlift is controlled to get the required liquid flow. Vacuum aided airlift is a secondary liquid transfer mode which is to be used when the primary mode of transfer is unavailable. The required submergence is only created by vacuum generated by an air ejector in the vent pot vent line. The required submergence can be controlled by adjusting the vacuum created by the air ejector. Emergency airlift is a type of vacuum-aided airlift used for emergency transfer operations. It is employed only in extreme cases, when all other transfer modes provided inside the cell are inoperable. Steam jets are used to transfer active liquid containing no or very less plutonium, such as those in the uranium processing section, evaporation section and waste tank farm. Vacuum killing provision is provided for steam jet transfers to prevent activity blowback in the service piping area.

6. Piping layout in 3D modeling

Piping layout plays a significant role in arriving at the shielded space requirements of a process cell. Erection of large number of small-bore pipe lines (mostly 8 NB to 25 NB) in a confined space, leads to high piping density. If the space is not optimized, it may lead to extraordinary delays in the piping erection and may also pose problems during inspection and maintenance. Owing to the high density of piping, normally the piping layouts are prepared using advanced 3D piping modeling softwares.

Design considerations for piping layout

The major feature that has to be considered in the preparation of piping layout inside the process cells include the provision of exclusion zones like access platform, in-service inspection, material handling during erection & maintenance, adequate clearance and spacing for pipes & pipe supports. Pipe bends are preferred instead of elbows and are made from pipe itself to avoid the additional weld joints. Generally, Pipe bends of radius 5 times the nominal pipe diameter is used for pipe sizes less than or equal to 40 NB. For sizes above 40 NB, miter bends or elbows are provided. Process lines are provided with adequate slope to avoid the liquid holdup after the liquid transfer. This helps prevent the corrosion of pipelines and avoids the stagnation of radioactive liquid in the lines. Sufficient center to center spacing between the pipelines is maintained for ease of fabrication and inspection including radiographic examination of weld joints. Adequate additional area for spare/additional piping is to be envisaged during the design stage. A proper routing sequence is followed to avoid interference between piping. Intercell lines (lines connecting one cell to another) are routed first to minimize the interference with other piping inside the cells. Pipe lines are grouped and laid along the cell wall to facilitate support provision. Thermocouple lines are routed with higher bend radius and minimum no. of bends. Steam ejector is routed with minimum of 1 mtr straight run after the discharge. Piping is routed tier-wise along the cell wall and they are provided with combined pipe supports at regular intervals. Adequate flexibility has been incorporated in steam lines and all lines connecting to evaporators to accommodate thermal expansion. The 3D view of tanks with supports and its piping in the typical process cell is shown in figure 7.

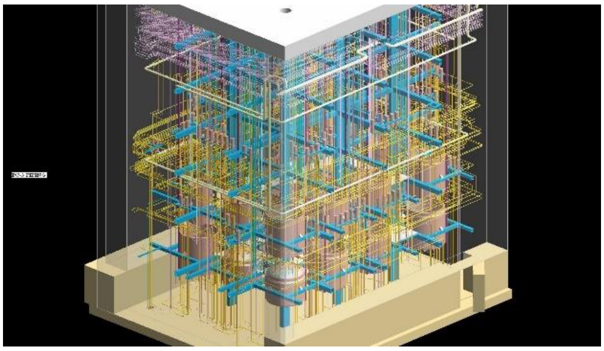

Figure 7. 3D view of tanks, supports and its piping in the typical process cell.

Construction, isometric drawings are also generated from the 3D model to enable efficient execution of construction activities.

7. Conclusion and future Outlook

The reprocessing of spent nuclear fuel from fast reactors presents unique technical challenges arising from its short cooling time, high burnup, and elevated plutonium content. As discussed in this paper, the adoption of an aqueous-based PUREX process for fast reactor fuel requires the integration of robust engineering and safety features, including critically safe equipment, remote operation and maintenance systems and leak-tight containment structures. The design experience gained, demonstrates that these challenges can be effectively addressed through systematic process design, appropriate engineering and operational practices.

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Design of Robust Solvent Extraction Flow-Sheet for Reprocessing of Fast Reactor Fuels by Indigenous Computer Code and Simulation

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Abstract

Solvent extraction operation is the major unit operation in nuclear fuel reprocessing. The computational codes employed for solvent extraction process design and flow-sheet simulation in reprocessing are not available in open literature. A rigorous Solvent Extraction Equilibrium SPEciation Calculation (SEESPEC) code based on solvent extraction thermodynamics was developed in-house for the prediction of distribution coefficient of U(VI), Pu(IV) and HNO_3. The SEESPEC code has been used for the process flow-sheet design and optimization in DFRP. This article discusses the salient feature of the SEESPEC model and the application of the model in various process stages of fast reactor fuel reprocessing.

Keywords: *Solvent extraction, reprocessing, computer code, thermodynamics, uranium, plutonium.*

1. Introduction

Solvent extraction is one of the important unit operations in the spent nuclear fuel reprocessing by PUREX process. The solvent extraction operation is usually carried out in a multistage counter-current contactor for the maximum recovery of the desired product with minimal solvent requirement and high decontamination factor. Therefore, the design of a multistage solvent extraction flow-sheet is vital for the sustained and safe operation of the reprocessing plant, especially for the plutonium rich fuels in fast reactor fuel reprocessing. Higher concentration of plutonium and burn-up achieved in fast reactor fuels introduces additional complexity during the solvent extraction operation prone to third phase formation during extraction and incomplete recovery of plutonium in the stripping cycle due to the presence of solvent degradation products [1]. It should be noted that the hydrolytic and radiolytic degradation products formed in the solvent phase are quite significant in fast reactor fuel reprocessing as compared to thermal reactor fuel reprocessing.

To understand the actinide profiles in reprocessing flow-sheet and to prevent the loss of actinides in the process condition, there is a requirement of computer codes and models for describing the extraction of actinides in organic phase and simulate the solvent extraction flow-sheet in the plant. Since the computational simulation offers a convenient way to study the multiple options rapidly and to arrive at the optimum parameter quickly, it saves lot of costly and time-consuming experimental effort involving radioactive material, especially plutonium. Also, simulation gave the convenience and flexibility to study the condition which leads to the third phase formation and criticality limits which are difficult/unsafe to perform in experiments. However, such computational codes are not available in open literature even for thermal reactor fuel reprocessing, due to confidential policy adopted by the respective countries and secrecy involved in safeguarding the nuclear material. The primitive and empirical declassified computer codes available in open literature for solvent extraction process simulation are also not suitable for Indian reprocessing plants owing to the exclusion of degradation products in the model, higher plutonium content in the feeds e.g. in fast reactors [2]. In view of this, an indigenous computer code is necessary for fast reactor fuel reprocessing.

The multistage flow-sheet simulation consists of two parts wherein the first part computes the distribution coefficient of actinides in organic phase and the second part solves the set of differential mass balance equation to determine the stage concentration profiles. The accuracy of the actinide profile simulation in a multistage solvent extraction bank depends upon the predictive accuracy of distribution coefficient of actinides. In addition, it is necessary to include the influence of degradation product di butyl phosphate (HDBP) in the organic phase for the realistic simulation of stripping operation. Therefore, an in-house computer code SEESPEC (Solvent Extraction Equilibrium SPEciation Calculation) was developed for the accurate prediction of distribution coefficient of U(VI) and Pu(IV) even in the presence of HDBP [3, 4]. The SEESPEC code is the semi-theoretical model where the aqueous phase activity coefficients of various species were estimated through eUNIQUAC activity coefficient model and the organic phase activity coefficients were assumed unity. The equilibrium constants of the various species forms in the organic phase and the interaction parameters in the activity coefficient model were estimated using the experimental data.

This article highlights the development of the SEESPEC code for the prediction of distribution coefficient of actinides, and employment of the code for design simulation and optimization of solvent extraction flow-sheet in the Demonstration fast reactor fuel reprocessing (DFRP) facility.

2. Description of model

2.1. Distribution coefficient model (SEESPEC)

In this theoretical approach, the extraction of U(VI) and Pu(IV) in an organic phase is considered as a complex formation reaction, shown in Eq. (1). The species in the organic phase are denoted as over bar.

$$M^{n+} + n\,NO_3^- + z\,\overline{TBP} \leftrightarrow \overline{M(NO_3)_n.zTBP} \qquad \textbf{(1)}$$

The thermodynamic equilibrium constant (K) for the above reaction based on the law of mass action is shown in equation 2.

$$K = \frac{a_{\overline{M(NO_3)_n.zTBP}}}{a_{M^{n+}}\, a^n_{NO_3^-}\, a^z_{\overline{TBP_f}}} \qquad \textbf{(2)}$$

According to the law of mass action principle, the thermodynamic equilibrium constant is the quotient of chemical activity of products to the activity of the reactants. By rearranging the above equilibrium relation, the distribution coefficient of metal ion (D_M) can be expressed as follows

$$D_M = K\,[NO_3^-]^n\,\left[\overline{TBP_f}\right]^z \frac{\gamma^n_{NO_3^-}\,\gamma_{M^{n+}}\,\gamma^z_{\overline{TBP_f}}}{\gamma_{\overline{M(NO_3)_n.zTBP}}} \qquad \textbf{(3)}$$

where $[\mathrm{NO_3^-}]$ and $[\mathrm{TBP_f}]$ are the concentration of nitrate ion and free TBP respectively, γ is the activity coefficient of species. It is noted that the distribution coefficient is the function of nitrate ion concentration, free TBP concentration and the ratio of activity coefficient of species. The unknown parameter in Eq. (3) is the equilibrium constant and activity coefficients. Determination of activity coefficient of the species involved in Eq. (3) is very complex and in the present study, it was determined by electrolyte UNIversal QUAsi Chemical (eUNIQUAC) model. The interaction parameters in the activity coefficient model and the thermodynamic equilibrium constant (K) were estimated by nonlinear optimization. The objective function is formed by the sum of squared deviation between the experimental and estimated distribution coefficient as follows

$$f_{obj\_fn} = \sum_i \left(D_{M,exp,i} - D_{M,pred,i}\right)^2 \qquad \textbf{(4)}$$

The unknown parameters in Eq. (3) is estimated by the minimization of the above objective function (eq. 4). In Eq. (3), the activity coefficient of species in the organic phase is assumed as unity, since the interaction of species in the organic phase is simple as compared to the aqueous phase. In view of this, the present approach for modeling of distribution coefficient can be termed as semi-theoretical model owing to the assumption of ideal nature of organic phase.

2.2. Mass transfer model

The mixer can be considered as the single perfectly mixed equilibrium stage due to the intense agitation where the rate of mass transfer is sufficient to attain equilibrium. The multiple mixer and settler units are arranged in series to form a mixer settler cascade. The typical mixer settler cascade arrangement with counter current flow of aqueous and organic phase is shown in figure 1, where 1 refers the 1st stage where aqueous feed enters and f refers the final stage.

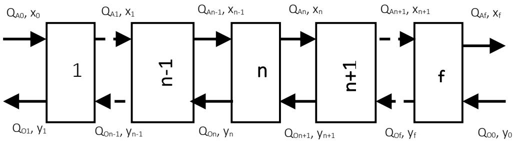

Figure 1. The schematic of mixer settler cascade.

The component mass balance for the two combined liquid phase in the nth mixer stage is

$$\frac{d\left(V_{Am}X_{m,n} + V_{Om}Y_{m,n}\right)}{dt} = Q_{A,n-1}X_{n-1} + Q_{O,n+1}Y_{n+1} - Q_{A,n}X_n - Q_{O,n}Y_n \qquad \textbf{(5)}$$

where V_{Am} and V_{Om} refers the volume of the mixer occupied by aqueous and organic phase respectively, Q_A and Q_O refers the volumetric flow rate of the aqueous and organic phase and the subscript n and n+1 refers the stage from which the stream is flowing. It is usually assumed that the mixer holdup volume (V_{Am} and V_{Om}) will be directly proportional to the respective phase flow rate to the mixer. After rearranging and substituting Y_n in terms of distribution coefficient relation $Y_n = D X_n$ as

$$\frac{dX_{m,n}}{dt} = \left(Q_{A,n-1}X_{n-1} + Q_{O,n+1}DX_{n+1} - Q_{A,n}X_n - Q_{O,n}DX_n\right)\left(\frac{1}{h_a+h_oD}\right) \qquad \textbf{(6)}$$

Now the above equation is only the function of X and distribution coefficient (D) values that were estimated from SEESPEC code. By solving the above first order ordinary differential equation yields the concentration of exit stream as the function of time. Similarly, solving the equation for n stage produces the concentration profile for the mixer settler cascade.

3. Results and Discussion

3.1. Validation of model

The salient features of the SEESPEC code are the inclusion of partial dissociation of nitric acid and speciation of Pu(IV) in aqueous nitric acid solution. The formation of various Pu(IV) species in nitric acid solution such as mono-, di-, tetra- and hexa-nitrate species are considered and the relative concentration of Pu(IV) nitrate species as well as the distribution coefficient of Pu(IV) and U(VI) in 1.1M solution of TBP in n-DD as a function of nitric acid concentration were estimated [5]. Figure 2 shows the speciation of U(VI), Pu(IV) and HNO_3 in the aqueous and organic phase as a function of nitric acid concentration at equilibrium obtained during the extraction of U(VI) and Pu(IV) (organic phase: 1.1M TBP/n-DD, aqueous phase: 0.2M U(VI), 0.2M Pu(IV) and HNO_3 (0.3 to 6 M))

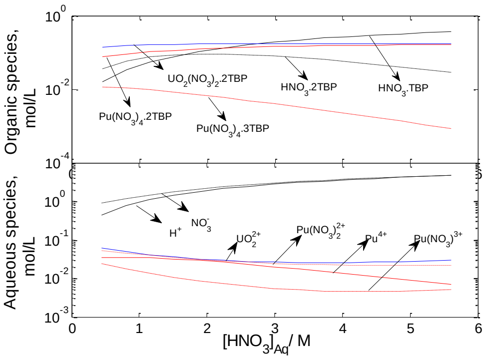

Figure 2. The modelled speciation of U(VI), Pu(IV) and HNO3 in the aqueous and organic phase as a function of nitric acid concentration at equilibrium obtained during the extraction of U(VI) and Pu(IV) (organic phase: 1.1M TBP/n-DD, aqueous phase: 0.2M U(VI), 0.2M Pu(IV) and HNO3 (0.3 to 6 M).

Figure 3 shows the variation in the distribution coefficient of Pu(IV) estimated from SEESPEC code, plotted as a function of U(VI) and Pu(IV) concentration in 0.5 M nitric acid. The SEESPEC distribution coefficients are compared with the experimental data reported in literature. It can be seen that the distribution coefficient of Pu(IV) increases with the decreasing concentration of U(VI) in aqueous phase, whereas the K_d of Pu(IV) increases with increase in the concentration of Pu(IV). It is important to note that the SEESPEC code precisely follows the experimental data and falls on the mesh in the 3D plot.

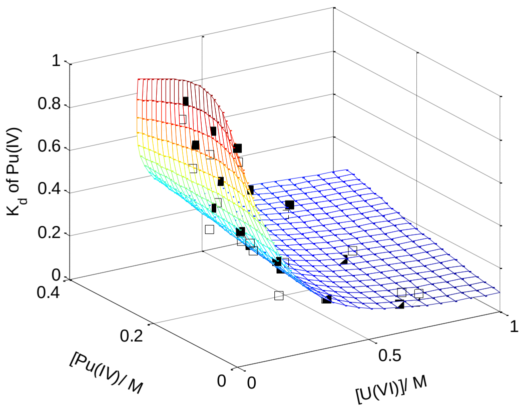

Figure 3. The distribution coefficient of Pu(IV) as a function of aqueous U(VI) and Pu(IV) concentration for a fixed HNO_3 concentration of 0.5 M (open squares - Experimental data, filled squares - Calculated by SEESPEC).

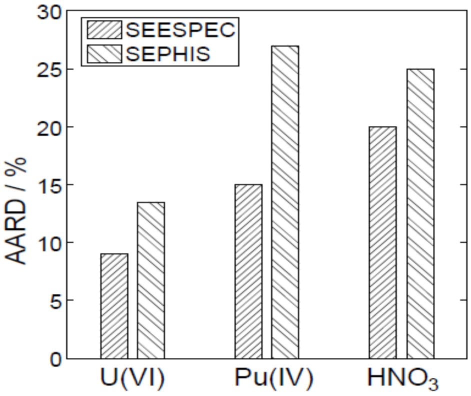

Figure 4. The The comparison of AARD between the SEESPEC and SEPHIS code for estimating the distribution coefficient of U(VI), Pu(IV) and nitric acid in 1.1 M TBP/n-DD.

Similarly, the distribution coefficient of Pu(IV) was estimated from SEESPEC code at all acidities and the variation in the percentage error of the estimated value from the experimental data at various uranium and plutonium concentration.

The average absolute relative deviation (AARD, %) between the experimental and model estimated distribution coefficient of U(VI), Pu(IV) and HNO_3 is shown in figure 4. The data were compared with those obtained using the other code, SEPHIS. The estimated distribution coefficient of Pu(IV) at lower nitric acid concentrations (< 1 M HNO_3) using the SEESPEC model shows lower deviation as compared to the SEPHIS model. It is noted that the AARD of SEESPEC is nearly half of that determined by SEPHIS especially for the estimation of distribution coefficient of Pu(IV).

3.2. Flow sheet design and optimization

The SEESPEC code was extensively used for the design, optimization and safety analysis of undesirable event during the operation of various solvent extraction flow sheet in DFRP. The DFRP plant consists of first co decontamination (HA), co stripping (HC), aqueous partitioning (2B) and U stripping cycle (2C). These solvent extraction cycles differ from the general PUREX process followed for thermal fuel reprocessing. The first co stripping (HC) and aqueous partitioning (2B) cycles are not followed in the standard PUREX process. Also the composition of FBTR fuel (70% PuC & 30% UC) and its burnup of 155 GWd/T presents unique challenges during the reprocessing [6]. Some of these challenges are taken care through modifying the flow sheet parameters by extensive simulation using SEESPEC code. The important results of the simulation studies are discussed below [7].

3.2.1. First co decontamination cycle (HA)

In the first co-decontamination (HA) cycle, the number of stages required for the extraction of uranium & plutonium and scrub stages needed to achieve the required decontamination from fission products was estimated using SEESPEC code. Six to nine stages are adequate for complete extraction of actinides. Similarly, the Zr(IV) and Ru(III) concentration in the organic product can be brought down to less than 1 mg/L with the employment of nine scrubbing stages. The scrub acid flow rate and concentration were optimized based on the number of scrubbing stages. The operating parameter such as aqueous/organic feed flow rate are also verified in the flow-sheet and optimized for minimizing the loss of fissile material in raffinate. The other constraints such as the third phase formation limit and presence of Pu(VI) concentration in the aqueous feed must also be taken care during the optimization of A/O ratio. These were estimated using SEESPEC code

3.2.2. First co stripping cycle (HC)

The first co stripping cycle is the unique flow sheet adopted for the combined striping of both U & Pu in a single cycle. This is carried out by adopting dual stripping method where Pu(IV) and U(VI) are stripped at 0.5M and 0.01M HNO_3 concentration respectively [8]. This is to avoid Pu(IV) polymerization at dilute HNO_3 concentration below 0.3M. The first co stripping cycle was successfully operated at CORAL reprocessing facility in IGCAR. Due to the interference of solvent degradation product such as HDBP, the complete stripping of Pu(IV) and U(VI) from the organic phase is difficult to achieve. This results in the requirement of separate recovery cycles for Pu(IV) in order to meet the lean organic phase for waste disposal limit. In DFRP the problem of Pu retention in the lean organic phase was addressed by providing increasing the number of stages for Pu(IV) stripping from 8 to 11 and also modifying the process parameters with the help of simulation studies using SEESPEC code. The SEESPEC modified process parameters have been tested and implemented in the stripping bank operating in the plant.

The steady state and dynamic simulation were carried out using SEESPEC code to understand the stripping behavior of Pu in the presence of HDBP [9]. The addition of U(IV) helps in the stripping of Pu coordinated with HDBP. The optimum U(IV) concentration and location of U(IV) feed in the bank was finalized based on the simulation studies. The minimum U(IV) concentration required for the near complete stripping of Pu(IV) as a function of HDBP concentration was determined. Another important finding from the simulation is the initial transient period during the startup. At the startup of the HC contactor, the Pu(IV) concentration in the lean organic phase is going through a maximum and slowly comes down with time. This initial Pu peak in the lean organic is responsible for the majority of the Pu found in the lean organic phase [10].

3.2.3. Aqueous partitioning cycle (2B)

The partitioning of uranium and plutonium in fast reactor fuel reprocessing is based on aqueous partitioning procedure and it is quite different from the thermal fuel reprocessing, wherein Pu(III) is selectively stripped from the loaded organic phase contains both U(VI) and Pu(IV). However, in DFRP, the co-stripped U(VI) and Pu(IV) is sent to the conditioner where Pu(IV) is converted to Pu(III) using U(IV). Then the U(VI) present in the aqueous solution is separated from plutonium by solvent extraction using 30% TBP/n-DD. In order to meet the product purity, the optimized flow sheet parameters such as A/O ratio, scrub flow rate and concentration were arrived after the detailed simulation of various cases using SEESPEC code. The U(VI) present in the loaded organic is stripped with the use of 0.01M HNO_3 solution. The simulation was carried out to estimate the optimum A/O ratio as a function of U(VI) concentration in the loaded organic for the near complete stripping of U(VI).

3.3. Future directions

To further extend the application of SEESPEC code to the reprocessing plant, an attempt is being made to develop the plant level simulation software for catering to the process monitoring of the plant, operator training simulator for training the staff and to understand the flow-sheet results. In addition, the protocols are under development by linking the plant instrumentation systems to the software for near real-time material monitoring and dynamic optimization and control of the plant in real time. Since the computational cost of running the software on SEESPEC is time intensive, the real time optimization and control of the plant require advanced artificial intelligence protocols such as machine learning, data science and data analytics. The machine learning protocols embark on the models and algorithms trained from the data without running on the programmed SEESPEC software for every task. Therefore, the machine learning algorithm is being trained from the SEESPEC code and therefore, the protocol can predict the process performance without running into the SEESPEC algorithm each time. As a result, the computational time required for optimization and control of the plant can be dramatically reduced to near real time.

The other forms of AI namely the data science, and data analytics, derive knowledge pattern and predictive power from the massive data generated in reprocessing plant operation. The historical data accumulated over the period of time serves as valuable asset for predicting the process performance of the plant, especially the data generated during abnormal process behaviours or circumstances that are not considered in the design phase. The collected data base from the plant combined with effective analytics and superior machine learning algorithms shall be utilized to distil the required plant knowledge patterns from the data ocean. Such knowledge can be utilized to identify the current limitations and further improvement of the plant performance.

4. Conclusion

The development of indigenous code SEESPEC for the solvent extraction flow sheet simulation was highlighted. The thermodynamic model was developed for the prediction of distribution coefficient of major solutes such as U(VI), Pu(IV) and HNO_3. The activity coefficient of various species in the aqueous phase was estimated by eUNIQUAC activity coefficient model. The effect of HDBP on the distribution coefficient of major solutes were modeled by considering the various organic species forms during the extraction. The equilibrium constant of the organic phase species was estimated from the experimental distribution coefficient data obtained for various metal ion and nitric acid concentration. The accuracy of prediction for the distribution coefficient of U(VI), Pu(IV) and HNO_3 were improved over the literature reported SEPHIS model. The SEESPEC code was extensively applied for the flow sheet design and parameter optimization in DFRP to achieve the desired product purity and minimizing the waste generation.

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Design and Development of Head End Equipment for Fast Reactor Fuel Reprocessing

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Abstract

Reprocessing plant comprises of head end operations starting from receipt of fuel Sub-Assembly (SA) to chopping followed by process operations starting from dissolution to extraction and precipitation. Head-end operations are essential as they involve the precise mechanical operations include dismantling, separation of foot piece and fuel pins, chopping etc. inside the shielded cells. SA's are received from the reactor site through Single Subassembly Transfer Cask (SSTC) and Three Subassembly Transfer Cask (TSTC). The SA's will be brought inside the cell using a SA transfer system. SA Horizontal Grappler Mechanism (SAHGM) and Vertical SA Grappler Mechanism (VSAGM) are augmented in receipt & storage cell to handle the SA's horizontally and vertically. The SA tilting mechanism is meant for tilting of the SA from horizontal to vertical alignment and vice versa. Similarly, the ICTS facilitates the transfer of SA's between the receipt cell and the dismantling cell using a Double Door Transfer Port (DDTP). Dismantling machine is used to remove the hexagonal wrapper. Foot Piece Removal System (FPRS) is engineered to effectively separate the foot piece and its support rails from the fuel pin bundle. Due to the slender nature of fuel pins, a fuel pin loading mechanism (FPLM) is designed to segregate and load the fuel pins in to a can/magazine. The pins in the magazine are then fed into a chopper, where they are chopped into small pieces and allowed to fall into the dissolver for dissolution. These sequential stages ensure a seamless transition from the mechanical disassembly of the fuel to the chemical extraction of the fissile material.

Keywords: *Head-end operations, SA Transfer system, Grappler, Tilting mechanism, Inter cell Transfer, Dismantling, Foot piece, Fuel pin retrieval, Chopper, Dissolver.*

1. Introduction

Several unique equipments have been developed for the processing of the mixed carbide spent fuel from FBTR and have been successfully demonstrated in COmpact Reprocessing of Advanced fuel in Lead cells (CORAL), which is continuing to process the spent fuel from FBTR [1]. The Indian fast reactor program is stepping into the commercial domain with the construction of Prototype Fast Breeder Reactor (PFBR) at Kalpakkam, which is undergoing commissioning trials. The reprocessing of PFBR spent fuel will be demonstrated in DFRP and for this purpose special head end process equipments such as fuel transfer system capable of handling PFBR sub assembly, Laser based sub assembly dismantling system for precise cutting and dismantling of the sub-assembly for retrieval of fuel pins, stainless steel special incell crane with all motors outside the hot cell meeting the site requirements, multi pin chopper for increased throughput and Titanium fuel dissolver for PFBR fuel have been developed. This equipment's have been designed, developed and manufactured indigenously and have been rigorously tested and their design was validated. For some of these equipment's full scale mockup has been manufactured, tested and their feedback has been incorporated in the design. Hot validation of these equipment's will be carried out in DFRP and the experience will be utilized to fine tune and optimize the design for the future reprocessing plants namely the Fuel Reprocessing Plant of Fast Reactor Fuel Cycle Facility.

2. Overview of Head end equipment

Head End Facility (HEF) houses the equipment for head end operations such as subassembly (SA) handling, sodium cleaning, transfer, dismantling, fuel pin handling, transfer, chopping and dissolution. Typically, HEF consists of four cells, (i) Receipt & storage cell, (ii) Dismantling cell, (iii) Buffer cell and (iv) Chopper & Dissolver cell. Subassembly receipt, sodium cleaning and storage is done in receipt & storage cell. SA dismantling and fuel pin bundle separation is done in dismantling cell. Fuel pin retrieval, handling & transfer and solid waste transfer is done in buffer cell. Fuel pin transfer, chopping and dissolution is done from chopper & dissolver cell. All these cells are equipped with in-cell cranes and master slave manipulator for remote operations and maintenance. For these operations, several first-of-its-kind (FOIK) systems [2] are required due to the specific nature of fast reactor spent fuel.

Accordingly, the following sections are organized to highlight the equipment used for fuel SA receipt, handling, transfer and dismantling, retrieval of fuel pin, loading into magazine, and the chopping and dissolution processes.

3. Receipt, handling and transfer of fuel SA

3.1. PFBR spent fuel transfer system

In PFBR, the irradiated spent fuel subassemblies are stored and cooled within the reactor vessel for a period of 480d. Further cooling is done at spent subassembly storage bay using de-mineralized water. The spent subassembly is transported to DFRP for the initial demonstration of reprocessing using single subassembly transport cask (SSTC) and to FRFCF using the three subassembly transport cask (TSTC). Schematic of cask with support structure is shown in Figure 1.

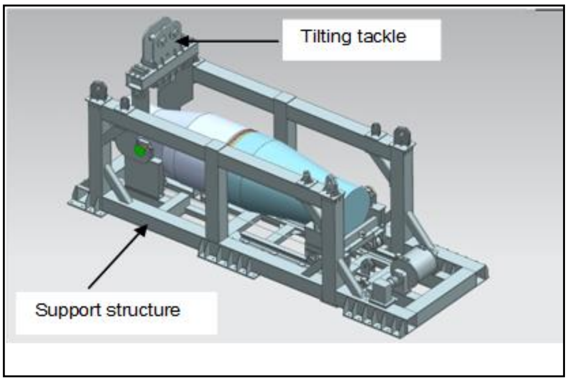

Figure 1. Schematic of SSTC with support.

AERB SC TR – 1 stipulates the regulations for the safe transport of radioactive material in public domain under normal and accidental conditions. Movement of cask is within the site of Kalpakkam during non-busy hours with a restricted speed limit of maximum 10 km/h.

3.2. Sub Assembly Transfer System (SATS)

In the head end system of Fuel Reprocessing Plant (FRP), PFBR Sub-Assembly (SA) is received horizontally in a Single SA Transport Cask at HEF/ DFRP and Three SA Transport Cask (TSTC) at FRP/FRFCF. After engaging the SSTC or TSTC with the CSU, the SA Transfer System (SATS) is engaged with the SA basket plug. Four pair of wheels are provided on the basket to move horizontally from SSTC/TSTC to receipt cell and vice versa. The basket is pushed with the help of SATS which in turn translates it to inside the Hot Cell. After the SA is received, SATS is used to retrieve the basket back into cask. Figure 2 shows the schematic of SATS at fuel handling area of FRP.

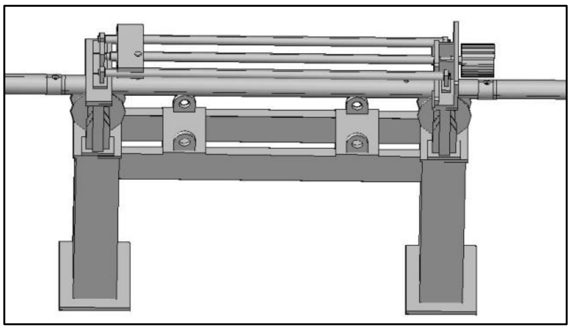

Figure 2. schematic of SATS at fuel handling area of FRP

3.3. Incell cranes

Incell crane for hot cell applications has been developed for handling different types of loads remotely. The Incell crane plays a significant role in fuel handling, hull transferring, remote operation and maintenance of hot cell equipment's, etc. inside the hot cells, where radioactivity and high concentration nitric acid environment prevails.

Figure 3. Schematic of mock-up incell crane.

The design of incell crane is first of its kind. Since the in-cell crane has to be erected inside an α-tight hot cell, it demands compact, robust design and shall be amenable for remote operation and maintenance. Innovative concepts with special mechanisms are deployed in order to bring all the wear & tear components requiring periodic maintenance such as drive motors and limit switch systems etc. outside the hot cell and kept in a drive box for contact maintenance. The drive shafts are passing through pipe in pipe construction connecting the hot cell lining and drive box to ensure α tight containment. Schematic of mock-up incell crane is shown in Figure 3.

3.4. Vertical Sub Assembly Grappler Mechanism (VSAGM)

The reprocessing plant is designed to reprocess spent fuel from PFBR. In the head end system of Fuel Reprocessing Plant (FRP), PFBR Sub-Assembly (SA) is received and stored temporarily before further operations. For safety and stability, the interim dry storage is provided with array of vertical cavities on which SAs will be stored. The vertical handling of SAs will be carried out by incell crane with the help of a Vertical SA Grappler Mechanism (VSAGM). Therefore, VSAGM should be designed in compliance with the SA head for locking and unlocking provisions considering safe and redundant features during lifting/handling of PFBR SA. Figure 4 shows the conceptual drawing of VSAGM.

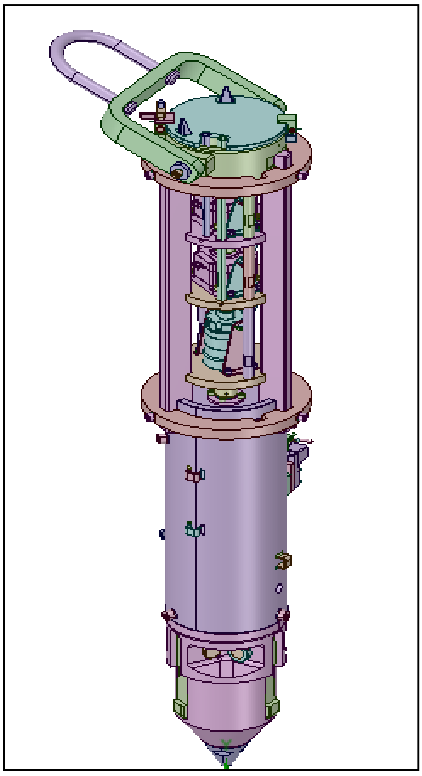

Figure 4. Conceptual drawing of VSAGM.

3.5. Sub-Assembly Horizontal Grappler Mechanism (SAHGM)

The SA Horizontal Grappler Mechanism (SAHGM) facilitates the horizontal transfer of spent fuel subassemblies (FSA) and Blanket Sub-Assembly (BSA) within the head end area of the Fuel Reprocessing Plant (FRP) at the Fast Reactor Fuel Cycle Facility (FRFCF). For the design the total weights of the FSA and BSA are considered. The mechanism has a positive locking system and it engages passively. The operation is performed with the aid of MSM and incell crane. Key components of the SAHGM include two grappler connected by tie beam with positive locking mechanism. The prototype of SAHGM is shown in Figure 5. The operation consists of lowering SA gripping mechanism over SA with help of crane, engaging of counter weight based positive locking mechanism and lifting of SA gripping mechanism with SA. The counter weight based positive locking mechanism engage passively without external assistance. The disengagement of locking mechanism shall be achieved with help of MSM, when SA needs to be placed over the flat surface.

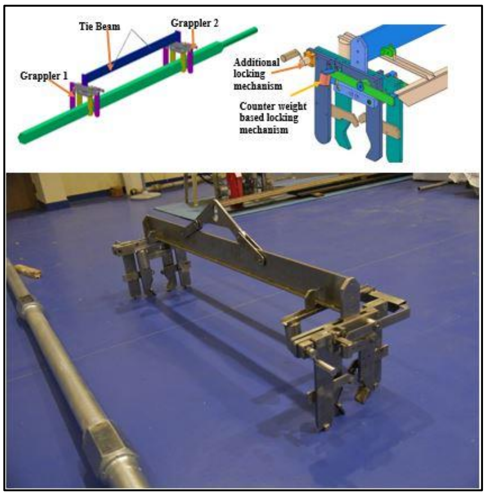

Figure 5. Schematic of mock-up SAHGM.

3.6. Sodium cleaning system

Sodium cleaning system is required for the removal of residual sodium sticking to the fuel sub-assemblies (FSA) received at Head End Facility (HEF) from FBTR. Normally a small quantity of sodium will be sticking to the surface of the FSA when it is taken out of reactor for storage. As this sodium and its reaction products with air are likely to interfere with chopping process, it has to be removed. Hence sodium cleaning has to be in HEF by steam/nitrogen process. Schematic of FBTR sodium cleaning system is shown in Figure 6.

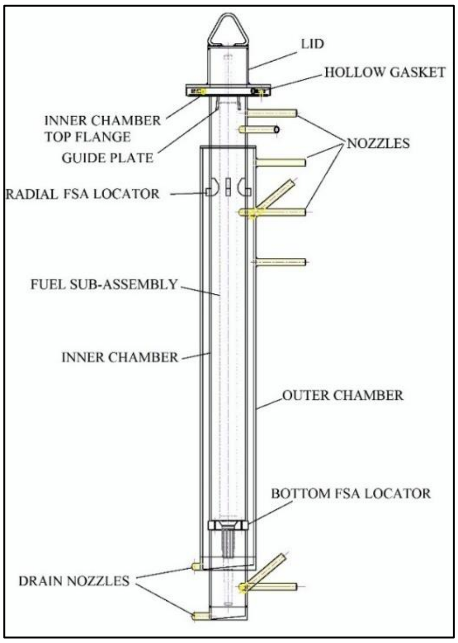

Figure 6. FBTR sodium cleaning system.

4. Dismantling and retrieval of fuel pins

4.1. Laser Dismantling Machine

Dimensional Measurements-cum-Laser Dismantling Machine (DMLDM) has been designed and developed for Head End Facility (HEF) of DFRP, for remote dimensional measurement of PFBR fuel subassemblies (FSAs) for the assessment of dimensional changes due to irradiation and also dismantling the FSAs for the retrieval of fuel pins for reprocessing at DFRP. This is a 5+1 axes machine viz. two X-axes, Y-axis, Z-axis, ϴ1-axis (FSA holder rotation) and ϴ2 axis (motorized cutting wheel). The design of the machine was carried out in compliance with ASTM C1533-02 "Standard Guide for General Design Considerations for Hot Cell Equipment components". Figure 7 shown the schematic of dismantling machine.

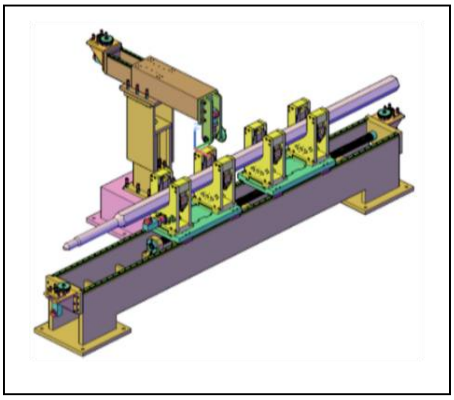

Figure 7. Schematic of dismantling machine.

4.2. Foot piece removal system (FPRS)

Foot piece removal system is designed to separate the foot piece from the remaining part of the fuel pin bundle. The foot piece along with the support rails are separated from the guide rails and fuel pin bundle. The various sub-systems in the Foot Piece Removal System are

- Finger gripper assembly
- Foot piece fixture and drive unit
- FSA fuel pin bundle handling station.

The mock-up foot piece removal system is shown Figure 8.

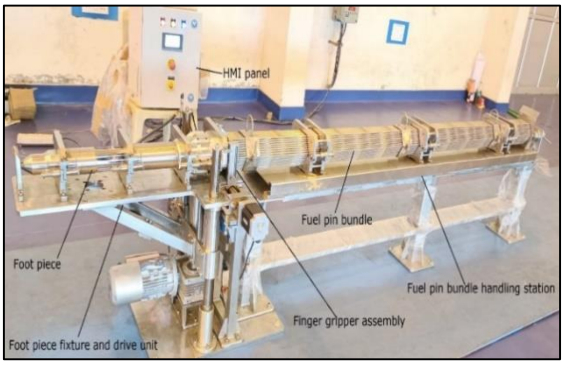

Figure 8. Schematic of mock-up Foot Piece Removal System.

4.3. Fuel pin retrieval system

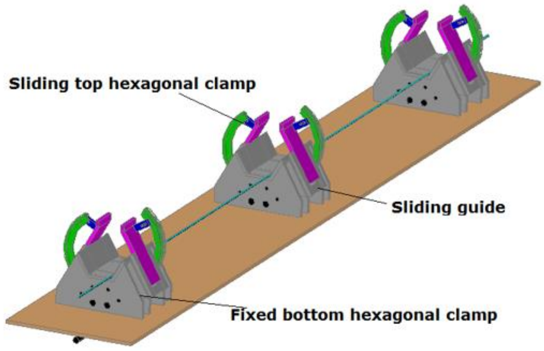

Figure 9. Schematic of SHFPBHS.

Once the foot piece is removed, the fuel pin bundle is held in the hexagon matrix with the help of guide rails and hexagonal grippers. Fuel pin retrieval system is designed to retrieve fuel pins from the fuel pin bundle and load them into the fuel pin tray (Figure 9).

It consists of

- Sliding hexagonal fuel pin bundle handling station (SHFPBHS)
- Fuel pin retrieval mechanism
- Variable pick and place mechanism
- Fuel pin tray assembly

5. Handling and loading of fuel pins to magazine and chopping

5.1. Inter Cell Transfer System (ICTS)

The ICTS enables horizontal movement of spent FSA and BSA within the head end area of FRP/FRCFC, through a double door mechanism to prevent radioactive cross-contamination of cells. Key components include the transfer carriage, Engagement Disengagement mechanism for Y motion, and pushing mechanism for X motion, all equipped with necessary instrumentation. Design priorities include indigenous, compact, lightweight, modular construction with high fit and tolerance for interchangeability. Material selection focuses on strength and corrosion resistance, with provisions for remote maintenance. Schematic of mock-up ICTS is shown in Figure 10.

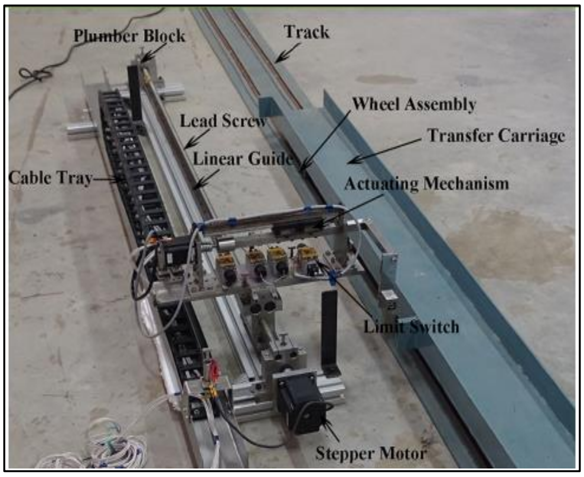

Figure 10. Photograph of mock-up ICTS.

5.2. Double door transport port (DDTP)

The Double door transfer port (DDTP) system is a square shaped tunnel equipped with door on either end. All the radioactive items or tools/tackles will be transferred in between the hot cells, through a DDTP system that prevents the radioactive cross-contamination and maintain differential pressure.

Figure 11. Photograph of mock-up double door transfer port.

DDTP system shall have remotely operated leak tight doors at both the ends to isolate the cell atmosphere. Based on cell layout and feasibility study of various options for the remotely operated leak tight door mechanism are simulated and designed. DDTP system is designed as per ASTM C-1533, Standard Guide for General Design Considerations for Hot Cell Equipment. Schematic of mock-up DDTP system is shown in the Figure 11. The Double Door Transfer Port (DDTP) system is consists of the following sub-systems.

- Transfer port and support
- DDTP door-1 Motor-gear driven system
- DDTP door-2 Pneumatic driven system

At one end of the port, remotely operated motor-gear driven leak tight door systems and at other end, remotely operated pneumatic driven leak tight door systems are provided.

5.3. Spacer Wire Removal System (SWRS)

A single PFBR FSA consists of 217 fuel pins whereas 61 fuel pins in FBTR FSA. The fuel pins are helically wound with a spacer wire. Spacer wire poses various challenges during reprocessing operations, especially in chopping, hull agitation and hull transfer system. So it was planned to remove the spacer wire prior to chopping operation.

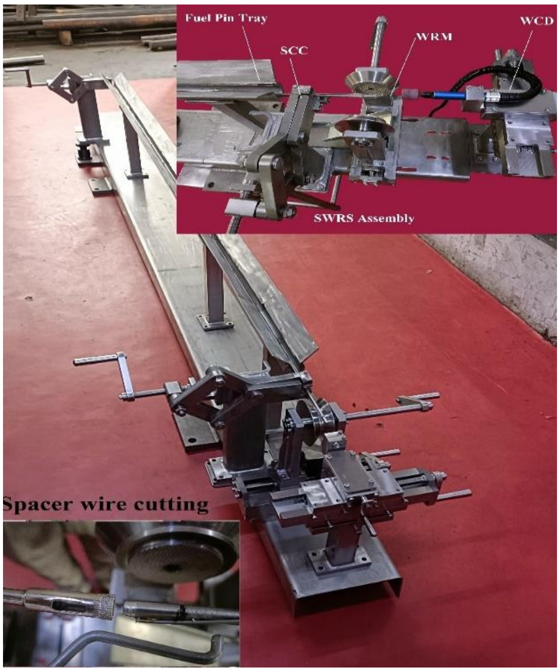

Figure 12. Photograph of mock-up SWRS.

Towards this, a prototype Spacer Wire Removal System (SWRS) is designed as a proof of concept consisting of

- Fuel Pin Tray
- Self-centering clamp (SCC)
- Wire cutting device (WCD)
- Wire Retrieval Mechanism (WRM)

The schematic of mock-up spacer wire removal system is shown in Figure 12.

5.4. Fuel Pin Loading Mechanism (FPLM)

The fuel pins will be retrieved from the fuel pin bundle and loaded into the Fuel Pin Can (FPC) / magazine for progressive chopping of fuel pins by a multipin chopper. Remote handling of fuel pins is very challenging due to highly slender in nature of the fuel pins. Towards ease in handling/operation, a Fuel Pin Loading Mechanism (FPLM) is designed, developed and manufactured to load the 10 number of fuel pins in to the FPC / magazine. All the individual components of the FPLM are designed with modular construction, amenable for remote operation and maintenance by using master slave manipulator and in-cell crane. FPLM consists of:

- Fuel pin tray assembly
- Fuel pin feeding drive roller assembly

The schematic of mock-up FPLM is shown in Figure 13.

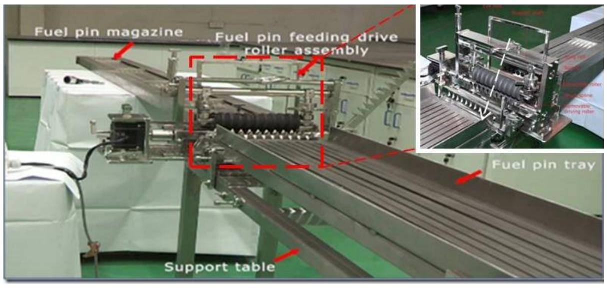

Figure 13. Photograph of mock-up FPLM.

5.5. Chopper

In Fast Reactor Fuel cycle facility (FRFCF), chop leach process is adopted. In this process the fuel alone is dissolved leaving the clad as hull. Chopping is done by exposing the surface area of the fuel to the dissolvent. The important mechanical components of the chopper are magazine, Stepper motor with trolley and pinion assembly, Gripper block, Stationary block, Multiple Pin Cutter, Pneumatic cylinder and Linear Actuator. After dismantling of FSA & RSA, these pins are loaded in a magazine. The magazine consists of array of ten tubes having slot at the bottom.

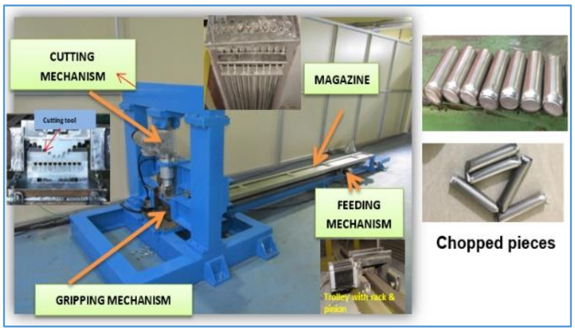

Figure 14. Photograph of mock-up multi-pin chopper.

Movement of the fuel pins takes place with the help of rack and pinion arrangement, where the stepper motor assembly drives the pinion. These fuel pin grips by gripper block driven by double acting pneumatic cylinder. The stationary block, gripper block has to be designed such that the unwinding of spacer wire can be avoided. The capacity of pneumatic cylinder ensures that the pin does not get punctured during gripping and it generates enough force that pin doesn’t pulled out while cutting. Cutting mechanism consists of specially designed multipoint cutting tool driven by electrical actuator. The schematic of mock-up multi-pin chopper is shown in Figure 14.

6. Dissolution and hull transfer system

6.1. Dissolver

Dissolution of chopped irradiated fuel discharged from the fast reactor in nitric acid is the first chemical step in nuclear fuel reprocessing. A novel thermo-syphon type dissolver [3] is indigenously designed and developed to dissolve the irradiated chopped fuel pins. Presence of high concentration of nitric acid at boiling point in the dissolver, a high corrosive oxidative atmosphere prevails, which necessitates selecting refractory material. Considering the factors like corrosion resistance, easy availability, fabrication expertise and adequate experience in dissimilar joint fabrication (indigenously developed) etc., titanium was selected as the material of construction. The thermo-syphon type dissolver consists of vertically placed two limbs viz the fuel charging limb (hot limb) and the cold limb. These two limbs are interconnected by three limbs namely bottom, intermediate and top limb. As the reprocessing plant capacity increases (Pilot plant CORAL to industrial scale plant FRFCF), requirement of size of dissolver also increases with geometric intricacy, requires special attention in design, fabrication, inspection and remote operation and maintenance aspects.

6.2. Hull drum

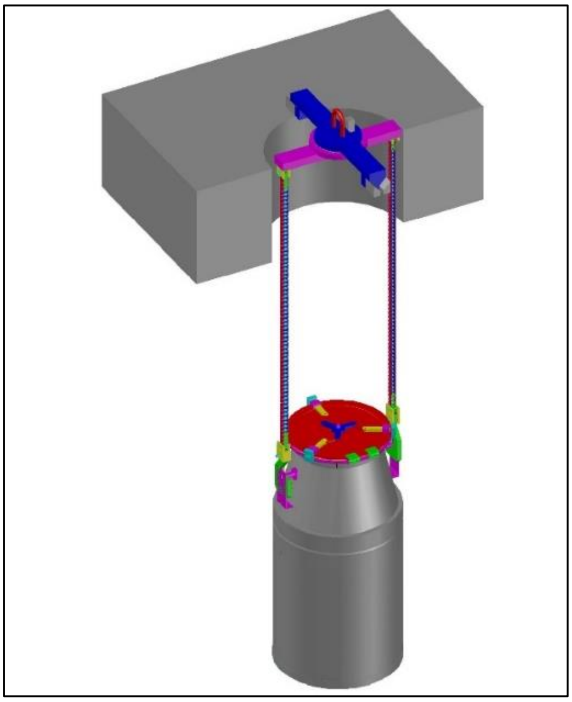

Figure 15. Schematic of hull drum.

The design of alpha-tight hull drum is based on the volume of hull waste generated from reprocessing of FSA and RSA. It is designed to collect FSA waste and RSA hull waste. In view of the above requirement, the required volume of hull drum is calculated, including free board, along with its lifting mechanism as shown in Figure 15.

The solid waste drums (hull & hardware) are individually inserted into a vertical shipping cask and transferred to WMP of FRFCF for further treatment. Due to very high beta and gamma activity, shielding provision is required throughout the transportation from FRP to WMP. In view of that, a vertical cask is designed to provide as per AERB guide lines.

6.3. Hull tilting system

A Hull tilting system (HUTS) is designed for the FRP of FRFCF to handle hulls generated from chopped BSA pins and FSA pins after dissolution. Hulls are transferred from the dissolver basket to the buffer cell via a chute. The dissolver basket, carrying hulls, is lifted, positioned, and lowered into the HUTS cage for tilting. A gearbox with a motor drives the tilting assembly. HUTS comprises key mechanical components including the cage, locking plate, hinge support, electrical motor with gearbox, and associated instrumentation. Schematic of mock-up hull tilting system is shown in Figure 16.

Figure 16. Schematic of mock-up hull tilting system.

7. Conclusion and future Outlook

The major head-end operations include the receipt, handling, and transfer of fuel subassemblies (SAs), followed by the dismantling and retrieval of individual fuel pins. Once retrieved, the next step involves the precise handling and loading of these pins into magazines for chopping. Finally, the chopped pins undergo dissolution. For the reprocessing of fast reactor fuel, all the required First-of-Its-Kind (FOIK) equipment is indigenously developed and continuously evolved through conceptual design and prototype fabrication to establish functionality and reliability, with the long-term goal of achieving automation and reducing human exposure. The actual systems will be developed incorporating feedback obtained from the prototype and deployed in the plant. These systems are modular in construction and accessible for remote maintenance. Based on the prototype testing, some of these equipment's are implemented in the plant while others are in testing phase.

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Unique and Special Equipment for Reprocessing Fast Reactor Fuel

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Abstract

The fast reactor (FR) fuel reprocessing involves handling of high burn-up and short cooled fuels. The radioactivity level of FR spent fuel is about 10 to 15 times higher than the thermal reactor fuel and the specific activity handled in the order of 1000 Ci/l as compared to 200 Ci/l. The Pu/(U+Pu) ratio in FR reprocessing in the range of 0.15 to 0.7 as compared to 0.002 to 0.004 in thermal nuclear reactor reprocessing. Because of high specific activity and high concentration of Pu, FR reprocessing plant requires specialized equipment over thermal nuclear reprocessing plant. The few unique equipment developed for fast reactor fuel reprocessing application are:

1. Annular Centrifugal Extractor for Solvent Extraction Unit Operation
2. Cross current air sparged mixer settler for primary solvent purification
3. Stirred tank continuous precipitator and thickener for Plutonium Reconversion
4. Helical coil based pulsating pump for Fluid transport and metering flow

The development of the equipments are described in this report.

Keywords: *Annular Centrifugal Extractor (ACE), Cross current air sparged mixer settler, Continuous precipitator, Pulsating pump, Equipment for nuclear reprocessing*

1. Annular Centrifugal Extractor

The solvent degradation is one of the major problems in spent fuel reprocessing. Conventional equipment such as mixer settler, pulse column and other extractors has residence time in the order of one to ten minutes per stage, and it leads to solvent degradation and necessitates purification of the solvent after every pass in the process. To address the above problem the short residence contactor was developed. For fast breeder nuclear reprocessing plant necessitated the development of fast contactors.

In 1970's, Savannah River Laboratory (SRL) developed paddle type centrifugal extractor (CE) with air controlled aqueous weir to control the interface position inside the separating zone, which facilitates the handling of a wide range of aqueous and organic densities at different organic and aqueous phase flow ratios (O/A ratios). Later, Argonne National Laboratory (ANL) modified the paddle type into an annular type centrifugal extractor (ACE). ACE have potential for wide/broad usage in different process industries especially in biochemical, pharmaceutical, chemical, hydrometallurgical, waste treatment and nuclear industries. ACE has several advantages such as compact design, less holdup, high throughput, high mass transfer performance, low residence time, small footprint, low height, etc. Owing to these advantages it is also being used in nuclear fuel reprocessing especially in fast breeder reactor fuel reprocessing to overcome or address the solvent degradation. Different sizes of annular centrifugal extractors were developed in Indira Gandhi Centre for Atomic Research (IGCAR). The 30 mm ACE was developed and deployed in the CORAL facility and successfully operation for more than two decades. Similarly, 40 mm ACE was developed and hot commissioned in the DFRP plant.

1.1 Working principle of ACE

The schematic diagram of ACE is shown in Fig. 1. During normal operation heavy phase (blue) and light phase (yellow) enters through ACE via aqueous pipe (1B) and organic pipe (1A) in-between stationary bowl (2) and rotating bowl (3). Both phases mixed vigorously in annular region by shear force/ skin friction due to high-speed rotation of inner bowl and turbulent liquid-liquid dispersion promotes the mass transfer between two phases. Mixed phase (dispersion) flows down by gravity in annular region and enters inside rotating bowl through orifice (5) via bottom baffle (4) (mixing vane). Main function of bottom baffle is to break the vortex formation and provided in the bottom region which are either attached to the base of the outer cylinder or to the bottom of the rotating cylinder in the case of paddle type contactors. Dispersion entering inside the rotating bowl is deflected towards the wall of the rotating cylinder by deflecting baffle (6) to enhance the centrifugal separation of heavy and lighter phases. Dispersion phase gets separated and also flows from bottom to top inside the rotating bowl and it is confined inside the chambers (four to eight numbers) by vertical baffles (7). The rotating cylinder imparts a rigid body rotation to the liquid and creates vertically cylindrical free surface of liquid and it is coaxial with the axis of rotation because of high 'g'. The central portion is occupied by air. The dispersion entering at the bottom gets separated as it moves upwards. The separation rate of dispersed phase depends upon density difference between continuous and dispersed phase, viscosity of continuous phase, drop size distribution, settling velocity of dispersed phase under centrifugal acceleration ($r\omega^2$) and coalescing behaviour of the two phases. For complete separation (which is considered to be a flagship advantage of ACEs) of dispersion, adequate height of rotating bowl needs to be provided for a given level of centrifugal acceleration. Separated lighter phase (yellow) flow over light phase weir (8) and it is thrown out in to lighter phase collection chamber (11) located at stationary bowl. Similarly, heavy phase (blue) flows over heave phase weir (10) via under flow (9) and it is thrown out in to heavy phase collection chamber (12) located above lighter phase collection chamber at stationary bowl. Finally, both heavy phase and light phase comes out or flows in to adjacent stages through light phase outlet (13A) and heavy phase outlet (13B).

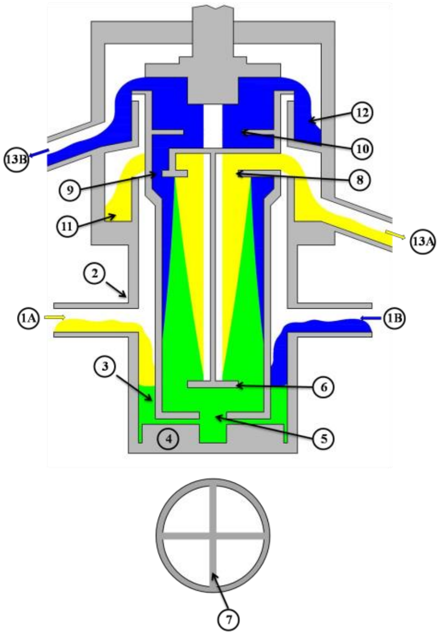


Figure. 1. Schematic diagram of the annular centrifugal extractor (ACE).

(1A) Light phase inlet, (1B) heavy phase inlet, (2) Stationary cylinder, (3) rotating cylinder, (4) radial baffles on the stationary bottom plate, (5) central opening for rotating cylinder, (6) deflecting baffle in the rotor, (7) vertical baffles in the rotor, (8) light phase weir, (9) under flow region for heavy phase, (10) heavy phase weir, (11) light phase collection chamber, (12) heavy phase collection chamber and (13A and B) outlets for light and heavy phases, respectively.

The photographic view of 20 stages extraction bank installed in DFRP plant is shown in Fig. 2. The employment of the ACE in the nuclear fuel reprocessing is a challenging task due to remote maintenance requirement. It also requires specialized tools for remote maintenance. Frequent motor/ bearing failures due to overheating, corrosive, and radioactive environments during the multistage operation. The subsequent repair needs a longer period to bring back the multistage ACE system back to the operating condition. During the maintenance, the stagnant organic phase present inside the multistage ACE gets exposed to a high radiation dose and leads to the formation of degraded products. Further, the degraded solvent drastically affects the stripping performance, reduces the decontamination factor, and increases the plutonium and uranium losses. After detailed literature analysis and considering the flow conditions, a stationary bowl with an interstage inclined overflow line was developed to ensure smooth operation during motor or bearing failure. The performance of the abovementioned inclined system has been investigated.

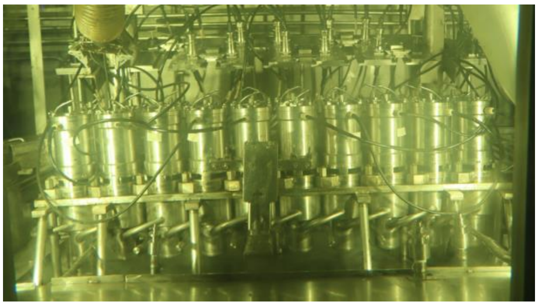


Figure 2. Photographic view of 20 Stages ACE extraction bank installed in DFRP plant.

1.2. Inter stage inclined overflow line

A four-stage CE 30 mm centrifugal extractor bank having stationary bowls with interstage inclined overflow lines (6) was designed and developed. Its photograph and schematic drawing are shown in Fig. 3 and 4, respectively.

The top end of the inclined overflow line is connected below the organic collection chamber in the annular region at the adjacent stationary bowl, and the bottom end is connected below the aqueous and organic inlets of its annular region. It ensures that the aqueous phase flows from bottom to top from one inclined overflow line and the organic phase flows from top to bottom from another inclined overflow line connected in between the failed stage and adjacent stages. The orientation of the inclined overflow line is made such that the aqueous and organic phases follow their paths in the multistage ACE system.

Figure. 3. Four stages 30 mm ACE stationary bowl with inter-stage inclined overflow line. (1) Aqueous feed pipe, (2) Organic feed pipe, (3) Aqueous outlet, (4) Organic outlet, (5) Aqueous inter-stage line and (6) Inclined over flow line.

1.3. Working principle of inclined overflow line

The aqueous phase enters into the first stage through the aqueous inlet line (1) (Fig. 4) and is mixed with the organic phase (coming from the second stage) in the first-stage annular region. The two-phase mixture generated in the annular zone is separated inside the rotating bowl by centrifugal acceleration. The aqueous phase comes out from the first stage and flows into the second stage through the aqueous interstage line (5). Similarly, the organic phase enters into the fourth stage through the organic inlet line (2) and is mixed with the aqueous phase coming from third stage into the fourth stage annular zone through the aqueous interstage line (5). The two-phase mixture gets separated inside the fourth stage rotating bowl by centrifugal action. The separated organic phase flows into the third stage through an organic interstage line (7). Both the aqueous and organic streams flow in a counter current mode from the first to the fourth stage and from the fourth to the first stage, respectively. Finally, the aqueous phase comes out through the aqueous outlet (3) from the fourth stage, and the organic phase comes out from the cascade through the organic outlet (4) from the first stage.

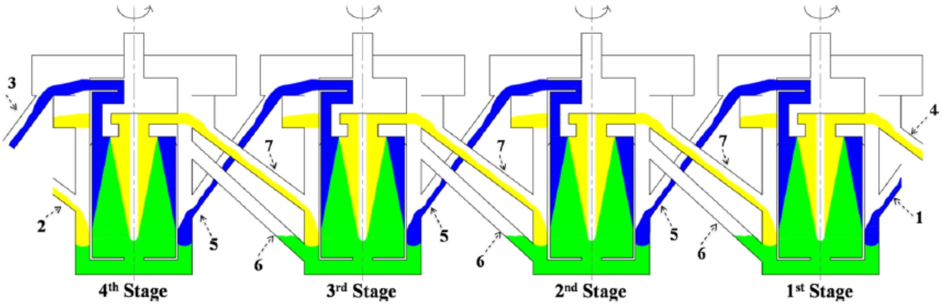


Figure 4. Schematic drawing of the four-stage 30 mm centrifugal extractor stationary bowl with an interstage inclined overflow line (blue color, aqueous phase; green color, mixture; yellow color, organic phase).

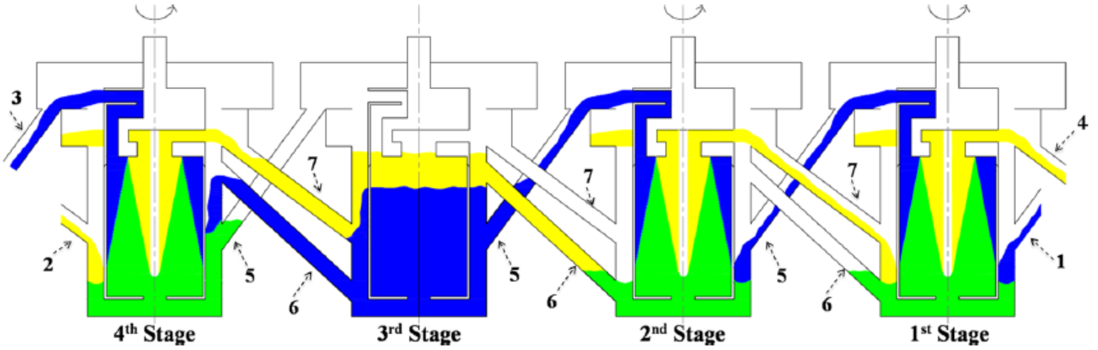


Figure 5. Schematic view of aqueous and organic flows in stationary bank in presence of overflow line with third stage motor failed condition (blue color, aqueous phase; green color, mixture, yellow color, organic phase).

Failure or malfunction of any stage in the multistage unit leads to accumulation of the aqueous and organic phases inside the failed stage and disturbs the counter current mode operation. To handle this adverse condition, an inclined interstage overflow line (6) is incorporated in the multistage ACE system. In the motor failed stage (third stage in Fig. 5), the aqueous phase settles at the bottom, and the organic phase accumulates at the top of the mixing chamber. Aqueous stream flows from the bottom of the motor failed stage (number 3) to the top of the left running stage (number 4) via the inclined overflow line (6) due to the hydraulic head. The organic stream flows from the top to the bottom of the right running stage (number 2) via another inclined overflow line (6) by gravity. The incorporated inclined overflow line (6) ensures smooth counter current mode operation without any interference. All the other active three stages (1, 2, and 4) continue to operate even though stage 3 failed. The inter-stage inclined overflow line is developed and demonstrated the smooth operation of multistage ACE in the event of any single stage motor or bearing failure.

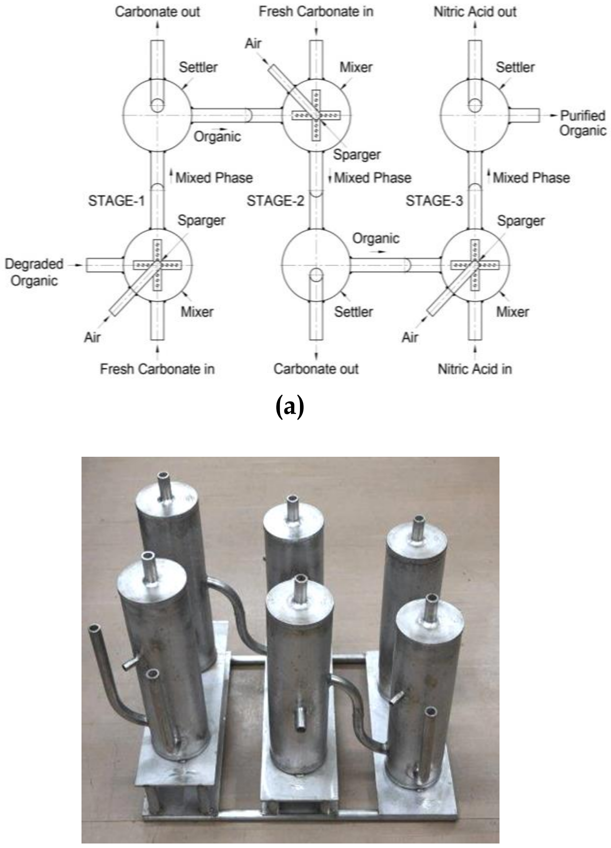
2. Cross current air sparged mixer settler

Solvent degradation is more severe during reprocessing of the spent nuclear fuels arising from fast reactor, due to high burnup in comparison to thermal reactor spent fuel reprocessing. This results in the generation of large amount of organic waste that calls for the development of sustained methods for the regeneration of the solvent.

Most commonly the degraded PUREX solvent is washed with sodium/hydrazine carbonate solution since majority of the degradation products are acidic-in nature. It is essential to have an appropriate equipment for washing the degraded solvent. Impeller driven mixer settlers were used in radiochemical plants for primary solvent treatment using aqueous reagents. Mixer settler has the advantage of less head room requirement, but the impeller type design has the demerit of maintenance arising due to moving parts and electric components associated with the drives. Hence, it was decided to design and develop solvent treatment equipment which is convenient for remote operation and maintenance, less probable to choking, and efficient in performing primary treatment of solvent. The newly developed solvent treatment equipment is a continuous, cross current, 3-stage air sparged mixer-settler.

3. Air sparged mixer settler

Fig. 6 shows the schematic and photograph of the three-stage, continuous, cross current, air sparged mixer settler developed for treating the degraded organic phase. The equipment consists of three stages and each stage consists of a mixer vessel and a settler vessel. Each mixer is provided with a cross-type air sparger having 3 mm perforations along its branch. Aqueous and organic phases entering each mixer are mixed by air sparging and the mixed phase overflows to the respective settler through the overflow line provided at an elevation of 210 mm from the base of the mixer. The mixed phase gets disengaged at the settler and organic phase goes out through the overflow line provided at an elevation of 200 mm. The aqueous phase flows out through the underflow line through the outlet provided at an elevation of 179 mm from the base of the settler. Each stage is provided with a difference in elevation of 50 mm for the organic phase to flow by gravity across all the stages starting from the first stage at the highest elevation and last one at the lowest.



(a)

(b)

Figure 6. Schematic (a) and photograph (b) of the three-stage, air sparged mixer settler.

Hydrodynamic and mass transfer experiments were performed using the air sparged mixer settler. Fig. 7 shows the variation in concentration of DBP at the outlets streams during the mass transfer experiment. 30% (v/v) TBP-nDD mixture containing 3 g L^{-1} DBP was used as the simulated degraded solvent. 0.5M sodium carbonate and 3M acid was used for treating the solvent. Almost complete removal of DBP occurred in first stage itself.

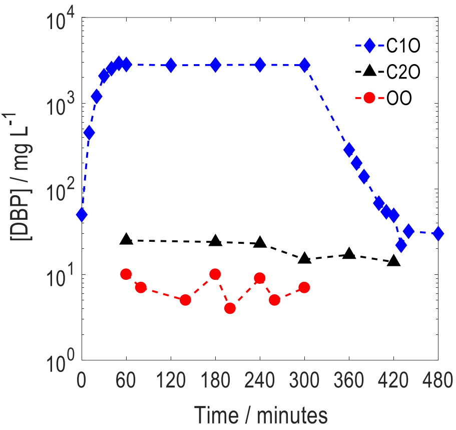


Figure 7. Variation in concentration of DBP at carbonate 1 out (C1O), carbonate 2 out (C2O), organic out (OO) as a function of time.

Concentration of DBP in the first stage aqueous outlet (C1O) has reached close to organic feed in 50 minutes thereafter remained constant till the organic feeding was stopped. After stopping the organic feed, the DBP concentration in C1O has gradually come down to less than 50 ppm within two hours. DBP content in the treated organic (OO) as well as second stage aqueous outlet (C2O) were very negligible (less than 20 ppm) as shown in the Fig. 7.

4. Stirred tank continuous precipitator and thickener for Plutonium Reconversion

The plutonium reconversion operation in the PUREX process is conventionally performed by a series of batch or semi-continuous systems. These systems have some limitations, such as low throughput, inconsistent product quality, and extensive manual intervention, leading to worker exposure, which are unfavourable and inefficient for large-scale reprocessing plants. In contrast, a continuous process typically offers higher throughput, reduced cycle times, reduced manual interventions, and consistent product quality, making it a safer, more efficient and reliable to the conventional batch or semi-continuous systems. In this context, a glove box-adaptable continuous precipitation and sedimentation system, as shown in Fig. 8, was developed for plutonium oxalate precipitation and separation. The system consists of a stirred-tank precipitator equipped with a standard impeller and a sedimentation column having a slow-rotating rake.

Figure 8. Continuous precipitation and sedimentation system.

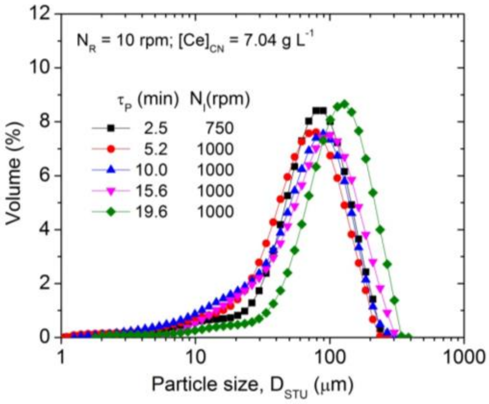


Figure 9. PSD of cerium oxalate as a function of residence time.

A comprehensive evaluation of the new system was carried out using Cerium(III) as the non-radioactive surrogate for plutonium. The effects of critical process parameters, such as residence time, metal ion concentration, and rake speed, on the key performance indicators of the system, such as throughput, conversion, median particle size, solid recovery, and volume reduction, were assessed in detail to optimize the system. The optimum condition identified ensured the continuous steady-state operation of the system, achieving a throughput of 60-152 g Ce per hour,

>99% conversion, cerium(III) oxalate particles with a median size of 60-70 μm, >99.5% solid recovery and 87-94% volume reduction by gravity sedimentation.

Lower cerium concentrations (<2.7 g L^{-1}) led to decreased conversion and increased solid carryover affecting overall performance of precipitation. Fig. 9 shows the particle size distribution as function of resident time. The results showed the strong influence of residence time on the relative rates of growth and nucleation mechanisms and the density of the cerium(III) oxalate particles.

5. Helical Coil Based Pumping System

The pump used in the nuclear reprocessing plant should be highly reliable, with minimum moving parts and zero leak rates. Generally, four methods of liquid transfer systems without moving parts are used to handle radioactive liquids and they are (i) steam jets, (ii) airlift, (iii) vacuum and (iv) blow cases, etc . Later, several fluidic devices based pumps were developed and successfully operated in the nuclear reprocessing facility and they are Reverse flow diverter (RFD), Pulse Jet Mixer (PJM), Double Diode Pump (Vortex Diode), Vacuum Operated Slug Lift (VOSL), etc. Centrifugal extractor is widely used in solvent extraction operation, especially in fast reactor reprocessing application. Flow fluctuation in centrifugal extractor system drastically affects the steady-state operation and reduces its mass transfer performance. In fast reactor reprocessing flow sheet, throughputs for aqueous and organic phases vary from few mL/min to L/min. The airlift pumping system is not suitable for low throughput metering operation. Other fluidic pumps are also not suitable for low throughput as well as metering operation. To overcome the above problem helical coil based fluidic pump is developed.

The working principle of helical coil based pulsating pump is similar to reciprocating pump. In reciprocating pump, piston pushes the liquid to and fro in closed compartment. Outlet valve opens during pumping cycle and inlet valve opens during refill/suction cycle. Similarly, in helical coil based pulsating pump, liquid is pulsed in pulsing leg and helical coil connected at inlet line act as a leaky check valve during pumping cycle and open valve during refill cycle respectively. Main advantages of helical coil based fluidic pump is (i) mixing of air with process liquid during normal operation is nil, (ii) pumping capacity is independent of liquid operating temperature, (iii) easily scalable, (iv) it can be used as a metering pump and (v) air never enters inside the feed tank.

Schematic diagram of helical coil based fluidic pump is shown in Fig. 10 and photographic view of helical coil is shown in Fig. 11.

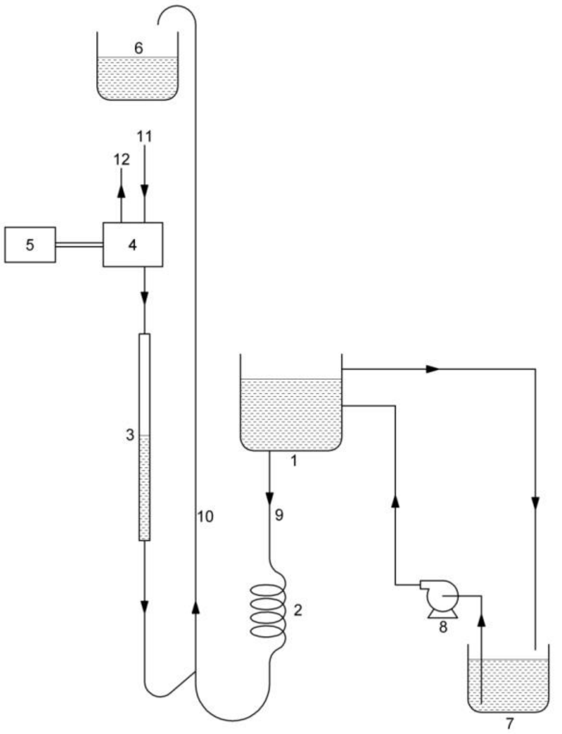


Figure 10. Schematic diagram of the Helical coil based fluidic diode pump (1) feed tank, (2) helical coil, (3) Pulsing limb, (4) three ways solenoid valve, (5) Timer, (6) Discharge tank, (7) Bottom tank, (8) Pump, (9) liquid inlet line to helical coil from feed tank, (10) fluid discharge line to discharge tank, (11) Compressed air inlet line to solenoid valve and (12) vacuum line connected to air ejector through solenoid valve.

During filling cycle, fluid flows from feed tank (1) to pulsing limb (3) through liquid inlet line (9) and helical coil (2) by gravity and vacuum (12). During pumping cycle, fluid inside the pulsing limb (3) is pressurized by compressed air (11 and 4) and fluid from pulsing limb flows into discharge tank (6) through discharge line (10) and also part of the fluid flows into feed tank (1) through helical coil (2) and inlet feed line (9). During pumping cycle pressure drop inside the helical coil (2) is much higher than the pressure drop (due to elevation) in discharge line (10) because of flow in turbulent region. Pulsing limb (3) is connected with compressed air line (11) with air pulsing unit which consists of a timer (5) with a three-way solenoid valve (4) and vacuum line (11) which is connected with air ejector to control the pressure during pumping cycle and refill cycle. During pumping experiment, feed tank (1) was maintained with constant fluid level by pumping the liquid from bottom tank (7) through pump (8).

Figure 11. Photographic view of helical coil.

5.1. Helical coil based pulsating pump details

Pumping experiment was conducted in the helical coil based pulsating pump. The above helical coil based pulsating pump was operated at different pulsing on and off time. During experiments, it was observed that the refill time less than 5 second leads to entering of air inside the discharge pipe. It was mainly due to the time required to refill the liquid in pulsing limb was found to be much lower than the discharge throughput. Similarly pumping time less than 1.3 second leads to no pumping. Helical coil based fluidic diode pump pumping experiment results are shown in Fig. 12.

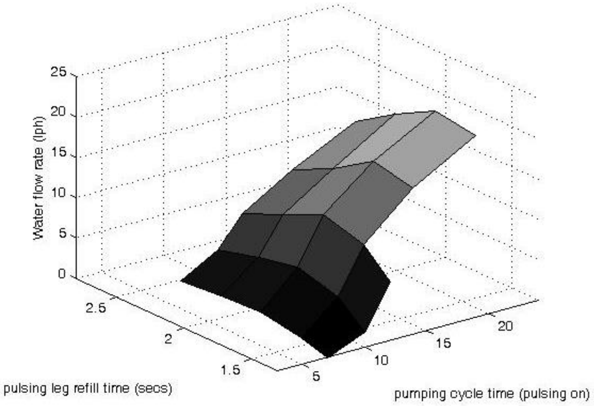


Figure 12. Pumping capacity of helical coil based fluidic pump.

The novel helical coil based pulsating pump was demonstrated as a pump and it may be used in nuclear reprocessing applications as well as other process industries to handle hazardous liquids

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Concurrent Development and Commissioning of Indigenous Hot Cell Equipment for Accelerated Plant Construction

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Abstract

Fast Reactor Fuel Reprocessing Plants (FRFRP) have both process cells & hot cells. Process cells accommodate static components like process vessels & massive piping system, while hot cells house many motorized equipment, fixtures & gadgets, calling for maintenance. Process cells of Demonstration fast Reactor Fuel Reprocessing Plant (DFRP) of IGCAR house a striking number of process vessels, surrounded by a labyrinth of high-density piping system, measuring a whopping 150 Km (approx.) & a staggering number of radiographic quality butt weld joints, to the tune of 1,50,000 nos. The above specification inevitably calls for prolonged construction time. The article throws light on the ingenuity ingrained during the construction phase of DFRP of RpG, IGCAR to optimize the project construction time, hence the project cost. Developing equipment for hot cells of FRFRP is tricky, challenging & nonconventional in view of the high radiation, acid fume laden ambience. The cells are designed leak tight and maintenance of the equipment shall be done without breaching the harsh ambience inside. Hence the equipment shall be constructed robust, simple but multi-functional, modular, with minimum drives, with limitation on the material of construction, without sensors, etc. Main focus of this article is on the indigenous customized special purpose automated equipment, fixtures, fittings, etc. installed in the alpha tight Hot cells of DFRP, complying the above listed stipulations.

Keywords: *fast reactor fuel reprocessing, Hot cell Equipments, Project Construction*

1. Construction Experience

Pipe Bend Fixture

High dense piping as shown in the Fig. 1a is inevitable in any FRFRP, which calls for innumerable pipe loops with less width. Fabrication of less width loops (Fig. 1b) in "Hydraulically operated ram type bending machine (Fig. 1c)", mandates, forming two independent spools with 90^0 bends, followed by welding of the two spools as shown in the Fig. 1b. This calls for additional butt weld & associated NDT, hence manpower& cost.

In DFRP, a fixture was fabricated attachable to Ram type bending machine, as shown in the (Fig. 1c) to fabricate less width loop, saving 1000 s of butt joints (Fig. 1d). The numbers would be way above this in FRP, FRFCF, which is a large-scale reprocessing plant.

Instant Tack Weld Free Pipe Butt Weld Fixture

Construction of FRFRP is a prolonged activity due to the complex high density and unimaginably long piping system & hence the massive pipe weld joints involved. The two activities roughly share a whopping 50% of the total construction time. The time saved on deploying the tool in piping projects like FRP or any other similar projects would be phenomenal (Fig. 1e).

Butt Weld Fixture

The average time taken for fit-up of a single butt joint is around 45 minutes, including the tack welding, inspection involved, etc. A pipe butt weld fit-up fixture has been developed in DFRP, which is instantaneous, requiring no tack weld, hence no need for tack welder & grinding machine operator. Since the fit-up is machine made after gaining confidence even the fit-up inspection can be waived off (Fig. 1f, g).

Highlights

- Pipe tack welding for fit-up is not required.
- Tack welder & Grinding operator are not required
- Start & stop of welding during grinding the tack welds is not applicable.
- Defects connected with tack welding done during fit up are eliminated.
- Time saving per joint is approx. 45 min.
- No. of joints in DFRP is approx. 150000 joints
- No. of joint in FRP in 3-fold.
- Huge saving in project time and cost.

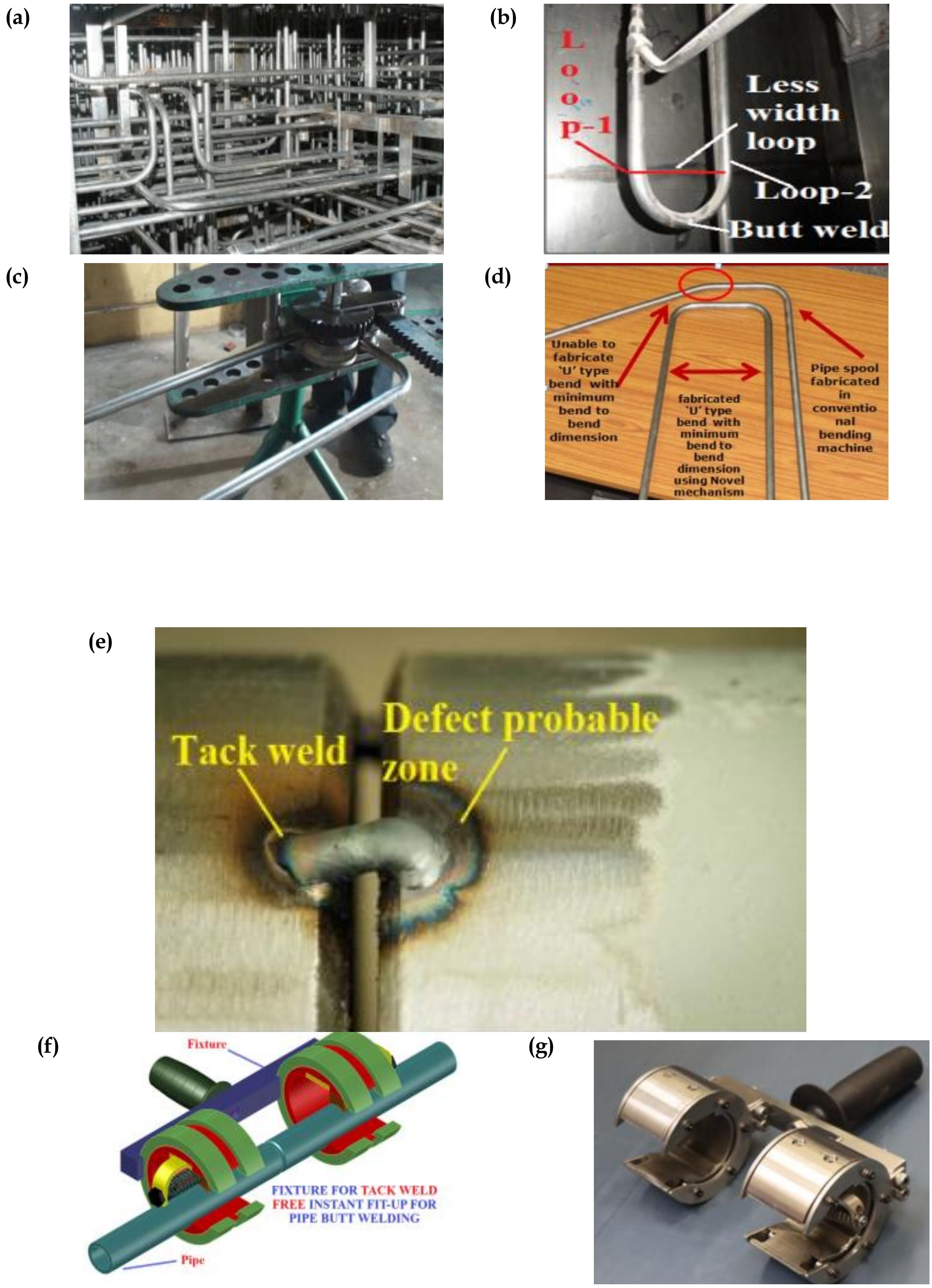


Figure 1. (a) Typical piping in DFRP, (b) Typical less width pipe loop, (c) Bending machine Fixture, (d) Loop formed with & without fixture (e) Tack weld (Typical), (f) Image of Butt weld fit-up fixture, (g) Mock-up Butt weld fit-up fixture.

2. Manipulator Reach Simulator for Hot Cells

Contactor cells, sampling cells, analytical robot cells have multiple equipment like centrifugal extractor banks, MSM based sampling stations, remote sampling stations, Analytical sample handling robot, capper unit, etc.

Since these cells handle high volume Pu in solution form, the cells are constructed leak tight using stainless steel plates with external lead wall for compensatory shielding (Fig. 2). These cells house 90% of the total motorized equipment of FRFRP and are supported with MSM, in-cell crane & viewing windows to facilitate operation & maintenance.

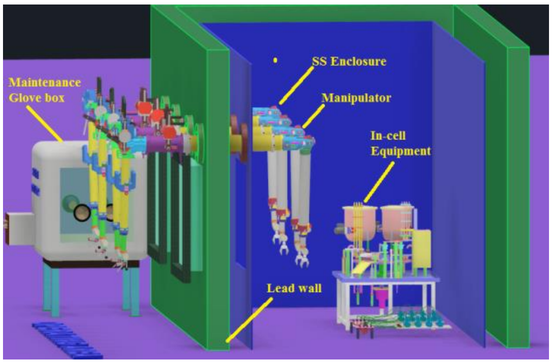


Figure 2. SS containment with Lead shielding.

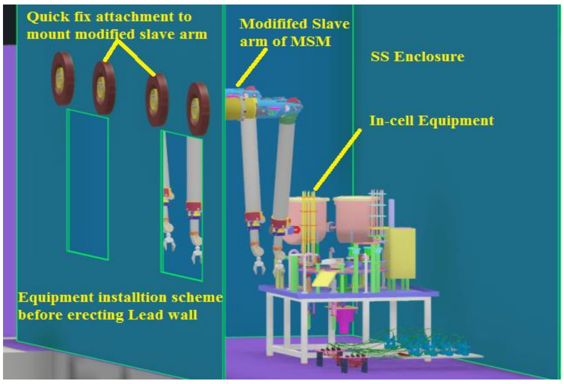


Figure 3. Proposed modified wall hung slave arm.

Equipment Erection

MSMs are installed on the lead wall. Equipment erection inside the cell would be validated based on the reach of the MSM. However, the lead wall erection will be a fag end activity for many practical reasons. However, in order to complete the equipment erection & piping erection prior to lead wall erection, MSMs would be mounted on temporary support in all locations to facilitate equipment layout finalization & will be removed later after completing the erection work, which is a high skilled, laborious, time-consuming operation.

As an alternate, installation of equipment in all cells can be completed with just a pair of slave side arms, cannibalized from a set of manipulators devoid of the internals, master arm& counter weight wherein the final weight of the slave arm would be around 20% of an actual MSM.

The modified slave arm (Fig. 3) of MSM shall be fixed with ball socket unit in all the joints. The movement of all the ball socket joints shall be restricted suitably with stoppers to simulate a real MSM movement in all the axes.

Features shall be added to the modified slave arm to facilitate the clamping of slave arm instantly to the inside wall of the containment. With the aid of the pair of slave arms clamped to the containment wall, equipment reach verification can be completed precisely in one go by manually moving the slave arm, which will imitate exactly the reach of a real MSM. The approach would save lots of time, manpower.

3. Modular Remote Sampling Station (MORSS)

Sampling of process solution provides vital information pertaining to plant performance, such as the decontamination factors achieved, the recovery obtained, etc., hence is crucial for FRFRP. A robot was developed by an external agency for collection of process sample solution from nozzles (Fig .4) beyond manipulator reach.

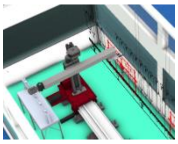
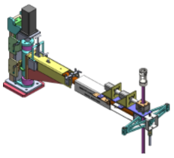
The robot could not be commissioned due to reasons listed in the Table 1 below. Hence an indigenous electromechanical equipment was developed in-house for the task & utilized successfully for remote sampling in all campaigns of DFRP.

Figure 4. Row of sampling nozzles.

Special Features of MORSS

The unit has features like, modular in construction, fastener free integration of modules, no sensors for engaging bottle with nozzle, passively safe with mechanical inter locks.

Table 1.

Limitations in		Merits in Department version
Initial Robot installed in cell	Design proposal (TYP)	
		
Complex mechanism		Fastener free, Simple & modular
Includes: 10 motors, 30 limit switches &10 bearings		2 motors, 3 limit switches & 6 bearings
50% of parts not reachable for maintenance		100% accessible. Entire unit can be assembled remotely in 1 hour
Sprawls over 60% of floor space		Occupies < 5% floor space
Approximate cost- 150 to 200 lakhs		9 Lakh for 3 units, per cell
Gross weight incl. support – 2000Kg. (approx.)		50 Kg. (approx.)

Highlights

- Robust, modular compact, Fastener free, remote assembly in 1 hour.
- No sensor used, till high accuracy & repeatability.
- Above 95% weight reduction against initial version
- 94% Cost reduction against initial version
- Above 80% reduction in equipment foot print
- 70% reduction in spares, hence reduction in maintenance probability
- Mechanical interlocks for better reliability
- Passive safety features enabled for equipment safety.

4. Analytical Sample Handling Robot (AHR)

AHR was developed as a replacement to a complex robot, designed & developed by an external agency (Table 2).

Construction

AHR has three units namely,

a) Capper unit, b) Pipette unit c) Indexing table

Table 2. AHR developed by external agency against the indigenous version.

AHR DEVELOPED BY EXTERNAL AGENCY	MODULAR INDIGENOUS AHR
	
Involves complex mechanism.	Simple mechanism & modular in
10% (approx.) success in capping/de-capping	100% successful operation.
Huge number of maintainable parts.	Includes very minimum maintainable parts.
Frequent calibration required for capping	No sensor & no calibration is required
Replacement of 50 % of maintainable parts are not feasible.	Replacement of all the parts is possible in very short time.
Spares are non-standard, hence difficult to maintain.	Readily available spares are only used.
Approximate cost – Rs. 150 lakhs.	Approximate cost Rs. 20 lakhs.

5. Hermetic Sealing Solutions for Power and Data Cable Bundle Feed-Through

Vessels, piping & motorized equipment, gadgets, etc., housed in hot cells are having features to serve for a prolonged life. Contrarily the primary cables routed in bunch (Fig. 5a) from outside to inside, vide pipe embedment are non-replaceable hence decides the plant life. Feed through available in market lack replacement feature. Considering the gravity of the issue an in-house feed through for cable bunch with cable replacement feature was developed and used in DFRP. In DFRP around 100 nos. of the feed through have been used.

Figure 5. (a) Cable bunch, (b) Cable feed through.

Highlights

- Leak tested @ 3.0 Kg./sq.cm using air soap solution.
- Helium leak tested @ 1.0 Kg./sq.cm.
- Helium leak tested at a –ve pressure of 10^{-7} mbar-l/s.

The product (Fig. 5a,b) is patented, Technology transferred, with good response from market.

6. Fixture for Remote Installation of Wall Mounted Fittings

Illumination of Hot cells are taken care by metal halide bulb, which are isolated from the cell ambience by glass dome (Fig. 6a). Provision shall be available for the replacement of the glass dome from outside. A fixture (Fig. 6b) was developed at RpG in this context.

The fixture has been tested & installed in the hot cells of DFRP. The fixture can be used for the installation of any wall mounted fitting like small motors, sensors, gauges, etc., blindfold in a very small time. Hence is suitable for hot cell application. The fixture has been patented and technology transferred to a private firm.

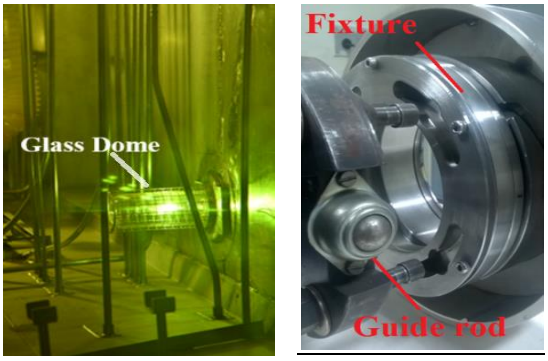


Figure 6. (a) Glass dome installed at site (b) View of Fixture & guide rod.

Highlights

- Blind fold installation
- No manipulators /in-cell crane required for installation.
- Equipped with self-locking feature.
- Leak tested at 3.0 kg (g)./sq.cm. (air soap)
- Leak tested at 1.0 kg (g)./sq.cm. (Helium)
- Replacement in < 3 minutes
- Maintenance free

7. Rotary Linear Actuator driven by Single Motor

Rotary linear actuator (RLA) is a special tool for providing both rotary & linear motion simultaneously or separately as required to the job. Two types of rotary linear actuators as described below are available in market.

- RLA driven by two electrical motors
- RLA driven electro pneumatically

RLA driven by single motor was developed in house, demonstrated & installed in one of the critical equipment called, Analytical sample handling robot in a hot cell of DFRP, which is working satisfactorily. The unit can be used as a universal tool for assembly/ disassembly of threaded parts remotely without the need for senor. The merits of indigenous RLA are listed in Table 3 below.

Table 3. Comparison of RLA based capper fabricated with:

Two motors	Electro-pneumatic	Single motor
		
Sensor required	Sensor required	Not required.
PLC based	PLC based	Not required
Motor synch.	Not required	Not required
Calibration for job with diff. thread spec	Not required	Not required
Not safe for idle rotation	Partially safe	100% safe
Needs Frequent recalibration	Requires to some extent	Not required
Complex mech.	Complex mech.	Sleek
High maintenance	High maintenance	No maintenance
Multiple operations in one axis- yes	Not Suitable	Suitable
High cost	High cost	Cheap
Tedious design	Tedious design	Plug and play

Highlights

- Driven by single motor, hence no need of synchronization between drives.

- No sensors required
- Self-tolerant to handle job with different thread pitch without PLC correction
- No need for calibration/ recalibration
- Immune to idle rotation threaded parts before engagement happens with counterpart.
- Ideal remote tool for assembly/ disassembly of threaded parts

8. Remotely Operated Centrifugal Extractor Bowl Removal Fixture

Centrifugal extractor is the heart of the FRFRP used for extraction & stripping processes. CORAL & DFRP have many centrifugal extractor banks (Fig. 7a) with a total of 40 & 150 stages respectively. Each stage has a bowl screwed to a motor shaft (Fig. 7b). The weight ratio of motor to bowl (Fig. 7c) is 90%.

(a) Row of CE Banks (b) Motor bowl assembly (c) Removed Bowl

Figure 7. CE Bank: Motor Bowl assembly.

During operation the bowl gets chocked and shall be removed & disposed of as a high active alpha bearing solid waste. Due to constraints in hot cell, the chocked stage is brought to maintenance glove box for decoupling the bowl from motor, which ensued only 10% success rate. Consequently, the motor needed to be disposed of as a high active alpha bearing waste, which is not at all an acceptable procedure.

A versatile hot cell compatible fixture (Fig. 8) was developed for coupling/ de-coupling of bowl and implemented in DFRP & CORAL.

Highlights

- Alpha bearing solid waste weight reduction by 90%.
- Enables hot cell dismantling.
- Operator dose exposure is eliminated.
- Blind fold fixture pin – bowl engagement.
- Toggle clamp enabled design to secure Bowl firmly, hence MSM friendly
- Coupling/decoupling (Fig. 7c) of bowl completed in a couple of minutes.
- Motor-Bowl assembly can be loaded in the fixture in any orientation.

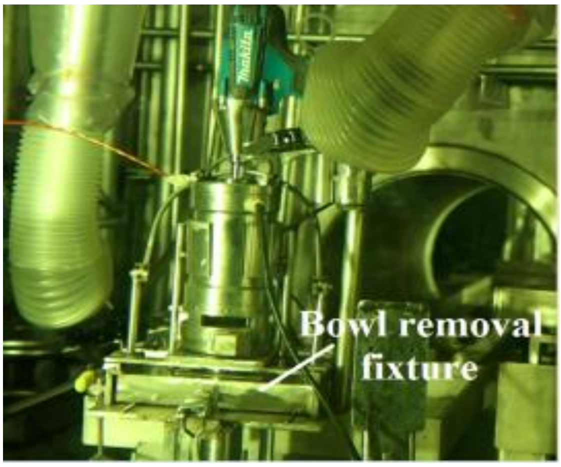
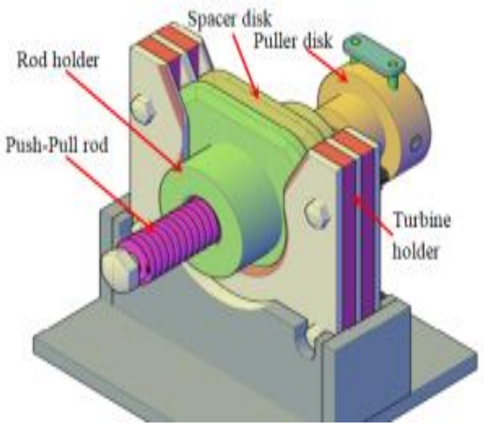


Figure 8. Fixture during Bowl removal in hot cell

9. Remotely Operated Centrifuge Turbine Bearing Removal Fixture

Spent fuel in DFRP undergoes chopping and dissolution in nitric acid, followed by feed clarification in a pneumatic centrifuge to separate un-dissolved particles. Bearing replacement in the centrifuge turbine assembly has been complex, time-intensive, and radiation-exposing, a specially designed Centrifuge Turbine Maintenance Device (CTMD) (Fig 9. a,b) has been developed for easy removal and assembly of components, addressing challenges in conventional methods.

(a)



(b)

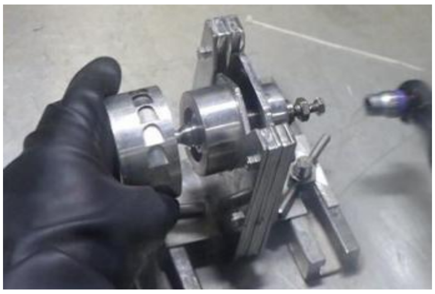


Figure 9. (a) Conceptual view of CTMD (b) Bearing being removed by fixture.

10. CE Motor Removal Fixture with Blind Fold Access

Centrifugal extractor (CE) is the heart of the Fast reactor reprocessing plant. Each CE bank has two rows, each row having 10 motor bowl assemblies. Each motor base is secured to the bed by 2 fasteners. The fasteners securing the motor-assembly are not visible & lie below a network of cables & pipe lines. Accessing fastener is very difficult, & has hitches like, holding the long neck wrench upright, countering the high torque while screwing/ unscrewing the nut. A fixture (Fig. 10) was developed in RpG, having a guide rod and slider to address all the issues listed above.

Consequently, the operator can reach the hidden fasteners blind fold and carry out the operation effortlessly in a very short time.

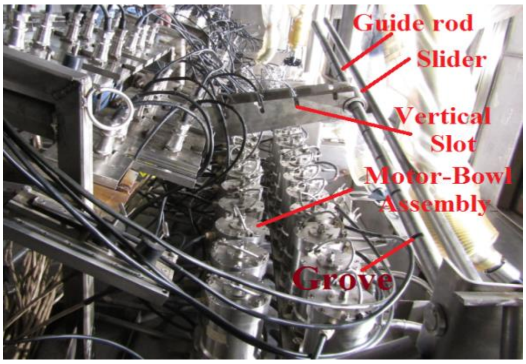


Figure 10. CE with motor removal fixture.

11. Pipe embedment for In-situ cable replacement with connector in hot cell

Vessels, piping & motorized equipment, gadgets, etc., housed in hot cells are having features to serve for a prolonged life.

Figure 11. Cable connector.

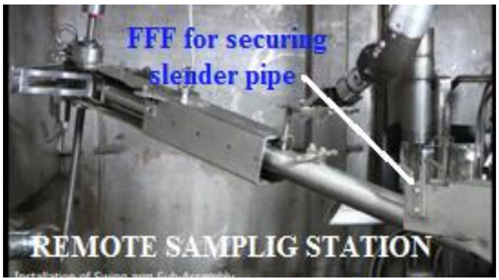
Contrarily the primary cables routed in bunch from outside to inside, vide pipe embedment are non-replaceable hence decides the plant life. Currently there is no mechanism available for cable replacement from a bunch. Hence sufficient spare cables with metal connectors (Fig. 11) are loaded. If the spare cables also become defective the associated equipment can’t be operated and may lead to permanent plant shutdown. A special EP was developed at RpG, which permits replacement of cable along with connector form hot cell, thereby by supports plant life extension.

12. Fastener Free Fixture (FFF) for Instant Mounting of Structures/ Fittings Remotely

Generally, any equipment is constructed by an assembly of structure, fittings & plates. Adding simplified remote installation features to these parts would facilitate assembly/ disassembly of a complete equipment remotely. A fastener free fixture for remote installation of tubular product was developed, tested & implemented in Remote sampling station (Fig. 12a,b). Similarly, a fastener free fitting (Fig. 12c) was developed for the remote installation a motor. This fixture also has features to enable installation simply by the pull of a lever, hence suitable for hot cell application.

(a)

(b)



. **(c)**

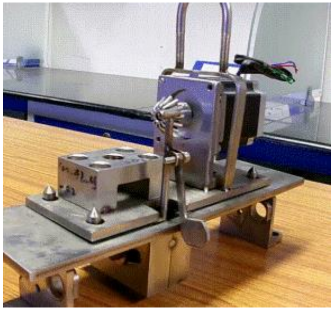


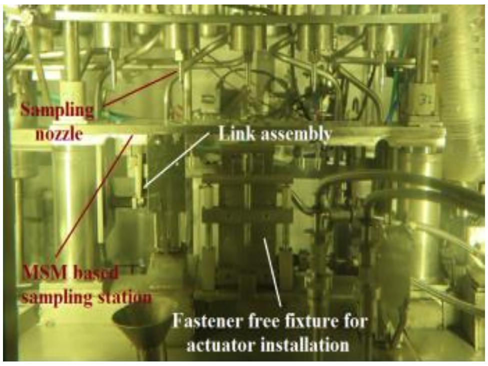
Figure 12. (a) FFF for structure, (b) Fixture installed in hot cell (c) FFF for fittings.

13. MSM based Sampling Station

Sampling of process solution is essential for effective process control and material accounting. MSM based sampling station (Fig. 13a) operated by a single pneumatic actuator providing movement to the bottle holding tray and the stripper assembly was developed & implemented. Despite both the bottle holding tray & the stripper assembly are designed to travel at different relative velocities to engage at different points of time.

The station is also equipped with a fastener free mechanism (Fig. 13b) for replacement of pneumatic actuator remotely in a very short time.

(a)



(b)

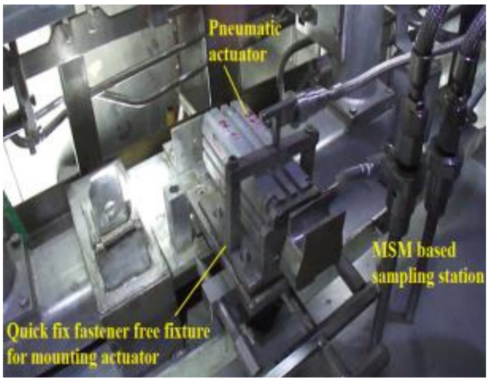


Figure 13. (a) MSM based sampling station (b) Pneumatic actuator fixture.

14. Electrically Operated Jacks for Handling Heavy Concrete Plug

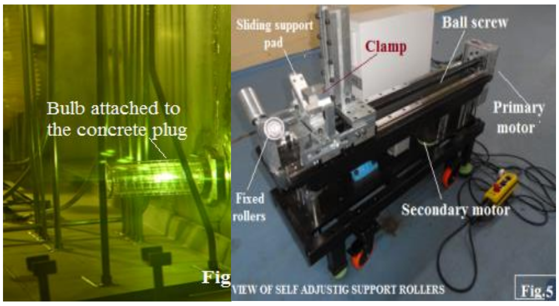


Figure 14. (a) Bulb fixed with CP Jack (b) Heavy duty CP Jack.

DFRP many electrical plug (Fig. 14a) ports closed with concrete plugs, which are 1.8 mtr. long and l & weighing around 250 Kg. An electrically operated heavy-duty jack (Fig. 14b) was developed and demonstrated for loading & unloading of the concrete plug in the electrical wall port to minimize the operator effort.

15. Light Duty Lead Plug Jack

The lead walls of hot cells in DFRP are provided with multiple electrical ports, closed with lead plugs measuring 600 mm long & weighing around 40 Kg.

An electrically operated light duty jack (Fig. 15) was developed and demonstrated for handling the lead plugs in the electrical port of the lead wall to minimize the operator effort.

Figure 15. Low duty LP Jack.

16. Spacer Wire Removal Fixture (SWRF) for FBTR Fuel Pin for DFRP

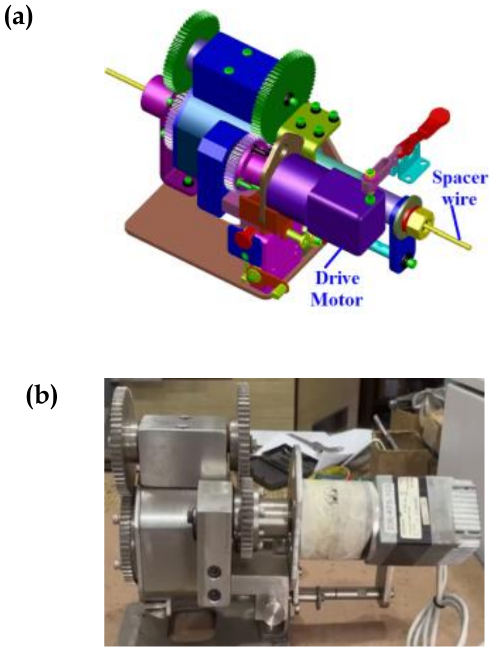


Figure 16. (a) Conceptual view of SWRF, (b) Mock up unit of SWRF.

Dissolver is a crucial equipment of fast reactor fuel reprocessing plant. Input to dissolver is chopper, which feeds chopped fuel pins vide chute into the dissolver. Chopping the fuel pins along with spacer poses many issues during chopping, dissolution & also while disposing the same. To address the issue, a tool (Fig. 16a, b) is being developed to remove spacer wire before feeding the same to chopper.

Since the cell is already commissioned, the tool shall be compact, modular, sporting features to be installed remotely, in tandem with remote operation & maintenance.

17. Development of A Reliable Mechanical System for Overload Protection for Hot Cell Cranes

Hot cells in DFRP are installed with in-cell cranes for operation & maintenance of in- cell equipment. Many operations inside the cells are hinged on the availability of the same. Hence non-availability of in-cell crane may lead to even permanent shutdown of the plant, which is catastrophic.

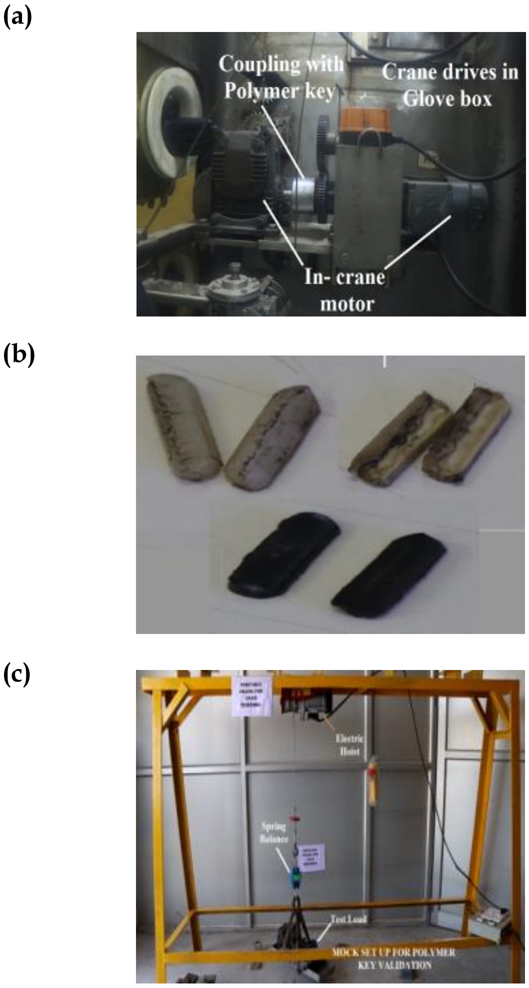


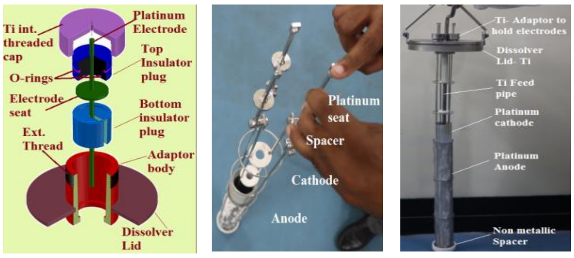
Figure 17. (a) Maintenance glove box with In-cell crane motors (b) Failed Polymer key samples & (c) Mock testing arrangement.

The electrical motor for the in-cell cranes are housed in a glove box (Fig. 17a) outside the cell, to enable contact maintenance. The maintenance glove box doesn't have enough space to introduce any standard overload protection device. Electrical overload protection devices are less sensitive and any error may lead to irreparable damage. Hence both options were dismissed.

In view of the above limitations, a new simple fool proof & cost-effective strategy was developed, whereby the metal key of the first coupling connecting the driver & driven shaft of the power transmission line was replaced with non-metallic key (Fig. 17b) to facilitate its failure well below the safe load of the chain or rope

18. Titanium Electrolyzer with Platinum Electrodes

DFRP has the mandate of reprocessing mixed carbide fuel on regular basis and demonstration of mixed oxide fuel. For MOX fuel, dissolution shall be supplemented with electrolytic dissolution. Electrolytic dissolver body is made of titanium, while electrodes (Fig. 18b) are cylindrical, long & made of platinum, wherein small diameter cathode is mounted within the anode concentrically with a narrow annular gap in-between. The electrodes are suspended from the lid (Fig. 18c).



(a) Adaptor with insulator plug (b) Electrodes on assembly (c) Dissolver lid with electrodes

Figure 18. Electrode of Electrolyzer.

The equipment has been fabricated with the following features.

- Dissolver lid ensures a leak tight with the body without using fastener.
- Provision for remote assembly/ disassembly of electrodes
- Cable replacement feasibility

The electrolytic dissolver unit was tested satisfactorily performing electrolysis of concentrated nitric in simulated conditions corresponding to parameters of irradiated MOX.

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Challenges faced and Developments in Electrical, Electronics and Instrumentation

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Abstract

Fast Reactor Fuel Reprocessing Plants require specialized Electrical, Electronics, and Instrumentation systems to meet the requirements of process operation and safety. The radiation fields in the hot cells are particularly high due to the large burnup of the spent fuel. The high fissile material content of the spent fuel poses a significant risk of accidental criticality at all stages of the process. Many of these systems operate in the harsh radiological and chemical ambience of the hot cells. The operation and maintenance of the systems pose unique challenges. The paper describes the novel electrical, electronics, and instrumentation systems developed and deployed in the fast reactor fuel reprocessing plants. The novel systems include the heat pipe based LED Light, Passive neutron assay based waste drum assay system for fissile material, Averaging Pitot Tube based stack flowmeter, Dual phosphor scintillator based radiation monitors, Soft Dissolver Temperature Sensor based on Artificial Neural Network, Control System with wireless signal transmission, Control System for vacuum based liquid transfer, Speed sensing system for Centrifuge, and VFD based Control System for Centrifugal Extractor Motor Bank. The experience of operating the Electrical, Electronics, and Instrumentation systems in the Plant is also provided in the paper.

Keywords: *LED Light, Passive neutron assay-based waste drum assay system, Averaging Pitot Tube based stack flowmeter, Dual phosphor scintillator-based radiation monitors,*

1. Introduction

Fast Reactor Fuel Reprocessing Plants reprocess the spent fuel from fast nuclear reactors. The spent fuel from fast reactors is characterized by high burnup of up to 155 GWday/T. These specialties of the spent fuel pose a challenge not only to the process design and operation but also for the electrical, electronics, and instrumentation systems. The radiation fields in the hot cells are particularly high due to the large burnup of the spent fuel. Many of these electrical, electronics, and instrumentation systems operate in the harsh radiation and chemical ambience of the hot cells. The operation and maintenance of the systems pose unique challenges.

2. Heat Pipe based LED Light

The Plant required the development of a special lighting system for the lead wall shielded hot cells. The design of this lighting system was a challenging task because of the confined opening (90 mm diameter) for the entry of light fixtures, limited provision for cooling and high illumination requirement.

A unique heat pipe based 50W LED light fixture was developed for this application. Heat pipes

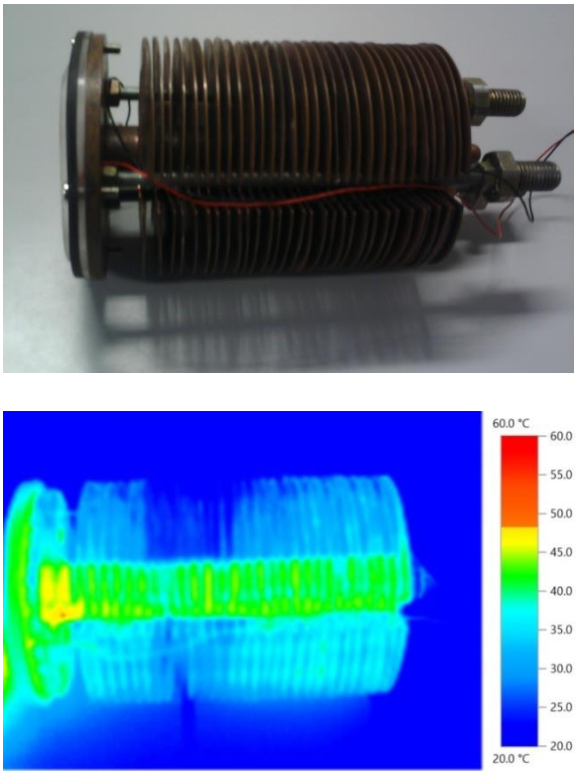

Figure 1. Optical and Thermal image of 50 W Heat Pipe based COB LED lamp with copper fins.

employ phase change to transfer thermal energy from one point to another by the vaporization and condensation of a working fluid. In this design, a cylindrical heat pipe and de-ionized water as the working fluid is used (Fig. 1). Copper fins are provided on the heat pipe for further heat dissipation into the ambient air.The LED used is the Chip On Board (COB) type and has a relatively high luminous efficacy 140 lm/W. The high efficacy provides high light output with reduced heat generation. The average illumination was over 2000 Lux in the hot cell as viewed for the Radiation Shielding Window. Various components of the Heat pipe based LED light fixture was tested in gamma radiation chamber and found to performsatisfactorily in terms of long time operation and easy maintainability.

3. Passive Neutron Based Waste Drum Assay System for Fissile Material

It is essential to quantify the plutonium in the alpha-active solid wastes generated during the reprocessing of fast reactor spent fuel. Neutron waste assay systems measure the uranium or plutonium content of waste containers or objects. They can be either PASSIVE (measuring spontaneous fission neutrons) or ACTIVE (measuring induced fission neutrons). In the passive system, the inherent emission of neutrons by the even-even isotopes of plutonium is utilized and this is suitable for assaying the plutonium in the waste drums. The sensitivity is dependent on the isotopic composition. A complete system based on the passive neutron counting technique using He-3 detector has been designed and fabricated indigenously (Fig. 2). The system consists of eight He-3 based neutron detectors embedded in High-Density Polyethylene (HDPE) moderating medium and configured in a semi-circular shape (Fig. 3) which encircles half of the drum containing the waste to be assayed. Each detector has a sensitivity of 130 cps/nv. Gamma tolerance of 20 mSv/h is achieved using lead shielding.

Studies were carried out to determine the effect of the matrix on the counts obtained using a 1g plutonium source of standard isotopic composition. Extensive calibration was carried out with different known quantities of Pu distributed in a heterogeneous manner simulating different gamma fields as background. The system can be used for the drums having up to 2 R/h surface dose.

Figure 2. Waste Drum Monitoring System.

Figure 3. Arrangement of He-3 detectors.

4. Dual Phosphor Scintillator based Radiation Monitors

A Dual phosphor scintillator is sensitive to both alpha and beta radiation and the two types of radiation can be individually measured using pulse shape discriminating electronics. This two-in-one sensor provides significant savings of space and cost as compared to the conventional radiation monitors which could detect only one type of radiation. This system also leads to saving of time when checking for radioactive contamination, measuring the radioactivity in samples etc. The developmental activity carried out in-house includes optimizing the parameters like the thickness of the dual phosphor and the shape discriminating parameters of the electronic circuit in order to obtain maximum efficiency, minimum noise and minimum cross-talk. Various types of radiation monitors were developed, qualified and deployed in the Plant including hand & clothing contamination monitors (Fig. 4), continuous air monitor, alpha & beta counting systems and contamination survey meters

Figure 4. Dual phosphor scintillator based hand & clothing contamination monitor and its internals.

5. SiPM based Alpha Beta Hand Foot and Cloth Contamination Monitor (ABHFCM)

Radioactive contamination monitor in a radio-chemical plant is required for monitoring the alpha-beta radioactive contamination on hand, foot and cloth of the personnel working in the plant. The instrument comprises of seven Alpha-Beta detectors and seven channel electronic module with display unit. It generates audio-visual alarms on detection of contamination above preset limits. It also alerts the user if the monitoring process is terminated before the preset monitoring time. The detected events are counted over time and displayed digitally in cps or cpm in the display unit. The unit stores measurement data and also can communicate with external data acquisition system via ethernet. The monitor performs reliably round-the-clock in the nuclear plant environment. These monitoring devices are installed at the exits of radiation controlled areas in nuclear power plants and other facilities, where radioactive materials are handled. Along with the conventional radiation detectors, requirement of other type of detectors have been growing for various nuclear applications. Over the last few years, internationally several scintillation fiber based detectors are developed for hand, foot and clothing contamination monitor. This development introduces latest detector technology to develop compact monitors with improved performance. One such development is the use of dual phosphor detector with SiPM and optical fiber (wavelength shifting fiber for light coupling). The advantages of this detector over the conventional detectors are: 1. Single detector for monitoring alpha and beta contamination, 2. SiPM operates at 55±3VDC and hence the use of High Voltage in the unit is avoided, 3. Detector coupled with SiPM and wavelength shifting fibers makes the system simple compared to conventional Photo Multiplier Tube (PMT) based unit, 4. The efficiency is comparable to PMT based unit, 5. The efficiency does not vary much over the entire surface area of the detector, 6. Foot detection of alpha and beta particles is possible whereas the conventional unit uses GM tube for foot detection where only beta detection is possible and 7.economic and rugged. Efficiency variation study at different position of the detector was carried out with alpha (Am241) and beta (Sr 90) sources (Fig. 5 & 6).

Figure 5. Alpha Beta Hand Foot and Cloth Contamination monitor using SiPM and optical fiber.

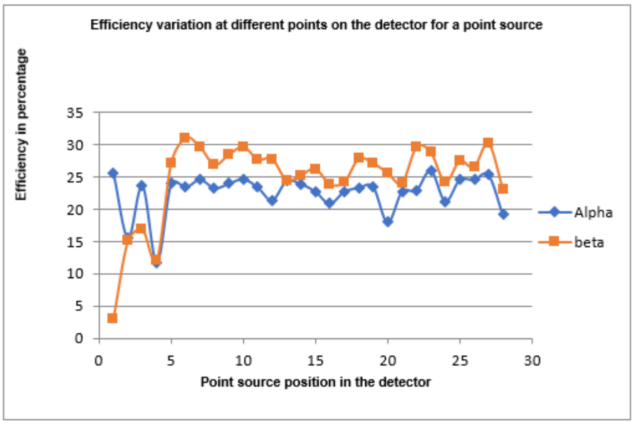

Figure 6. Efficiency variation at different points on the detector for a point source.

6. Stack Monitoring System

Stack effluent monitoring is required for measuring the particulate air borne radioactive contamination at Reprocessing Plant. Stack effluent monitoring for DFRP comprises of six channels (i.e. gross alpha – 1 channel, gross beta – 1 channel, Krypton – 2 channels and Iodine – 2 channels). The gross alpha and beta are detected by dual phosphor detector which is a sandwich of ZnS(Ag) and plastic scintillator. The detection is done by detecting the particulates collected over the high efficiency particulate activity (HEPA) filter papers. Krypton-85 is detected by allowing the air to pass through a large volume chamber containing a cylindrical plastic scintillation detector coupled with suitable PMT. Iodine is captured in charcoal and the same is measured by NaI(Tl) flat type crystal. The Duct air is continuously sampled @ 50 lpm using a pump. The set up was developed and commissioned at DFRP stack monitoring room (Fig. 7 & 8). The Suction system provides the required suction for drawing air through the filter and chamber from the air sampler assembly. The system contains an inlet header and outlet header connected to the duct. Each header has nozzles with needle valve to be accessed by quick fit coupling arrangement. Also common inlet port of the inlet header and common outlet port of the outlet header is provided with isolation valves. The inlet header is fixed horizontally at a height assessable to the operator. The outlet header and the intermediate headers are placed within the table fabricated to place the detectors along with the lead shielding. A noise free, dry type, oil less vacuum pump is used for suction, mounted over the anti-vibration pads. The two air sampling lines used for bifurcated detection has a Rotameter of 100 LPM flow capacity. The detector modules have in-build high voltage circuit and pulse shaping circuits. The electronic unit contains counter module having six independent counting channels corresponding to each of the detector outputs. The detected events are registered by the counter module. Display & control module display the activity as counts per minute or counts per second and generates audio & visual alarm if the same exceeds a pre-set value. The electronic unit is connected to the plant SCADA via Modbus RTU protocol over Ethernet interface which enables it to be logged at the plant server.

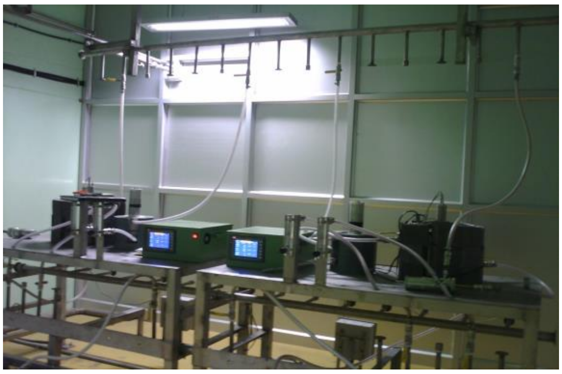

Figure 7. DFRP stack monitoring system.

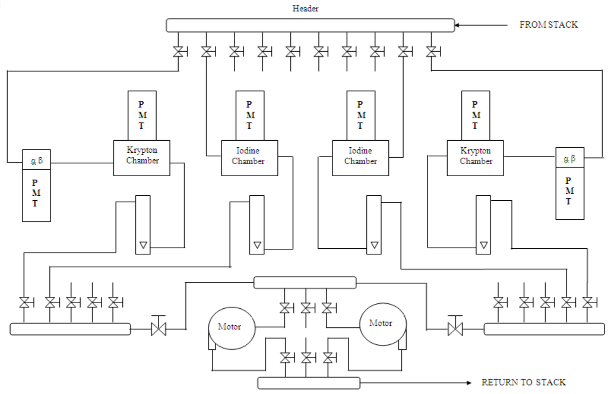

Figure 8. Schematics of DFRP stack monitoring system.

7. SiPM based spot contamination monitoring

Portable Alpha & Beta combined counting system for spot contamination is required for monitoring the particulate alpha & beta radioactive contamination on a small or narrow area as on the inner surface of a pipe. The contamination monitor contains a detector, pre-amplifier placed very close to the detector, counting electronics and readout electronics with display unit. Conventional detectors with PMT cannot be used for this application due to size limitation and hence a new detector with Silicon Photo Multiplier (SiPM) was developed to meet the field requirements. Unlike the conventional PM tubes, SiPM is an active semiconductor device consisting of multiple Avalanche Photo Diodes (APD) arranged to form an array in the sensitive area (Fig. 9). The SiPM is a very compact device. For detecting the radiation, 0.25 mm thick dual phosphor scintillator detector was used. SiPM with a photo sensitive area of 6x6mm was used as the photon detector. The dual phosphor detector was optically attached to the SiPM covering the sensing area. The detector and photon sensor were covered with Mylar sheet to arrest the light leakage. The biasing voltage of SiPM was 58 V

DC. The gain of the SiPM is 7 × 105.The Pre-amplifier circuit was assembled on a PCB of dimensions 85 × 12 × 10 mm. The SiPM is operating in reverse biased condition and the cathode is connected with the positive supply through a current limiting resistor. The output from the preamplifier was fed to the signal processing circuit. The alpha and beta discrimination was done based on amplitude and pulse width. The pulses due to Alpha and Beta were separated and counted. The same was displayed on a HMI. The system was tested using various standard radioactive sources. On testing with Pu 239 alpha source and Sr90-Y90 beta source, efficiency was found to be 20% and 22% respectively. A natural Uranium source also used to test combined Alpha & Beta response of the system, the alpha efficiency was found to be 30% and beta efficiency was found to be 45%. The average background was found to very less for alpha and beta. The system is satisfying our required criteria for contamination measurement.

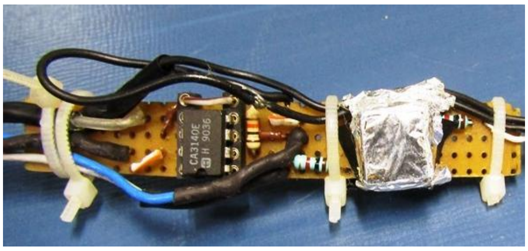

Figure 9. Detector unit.

8. Diagnostic System, Surveillance and the Reliability of the Criticality Alarm System

Criticality Alarm System (CAS) is mandatory in nuclear fuel fabrication and reprocessing facilities for round-the-clock monitoring of criticality accident. CAS is used for generating alarm for evacuation in the event of a criticality incident. The CAS consists of three independent channels and alarm module. Each channel contains an ionization chamber, preamplifier, and an electronic module. The ionization chamber is a gamma-based detector and it has a sensitivity of $3x10^{-8}$A/Gy/h. The CAS shall not fail to detect any criticality event or trigger a false criticality alarm. To meet this requirement, online fault diagnostics, surveillance for CAS electronics from the control room is mandatory (Fig. 10). CAS generates a criticality alarm based on 2 out of 3 (2-o-o-3) voting logic using a relay that operates in a fail-safe mode, i.e., the alarm relay de-energises on a criticality alarm condition. Maintaining this system with maximum availability and minimum false criticality alarm probability is a challenging task. A detailed Failure Mode and Effect Analysis (FMEA) was carried on each element of the CAS electronics. The failure of low voltage and high voltage power supplies of the CAS were not annunciated for immediate corrective action. The failure of battery isolation diode and low battery indication were also not available. Based on the analysis, it was identified that such failures require an alarm annunciation in the control room. A diagnostic system is developed and implemented in the CAS along with reliability analysis. The development of new diagnostic system, estimation of reliability and surveillance carried out on CAS for prevention of false criticality alarms.

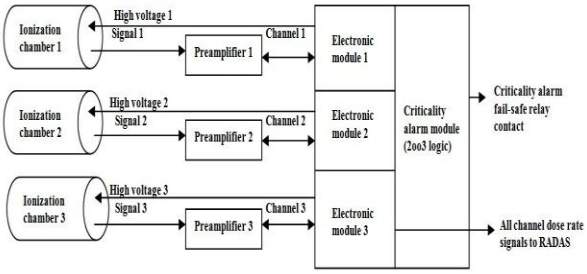

Figure 10. Criticality Alarm System.

The Failure In Time (FIT) data of each component is calculated using MTBF calculator software. The FIT data is added for different electronic components according to its quantity. It includes resistors, capacitors, relays, diodes, transistors, ICs and connectors which are calculated per billion hours. The MTBF of CAS channel PCB and alarm module PCBs are estimated to be as $3.46x10^{6}$ h and $9.32x10^{6}$ h respectively. The estimated failure rates for the additional PCBs are acceptable.

The fault diagnostic system is developed to detect the failures associated with AC mains, low voltage, high voltage, system on battery, battery isolation diode and channel alarm of each CAS channel. The MTBF of the diagnostic system PCBs are estimated to be better than $3.46x10^{6}$ h. The channel dose alarm is selected at the dose rate of 10 mGy/h and the alert alarm set-point is selected in RADAS between 0 to + 4 mGy/h for detection of fluctuations due to noise/drifts. Initially, since there was no diagnostic system provided in the CAS and false criticality alarms were generated. After the implementation of the diagnostic system, failures were detected and annunciated. There were no false criticality alarms from the CAS. Rigorous periodic surveillance procedures are carried out daily, weekly (Fig. 11), quarterly

(Fig. 12), and yearly to ensure maximum availability. The design of diagnostic system along with periodical surveillance procedures meet the intended application.

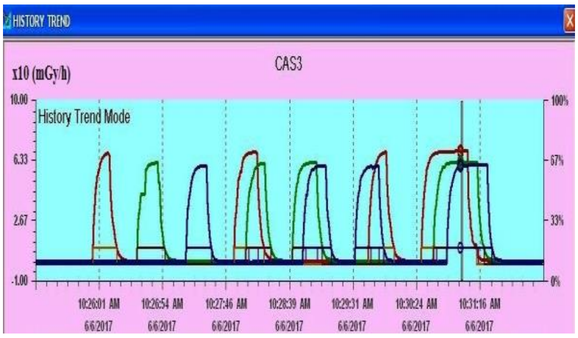

Figure 11. Weekly electronic surveillance test.

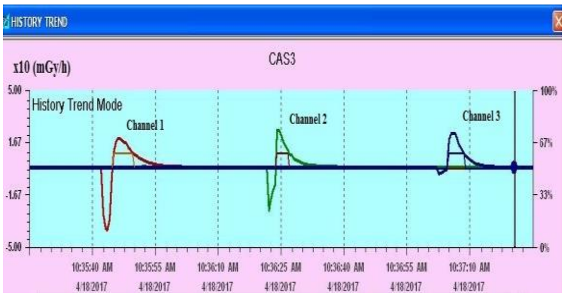

Figure 12. Quarterly channel loop surveillance test.

9. Soft Dissolver Temperature Sensor based on Artificial Neural Network

The dissolver equipment in a reprocessing plant has multiple thermocouples inside thermowell for temperature measurement that are located at various points of the equipment. The thermocouple has a risk of failure and it is difficult to replace them. As an experiment, data driven modelling of the main thermocouple (in the liquid) of the dissolver was carried out using the recorded data from the actual operation of the system. The data of the readings of the thermocouple in the liquid and the thermocouple in the vapour was logged (Fig.13). The modelling was done using Artificial Neural Network (ANN). In ANN, the model used was Nonlinear Autoregressive Exogenous (NARX). This is a time series modelling which uses the current and past values of the parameter to estimate the required value. The estimate of the liquid temperature was based on the vapour temperature showed an excellent relationship with the actual liquid temperature and the error was minimal (Fig.14). In the eventuality of the failure of the thermocouple in the liquid, the thermocouple in the vapour will be used to estimate the temperature of the liquid using ANN which can be used to control the dissolution process. This has shown that a soft sensor can be developed and used if replacement of physical sensor is not possible.

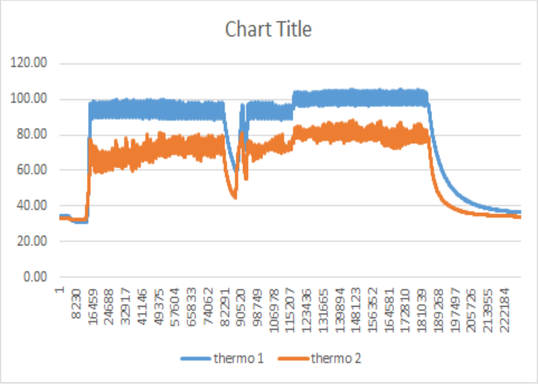

Figure 13. Time series graph of Liquid and Vapour temperature.

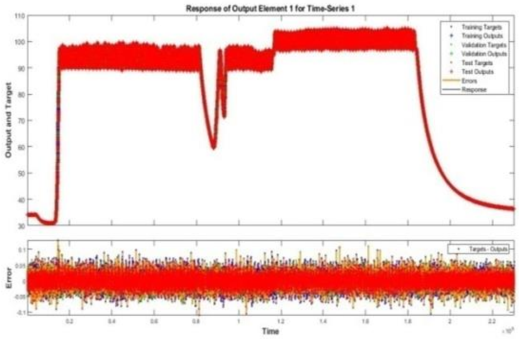

Figure 14. Estimated and actual liquid temperature and error.

10. Averaging Pitot Tube based Stack Effluent Flow meter with wireless communication

The stack is essential for the ventilation system of every Nuclear Plant. Measurement of the flow rate of effluent air through the stack provides confidence regarding the health of the ventilation system. However, this measurement is challenging due to site conditions. For the stack duct, its cross-sectional dimensions are very large, the straight length available is low and its location makes the access by personnel difficult. The averaging pitot tube (APT) was selected as the flow sensor as they are rugged, accurate, and are easy to maintain. To sense the flowrate, the pressure difference between the stagnation pressure and static pressure is measured. The stagnation pressure is averaged across the length of the duct by the APT using several taps. As the availability of the straight lengths of the duct is less, three APTs are installed to obtain a more accurate measurement (Fig. 15).

In order to overcome the difficulties in installing the long signal and power cabling, wireless signal transmission technology and in-built battery power were employed (Fig. 16). The wireless field devices communicate with the field wireless access point through the industrial automation wireless communication standard ISA 100.11a.

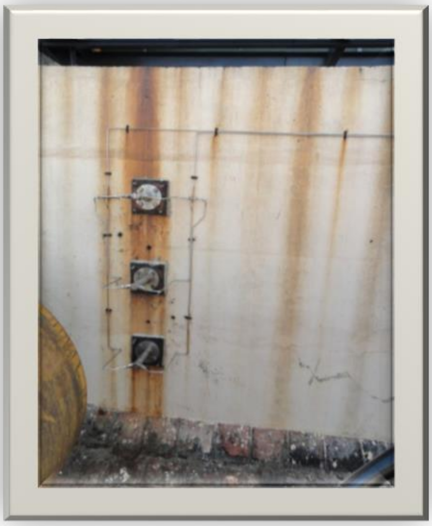

Figure 15. Time Three APTs in the exhaust duct.

Figure 16. DP transmitter with battery powered wireless transmitter.

11. Control System for Vacuum based Liquid Transfer

In reconversion lab of Reprocessing plant, majority of controls are based on open loop control system. This includes liquid transfer systems, sampling & sparging. Different methods adopted in liquid transfer system are airlift, vacuum transfer using an air ejector. Tank level is measured using purge-based instrumentation. Vacuum transfer (VT) method is adopted to carry out all liquid transfers. In VT, the required vacuum is created in destination tank by using an air ejector. The required motive air pressure supplied to air ejector is set at a constant value by manually adjusting the pressure reducing valve (PRV) in main air header. The air supply to a particular transfer is controlled by energizing a solenoid valve (SV) connected in designated nozzle of air header. A soft switch is provided in SCADA to operate the liquid transfer system. A hand operated valve is kept open for the transfer to take place.

In a typical vacuum transfer system (Fig. 17), vacuum is created by setting the header pressure in destination tank required for initiating liquid transfer. An interlock is provided to cutoff the SV based on the destination tank high level (≥ Level High Set point). During cutoff, due to sudden change in negative pressure, liquid will entrain in level purge line of the destination tank. Due to this the real time level, density & pressure values of the destination tank are not read correctly, mostly it indicates low or zero level. Because of this problem, the high level cut off will not work which may even lead to overflow of process liquid. In order to overcome this problem, a control system has been developed which is implemented by using a Programmable-logic Controller (PLC). The conventional PRV installed in the main air header is replaced by an electronic PRV to control the motive air pressure. The analog output of the PLC is connected to the electronic PRV. After the start of the transfer, a ramp output logic is initiated in the PLC to reduce the header pressure at a steady rate, till the target value of the negative pressure in destination tank is reached. All the pressures and timings are set by field trials.

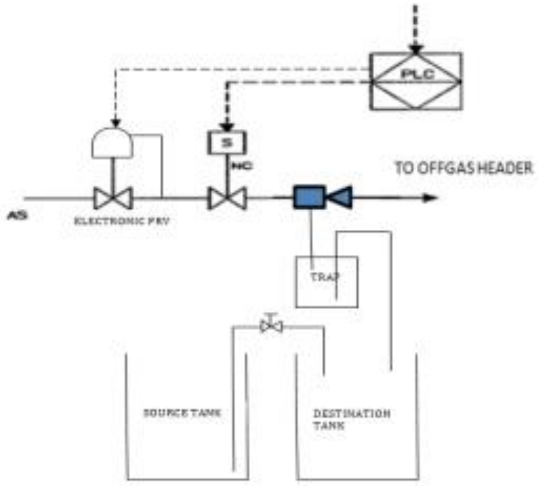

Figure 17. Automated vacuum based liquid transfer.

12. Remote Sampling System

The indigenously designed and fabricated RSS differs from the MSM based system in the method of handling the sample bottles. PLC based motion control is implemented to stretch swing arm to hold a sample bottle at a specific row column coordinate selected by the operator. The arm and the sample bottle actuator are stepper motor controlled and are remotely operable through a HMI pendant screen, as shown in Fig. 18. All motors operate in a preprogrammed sequence to accomplish the sampling task. The swing arm which is the heart of the system is rotated around a vertical shaft for carrying out the entire process. The sample collection is achieved with only two axis of motion.

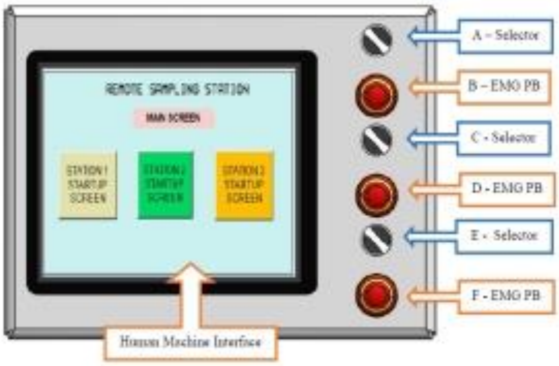

Figure 18. HMI screen for RSS.

13. Centrifugal Extractor I & C

The solvent extraction process uses annular centrifugal extractors, each of the extraction, stripping cycles is equipped with a centrifugal extractor bank consisting of multiple motors. Each centrifugal extractor is operated individually by a 3 phase custom build induction motor. Speed of each motor is within 3300-3600 RPM. The configuration is given in figure 19. The motors have been designed to prevent damage by radiation and acid corrosion. These motors have been grouped in two sets as group A running forward and group B running in reverse direction to have the unidirectional liquid flow in the CE bank. The speed of the individual motor is continuously monitored and any one deviate 10% of the operating speed the entire bank is automatically tripped off and inlet feed of the bank also cut off to avoid solution over flow.

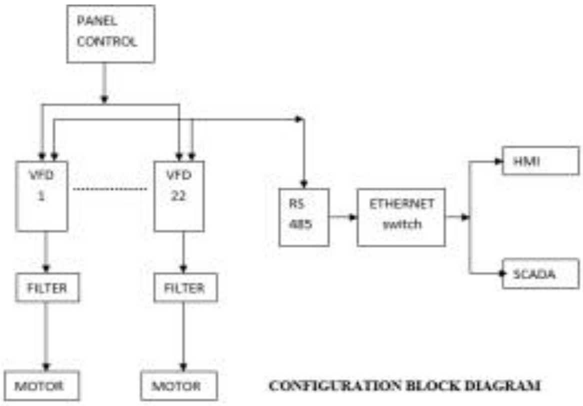

Figure 19. CE Motor I&C Block Diagram.

The variable Voltage Variable Frequency Drive (VVVFD) is operated with sophisticated control technology to control the motors for different mode of operation as per the process requirement (constant speed, constant torque, and constant power). One of the process requirements is constant speed operation for centrifugal extractors. To achieve the constant speed the drives are operated by closed loop with conventionally by complicated filter algorithms from the sampled stator current as a feedback signal for the closed loop control within the drive itself, which is called sensor less vector control.

In this mode of operation, the motor currents of each motor (connected through a drive) are continuously observed. The physical status of the motors is decoded and displayed at any time such as normal running, overloading, stalled, physical status of connection or de-connection (motors are used to connect with allied connector remotely inside the cell for easy maintenance), one phase open and mechanical problem (over vibration) etc. This online diagnosis feature reveals the motor status at running.

14. Development of sensor for monitoring speed in centrifuge

The dissolved solution after the chopped spent fuel contains fine solid particles. High speed centrifuges are used for separation of these solid particles. The presence of these particles will lead to choking of the liquid transfer lines and accumulate in the interfaces in the solvent extraction step degrading the solvent as well as creating operational difficulties. The rotor of centrifuge consists of a bowl with a shaft. There is a collet attached to the shaft which acts as a target material for speed sensing. Centrifuge is rotated by air operated turbine. The desired operating speed of the centrifuge is 18000 rpm (app). Speed is a critical parameter for centrifuge. As the speed reduces below operating speed, liquid will not get clarified and if speed increases beyond operating speed, mechanical equipment damage will occur. Hence monitoring the speed of centrifuge is mandator for smooth operation of centrifuge. Proximity sensors available in the market were used for measuring the speed previously. This sensor will produce a pulse of predefined output voltage when it comes in proximity with the target material, irrespective of the distance between them. The failure rate of the sensor's in-built electronics was high when it was used in Hot cell. The frequent failures and replacements of sensors in Hot cell caused drastic increase in man Rem expenditure of the plant. Hence a new type of sensor was planned to develop which will cater the following requirements. The sensor should be able to withstand the effect of high radiation, acid and organic fumes and maintainable via Master Slave Manipulators (MSM). The output shall be linearly varying with respect to the distance between sensor and target and also to have a reliable sensing distance up to 8mm.

The sensor developed consists of two parts: a) Sensor and b) Drive electronics. The sensor is a coil wound on suitable former and housed in proper mechanical support. The coil senses the speed and gap between the tip of Proximity Sensor and the target material based on eddy current principal. The target material is made up of magnetic permeable material (SS304L). The Drive electronics consist of electronic circuit to sense the change in magnetic flux and generate variable pulse frequency to measure the speed and linear DC voltage for measuring gap. The electronics is housed in a metallic enclosure and connected to the Proximity sensor by co-axial cable. The sensor is powered by 48V DC. The output signal from the sensor is pulsating DC.

Figure 20. Sensor & Electronics.

The sensor coil is excited by a high frequency oscillator. The target produces Eddy current. The Eddy current produces a secondary magnetic field that opposes and reduces the intensity of the original field. This interaction is called the coupling effect. The strength of the electromagnetic coupling between the sensor and target depends on the gap between them. Signal conditioning electronics senses the effect of impedance variations as the gap changes and translates them into a usable displacement signal. The pulsating DC output from the sensor was converted into a standard voltage pulse and then counted. The read-out electronics for the sensor was developed in-house to display the speed. The read-out electronics consists of an input filter board, a counter board and an HMI. The filter board eliminates the DC component of the signal; the counter board consists of a comparator and a non-retriggerable monostable multivibrator. HMI was programmed to display the speed. The speed output is made available at the plant SCADA via Ethernet (Fig. 20 and 21).

Figure 21. Display Unit.

15. Evaluation of commercial video cameras for Hot-cell Applications

Analog video cameras are generally used for visual monitoring in the hot-cells of reprocessing plants. The video camera should withstand high gamma doses and dose rates of up to thousands of Rad and Rad/h. It should also be able to withstand the corrosive effect of vapours of nitric acid and organic solvents. The weight of the

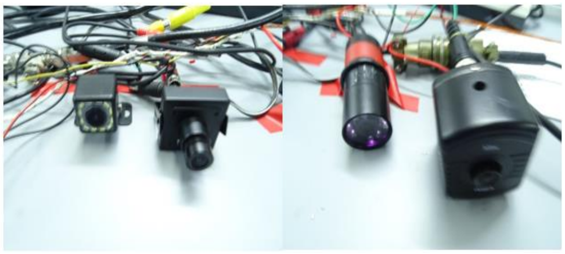

Figure 22. Photographs of CMOS colour, CCD based B & W and CCD based bullet camera.

cameras shall be low for easy handling with manipulators. Analog cameras available in the market are Black & White (B&W) or colour. The sensing technology is a Charge Coupled Device (CCD) or Complementary Metal Oxide Semiconductor (CMOS). These types of cameras (Fig. 22) are evaluated by irradiation in the gamma chamber to estimate each camera's radiation dose tolerance capability. Four analog cameras selected for evaluation are CMOS-based Colour, CCD-based B&W and two different CCD-based colour cameras. The cameras were connected to a Digital Video Recorder (DVR) and display system for a long period in the laboratory to identify any inherent faults. The cameras are placed in the gamma chamber (Fig. 23) and it was irradiated at a dose rate of 70 kR/h. The radiation dose tolerance of each camera is given below.

The CMOS colour camera CCD-based colour-I, CCD-based colour-II, CCD-based B&W camera has the better radiation dose tolerance in decreasing order. The CMOS colour camera and CCD-based bullet Camera-I are the most suitable for hot-cell applications. These cameras provide the best radiation dose tolerance.

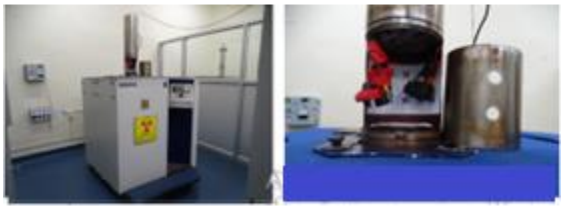

Figure 23. The Gamma chamber and four cameras are arranged in the chamber for the irradiation test.

16. Conclusion

Several types of Electrical, Electronic, and Instrumentation systems have been designed and developed in-house for the special requirements of Fast Reactor Reprocessing Plants. They have been implemented and tested to meet the challenging environmental requirements. All the systems have been successfully deployed in the Reprocessing Plants and are operating continuously and reliably.

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Design and Commissioning Aspects of Ventilation and Off–Gas Systems in Fast Reactor Fuel Reprocessing Plants

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Abstract

The main function of a radioactive building ventilation and filtration system is to protect the personnel from air-borne contamination. Ventilation and Off gas system plays a major role in a reprocessing plant as safety systems in ensuring that flow of air is always from areas of lower contamination potential to areas of higher contamination potential. This is achieved by adopting zoning concept and by maintaining areas of higher activity at a more negative pressure with respect to its preceding area.

This article enumerates the general design philosophy and principles followed in Ventilation and Off gas system for a typical Fast reactor reprocessing plant. It outlines the basis for system classification and general safety criteria as recommended by the guidelines and standards. The commissioning and readiness of all off gas systems is one of the mandatory requirements before hot runs. Towards this objective, various steps to be followed for troubleshooting and ventilation balancing for integrated commissioning of the system are presented.

Keywords: *Reprocessing, Ventilation, Off gas system, negative pressure, filters, containment, air flow*

1. Introduction

The ventilation System in Fuel Reprocessing plant is designed to keep air activity in operating areas under control, to avoid exposure through inhalation and for comfortable working environment for plant personnel. The design philosophy is based on progressive negative pressure gradient between Green, Amber and Red zones thus ensuring air flow pattern from low contamination potential areas to higher contamination potential areas. The air flow increases the efficiency of physical barriers and minimizes the potential for spread of contamination. The ventilation system thus reinforces the physical containment.

The plutonium quantities handled in Fast reactors are higher compared to thermal reactors. Hence while designing ventilation system for Fast reactor reprocessing plants, static containment of process cells is also to be considered in comparison to thermal reprocessing, where dynamic containment is generally adequate. This necessitates the leak tightness of cells, stringent filtration systems and the adoption of supply side Blowback filters.

Off–gas system is one of the main safety systems in a reprocessing plant. The function of this system is to create vacuum in the process tanks and to exhaust the vapours generated from tanks after suitable treatment and filtration.

2. System Classification

2.1. Ventilation system

The Supply-Exhaust systems are classified based on Contamination potential, Area Layout and process systems. The Factors considered for system classification are:

a) Degree of contamination potential based on zoning
b) Process systems and Plant Layout – viz. Head end, waste vault, Pu cells, U cells, Pu reconversion lab, U reconversion lab
c) One to one Supply and exhaust system – for the ease of providing interlocks between corresponding supply and exhaust fans
d) Conditioned or Non conditioned air requirements for different areas

Conditioned air is generally provided for operating areas of Process cells, Head end cells, Pu Reconversion Lab, Analytical Lab, control room and Health Physics rooms.

The corresponding AHU systems are provided nearer to these areas so that Ducting lengths are minimized.

2.2. Off–gas system

The gases and vapors generated in the process vessels and equipments during various operations in the plant are called Off-gases. Depending on their nature, the off-gases are treated before discharge. Based on the source of generation, they are classified into:

a) Dissolver Off-gas system (DOG)
b) Vessel Off-gas system (VOG)
c) Evaporator Off-gas system (EOG)
d) Glove box Off-gas system (GBOG)
e) Containment box Off-gas system (CBOG)
f) Waste vault Off-gas system (WVOG)

Scrubbing followed by de-entrainment and filtration is carried out based on the nature of the Off-gas. The classification of the Off-gas system is based on this. Since GBOG and CBOG streams do not contain much of oxides of Nitrogen and Nitric acid vapor, no scrubbing is provided in these two systems, whereas for systems like DOG, and VOG, scrubbing is provided. Each off-gas system is provided with an individual HEPA filtration system.

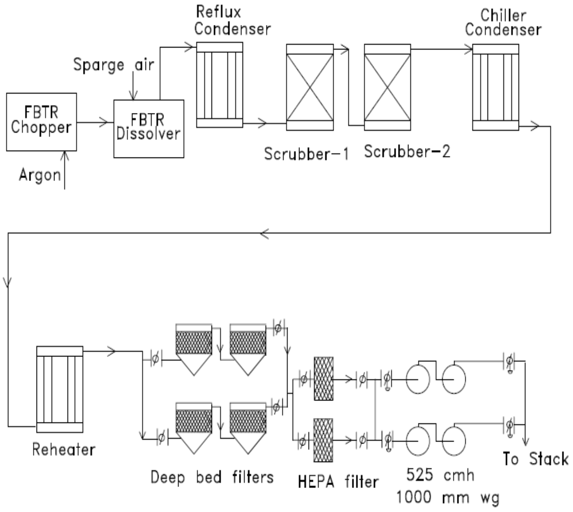

Figure 1. Typical DOG Off–gas scheme.

The different off-gas systems are designed to treat the highly contaminated off-gas streams at the source itself before mixing with the Ventilation air of Plant Exhaust system. For example, the off-gases generated during Chopping and Dissolution is treated by the Dissolver Off-gas System. The dissolver off-gas stream contains radioactive gases, particulates as well as oxides of Nitrogen. After treating the Off-gas stream by wet scrubbing and filtration, the dissolver Off-gas stream is discharged through Stack. (Fig. 1).

Wet scrubbing is followed to scrub Iodine, oxides of Nitrogen, Nitric Acid vapors and coarse particulates. The fine particulates are removed progressively by de-entrainment devices like Cyclone Separators, Deep bed filters and High Efficiency Particulate Air (HEPA) filters.

Air in the Off-gas stream will be saturated with water vapour after wet scrubbing. Both HEPA and Deep bed fiber glass wool filter are sensitive to moisture. Hence the dew point of air in Off-gas stream shall be kept below the ambient temperature. This is achieved by passing the off-gas stream through a chiller condenser and re-heater after wet scrubbing. Wherever high dust load in Off-gas stream is expected, a deep bed filter with high dust bearing capacity is provided at source in each of the Off-gas stream, since HEPA filters have low dust bearing capacity.

3. Safety Criteria and Design Philosophy

The general principles of safety criteria [1-5] followed are:

3.1. Ventilation system

a) Ventilation and Off-gas system shall comply with regulatory limits and maintain exposures to occupational workers and general public at levels that are 'as low as reasonably achievable' (ALARA).
b) The system shall be designed to confine air borne radioactive contamination as close to the source of origin as practicable and to prevent release of radioactive particulates, noxious fumes and vapors into areas occupied by plant personnel.
c) Negative pressure differential shall be maintained between the confinement zones and outside atmosphere to ensure that airflow is always from zone of lower contamination potential to zone of higher contamination potential.
d) Design shall ensure that reversal of pressure differential never takes place from higher active to lower active area.
e) Ventilation systems shall be designed so that the failure of any one component (equipment or control device) will not affect the continuous availability of the ventilation system.
f) Status of Exhaust Fans and Off-Gas Fans and Alarm indication for their failure shall be available in the Main Control Room.
g) Standby fans and filter banks shall be provided for the Ventilation Systems. Emergency power supply (Class III power supply) shall be provided to Exhaust Fans and components, including instruments and controls.

h) The ventilation systems shall be constructed of fire-resistant materials and shall include fire-resistant filters, heat and smoke detectors and alarms to restrict the spread of fires.
i) Ventilation systems shall be capable of operating during a fire in the areas they ventilate. Provision to discontinue supply air to the spaces affected by fire shall be made.
j) The ventilation systems shall be designed to withstand maximum wind velocity conditions expected in the site.
k) Components of the ventilation systems shall be designed for structural integrity to withstand the effects of design basis earthquakes.
l) The Exhaust System in Re-Conversion Glove boxes handling Plutonium in powder form shall be provided with minimum two stages of HEPA filtration prior to release to stack.
m) Air locks shall be provided, where frequent access between continuous occupancy areas and controlled access areas is envisaged and where airflow must be unidirectional.
n) Glove boxes which require opening of rear access doors (as in Reconversion Laboratory) for maintenance shall be provided with double skin (barrier). The body of the glove box shall be the primary barrier and the physical enclosure surrounding the glove boxes shall be the secondary barrier. The glove boxes and the space in between the glove boxes and secondary barrier shall be exhausted independently.

Construction and Layout

a) While selecting the material of construction for the ventilation systems, the following factors shall be considered a) Structural strength b) Corrosion resistance c) Fire resistance and d) Long operating life.
b) The design and construction shall provide for convenient inspection, maintenance and replacement of critical components. Floor drains shall be provided in Filter bank area and Stack.
c) Housings, filter mounting frames, and ducts shall be designed to withstand system pressure changes without distortion, vibration, fatigue etc.
d) Under plant black out condition, the exhaust dampers shall remain in stay put condition to ensure continuous ventilation through the stack.

3.2 Off–gas system

a) The face velocity across the opening of the dissolver shall be at least 0.5 m/s when dissolver is open to the cell to prevent diffusion from the Dissolver to Cell.
b) The release of radioactive species by air route through stack, during all plant operating conditions shall be limited to the levels stipulated by regulatory authorities.
c) The release of noxious gases shall comply with state statutes and regulations.
d) The system shall be designed to resist fire, corrosion and thermal effect.
e) Off–gases shall be treated near the point of generation to remove corrosive and radioactive gaseous species.
f) Negative pressure in the respective off-gas generating equipment shall always be maintained relative to surrounding areas under all plant operating conditions.
g) The condensate accumulation in the off-gas system shall be prevented.
h) Provision for decontamination and maintenance of off–gas system equipment with minimum personnel exposure shall be made.
i) Radiation exposure to plant personnel shall be minimized by suitably routing off–gas ducts.
j) Design provision shall be made to prevent backflow from one off–gas system to another.
k) Off–gas filter room shall be suitable for easy de-contamination.
l) Radiation monitoring and pressure drop measurement shall be provided.
m) Design provision shall exist for easy removal of the filters with minimum radiation exposure.
n) Sampling ports for testing deep bed filter efficiency shall be provided.
o) Two stages of HEPA filtration system shall be provided in the off–gas system before discharge to stack for containment box and Glove box off–gas system.

In general, for all radioactive areas, necessary air changes are provided based on contamination potential, Annual Limit on Intake (ALI) of radio-nuclides and Derived Air Concentration (DAC) as specified by regulatory authorities. To protect plant personnel and general public from air borne radioactive contamination, the exhaust air from all radioactive buildings and areas are filtered before releasing to atmosphere through a stack. Chemically toxic gases are appropriately treated before being released to the atmosphere so that the release is within the limits set by the State Pollution Control Board.

4. Ventilation Flow Model and Air changes

Different types of ventilation models are followed in reprocessing plants. In any scheme, it should be ensured that minimum air flow (0.5 m/s) between zone barriers are maintained. [2]

A typical ventilation model followed in the plant is shown in Fig. 2.

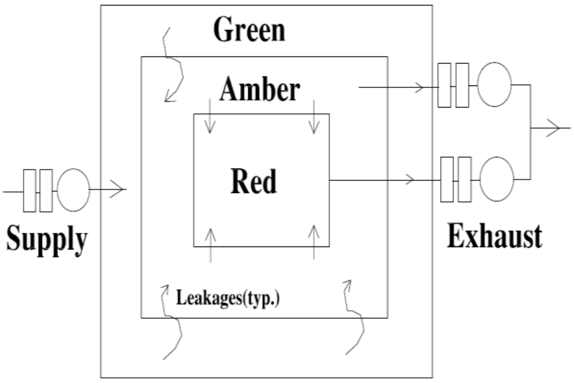

Figure 2. Typical ventilation model.

Each scheme has its inherent advantage and disadvantages. A single exhaust system for the whole plant would result in entire plant shutdown in the event of fan failure. Whereas multiple exhaust systems give reliability and flexibility in operation, but with increased costs. A designer should be able to choose a model appropriate for the plant considering all factors involved.

Table 1. Air changes and negative pressure in different Zones.

Zones	Air Changes per hr (ACPH)	Negative Pressure (mm wg)
Green Zones	2 – 5	-2 to -5
Amber Zones	5 – 10	-2 to -8
Red Zones	3 – 40	-10 to -25

For commercial Fast reactor reprocessing plants with large layouts, providing direct supply to all areas (as existing in other reprocessing plants) would result in huge duct layout, capital cost and fan sizes. In order to minimize the same, cascaded flows (or passive flows) between areas are considered for such plants.

The Guidelines and standards [2-5] have recommended following wide range of air changes and negative pressures for a typical reprocessing facility (Table 1).

Examples of Green zones are change rooms and HP rooms. Cell operating area and Reconversion area belong to Amber category and Process cells, Glove boxes are classified as Red zones.

5. Commissioning aspects and Challenges

The commissioning and readiness of all off–gas systems and the balancing of ventilation system are among the mandatory requirements before hot runs in any reprocessing plant. Balancing of the system is important to achieve both required air changes and vacuum in different areas. During pre-commissioning stage, it is pertinent to carry out thorough on-site examination of each off–gas system to ensure the ducting systems are in place and fully connected to the respective tanks/equipments.

The system corrections which may be required at this stage are classified into two types.

Connectivity errors

Commissioning modifications

The former involves correcting the design scheme errors that can arise between inside and outside cells, whereas the latter is due to the system not performing as per design intent or due to construction damages or corrosion of EP (embedded parts) etc. It is prudent and desirable that all corrections required are duly identified during acid-TBP runs and the modifications done well before taking up uranium runs. Moreover, personnel entry inside containment box/cell spaces if required, for air flow measurements should be completed before uranium runs.

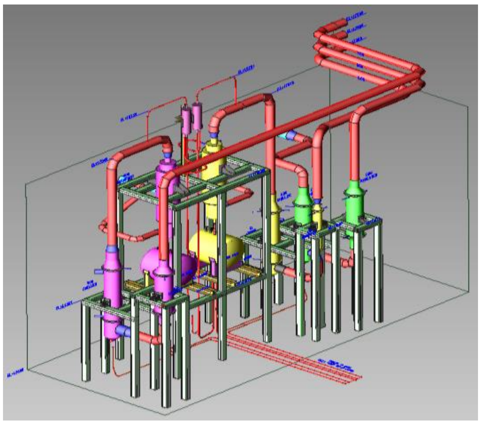

Figure 3. Layout of Scrubber, Chiller condenser and reheater.

Troubleshooting the system involves identifying leaks at various points so that desired vacuum is achieved in the tanks and equipments. Other commissioning activities which are common in initial stages include duct tracing, connectivity verification and sealant application in Deep bed filter lids etc.

The functionality of individual system equipments viz. Scrubbers, Chiller condensers and reheaters (Fig. 3) needs to be verified. In situ DOP Filter testing [4] has to be carried out for Deep bed filters and HEPA filters to ensure they are satisfactory and meeting the requirements.

Since the complete closure of containment box glasses along with MSM (Master slave manipulators) are possible only before hot runs, the integrated commissioning of containment box Off–gas system along with cell space will be carried out at this stage. The required pressure gradient between the box and cell needs to be achieved before taking up hot runs.

Fumehoods wherever located in Active labs and Reconversion labs should have a face velocity of 0.5 m/s [2,5] across the openings. The number of fumehoods and their opening sizes need to be suitably balanced to achieve the same.

Another important system is the Double skin enclosure in Reconversion labs (Fig. 4) which have to be made functional before hot runs by achieving the desired face velocity across port openings.

Figure 4. Reconversion lab Double skin system.

Ventilation balancing of the entire plant is mandatory and it involves the following general activities.

- Closure of all Doors in areas (both normal & Fire Doors)
- Identifying and arresting all undesired openings and leaks
- Operation of Supply and Exhaust fans at rated flows (Exhaust fans are always sized at higher flows than supply fans)
- Closure of specific supply grills at appropriate areas – to get the required Green to Amber flow gradients
- Closure of certain Exhaust grills/dampers to get required vacuum and flows
- Dampers throttling, wherever required
- Measurement and consolidation of air changes and vacuum obtained in all areas and process cells - to verify meeting the requirements
- To verify flow directions and gradient

6. General Design principles

In general, there is no steady arising rate (leakage) of contaminants in a Fast reactor reprocessing plant. They are built to high standards of containment and a continuous loss of material would not be acceptable. The situation would be a discrete release followed by a period of cleanup. Typically, there may be releases during the plant activities viz. waste transfer operation, bagging out of materials, maintenance etc. Following the activity, the area is cleaned and brought back to original situation.

Under these conditions, following equation is applicable for air activity decay. [6]

$C = C_o e^{-n t}$

where,

C_o = Original concentration at time zero, Bq/m^3

C = Concentration after time t hrs, Bq/m^3

n = no. of air changes per hr (ACPH)

t = time in hrs

The above equation implies that air activity of 20 DAC (Derived air concentration, Bq/m^3) in an area can be brought down to 1 DAC in 1 hr, if 3 air changes are given. Similarly, for 5 air changes, 140 DAC to 1 DAC is achieved in 1 hr.

Hence lower ACPH are effective and sufficient for larger operating plants.

The general Ducting Design principle followed is based on pressure drop of 0.8 Pa/m.

The Heat transfer Equipments in Off–gas system are designed as per conventional heat exchanger principles based on TEMA Class-C. Since, the Dissolver reflux condensers are designed with inerts, depending on the partial pressure of vapour, condensation starts from one temperature and ends at another temperature. The overall heat transfer coefficients obtained in these condensers are less and hence sizes required are bigger.

However, evaporator condensers are designed for pure vapour condensation and hence overall heat transfer coefficients are higher. Chiller condensers are always designed for 100 % saturation of air.

While designing Off–gas headers, care has to be taken to maintain proper slopes, to segregate the headers based on both activity and Pu concentration and also to provide header drains appropriately. The Off–gas ducting portions inside cells shall be of sch.40 thickness whereas sch.10 is chosen outside the cells. The overall decontamination factor (DF) achieved in Off–gas equipments shall be minimum of 10^7 to meet stack apportionment limits for the plant.

The sizing of Off–gas fans shall be done to cater both normal and abnormal conditions. In addition to the pressure drops of all equipments under normal conditions, the system should be designed for abnormal conditions e.g. Dissolver lid opening, choking of HEPA filters, Gauntlet/booting failure, Boiling of solution etc.

7. Conclusion

This article has highlighted the general design philosophy and principles to be followed in a typical Fast reactor reprocessing plant.

It also outlines the basis for system classification and general safety criteria as recommended by standards. Typical ventilation models and air changes followed are presented. For the large layout plants, cascaded flows between areas have to be adopted so as to have smaller ventilation systems with overall reduction in Exhaust flows. This approach would result in significant cost savings due to lower capital cost of fans, duct work and lower running costs. It is imperative that all safety criteria and operational issues are considered at the earliest design stage itself.

Various steps to be followed for troubleshooting and ventilation balancing are described for successful commissioning of the facility. Improved strategies for Iodine removal methods viz. Silver impregnated zeolite/alumina are to be developed for adsorbing Iodine-129 which can further reduce the environmental releases and apportionment.

Though standard design practices are followed in conventional ventilation systems, extensive scope exists in the areas of automation and filter removal. Better designs with Remotely operable Bag in Bag out filter removal systems should be implemented. This is particularly relevant for Fast reactor reprocessing where activity levels are higher resulting in significant gain in man rem exposure.

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Waste Management Aspects of the Fast Reactor Fuel Reprocessing

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Abstract

India follows a closed nuclear fuel cycle options with reprocessing of spent fuel as an integral part of the three stage nuclear programme. The closed fuel cycle is based on the vision to sustain the nuclear energy programme with better utilization of the available nuclear resources and to reduce the concerns on the radioactive wastes generated from the reactors. The closed fuel cycle depends on the reprocessing of the irradiated fuel, for providing the necessary fissile material for fast breeder reactor (FBR) programme. Reprocessing of the irradiated fuels from the FBR's with a high burnup of up to 150 GWd/t with a short cooling period, is the key for sustaining the fast breeder programme. In this case, the challenge of managing the waste from the reactors is reduced but during reprocessing operation, waste streams in different forms are generated during the recovery of fissile product. This article discusses the waste management aspects of the fast reactor fuel reprocessing including the types of waste, concerns in handling, storage, transportation and also provides the information on the treatment, conditioning, storage and final disposal strategies followed by the waste managers authorized for final disposal of these wastes.

Keywords: *closed fuel cycle, FBR, reprocessing, spent fuel, waste management.*

1. Introduction

The spent fuel discharged from the Indian Pressurised Heavy Water Reactors (PHWRs) is reprocessed to recover fissile material and to provide necessary fuel for the FBRs, which would be otherwise disposed of as a waste. The spent fuel from the FBRs is reprocessed to continue the efficient use of nuclear materials. During FBR spent fuel reprocessing different types of wastes are generated. The radioactive wastes need to be safely handled due to the radioactivity. This is the major reason for prioritizing the waste management aspects to sustain the nuclear energy programme. The forms of waste generated from the fast reactor fuel cycle are more or less as that of thermal reactor fuel cycle. The main challenge is the high burnup of the spent fuel and the processing of short cooled fuel which causes the high level of radioactive fission products present in the waste (Fig. 1) and the increase in the waste volume generation.

CORAL (Compact facility for Reprocessing Advanced fuels in Lead Mini Cell) was a pilot facility which has been in operation for the last two decades. It has been reprocessing FBTR spent fuel and provided the hands-on experience on the reprocessing of fast reactor fuel reprocessing along with the knowledge on waste generated during the operation including the types/ forms of waste generated from a reprocessing plant of its kind. This provided the confidence in designing the waste management system for the Demonstration Fast reactor fuel Reprocessing Plant (DFRP). DFRP is successfully hot commissioned in the recent past.

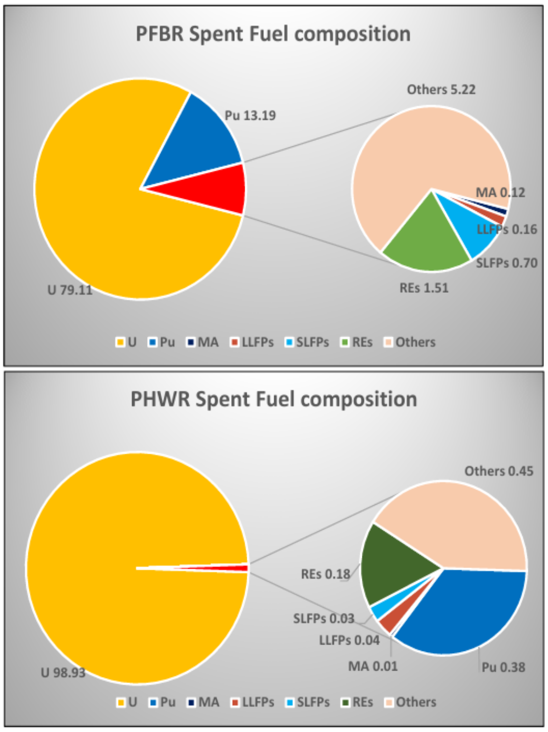

Figure 1. Spent fuel composition (%) for different fuels.

The waste management philosophy followed is framed mainly with the objective of protection of the human health and the environment.

Radioactive Waste management ensures that the release of radioactivity in any form to the nature never exceeds the permissible limits as per the radiation protection principles. Accurate assaying techniques are necessary for different types of waste to enable the necessary treatment for a better waste management from Fast Reactor fuel reprocessing. Waste management is planned from the initial design stage to the final stage of storage or disposal of the different waste generated in the plant. The best practices for minimization, treatment of waste at the source of generation, storage and disposal to effect safe and proper management of the waste are followed.

Technologies for the retrieval of the actinides and the valuable fission products from fuel reprocessing waste streams, for recycling them back to the reactor or to divert them for various societal applications is being developed and implemented. The use of radionuclides for societal uses postpones the management of waste but ultimate disposal upon the end of usefulness is being considered by the waste managers. In this context, waste management aspects ensure a complete control of the wastes till the natural radioactivity decay reduce the activity to a safe level.

2. Radioactive wastes from aqueous reprocessing

2.1. Process of Waste generation

Aqueous reprocessing of spent fuel follows chopping of fuel pins, dissolving the spent fuel in nitric acid medium and solvent extraction which results in fission products (FP) remaining in the aqueous solution (raffinate) leaving it as a highly radioactive solution. The raffinate is vitrified to produce the waste in a solid matrix. Amount of the vitrified waste depends on the fission product content in the waste. Solid wastes are also generated during the operation and maintenance. The off-gases released from the various processes, exhausted from the tanks, equipment, containment systems and the exhaust from the areas comprises the gaseous waste.

2.2. Classification of Wastes

For management of the waste, classification of the waste is of paramount importance. The radioactive wastes generated during reprocessing are classified based on the physical form, source of generation, chemical nature, radionuclide content and the nature of conditioning for storage or disposal. The radioactive concentrations differ in the streams based on the source of generation. Broad classification is based on their physical form as Solid, Liquid and Gaseous waste. Table. 1 gives the details of the different types of waste under the solid, liquid and gaseous forms and their classification based on their generation in the plant.

Solid Waste:

Solid waste are wastes that arises during operation, maintenance and repair. They are classified as compressible/ non-compressible, combustible and non-treatable. Further classification is based on the amount of radioactivity and type of radionuclide present in the waste.

Liquid Waste:

Liquid wastes are generally classified as exempted, low level, intermediate level or high level based on the activity levels, based on chemical properties – as process or analytical waste, as organic or aqueous waste and based on the radionuclides present.

Gaseous waste:

Gaseous wastes generated in reprocessing plants are of low activity. The concentration of the contamination in the air in handling of radionuclides during operation and maintenance, depends on the type of operation and amount of material handled. Based on this gaseous waste are classified as exhaust gases from areas and off–gases from process operation. Further the exhaust from cells, containment box and Glove box are handled separately and the off–gases from dissolver, evaporator and process vessels are handled separately before discharge. The off–gases after the treatment processes in the plant are discharged through a stack.

2.3. Categorization of waste

Categorization of different classified wastes (column f of Table 1, based on their activity concentration taking the experience from the operating plants, helps in segregation, undertaking treatment methods, storage and final disposal. Table 2 and 3 gives the categorization of Solid and Liquid waste, based on their radioactivity concentration.

Table 1. Classification of wastes from Fast Reactor Fuel Reprocessing.

<table>
<tr><td colspan="8">Classification of Radioactive wastes from reprocessing plant</td></tr>
<tr><th>Physical Form</th><th colspan="2">Chemical Nature</th><th>Radioactive Nature</th><th colspan="2">Type</th><th>Waste - Example</th><th>Category</th></tr>
<tr><th>(a)</th><th colspan="2">(b)</th><th>(c)</th><th colspan="2">(d)</th><th>(e)</th><th>(f)</th></tr>
<tr><td rowspan="8">Solid Miscellaneous</td><td colspan="2" rowspan="4">NIL</td><td rowspan="3">Non – α, Low active</td><td colspan="2">Combustible</td><td>Paper, Cotton waste, clothes</td><td>Cat I</td></tr>
<tr><td colspan="2">Compactable</td><td>Rubber, Plastic, PVC sheet, Gloves, laboratory glass ware, HEPA filters</td><td>Cat I</td></tr>
<tr><td colspan="2">Non-Treatable (Metal)</td><td>Metallic parts, Tools, Analytical metal gadgets, Fuel handling gadgets</td><td>Cat I</td></tr>
<tr><td>Non – α, High active</td><td colspan="2">Non-Treatable (Off–gas Filters)</td><td>Deep Bed Filters of off–gas of High-level Waste storage</td><td>Cat III</td></tr>
<tr><td colspan="2" rowspan="4">NIL</td><td>α, Low active</td><td colspan="2">Compactable</td><td>Glove box waste – Gauntlet, Gloves, Plastic, Glassware, HEPA filters</td><td>Cat IV</td></tr>
<tr><td rowspan="3">α, High active</td><td colspan="2">Non-Treatable (metal)</td><td>Centrifugal extractor bowls, Centrifuge Bowl, SS bearings, Other Metal Tools, Sampling gadgets like Analytical extraction vials, tips</td><td>Cat IV</td></tr>
<tr><td colspan="2">Non-Treatable (Off–gas Filters)</td><td>Deep Bed Filters of off–gas of Process facility</td><td>Cat IV</td></tr>
<tr><td colspan="2">Non-Treatable (Dissolver cell metal waste)</td><td>Chopper Tools, Centrifuge bowls, Electrodes, SS bearings, Other Metal Tools</td><td>Cat IV</td></tr>
<tr><td rowspan="7">Liquid</td><td rowspan="5">Process</td><td rowspan="4">Aqueous</td><td rowspan="2">Non – α</td><td rowspan="2">LLW</td><td>Non-Detergent</td><td>Condensate - Steam, Waste Evaporator</td><td rowspan="2">Cat II</td></tr>
<tr><td>Detergent</td><td>Sink & Shower Decontamination</td></tr>
<tr><td>α</td><td colspan="2">ILW</td><td>Raffinate Evap Condensate, Scrub liquid</td><td>Cat IV</td></tr>
<tr><td>α</td><td colspan="2">HLW</td><td>Raffinate from extraction cycles</td><td>Cat V</td></tr>
<tr><td>Organic</td><td>α</td><td colspan="2">OLW</td><td>Lean Organic from extraction cycles</td><td>Cat IV</td></tr>
<tr><td rowspan="2">Analytical</td><td>Aqueous</td><td>α</td><td colspan="2">Special Aqueous</td><td>Aqueous waste in generated during analytical operation</td><td>Cat IV</td></tr>
<tr><td>Organic</td><td>α</td><td colspan="2">Special Organic</td><td>Alcoholic waste, TTA/ TOPO waste generated during analytical operation</td><td>Cat IV</td></tr>
<tr><td>Gas</td><td colspan="2">NIL</td><td>Non – α</td><td colspan="2">Cell / GB Exhaust, off–gas from process</td><td>Exhaust from Cells, Containment Box, Glove boxes Exhaust & Treated off–gas from Dissolver system, Evaporator System and form Vessels.</td><td>Cat II</td></tr>
</table>

Table 2. Solid Waste categorization.

Type	**AERB Category**	**β activity max (Bq/g)**	**α activity max (Bq/g)**	**Cat**
(a)	**(b)**	**(c)**	**(d)**	**(e)**
Non – α, Low active				
Combustible	B1A3	1×10^4	2×10^3	I
Compactable				I
Non-Treatable (Metal)				I
Non – α, High active				
Non-Treatable (Off–gas Filters)	B3A3	$>2.5 \times 10^6$	2×10^3	III
α, Low active				
Compactable	B2A4	9×10^4	$> 4 \times 10^3$	IV
Non-Treatable (metal)	B3A4	2×10^6		IV
Non-Treatable (Off–gas Filters)	B3A4	2×10^6		IV
α, High active				
Non-Treatable (Dissolver cell metal waste)	B4A4	$>2.5 \times 10^6$	$> 4 \times 10^3$	IV

Table 3. Liquid Waste categorization.

Chemical Nature	**Radioactive Nature**	**Type**	**AERB Category**	**βγ Bq/ml**	**α Bq/ml**	**Cat**
Process Waste						
Aqueous	Non – α	LLW	B2A2	37	3.7	II
	α	HLW	B4A3	3.7×10^6	3.7×10^4	IV
	α		B5A3	$>3.7 \times 10^8$	$>3.7 \times 10^4$	V
Organic	α	OLW	B4A3	3.7×10^5	3.7×10^4	IV
Analytical Waste						
Aqueous	α	Special	B4A3	3.7×10^4	3.7×10^3	IV
Organic	α	Special	B4A3	3.7×10^4	3.7×10^3	IV

3. Waste Management philosophy

The solid and liquid wastes generated in any reprocessing plant are segregated at source and collected and stored temporarily at the designated location at site. The treatment for the final disposal is by the waste management facility. Only the gaseous waste after treatment is discharged directly to the environment within the permissible levels.

3.1. Solid waste

Low level active solid wastes that are non-alpha (Ref. **Non–α, Low active** in Table 2.) are segregated, packed in PVC covers, sealed and appropriately tagged are collected from the source of generation and kept in the Mild steel (MS) drums, for shipment in the transit area for storing wastes which has facility for material handling. High active non-alpha wastes (Ref. **Non – α, High active** in Table 2.) are packed in a SS Container, in turn in a shielded cask for shipment to the transfer point. MS drums/ container will be transferred to Near surface disposal facilities (NSDF) after measuring the dose on the surface of the drum/ container to be less than 200 mR/hr and assaying the waste for ensuring the α activity levels to be less than 4000 Bq/g. Monitoring and periodical radiation surveillance in the interim storage is to be carried out. Provision for decontamination and approach area for loading the waste into the truck are of importance in the transit area meant for storage.

Treatable alpha bearing, low βγ wastes (Ref. **α, Low active** in Table 2.) are doubly sealed in PVC covers sealed and appropriately tagged, transferred to the designated interim storage and placed in Stainless steel (SS) containers. Periodic surveillance for the radiological status of the interim storage area needs to be carried out. High active metallic wastes, Hulls and Non-treatable alpha bearing wastes (Ref. **α, High active** in Table 2.) having more than 200 mR/hr radiation field are stored inside the Cell and collected in SS containers, loaded in a **α - tight,** container and transferred in a shielded cask to the appropriate waste managers. In-cell material handling provisions like remote handling gadgets, master slave manipulators and in-cell cranes are required by design for these solid wastes.

3.2. Liquid waste

Liquid wastes are segregated at source as potentially active and low active wastes collected as per the category in the Holdup tanks at the designated areas for the storage. Process waste is collected at the source and on sampling and analysis for the meeting of the radioactivity levels only it can be routed to the storage tank. Decontamination waste is routed to the storage tank through Holdup tank. Low Level liquid wastes are stored together in delay tank and sent to waste management facility for treatment and disposal (Ref. **Process Waste - Aqueous, Non – α, LLW** in Table 3.). High level aqueous and organic waste (Ref. **Process Waste –Aqueous/Organic, α, HLW/OLW** in Table 3.) needs to be stored in underground storage tanks for delay and decay of the activity levels as per the requirement of waste treatment management methods. The tanks are with cooling provision for extracting the heat from the waste. The categorization enables suitable processes to be adopted in the waste management plants. The main concern in the HLW management is the transfer of the waste without leaching of the waste in case of a breach. This is achieved by design in which the waste is transferred through a primary pipe that is enclosed in a secondary outer pipe isolated from the environment by a stainless steel box that is placed in an underground RCC trench. The high level wastes after storage period, on analysis for meeting the radioactivity levels are transferred to waste management facility for further treatment and disposal.

The analytical wastes (Ref. **Analytical -α, Special waste** in Table 3.) are collected and on sampling and analysis for the meeting of the radioactivity levels are stored in container in a designated area and are transferred to the waste management facility in truck.

3.3. Gaseous waste

Based on the source of generation, the gaseous waste generated during the process are treated and filtered before discharging to the environment. The exhaust air from the various areas is discharged after filtration through HEPA filters to remove radioactive particulates. The off–gases from the process are handled and treated separately based on the source of generation. The exhaust air and off–gases are finally discharged at the discharge point, though the stack. The height of the stack is decided such that during normal operation of the plant the dose to the members of the public at the site boundary shall be within the regulatory cleared dose apportionment limits. The activity of effluents released to the environment is continuously monitored by radiation monitors provided in the stack.

Table 4. Solid Waste treatment.

Type	Waste - Example	Treatment
(a)	**(b)**	**(c)**
Non – α, Low active Cat I		
Combustible	Paper, Cotton waste, clothes	Incineration
Compactable	Rubber, Plastic, PVC sheet, Gloves, HEPA filters	Compaction /Drum pelletization by Bailing press, Melt densification
α, Low active Cat IV		
Compactable	Glove box waste – Gauntlet, Gloves, Plastic, Glassware, HEPA filters	Compaction/ Drum pelletization

4. Overview of waste Treatment & conditioning for final disposal

The solid and liquid wastes generated in any reprocessing plant is subjected to appropriate treatment methods. As the radioactive wastes are of different types, treatment processes developed and in practice are of different kind based on the type of the waste. This step is required to further segregate the waste and to decide on the disposal methods as radioactive waste or for disposal as non-radioactive waste for the streams when the content of radioactivity is within the regulatory limits.

4.1. Solid waste

Radioactive solid waste treatment includes volume reduction by methods like compaction; incineration and melt densification (Table 4). By the way of combustion, a volume reduction of about 50 and by compaction by bailing process volume reduction of about is 5 is obtained.

The main challenge in handling and treatment of fast reactor fuel reprocessing solid waste is the associate α activity in the waste. One of the emerging technologies is underway for the treatment for high active metallic hulls is supercritical extraction process which enables the removal of radioactivity by using supercritical fluids at controlled conditions as extracting solvents.

4.2. Liquid Waste

The low-level liquid wastes require treatment using different techniques to reduce their activity levels to the permissible levels authorized for discharge (Table 5). Treatments like evaporation to concentrate the waste and techniques like ion exchange, chemical precipitation and membrane processes to remove the radioactivity are used separately or in combination to achieve the required treatment. Acid digestion of waste followed by evaporation reduces the waste volumes significantly. High level liquid wastes on volume reduction by evaporation, is stored in stainless-steel tanks with cooling provision. After the interim storage, the HLW conditioning is by a process called vitrification in which the waste is immobilized in a stable and inert glass matrix and the conditioned waste is stored in an interim continuously cooled retrievable storage space till its disposal in deep geological repositories.

Table 5. Liquid Waste treatment.

Type			Waste - Example	Treatment
(a)			**(b)**	**(c)**
Maximizing low level waste volume and minimizing the high level waste volume				
Aqueous	LLW Cat II	Non-Detergent	Condensate - Steam, Waste Evaporator	Neutralization, Chemical Treatment followed by Ion exchange, cementation
		Detergent	Sink & Shower Decontamination	Neutralization, Ion exchange, Dilute and disperse
	ILW Cat IV		Raffinate Evaporator Condensate, Scrub liq	Evaporation with de-nitration
	HLW Cat V		Raffinate from extraction cycles	Vitrification
Organic	OLW Cat IV		Lean Organic from extraction cycles	Solvent wash treatment for recycling, Alkaline hydrolysis

Distillation technique for organic waste treatment is the area in research. Organic liquid wastes on chemical and radiolytical degradation during process, are treated by alkaline hydrolysis process and the resultant diluent free of organic and radioactivity is subjected to incineration and the heavier bottom aqueous from the process is immobilized in cement. The vitrification process with the associated melter technology and research on the matrix are well developed by the waste management facilities.

4.3. Gaseous waste

Off–gas systems to treat the gaseous waste in order to limit the discharge of activity through stack route is available in the reprocessing facilities (Table 6). The off–gas system and the associated equipment in the system are selected based on the source of generation of the gaseous waste, specific activity, type of activity, particulate density, radioisotope present and its concentration. Wet scrubbers, chiller condensers, demisters are provided for gas cleaning as the gases generated are in contact with liquid streams of the process. Final filtration for particulate radionuclides is achieved by high efficiency particulate air (HEPA) filters. Deep bed filters and reheaters are provided in the upstream of HEPA to avoid clogging of HEPA filters due to condensation of water vapour in the gaseous stream. The gaseous wastes are continuously monitored to ensure that there is no unintended release of radioactivity, by stack sampling.

Table 6. Gaseous Waste treatment.

Origin of waste	Treatment
(a)	(b)
Process equipment	Cyclone separation, Scrubbing, Humidity adjustment, Filtration (separate system for equipment and vessel based on its nature)
Process vessels	
Exhaust of Glove boxes/ /containment	HEPA Filtration

5. Final Disposal strategies

Dilute and disperse, Concentrate and contain, Delay and Decay are the strategies followed for final disposal (Table 7).

The dilute stream from the Low-level liquid/ Intermediate level waste treatment for removal of radionuclides to the level possible, needs be later diluted and dispersed in water bodies. The concentrated stream/ solids with higher activity are immobilized in cement matrix before disposal in near surface disposal RC trenches. Vitrified high level liquid waste in canisters with a surface dose >50 R/hr are stored in an interim storage and later needs to be disposed in Deep geological repository (DGR).

α bearing solid waste with α activity< 4000 Bq/g are stored in RC trenches and if the α activity levels are more than that, the waste is stored in Tile holes (Table 7).

Non α solid waste with low βγ activity with surface dose < 0.2 R/hr are stored in stone lined earth trenches, between 0.2 R/hr to 50 R/hr is disposed in RCC trenches and those with high βγ activity with surface dose >50 R/hr are stored in Tile holes. Reprocessing facility or any reactor site has a near surface disposal facility co-located in the site (Table 7).

Table 7. Waste Disposal strategies.

Surface dose/ activity R/hr	Waste Type	Disposal strategy
(a)	(b)	(c)
0.2 R/hr	Paper, rubber, concrete, cotton	Stone lined earth trenches
2 R/hr	Filters, Contaminated equipment,	RC trenches
2-50 R/hr	Concentrates/ solids from treatment	RC trenches
>50 R/hr	Highly contaminated equipment/ metal parts, High level solid from treatment	Tile holes
α activity < 4000 Bq/g	Solidified alpha bearing waste	RC trench for βγ < 50 R/hr Tile hole for βγ > 50 R/hr
α activity > 4000 Bq/g		Tile hole as interim storage

6. Challenges in Waste Management

The research and development in waste management is the area in which no international technical support is available. Indigenously developed process and technology are available based on the research and development works in Bhabha Atomic Research Center at Trombay and waste management plants based on this are operational at Trombay, Tarapur and Kalpakkam to cater the needs of the wastes arising from spent fuel reprocessing facilities. The reprocessing and associated waste management facilities are planned to be made operational to meet the demands of the increase in the capacity of nuclear power. Further improvement in the technologies, material development, development of automated remote handling gadgets, selection of disposal site with long-term surveillance plan, studies on safety & environmental impact and to address the public concern for sustainability of the nuclear industry are the main challenge of the waste management aspects. New processes are designed to reduce the waste discharge and to recover the valuable radioisotopes at large scale as resource for societal applications. This reduces the environmental impact due to the radioactive waste. Waste management approaches and processes for the need of wastes from the future advanced fuel cycles for fast breeder reactors and from the non-aqueous fuel reprocessing are being developed.

7. Role of partitioning technology and waste management aspect

Partitioning technology is a promising back-end technology for reducing the burden of the waste management by the following ways.

i. Increasing the recovery of U & Pu and avoiding the loss of U & Pu in waste streams.
ii. Recovery of Long-lived fission products and minor actinides from HLW to reduce the radiotoxicity of the resultant waste and to reduce the hazard potential thus lowering the monitored repository life.
iii. Reduce the radiotoxicity of Long-lived fission products and minor actinides by transmutation.
iv. Reduces the interim storage and repository requirement.
v. Processes with assured reduction in secondary waste volume
vi. Simplifies reprocessing and radioactive waste management by removing the gamma/ heat producing short live fission products like Cs and Sr and using them as radiation sources in medicine and industrial applications.
vii. Removal of Neptunium for use in space technology
viii. Removal of problematic element in the vitrification processes like Ruthenium.

8. Waste Assaying Techniques

The solid waste drums have to be characterized, before storage and disposal as per the regulatory requirement. The allowable limit for disposal of non-alpha bearing solid active waste is α < 4000 Bq/g. The waste with α > 4000 Bq/g are considered as alpha bearing solid active waste which needs to be stored at site for final ultimate disposal in DGRs. Two methods developed for solid waste assaying are 1) Gamma spectrometry 2) Neutron counting technique like active neutron interrogation or passive neutron counting. The type of the assaying technique used depends mainly on the source of the waste drum.

a. __Drum Monitoring__

i. __WASCT__

Waste Assay Scanning Computer Tomography (WAST) is a method of segmented gamma scanning of the waste drums. It is based on the Gamma Spectrometry estimation method based on the gamma signature of the nuclear material of interest like Pu isotopes, U-235 and Am-241. For assay of Plutonium it uses the gamma signature of Pu 239.

Figure 2. WASCT system for assaying solid waste.

This system is ideal for use of wastes containing low β, γ activity. High purity Germanium (HPGe) gamma detector is used for fission product quantification. This system consists of a drum rotating mechanism with a provision to move the drum up down precisely (Fig. 2). This technique is ideally suited for wastes generated from glove boxes.

ii. __Passive neutron-based monitoring__

Passive neutron-based monitoring is based on the counting of the passive neutrons emitted from the spontaneous fission of the Pu isotopes in the waste assayed. This system is ideal for use of wastes containing medium β, γ activity with a surface radiation level of 2 R/hr on the drum and low alpha content (Fig. 3).

Figure 3. Passive Neutron counting solid waste drum.

b. <u>Hull Monitoring</u>

Hulls are SS material of spent fuel pin that are left behind in the dissolution step of aqueous fuel reprocessing and is a high active solid waste. This waste has to be assayed for the amount of alpha content left behind after dissolution in the chopped clad material. Even though the amount left behind is less quantity, it would have effect on the material balance in total amount considering the capacity of the plant.

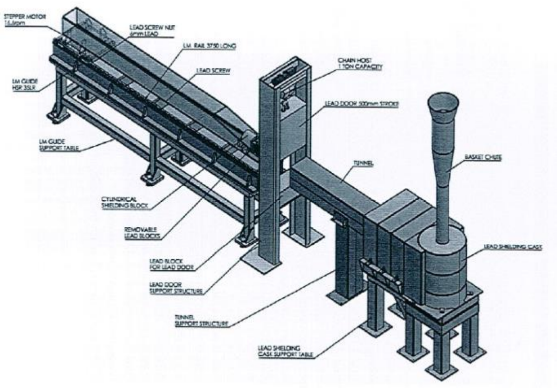

Figure 4. Hull monitoring system.

Gamma spectrometric technique is used for the assay of Pu in the hulls by indirect technique. The method developed uses the gamma signature of Ce-144 to quantify the Pu -239. As Ce is chemically similar in nature to Pu, the Ce-144 to Pu ratio in the hulls will also be similar. The main challenge in this technique is that the technique measures the activity of Ce in the high background activity of spent fuel. Collimators are used to reduce the background. A hull monitoring system is shown in Figure 4.

9. Research in waste management aspects

Research and technology development in the waste management field is facing new horizons and is always having a scope for improvement. The following are the areas in which research and development can be thought of, initiated or continued for achieving the long-term goals in the waste management aspects.

a) Partitioning technologies and process development that can be economically adopted in plant scale, for the waste from fast reactor fuel reprocessing plants.
b) Reliable assaying techniques for more quantitative results on the radionuclide present in the waste.
c) Research on promising alternative non-aqueous processes and the associated waste management technologies

10. Conclusion and future Outlook

Waste management practices is one of the important aspects in sustaining the nuclear industry in any country. India has decades of experience in the field since the earlier stages of nuclear programme. The waste management facilities have pioneered the classification, categorization, process development and conditioning technology for the different types of waste for the final disposal of the waste. The scope for research on process and technology for the future fast rector fuel cycles and non-aqueous reprocessing wastes are being addressed. The waste management at the facility level has importance by continuous research on the development of processes, flow sheet studies for better recovery of fissile material and reducing its loss in the waste and plant scale recovery of the radionuclides of societal importance. Like the demonstration facilities for separation of radionuclide from PHWR spent fuel reprocessing waste, scaled up demonstration facilities for partitioning of the valuable radionuclides from the fast reactor fuel reprocessing wastes is an immediate need for gaining experience and for reducing the heat and the long-term burden in the final waste. The consideration on the cost effectiveness of the process and technologies adopted for waste management, per unit of the nuclear power production is important as the overall unit cost of the power depends on all the components.

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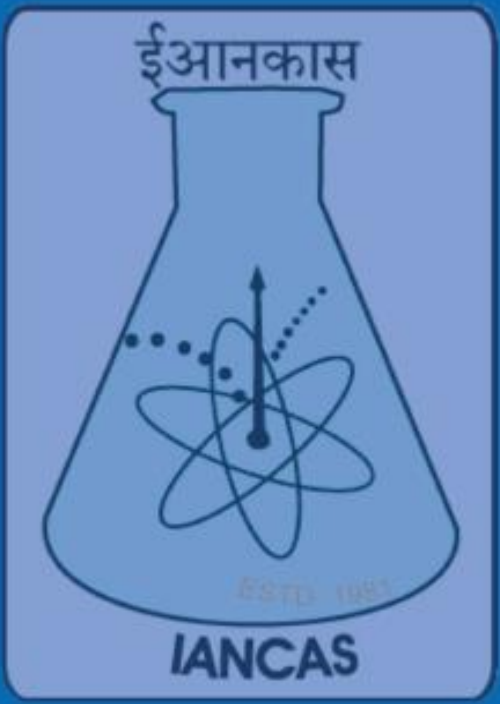

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