HIGH-LEVEL DEFENSE WASTE SOLIDIFICATION
AT THE SAVANNAH RIVER PLANT

by

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Introduction

The Savannah River Plant (SRP), operated by the Du Pont Company for the U.S. Department of Energy, is on a 300-square-mile site near Aiken, South Carolina, approximately 20 miles downstream from Augusta, Georgia (Slide 1). The site contains a nuclear fuel fabrication plant, three operating reactors, two fuel reprocessing plants, and a facility for producing heavy water. SRP started producing nuclear materials for national defense programs in 1953 and is now the only plutonium-producing plant in the United States. Other materials produced have been used for space programs and civilian uses.

Radioactive wastes generated in the SRP nuclear materials production processes are currently being stored in approximately 35 underground carbon steel tanks (Slide 2). These tanks

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contain about 20 million gallons of waste, which consists of an insoluble sludge layer covered by a layer of soluble salt cake and saturated salt solutions.

Defense Waste Solidification Process

The Savannah River Laboratory (SRL) is developing a process for the solidification of the radioactive waste in a stable form. In the reference design process for high-activity waste, the salt cake will be dissolved, and the resulting solution will be used to slurry the sludges from the tanks where they are currently stored. The salt solution will be separated from the sludge by centrifugation. More than 99% of the $^{137}$Cs and residual $^{90}$Sr will be removed from the salt solution by ion exchange. The remaining liquid will be evaporated to salt cake (mostly sodium nitrate) that contains only a small fraction of the total biologically hazardous radioactive material. The $^{137}$Cs and $^{90}$Sr will be combined with the sludge (containing most of the $^{90}$Sr and $^{239}$Pu). The combined eluate and sludge will be incorporated into borosilicate glass and shipped to a Federal repository (Slide 3).

Before the glass is formed, the sludge will be dried in a spray drier. The dried waste is mixed with a glass frit formulation (Slide 4) and melted at 1150°C. The molten glass will be poured into 2-foot-diameter by 10-foot-tall stainless steel canisters (Slide 5). The glass and canister will prevent the dispersion of radionuclides into the biosphere. Although the repository is expected to remain dry, the glass form is designed
to resist leaching by water. Therefore, the effects of water on glass are being studied.

Leaching Studies

The rate at which the radionuclides can be removed from the glass by water is called the leachability of the glass. Leach tests at the Savannah River Laboratory are determining the durability of glass waste forms in situations where water comes in contact with glass. The leachability of glass is used to determine whether or not a good glass has been made and will be a primary consideration for determining if glass is a suitable form for the disposal of radioactive waste. Leachabilities of $10^{-5}$ g/(cm²)(day) in water at 100°C are considered to be good.

In our program to find the optimum glassmaking formulation for the vitrification of sludges, Soxhlet leach tests have been used for comparison of glasses (Slide 6). This test is an accelerated test which gives durability data on glass in only 24 hours. Crushed glass is used for these tests to provide an increased glass surface area for reaction. In the test, continuously distilled water passes over the glass, and the leached glass components are concentrated in the bottom of the flask. The quantity of elements leached is determined by analysis of the leach solutions and by measuring the weight loss of the glass.

Accelerated leach tests are supplemented by long term leach tests designed to simulate actual leaching during long term
storage in a geologic repository. These long-term tests are at room temperature and at 90°C. In these tests, crushed glass is placed directly in the water leaching apparatus (Slide 7). The glass used in these tests contained simulated waste, spiked with radioactive tracers. The quantities of components which leach out of the glass are determined by radiometric analysis of the leaching solution.

One of the radionuclides of major concern is $^{137}$Cs. This element is one of the first to be leached from glasses. Therefore, it was used as a tracer in glass leaching experiments described above. $^{137}$Cs leaching data from typical glasses is shown in Slide 8.

Similar tests with actual radioactive glasses at room temperature give comparable results (Slide 7).

Program

Future tests at SRL will simulate all details of a repository: chemical conditions as well as temperature. Analysis of glass surfaces after leaching will help determine release mechanisms for the various radionuclides.
Slide 1. Location of Savannah River Plant
Slide 2. Photographs of Waste Tanks During Construction
Residual Solids

Soluble Salts

Insoluble Solids

Cs, Sr, Pu

To Salt Solidification
<table>
<thead>
<tr>
<th>Component</th>
<th>Wt Percent</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Glass Frit</strong></td>
<td></td>
</tr>
<tr>
<td>SiO$_2$</td>
<td>40.4</td>
</tr>
<tr>
<td>Na$_2$O</td>
<td>16.2</td>
</tr>
<tr>
<td>B$_2$O$_3$</td>
<td>7.2</td>
</tr>
<tr>
<td>TiO$_2$</td>
<td>7.2</td>
</tr>
<tr>
<td>CaO</td>
<td>4.5</td>
</tr>
<tr>
<td>Li$_2$O</td>
<td>2.9</td>
</tr>
<tr>
<td><strong>Waste Oxides</strong></td>
<td></td>
</tr>
<tr>
<td>Al$_2$O$_3$</td>
<td>3.1</td>
</tr>
<tr>
<td>Fe$_2$O$_3$</td>
<td>12.1</td>
</tr>
<tr>
<td>MnO$_2$</td>
<td>3.4</td>
</tr>
<tr>
<td>U$_3$O$_8$</td>
<td>1.1</td>
</tr>
<tr>
<td>NiO</td>
<td>1.5</td>
</tr>
<tr>
<td>Na$_2$SO$_4$</td>
<td>0.4</td>
</tr>
</tbody>
</table>
LEACHING APPARATUS

Condenser

Water

Crushed Glass