

Salt Removal from the Uranium Deposits of Electrorefiner

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1. Introduction

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 [1,2]. The solid cathode processing is necessary to separate the salt from the cathode since the uranium deposit in a solid cathode contains electrolyte salt. In the liquid cathode, cadmium metal should be removed to recover actinide product.

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 [3]. 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 [4].

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 [5-6].

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 electrorefiner. Therefore, wide evaporation area or high distillation temperature is necessary for the successful salt separation.

In this study, the solid-liquid separation was proposed prior to distillation of salt and a feasibility of the separation of the liquid salt by a metallic wire mesh (sieve) was tested for the reduction of the burden of the following vacuum distillation process.

2. Experimental

The solid-liquid separation experiments were carried out on the sieve with a stainless steel mesh. Fig. 1(a) shows a photograph of a sieve assembly including a stainless steel mesh, a stand and a bottom salt container for the solid-liquid separation. The mesh has holes with a diameter of 150 μm . The liquid salt separation experiments were carried out with a LiCl-KCl and

LiCl-KCl-UCl₃ salt systems. In the LiCl-KCl-UCl₃ system, the content of UCl₃ was 5 wt%.

The salt evaporation experiments were carried out after the liquid salt separation experiment 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 400mm. The temperature in the crucible was measured by using a thermocouple that is connected to the centre of the tower.

3. Results and Discussion

Basically, the amount of adhered salt in the solid cathode deposit is much more than that of cadmium metal in the liquid cathode, since the uranium is the main component of the spent fuel. The fraction of adhered salt in uranium deposits is more than 20 % [6]. Therefore, the capacity of salt distiller should be large enough.

For the achievement of high throughput performance in the salt separation process, it was proposed on the combination of a liquid salt separation and a vacuum distillation process. In the combined process, the adhered salt in the uranium deposits is separated by heating on the sieve at low temperature compared to the vacuum evaporation and then the residual salt is distilled at a high temperature.

The adhered salt contained in the uranium deposits is separated in the solid-liquid separation column at a low temperature and the remained uranium deposits are transferred to the vacuum distillation tower. The residual salt in the deposits is removed further by vacuum distillation. The final salt removed uranium deposits are transferred to the uranium casting process for the ingot preparation.

It was examined on the feasibility of the combination process of a liquid salt separation and a vacuum distillation process. The solid-liquid separation experiments were carried out by the use of a stainless steel mesh. The mesh has holes with a diameter of 150 μm . Two lumps of salt (about 100g) were placed on the mesh and heated at 450 $^{\circ}\text{C}$ for 0.5 hr. Fig. 1(b) shows a photograph of a frozen LiCl-KCl salt collected in the bottom container. As shown in Fig. 1(b), most of the LiCl-KCl eutectic salt flew down by gravity through the mesh. At 400 $^{\circ}\text{C}$, the salt was melted on the sieve but did not flew through the mesh holes.

It was also tested the possibility of the separation of LiCl-KCl- UCl_3 , since the residual salt in cathode deposits contains uranium chloride. For the experiment of the uranium chloride containing salt, lumps of the salt (about 50 g) was heated on the mesh during 1 hr at 450°C. More than 98 wt % of the initial salt remained on the mesh. This phenomenon was probably caused by a viscosity of the uranium chloride containing salt. In the experiment of heating at 500°C during 0.5 hr, less than 1.0 wt % of the initial salt remained on the mesh surface as a thin film as shown in Fig. 2. Therefore, it could be concluded that the liquid salt can be separated from the uranium dendrites above 500°C.

Generally, the uranium deposits (dendrites) on the solid cathode are much larger than 150 µm. From the above results, the good deal of salt retained in the uranium deposits can be separated within 0.5 hr at low temperature before distillation. Therefore the amount of salt to be distilled in the uranium deposits can be highly reduced and the design capacity of salt distiller can be reduced.

In the experiment for the distillation of residual salt, the dendrite remained salt was successfully evaporated.



Fig. 1. Photographs of (a) a sieve assembly for the solid-liquid separation and (b) a frozen LiCl-KCl salt collected in the bottom container.



Fig. 2. Photograph of the salt (LiCl-KCl- UCl_3) residue on the sieve after the solid-liquid separation experiment.

4. Conclusions

In this study, the separation of the liquid salt prior to distillation of the LiCl-KCl eutectic salt was proposed for the high throughput performance of the salt removal process in the uranium deposits of the solid cathode. It was found that the reduction of the burden of the distillation process was feasible by the liquid salt separation prior to the salt distillation. It could be

concluded that the liquid salt can be separated from the uranium dendrites above 500°C. The residual salt remained in the dendrite after the liquid salt separation was successfully removed by the vacuum distillation.

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