Development of Decontamination Technology for Separating Radioactive Constituents from Contaminated Concrete Waste

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1. Introduction

The large amount of contaminated concrete produced during decommissioning procedures and available decontamination. In Korea, more than more than 60 tons of concrete wastes contaminated with uranium compounds have been generated from UCP (Uranium Conversion Plant) by dismantling. A recycling or a volume reduction of the concrete wastes through the application of appropriate treatment technologies have merits from the view point of an increase in a resource recycling as well as a decrease in the amount of wastes to be disposed of resulting in a reduction of a disposal cost and an enhancement of the disposal safety. For unconditional release of building and reduction of radioactive concrete waste, mechanical methods and thermal stress methods have been selected. In the advanced countries, such as France, Japan, Germany, Sweden, and Belgium, techniques for reduction and reuse of the decommissioning concrete wastes have applied to minimize the total radioactive concrete waste volume by thermal and mechanical processes. It was found that volume reduction of contaminated concrete can be achieved by separation of the fine cement stone and coarse gravel. Typically, the contaminated layer is only 1~10mm thick because cementitious materials are porous media, the penetration of radionuclides may occur up to several centimenters from the surface of a material [1]. Most of the dismantled concrete wastes are slightly contaminated rather than activated. This decontamination can be accomplished during the course of a separation of the concrete wastes contaminated with radioactive materials through a thermal treatment step of the radionuclide (e.g. cesium and strontium), transportation of the radionuclide to fine aggregates through a mechanical treatment step. Concrete is a structural material which generally consists of a binder (cement), water, and aggregate. The interaction between highly charged calcium silicate hydrate (C-S-H) particles in the presence of divalent calcium counter ions is strongly attractive because of ion-ion correlations and a negligible entropic repulsion. In the temperature range 100-300^{o}C, these evolutions are mainly attributed to the loss of the bound water from the C-S-H gel. Volume reduction studies for radioactively contaminated dismantled concrete wastes have been carried out by using thermal and mechanical processes. The series of semi-pilot tests have investigated the characteristics of a separation of the aggregates and the distribution of the radioactivity into the aggregates from the volume reduction point of view using uranium contaminated light weight concrete from the UCP.

2. Method and Result

Pilot tests were performed with radioactively contaminated dismantle concrete waste generated from UCP. The characteristics of the distribution and absorption ratio of separated aggregates were investigated. In these experiments, the heating and crushing time of processes were fixed at 40 min and using about 80 kg of light concrete waste **Fig. 1, 2** shows the experimental separation procedure. The concrete wastes were crushed for a size reduction. In the jaw crusher the input concrete is crushed to about 40mm diameter parts. The crushed concrete rubble is separated into groups of coarse aggregate (>5mm), fine aggregate (1-5mm), and cement paste (<1mm). The separated concrete rubble is heated to about 500℃ for 40 minutes at furnace. The classified each aggregate was taken by a standard cylindrical vial at 20ml and 80ml for an activity analysis. The aggregate samples were analyzed with a MCA (multi-channel-analysis) of a high-purity germanium detector to evaluate the radionuclides distribution.

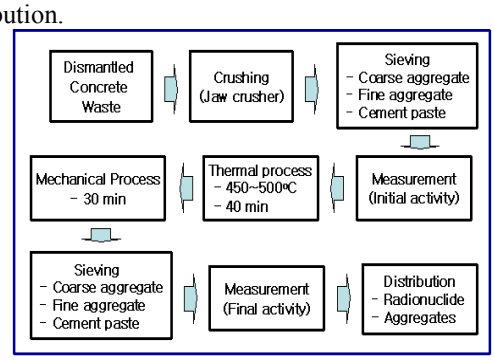

Fig. 1. Experimental procedure

Fig. 2. The picture of pilot test

Fig. 3 shows the distribution of uranium on separated aggregated by thermal and mechanical treatment. This experiment was performed with 80kg/batch for contaminated uranium light concrete. Concrete rubble by thermal and mechanical method was separated into group of gravel, sand and cemented paste. The specific activity of gravel and sand aggregate was 0.34Bq/g. 0.56 Bq/g, respectively. The specific radioactivity of the gravel and sand aggregate became much lower and most of the radioactivity was concentrated into the cement paste.

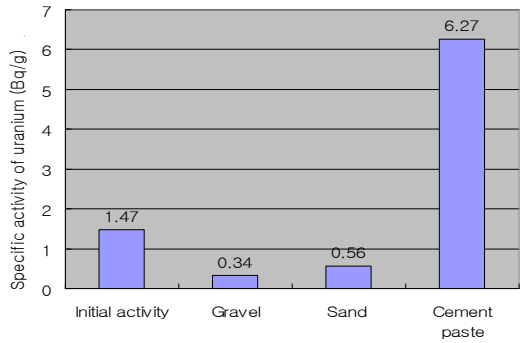

Fig. 3. Specific activity of uranium on separated aggregates by semi-pilot test

Fig. 4 shows the distribution of the separated aggregates by pilot test. The general light weight concrete could be separated into gravel sand aggregates of more than 79% with a low specific radioactivity. The uranium was removed easily from the concrete wastes by heating to weaken the adherence force between the cement matrix and aggregates by mechanical treatment. It is possible reducing the volume reduction of light concrete waste contaminated with uranium compounds generated from UCP by about 260 ton.

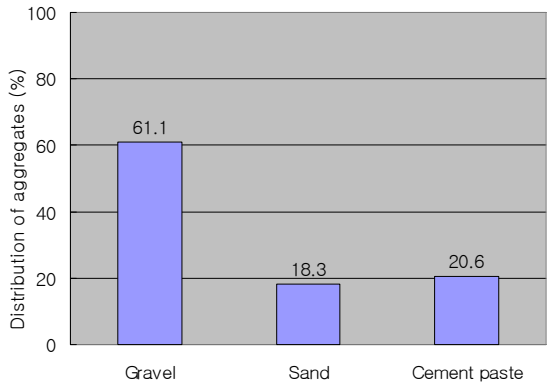

Fig. 4. Distribution of separated aggregates by semi-pilot test.

The main source of cohesion in cement paste is the nano-particles of calcium silicate hydrate (C-S-H), which are formed upon the dissolution of the original tricalcium silicate (C_3S). The interaction between highly charged C-S-H particles in the presence of divalent calcium counter ions is strongly attractive because of ion-ion correlations and a negligible entropic repulsion. Sylwester et al. [2] have shown that U remains in the form of UO_2^{2+}. Evidence of inner-sphere interactions on both treated and untreated cements at all pH values was found, with UO^{2+}_2 complexing with the mineral surface via sharing of equatorial oxygen. The coordination environment of U (IV) in the co-precipitated samples depended on the synthesis method, and the spectra differed from those for the sorption samples.

3. Conclusions

Radionuclides were easily removed from the concrete waste by thermal treatment to weaken the adherence force between the cement matrix and the aggregate follow by mechanical treatment. Radioactivity is mainly concentrated in the porous cement paste. A volume reduction of activated concrete waste and uranium concrete waste was achieved by up to about 80%, 75% respectively. It was expected that greater part of concrete wastes generated from UCP could be a volume reduction.

4. Future work

Pilot tests will perform with dismantled concrete waste generated from UCP using modified semi-pilot plant in the near future. **Fig. 5** shows the schematic of modified pilot test.

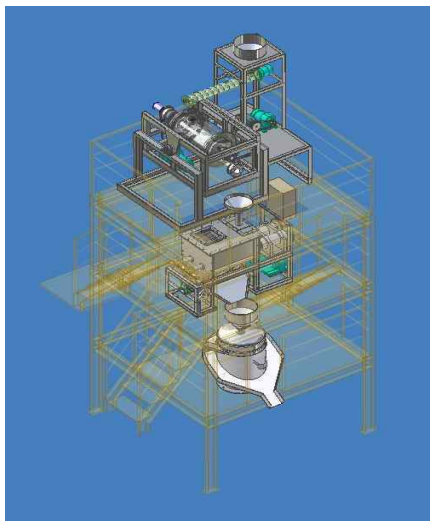

Fig. 5. Schematic of modified pilot test

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[2] Sylwester, E. R., Allen, P. G., Zaho P. and Viani, B. E., “Interactions of uranium and neptunium with cementitious materials studied by XAFS”, Mat. Res. Soc. Symp. Proc. 608, pp. 307-312, (2000)