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Waste Processing Demonstrations Within the Waste Pretreatment and ProcessingProgram of the Tank Focus Area*

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ABSTRACT

The U.S. Department of Energy (DOE) Office of Environmental Management has created a new approach for the development of technology for DOE environmental remediation problems. Previously, technology development was conducted on a site–by–site basis and managed by multiple organizations. This new DOE strategy consolidates and focuses all technology development efforts on five priority problems. The remediation of the 1×10^8 gal of radioactive waste in the underground storage tanks (USTs) at five DOE sites is one of the priority problems. All tank remediation projects will be integrated into the Tank Focus Area (TFA). The TFA mission is to manage an integrated technology development program that results in the safe and efficient remediation of UST waste across the DOE complex. The TFA has divided its efforts into several areas such as safety, characterization, retrieval, pretreatment and processing, immobilization, and site closure. A key predecessor of the Waste Pretreatment and Processing Program (WPPP) of the TFA was the Waste Processing and Disposal Program (WPDP) of the Underground Storage Tank – Integrated Demonstration. Nearly all of the FY 1995 WPDP projects have been transferred into the WPPP. These WPPP projects can be divided into four systems: cesium removal, comprehensive sludge and supernate, out– of–tank evaporation, and cross–flow filtration. The current status of these WPPP projects is presented. The goal of the projects is to minimize the volume of high–level waste and the radioactivity in low–level waste.

INTRODUCTION

The U.S. Department of Energy (DOE) has identified five priority remediation problems, which include the remediation of the 1×10^8 gal of high-level and low-level radioactive waste in 334 underground storage tanks (USTs) at its Hanford, Savannah River, Oak Ridge, Idaho, and Fernald sites. With the exceptions of Fernald and Idaho, the waste was generated primarily from plutonium production processes such as PUREX, REDOX, and bismuth phosphate separations. The waste in a typical UST is partitioned into three aqueous phases: saltcake, supernate, and sludge. The radioactivity of the waste, 1×10^{11} Bq/L or 3 Ci/L, is primarily due to ^{137}Cs and ^{90}Sr. Major inorganic species result from aluminum nitrate and concentrated nitric acid that were used for metallic dissolution and from ferrocyanide that was used for cesium removal. Prior to storage at Hanford, Savannah River, and Oak Ridge, the pH of the waste was adjusted to 10–14 with the addition of sodium hrydroxide. The largest single constituent is sodium nitrate.

The DOE's Office of Environmental Management (EM) is responsible for the remediation, waste minimization, and environmental compliance activities at all DOE sites. EM has determined that cleanup costs for the USTs would be prohibitive using the current baseline technologies. Therefore, EM has established an extensive technology development program that main objectives are to reduce cost, minimize risk, and accelerate schedule. Currently, site–specific technology development projects are directed by local Waste Management organizations at the UST sites while the EM Office of Technology Development (OTD) utilizes the entire DOE complex to develop and demonstrate new UST remediation technologies. The OTD research and development activities on waste processing are conducted by the Efficient Separations and Processing (ESP) program while demonstrations, tests, and evaluations were performed by the Underground Storage Tank – Integrated Demonstration (UST–ID). In an effort to consolidate these various projects, EM created the Tank Focus Area (TFA) in late 1994. The entire UST–ID has already been incorporated into the TFA. In addition, the TFA is responsible for the technical integration of all tank waste remediation projects of local Waste Management organizations and OTD programs such as the ESP program; the Characterization, Monitoring, and Sensor program; and the Robotics program.

The TFA has developed six goals to measure the success of this new approach:

- Develop technology to meet waste tank needs across the DOE complex and apply all successfully demonstrated technologies.
- Provide an environment for cost-effective technology.
- Have an efficient management system for tank technology development.
- Clarify regulatory requirements that impact and drive technology development.
- Increase commercialization of tank technologies.
- Improve stakeholder satisfaction.

The management of TFA has divided the technology development efforts into six areas: safety, characterization, retrieval, pretreatment and processing, immobilization, and site closure. The Waste Pretreatment and Processing Program (WPPP) of the TFA is a key component since the amount and efficiency of the selected pretreatment processes will directly impact on the overall remediation costs of the USTs. The WPPP consists of eight separation projects from the former Waste Processing and Disposal Program (WPDP) of the UST–ID. The WPPP and former WPDP share four primary objectives (1):

- Demonstrate the removal of radionuclides and chemical toxicity from low-level waste.
- Demonstrate the removal of the constituents contributing to excess volume of high-level waste (HLW) from sludges and acidic wastes.
- Demonstrate technologies to maximize releasable or reusable fractions from the wastes.
- Demonstrate technologies to minimize the requirements for pretreatment chemicals.

These WPPP projects, as shown in Figure 1, can be divided into four systems: cesium removal, comprehensive sludge and supernate, out–of–tank evaporation, and cross–flow filtration. The cesium removal system, which is the largest with respect to funding levels, is comprised of the projects on the Cesium Extraction Testing (resorcinol-formaldehyde), the Hot Cell Testing on Cesium Sorbents, and the Cesium Removal Demonstration (CRD). In addition, part of the Comprehensive Supernate project supports the cesium removal system. The primary goal of the cesium removal system is to begin a 25,000–gal demonstration at the Melton Valley Storage Tanks (MVSTs) in Oak Ridge by the end of FY 1996. The other projects support the CRD through evaluation of potential cesium sorbents with batch and column tests. The remainder of the Comprehensive Supernate project, which also evaluates strontium and technetium removal processes, is part of the comprehensive sludge and supernate system. Two other projects, Technical Interchange with CEA and TRUEX Applications, are part of the comprehensive sludge and supernate system as well. These two projects focus on DIAMEX and TRUEX processes, which are competing solvent extraction techniques for the removal of tranuranics (TRUs) from acidic waste streams. While the out–of–tank evaporation and cross–flow filtration systems have only a single project, these programs are already heavily integrated with other local Waste Management projects. The current status of the former WPDP projects will be discussed.

RESULTS

Cesium Removal System

<u>Cesium Extraction Testing by J. P. Bibler of the Westinghouse Savannah River Company</u>. The primary mission of this project is to provide technical support to all DOE projects that evaluate the ability of the resorcinol–formaldehyde resin to remove cesium from the high-caustic, high-sodium, radioactive wastes at Hanford, Savannah River, and Oak Ridge. Waste Management personnel at Hanford and Oak Ridge

are strongly considering resorcinol for their baseline cesium removal technology. The resin can selectively remove micromolar amounts of Cs^+ (2) from solutions that are up to 11 M Na^+. Studies have shown that the resin is resistant to radiation as high as 5×10^8 R. Since the resorcinol resin is a condensationpolymer of the potassium salt of resorcinol and formaldehyde, it can be produced inexpensively in bulk quantities at a cost of $17.60/lb.

This project has three major activities, which include small–column tests on real Savannah River Site (SRS) supernate, a modeling study on ion–exchange processes with Purdue University, and resin degradation tests with Clark Altanta University. Preparations for the first task have been completed to test the resorcinol resin with real SRS supernate in a radioactive cave. The initial test indicated that the SRS HLW supernate must first be filtered to prevent plugging of the column by small particles. After the supernate has been satisfactorily filtered, two 5–mL columns of the resin will be used in series to treat 4 L of SRS supernate. Samples of the effluents from the first and second columns will be taken at regular intervals. The gamma counting will be used to determine a column profile for the resin. The second task utilizes a computer simulation package that was developed by researchers at Purdue University. The VErsatile Reaction and SEparation (VERSE) simulator is being used to design and develop ion–exchange processes for the removal of cesium from liquid nuclear waste. SRS and Pacific Northwest Laboratory personnel have conducted extensive column tests with simulants on the resin. Researchers at Purdue University have analyzed these results to determine void fraction, multicomponent ion–exchange isotherm correlations, and mass transfer parameters for the resin. With this information, the VERSE simulator should be able to predict effluent histories when system parameters, operating conditions, and modes of operations are provided. Bench-scale column experiments are currently under way to validate the VERSE predictions. The third task, determining the mechanism for resorcinol degradation, was initiated because the resin has a shelf life of approximately 3–5 years in storage. Researchers at Clark Atlanta University have determined that the resin degradation is due to a hydroxide-assisted oxidation to quinone and hydroquinone fragments. In order to prevent degradation in storage, the resorcinol resin should be stored dry and free of residual potassium hydroxide from the manufacturing process. The latter criterion may require that the resin be washed or converted to the acid form prior to shipment and storage.

<u>Comprehensive Supernate – Cesium Studies by B. Z. Egan of the Oak Ridge National Laboratory</u>. The primary mission of this project is to develop the preferred supernate processing flowsheet. In addition, realistic performance assessments for each individual process are underway, and operating results that are needed for the pilot-plant design are being generated. In order to perform realistic evaluations, approximately 5 L of actual sludge/supernate waste from the MVST W-25 at Oak Ridge National Laboratory was successfully transferred to a hot cell. The supernate was filtered through 0.45-µm filters before it was analyzed. The concentrations of sodium, nitrate, potassium, and ^{137}Cs in filtered tank W-25 supernate are 3.87 M, 3.8 M, 0.36 M, and 2.5E+5 bq/ml, respectively. The sorption measurements were made using batch equilibrium tests. Sorption data for equilibration times of 0.25, 2, 24, 72, and 144 h were determined for each exchanger tested. In these experiments, the mass of the exchanger was 0.050 g and the supernate volume was 10 mL; therefore, the solution/solid ratio was 200/1. In addition, equilibration isotherms and kinetic data were determined by varying the masses of the exchangers from 0.100 g to 0.005 g while the volume of the supernate, 10 mL, was held constant for mixing times of 0.25 to 144 h. Tests have been performed on the following exchangers: resorcinol–formaldehyde (SRR), Duolite CS-100, crystalline silicotitanate (CST), granular (gr) potassium cobalt hexacyanoferrate (KCoCF), fine powder (fp) KCoCF, hydrous titanium oxide/potassium cobalt hexacyanoferrate composite (HTiO/KCoCF), and titanium monohydrogen phosphate/sodium cobalt hexacyanoferrate composite (TiHP/NaCoCF) (3). Results of batch tests are reported as percentage removal (% R), and distribution ratio (D). These values are calculated in the following manner:

$\% R = 100[(C_o - C_t)/C_o]$ and

$D = [(C_o - C_t)/C_t][V/m]$ (units are mL/g).

The pretest and posttest count rates of ^{137}Cs are denoted by C_o and C_t, respectively. The quantity D is the ratio of the concentration of a radionuclide sorbed on the ion exchanger to the concentration remaining in the test solution after a specified mixing time, where (V) is the volume of supernate and (m) is the mass of exchanger. The results of these batch tests are presented in Table 1. Additional sorbent tests with MVST W-29 supernate from 1989 are under way. In addition, batch tests of a new IBC Advanced Technologies cesium sorbent, SuperLig 644, will also be tested in batch .

Hot Cell Testing on Cesium Sorbents by D. D. Lee of the Oak Ridge National Laboratory. This new project for FY 1995 will conduct column tests on a series of the potential cesium sorbents for the CRD. The CRD candidate sorbents include SSR, potassium cobalt hexacyanoferrate, CST, and SuperLig 644. Additional sorbents from the ESP program will be tested as these sorbents complete their developmental stage. The initial tests will process approximately 2 L of MVST supernate. The goal of each test is to reach a minimum of 50% breakthrough. The results from the bench–scale column tests and from the batch tests of the Comprehensive Supernate project will be compared and used in the sorbent down–selection for the CRD. In addition, the column experiments are expected to provide engineering design parameters for CRD. The rate of removal, the amount of needed sorbent, loading capacities, and pressure drops are important design parameters. These results will determine the size of the ion–exchange columns and the volume of loaded sorbent that will require final disposal.

Refurbishment of the hot cell and clearance documentation to use MVST supernate in the column tests of cesium sorbents have been completed. Based on the waste compositions of the MVSTs, the principal investigator and the CRD team have determined that the first column tests should be conducted with supernate from tank W-27. The final determination of the MVST for the CRD will be made based on consultations with the local Waste Management organization. The concentrations of sodium, nitrate, potassium, and ^{137}Cs in tank W-27 supernate are 4.13 M, 5.52 M, 0.29 M, and 5.3E+5 bq/ml, respectively. Since the pH of the supernate in tank W-27 is 7.2, sodium hydroxide will be added to the supernate so that the Oak Ridge supernate will more closely resemble the supernates at Hanford and Savannah River. A 50–L sample of W-27 supernate was retrieved and transported to the hot cell facility. The initial evaluation of cesium sorbents have been started. In addition, the effects of the sodium hydroxide additions will be analyzed.

Cesium Removal Demonstration by J. F. Walker of the Oak Ridge National Laboratory. This new project for FY 1995 will construct a modular, mobile ion-exchange system and will select and demonstrate the baseline cesium sorbent for the MVSTs. The CRD, which is a joint project between Waste Management and OTD, will address one of the highest–priority needs that the local Waste Management organization has identified for the MVSTs. The ^{137}Cs concentrations in most of the MVST supernates are already too high for grout stabilization and subsequent handling and transportation. The ion-exchange system will be installed in an existing containment facility and operated to remove ^{137}Cs from 25,000 gal of MVST supernate during FY 1996 and FY 1997. The system will be operated by remote control, and it will be designed for contact-handled maintenance. The loaded ion-exchange material will be moved to a hot cell for interim storage as a solid secondary waste. A local carrier for the loaded sorbent has been obtained. The carrier, which weighs 12,000 lb, has a shielding equivalent of 3 in. of lead. The material will then be sent to Savannah River for vitrification if feasible or to the Nevada Test Site for long-term storage. At the conclusion of the demonstration, the CRD system will be transferred to the local Waste Management organization for long-term remediation activities.

The preliminary specifications for the design and construction of the CRD system have been completed. These specifications are currently undergoing an internal review by personnel in the Engineering Development Section at Oak Ridge National Laboratory. The final design and construction of the ion-exchange system will be performed by a commercial vendor with expertise in the areas of radioactive materials processing and remote operations. Discussions were held with representatives from several different companies concerning the design and construction of the CRD system. Each company indicated that it would be interested in bidding for the design and construction of the CRD system in June

1995. Each company provided comments and suggestions on the proposed skid–mounted system. The preliminary cost estimates for the design and construction were between $150,000 and $250,000. The design of the system would take 4 to 6 months. Construction of the facility could be completed during the first 6 months of FY 1996.

Comprehensive Sludge and Supernate System

TRUEX Applications by G. F. Vandegrift of the Argonne National Laboratory. This WPPP project continues to broaden the applicability of the TRUEX process for use on HLW and TRU waste streams. The TRUEX process (4) is a solvent extraction process that uses octyl (phenyl)-N,N-diisobutylcarbamoylmethylphosphine oxide (CMPO) as the key ingredient. The TRUEX process can effectively separate the TRU components from aqueous nitrate and chloride solutions. These solutions are typically generated in nuclear fuel reprocessing plant operations in plutonium production and purification. A successful separation reduces the amount of TRU waste that must be processed by the HLW vitrification plant. Most of the resulting solutions from the TRUEX process can be disposed of as a low-level waste, which will greatly reduce the overall cost of final disposal. OTD demonstrated the TRUEX process with dissolved Mark 42 targets at Oak Ridge during FY 1993 (5). Currently, the TRUEX process is the reference process for TRU removal at Idaho National Engineering Laboratory.

In an effort to increase the utility of the TRUEX process, the Center for TRUEX Technology Development developed the Generic TRUEX Model (GTM) as a tool for designing site- and feed-specific TRUEX flowsheets and for estimating the space and cost requirements for installing a TRUEX process (6). Recent improvements to the GTM included the following: (1) the number of aqueous species for which concentrations can be calculated was increased from 146 to over 200, and (2) the accuracy of the distribution ratio calculations and solvent loading was improved by the addition of the extraction due to tributylphosphate. The Waste Management organization at Idaho is currently funding the additions of two important constituents, Hg^{2+} and Cl^-, in their sodium–bearing waste in the GTM.

Commissariat à l'Energie Atomique Interchange by R. T. Jubin of the Oak Ridge National Laboratory. Numerous countries such as France, Japan, and England are continuing to fund the development of new technologies for the remediation of nuclear waste. These efforts have produced several separation technologies that are relevant to DOE remediation needs. Therefore, it is in the national interest to expand our collaborative agreements with the major foreign organizations that have developed appropriate remediation technologies. In particular, the Commissariat à l'Energie Atomique (CEA) Interchange is a prime candidate for a collaboration due to its long-term development program on the separation of long-lived radionuclides.

Technical interchange, which began in early 1994 with the placement of an ORNL engineer at the CEA, is focusing on the French DIAMEX process, which uses diamides as the extractants for the actinides in acidic solutions. At the conclusion of the interchange at the end of FY 1995, the DIAMEX process will be evaluated against comparable DOE technologies such as the TRUEX process. During the technical interchange, the ORNL engineer has focused on three key research and development areas for the DIAMEX process. The first area is to develop a method to control or suppress the extraction of molybdenum (VI) by malonamides. The removal of molybdenum is important in the control of third phase formation. A flowsheet has been designed to demonstrate the suppression of molybdenum and zirconium. The underlying approach of the flowsheet was based on the use of two separate scrubbing sections. Hydrogen peroxide will be used to complex the molybdenum while the zirconium will be complexed with ketomalonic acid. The proposed flowsheet utilizes 21 stages for the extraction and molybdenum scrub and 16 stages for the zirconium scrub and secondary extraction. Analysis of this flowsheet has been initiated to determine the required run duration and to identify possible reflux situations that could occur during the startup. A demonstration run with simulants is expected to occur in May 1995. The second area focused on the hydraulic problems that were observed during the hot demonstration of the DIAMEX process in the CYRANO hot cell. Flow problems occurred in the mixer/settler of the extraction bank. Therefore, the number of stages that could be utilized was limited. Several hydraulic tests have shown that a modified impeller has eliminated the flow problems. The third area addresses the use of centrifugal contactors for the DIAMEX process. The rapid contactors will be evaluated as a means to improve decontamination factors through kinetics effect.

Comprehensive Supernate – Strontium and Technetium Studies by B. Z. Egan of the Oak Ridge National Laboratory. In addition to the cesium sorbents, the comprehensive supernate project is evaluating technologies to remove strontium and technetium from alkaline supernate after the cesium has been removed. For the strontium batch tests, cesium and strontium were removed from a sample of MVST W-29 supernate. The Sr concentration in the supernate was then adjusted to 2 mg/L using a stock solution of strontium nitrate with ^{85}Sr tracer. The solution was mixed for several days and then filtered through 0.45– and 0.2–μm nylon filters to remove any strontium particulates. Seven different sorbents were tested for the removal of strontium from the supernate. In each test, the mass of sorbent equivalent to approximately 50 mg of air–dried material was mixed with 10 mL of supernate for 2 h. The strontium distribution coefficients were 4200 mL/g for sodium titanate on polyacrylonitrile (PAN), 4300 mL/g for titanium monohydrogen phosphate on PAN, 1200 for CST, 990 for Duolite C-467, 850 for sodium titanate (Cerac), 650 for Chelex 100, 620 for Amberlite IRC-718, and 210 for SRR.

The technetium concentration in the MVST W-29 supernate is 3.2E−7 *M* while the technetium concentrations at Hanford typically range from 6E−7 to 4E−4 *M*. After the cesium and strontium were removed from the W-29 supernate, the technetium concentration was raised to 4E−5 *M* through the addition of ammonium pertechnetate. In each batch test, a 10-mL sample of adjusted supernate was mixed with an air–dried equivalent of 0.05 g of exchangers for 24 h. For Reillex HPQ, the technetium distribution coefficients were 624 and 511 mL/g for the hydroxide and nitrate forms, respectively. The technetium distribution coefficients were 786, 412, and 535 mL/g for Reillex 402, the hydroxide form of Amberlite IRA-400, and the nitrate form of Amberlite IRA-904, respectively.

Out–of–Tank Evaporator System

Out–of–Tank Evaporator Demonstration by A. C. Lucero of the Oak Ridge National Laboratory. This new project between OTD and the Waste Management organization at Oak Ridge will demonstrate the feasibility of a mobile evaporator for the treatment of UST supernate. The Out–of–Tank Evaporator Demonstration (OTED) will process 25,000 gal of supernate from a MVST tank. Approximately 19,000 gal of concentrate will be returned the MVST, while 6,000 gal of condensate will be collected and transferred to the Oak Ridge National Laboratory process waste system. The OTED has chosen to evaporate the supernate from W-24 based on preliminary characterization data on the MVSTs because the nitrate concentration would remain below 5 *M* even after a 25% volume reduction. The current concentrations of sodium, nitrate, potassium, and ^{137}Cs in MVST W-24 are 3.25 *M*, 3.18 *M*, 0.53*M*, and 1.1E+6 bq/ml, respectively. With a 25% volume reduction, the concentrations of Na, NO_3^-, K, and ^{137}Cs in W-24 would be 4.33 M, 4.25 M, 0.71 M, and 1.45E+6 bq/mL, respectively. Personnel at Argonne National Laboratory are performing tests with a W-24 simulant and a small bench-scale evaporator that is comparable to the OTED evaporator. These Argonne National Laboratory experiments will help to identify potential problems and to determine the relationship between the performance of small-scale evaporators on surrogate solutions and full-scale units on actual supernate.

A skid-mounted single-stage subatmospheric evaporator unit will be delivered by Delta Thermal Inc. by May of 1995. The evaporator will then be installed in Building 7877. Temporary shielding, heating and cooling systems, and tanks for the OTED will be added as necessary to meet Oak Ridge National Laboratory safety requirements. The OTED should complete its evaporation phase by the start of FY 1996. The evaporator will then be decontaminated and transferred to the local Waste Management organization for additional evaporation campaigns. The OTED will determine processing capabilities such as decontamination factors, identify potential operation/maintenance problems with remotely operated evaporators, and evaluate the feasibility of decontaminating the evaporator systems for hands-on maintenance or movement to another site.

Cross–Flow Filtration System

High–Level–Waste Process Filter Testing Program by D. J. McCabe of the Westinghouse Savannah River Company. This new project is evaluating the filtration needs for specific pretreatment processes that are proposed for the UST remediation efforts at Hanford and Oak Ridge. The personnel from the two tank sites have described their solid/liquid separation needs and their relative importance. In addition, the status of solid/liquid separation projects at Hanford and Oak Ridge has been discussed. Since SRS personnel have years of experience with filtration for its remediation efforts, this SRS project has assisted the other sites in the development of test strategies for their proposed waste stream flowsheets. In particular, the SRS personnel have extensive knowledge in the testing and operation of cross–flow filters. This project is currently ranking the solid/liquid separation needs based on the site priority and the potential for success with cross–flow filters.

This project will then initiate cross–flow filter tests on the highest–ranked solid/liquid separation need. The concentration of TRU sludges at Oak Ridge and removal of sludge particulates from Hanford supernate are leading candidates for the initial tests. After the waste stream is selected, simulant preparation and characterization will be done in collaboration with the appropriate Hanford or Oak Ridge personnel. This simulant formulation will be the key to a successful program. The initial experiments are expected to be conducted with a commercial metal or ceramic filter. The tests will primarily be run on a laboratory-scale cross–flow filtration unit. This study will characterize waste behavior through the examination of filter capacity, operating conditions, removal efficiencies, pump requirements, and chemical cleaning. After the engineering conditions and operation parameters have been determined, it is expected that a cross–flow filtration demonstration on the actual waste stream will be conducted with the support of this project.

CONCLUSIONS

The greatest technical and financial challenge facing DOE is the remediation of the USTs. In an effort to reduce remediation costs, improve safety, and minimize delays, the WPPP has been conducting demonstration, tests, and evaluations on new separation and treatment technologies. The current WPPP projects focus on cesium removal, comprehensive sludge/supernate processing, solid/liquid separation, and mobile evaporators. The cesium removal projects are designed to support the 25,000–gal cesium removal demonstration at Oak Ridge. The objectives of the comprehensive sludge/supernate studies are to develop a complete system-level plan for handling sludge/supernate and to evaluate TRUEX and diamide solvent extraction processes for TRU waste streams. The WPPP projects on cross-flow filtration and mobile evaporators are directly addressing key needs that have been identified by the tank sites. The WPPP projects have and will continue to impact the selection of the processes for the full-scale remediation of the USTs.

Acknowledgements

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Table 1. Batch adsorption data showing the effect of the mixing time on the removal of cesium from MVST W-25 supernate

	Mixing time (h)														
	(0.25)			(2)			(24)			(72)			(144)		
Exchanger	D	%R	pH	D	%R	pH	D	%R	pH	D	%R	pH	D	%R	pH
CS-100	34	15.3	12.2	35	20.0	12.2	34	15.0	12.2	42	20.0	12.6	44	22.0	12.6
SRR	138	41.0	12.6	763	79.3	12.6	736	79.5	12.6	764	79.2	12.6	641	78.7	12.6
CST	451	71.5	12.6	662	77.4	12.6	672	77.5	12.6	672	77.7	12.6	958	83.8	12.6
KCoCF (gr)	36,900	99.5	12.3	46,200	99.6	12.3	36,900	99.5	12.3	36,300	99.5	12.2	26,000	99.3	12.2
KCoCF (fp)				49,900	99.6	12.3	66,800	99.7	12.3	18,630	99.0	12.2			
TiHP/NaCoCF				1,140	96.9	9.4	3,105	98.8	9.4	3,960	99.1	9.4			
HTiO/KCoCF				110	72.3	12.3	5,550	99.3	12.3	5,500	99.3	12.3	5,530	99.3	12.3

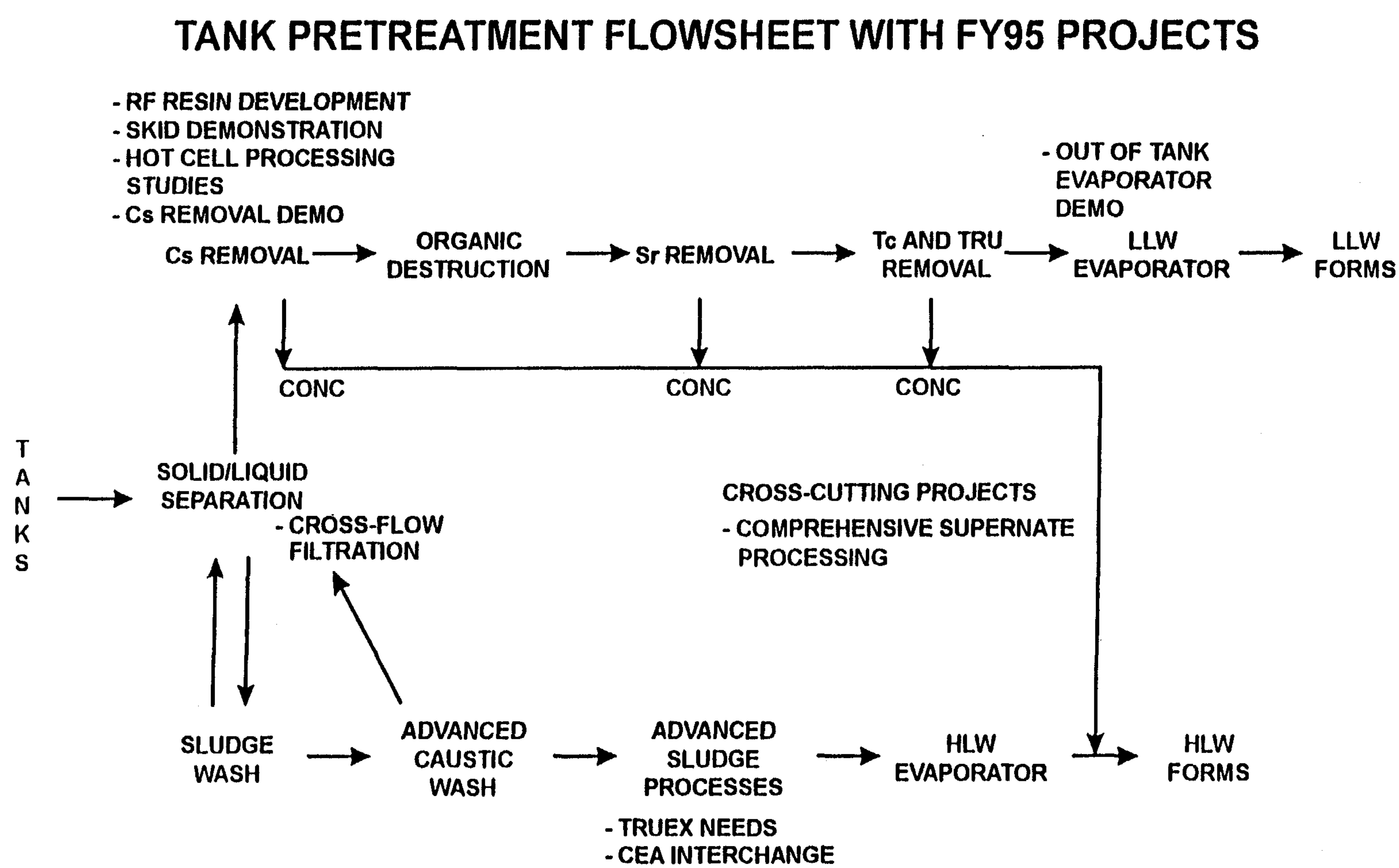

Fig. 1 Relationship diagram for all projects in the Waste Processing and Pretreatment Program