Assessment of the Characteristic Aggregates during a Decontamination of Contaminated Concrete Waste

B.Y. Min, W.K. Choi, W.Z. Oh, C.H. Jung, J.W.Park
Korea Atomic Energy Research Institute, D&D Technology R&D Dept., P.O. Box 105 Yuseong, Daejeon
bymin@kaeri.re.kr

1. Introduction

During a decommissioning of nuclear plants and facilities, large quantities of slightly contaminated concrete wastes are generated. The exposure to radiation over many years could be hazardous to human health. In Korea, the decontamination and decommissioning of the retired TRIGA MARK II and III research reactors and a uranium conversion plant at the Korea Atomic Energy Research Institute (KAERI) has been under way. Hundreds of tons of concrete wastes are expected from the D&D of these facilities [1]. 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 [2, 3]. Contaminated concrete waste can be of two forms, either a surface or bulk contamination. Bulk contamination usually arises from a neutron activation of nuclides during the service life on a component. Surface activity can be a loose contamination arising from a deposition of nuclides from an interfacing medium, and it also can be tightly bound. 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 such as a crushing, milling and sieving. Produced fine powder (paste) should be stabilized for the final disposal. Melting technology has been known as the one of the most effective technologies for a stabilization and volume reduction to the paste [4-6]. Therefore, a melting may be a last step in the decontamination of a contaminated paste.

The aim of this study was to establish the separation conditions for an optimum decontamination for the treatment of concrete wastes contaminated with radioniclides. The separation tests had been performed using the concrete wastes which were contaminated with the radioisotope (^{60}Co) generated from TRIGA MARK II and III in KAERI. To minimize the total radioactive waste volume of the radioactive concrete as a paste, the fine powder wastes were melted in a lab-scale direct current graphite arc melting system.

2. Method and Result

for the experimental test, this is non-radioactive concrete waste generated by dismantling the retired TRIGA MARK II and III research reactors. Figure 1 shows the separation process flow chart. In this figure, four main steps can be distinguished. In the first step concrete parts are crushed for a size reduction. After the 1st crushing using a joy-crusher, non-radioactive concrete wastes are absorpted during 24 hours in a radioisotope solution and dried during 1 hour at a room temperature. By means of a heating, the bond between the aggregate and paste matrix is reduced. The crushed aggregate was heated under 300, 500, 700^{o}C in a muffle furnace during 1 hour and in the next step a separation was performed by a mechanical process in a ball mill for about 1 hour. Separation of the concrete into its components gravel, sand and paste is based on a reduction of the bond between the cement matrix and the aggregates. Three size fraction of crushed aggregate, 5.6-16mm (named from gravel), 1-5.6mm (named from sand), below 1mm (named from paste) were taken for test.

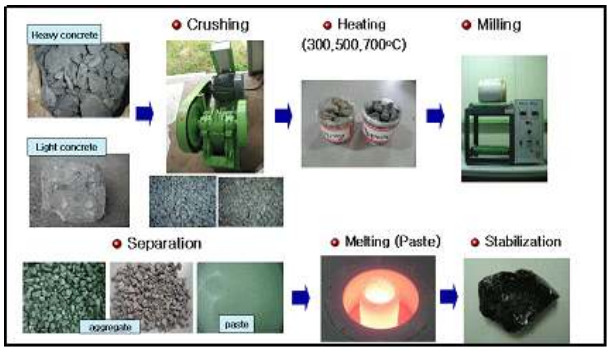

Fig.1. Experimental procedure

After the completion of a separation, the classified aggregate was taken by a standard cylindrical vial at 20ml and 80ml for a quantitative analysis. The aggregate samples were analyzed with a MCA (multi-channel-analysis) of a high-purity germanium detector for ^{60}Co to evaluate the radioisotope distribution. The final wastes such as the paste were melted in a lab-scale direct current graphite arc melting system. The direct current graphite arc furnace consists of one graphite electrode (20 mm diameter) and a copper crucible (100 mm inner diameter by 80 mm in depth) which serves as a counter electrode. The paste contaminated with the ^{60}Co was charged into the graphite crucible with a melting capacity of 500 cc. 200g of the paste wastes was melted under an ambient for 5 minutes.

Figures 1, 2 show the specific activity of the aggregate after the second milling using a ball mill. In this experiment, most of the radionuclide is removed from the gravel and sand. Most of the radionuclide is

transferred to the fine aggregate. The specific activity of ^{60}Co was below 0.4Bq/g in the gravel and sand. The recycling and reuse of the contaminated concrete waste was achieved by about 70% for the total contaminated concrete waste. The possible recycling and reuse of the contaminated concrete waste is influenced greatly by initial contamination level. That is, slightly contaminated concrete should be a possible reuse and recycle.

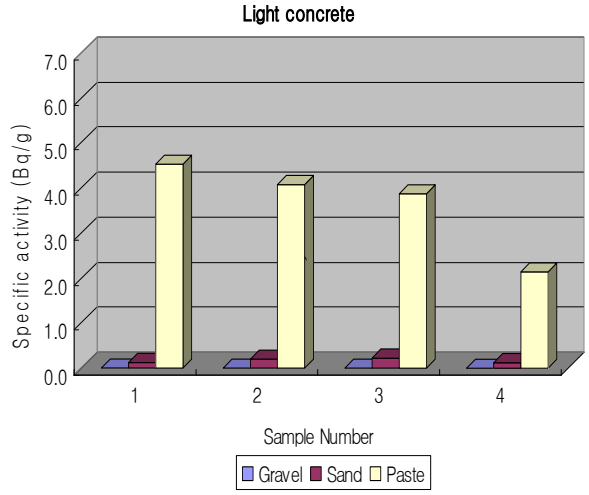

Fig. 1. Effects of specific activity for the second milling of the various light concrete samples

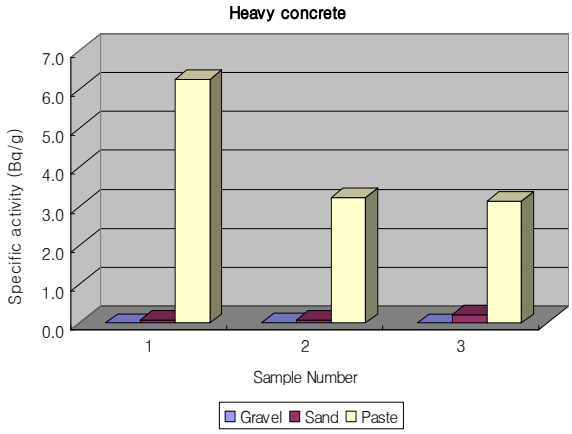

Fig. 2. Effects of specific activity for the second milling of the various heavy concrete samples

The fine powder generated from the separation process of the radioactive concrete wastes was a secondary radioactive waste. The volume reduction ratio (VRR) for the paste tests is shown in Figure 3. In this experiment, the melting test was carried out by only using the paste for stabilization without an additive material. The VRR for the paste of the heavy concrete is about 3. The VRR for the paste of the light concrete was reduced to a volume of little more that a half (factor 1/2.2).

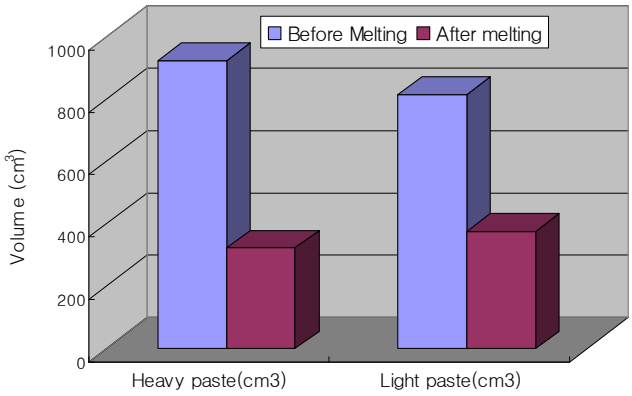

Fig. 3. Volume reduction of paste between heavy paste and light paste in arc furnace

3. Conclusions

The recycling and reuse of the contaminated concrete waste was achieved by about 70% for the total contaminated concrete waste. According to an increase of the amount of the paste and heating temperature, the removal of nuclide from the concrete waste linearly increased because the bond between the aggregate and paste matrix was reduced. For a stabilization of the second waste, slagging test was performed for the paste. The VRR for the heavy concrete was about 3. The VRR for the paste of the light concrete was reduced to a volume of little of more than half (factor 1/2.2) of that of the original waste

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