

## Salt Separation from Uranium Deposits in Integrated Crucible

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### 1. Introduction

Pyroprocessing has been developed for the recovery of actinide elements from spent fuel due to its advantages. Electrorefining is a key step in pyroprocessing. The electrorefining process is generally composed of two recovery steps – the deposit of uranium onto a solid cathode and the recovery of the remaining uranium and TRU elements simultaneously by a liquid cadmium cathode. The solid cathode processing is necessary to separate the salt from the cathode since the uranium deposit in a solid cathode contains electrolyte salt.

A physical separation process, such as distillation separation, is more attractive than a chemical or dissolution process because physical processes generate much less secondary process. Distillation process was employed for the cathode processing due to the advantages of minimal generation of secondary waste, compact unit process, simple and low cost equipment [1,2]. The basis for vacuum distillation separation is the difference in vapor pressures between salt and uranium. A solid cathode deposit is heated in a heating region and salt vaporizes, while non-volatile uranium remains behind [3].

It is very important to increase the throughput of the salt separation system due to the high uranium content of spent nuclear fuel and high salt fraction of uranium dendrites [4,5]. The evaporation rate of the LiCl-KCl eutectic salt in vacuum distiller is not so high to come up with the generation capacity of uranium dendrites in electro-refiner. Therefore, wide evaporation area or high distillation temperature is necessary for the successful salt separation. Kwon et al. proposed an integrated crucible for a high throughput salt separation [6].

In this study, the feasibility of the sequential salt separation in an integrated crucible was investigated by sequential operation of a liquid salt separation - a vacuum distillation.

### 2. Experimental

The salt separation experiments were carried out in an integrated sieve-crucible assembly in distillation tower. Fig. 1 shows a sieve for the solid-liquid separation and a crucible for the integrated sieve-crucible assembly. The mesh opening was 150  $\mu$  m.

The salt separation experiments were carried out in the vacuum distiller. The distiller was composed of a distillation tower with an evaporator and a condenser, a

control unit, and an off gas treatment system. The diameter of the tower at an evaporation area was 400 mm. The temperature in the crucible was measured by using a thermocouple that is connected to the centre of the tower.

About 80 g of uranium deposits was put into the crucible and heated at 600° C for 30 minutes for liquid salt separation. After the liquid salt separation step, the assembly was rotated and the temperature was increased to 820°C for the salt distillation. After distillation, the deposit samples were dissolved into nitric acid and the salt contents were measured by using ICP spectroscopy.

### 3. Results and Discussion

The experiment on the sequential operation of the liquid salt separation and salt distillation was carried out using an integrated sieve-crucible assembly. The uranium deposits are placed into the sieve side of the assembly and the adhered salt is separated by heating on the sieve. Then, the uranium deposits are moved to the crucible by the rotation of the sieve-crucible assembly and the residual salt is evaporated at an elevated temperature. Figure 7 shows the sieve and crucible with uranium deposits.

The remained salt in the uranium deposits was further separated by evaporation in the distillation tower for two hr at 820° C. After the salt distillation, the deposit samples were dissolved into the nitric acid and the salt contents were measured by using ICP spectroscopy. The salt content in the deposits was below 0.1 wt% after the sequential operation of the liquid salt separation and salt distillation. Figure 3 shows XRD pattern of uranium deposits after the liquid salt separation and salt distillation. The Li or K is not found on the XRD pattern since the salt content is very low. Uranium oxide was formed by the oxidation of uranium metal during sample preparation. The adhered salt in the uranium deposits was removed successfully. This residual salt after salt separation process can be removed completely during the melting of uranium metal in the following ingot preparation process.

Thus, the efficient salt separation work was realized in an integrated sieve-crucible assembly by the sequential operation of liquid salt separation and vacuum distillation because the operation procedure is simplified and no extra operation of cooling and heating is necessary for the change of the operation modes. Therefore, it could be concluded that the sequential operation of liquid salt separation and vacuum

distillation using an integrated sieve-crucible assembly is one of the effective ways for the achievement of a high throughput performance in the salt separation process.



Fig. 1. Photographs of (a) a sieve for the solid-liquid separation and (b) a crucible for the integrated sieve-crucible assembly.



Fig. 2. Photographs of uranium deposits (a) in a sieve and (b) in a crucible

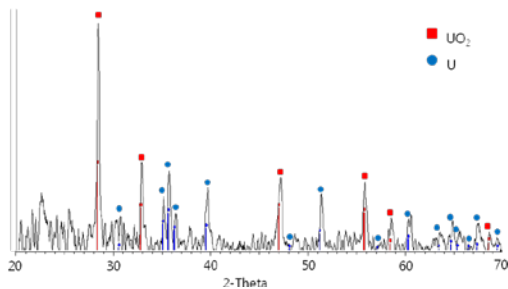


Fig. 3. XRD pattern of uranium deposits after the liquid salt separation and salt distillation

#### 4. Conclusions

In this study, the feasibility of the sequential salt separation in an integrated crucible was examined by sequential operation of a liquid salt separation - a vacuum distillation. The adhered salt in the uranium deposits was removed successfully. The salt content in the deposits was below 0.1 wt% after the sequential operation of the liquid salt separation - salt distillation.

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