Removal of Adhered Salt from Uranium Deposits in Pyroprocess

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Abstract

Pyroprocessing has been developed for the recovery of actinide elements from spent fuel due to its advantages of a compactness, a nuclear proliferation resistance, and a reduction of a secondary waste generation [1-2]. It was proposed to increase the throughput of the salt removal process by the separation of the liquid salt prior to the distillation of the LiCl-KCl eutectic salt from the uranium deposits in this study. The feasibility of liquid salt separation was examined by salt separation experiments on a stainless steel sieve. It was found that the amount of salt to be distilled could be reduced by the liquid salt separation prior to the salt distillation. It was found that the liquid salt can be separated from the uranium dendrites above 500℃. The residual salt remained in the uranium deposits after the liquid salt separation was successfully removed by the vacuum distillation. It was concluded that the combination of a liquid salt separation and a vacuum distillation process is an effective route for the achievement of a high throughput performance in the salt separation process because the amount of salt to be distilled in the uranium deposits can be highly reduced and the burden of salt distiller can be reduced.

REFERENCES

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