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Waste Form Development and Characterization in Pyrometallurgical Treatment of Spent Nuclear Fuel\*

J. P. Ackerman, C. Pereira, S. M. McDevitt, and L. Simpson

Argonne National Laboratory  
Chemical Technology Division  
9700 South Cass Avenue  
Argonne, IL 60439  
Telephone: (630) 252-8163  
Facsimile: (630) 252-5246

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# WASTE FORM DEVELOPMENT AND CHARACTERIZATION IN PYROMETALLURGICAL TREATMENT OF SPENT FUEL

J. P. Ackerman  
Chemical Technology Division  
Argonne National Laboratory  
9700 South Cass Avenue  
Argonne, IL 60439  
(630) 252-8163

C. Pereira  
Chemical Technology Division  
Argonne National Laboratory  
9700 South Cass Avenue  
Argonne, IL 60439  
(630) 252-9832

S. M. McDevitt  
Chemical Technology Division  
Argonne National Laboratory  
9700 South Cass Avenue  
Argonne, IL 60439  
(630) 252-4308

Lin J. Simpson  
Chemical Technology Division  
Argonne National Laboratory  
9700 South Cass Avenue  
Argonne, IL 60439  
(630) 252-9798

## ABSTRACT

Electrometallurgical treatment is a compact, inexpensive method that is being developed at Argonne National Laboratory to deal with spent nuclear fuel, primarily metallic and oxide fuels. In this method, metallic nuclear fuel constituents are electrorefined in a molten salt to separate uranium from the rest of the spent fuel. Oxide and other fuels are subjected to appropriate head end steps to convert them to metallic form prior to electrorefining. The treatment process generates two kinds of high-level waste — a metallic and a ceramic waste. Isolation of these wastes has been developed as an integral part of the process. The wastes arise directly from the electrorefiner, and waste streams do not contain large quantities of solvent or other process fluids. Consequently, waste volumes are small and waste isolation processes can be compact and rapid. This paper briefly summarizes waste isolation processes then describes development and characterization of the two waste forms in more detail.

## I. INTRODUCTION

Figure 1 shows that all high level wastes in the pyrometallurgical process originate in the electrorefiner. The input to the electrorefiner is chopped clad fuel or the metallic product from reduction of oxide fuels [1]. Outputs consist of two high-level waste streams (a metallic stream and a salt-borne stream) and relatively pure uranium metal.

These streams arise as follows: Spent fuel in steel baskets is introduced into the electrorefiner, whereupon its most reducing constituents — alkali, alkaline earth, rare-earth, and transuranic metals — are oxidized by actinide chlorides in the molten LiCl-KCl salt, forming metal

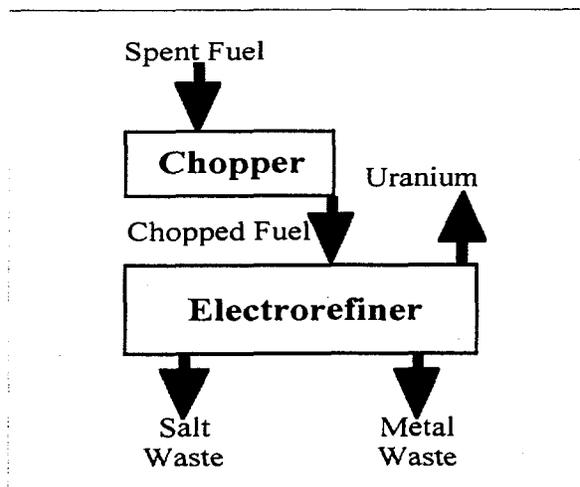


Fig 1. Material Flow in Electrometallurgical Treatment

chlorides. These chlorides are ultimately removed from the process as salt-borne waste.

A small quantity of a metal chloride, such as a uranium chloride or iron(II) chloride, is added to react with the fuel and maintain the actinide chloride concentration that is necessary to support electrotransport. An electric current is then passed to transport the bulk of the fuel (uranium) from the fuel baskets to a cathode, where it is harvested.

The metallic wastes are the cladding hulls and the "noble" metal fission products — that is, those metals that are less readily oxidized than the actinides. These metals are never oxidized. They remain largely in the baskets and make up the metal waste stream.

Alkali, alkaline earth, and rare-earth metals in spent fuel are oxidized in the electrorefining process and remain in the salt as cations. In addition, all of the transuranic fuel constituents are oxidized and removed with the salt waste stream, as is some of the uranium chloride in the electrolyte. Oxidation of these metals generates a stoichiometrically equivalent quantity of salt; this small quantity nicely matches the amount of salt that is unavoidably removed with the waste constituents in the salt-borne waste stream.

Conversion of the metal waste stream to a strong, corrosion-resistant metal waste form is relatively simple. The basket's contents are collected and melted to form an ingot that is basically a stainless steel-zirconium alloy. Any residual salt on the metal is vaporized in the melting step, collected, and returned to the electrorefiner. The ingot is the metal waste form. As discussed below, the composition of the alloy can vary over a reasonably wide range. Although the metal waste form is almost solely composed of the waste materials themselves, some fuels may require addition of small quantities of stainless steel or zirconium to adjust the waste composition.

The salt-borne waste stream is immobilized by sorption into an anhydrous aluminosilicate, zeolite A, which is used in the form of small granules. The crystal structure of zeolite A is made up of "cages" whose effective diameters are 7 Å and 11 Å. The waste salt is completely sequestered in these cages, and hence is effectively immobilized. The resulting material, called salt-occluded zeolite, is mixed with a lesser amount of glass frit, which serves as a bonding agent for consolidation of the particulate. Zeolite and frit are consolidated at elevated temperature and pressure to form a ceramic waste form. In the process, the zeolite is converted into sodalite, a naturally occurring mineral.

It is possible to extract the waste constituents from the salt and reuse the purified electrolyte in the electrorefiner by passing the salt through an ion-exchange bed of zeolite A that has been "preloaded" with pure electrolyte salt. The bed, which has concentrated waste constituent by exchange for lithium and potassium ions, becomes the feed materials to the waste fabrication process, resulting in a substantial reduction in volume of waste.

At present, most effort in waste form development is in support of a demonstration of electrometallurgical treatment that will be completed in July 1999 at Argonne's site in Idaho. Spent fuel from the Experimental Breeder Reactor (EBR-II) will be treated in this "hot" demonstration and both ceramic and metal wastefroms will be produced. Scaleup of waste form fabrication for the demonstration is the subject of companion papers at this conference [2-3]. Zeolite column technology is being developed in parallel to the demonstration; it will be the subject of a future paper.

## II. METAL WASTE FORM

### Nature of the Waste Form

The metal waste form comprises the metallic fuel components left behind in the electrorefiner charge basket. In addition to the cladding and noble metal fission products, zirconium from the fuel alloy and a minor amount of uranium will also be left in the metal waste stream. Stainless steel-zirconium (SS-Zr) waste form alloys have been developed for the consolidation and immobilization of these remnant metals [4-9].

Cladding hulls represent 80 to 99 percent of the remnant metallic wastes for the fuels being considered for electrometallurgical treatment. For EBR-II fuel, all cladding is stainless steel; the driver cladding is Type 316, D9, or HT9 stainless steel, whereas the blanket cladding is Type 304 stainless steel. In addition to stainless steel-clad fuels, Zircaloy-clad fuels can also be processed. Therefore, two baseline waste form compositions have been established: (1) stainless steel-15 wt% zirconium (SS-15Zr) for stainless steel-clad fuel and (2) Zircaloy-8 wt% stainless steel (Zr-8SS) for Zircaloy-clad fuel.

### Waste Form Fabrication

The basic process for waste form production involves melting and alloying the waste metals together in a high-temperature melting furnace. However, the process must also accommodate the evaporation of remnant electrorefiner salts clinging to the wastes. At present, the demonstration equipment for waste form melting does not accommodate salt collection, so the waste metals are processed through a separate distillation furnace at ~1000°C to remove the salts and then are melted in the casting furnace at 1600°C. A new melter is being designed to produce waste forms in a single processing step.

The existing waste form casting furnace is in the Fuel Conditioning Facility at ANL-West. It is an induction furnace previously used for injection casting of uranium-alloy fuel pins for EBR-II, but it has been modified to accommodate the metal waste form alloying process. The capacity of this furnace is ~5 kg; the molten metal cannot be poured from its crucible.

Development experiments for the SS-15Zr and Zr-8SS waste form alloys involved the generation of small-scale (20 to 40 g) and large-scale (1 to 5 kg) alloys using induction and resistance-element furnaces. The alloying temperature for both waste form alloys was determined to be ~1600°C. Because of the reactive nature of molten zirconium alloys, SS-Zr alloys must be melted under an inert argon atmosphere in yttrium oxide (Y<sub>2</sub>O<sub>3</sub>) ceramic crucibles. Most of the development alloys were generated using nonradioactive materials, but several

samples were generated that contained technetium, uranium, and plutonium.

The small-scale SS-Zr alloys were prepared in cylindrical yttria crucibles charged with Zr, stainless steels, and pure metal powders representing the noble metal fission products. A typical specimen size was ~15 mm diameter by 30 to 40 mm tall. These specimens were used for metallurgical examination and preliminary corrosion test samples. The samples were held at 1600°C for 1 to 2 h and cooled slowly (~7°C/min) during solidification to avoid unrealistic quench rates and to preserve the integrity of the yttria containment crucible. A high-temperature, tungsten element vacuum furnace with a controlled-atmosphere specimen chamber (Mo-30 wt% W alloy) was used for most small-scale alloys, but other furnaces were used for alloys containing radioactive materials.

Large-scale, nonradioactive ingots were prepared using a tilt-pour casting furnace connected to an inert-atmosphere glovebox [6]. The large-scale alloys generated in this furnace were used as source material for corrosion, mechanical, and thermophysical test samples. The alloy ingots were produced by slow-cooling the molten metal within an yttrium oxide crucible or by pouring the metal into casting molds preheated to 800°C.

#### Alloy Metallurgy

Details of the SS-Zr alloy development and metallurgy have been reported elsewhere [4-9], so only a brief summary is presented here. Because the SS-15Zr alloy will be made as part of the EBR-II demonstration, the most development effort has been devoted to this alloy. However, the metallurgy of as-cooled Zr-8SS has been characterized. Both of these metal waste form alloys are multiphase materials containing mixtures of metal and intermetallic phases.

The SS-15Zr alloy composition is near a binary eutectic between an iron-based solid solution phase ( $\alpha$ -Fe) and an iron-zirconium intermetallic; the eutectic temperature is near 1330°C. Figure 2 shows the microstructure from a typical SS-15Zr alloy. These alloys contain a two-phase structure. The dark phase is a ferritic Fe solid solution, and the bright contrast phase is a Laves-type intermetallic ( $AB_2$  crystal structure) designated as  $Zr(Fe,Cr,Ni)_{2+x}$ .

The Zr-8SS alloy microstructure contains a primary zirconium metal ( $\alpha$ -Zr) phase and a matrix of several secondary phases; it is not near a eutectic. The Zr-8SS solidus temperature is between 900°C and 1000°C, and the liquidus temperature is ~1500°C. The  $\alpha$ -Zr regions typically contain ~95 atom % Zr and an assortment of soluble metals. The multiphase matrix materials include secondary  $\alpha$ -Zr and intermetallic compounds that have been identified as  $Zr_2(Fe,Ni)$  and  $Zr(Fe,Cr)_2$ .

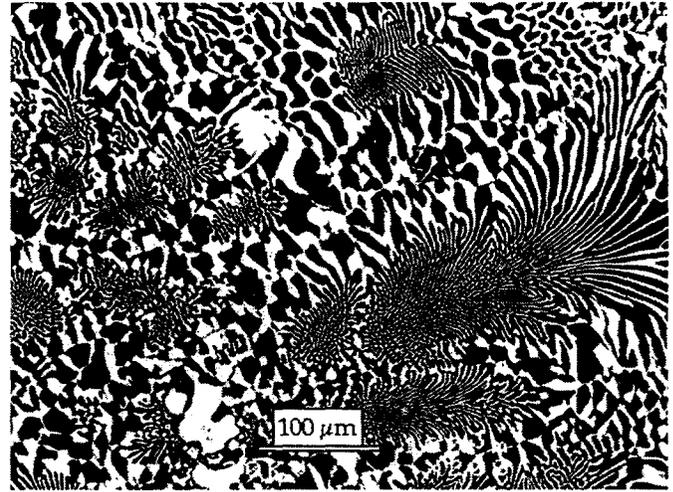


Fig. 2. Backscattered Electron Image (500x) of the SS-15Zr Eutectic Structure [6]. The dark phase is an iron solid solution, and the bright phase is the  $Zr(Fe,Cr,Ni)_{2+x}$  intermetallic.

To simulate the presence of the noble metal fission products, nonradioactive isotopes of several of the most representative noble metal elements were added to SS-15Zr and Zr-8SS alloys. These were ruthenium, niobium, molybdenum, palladium, and silver. In addition, radioactive samples were prepared using small quantities of technetium-99 ( $^{99}Tc$ ), the longest-lived isotope in the waste form.

The expected fission product content in production waste form alloys lies between 0.01 and ~4 wt% total noble metals, depending on the burnup of the treated fuel. For all of our alloys with additions below 4 wt% of any individual noble metal, all noble metals were observed to be in solution, distributed throughout the alloy phases. Because no discrete fission product phases are formed, fission product release from SS-Zr alloys will be controlled by the corrosion behavior of the SS-Zr alloys.

#### SS-15Zr Composition Constraints

It is evident that some zirconium metal must be added to achieve a nominal waste form composition of 15 wt% Zr from EBR-II waste. The driver waste will include a significant amount of zirconium (approx. 10 to 15 wt% Zr), but the blanket waste will not contain any zirconium. Therefore, a series of experiments was carried out to determine an allowable zirconium composition range. Simulated waste form alloys were generated with stainless steel, 1 to 4 wt% noble metals, and 0, 5, 10, 15, and 20 wt% zirconium. Previous experience with these compounds showed there was not a significant difference in corrosion resistance with Zr content [6-7,9]. We found that:

- 1) Zirconium plays a critical role in the incorporation and retention of fission products in SS-15Zr. The  $Zr(Fe,Cr,Ni)_{2-x}$  intermetallic is the "preferred" location for several noble metal fission products that have low solubility in iron (e.g., Nb, Pd, and Ag).
- 2) Without zirconium present to form the intermetallic, these same fission products will precipitate into noble metal-rich phases.
- 3) For alloys with only 5 wt% Zr, the microstructure has three primary phases: a) austenitic iron, b) ferritic iron, and c) the  $Zr(Fe,Cr,Ni)_{2-x}$  intermetallic. The intermetallic becomes saturated at relatively high noble metal compositions (approx. 2 to 4 wt% NM). If the noble metal content is low (as would be true for EBR-II blanket fuel), then all fission products may be fully dissolved in the alloy phases at 5 wt% Zr.
- 4) In practice, we find that 2.5-kg ingots with a nominal 15 wt% Zr charge have local compositions which vary from ~12 to ~18 wt% Zr.

Therefore, the specification for the SS-15Zr metal waste form was set at 5 to 20 wt% Zr (i.e., no less than 5 wt% Zr and no more than 20 wt% Zr at any location within a waste form ingot). Also, the nominal charge composition will be based on fission product content. For high-burnup fuels (i.e., ~4 wt% fission products), the charge shall not contain less than 15 wt% Zr. For low-burnup fuels (i.e., <0.5 wt% fission products), the nominal waste form composition may be as low as 8 wt% Zr. The specification is intentionally broad to accommodate a variety of fuels. Waste forms with nominal compositions between ~8 and ~15 wt% Zr are expected to always have local concentrations that are within the 5 to 20 wt% Zr specification.

### III. CERAMIC WASTE FORM

#### Nature of the Waste Form

The ceramic waste form is a composite of sodalite and borosilicate glass. Sodalite,  $Na_8Al_6Si_6O_{24}Cl_2$ , is one of the few chloride-bearing minerals and therefore a natural candidate for the halide-bearing phase in the ceramic waste form. Waste ions occupy specific crystallographic locations in the waste form, just as sodium and chloride ions do in natural sodalite. The borosilicate glass phase binds the sodalite crystals together into a monolithic waste form. The rare earths and actinides also form minor amounts of stable secondary phases that arise from the reaction of these chlorides with residual water in the zeolite from which the waste form is fabricated.

Sodalite does not readily sorb or exchange waste ions into its structure. In the waste form fabrication process,

therefore, the waste salt is first incorporated into zeolite A,  $Na_{12}Al_{12}Si_{12}O_{48}$ , which is then converted to sodalite by heat when the waste form is consolidated. The addition of glass promotes this reaction, but its primary role in the waste form is as a binder.

#### Waste Form Fabrication

The ceramic waste form fabrication process is illustrated in Fig. 3.

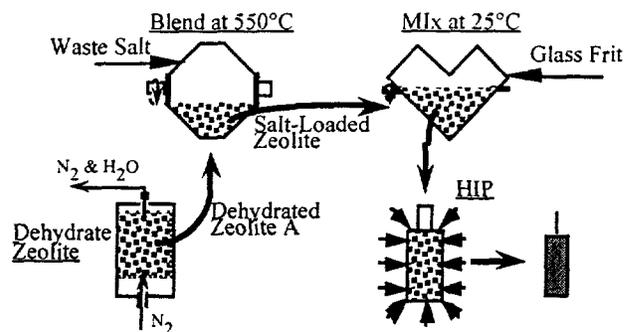


Fig. 3. Ceramic Waste Form Fabrication Process

There are four major unit operations: dehydration, hot blending, cold mixing, and hot pressing. Because zeolite A is extremely hydrophilic, granular zeolite A must be dehydrated prior to contacting with molten salt. In the first step of the fabrication process, a stream of dry nitrogen is passed through a bed of zeolite while it is heated in a stepwise manner to 525°C. The moisture content of the dehydrated zeolite is typically less than 0.3 wt%.

The dehydrated zeolite is transferred to a high-temperature blender, where it is combined with ground waste salt. The weight fraction of salt in the mixture is between 10 and 12%. This corresponds to the amount of chloride that can be fully incorporated into the sodalite structure that forms when the zeolite is heated in the final unit operation. The zeolite-salt mixture is tumbled at 550°C; a period of 20 hours is more than adequate to effect essentially complete salt sorption. The amount of free (unsorbed) salt is typically below 0.03 wt% of the total sample.

After hot blending, the salt-loaded zeolite is mixed with glass frit at ambient temperature. As in the hot blending step, the components are tumbled together to achieve a uniform mixture. It may be that the blending and mixing steps can eventually be combined. Next, the mixture is loaded into stainless steel canisters and compacted to ~60% of the final waste form density. The loaded canisters are evacuated, heated to 500°C, and seal-welded. Canister and contents are hot isostatically pressed (HIPed) at a maximum temperature of 900°C and a pressure of 25 ksi for one hour. Under these conditions the zeolite is transformed to sodalite.

The glass melts and flows between the sodalite particles to yield a monolithic structure.

#### Reference Waste Form Composition

During the development of the ceramic waste form, two candidate materials were investigated extensively: a composite of salt-loaded zeolite and glass ("glass-zeolite"), and a glass-sodalite composite. Each of the waste forms possesses an intrinsic advantage. Zeolite A has three times the capacity of sodalite for chloride because of its more open cage structure. Consequently, the volume of waste form that is generated per mass of salt to be discarded is two to three times less for the zeolite if the amount of glass is the same. On the other hand, sodalite is much more thermally stable—salt-loaded zeolite tends to convert to sodalite at elevated temperatures, with concomitant expulsion of salt. This superior thermal stability makes processing much more flexible and dependable. Both waste forms show excellent waste-ion retention in leach tests; the differences that do exist are minor.

Although the raw materials and processing steps for both waste forms are similar, only one of the composites could be selected for production and testing with "hot" salt as part of the July 1999 process demonstration. For development and comparison purposes, compositions and process conditions were established for each waste form, and a baseline was defined for each. Baseline compositions were 50:50 mixtures by weight of glass and salt-loaded zeolite (21 wt% salt) or sodalite (10 wt% salt) consolidated by a HIP process appropriate for each material. The same glass composition was used for both forms. All of the waste forms were characterized by X-ray diffraction, density, porosity, and both cesium and chloride releases in 3-day Materials Characterization Center type 1 (MCC-1) leach tests. Twenty eight-day MCC-1 tests were carried out with samples that showed good three-day durabilities.

Sample-to-sample variation in the properties of the baseline waste forms was small. Densities were typically between 2.3 and 2.45 g/cc. Accessible porosity was typically less than 0.5%. Minor secondary crystalline phases were sometimes observed for both waste forms. For the glass-sodalite composite, these included NaCl and nepheline. For the glass-zeolite composite, minor phases were NaCl, microsommitte, and sodalite.

Cesium releases, shown in Fig. 4 as cesium fractional mass release (Cs-FMR, the fraction of the cesium that was originally in the waste form that is released to solution) were typically an order of magnitude higher for the zeolite composite in three-day tests, somewhat less in 28-day tests. The releases are considered quite acceptable, however, even for the zeolite composite. Average chloride releases for the two waste forms were comparable. Alkaline earth and rare-earth releases were slightly lower for the zeolite composite, but two orders of magnitude lower than

the cesium releases measured for either composite. Because cesium is the fission product most readily leached from these and most other waste forms, its release was used as the primary performance criterion during waste form development.

In an attempt to enhance waste form performance, a number of process parameters were varied, including glass content, glass composition, salt content, temperature and pressure schedules in the HIP process, and the inclusion of additives in the waste form mixture. Fourteen commercial glasses were tested. One composition yielded the best properties for both waste forms; it was used in the baseline formulations. From cesium release rates in 3- and 28-day tests, the glass content had a much stronger effect on the behavior of the zeolite composite, as can be seen in Fig. 5.

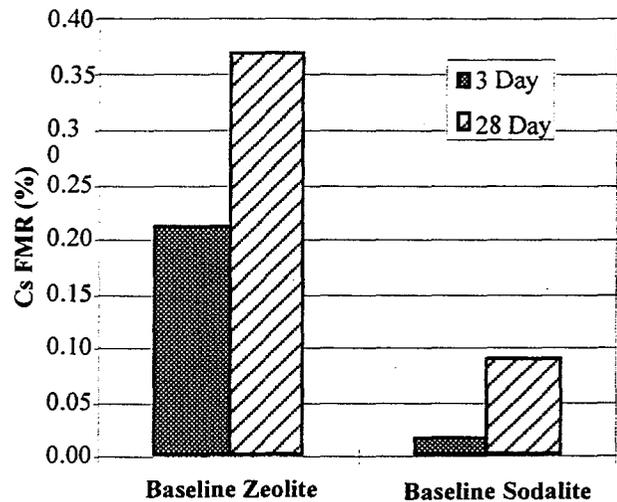


Fig. 4. Comparison of Cs Release from Baseline Samples

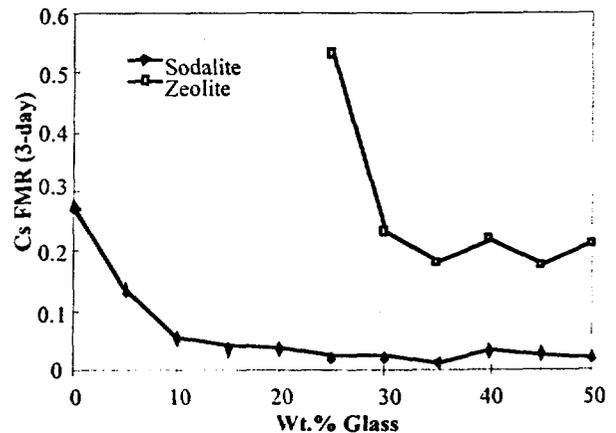


Fig. 5. Effect of Glass Content on Waste Form Behavior

Neither the salt content nor additives — including silica, alumina, and chabazite — significantly altered the waste form behavior, seen in the results of 3- and 28-day MCC-1 tests.

The selection of sodalite as the reference waste form was based finally on its substantially greater flexibility in processing and somewhat better overall fission product retention. Part of the intrinsic volume difference between sodalite and zeolite waste forms is compensated by the lower acceptable glass content of the sodalite composite. As can be seen in Fig. 5, cesium retention in the sodalite waste form is relatively constant over a range of glass compositions. Although a reference composition has been established, refinements to the reference process continue to be made in order to support scaleup and remote handling for the demonstration.

#### IV. WASTE FORM QUALIFICATION

To be accepted into the DOE Civilian Radioactive Waste Management System, the ceramic and metal waste forms must be reviewed and deemed to meet certain minimum physical characteristics and consistency guidelines as presented in the Waste Acceptance System Requirements Document (WASRD) for spent nuclear fuel and high-level waste [10]. For qualification and disposition of the waste forms in a geologic repository, additional information beyond product consistency may be needed to establish corrosion behavior for acceptance and to allow a "repository-relevant performance assessment."

We have begun qualification testing of the ceramic and metal waste forms and have identified a suite of tests needed to characterize them and determine their corrosion mechanisms. Our approach is based upon the WASRD guidelines, the testing methodology outlined in the American Society for Testing and Materials document C1174-91 [11], and results from our previous work in evaluation of appropriate methods for waste form testing. [9,12-13] An initial task matrix has been formalized.

Qualification testing will be used to determine the waste forms' physical attributes, initial and accelerated corrosion behavior, actual "service condition" behavior, and expected long-term behavior. The latter may be based, in part, on "natural analog" comparisons and canister material studies [12-14]. The testing will take place over periods of up to 20 years. The goal of behavior testing is to describe long-term behavior of the waste forms well enough to allow qualification for acceptance into the Civilian Radioactive Waste Management System.

Corrosion mechanisms and mechanisms of radionuclide loss will be inferred from the test results. Because a link may need to be established between long-term, repository-relevant behavior and results of short-term

consistency tests, the results will also be used to develop predictive behavior models and to evaluate the capability of the waste forms to meet yet-to-be-determined radionuclide containment requirements for periods of 10,000 years or more.

#### The Ceramic Waste Form

Development of test methods and data for the qualification of the ceramic waste form was carried out with glass-bonded zeolite samples [12-14]. This work resulted in a conceptual model for the mechanistic release of radionuclides based upon corrosion results from product consistency tests (PCT), vapor hydration tests (VHT), and MCC-1 tests. The results from this initial testing suggest that several factors influence the corrosion of the ceramic waste form, including test duration, sample surface area to leachant volume, temperature, leachant (composition, pH), waste form composition, and the type of corrosion mechanisms involved (ion exchange, dissolution). The test matrix and choice of specific tests for the present effort to qualify the reference glass-bonded sodalite waste form are based on this earlier work.

The information obtained from the present qualification effort should be useful both for modeling waste form reactions and for understanding the effects of parameters under controlled conditions. The response of granulate sodalite crystals with clay binders used in the reference ceramic waste form samples during exposure to potential and/or simulated repository conditions (i.e., aqueous corrosion, irradiation exposure, etc.) may be quite distinct from that of borosilicate glass and crystalline materials, such as zeolite A. The largest suite of tests will involve nonradioactive "reference" ceramic waste form samples, but tests will also be performed with reference samples containing uranium, plutonium, and actual radionuclides from the demonstration electrolyte.

#### The Metal Waste Form

Several test methods have been used to investigate the mechanism of corrosion processes associated with the metal waste form. [6-7,9] Electrochemical polarization and high-temperature immersion tests have provided the most useful information and will be the focus of our future qualification efforts.

MCC-1 tests were performed on metal waste form ingots. The ingots remained shiny and visibly unreacted after 10,000 hours in MCC-1 tests with 90°C simulated J-13 well water. Because of the exceptional corrosion resistance of the metal waste form, neither 90°C MCC-1 tests nor vapor hydration tests at 200°C can provide the quantitative product consistency information required for acceptance. However, initial results indicate that we might be able to enhance corrosion sufficiently to identify major

changes among production samples by using higher-temperature (200°C) MCC-1 tests.

The PCT was also investigated with a view to obtaining usable consistency information (adequate corrosion rates) for the acceptance process. The metal waste form was cooled with liquid nitrogen and crushed to provide the necessary 100-to-200-mesh (75-to-150- $\mu$ m) particle size distribution. Even though the crushing induced strain in the sample and significantly increased the exposed surface area, the normalized release of any specific element from the metal waste form was 10 to 10,000 times lower than that of glass waste forms tested under similar PCT conditions. Although the metal waste form performed well in the product consistency tests, the use of the test for product consistency and performance assessment may be limited by the extremely difficult sample preparation required and the significant amount of damage imposed upon the samples.

The vapor hydration tests provided useful information about the effect of unsaturated repository conditions on corrosion rates and alteration phase formation. The metal waste form's corrosion layer forms a protective barrier that impedes the continued corrosion of the surface. Vapor hydration and electrochemical corrosion rate test results show that the main corrosion mechanism is oxidation of the surfaces, and that the fission products are principally found in intermetallic phases. These intermetallic phases appear to be even more corrosion-resistant than stainless steel.

Electrochemical polarization measurements have provided useful corrosion information that can be compared with measurements from other metal materials [13,14]. Electrochemical polarization measurements are relatively simple to perform and provide an accurate corrosion rate for metal specimens. The simplicity of the technique allows a broad range of environmental conditions to be investigated. The technique is a standard in the metallurgical field, is often used to provide product consistency data, and would probably provide the most relevant measure of product consistency as it applies to acceptance of the metal waste form into the DOE Civilian Radioactive Waste Management System.

All test results indicate that both the stainless steel-rich and the zirconium-rich metal waste forms are exceptionally corrosion-resistant. Their behavior is similar to that of pure stainless steel and zirconium, respectively. Fission products are, therefore, unusually well retained.

## V. SUMMARY

Electrometallurgical treatment results in just two high-level waste forms; both are concentrated, low-volume forms that arise in the electrorefiner. The electrorefiner salt wastes are disposed of as a glass-bonded sodalite ceramic

waste form; it is made by sorbing the salt-borne fission products and any excess salt into zeolite A, mixing with glass powder, and consolidating at elevated temperature and pressure. A metal waste form is made by melting together the cladding hulls, noble metal fission products, and any zirconium that may have been present in the fuel alloy. Both waste forms evince excellent mechanical properties and durability; the metal waste form is exceptionally corrosion-resistant. Because the basic development of these waste forms is complete, most effort has been redirected to the qualification process.

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