CONF-970568--

Performance Modeling Applied to the Treatment and Disposal of a Mixed Waste at the SRS

by

J. B. Pickett

Westinghouse Savannah River Company
Savannah River Site
Aiken, South Carolina 29808

C. M. Jantzen
WSRC

J. R. Cook
WSRC

A. R. Whited
Dames and Moore
NY USA

R. A. Field
Clemson University
SC USA

A document prepared for 99TH ANNUAL MEETING, SYMPOSIUM PROCEEDINGS: WASTE MANAGEMENT TECHNOLOGIES IN THE CERAMIC AND NUCLEAR INDUSTRIES at Cincinnati, OH, USA from 5/4/97 - 5/7/97.

DOE Contract No. **DE-AC09-96SR18500**

This paper was prepared in connection with work done under the above contract number with the U. S. Department of Energy. By acceptance of this paper, the publisher and/or recipient acknowledges the U. S. Government's right to retain a nonexclusive, royalty-free license in and to any copyright covering this paper, along with the right to reproduce and to authorize others to reproduce all or part of the copyrighted paper.

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

This report has been reproduced directly from the best available copy.

Available to DOE and DOE contractors from the Office of Scientific and Technical Information, P. O. Box 62, Oak Ridge, TN 37831; prices available from (423) 576-8401.

Available to the public from the National Technical Information Service, U. S. Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

DISCLAIMER

**Portions of this document may be illegible
electronic image products. Images are
produced from the best available original
document.**

PERFORMANCE MODELING APPLIED TO THE TREATMENT AND
DISPOSAL OF A MIXED WASTE AT THE SAVANNAH RIVER SITE (U)

J. B. Pickett, C. M. Jantzen, and J. R. Cook
Westinghouse Savannah River Co., Aiken, SC 29808

A. R. Whited
Dames and Moore, Orchard Park, NY 14127

R. A. Fjeld
Clemson University, Clemson, SC 29643-0919

ABSTRACT

Performance modeling for Low Level Mixed Waste disposal was conducted using the measured leach rates from a number of vitrified waste formulations. The objective of the study was to determine if the improved durability of a vitrified mixed waste would allow trench disposal at the Savannah River Site (SRS).

Leaching data were compiled from twenty-nine diverse reference glasses, encompassing a wide range of exposed glass surface area to leachant volume ratios (SA/V), and various leachant solutions; all of which had been leached at 90°C, using the MCC-1 or PCT procedures (ASTM Procedures C1220-92 and C1285-94, respectively). The normalized leach rates were scaled to the ambient disposal temperature of 25°C, and compared to the allowable leach rate of uranium - which would meet the performance assessment requirements. The results indicated that a glass of above average durability (vs. the reference glasses) would meet the uranium leaching concentration for direct SRS trench disposal.

INTRODUCTION

Vitrification, the conversion of source materials into a solid amorphous glass matrix, has emerged as a viable treatment technology for the immobilization of Low-Level Radioactive Waste (LLW) and mixed waste (LLW/hazardous waste). Vitrification has been selected for the High Level Radioactive Waste (HLW) stored at the Department of Energy (DOE) West Valley, NY; Savannah River Site, SC; and Hanford Site, WA sites. The U.S. Environmental Protection Agency (EPA) has declared vitrification to be the "Best Developed Available Technology" for the treatment of the DOE HLW/mixed wastes [1]. The advantages of vitrification include immobilization of radioactive and hazardous constituents, substantial volume reduction, and a final wasteform with high chemical/physical durability.

Over the past two decades, substantial research has been conducted to vitrify the legacy of High Level Radioactive (HLW) waste remaining from the United States weapons production programs. More recently, vitrification has been proposed for the treatment of Low Level Radioactive (LLW) and mixed (radioactive/hazardous) wastes. Vitrification has been selected by the DOE to treat the low level fraction of the high level wastes stored at the Hanford reservation, to treat some of the mixed wastes stored at the Idaho National Engineering Laboratory site, and a mixed waste at Savannah River Site (SRS).

Final disposal of the vitrified wastes will depend on whether the final wasteform remains a hazardous waste by the EPA hazardous waste management Resource Conservation and Recovery Act (RCRA) regulations, or whether the waste can be de-characterized or "delisted" as hazardous. The SRS waste being vitrified is a RCRA listed F-006 wastewater treatment sludge from electroplating operations. The sludge was generated during the production of depleted uranium "targets" in the SRS "M-Area", for the eventual production of plutonium-239. The treated waste will remain a "listed" F-006 waste after treatment, and would therefore have to be disposed to a RCRA Subtitle C disposal facility. Disposal of the vitrified waste at the SRS is the preferred option, but the SRS low level waste disposal facilities are not permitted Subtitle C facilities, and no such facilities are anticipated at the SRS. Therefore, the SRS has submitted a "Delisting Petition" to the Region IV EPA for the vitrified M-Area sludge [2].

In order for low-level radioactive wastes to be disposed at DOE sites, the DOE Order 5820.2A mandates that a site-specific radiological Performance Assessment (PA) must be conducted. The final waste and disposal facility must comply with the performance objectives of the order. The time period for which compliance is required is not specified in the order, but it is generally assumed to that the performance objectives be met for a minimum of 10,000 years. [3] The key performance objectives for the SRS E-Area disposal are:

- Protect public health and safety in accordance with other applicable DOE orders (other than 5820.2A),
- Radioactive dose to public is less than 25 mrem/yr; all pathways,
- Prevent dose to inadvertent intruder to:
 - less than 100 mrem/yr for continuous exposure, or
 - less than 500 mrem for a single acute exposure, and
- Protect groundwater resources consistent with federal, state, and local requirements - usually assumed to be less than 4 mrem/yr from radionuclides or the Maximum Contaminant Limits (MCLs) for chemical constituents from water ingestion.

The performance assessment assumes that the integrity of the wasteform, the wasteform container, and/or the vault structure itself will degrade within the 10,000 year period. Therefore, only naturally occurring geologic phenomena are used to calculate the affect on the environment or potential human uses (or intrusion) from the disposed materials.

For example, the affect of a disposed waste on the groundwater, and its potential use as a human drinking water supply, is a function of the chemical mobility of the waste in the natural environment, and the distance and attenuation of the waste in the soil column between the waste and the water table. In the case of the M-Area plating line sludge, the final vitrified waste form will contain quantities of uranium isotopes which would exceed the SRS E-vault or trench Performance Assessment disposal limits. However, as stated above, the durability of the vitrified wasteform is not included in the performance assessment. The retention of the constituents in the glass matrix will inhibit the migration of the radioactive materials to the drinking water supply. If the durability of the glass were included in the performance assessment calculations, the authors believe that the final M-Area product could meet the performance objectives.

The objective of this presentation is to demonstrate that current laboratory leaching procedures can be used to "predict" the long-term leaching performance of vitrified wastes, and that specifically, the vitrified M-Area waste should meet the performance criteria with respect to groundwater protection for the SRS low level radioactive trench disposal. The Curies of the various uranium isotopes present in the M-Area sludge are shown in Table I.

Table I Curies of Uranium Isotopes In M-Area Sludge

<u>Isotope</u>	<u>Ci</u>
U-234	4.23
U-235	0.19
U-236	0.13
U-238	9.55

SRS PERFORMANCE ASSESSMENT GROUNDWATER MODELING

Previous performance assessments for the SRS have shown that the limiting pathway for public exposure is via transport to a drinking water well and subsequent ingestion [3]. The modeling of the M-Area waste considered both the radiological and chemical toxicity of uranium. The limits used were:

- radiological toxicity, 4 mrem/yr from uranium (equivalent to 45 $\mu\text{gm/L}$),
- chemical toxicity, a maximum proposed drinking water limit of 20 $\mu\text{gm/L}$ (20 ppb) uranium, due to heavy metal (kidney) toxicity [4].

The groundwater modeling code used for the SRS EAV performance assessment [3] and the 1987 SRS Groundwater Protection Environmental Impact Statement [5,6,7] is the "PATHRAE" code. This code was designed and accepted by the EPA for modeling the land disposal of LLW waste [8]. The PATHRAE code is based on one-dimensional advection through the unsaturated soil zone, and is primarily affected by water infiltration rate, fraction of saturation, porosity, soil density, and contaminant sorption coefficients. Saturated zone transport is a one-dimensional advection and 2 dimensional dispersion as a primary function of groundwater velocity, dispersion coefficients, aquifer porosity and soil density and contaminant sorption coefficients. The rate of infiltration to groundwater is

assumed constant over the entire period, in this case 10,000 years. The leach rate from the inventory is assumed constant, except for radiological decay and solubility influences. The modeling conducted in this presentation for the disposal of the vitrified M-Area waste at the SRS E-Area Low Level Waste disposal utilized the following waste and site specific parameters and disposal site assumptions. It was assumed that the waste would be distributed over the entire area of the waste trenches, to minimize a localized uranium concentration.

Table II. Parameters for PATHRAE Modeling of SRS E-Area Disposal Site

<u>Parameter</u>	<u>Value</u>	<u>Source (Ref. #)</u>
Trench Dimensions (m)	200 x 48	EAV-PA (3)
Downgradient Drinking Well Distance(m)	100	EAV-PA (3)
Annual Rainfall Infiltration (cm/yr)	40	EAV-PA (3)
Unsaturated Zone Depth (m)	9	EAV-PA (3)
Uranium Solubility (moles U/L)	2.0×10^{-6}	EAV-PA (3)
Drinking Water Uptake (L/yr)	730	EPA (4)
Uranium K_d (ml/g)	35	Sheppard (10)
Volume of Wasteform (m^3)	380	Pickett (2)
Density of Wasteform (kg/m^3)	2600	Pickett (2)
SRS Specific Hydrogeologic Factors	-	EAV(3) and Ryan (9)

The PATHRAE code calculates contaminant leaching, and subsequent groundwater concentration at the drinking water well, based on a constant fractional leach rate $\bar{A}_L$ (yr^{-1}). This $\bar{A}_L$ is multiplied by the initial inventory of each isotope I_0 (Ci) to obtain a preliminary value for the annual activity release in Ci/yr. The preliminary value is corrected for isotopic radioactive decay and solubility limitations, and corrected to human dose equivalents in millirem/year (mrem/yr). The significant decay products for the M-Area uranium inventory, in a 10,000 year period, are U-238 and U-234. At a fractional release rate of $6.0 \times 10^{-6}/yr$, U-238 causes 2.5 mrem/yr dose, U-234 causes a 1.25 mrem/yr, and all other radionuclides provide a dose of 0.25 mrem/yr, for a total of 4.0 mrem/year. Since the major isotope present is U-238 (Table I), this translates to a concentration of 45 μg U/L (45 ppb). The fractional release ($\bar{A}_L$) resulting in a uranium concentration of 20 $\mu g/L$ (20 ppb) is $2.65 \times 10^{-6}/yr$. This fractional release from the E-Area trenches, of 2.65% of the uranium inventory over 10,000 years, is the release rate which would meet the health based drinking water goal.

GLASS CORROSION

Corrosion Mechanisms And Glass Dissolution

Waste glass corrosion involves a series of complex processes which are highly dependent on environmental conditions and initial glass composition. The model most widely accepted at present involves three primary mechanisms (11):

- diffusion controlled ion-exchange (diffusion of water into the glass, and exchange of hydrogen for soluble cations),
- dissolution of the glass matrix, (complete breakdown of the glass structure via hydrolysis reactions), and

- formation of thermodynamically stable products from the glass constituents (crystalline phases on the glass surface, or precipitation of amorphous phases).

The diffusion controlled ion-exchange dominates the early stages of glass dissolution, while glass matrix dissolution dominates the latter stages (12). The surface layers may reduce leaching by forming a protective barrier [13], or could cause leaching to increase by altering the solution chemistry [11,14].

While these three corrosion mechanisms control the mechanism by which glass dissolves, solubility considerations determine the dissolution rate. The dissolution rate is highest initially, when the chemical potential gradient between the glass and the solution is the greatest [15]. As soluble constituents are released, the driving concentration gradient is decreased, and the corrosion rate declines. In most leaching environments, highly soluble elements such as sodium, lithium, and boron are almost completely released into solution. Components such as silicon, aluminum, and iron become supersaturated as leachant chemistry changes, and can be re-precipitated as surface layers. If the vitrified wastefrom is subjected to limited infiltration of groundwater over long time periods, then the contacting solution may experience saturation with respect to all but the most soluble components. Solution saturation, especially with respect to the primary glass former, silicon, causes the dissolution rate to decline dramatically [11,16].

Most long-term leaching data suggest that the overall glass corrosion rate should decrease with time[12]. Test conducted with high Surface Area (SA) to Volume (V) ratios, at durations long enough to reach silicon saturation, generally show that dissolution rates approach a low, steady-state value [17].

Glass Leach Rate Calculations

An analytical model that reliably predicts contaminant leaching from glass in a disposal environment has yet to be developed. However, a substantial amount of empirical leaching data has been generated in the last two decades, during the development of durable glass wastefroms for the stabilization of radioactive and mixed wastes. Standard leaching test protocols report the concentration of the elemental glass components (i.e., silicon, boron, sodium) observed in the leaching solution, at various times. The leach solution concentration is normalized to based on the concentration of each component in the initial glass. The Normalized Concentration NC_i is defined as:

$$NC_i = \frac{C_i}{F_i} \text{ where,} \quad (1)$$

NC_i = normalized concentration of the element i in the leachant (g/m^3 , or ppm),

C_i = concentration of element i in leachant (g/m^3), and

F_i = mass fraction of element i in glass (g/g glass)

If leaching data exist for two different times, at otherwise identical conditions, an approximation of the Normalized Leach Rate (NLR), at average time, can be calculated.

$$NLR_i = \frac{(NC_i)_n - (NC_i)_{n-1}}{(SA/V) (t_n - t_{n-1})} \quad (2)$$

where;

NLR_i = normalized leach rate for element i (g/m²-day)

SA = exposed glass surface (m²), and

V = leachant volume (m³)

$t_n - t_{n-1}$ = delta time (days)

This approximation is poor during the early stages of glass dissolution, when the rate is affected strongly by the large differences in glass vs. solution concentrations. However, over long time periods (months to years), and in situations where the surface area is large compared to the volume of leachant, the NLR_i will remain relatively constant. High SA/V conditions are expected in disposal sites where groundwater infiltration is limited due to climatic or geological conditions, and the surface area of the waste is significant.

Boron has been selected as a good indicator of the overall leach rate of a glass, since it is a key component of boro-silicate glasses, is among the most soluble of glass constituents, and does not tend to saturate the solution or form secondary precipitation products. NLR_i can be approximated as NLR_{boron} .

If both the NLR_i and the effective surface area are assumed to be relatively constant over a delta period of time t_n to t_{n-1} , then the total mass release (M_{Total}) can be calculated by:

$$M_{\text{Total}} = (NLR_{\text{boron}}) \text{ g/m}^2\text{-day (SA) m}^2 \text{ (delta time) days} \quad (3)$$

The mass release of less soluble glass components, such as uranium, will be a fraction of the boron release. A conservative assumption for these calculations is $NLR_{\text{uranium}} = NLR_{\text{boron}}$. Therefore, the total mass release of uranium at time t can be found as:

$$M_{\text{Total uranium release}} = (NLR_{\text{boron}}) \text{ g/m}^2\text{-day (SA) m}^2 \text{ (delta time) days} \quad (4)$$

$$\text{then the fractional release} = \frac{M_T}{M_{\text{initial g}}} = \frac{(NLR_{\text{boron}}) \text{ g/m}^2\text{-day (SA) m}^2 \text{ (days)}}{M_{\text{initial g}}} \quad (5)$$

PATHRAE vs. GLASS LEACH RATE MODELING

PATHRAE's use of a constant fractional leach rate $\bar{A}_L$ is therefore similar to the constant approximation of glass leaching based on boron release and surface area to volume of leachant ratio. I. e. the PATHRAE fractional release is:

$$\bar{A}_L \text{ (yr}^{-1}\text{)}, \text{ fractional release/year} = \frac{M_T \text{ g released by time } t}{M_{\text{initial g}}} / \text{time (years)} \quad (6)$$

Substituting the average fractional release from the leaching model; Eq. 5, gives

$$\bar{A}_L(\text{yr}^{-1}) = \frac{(\text{NLR}_{\text{boron}} \text{ g/m}^2 \cdot \text{day}) (\text{SA, m}^2) (\text{delta time, days})/\text{years}}{\text{Mass initial g}} \quad (7)$$

This equation assumes that the Normalized Release Rate for the vitrified M-Area waste remains relatively constant over the 10,000 year time frame. This is reasonable, considering that in order to meet the performance assessment, the glass must release no more than 2.65 % of the uranium in the 10,000 year time frame.

In the specific case of the modeling of the release of uranium from a vitrified waste in an E-Area trench, the PATHRAE $\bar{A}_L$ can be related to the approximate glass leaching. The final volume of vitrified waste is assumed to be ~100,000 gallons, or 380 m³. Since the density of the final glass is ~2.6 x 10⁶ g/m³, then the final mass will be ~9.9 x 10⁸ g.

The glass will be prepared as small glass ovoids, approximately 13 mm in diameter. The volume (V) of a single ovoid = $\frac{2}{3}\pi r^3 = 5.75 \times 10^{-7} \text{ m}^3$. The surface area of a single ovoid = $\pi r^2 = 3.98 \times 10^{-4} \text{ m}^2$. The number of ovoids can be estimated by dividing the total volume V by the volume per ovoid = $380 \text{ m}^3 / 5.75 \times 10^{-7} \text{ m}^3 = \sim 6.6 \times 10^8$ ovoids. The surface area (SA) of the vitrified waste will be $\sim 6.6 \times 10^8 \times 3.98 \times 10^{-4} \text{ m}^2 = 2.6 \times 10^5 \text{ m}^2$. To relate $\bar{A}_L$ to the leach rate calculations for this case, substitute the M-Area parameters in equation 7;

$$\bar{A}_L(\text{yr}^{-1}) = \frac{(\text{NLR}_{\text{boron}} \text{ g/m}^2 \cdot \text{day}) (\text{SA, m}^2) (\text{delta time, days})/\text{yr}}{\text{Mass initial g}} \quad (7)$$

or, rearranging,

$$\text{NLR}_{\text{boron}} \text{ g/m}^2 \cdot \text{yr} = \frac{\bar{A}_L \text{ yr}^{-1} (\text{Mass initial g})}{(\text{SA, m}^2)} \quad (8)$$

$\text{NLR}_{\text{boron}}$ = Average bulk corrosion rate, g/m²·yr

SA = initial effective area = $2.6 \times 10^5 \text{ m}^2$,

M_{glass} = initial mass of glass = $9.9 \times 10^8 \text{ g}$

$$\text{NLR}_{\text{boron}} \text{ g/m}^2 \cdot \text{yr} = \frac{\bar{A}_L \text{ yr}^{-1} (9.9 \times 10^8 \text{ g})}{(2.6 \times 10^5 \text{ m}^2)} = \bar{A}_L \text{ yr}^{-1} 3.8 \times 10^3 \text{ g/m}^2 \quad (9)$$

Assuming $\text{NLR}_{\text{uranium}} = \text{NLR}_{\text{boron}}$, then

$$\text{NLR}_{\text{uranium}} = \sim 3800 \bar{A}_L \text{ yr}^{-1} \text{ g/m}^2 \quad (10)$$

This is a direct relationship between the leaching of the M-Area glass beads and the PATHRAE fractional leach rate parameter. The PATHRAE code was used previously to calculate a fractional release rate $\bar{A}_L = 2.65 \times 10^{-6} \text{ yr}^{-1}$, giving a drinking water concentration of 20 ppb. Substituting this into equation 10 gives:

$$\text{NLR}_u = (3800 \text{ g/m}^2)(2.65 \times 10^{-6} \text{ yr}^{-1}) = 0.010 \text{ g/m}^2\text{-yr} = 2.7 \times 10^{-5} \text{ g/m}^2\text{-day} \quad (11)$$

Therefore, the normalized average bulk glass uranium release rate ($\text{NLR}_{\text{uranium}}$) that would meet the 20 ppb drinking water standard is $\sim 2.7 \times 10^{-5} \text{ g/m}^2\text{-day}$.

ACCEPTABILITY OF M-AREA GLASS LEACH RATE

Now that a normalized uranium leach rate has been calculated for the vitrified M-Area waste, it can be determined whether the actual waste will meet this level. Three methods can be used:

- 1) comparison to natural glasses such as obsidian and tektites,
- 2) comparison to other glasses using standard leaching procedures, and
- 3) long term leaching studies with actual M-Area glasses.

Natural Glasses

The first method is appropriate to obtain a rough estimate of whether the M-Area waste glass can meet the $2.7 \times 10^{-5} \text{ g/m}^2\text{-day}$ leach rate, under ambient conditions. Jantzen has compared the durability of natural glasses (10^5 to 10^6 years) to ancient man-made glasses (10^3 years)[18] and found that these glasses bounded the expected durability of nuclear waste glasses. Advanced waste glasses should approach 10^6 years in a repository environment. Since the PATHRAE code calculations indicated that a release of 2.65% per 10,000 years would meet the 20 ppb drinking water limit, a total release would occur in 3.8×10^5 years. This is consistent with the durability of nuclear waste glasses.

Standard Leaching Test Results

A number of long term static leach test results were collected from the literature. Twenty-nine studies, from seven different references, were utilized to form a reference database [20-26]. All the tests were conducted at 90°C , with a variety of leachants and SA/V ratios. The times ranged from 84 to 920 days, at a minimum SA/V of 340 m^{-1} . An average long term normalized leach rate for boron was determined from each of the 29 studies. Using the Arrhenius equation, the $\text{NLR}_{\text{boron } 90^\circ\text{C}}$ data was then corrected to an ambient temperature of 25°C . An activation energy of 75 kJ/mole for the Arrhenius calculation was selected from the average of the range activation energies of 60-90 kJ presented in Reference 19. The temperature corrected leaching data ranged from 1.0×10^{-6} to $1.6 \times 10^{-4} \text{ g/m}^2\text{-day}$. A lognormal probability plot (Figure 1) showed that seventy % of the glasses from the 29 different reference glasses would satisfy the

calculated leach rate of 2.7×10^{-5} g/m²·day needed to meet the 20 ppb groundwater concentration.

CONCLUSION

The allowable leach rate needed to meet the drinking water standard for uranium at a groundwater well downgradient of an E-Vault trench is well within the durability attained by modern nuclear waste glass formulations. Empirical leaching studies to confirm that actual M-Area glasses will meet this standard are now underway.

ACKNOWLEDGMENTS

This presentation was prepared in connection with work conducted under DOE Contract No. DE-ACO9-89SR18035, and a S. C. Universities Research and Education Foundation under Department of Energy contract DEFCO993SR-18262. The DOE's Mixed Waste Focus Area, EM-50 also contributed to the funding. The work discussed in this paper is based primarily on work conducted by A. Ryan Whited to prepare a thesis for presentation to the Graduate School of Clemson University, in partial fulfillment of a Master's Degree.

Lognormal probability plot for long-term leach rates with allowable leach rates from performance assessment calculations.

REFERENCES

1. U. S. EPA, "Land Disposal Restrictions for Third Scheduled Wastes, Final Rule", *55 Federal Register* 22627 (1990).
2. J. B. Pickett, "Upfront Delisting Petition for Vitriified M-Area Plating Line Wastes", WSRC-TR-96-0244, U. S. Department of Energy, Savannah River Site, Aiken, SC 29898, 1996.
3. "Radiological Performance Assessment for the E-Vaults Disposal Facility", WSRC-RP-94-218, Rev. 0, WSRC, Aiken, SC 29898.
4. U. S. EPA, "40 CFR Parts 141 and 142, National Primary Drinking Water Regulations; Radionuclides, Proposed Rule", *56 Federal Register* 33050 (1991).
5. U. S. DOE, "Final Environmental Impact Statement, Waste Management Activities for Groundwater Protection, Savannah River Site, Aiken, SC", DOE/EIS-0120, (1987).
6. B. B. Looney, C. M. King, and D. E. Stephenson, "Environmental Information Document, Quality Assurance Program for Environmental Assessment of Savannah River Plant Waste Sites", DPST-86-725, E. I. Dupont de Nemours & Co., Savannah River Laboratory, Aiken, SC 29808 (1986).
7. D. E. Stephenson, C. M. King, B. B. Looney, and M. W. Grant, "Environmental Information Document, Methodology for Predictive Modeling of Environmental Transport and Health Effects for Waste Sites at the Savannah River Plant", DPST-86-710, E. I. Dupont de Nemours & Co., Savannah River Laboratory, Aiken, SC 29808 (1986).
8. G. B. Merrell, V. C. Rogers, K. K. Nielson, and M. W. Grant, "The PATHRAE Performance Assessment Code for the Land Disposal of Radioactive Wastes", Report to the U. S. EPA, RAE-8469/3, Rogers and Associates Engineering Corp., Salt Lake City, UT (1985).
9. A. Ryan Whited, "Vitriified Waste Form Performance Modeling Applied to the Disposal of a Mixed Waste Sludge at the Savannah River Site", A Masters Degree Thesis Presented to the Graduate School of Clemson University, Clemson SC 29643 (1996).
10. M. I. Sheppard and D. H. Thebault, "Default Soil/Liquid Partition Coefficients, K_{ds} , for 4 Major Soil Types: A Compendium", *Health Physics* **59** [4], 471 (1990).
11. W. L. Bourcier "Critical Review of Glass Performance Modeling", ANL-94/17, Argonne National Laboratory, Chicago, IL. (1994).
12. J. C. Cunnane and J. M. Allison, "High Level Waste Glass Compendium; What is Tells Us Concerning the Durability of Borosilicate Waste Glass", *Mat. Res. Soc. Symp. Proceedings*, **333**, 3 (1994).
13. X. Feng, "Surface Layer Effects on Waste Corrosion", *Mat. Res. Soc. Symp. Proceedings*, **333**, 55 (1994).
14. W. L. Ebert, J. K. Bates, E. C. Buck, and C. R. Bradley, "Accelerated Glass Reaction Under PCT Conditions", *Mat. Res. Soc. Symp. Proceedings*, **294**, 569 (1993).
15. A. J. G. Ellison, J. J. Mazer, and W. L. Ebert, "Effect of Glass Composition on Waste Form Durability: A Critical Review", ANL-94/28, Argonne National Laboratory, Chicago, IL. (1994).

16. B. Grambow, "Remaining Uncertainties in Predicting Long Term Performance of Nuclear Waste Glass From Experiments", *Mat. Res. Soc. Symp. Proceedings*, **333**, 167 (1994).
17. L. R. Pederson, C. Q. Buckwalter, and G. L. McVay, "The Effects of Surface Area to Solution Volume on Waste Glass Leaching", *Nuclear Technology*, **62**, 151 (1983).
18. C. M. Jantzen, "Prediction of Nuclear Waste Glass Durability from Natural Analogs", *Adv. Ceram.* **20**, *Nuclear Waste Management II*, D. E. Clark, (Ed), American Ceramic Society, Westerville, OH (1986).
19. W. B. White, "Dissolution Mechanisms of Nuclear Waste Glasses: A Critical Review", *Adv. Ceram.* **20**, *Nuclear Waste Management II*, D. E. Clark, (Ed), American Ceramic Society, Westerville, OH (1986).
20. K. Lemmens and P. V. Iseghem, "The Long-Term Dissolution Behavior of the Pamela Borosilicate Glass SM527-Application of SA/V as Accelerating Parameter", *Mat. Res. Symp. Proc.*, **257**, 47 (1992).
21. S. Fillet, E. Vernaz, J. L. Nogues, and N. Jacquet-Francillion, "Corrosion Rate of Nuclear Glasses in Saturated Media", *Adv. Ceram.*, **20**, *Nuclear Waste Management II*, D. E. Clark (Ed.), Am. Ceram. Soc., Westerville, OH (1986).
22. E. Vernaz, T. Advocat, and J. L. Dussossoy, "Effects of SA/V Ratio on the Long-Term Corrosion Kinetics of R7-T7 Glass", *Ceram. Trans.*, **9**, *Nuclear Waste Management III*, J. G. Moore, (Ed.), Westerville, OH (1990).
23. W. L. Ebert and J. K. Bates, "A Comparison of Glass Reaction at High and Low Glass Surface/Solution Volume", *Nuclear Technology*, **104**, 372, (1993).
24. X. Feng, J. K. Bates, E. C. Buck, C. R. Bradley, and M. Gong, "Long-Term Comparison of Dissolution Behavior Between Fully Radioactive and Simulated Nuclear Waste Glasses", *Nuclear Technology*, **104**, 193, (1993).
25. J. K. Bates, *et al.*, "ANL Technical Support Program for DOE Environmental Restoration and Waste Management, Annual Report, October 1992-September 1993", ANL-94/19, Argonne National Lab., Chicago, IL (1994).
26. W. G. Ramsey, "Glass Dissolution of the System $\text{Na}_2\text{O}-\text{B}_2\text{O}_3-\text{SiO}_2-\text{Al}_2\text{O}_3-\text{Fe}_2\text{O}_3-\text{CaO}$ ", Doctoral Dissertation, Clemson Un., Clemson, SC (1995)