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INNOVATIVE TECHNOLOGY
Summary Report

Vitrification of Ion Exchange Materials

Tanks Focus Area

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Summary Report

Vitrification of Ion Exchange Materials

OST Reference #81

Tanks Focus Area

Demonstrated at
Savannah River Site
Aiken, South Carolina

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Summary Report

Purpose of this document

Innovative Technology Summary Reports are designed to provide potential users with the information they need to quickly determine if a technology would apply to a particular environmental management problem. They are also designed for readers who may recommend that a technology be considered by prospective users.

Each report describes a technology, system, or process that has been developed and tested with funding from DOE's Office of Science and Technology (OST). A report presents the full range of problems that a technology, system, or process will address and its advantages to the DOE cleanup in terms of system performance, cost, and cleanup effectiveness. Most reports include comparisons to baseline technologies as well as other competing technologies. Information about commercial availability and technology readiness for implementation is also included. Innovative Technology Summary Reports are intended to provide summary information. References for more detailed information are provided in an appendix.

Efforts have been made to provide key data describing the performance, cost, and regulatory acceptance of the technology. If this information was not available at the time of publication, the omission is noted.

All published Innovative Technology Summary Reports are available onthe OST Web site at http://ost.em.doe.gov under "Publications."

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SECTION 1

SUMMARY

Technology Summary

Most radioactive tank waste can be treated and disposed of as low-activity waste (LAW) if high-activity radionuclides such as cesium are removed. Disposal as LAW can generate significant cost savings over disposal as high-level waste (HLW).

Ion exchange is a process that safely and efficiently removes radionuclides from tank waste. The basic principle of ion exchange technology is that ions in a solution are exchanged with ions on a solid. A solid (called a sorbent) with high affinity for a particular ion is said to be selective for that ion. When a waste solution contacts a sorbent, the sorbent adsorbs certain ions until it becomes loaded. Depending on the sorbent, adsorbed ions may be washed from the ion exchange material by contacting the sorbent with a solution of a different pH. However, some sorbents bind certain ions very strongly and are nonregenerable.

Cesium and strontium account for a large portion of the radioactivity in waste streams from U.S. Department of Energy (DOE) weapons production. Crystalline silicotitanate (CST) is an inorganic sorbent that strongly binds cesium, strontium, and several other radionuclides. Developed jointly by Sandia National Laboratory and Texas A&M University, CST was commercialized through a cooperative research and development agreement with an industrial partner. Both an engineered (mesh pellets) and powdered forms are commercially available.

Cesium removal is a baseline in HLW treatment processing. CST is very effective at removing cesium from HLW streams and is being considered for adoption at several sites. However, CST is nonregenerable, and it presents a significant secondary waste problem. Treatment options include vitrification of the CST, vitrification of the CST coupled with HLW, direct disposal, and low-temperature processes such as grouting. The work presented in this report demonstrates that it is effective to immobilize CST using a baseline technology such as vitrification.

Vitrification produces a durable waste form. CST vitrification was not demonstrated before 1996. In FY97, acceptable glass formulations were developed using cesium-loaded CST obtained from treating supernatants from Oak Ridge Reservation (ORR) tanks, and the CST was vitrified in a research melter at the Savannah River Technology Center (SRTC). In FY98, SRS decided to reevaluate the use of in-tank precipitation using tetraphenylborate to remove cesium from tank supernatant and to consider other options for cesium removal, including CST. Hanford and Idaho National Engineering and Environmental Laboratory also require radionuclide removal in their baseline flowsheets.

Demonstration Summary

Crucible studies and bench-scale demonstrations were conducted to prepare glass containing only CST and glass containing both CST and sludges from different sites. The Tanks Focus Area studies at SRTC are briefly described below.

Oak Ridge National Laboratory (ORNL) Demonstration

The cesium removal demonstration at ORR used CST as the ion exchange sorbent. From September 1996 through June 1997, the cesium removal system processed 31,000 gal of Melton Valley Storage Tank waste. Approximately 15% of the loaded CST from the demonstration was sent to SRTC for a vitrification demonstration (see Figure 1).

Crucible studies performed in FY96 developed an acceptable glass formulation for CST-only glass. Additional crucible tests were conducted to increase the glass waste loading and optimize the glass composition. These studies demonstrated that glass could be loaded with CST up to 60 weight percent (wt %).

A radioactive demonstration carried out in August 1997 processed radioactive CST in the SRTC shielded-cells melter system. The remotely operated process included the preparation of the melter feed, vitrification at 1150°C, and analysis of the glass product for durability. The campaign immobilized approximately 20 kg of cesium-loaded CST in 80 h of shielded-cells melter operation. The glass contained 50–55 wt % CST.

Figure 1. Vitrified material.

SRS Glass Formulation

Removal of cesium, strontium, and plutonium from tank supernatant by ion exchange using CST is among the alternative pretreatment methods to in-tank precipitation being considered at SRS. This inorganic material was shown to selectively remove these elements from supernatant. As part of a Tanks Focus Area project, cesium-loaded CST sorbent was combined with the Defense Waste Processing Facility (DWPF) HLW sludge and glass-forming chemicals and then vitrified. Initial glass formulation efforts indicate that a reasonable waste loading of both sludge and CST can be achieved in DWPF glass.

Hanford Glass Formulation

At the Hanford Site, 177 underground storage tanks contain approximately 250,000 m^3 of waste generated from weapons production. The bulk of the cesium is contained in the supernatant and saltcake. Waste pretreatment is expected to include cesium removal. Though no specific technology has been defined, CST is a candidate for the ion exchange sorbent. A glass formulation incorporating CST and Hanford sludge was developed in FY97.

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Other

All published Innovative Technology Summary Reports are available on the OST Web site at http://ost.em.doe.gov under “Publications.” The Technology Management System, also available through the OST Web site, provides information about OST programs, technologies, and problems. The OST reference number for Vitrification of Ion Exchange Materials is 81.

SECTION 2

TECHNOLOGY DESCRIPTION

Overall Process Definition

The primary technology demonstrated under this project is the vitrification of a CST waste form in the SRTC research melter. The CST contains significant quantities of titanium. This element has historically been difficult to incorporate into the glass matrix. The major objective was to produce a CST waste form suitable for disposal at an appropriate repository.

The SRTC research melter is a joule-heated melter. Inconel® electrodes provide the power to maintain a melt pool temperature of 1150°C. The cylindrical melt chamber is 8 inches in diameter and 6 inches deep and holds approximately 10 kg of glass. Two additional heaters are located in the melter above the melt pool to provide supplemental heat, which increases the melt rates by vaporizing any water from the feed. Tilting the entire melter initiates glass pours. The glass flows from the melt pool, through a riser cut in the refractory, and out a heated pour spout into 0.5-L stainless steel beakers. Figure 2 is a schematic of a typical joule-heated melter.

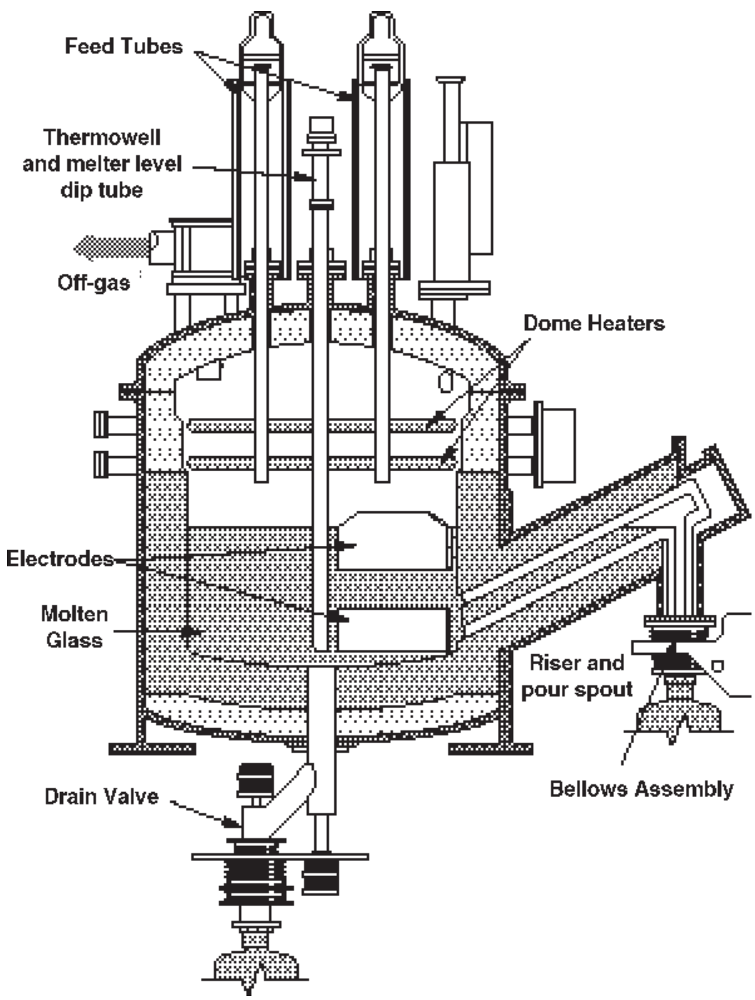

Figure 2. Joule-heated melter.

SRTC performed glass durability tests using the product consistency test (PCT). The PCT is a crushed-glass leach test that measures the releases of boron, lithium, sodium, and silicon from the glass in 90°C ASTM Type 1 water over a period of seven days. The PCT creates accelerated leaching conditions.

System Operation

Table 1 summarizes the system operation requirements for Shielded Cells Facility tests and melter system demonstrations.

Table 1. System operations requirements

Special operational parameters	Due to the highly radioactive nature of the loaded CST, all radioactive testing is performed remotely in a shielded facility. The formulation for CST vitrification must produce a homogeneous waste form with durability, liquidus, temperature, viscosity, and characteristics compatible with disposal requirements. The process generates highly radioactive glass. The waste form must be disposed at an appropriate repository. CST contains a significant amount of titanium, an element known to cause crystallization in borosilicate glass. Glass waste forms with elevated titanium must be obtained without crystals. The glass produced must be more durable than the Environmental Assessment (EA) glass. The HLW acceptance criteria state that any glass produced must be at least two standard deviations below the results of the EA glass to be accepted.
Materials, energy, other expendable items	The melter must receive a homogeneous feed, and the melter feed system must withstand the high radiation fields. Costs associated with vitrification suggest that waste loading should be as high as possible without exceeding waste acceptance criteria.
Personnnel required	The personnel operating the processes need to have knowledge of the technology and remote-handling skills.
Secondary waste stream	Volatilization in the melter creates the potential for cesium in the off-gas. However, the melter recovers and recycles cesium in the off-gas.
Potential operational concerns and risks	Remote handling is required due to the radioactive components in the tank waste.

SECTION 3

PERFORMANCE

Demonstration Plan

Table 2 summarizes the various CST immobilization and demonstration activities conducted during this project. The project included crucible studies and a large-scale demonstration of CST vitrification using loaded CST from Oak Ridge Melton Valley Storage Tanks supernatant.

Table 2. CST vitrification demonstration summary

Year	Demonstration description	Major objectives
1996	CST-only glass formulation	Produce acceptable glass formulation to accommodate elevated levels of titanium. Identify the maximum CST waste loading.
1997	ORNL CST vitrification	Vitrify in a manner to keep emissions of hazardous material well below regulatory levels even in a much larger melter system. Meet the waste acceptance criteria for disposal at Nevada Test Site. Determine whether feed systems developed for HLW sludges are appropriate for CST delivery.
1998	CST-DWPF sludge glass and CST-Hanford sludge glass	Determine the type and amount of glass-forming chemicals necessary to combine nonradioactive sludge and unloaded CST to in a glass. Develop glass formulations with high CST and sludge waste loading. Produce durable glass using actual waste.

System Performance

Demonstration results are summarized in Table 3 and discussed below. These studies demonstrated acceptable glass formulations for Oak Ridge, Hanford, and Savannah River Site (SRS) with waste loadings as high as 60%.

CST-Only Glass Formulation

Table 4 summarizes scoping studies performed with unloaded CST and reagent grade glass chemicals. Several variations in the borosilicate glass composition at various waste loadings were prepared and tested. The samples were vitrified in an 1150°C furnace for 4 h. Afterwards the samples were transferred to a 900°C furnace for 2 h.

A Plackett-Burman screening design was subsequently used to increase waste loadings and determine a more optimal frit. Twelve glasses were prepared and analyzed to determine the crystalline content. Durability testing indicated that all of the compositions were significantly more durable than the EA glass. The sample that did not contain crystals at 65 wt % CST was the best formulation.

ORNL CST Vitrification

Loaded CST from treating Oak Ridge Melton Valley Tanks supernatant was successfully vitrified. Before the vitrification in the SRTC Shielded Cells Facility, a glass formulation was needed to incorporate ORNL CST sorbent without crystallization. Five glass formulations were tested for vitrifying dried CST. One of the five formulations produced acceptable glass and easily incorporated CST loadings of 40 and 55 wt %.

Table 3. Results of CST immobilization technology demonstrations

Technology	Demonstration elements	Results
CST-only glass formulation	CST and reagent-grade glass-forming chemicals are mixed and vitrified in a platinum crucible.	Scoping studies indicated that durable glasses could be produced with 50 wt % CST. Plackett-Burman design screening experiments showed that durable glasses with up to 65 wt % CST were achievable.
ORNL CST vitrification	Crucible studies were conducted using five formulations for pretreated (but unloaded) CST at waste loadings of 50 wt %. Efforts were made to correct problems associated with plugging in the slurry feed system for the melter. Slurry feed and dry feed melter systems were tested.	Only one of the five formulations was acceptable. The formulation could easily incorporate ORNL CST loadings between 40 and 55 wt % dry CST. For the slurry feed, a pumping/mixing setup using a recirculation loop successfully operated at a rate of approximately 27 cm^3/min. The dry feed system was selected for use during the radioactive melter campaign.
CST-DWPF sludge glass	Both the sludge and CST loading were varied, but the sludge waste and vitrification loading was kept close to the current DWPF loading of 28 wt % sludge oxides.	Durable glass was made with up to 20 wt % CST and 30 wt % DWPF sludge (no crystals were detected).
CST-Hanford sludge glass	The simulated Hanford sludge and CST loading were varied. The target sludge loading was 45 wt % oxides and a CST loading of 7.5 wt %.	Preliminary results demonstrate a durable borosilicate glass waste form with 45 wt % sludge and 7.5 wt % CST loadings.

Table 4. Summary of purpose and results of scoping tests for CST glass

Measurement	Method	Result
Durability	PCT test	Durability generally tends to improve with increased waste loading. The 50 wt % CST loading had the highest acceptable durability with no crystallization.
Density	Bouyancy	Glass density increases with increased waste loading.
Crystallinity	X-ray diffraction	Some glass formulations up to 50 wt % CST did not form crystals after heat treatment.
Viscosity	Glass samples from the 1150ºC furnace were poured into a stainless steel pan simultaneously with a glass of known viscosity	The 40, 50, and 60 wt % CST glasses all had acceptable viscosities.
Metal content	Scanning electron microscopy/ electrical conductivity	No reduced metals were found, indicating that the glass would be compatible with the melter.

For the large-scale CST immobilization demonstration, the melter feed system remotely mixed the CST and chemicals to obtain a homogeneous feed for the melter. Early efforts focused on the delivery of the CST resin to the melter as slurry. Unfortunately, the low melter-feed delivery rate allowed solids to leave suspension. As shown in Figure 3, the melter feed system was modified for the demonstration as follows:

- A recirculation loop using a diaphragm pump was added. Serious plugging was encountered in the recirculation loop (especially in the needle valve). The needle valve was replaced with a ball valve.

- The diaphragm pump and the agitator in the feed drum agitated feed material through recirculation. A 90-degree elbow added to the system increased the turbulence in the recirculation line.

- The implementation of a recirculation loop in conjunction with agitation demonstrated that CST and glass formers could produce a melter feed rate of approximately 27 cm^3/min.

- Since the slurry feed system required the addition of water to the CST, the condensate generated during vitrification generated additional waste. Therefore, a dry feed system was also tested and ultimately selected.

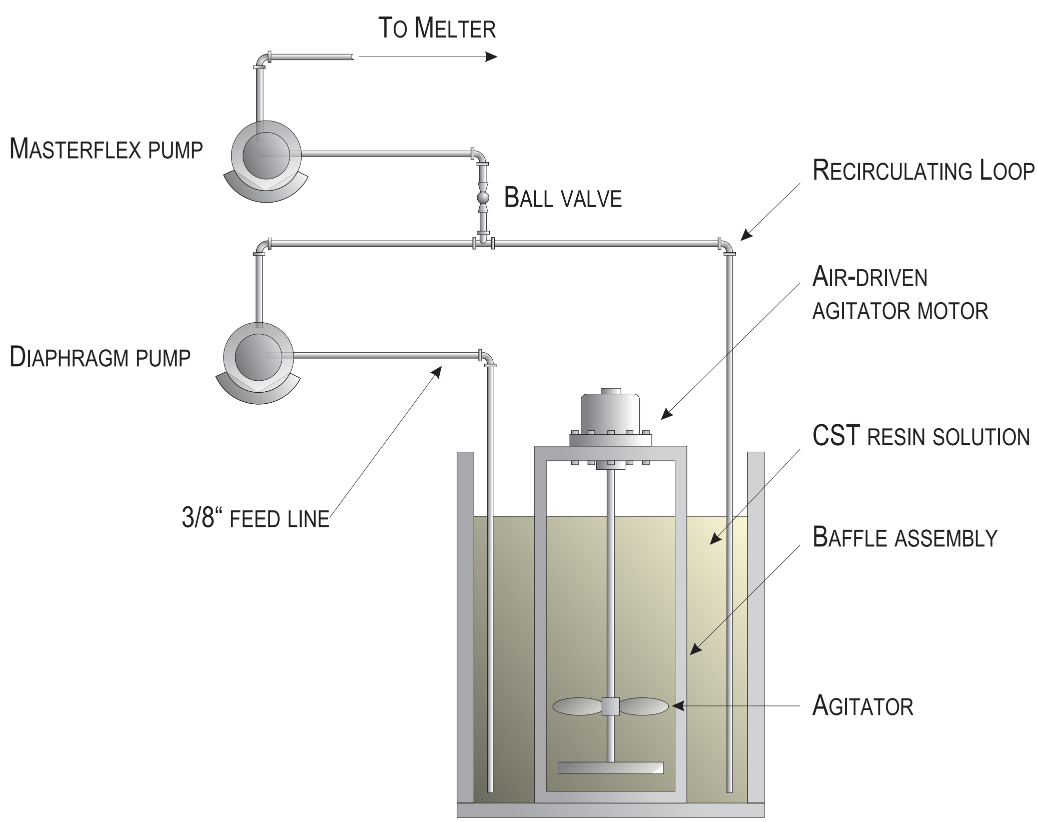

Figure 3. Schematic of the melter feed tank system in the shielded cells.

Before the initiation of CST feeding, the melter contained approximately 10 kg of glass. During the CST campaign, the melter ran continuously for 85 h, producing nearly 30 kg of glass. Since the melter contained a different glass prior to the start of the CST run, samples of CST glass were taken from the last can poured during the campaign. Tests determined that the glass had acceptable durability.

CST-DWPF Sludge Glass

Acceptable glass formulations were developed to incorporate CST into DWPF sludge glass. Scoping studies quantified the type of frit required to produce a durable CST-DWPF sludge glass with the appropriate liquidus and viscosity processing properties. Ten samples were prepared and separately vitrified in platinum crucibles. Each of these samples represented a different glass composition.

After approximately 4 h at 1150°C, the glasses were removed and either air-cooled to room temperature or slow-cooled to room temperature. Some of the glass samples were removed from the furnace and poured to determine the viscosity range of the glass. After the glasses had completely cooled, they were removed from the crucible and ground for analyses. All glasses were determined to be acceptable.

After the initial scoping studies, a Plackett-Burman screening experiment was performed in an attempt to increase the CST loadings. Initially, 12 compositions were tested. A waste loading of 28 wt % sludge oxides and 5 wt % CST was chosen for the radioactive demonstration.

The radioactive demonstration was conducted using actual sludge from SRS Tank 51 along with CST from the demonstration at ORNL. The glass-forming chemicals were combined with the sludge and CST, poured into a platinum crucible, and then vitrified in a muffle furnace. After approximately 4 h at the melt temperature, the crucible was removed from the muffle furnace and poured into a stainless steel pan. There were no visible differences between the radioactive and nonradioactive glasses. The glass poured easily from the crucible and was in the same viscosity range as the nonradioactive glass sample, indicating that it would be acceptable in the DWPF melter.

CST-Hanford Sludge Glass

Glass formulations were also developed for CST mixed with Hanford Site HLW sludge. The preliminary results indicate that sufficient quantities of CST and HLW sludge can be loaded into a borosilicate glass to make this approach competitive with other HLW immobilization options.

SECTION 4

TECHNOLOGY APPLICABILITY AND ALTERNATIVES

Competing Technologies

Vitrification of the cesium-loaded ion exchange material offers a number of benefits:

- It is less expensive than many of the technologies available.
- It offers a large volume reduction.
- It produces a very durable waste form (suggested as the “best demonstrated available technology” for several waste streams).
- It is an established technology.
- It can be used for a variety of waste streams.
- It produces a waste form that is resistant to radiation damage.

Although vitrification is generally considered to be the technology of choice for immobilization of highly radioactive materials, a number of other technologies are available or are under development. Table 5 compares the most predominant options to conventional vitrification.

Table 5. Conventional vitrification vs other technologies

Technology	Primary advantages	Primary disadvantages
Cement-based processes	Cost	Durability, volume
Metal-based process	Less radiation damage	Cost, not established
Plasma vitrification	Possible durability	Cost, not established
Ceramic-based process	Possible durability	Cost, not established
Direct disposal	Cost	Durability

Vitrification is a very flexible technology. With appropriate additives, melt conditions, and off-gas system, it can be used for solidification of most waste streams. Vitrification is currently used for solidification of domestic waste, hazardous waste, low-level radioactive waste, high-level radioactive waste, and mixed waste.

A cost comparison between vitrification and the other technologies is difficult to make at this time. Cement-based processes are available, but because of the high solubility of cesium, cement-based final waste forms may not be an acceptable option. The next three technologies listed in Table 5 are not yet widely available with the remote handling needs that would be required for this project. Direct disposal of CST is a “no-treatment” alternative to vitrification of CST. There may also be low-temperature technologies for immobilization of CST.

Technology Applicability

Vitrification of ion exchange material has important applications to a number of government-owned facilities in the United States. ORNL has radioactive liquids that must be solidified and disposed. The radioactive cesium concentrations in much of this waste are high enough to require that the cesium be removed before disposal. Hanford is also evaluating use of the ion exchange material to remove cesium from its high-activity wastes. SRS and Idaho National Engineering and Environmental Laboratory are considering using ion exchange materials as backup technologies to their baseline flow sheets. In any of these applications, vitrification could be used to solidify the resulting cesium-loaded ion exchange material.

Patents/Commercialization/Sponsors

SRTC is in the process of patenting glass formulations. CST was developed jointly by Sandia National Laboratory and Texas A&M University. It was commercialized through a cooperative research and development agreement with an industrial partner. Both an engineered (mesh pellets) and powdered forms are commercially available.

SECTION 5

COST

Methodology

The CST immobilization process is an enabling technology. It allows the use of highly efficient CST for radionuclide removal from liquid wastes by providing a disposal technology for the secondary waste stream (loaded CST). Potential cost savings are realized from enabling the use of CST sorbent instead of Duolite™ CS100 sorbent in an out-of-tank ion exchange column at Hanford. Potential savings from using CST instead of CS100 are in the hundreds of millions of dollars (Demuth and Williams 1997). This savings estimate assumes no increase in HLW volume from loaded CST because acceptable glass formulations are available that incorporate loaded CST and sludge into HLW glass without increasing glass volume.

Cost Analysis

CS100 sorbent is the baseline for radionuclide separation from liquid-based waste in Hanford's underground storage tanks. Demuth (1998) and Slaathaug (1995) report capital costs for radionuclide removal to be $792 million and operating and maintenance (O&M) costs to be $276 million in 1995 dollars. Adjusted to 1999 dollars, capital costs for radionuclide removal are $909 million, and O&M costs are $317 million. Elution and regeneration costs are $311 million. These cost estimates assume that the ion exchange uses CS100 and that a decontamination factor of 98% is achieved.

Demuth and Williams (1997) estimate the cost savings from the use of CST instead of CS100 for the life cycle of waste treatment at Hanford. They account for new waste inventories and differentiate between Hanford's phases I and II of the privatization effort, making the following assumptions:

- Sludge washing with dilute caustic is the baseline for solids-based waste from underground storage tanks. Enhanced sludge washing (ESW) is an alternative technology that generates additional quantities of liquid-based waste that must be treated by cesium removal.
- The elutable CS100 organic resin survives ten regeneration cycles prior to disposal. Sodium is added to neutralize the cesium-loaded eluate, and this increases the final amount of HLW glass.
- The CST resin is inorganic and nonregenerable.
- The organic CS100 resins are disposed of following the final elution with no increase in waste volume.
- The loaded CST is disposed of as HLW with no increase in final waste volume using CST-sludge glass formulations developed by the Tanks Focus Area.
- The capital and operating costs for a cesium removal facility are similar for both types of ion exchange sorbents, however CS100 requires resin regeneration.
- The cost for resin regeneration is approximately $7/kg–liquid (1997 dollars).
- The cost for HLW vitrification is $374/kg of HLW glass (1997 dollars).

Cost Conclusions

Table 6 shows the effect of ion exchange sorbent on cost savings for the range of processing alternatives for underground storage tank waste at Hanford (Demuth and Williams 1997). The potential cost savings from using CST instead of CS100 range from $667 million to $838 million. Cost savings increase from the use of CST if ESW is used instead of simple sludge washing. Additional quantities of liquid wastes are generated with ESW that must be treated. Also, more secondary waste is generated from elution of CS100, resulting in greater HLW volume.

Table 6. Cost savings from cesium removal for various sludge washing alternatives

Baseline tank waste treatment alternative[a]		Cost savings (millions of 1997 dollars) CST	CS100	Increase in cost savings from CST rather than CS100
Sludge washing	Phase I	113	0	113
	Phase II	554	0	554
	Total			667
Enhanced sludge washing	Phase I	235[b]	45	190
	Phase II	1,984[b]	1,336	648
	Total			838

[a]Phase I is proof of concept and commercial demonstration; Phase II is full-scale production.
[b]Cost savings due to eliminating resin regeneration plus approximate 1% decrease in HLW volume.

Since the study above was completed, sludge wash factors for Hanford have been significantly revised. The sludge wash factors interact with the CST performance to significantly impact the HLW volume and consequent remediation cost savings. A revised study soon to be issued indicates the potential CST cost savings at Hanford are $372 million for Phases I and II of privatization at Hanford and range from $102 million to $503 million. These cost savings (in 1999 dollars) are less than those shown in Table 6 because they assume some increase in HLW volume from the use of CST.

SECTION 6

REGULATORY AND POLICY ISSUES

Regulatory Considerations

Management of HLW, transuranic (TRU) waste, and low-level waste (LLW) is addressed by DOE Order 5820.2a, which is being replaced by DOE Order 435.1. The revised directive will call for performance-based and risk-based requirements.

Waste Acceptance Criteria
Waste forms generated by sites such as Savannah River and Hanford will likely require disposal as HLW. The Office of Civilian Radioactive Waste Management System (OCRWM) handles disposal of HLW. Currently, no protocols exist for the acceptance and transportation of HLW. They are being jointly developed by OCRWM and the Office of Environmental Management under a memorandum of agreement.

The waste form for ORNL waste must meet the requirements for disposal at the Nevada Test Site (NTS):

- Curie content: The radionuclide content of the waste forms must not exceed the Class C limit (4,600 Ci/m^3 for cesium).
- Mixed waste: NTS will not accept mixed waste generated outside the state of Nevada. Since the cesium-loaded CST is nonhazardous, it was necessary to demonstrate that the glass passed the toxicity characteristic leaching procedure (TCLP) to confirm the nonhazardous nature of the vitrified waste form. An alternate leaching test for durability testing of glass forms is the PCT.
- TRU waste: NTS does not accept TRU waste for disposal. The NTS Waste Acceptance Criteria (WAC) requirement for TRU is that the concentration of alpha-emitting transuranic nuclides with half-lives greater than 20 years must not exceed 100 nCi/g.
- Radionuclide reporting: The NTS WAC lists the radionuclides and the levels at which they must be reported for each waste stream.
- Particle size: To ensure that the waste form is contained, NTS limits the percentage of particulates. The waste form must contain less than 1 wt % of less-than-10-micrometer-diameter particles and less than 15 wt % of less-than-200-micrometer-diameter particles. These requirements are readily met with a vitrified waste form.
- Free liquids: The waste must contain no free liquids. No liquids survive the vitrification process in the melter at 1500°C. Proper administrative procedures and containerization techniques must preclude water intrusion during storage, handling, and transportation.
- Other requirements: The NTS WAC precludes or limits gases, etiologic and chelating agents, polychlorinated biphenyls, explosives, and pyrophorics in the waste forms. The process of vitrification of CST to produce a borosilicate glass waste form destroys these materials.

Secondary Waste Streams
CST technologies generate minimal amounts of secondary waste compared to treatment technologies that require resin regeneration. Resin regeneration produces contaminated liquid that must be neutralized with sodium and eventually vitrified. Since sodium is a limiting constituent in HLW glass, the final HLW glass is increased.

CERCLA/RCRA Considerations
This technology is currently being considered for wastes regulated by the Resource Conservation and Recovery Act (RCRA). Hazardous and dangerous waste permit(s) will be required to operate treatment

facilities. Treatment of wastes regulated by Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) may be considered at a later date. CERCLA considerations are discussed below.

Human Health and Environment
The overall protection of human health and the environment is high. Vitrifying waste enables its permanent disposal and prevents its release into the environment. CST glass formulations minimize the amount of HLW resulting from waste treatment.

Compliance with Applicable or Relevant and Appropriate Requirements (ARARs)
Compliance with ARARs is required when the waste is disposed of on site. Vitrified HLW glass containing CST will be sent to an off-site repository for disposal. If CERCLA waste has been immobilized, the off-site disposal facilities must be qualified to accept waste from a CERCLA site.

Long-Term Effectiveness and Permanence
The vitrification process produces a very durable, homogeneous glass. The waste glass is expected to be stable over a period of time up to several hundred thousand years. This is important since unwanted leaching and migration of radioactive waste from the waste glass could pose risk to future generations.

Reduction of Volume, Mobility/Toxicity
The HLW fractions produced from the CST immobilization processes are of a much smaller volume and are more stable than if no treatment were conducted. Therefore, reduction of toxicity, mobility, and volume of HLW is more effective than doing no pretreatment or an alternative type of treatment.

Implementability
Full-scale implementation is not complex. The remote-handling designs and procedures already exist, all equipment and reagents are commercially available, people are currently trained in this process, and regulatory permits can be obtained.

Costs
Money can be saved over the baseline by not regenerating the resin used for radionuclide removal and by vitrifying resin and sludge simultaneously using acceptable glass formulations.

State and Community Acceptance
State and community acceptance are addressed as part of the total remedial action. The technology will likely improve acceptance for the remedial action because the technology reduces the volume of HLW compared to the baseline.

Safety, Risks, Benefits, and Community Reaction

Worker Safety and Potential Exposures
Worker protection is required. Remote operations and protective equipment are required for waste handling and maintenance of equipment for both the baseline and competing technologies.

Community Safety
There are potential community safety issues associated with accidents during transportation of radioactive glass to an off-site repository. Such accidents are of low probability. Shielded trucks and canisters protect the public from excessive exposures. Use of CST immobilization technology will positively impact community safety issues because the volume of HLW glass is lower than the baseline.

Potential Environmental Impacts
Potential environmental impacts are addressed in an Environmental Impact Statement as part of the total remedial action. Use of CST glass formulations will have no impact on the overall environmental impacts.

Liability Risk
Liability risk is addressed as part of the total remedial action. Use of CST glass formulations should have no impact on the liability risk.

Socioeconomic Impacts and Community Perceptions

No adverse socioeconomic or community reactions are anticipated.

SECTION 7

LESSONS LEARNED

Implementation Considerations

The following should be evaluated prior to implementing this technology:

- Mixing and pumping properties of CST are different than for HLW sludges. Fine particles are produced from the mixing of CST, and CST is difficult to maintain in a well-mixed slurry. Recirculation loops or dry feed systems may be required for the melter.

- There is potential for accumulation of volatile radionuclides in the melter off-gas system. Some material compatibility issues may be associated with minor components in the loaded CST resins.

- Heat loads associated with cesium loading in the glass waste form should be assessed for any applicable limits or relationships to disposal site requirements.

Technology Limitations and Need for Future Development

Research into the vitrification of cesium-loaded CST has yielded several lessons for CST-only glass:

- Glass durability: Glass durability is determined using the PCT. Results from these tests indicated that the durability generally tends to improve with increasing waste loading. Of the samples reported, the 60% CST loading was the highest waste loading that had acceptable durability and no crystallization. This is significantly higher than originally anticipated and should lead to significant waste volume and operating cost reductions.

- Compatibility with melter: For the waste to be processed, the melt must have a viscosity that will allow it to be poured. Glasses with 40, 50, and 60% CST loading were prepared, melted, and measured for viscosity. All were in the acceptable range for the melter and therefore should be compatible with the melter.

- Waste loading: Crystal formation in the glass waste form reduces durability. Tests were conducted at various waste loadings to determine the upper bounds for the formation of crystals in the glass. Loadings up to 64 wt % were acceptable.

- Direct disposal of CST or a low-temperature process for a CST-only waste form is a potential alternative to vitrification. There is little or no data on the leachability of cesium-137 from unconsolidated CST. There is little data on the long-term stability of CST loaded with radioactive cesium. Such information needs to be available to assess the suitability of loaded CST for long-term storage.

APPENDIX A

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APPENDIX B

ACRONYMS

ASTM	American Society for Testing and Materials
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CST	crystalline silicotitanate
DOE	Department of Energy
DWPF	Defense Waste Processing Facility
EA	Environmental Assessment
ESW	enhanced sludge washing
HLW	high-level waste
LAW	low-activity waste
LLW	low-level waste
NTS	Nevada Test Site
OCRWM	Office of Civilian Radioactive Waste Management System
O&M	operating and maintenance
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
OST	Office of Science and Technology
PCT	product consistency test
RCRA	Resource Conservation and Recovery Act
SRS	Savannah River Site
SRTC	Savannah River Technology Center
TCLP	toxicity characteristic leaching procedure
TRU	transuranic
WAC	waste acceptance criteria
WSRC	Westinghouse Savannah River Company