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Development of solvent extraction scheme for reprocessing of advanced heavy water reactor spent fuel using *N,N*-Dihexyl octanamide as extractant

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ABSTRACT

Distribution studies on Th(IV), Pa(V), U(VI) and Pu(IV) were carried out employing tri-*n*-butyl phosphate (TBP) and *N,N*-dihexyl octanamide (DHOA) as extractants under proposed advanced heavy water reactor (AHWR) spent fuel feed conditions. DHOA was found promising for selective extraction of U/Pu over Pa and Th as compared to TBP. Both Pu and Th formed trisolvated species with DHOA at 4 M HNO_3. Batch solvent extraction experiments were carried out for selective extraction of U and Pu from simulated AHWR spent fuel dissolver solution using 0.36 M DHOA and 0.18 M TBP dissolved in *n*-dodecane as solvents. Quantitative extraction of uranium and plutonium was achieved under experimental conditions leaving bulk of thorium in the raffinate. The co-extracted thorium from the organic phase was proposed to be scrubbed using 4 M HNO_3. A series of non-salt forming reductants were evaluated for their suitability for the partitioning of plutonium. Uranium from the Pu lean organic phase was proposed to be stripped using 0.01M HNO_3. The extraction behavior of fission products was also examined under the simulated AHWR spent fuel feed solution. Based on these observations, a reprocessing scheme has been proposed for the reprocessing of three component U, Pu and Th system arising out of the irradiated (Th, Pu)O_2 pins.

Keywords: Reprocessing; Uranium; Plutonium; Dialkyl amide; AHWR; TBP

1. Introduction

Advanced heavy water reactor (AHWR) is being developed in India with the specific aim of utilizing thorium for power generation. This reactor will serve as a predecessor to the third-stage reactors which will be based on Th-^{233}U fuel cycle and will provide the much needed vital information to initiate this fuel cycle. The initial core will be made up of entirely (Th, ^{239}Pu) MOX fuel assemblies, each being made of 54 fuel pins. Progressively, the ^{233}U bred in the (Th, ^{239}Pu) MOX fuel pin will be recovered and recycled as (Th, ^{233}U) MOX. At equilibrium, the core of AHWR will consist of composite fuel assemblies each having 24 nos. of (Th, ^{239}Pu) MOX pins and 30 nos. of (Th, ^{233}U) MOX pins. AHWR spent fuel adds new dimension to reprocessing by the presence of Pu in the spent fuel. The spent fuel cluster before reprocessing would undergo disassembly for segregation of (Th, Pu)O_2 and (Th, ^{233}U)O_2 pins. The (Th, ^{233}U)O_2 pins will undergo a two stream reprocessing process i.e., separation of thorium and uranium whereas the (Th, Pu)O_2 pins will require

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a three stream reprocessing i.e., separation of thorium, uranium and plutonium [1]. At the reprocessing facility, the pins containing (Th, Pu)O_2 MOX and pins containing (Th, ^{233}U)O_2 MOX will need to be processed in separate set of cycles. This requires additional provisions in the reprocessing plants and invokes the integration of PUREX and THOREX processes in some combination employing tri-*n*-butyl phosphate (TBP) as an extractant. However, separation scientists identified certain problems with the use of TBP as extractant viz. third-phase formation and low separation factor (SF) values of U(VI) and Pu(IV) over Th, as well as poor decontamination factor (DF) values of U and Pu with respect to fission products [2].

In this context, Siddall reported that in high radiation fields or other drastic conditions, the amides have an outstanding advantage for solvent extraction over TBP and all other similar compounds. The primary radiolytic and hydrolytic fragments from the amides should be amines and carboxylic acids. Rather large amounts of these fragments should be tolerable without serious interference in the process. There are no deleterious fragments like dibutyl phosphoric acid (DBP) that arises from TBP. In that sense, the amides should have a very high tolerance to radiation, even though they prove not to be more radiation resistant [3]. Musikas et al. further substantiated these observations on the completely incinerable *N,N*-dialkyl amides as alternative extractants to TBP [4–6].

Our group at Radiochemistry Division, BARC has been actively pursuing an R&D program to identify the promising alternatives to TBP in the PUREX as well as in the THOREX process. Extensive work has been carried out on completely incinerable, high molecular weight *N,N*-dialkyl aliphatic amides as potential alternate extractants for actinides [7]. Apart from their desirable extraction behaviour, amides offer the possibility of tuning their physico-chemical properties by the judicious choice of the alkyl group. Whereas straight chain *N,N*-dihexyl octanamide (DHOA) was found suitable for reprocessing of uranium based spent fuel; branched chain *N,N*-di(2-ethylhexyl) isobutyramide (D2EHIBA) was distinctly better than TBP for the reprocessing of (Th-^{233}U) based fuels [8–10]. However, the latter had a limitation for the reprocessing of three component spent fuels involving thorium, uranium and plutonium essentially due to poor extraction of plutonium [11–13].

In this context, DHOA was evaluated as an alternative extractant to TBP for the reprocessing of three component AHWR spent fuels [14]. In this work, efforts have been made to optimize the conditions for the selective extraction of ^{233}U and Pu over Pa, Th from simulated dissolved AHWR spent fuel solution. Based on the batch extraction data, a scheme has been proposed for the reprocessing of AHWR fuel.

2. Experimental

2.1. Materials

DHOA was synthesized in house as per the procedure described elsewhere [7]. TBP and *n*-dodecane used in the present work were of analytical reagent grade. ^{233}U and Pu (mainly ^{239}Pu) tracers (~10^{-4} M) were purified by anion exchange and by HTTA (2-theonyltrifluoroacetone) based solvent extraction procedures [15,16]. Their purity was checked by alpha and gamma ray spectrometry. Pu valency in the aqueous phase was adjusted and maintained in the tetravalent state by the addition of $NaNO_2$. ^{233}Pa was separated from irradiated ^{232}Th (in APSARA reactor) and from ^{237}Np (as a decay product) by a radiochemical technique using diisobutyl carbinol (DIBC) as extractant [17]. Its purity was checked by gamma ray spectrometry. Thorium (nitrate form) obtained from Indian Rare Earths Limited, Mumbai, was used wherever necessary.

0.18 M, 0.36 M DHOA and 0.18 M TBP stock solutions were prepared by dissolving required weight of the extractant in *n*-dodecane. 1–4 M HNO_3 solutions were evaluated for scrubbing of Th from the loaded organic phases. 0.5 M solutions of different reductants/complexants viz. hydroxyl amine nitrate (HAN), diethyl hydroxyl amine (DEHAN), hydroxyurea (HU), acetohydroxamic acid (AHA), and acetaldoxime (AOX) dissolved in 2.0 M HNO_3 were used for the partitioning of Pu from the loaded organic phases.

2.2. Preparation of simulated AHWR feed solution

Simulated AHWR feed solution was prepared by mixing appropriate quantities of U, Pu and Th from their stock solutions so as to yield their respective final concentrations as 2 g/l, 2 g/l, and 100 g/l at 3.5 M HNO_3 respectively [18]. Required quantities of $Al(NO_3)_3$ and of concentrated HF were also added such that their concentrations were 0.1 M and 0.03 M, respectively.

2.3. Coating of glasswares

5% solution of dimethyl dichlorosilane (DMDCS), in toluene was used for coating inner surface of glass tubes used during Pa extraction studies to minimize its loss due to sorption on the walls. The solution was then drained out after 3 h and the tubes dried prior to their use for the solvent extraction studies [19].

2.4. Solvent extraction studies

Equal volumes (0.5–1.0 ml) of the pre-equilibrated organic phases and the aqueous phases under desired conditions were kept for equilibration in a water bath for 30 min at 25°C. Distribution ratio of metal ions (D_M) is defined as the ratio of concentration of metal ions (M: Th, Pa, U, Np, Pu, and Am, expressed in terms of radioactivity) in the organic phase to that in the aqueous phase. The measurement of radioactivity was carried out by liquid scintillation counting (^{233}U, Pu) or by gamma ray spectrometry (^{233}Pa, ^{237}Np, ^{241}Am). Thorium concentration in the organic and aqueous phases was determined by ethylenediamine tetraacetic acid (EDTA) complexometric titration. The correction for counts in both the phases due to Th and its decay products was done by performing experiments under identical conditions using natural thorium solutions. Material balance and the reproducibility of the distribution data was within error limits (±5%).

3. Results and discussion

3.1. Distribution behaviour of Th, U and Pu

Table 1 compares the distribution ratio values of U, Pu and Th under earlier proposed THOREX feed conditions (200 g/l Th + 0.03 M F^- + 0.1 M $Al(NO_3)_3$ at 4 M HNO_3) for 0.18 M (5%) and 0.36 M concentrations of TBP and DHOA. Though, D_{Pu} for DHOA is similar to that for 0.18 M TBP, the lower D_{Th} values result in improved Separation Factor (SF) values with DHOA. It is evident that 0.18 M DHOA offers better SF values for both U and Pu over Th, yet the low D values for U and Pu (D_U: 1.1 and D_{Pu}: 1.3) are a limitation. On the other hand, 0.36 M DHOA appears to be a good compromise between SF and distribution ratio values.

Earlier studies on the recovery of ^{233}U from irradiated thoria (from CIRUS reactor) used 0.18 M TBP in *n*-dodecane as extractant and a feed solution ~200 g/l Th + ~0.2 g/l ^{233}U + ~225 mCi/l β activity + ~60 mCi/l γ activity + 0.03 M F^- + 0.1 M $Al(NO_3)_3$ at 4 M HNO_3 [15]. In view of the average enrichment of 3.5% Pu in proposed AHWR fuels, the concentration of thorium in the spent fuel feed solution was proposed to be kept ~100 g/l to limit the fissile content to <7 g/l to maintain a safe concentration from the criticality viewpoint. Further studies were, therefore, carried out on a new composition of simulated AHWR feed solution containing U, Pu and Th concentrations as 2 g/l, 2 g/l, and 100 g/l at 3.5 M HNO_3 respectively [18]. Required quantities of $Al(NO_3)_3$ and concentrated HF were also added to get a their concentration of 0.1 M and 0.03 M, respectively. However, it should be noted that organic-to-aqueous phase ratio (O/A) was increased in the case of DHOA beyond ligand concentration 0.18 M to avoid third-phase formation under THOREX feed conditions [14]. By contrast, no such problem was encountered in the presence of 100 g/l Th in the proposed AHWR feed solution using DHOA as the extractant.

The data suggested that under co-current mode, 6 and 10 contacts were required for quantitative extraction of U and Pu employing 0.18 M TBP/*n*-dodecane as extractant. By contrast, only 5 contacts of 0.36 M DHOA/*n*-dodecane were sufficient for achieving quantitative extraction. Dhami et al. reported that 7 and 9 contacts were required for quantitative extraction of U and Pu, respectively, employing 0.18 M TBP/*n*-dodecane as extractant [18]. Under these experimental conditions, thorium extraction was expected to be 38.6% (10 contacts) and 25.3% (5 contacts), for 0.18 M TBP and 0.36 M DHOA solutions, respectively. However, it has to be noted that the TBP concentration studied in this work is half that employed for DHOA.

3.2. Scrubbing studies

During the extraction step, co-extracted thorium is required to be scrubbed using appropriate concentration of nitric acid. Table 2 lists the distribution data of U and Pu at 1–4 M HNO_3 for both the loaded organic

Table 1
D_M as a function of DHOA concentration under THOREX/AHWR feed condition; Diluent: *n*-dodecane; THOREX feed: 200 g/l Th + ~0.2 g/l U + 0.03 M F^- + 0.1 M $Al(NO_3)_3$ at 4 M HNO_3; AHWR feed: 100 g/l Th + 2 g/l U + 2 g/l Pu + 0.03 M F^- + 0.1 M $Al(NO_3)_3$ at 4 M HNO_3; T: 25°C

[Ligand], M	TBP			DHOA		
	D_{Th}	D_U	D_{Pu}	D_{Th}	D_U	D_{Pu}
0.18	0.04	2.3	1.6	0.01	1.1	1.3
0.18[#]	***0.05***	***2.1***	***1.1***	***0.01***	***1.8***	***0.8***
0.36	0.043	2.4	2.0	0.05	4.0	5.7
0.36[#]	–	–	–	***0.06***	***3.1***	***3.3***

[#] Values in italics and bold refer to simulated AHWR feed solution.

Table 2
Scrubbing of U and Pu from loaded 0.18 M TBP and 0.36 M DHOA phases; Diluent: *n*-dodecane; O/A: 1; T: 25°C

Extractant	[HNO_3], M	D_U	D_{Pu}
0.18 M TBP	1.0	0.6	0.2
	2.0	1.8	0.5
	3.0	2.7	0.8
	4.0	***2.8***	***1.2***
0.36 M DHOA	1.0	0.3	0.1
	2.0	1.0	0.5
	3.0	1.7	1.2
	4.0	***2.3***	***2.4***

phases. The aim was to minimize the loss of U and Pu to the scrub solution. Thorium could not be detected in the loaded organic phase under all scrubbing conditions (D_{Th}: <10^{-2}).

Based on these observations, 4 M HNO_3 solution was found suitable for scrubbing of Th with minimum loss of U and Pu in the scrub solution. Whereas 0.18 M TBP appears better with respect to U loss; 0.36 M DHOA is promising with respect to Pu loss in the scrubbing cycle. Typical D_U and D_{Pu} values at 4 M HNO_3 are 2.8 and 1.2 (for 0.18 M TBP), and 2.3 and 2.4 (for 0.36 M DHOA).

3.3. Partitioning studies

In view of the low efficiency of reduction of Pu^{4+} to Pu^{3+}, large quantities of natural uranous salt (~10 times the stoichiometric amount of Pu^{4+}) are required to be generated from a specially designed electrolyzer to generate natural U^{4+} salt for partitioning of Pu from U. However, this will result in isotopic dilution of ^{233}U during AHWR fuel reprocessing. Therefore, it calls for (a) the evaluation of new non-salt forming reductants, and (b) development of equipment for in-situ reduction of Pu^{4+} to Pu^{3+}, for partitioning of Pu from U present in the loaded organic phases(s) [2]. It may be noted that electrochemical in-situ reduction of Pu^{4+} to Pu^{3+} will also lead to simultaneous generation of some U^{4+} from U^{6+}, which is acceptable because the generated uranous ion will act as additional reductant for Pu^{4+} to Pu^{3+}. Plutonium partitioning studies were, therefore, carried out using 0.5 M solutions of different strippant/reductant solutions prepared in 2 M HNO_3 to minimize the loss of uranium in the strip solution.

Table 3 lists the distribution data of uranium and plutonium along with their retention % in the organic phases. Most of the strippants appear very effective for the partitioning of Pu from loaded organic phases. Uranium retention was observed to be ~65% for 0.18 M TBP and ~41% for 0.36 M DHOA as the extractant during the partitioning cycle. Accordingly, the uranium lost in the partitioning cycle needs to be re-extracted after adjusting the feed acidity suitably (~7 M HNO_3). The higher uranium retention by 0.18 M TBP is an advantage over 0.36 M DHOA as solvent. The data suggested that at least two-three stages were required for quantitative stripping of Pu from the loaded TBP phase. By contrast, only two stages were sufficient for Pu stripping from loaded DHOA phase. Loaded uranium can be stripped by dilute acid (0.01 M HNO_3) solution.

Table 3
Partitioning of U and Pu from scrubbed 0.18 M TBP and 0.36 M DHOA phases; Aqueous phase: 0.5 M solution of different reductants at 2 M HNO_3; O/A: 1; T: 25°C

Extractant	Reductant	D_M		Retention, %	
		U	Pu	U	Pu
0.18 M TBP	HAN	2.0	0.02	66.7	1.6
	DEHAN	2.2	0.01	70.0	1.1
	HU	1.3	0.002	57.3	0.2
	AHA	1.9	0.01	66.0	1.0
	AOX	2.3	0.004	69.5	0.4
0.36 M DHOA	HAN	0.7	0.009	41.2	0.9
	DEHAN	0.7	0.002	41.2	0.2
	HU	0.6	0.001	36.3	0.06
	AHA	0.8	0.004	43.8	0.4
	AOX	0.8	0.002	43.5	0.2

These studies suggested that 0.5 M HAN/AHA/HU/AOX solutions at 2 M HNO_3 solutions can be used as strippant for the partitioning of Pu during the reprocessing of AHWR spent fuels.

3.4. Protactinium extraction studies

Initially, the AHWR spent fuels are expected to reach a burn-up of 15000 MWd/Te which will be further raised to 24,000 MWd/Te under equilibrium conditions. Plutonium in AHWR burns faster due to large absorption cross-section that leads to loss in reactivity. However, it provides an option to reconstitute the fuel cluster after an average discharge burn-up of 24,000 MWd/Te. During this process, only $(Th, Pu)O_2$ pins will be replaced by fresh fuel pins. Thus, it is possible to obtain an additional burn-up of ~20,000 MWd/Te from the reconstituted cluster. The cluster reconstitution improves ^{233}U production and reduces the reprocessing load due to increased average cluster burn-up. Therefore, it would be desirable to reprocess some of the fuel bundles after short-cooling every year to achieve a burn-up of ~ 40,000 MWd/Te and to get a pure ^{233}U product from the short cooled spent fuel [1,2].

In this context, the extraction profile of protactinium becomes important. Fig. 1 shows the variation of D_{Pa} values with aqueous phase acidity (1–6 M HNO_3) for 0.18 M TBP, 0.36 M DHOA solutions as solvents. There is a gradual increase in D_{Pa} with increased aqueous phase acidity and 0.36 M DHOA shows better extraction than 0.18 M TBP. This behavior can be explained in terms of the following equilibrium between the hydrolyzed cationic, neutral and anionic species of Pa in nitric acid medium. The interchange between these species is fast [20].

$$\left[\mathrm{Pa}(\mathrm{OH})_2(\mathrm{NO}_3)\right]^{2+} \leftrightarrow \left[\mathrm{Pa}(\mathrm{OH})_2(\mathrm{NO}_3)_2\right]^{+}$$
$$\leftrightarrow \left[\mathrm{Pa}(\mathrm{OH})_2(\mathrm{NO}_3)_3\right] \leftrightarrow \left[\mathrm{Pa}(\mathrm{OH})_2(\mathrm{NO}_3)_4\right]^{-} \leftrightarrow \ldots\ldots$$

(***extractable species***)

Interestingly, there was a decrease in D_{Pa} values under simulated AHWR/THOREX feed conditions for both

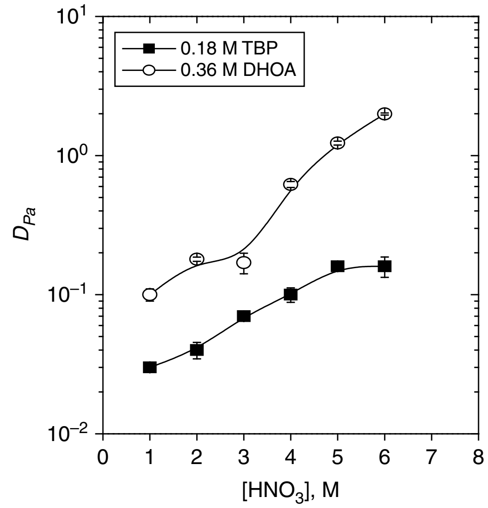

Fig. 1. Variation of D_{pa} with aqueous phase acidity.

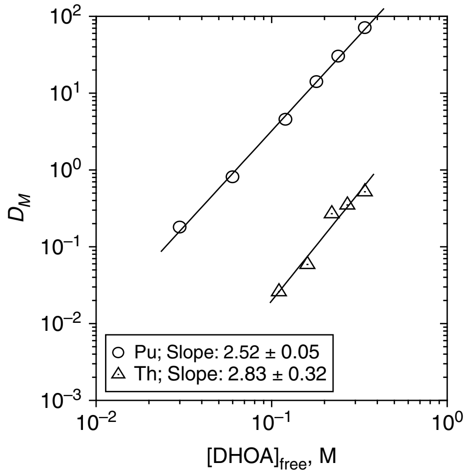

Fig. 3. D_{pu} and D_{Th} as a function of DHOA concentration; [Pu]: ~10^{-4} M; [Th]: ~10^{-4} M; [HNO_3]: 4 M.

the extractants. Typically for 0.18 M TBP, the D_{Pa} values decreased from 0.1 (no Th) to 0.07 (AHWR feed) and 0.05 (THOREX feed). The corresponding values for 0.36 M DHOA were 0.6, 0.35, and 0.03, respectively. This behavior was attributed to (a) thorium loading in the organic phase, and (b) the formation of inextractable anionic nitrate species due to the presence of ~6–7 M NO_3^- ions. Similar observations were made during the studies on extraction of protactinium in the presence of macro concentrations of thorium using D2EHIBA as extractant [21]. Fig. 2 suggests that DHOA appears particularly promising for the preferential extraction of U and Pu from dissolver solution of AHWR spent fuel arising out of (Th-Pu)O_2 pins in view of improved SF of U and Pu over Th. However, SF values for U and Pu over Pa are relatively lower compared to those of 0.18 M TBP. It is evident from Fig. 3 that both Pu and Th form trisolvated species with DHOA; while disolvated and trisolvated species are reported to be formed in the case of TBP [22]. Therefore, D_{Pu} value is significantly enhanced with increased DHOA concentration as compared to that of TBP. DHOA offers better separation of U and Pu over Th under AHWR/THOREX feed conditions.

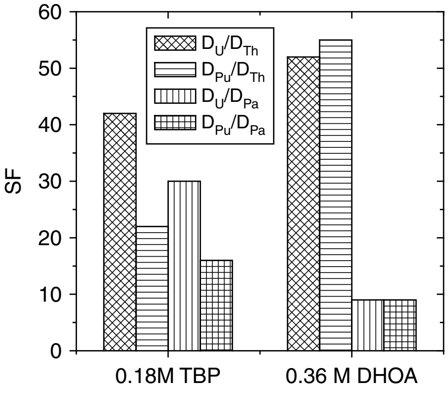

Fig. 2. Comparison of TBP, DHOA for U/Pu separation from Th and Pa under AHWR feed conditions.

3.5. Extraction behavior of minor actinides

The concentration of minor actinides produced in the discharged fuel of AHWR (viz. ^{231}Pa, ^{237}Np, ^{241}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, and ^{244}Cm) for different fuel cycles (PWR, PHWR, AHWR) were compared using computer code ORIGEN-2 [24]. The average burn-up of the AHWR fuel was taken as 24,000 MWD/Te. This study suggested that the production of minor actinides per unit energy was less in AHWR fuel cycle than those of PWR fuel (~29%), but it was comparable to the case of PHWR fuel. It should be noted that the minor actinides in the Th-U fuel per unit energy (TWh or GWy) are significantly less compared to uranium fuel. This is essentially because except for ^{231}Pa, most of the minor actinides are having mass numbers much higher than ^{232}Th, so at least 5 to 6 capture and decay reactions are required to generate those minor actinides. By contrast, the minor actinides per unit energy (TWh or GWy) for the plutonium pins are relatively higher because of Am and Cm isotopes. Typically, the concentrations (g/Te) of different minor actinides in AHWR spent fuel with average burn-up of 24,000 MWD/Te (~21000 MWD/Te for Th-Pu & 27000 MWD/Te for Th-U), and 10 y of cooling are: ^{231}Pa (2.2), ^{237}Np (5.7), ^{241}Am (512), ^{243}Am (25.7), ^{242}Cm (4.2×10^{-3}), ^{243}Cm (0.21), and ^{244}Cm (1.87) [23].

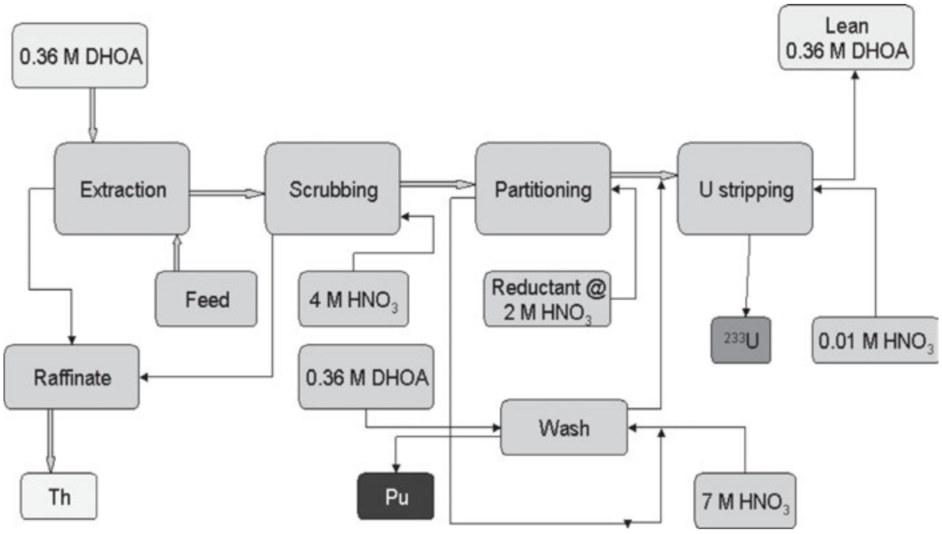

Fig. 4. Proposed solvent extraction scheme for the reprocessing of three component AHWR spent fuel using DHOA as extractant.

The reprocessing task for AHWR spent fuel gets further complicated due to the formation of minor actinides such as ^{231}Pa, Np, Am and Cm isotopes. In this context, the extraction behavior of ^{237}Np (no valency adjustment) and ^{241}Am, were investigated under proposed AHWR spent fuel feed conditions employing 0.18 M TBP and 0.36 M DHOA as extractants. Interestingly, both extractants showed appreciable extraction of Np (D_{Np}: 0.96 (0.18 M TBP) and 2.0 (0.36 M DHOA)). However, 0.36 M DHOA appears attractive for co-extraction of Np along with U and Pu. On the other hand, negligible extraction of Am was observed for both the extractants (D_{Am}: $<10^{-3}$). This study suggests that a separate actinide partitioning step is required after the reprocessing of the three components AHWR spent fuel.

3.6. Fission/activation products extraction studies

Batch extraction experiment was performed on simulated AHWR feed solution spiked with a diluted aliquot of High Level Waste (HLW) sample from the PUREX process stream as the aqueous phase and 0.18 M TBP and 0.36 M DHOA solutions in *n*-dodecane as extractants. Addition of HLW helped to get composite fission product activity in the simulated AHWR feed solution. After equilibration, the phases were separated and assayed for individual fission/activation products' (viz. ^{137}Cs, ^{144}Ce, ^{106}Ru, ^{60}Co etc.) activity using gamma spectrometry. The distribution ratio values for these radionuclides were of the order of ~10^{-4} to 10^{-3} suggesting better decontamination of U and Pu for both the solvents (viz. 0.18 M TBP and 0.36 M DHOA solutions in *n*-dodecane) over fission/activation products present in the AHWR feed solution.

3.7. Proposed solvent extraction scheme for AHWR spent fuel

Based on these solvent extraction studies, a reprocessing scheme has been proposed for AHWR spent fuel employing 0.36 M DHOA as extractant (Fig. 4). Scrubbing and stripping solutions were 4 M HNO_3 and 0.5 M HAN/AHA/HU/AOX in 2 M HNO_3, respectively. However, extensive mixer settler studies are still required to arrive at the exact operational conditions for the reprocessing of AHWR spent fuels using DHOA as the extractant. In view of relatively higher limiting organic concentration (LOC), thorium recovery from raffinate can be carried out by long-chain dihexyl decanamide (DHDA) [14]. After Pu partitioning step, the stripped aqueous phase can be given a wash with fresh 0.36 M DHOA solution to recover uranium, which can be finally be stripped with 0.01 M HNO_3 solution.

4. Conclusions

Batch extraction studies were carried out to compare the extraction behavior of 0.18 M TBP and 0.18 M/0.36 M DHOA in *n*-dodecane and to propose a reprocessing scheme for the three-component system (U, Pu, Th) under simulated AHWR feed solution arising from irradiated (Th, Pu)O_2 fuel. Studies clearly indicated that DHOA appears particularly promising for the preferential extraction of U and Pu from dissolver solution of

AHWR spent fuel arising out of (Th, Pu)O_2 pins. However, SF values for U and Pu over Pa are relatively lower as compared to those of 0.18 M TBP. Conditions for scrubbing of co-extracted thorium and for the partitioning of plutonium from uranium were optimized as 4 M HNO_3 and reductants like 0.5 M HAN/AHA/HU/AOX at 2 M HNO_3, respectively. Appreciable extraction of Np was observed for both extractants (~50% (0.18 M TBP); ~66% (0.36 M DHOA) at O/A: 1) under simulated AHWR feed conditions. By contrast, there was negligible extraction (D_M: 10^{-4} to 10^{-3}) of Am and fission/activation products (viz. ^{137}Cs, ^{144}Ce, ^{106}Ru, ^{60}Co etc.) suggesting better decontamination of U and Pu. Based on the co-current extraction data, a solvent extraction reprocessing scheme has been proposed for the three component AHWR spent fuel, employing 0.36 M DHOA as extractant.

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