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ATLANTIC RICHFIELD HANFORD COMPANY
QUARTERLY REPORT
WASTE MANAGEMENT AND TRANSPORTATION
TECHNOLOGY DEVELOPMENT
APRIL 1975 THROUGH JUNE 1975

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ABSTRACT

This document is the fourth in a series of quarterly reports on work in support of long-term management of Hanford high-level wastes. The work reported here was performed during the period April through June 1975. The specific topics discussed are grouped into the subject areas of:

- *Storage System Integrity and Engineered Improvements*
- *Waste Retrieval*
- *Waste Immobilization and Storage*
- *Contaminated Equipment Volume Reduction*

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INTRODUCTION

The high-level radioactive wastes resulting from Hanford operations must be stored or disposed of in a manner that at all times positively assures the protection of the public and the environment. As currently envisioned, long-term Hanford waste management plans are directed toward eventual removal of waste from underground storage tanks, conversion of the waste to an improved stabilized form, and then either interim storage of the immobilized waste in an engineered facility or terminal storage in a location that does not require controlled management. The entries in this report under Storage System Integrity and Engineered Improvements are directed toward establishing a data base and methodology for predicting the length of time that waste can safely remain in the tanks, and determining and evaluating engineered improvements which might be used to extend the safe in-tank storage period. Identification, development, and demonstration of the technology necessary for retrieval of the waste from underground tanks are discussed under Waste Retrieval. The Immobilization and Storage section of the report provides a summary of work associated with technology for the conversion of retrieved waste to an immobilized form satisfactory for long-term storage, and with the evaluation of alternative engineered and terminal

storage methods. Contaminated Equipment Volume Reduction addresses the development and demonstration of methods for reducing the volume of radioactively contaminated metallic items to a size and form suitable for long-term storage or disposal.

Sponsorship of these continuing programs will be assumed during the next quarter by the Energy Research and Development Administration. The Contaminated Equipment Volume Reduction Program is under the Division of Environmental Control Technology; the other programs are under the Division of Nuclear Fuel Cycle and Production. Subsequent reports will acknowledge the change.

SUMMARY

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Laboratory experiments to determine combustion limits for various sodium nitrate-organic mixtures were initiated and detonation testing of selected mixtures was concluded by Stanford Research Institute.	6
Determination of the long-term integrity of high-level waste storage tank concrete will be aided by the results of experiments in progress by the Portland Cement Association. Short-term temperature resistance studies at 21° and 177° C were completed and specimens prepared for prolonged exposure studies.	7
Preparation of the preliminary fault tree analysis report and the fault tree analysis computer program document was essentially completed.	9
Progress was achieved in preparation of the computer model for predicting radionuclide transport in soil. Preparation of a report describing the model was started.	10

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Preparation and review of documentation defining the distribution of hydraulic conductivity and transmissivity across the Hanford Reservation, with emphasis on the high-level waste storage areas, was continued in support of long-term groundwater management studies.	11
The mechanical retrieval development phase of waste retrieval studies received added impetus. A contract was placed with Programmed and Remote Systems Corporation for the preliminary design of a mechanical waste retrieval system.	13
Two agitator pumps and their hydraulic power units were received for bench-scale tests of hydraulic retrieval methods. Preparation of a test area in the 202-S Building was started.	14
As part of the materials characterization studies for the wastes to be retrieved, the sodium nitrate crystal size distribution in salt cake solids was determined for solids made from synthetic terminal liquor. Models were formulated for diatomaceous earth/salt cake and U Plant sludge.	15
A Gradall extendable boom and auxiliary equipment were procured and detailed design of necessary modifications started by Automation Industries, Inc., Vitro Engineering Division.	17
Leach rate data for both Lean and Rich Clay processes were determined to provide another basis for comparison.	18
The kinetics of cesium uptake were studied, and half-reaction times reported for the Lean Clay process. Related work on rich clay suggests that cesium and strontium may occupy different sites in the product.	19
Heats of reaction for calcined and uncalcined clay-waste reactions show that heat must be removed from calcined clay-waste reactions to prevent vaporization of the liquid phase.	20
Thermogravimetric analysis of clay- NaNO_3 and clay-synthetic sludge mixtures showed that the "Clay Calcination" process can be operated at lower	22

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temperatures than previously thought, thus reducing Cs volatilization and simplifying processing equipment.

Designs and cost estimates were prepared for a non-radioactive aqueous silicate pilot-plant. 24

Compositions of Redox and Purex sludge were determined for use in extensive Silicate Melt process screening tests to be conducted next quarter. 25

The corrosiveness of simulated Hanford waste glasses on six refractories was determined. Monofrax K-3 was the most resistant refractory tested and will be used in the one-ton-per-day glass melter which is scheduled to be completed by Battelle-Northwest next quarter. 29

Laboratory and engineering plans were prepared for identification, development, and evaluation of alternative immobilization concepts. 32

Scouting studies were conducted for cesium, strontium, and technetium sorbents, and some experimental work was performed to obtain equilibrium data for use in Radionuclide Removal flowsheet development. 34

Norman Engineering Company was retained to prepare a preliminary engineering evaluation and capital cost estimate for each of the various immobilization and storage alternatives delineated in the last quarterly report. 39

As part of the Contaminated Equipment Volume Reduction work, the plasma arc furnace torch replaced the sweeping electrode electric arc as the size-reduction method to be investigated. Single-step size-reduction and meltdown tests using powdered silicon and gaseous oxygen and powdered silicon and sodium nitrate were unsuccessful and will not be investigated further. 40

STORAGE SYSTEM INTEGRITY AND ENGINEERED IMPROVEMENTS

The objectives of this program are:

1. Evaluation of the present underground storage system to predict how long it will provide safe storage.
2. Identification, development, and testing of engineered improvements which could be utilized to extend the safe storage period of underground tanks.
3. Definition of the Hanford groundwater flow system and determination of the extent of groundwater management necessary to control rate and direction of radionuclide movement in the high-level waste storage areas.

SALT CAKE STABILITY - G. A. Beitel and L. E. Bruns

Since potential explosive or combustion reactions between organic materials and salt cake nitrates are of concern in activities associated with the long-term management of Hanford high-level wastes, studies and tests have been initiated to determine if such reactions are possible. Shock sensitivity tests conducted in July 1974 by Stanford Research Institute (SRI) on several sodium nitrate-organic resin mixtures showed no evidence of shock-initiated detonation. Following completion of a literature survey and calculation of maximum theoretical flame temperatures for mixtures of NaNO_3 , carbon, and water, laboratory experiments were designed to determine actual combustion limits for various NaNO_3 -organic mixtures. Preliminary tests conducted in Atlantic Richfield Hanford Company laboratory facilities with mixtures of NaNO_3 and activated charcoal confirmed the

theoretical calculations.

Testing of various NaNO_3 -organic mixtures was continued in ARHCO laboratory facilities during this report period. Some preliminary results were obtained. A mixture of 85% NaNO_3 and 15% Dowex 21-K resin (as-received) could not be ignited. A mixture of 88% NaNO_3 and 12% charcoal would not burn if ignited at room temperature but would burn if held initially at 150°C . A nearly stoichiometric mixture of 80% NaNO_3 and 20% charcoal was the hottest and easiest-to-burn mixture tested. Mixtures with the latter composition and up to 20% moisture content (mixture mostly fluid at 20% moisture) were successfully ignited and burned.

Detonation studies using standard gap test methods were conducted by SRI. Tests with 5-cm diameter samples of simulated salt cake-organic mixtures showed no reaction. Tests using about 5 liters of NaNO_3 -resin mixtures in 15-cm diameter tubes showed definite reactions in three of four tests following detonation of an explosive on the opposite side of the gap from the samples. The test reaction velocities shown in Table I approached, but did not equal, detonation velocity of 1500 m/sec. The significance of these reaction velocities has not yet been determined.

TABLE I
DETONATION TEST RESULTS WITH 15-cm SAMPLES
OF SODIUM NITRATE-RESIN MIXTURES

Test Mixture		Reaction Velocity
% NaNO_3	% Resin	m/sec
80	20	1100
85	15	1100
90	10	None
95	5	1100

No further detonation testing is planned at this time but may be considered following review of the final SRI report.

Laboratory testing of NaNO_3 -resin mixtures (both dry and containing moisture) will be continued and a report on chemical stability of salt cake-organic mixtures will be prepared during the next quarter. Analytical equipment to determine the organic-carbon content of actual salt cake will be received.

CONTAINMENT INTEGRITY - E. L. Moore and F. R. Vollert

The feasibility of continuing to use the Hanford waste storage tanks as an interim storage mode will be highly dependent on an evaluation of the effects of corrosion and deterioration on the tanks. Because of the variety and range of compositions of wastes affecting the evaluation, experimental programs have been developed to investigate corrosion of the mild steel liner and reinforcing bars and deterioration of the concrete tanks.

A program utilizing electrochemical techniques to study the effects of corrosion on the carbon steel liner was conducted by Battelle Memorial Institute - Columbus Laboratories. Potentiodynamic polarization, constant strain-rate, and linear polarization experiments on mild steel exposed to simulated solidified wastes were completed. Although the final report has not been received, interim reports indicated that general corrosion was low with no pitting detected. Stress corrosion cracking by several of the highly caustic waste compositions was minor-to-moderate.

The results of experiments on concrete temperature initiated by the Portland Cement Association (PCA) during this report period will be used in the determination of the

long-term integrity of high-level waste storage tanks. Elastic properties, tensile splitting strengths, compressive strengths, and Poisson's ratio will be determined as a function of temperature, exposure time, and curing time. The following tests were completed on Hanford concrete specimens of 3000 and 4500 psi strength following short-term exposure:

Seven compressometer tests (21° and 177° C).

Eight sonic tests (21° C).

Eight sonic tests (21° and 177° C).

The specimens exhibited compressive strengths well above the minimum required field strengths. A final report on this phase of the studies is due early next quarter. Preparation of specimens for the second phase of the program was initiated. The second phase will involve investigation of prolonged exposure to the test temperatures. The program is being coordinated and directed by Battelle Pacific Northwest Laboratories.

A separate contract was awarded to PCA to investigate deterioration of concrete exposed to simulated waste solutions.

During the next quarter the final report on the corrosion studies conducted by Battelle Memorial Institute - Columbus Laboratories will be reviewed and the determination made as to whether more extensive testing is warranted. Curing of the specimens for the second phase of the concrete temperature resistance program will be completed and heating of the specimens will be initiated. Testing of concrete specimens in simulated waste solutions will begin.

STORAGE SYSTEM ANALYSIS - W. J. Van Slyke

This task was originally identified as Fault Tree Analysis. Early work focused on obtaining a valid fault tree diagram for the high-level waste storage system, developing computer fault tree analysis techniques, and developing a computer-graphics method for producing fault tree diagrams for reports. The objective was changed to identifying those high-level waste storage system characteristics which should receive the highest priorities of attention with respect to engineered improvements or administrative procedures.

The preliminary draft report was completed in January and revisions to the program completed during the last quarter. The code now incorporates probability calculation routines in accordance with the methodology espoused in the Rasmussen Report, WASH-1400.

In response to a request by the Energy Research and Development Administration, Richland Operations Office, revision of the preliminary fault tree analysis report for issuance as a limited distribution report was initiated. Preparation of the fault tree analysis computer program document, including a short program which writes a table of fault tree component interrelationships, was also in progress during this report period.^[1] The report will be submitted to the Argonne Code Center.

Future work will include completion of the preliminary fault tree analysis report and the fault tree program document.

RADIONUCLIDE MIGRATION IN SOILS - R. A. Deju

The objectives of this program are to (1) define the Hanford groundwater flow system and its hydrological properties, (2) predict the long-term movement of radionuclides, (3) predict the impact of off-site irrigation, (4) identify radionuclide transport mechanisms and rates, (5) develop a computer model of radionuclide movement from high-level waste areas, and (6) assess the impact on man of radionuclide movement through the Hanford environment.

Geophysical gravity field data have been collected from 600 locations throughout the Hanford Reservation. Studies have been initiated to evaluate the effects of both the irrigation activities in the Cold Creek Valley and the aquaculture project located just northwest of the Reservation. Development of the computer model has been initiated.

The geophysical gravity field data were evaluated during this report period. Maps were drafted to show the location and distribution of several gravity highs and lows within and adjacent to the Hanford Reservation. This work is essential in defining the tectonic structure and will aid in identifying potential radionuclide travel paths.

Part of the debugging and checking of the computer model was almost completed with a major effort directed toward overcoming the small-numbers problems arising from small values of capillary conductivity in very dry soils. A selected set of experimental data on moisture and heat fluxes, and corresponding soil properties, were obtained from Dr. Ray Jackson of Phoenix, Arizona, Laboratory of the Agriculture Research Service, U. S. Department of Agriculture. These and additional data from Dr. Jackson will be used to verify the code model.

A progress report was started which will present:

1. Equations obtained from a comprehensive literature search and a multidisciplinary evaluation to form the basis of a theoretical description.
2. A verbal description of the numerical solution methods utilized with more detailed description of Gear's Methods, etc., in appropriate appendixes.
3. Limitations and comparisons necessary to experimentally verify the theoretical description.

During the next report period the computer model will be completed and the program used to solve a series of initial boundary value problems. A program user's guide will also be provided. Sediment permeabilities and waste discharge volumes for the aquaculture project will be measured.

LONG-TERM GROUNDWATER MANAGEMENT - R. A. Deju and
R. E. Gephart

The hydrologic properties comprising the Hanford groundwater flow system are required to predict long-term radionuclide transport, control the rate and direction of radionuclide movement in the high-level waste storage areas, and determine groundwater management needs. Two geohydrologic maps have been drafted showing the distribution of hydraulic conductivity and transmissivity values over the Hanford Reservation. Corresponding computer plots were run using the Transmissivity Iterative Routine and Variable Thickness Transport models.

The review of documentation on defining the distribution of hydraulic conductivity and transmissivity across the Reservation, with emphasis on the high-level waste storage

areas, was continued this quarter. The maps were reviewed by several consulting geohydrologists to assure valid data interpretation. This effort is summarized in a report entitled *Transmissivity and Hydraulic Conductivity of Saturated Sedimentary Rocks in the Hanford Reservation*. This document was drafted this quarter and will be completed during the next report period.

WASTE RETRIEVAL

The objective of this program is to develop and demonstrate the technology necessary to retrieve radioactive wastes (salt cake, sludges, liquids, and accumulated debris) from underground storage tanks.

CONCEPT AND EQUIPMENT DEVELOPMENT - N. C. Rodewald and D. O. Schuler

The objectives of this task are:

1. Definition of areas of the waste retrieval program which will require technology development and prototype evaluation.
2. Identification of potential retrieval techniques.
3. Preliminary testing of the identified retrieval techniques to establish feasibility.
4. Conceptual design of prototype retrieval systems based on the results of feasibility testing.

Mechanical Retrieval - N. C. Rodewald and D. O. Schuler

The objective of this subtask is to demonstrate the feasibility of an in-tank waste retrieval system based on mechanical digging equipment. The primary concept presently

being evaluated is based on a remotely controlled hydraulic boom with various mining/digging tools mounted at the end. Norman Engineering Company of Los Angeles, California, is conducting an engineering evaluation of alternative retrieval concepts.

A contract was placed with Programmed and Remote Systems Corporation, St. Paul, Minnesota, for the preliminary design of a mechanical waste retrieval system with an option to procure prototype hardware pending approval of the design. Work at PaR under the contract has begun.

A document is in preparation which summarizes bench-scale testing efforts to date and describes the mechanical retrieval system based on the Gradall boom. Bids have been received on a waste conveyor and on a hydraulic demolition hammer.

A preliminary Waste Retrieval Facility study report was received from Norman Engineering Company. The report was reviewed by ARHCO personnel and comments forwarded to Norman for incorporation in the final report, which is to be completed by September 30, 1975.

Development work will continue on the mechanical retrieval systems. Bench-scale testing will be continued in support of mechanical retrieval tool selection.

Hydraulic Retrieval - N. M. Arnold and N. C. Rodewald

The objective of this subtask is to evaluate the feasibility of in-tank waste retrieval using hydraulic techniques. The primary technique presently under consideration is the use of a combination agitator-pump to agitate the salt cake in its own interstitial liquor and pump the resulting slurry out of the tank. This technique could also be used for the retrieval of sludge if the moisture content were sufficiently high.

Two agitator-pumps and their hydraulic power units were received from Picatti Brothers of Yakima, Washington. The two pumps are (1) a Vaughn Model 162 with a 5-cm discharge capable of passing 2.5 cm solids, driven by a 60 hp, 3600 r/min hydraulic motor; and (2) a Vaughn Model 300 "Scumbuster" with a 10.2-cm discharge capable of passing 6.1 cm solids, driven by a 75 hp, 1800 r/min hydraulic motor. The Model 162 was designed to pass through a 30.5-cm diameter riser. The larger Model 300 requires at least a 0.91-m diameter opening. A pump test loop has been designed for the Model 300. A suitable area for testing the Model 300 is being set up in the 202-S Building. The loop will be used to obtain hydraulic pump curves with salt cake liquid content as a prime variable.

Feasibility testing of hydraulic waste retrieval will continue. An area will be set up for the test and evaluation of the Model 162 pump. Both pumps will be evaluated in various synthetic representations of in-tank wastes.

Pneumatic Retrieval - N. C. Rodewald and D. O. Schuler

The objective of this subtask is to evaluate the feasibility of using pneumatic air transport techniques for the retrieval of in-tank wastes. Previous testing has proven pneumatic retrieval feasible for certain forms of waste. A pneumatic retrieval system would have to be combined with some type of mechanical or hydraulic breakup device.

A quotation was received from a vendor for a basic pneumatic system that could be used for on-site development testing. No procurement action is planned at this time but a basic pneumatic system may be obtained at a later time for test and evaluation.

Material Flow Properties - C. H. Delegard and N. C. Rodewald

The objective of this subtask is the characterization of the static and dynamic properties of salt cake, sludge, and other waste forms under conditions proposed for the retrieval operation. Data from this program and the Salt Cake Characterization program will aid in selecting retrieval concepts and furnish input to formulate realistic, simulated salt cake for testing retrieval equipment.

The sodium nitrate crystal size distribution in salt cake solids was determined for solids made from synthetic terminal liquor. The crystal size distributions at crystallization temperatures of 20° and 40° C were nearly identical, while 60° C crystallization slightly favored the growth of smaller crystals. The approximate size distribution was, by weight: 0.3 (<10 mesh), 0.2 (10 - 14 mesh) 0.3 (14 - 20 mesh), and 0.2 (>20 mesh). Well-drained salt cake made at this particle size distribution had a bulk density of 1.56 g/ml. Variations of this typical size distribution resulted in denser salt cake. However a volumetric phase diagram for these various salt cakes revealed little difference in the ratios of solid NaNO_3 /interstitial liquor/void volume. Generally, the volume fractions of the phases were 0.45 to 0.50 solid NaNO_3 , 0.30 to 0.45 interstitial liquor, and 0.10 to 0.30 void. The phase diagram results suggest that salt cake drainage and pumping may remove only 0.33 to 0.50 of the interstitial liquor in genuine salt cake. Therefore even in the best-drained salt cake significant quantities of interstitial liquor may be present during retrieval operations.

A diatomaceous earth/salt cake model was formulated. This material exhibited plastic behavior in viscometric studies. Viscosities ranged from 50 to 240 poises as the weight fraction of the diatomaceous earth in the model

ranged from 0.065 to 0.905. Diatomaceous earth content throughout Tank 116-TX is roughly 0.018 wt% but it is undoubtedly much higher on the top where the diatomaceous earth accumulated.

A U Plant sludge model was formulated based on U Plant effluent concentrations. This sludge, in weight fraction, is 0.31 NaNO_3 , 0.15 Na_3PO_4 , 0.023 Na_2SO_4 , 0.020 $\text{Fe}(\text{OH})_3$, and 0.50 H_2O ; it has a density of 1.495 g/ml.

Future work will include investigations of liquor retention in salt cake versus interstitial liquor surface tension.

COLD DEMONSTRATION

The objectives of this task are:

1. The design and fabrication of prototype waste retrieval equipment.
2. The test and evaluation of this equipment on representative (nonradioactive) waste forms and in-tank retrieval problems.
3. The selection of equipment and waste retrieval techniques for the hot demonstration.

Cold Test Facility - D. O. Schuler

A test stand or facility will be required to provide a place where prototype retrieval equipment can be mounted and tested in a manner simulating in-tank conditions.

Automation Industries, Inc., Vitro Engineering Division, has completed the preliminary cost estimates of four possible options for the cold test stand. Two of the four options were dropped. The two options remaining are (1) modifying

the 190-D reuse Basin in the 100-D Area and (2) building a completely new facility in the 200 West Area. Final site selection will depend on the available funding and the type of equipment selected for test.

A site will be selected and final criteria developed for a cold test stand. Possible funding alternatives will be investigated.

Gradall Mechanical Retrieval System - D. O. Schuler

A Gradall extendable boom, control cab, and the hydraulic power system were procured. Vitro Engineering was requested to begin detailed design of the necessary modifications and the new hardware required to adapt the Gradall boom for in-tank retrieval.

IMMOBILIZATION AND STORAGE

The objectives of this program are:

1. Identification, development, and demonstration of technologies for converting high-level waste to an immobilized form satisfactory for long-term storage.
2. Evaluation of alternative engineered and terminal storage methods.

AQUEOUS SILICATE PROCESS

The conversion of fission product wastes to solid, insoluble products suitable for interim or terminal storage is being investigated through the Aqueous Silicate process. This process utilizes reactions of clay with waste to form stable products. Basically, three processing options are

available in the Aqueous Silicate process. The first option, called the Rich Clay process, features low-temperature aqueous reaction of a stoichiometric excess of clay with waste solution or sludge. For the second option (the Lean Clay process), powders resulting from the stoichiometric aqueous reaction (again at low temperature) of waste solution with clay can be formed by addition of binders into products harder and of smoother surface than rich clay products. In the third option, called the Clay Calcination process, dry waste materials (such as salt cake or other dehydrated waste solutions) are reacted with clay at $\sim 600^{\circ}$ - 800° C to form water- and nitrate-free sodium aluminosilicate products.

Leach Rate Reduction Studies - C. H. Delegard

As an extension to data presented in the previous quarterly report,^[2] bulk leach rate data based on cesium have been determined for a lean clay product containing a zinc silicate binder. Using a calcined kaolin (J. R. Simplot's PAB[®], calcined) and a mixed kaolin-bentonite (Interpace Corporation's Indian Hill[®]), these studies show that leach rates decreased in the order: rich clay > lean clay-portland cement binder > lean clay-zinc silicate binder. The kaolin-bentonite products, as expected, gave lower leach rates than the comparable calcined kaolin products. Table II contrasts the leach data for these products. These leach data offer another basis of comparison for the Lean and Rich Clay process options in the Aqueous Silicate process.

TABLE II
LEACH DATA FOR RICH CLAY AND
LEAN CLAY AQUEOUS SILICATE PROCESSES

<u>Process/Clay</u>	<u>Leach Rate (g/cm²-day) *</u>	
	<u>24 hours</u>	<u>504 hours</u>
Rich Clay		
PAB, calcined	2.4×10^{-1}	1.4×10^{-2}
Indian Hill	2.6×10^{-3}	6.7×10^{-4}
Lean Clay-		
Portland Cement		
PAB, calcined	9.6×10^{-3}	4.4×10^{-4}
Indian Hill	1.2×10^{-3}	9.0×10^{-5}
Lean Clay-Zinc		
Silicate		
PAB, calcined	9.8×10^{-4}	4.6×10^{-5}
Indian Hill	1.5×10^{-4}	9.3×10^{-6}

$$*\text{Leach rate} = \frac{(\text{fraction Cs leached})(\text{weight sample, g})}{(\text{surface area sample, cm}^2)(\text{time of leaching, days})}$$

Mechanism of Cesium Fixation - C. H. Delegard

The kinetics of cesium uptake were studied for a lean clay product made from uncalcined kaolin (Georgia Kaolin Company's 6-T[®]). At 100° C, cesium uptake in the reaction solids generally followed the solids weight gain curve by increasing rapidly at first, then leveling out smoothly in two to three days reaction time. Half-reaction time was roughly one hour. This compares with the half-reaction time of 30 minutes for the uncalcined bentonite lean clay reaction. Complete reaction of uncalcined bentonite occurred in four to six hours. Cesium uptake data like these aid in designing reactors for full-scale Lean Clay processes.

In other related work, cesium leach rates from rich clay products were found to be unaffected by variations in ⁹⁰Sr concentration in the feed waste liquor. The ⁹⁰Sr

concentrations in the feed liquor taken from Tank 110-BX were varied from 0 to 1.5×10^5 $\mu\text{Ci/liter}$ by ^{90}Sr spikes. The invariance of the cesium leach rates with varying ^{90}Sr concentrations in the feed suggests that cesium and strontium may occupy different sites in the rich clay product.

Heat of Reaction of Clays with Synthetic Waste Solutions

G. S. Barney

Heats of reaction were measured at 70°C for reactions of synthetic waste with calcined and uncalcined kaolin and bentonite clays to form cancrinite. These data are compared in Figure 1.

Values of heat evolved for the uncalcined clays were significantly lower (kaolin = 49.5 ± 3.6 cal/g, bentonite = 46.7 ± 1.3 cal/g) than for the same clays after calcination (kaolin = 190.7 ± 3.5 cal/g, bentonite = 105.9 ± 3.2 cal/g). Measurements of several commercially available calcined kaolins yielded evolved heat values of 128.7 ± 2.5 cal/g for Satintone® (Engelhard Minerals and Chemicals Corporation) and 159 ± 15 cal/g for PAB® (J. R. Simplot). These results show that the source of the clays is significant. The estimated temperature rise of mixing one gram of clay with one milliliter of waste ranges from 38° to 41°C for uncalcined clays and from 87° to 156°C for calcined clays (assuming adiabatic conditions).

The calculations show that heat must be removed from reacting calcined clay-waste mixtures in order to prevent vaporization of the liquid phase.

FIGURE 1
HEAT EVOLVED (-Q) VERSUS TIME FOR
CLAY-WASTE REACTIONS AT 70° C

Clay Calcination Process - Intermediate Temperature Reactions

G. S. Barney

The thermal decomposition of NaNO_3 to Na_2O and NO_x is an essential reaction in the Clay Calcination process for waste immobilization. The minimum temperature necessary for this process has previously been assumed to be the decomposition temperature of NaNO_3 . Prior work on this process involved firing a cancrinite product (or directly mixed, solid waste and clay) at 700° to 900° C to form nepheline.

Thermogravimetric measurements on clay- NaNO_3 mixtures were performed in order to determine the temperature ranges of NaNO_3 decomposition. Weight loss curves for kaolin- NaNO_3 mixtures are given in Figure 2. Data from all experiments show that both bentonite and kaolin catalyze the decomposition of NaNO_3 . The decomposition ranges were 460° to 640° C for kaolin- NaNO_3 and 420° to 710° C for bentonite- NaNO_3 , compared with 575° to 900° C for pure NaNO_3 . Similar results were obtained for clay-synthetic sludge mixtures. These data indicate that the Clay Calcination process can be operated at lower temperatures than previously thought, thus reducing (or possibly eliminating) the problem of cesium volatility.

Compilation of Aqueous Silicate Laboratory Studies

C. H. Delegard and G. S. Barney

A document compiling laboratory data accumulated since January 1974 for the Aqueous Silicate process has been prepared and will be issued early next quarter. Topics covered include

- Kinetics of cancrinite formation.
- Heats of reaction.

FIGURE 2
TGA CURVES FOR KAOLIN - NaNO_3 MIXTURES

- Conceptual process flowsheets.
- Leach rate.
- Thermal stability.
- Radiation stability.
- Mechanical strength.
- Density and total volume.

Aqueous Silicate Engineering-Scale Development

C. M. Peabody

Development of the Aqueous Silicate process is at a point where pilot-plant equipment testing is needed. Specifically, in support of the Rich Clay process alternative, the following topics should be investigated:

- Operating conditions for different feed types and clays.
- Binder types and mixing techniques.
- Methods for water removal from product.
- Heat generation.
- Heat conductivity.
- Kinetics of product formation on a large scale.
- Feasibility of remote maintenance.
- Equipment reliability.

During the report period, designs and cost estimates were prepared for a nonradioactive, rich clay alternative pilot-plant. Designed for installation in the third floor of the 202-S Building, the plant would consist of clay and binder hoppers and feeders, a mechanical mixer, and a drying oven. The plant would be supplied with a synthetic feed

makeup system utilizing existing tankage and piping with only minor modifications.

SILICATE MELT PROCESS

The objective of this task is to develop and demonstrate processes for incorporation of Hanford high-level waste salt cake and sludges into immobile silicate-based glasses. Previously, formulations and processes for converting salt cake to soda-lime- and borosilicate-type glasses were devised and successfully demonstrated in batch laboratory-scale tests with both synthetic and actual salt cake; results of these studies were summarized in a previous report.^[3] Present laboratory-scale glass-making activities are (a) development and demonstration of suitable processes for conversion of sludges, free of salt cake, to glasses; and (b) construction of a small-scale (10 to 25 kg of glass per day) all-electric, continuous melter for eventual hot cell use with high-level waste. An equipment development program is also under way with construction and safety analysis of a one-ton-per-day, continuous electromelter to process simulated wastes.

Development of Glass Formulations for Redox and Purex Sludges

C. W. Hobbick and W. W. Schulz

In preparation for subsequent glass-making activities, effort was directed toward determining the composition of water-washed Redox and Purex process waste sludges. Available composition data are compiled in Tables III and IV.

Two separate samples of Redox process sludge from the 108-S, 109-S, 101-SX, 102-SX, and 105-SX Tanks and three samples from the 111-S Tank were obtained. Portions (100 to 150 grams) of each sludge were water-washed three times

TABLE III
PARTIAL COMPOSITION OF WASHED REDOX PROCESS SLUDGE

Inerts ^a wt%	Tank No.												
	108-S	108-S	109-S	109-S	110-S	111-S	111-S	111-S	101-SX	101-SX	102-SX	105-SX	105-SX
Al ₂ O ₃	18.2	12.3	28.9	<10.1	52.1	56.3	21.2	10.7	23.0	31.6	51.8	17.0	66.3
SiO ₂	13.9	8.23	28.4	3.48	3.64	1.81	4.40	7.31	0.82	8.76	8.55	0.62	3.67
Na ₂ O	17.4	20.5	30.7	23.0	2.11	2.55	8.44	1.41	2.04	0.66	5.38	0.61	3.51
Fe ₂ O ₃	0.29	5.06	1.26	5.34	1.27	<1.66	8.72	9.38	0.42	0.52	1.62	0.54	2.94
Cr ₂ O ₃	0.18	2.39	1.21	7.15	0.07	<0.09	18.5	17.3	0.86	3.05	0.25	0.07	0.22
CaO	0.98	0.07	0.84	2.28	0.05	<0.55	2.0	2.19	0.21	0.67	0.53	0.06	0.24
MnO ₂	0.05	0.94	<0.29	<0.82	0.26	<0.68	5.07	4.07	<0.03	0.04	0.29	0.21	1.05
Fission Products													
μCi/gram													
of Washed Sludge													
⁹⁰ Sr	94	927	360	1150	646	10	6710	6000	98	233	2260	1080	490
¹³⁷ Cs	260	127	-	86.3	10	1.8	-	38.8	4.4	272	246	9.3	92.9
¹⁰⁶ Ru- ¹⁰⁶ Rh	- ^c	683	553	1280	-	-	6800	6300	-	-	-	-	-
¹²⁵ Sb	-	22.5	-	-	-	-	-	142	-	-	32.8	7.4	52.1
¹⁵⁴ Eu	-	18.0	-	37.7	-	-	-	106	-	-	10.6	5.8	28.3
¹³⁴ Cs	2.4	30.4	-	-	1.3	-	78.3	-	0.2	0.6	3.2	0.5	3.6
Actinides ^b													
mg/gram													
Pu	0.0233	0.0102	0.111	0.0129	0.0183	0.002	0.0349	0.0605	0.0004	0.0101	0.00584	0.00456	0.0274
U	1.98	45.	37.6	27.8	7.49	0.235	29.4	68.	0.0318	0.0492	0.161	1.0	0.249

^aConcentration of sulfate compounds not yet determined.

^bConcentration of Np, Am, and Cm not yet determined.

^cZero or negligible quantities found.

TABLE IV
PARTIAL COMPOSITION OF WASHED PUREX PROCESS SLUDGE

Inerts wt%	Tank No.		
	104-A	104-AX	102-AX
Fe ₂ O ₃	42.0	42.8	24.2
Al ₂ O ₃	14.3	11.2	17.2
Na ₂ O	13.8	8.44	20.6
SiO ₂	7.78	10.4	5.47
MnO ₂	(12.0)	0.84	2.39
CaO	1.61	3.92	1.13
Cr ₂ O ₃	0.50	0.54	0.80
<u>Fission Products</u>			
<u>μCi/gram</u>			
<u>of Washed Sludge</u>			
⁹⁰ Sr	4.36 x 10 ⁴	8.60 x 10 ⁴	8.19 x 10 ³
¹³⁷ Cs	1.17 x 10 ³	4.09 x 10 ⁴	1.40 x 10 ³
¹²⁵ Sb	3.81 x 10 ²	1.73 x 10 ³	9.10 x 10 ²
¹⁵⁴ Eu	1.23 x 10 ²	-	3.40 x 10 ²
¹³⁴ Cs	41	-	40
<u>Actinides^a</u>			
<u>mg/gram</u>			
Pu	0.162	0.199	0.0712
U	3.0	12.3	0.127

^aConcentration of Np, Am, and Cm not determined.

using 1.5 ml of water per gram of wet sludge, dried for ~12 hr at 200° C, and weighed. The solids were then fused with KOH, and the resulting melt dissolved in ~1M HCl. In Tables III and IV components of the washed and dried sludge are expressed as anhydrous oxides even though it is recognized that not all are present in this form.

As noted in Table III, the composition of Redox process sludge varied widely from tank to tank and even within a given tank. The major component is aluminum; some of the sludges contain substantial concentrations of silicon and sodium. As expected, ⁹⁰Sr is the principal fission product

present in the washed sludge. All the washed Redox sludges contained appreciable amounts of plutonium. Analyses to determine other components of the sludge, particularly sulfate, are pending.

It is currently believed that two types of water-insoluble solids are present in tanks containing Redox wastes. The major component, hydrated Al_2O_3 , precipitated as the neutralized liquid aged in the tank. The remainder is chiefly hydrated heavy metal oxides and hydroxides [e.g., $\text{Fe}(\text{OH})_3$] which precipitated when the Redox waste was made alkaline. Current solids in the underground tanks are an intimate mixture, in various proportions, of both types of solids.

Results in Table IV were obtained with Purex process sludge which had been water-washed only once at a ratio of 10 mg water per gram of as-received, wet sludge. Additional washing very likely would have reduced the sodium and cesium content. The principal metallic components of washed and dried Hanford Purex process sludge are Fe, Al, Na, Si, Mn, and Ca. With the exception of Si, these are also the principal metallic components of at least one batch of washed and dried Savannah River Purex process sludge. Significantly, also, both the Hanford and Savannah River sludges contain about the same amounts of Fe, Al, and Ca; Hanford sludge contains considerably more Si than the Savannah River material while the latter contains more Mn.

The data in Tables III and IV have been used to select compositions of synthetic Redox and Purex sludges for use in extensive glass-making screening tests to be conducted next quarter. Plans are to refine and extend these compositions as more sludges are characterized.

Glass Melter Pilot-Plant Construction - D. J. Bradley and
H. A. Irby, Battelle Pacific Northwest Laboratories

This task is intended to demonstrate the feasibility of the Silicate Melt process on a pilot-plant scale, develop suitable process equipment and systems, and obtain scale-up and design data. Construction of a one-ton-per-day glass melter at Battelle-Northwest's 314 Building is progressing toward the goal of an October 1975 startup.

The steel support structure for the melter has been completed. All refractories needed for completion of the melter have been received, and diamond saw and diamond core drilling apparatus necessary for refractory installation have been set up in the 314 Building. At mid-June the melter bottom refractories were being assembled in place.

To gain information on the corrosive nature of simulated Hanford waste glasses on refractories, the Carborundum Company agreed to conduct a test on corrosion between simulated salt cake waste-glass and six of their glass-contact refractories. The composition of the simulated waste glass mix sent to Mr. R. Minor of the Carborundum Company, Refractories and Insulation Division, Niagara Falls, New York, is given in Table V.

The six refractories were arranged to form a small tank into which 21 pounds of simulated waste glass mix were placed, heated to 1350° C, and held overnight. By morning the mixture was thoroughly melted, transparent, viscous, light-yellow in color, and no fuming was noted. The temperature was then raised to 1400° C and held for 96 hours, after which the glass was drained and the tank slowly cooled. The bricks were then removed and the depth of the metal line cut, as well as the volume lost, were measured. This procedure is essentially that described in American

TABLE V
SIMULATED HANFORD WASTE-GLASS FORMULATION

<u>Feed Formulation</u>		<u>Approximate Glass Composition</u>		
<u>Component</u>	<u>wt%</u>	<u>Component</u>	<u>wt%</u>	<u>mol%</u>
SiO ₂	39.9	SiO ₂	52.0	53.6
CaO	10.0	Na ₂ O	27.9	27.9
NaNO ₃	20.6	CaO	13.0	14.4
NaNO ₂	10.9	Al ₂ O ₃	6.5	3.9
NaAlO ₂	8.0	P ₂ O ₅	0.6	0.2
Na ₂ CO ₃	7.5		100.0	100.0
NaOH	1.0			
Na ₂ SO ₄	0.9			
NaH ₂ PO ₄	0.7			
NaCl	0.5			
	100.0			

Society for Testing and Materials (ASTM) Method C622-73. The results of this test (presented in Figure 3) show that Monofrax K-3, which was previously chosen as the glass contact refractory for the pilot melter, is the best refractory for this application.

Safety Review of the Silicate Melt Process - P. J. Pelto and J. D. Kaser, Battelle Pacific Northwest Laboratories

A conceptual facility description, a preliminary hazards analysis, and a fault tree diagram were prepared for the Silicate Melt process. The primary emphasis was on the high-temperature melting step.

As currently defined, the fault tree contains 132 "bottom" or initiating events. Work remaining on the fault tree analysis includes: (1) applying best estimates of probabilities and release fractions to the fault tree initiating events, (2) using appropriate computer codes to generate the cutsets (accident sequences) and rank them on the basis of risk, and (3) analyzing the high risk cutsets in more detail.

FIGURE 3
CORROSION TEST RESULTS

Even though the Silicate Melt process is only at the conceptual step, it is felt that the above approach will systematically point out the areas upon which to concentrate safety design efforts.

OTHER IMMOBILIZATION CONCEPTS

The objective of this task is to identify, investigate, and evaluate alternative processes (other than the Aqueous Silicate or Silicate Melt processes) for conversion of Hanford high-level waste to immobile forms suitable for long-term storage.

Laboratory Plans - M. J. Kupfer

The primary immobilization techniques to be investigated in the laboratory include, but are not limited to:

- Concrete with and without addition of monomers.
- Pressed concrete (at hot and ambient temperatures).
- Low-melting glass frits and glazes.
- Glass sintering.

Du Pont Savannah River Laboratory, Holifield National Laboratory, and Battelle Pacific Northwest Laboratories are presently investigating similar immobilization processes. At ARHCO emphasis will be placed on the immobilization of Hanford sludges and concentrated radionuclide fractions resulting from future Radionuclide Removal operations.

Additional work will be conducted on the immobilization of low-level solutions resulting from the Radionuclide Removal process.

Engineering Evaluation - C. M. Peabody

A literature survey of various radioactive waste immobilization and terminal storage methods is under way to determine their applicability to Hanford waste forms and volumes.

Immobilization methods under consideration include:

- Bituminization.
- Fixation in concrete or grout.
- Fixation in polyethylene.
- Glass processes (phosphate, borosilicate, aluminosilicate).
- Clay processes (high- and low-temperature, rich and lean mixtures).

Evaluation of the different processes will be conducted using the following criteria:

- Compatibility with ARHCO wastes.
- Volume changes.
- Radiolytic and thermal stability of product.
- Permissible heat load.
- Leach rates of product.
- Physical characteristics of product.
- Interim and/or terminal storage requirements.
- Economics.

RADIONUCLIDE REMOVAL STUDIES

These studies are concerned with devising, developing, and demonstrating processes for removal of all long-lived ($t_{1/2} > 25$ yr) radionuclides from Hanford terminal liquors and salt cake solution. The overall objective is to develop technology for partitioning such high-level wastes into a small volume of highly active waste requiring further immobilization for long-term storage and a large volume of chemical or very low-level waste which can be safely stored or disposed of by inexpensive means.

Current laboratory studies are focused on:

1. Screening and testing, with synthetic waste solutions containing appropriate radiotracers, various sorption and ion exchange processes and procedures for removal of ^{137}Cs , ^{90}Sr , and ^{99}Tc .
2. Testing radionuclide removal procedures with actual waste solutions as the latter become available.

Distribution Data for Various Cs, Sr, and Tc Sorbents

W. W. Schulz

In recent testing Duolite ARC 359, a phenolic-type cation exchange resin made by Diamond Shamrock Company, has been used exclusively for removal of ^{137}Cs from the various alkaline wastes. Technetium-99 removal is being accomplished using strong-base anion exchange resins such as Dow Chemical Company Dowex 1-X4 and 21K resins. Various sorbents are under consideration for removal of ^{90}Sr ; these include Duolite ARC 359 and Chelex[®]-100 ion exchange resins and $\text{NaTi}_2\text{O}_5\text{H}$, an inorganic sorbent. Chelex-100 resin, a chelating-type resin available from Bio-Rad Laboratories, has been reported by Savannah River personnel to sorb

strontium very effectively from alkaline solutions. The $\text{NaTi}_2\text{O}_5\text{H}$ sorbent (Sandia "Titanate") was obtained from R. Dosch of Sandia Laboratories, Albuquerque, New Mexico. Precipitation of inert strontium phosphate is also being studied as a means of removing ^{90}Sr from alkaline waste solutions. Hanford waste solutions for radionuclide removal operations will likely contain, in addition to ^{137}Cs , ^{99}Tc , and ^{90}Sr , small concentrations of other long-lived radionuclides (e.g., actinides). It is anticipated that procedures used to remove ^{137}Cs and, particularly, ^{90}Sr will also remove these other radionuclides.

Much of the experimental work completed this quarter was directed toward obtaining equilibrium data for distribution of ^{137}Cs , ^{85}Sr , and ^{99}Tc between aqueous NaNO_3 - NaOH solutions and various sorbents (Figures 4 and 5). Sorption of cesium and technetium on Duolite ARC 359 and Dowex 1-X4 resins, respectively, increases with increasing NaOH concentration while that of strontium by any of the three sorbents tested decreases (Figure 4). At constant NaOH concentration sorption of all three radionuclides decreases with increasing total sodium concentration (Figure 5). The data in both Figures 4 and 5 show that the Sandia "Titanate" exchanger has a very high affinity for ^{85}Sr in synthetic alkaline waste solutions.

FIGURE 4

SORPTION OF RADIONUCLIDES FROM NaNO_3 -NaOH SOLUTIONS:
EFFECT OF NaOH CONCENTRATION

FIGURE 5

SORPTION OF RADIONUCLIDES FROM NaNO_3 -NaOH SOLUTIONS:
EFFECT OF TOTAL SODIUM CONCENTRATION

Batch and Column Experiments with Actual Waste Solutions

W. W. Schulz

In tests with actual waste solutions the ^{137}Cs concentration of near-terminal liquor (3.6M NaOH-1.8M NaAlO_2 -2.7M NaNO_3 -1.7M NaNO_2) from the 110-S Tank was reduced from 5.2×10^5 $\mu\text{Ci/liter}$ to 0.14 $\mu\text{Ci/liter}$ by passage (25° C, 1 column volume/hour) through a series of five beds of 20-50 mesh, Na-form Duolite ARC 359 resin. The effluent from the final Duolite Column, after removal of small portions for the ^{99}Tc distribution ratio tests described subsequently, was passed (25° C; 1 column volume/hr) through a 10-ml bed of Sandia "Titanate" resin. A total of 31 column volumes of solution was passed through the bed of $\text{NaTi}_2\text{O}_5\text{H}$. The titanate resin removed 90 to 98% of the influent ^{90}Sr content. This amount of strontium removal is considerably less than predicted from distribution data obtained (Figures 4 and 5) with synthetic waste solutions.

Technetium-99 distribution ratios of 350 and 310, respectively, were measured when portions of the effluent from the Duolite ARC 359 resin were contacted with 50 to 100 mesh, NO_3 -form Dowex 1-X4 and 21K resins. In comparison, a ^{99}Tc distribution ratio of 290 was measured when a synthetic terminal liquor (4.0M NaOH-1.0M NaAlO_2 -2.7M NaNO_3 -1.7M NaNO_2) spiked with ^{99}Tc was contacted with Dowex 1-X4 resin.

In the next report period both batch and column experiments will be performed to firm up flowsheet conditions for ion exchange removal of ^{137}Cs and ^{99}Tc . Batch sorption and precipitation tests will be made to establish behavior of ^{90}Sr in actual waste solutions and to devise processes or combinations of processes for its satisfactory removal. Initial experiments to determine behavior of Am, Pu, and certain other long-lived radionuclides in these ion exchange

flowsheets will be performed.

ECONOMIC EVALUATION OF IMMOBILIZATION/STORAGE ALTERNATIVES

J. R. Irish

Selection of a waste immobilization and storage concept from the numerous combinations of alternatives is expected to be heavily dependent upon cost considerations. To gain insight into the cost impact problem, ARHCO personnel developed conceptual requirements and preliminary flowsheets for several alternative waste stabilization strategies.^[2] This information was transmitted to the Norman Engineering Company (NECO) of Los Angeles, California, which was retained to expand the material into scope designs and to estimate the capital requirements for each of the alternatives.

Draft reports detailing conceptual facility designs for three alternative waste immobilization schemes and one near-surface engineered storage mode were completed by NECO and reviewed by ARHCO personnel. Cost information based upon these designs is presently being compiled by NECO.

Continued work by NECO on this study in fiscal year 1976 has been redirected to place emphasis on the investigation/evaluation of alternative waste storage configurations. Final reports are expected at the end of September, 1975.

CONTAMINATED EQUIPMENT VOLUME REDUCTION

The objective of this program is to develop and demonstrate methods for reducing the volume of radioactively contaminated metallic items to a size and form suitable for interim or terminal storage. Although the problem of

disposition of contaminated equipment exists throughout the nuclear industry, this program is focusing its attention on the contaminated decommissioned equipment presently located in the 200 Areas at Hanford.

DECONTAMINATION EVALUATION - G. A. Beitel

A Purex dissolver has been selected to be decontaminated at the T Plant decontamination facility for the purpose of developing quantitative cost data for decontamination. Cost data will include manpower costs, chemical costs, facility costs, and liquid waste generation. Final arrangements for the decontamination test were made this quarter. The largest obstacle was the design and fabrication of a dunnage to allow the vessel to be transported to the decontamination facility. The test will be completed during the coming quarter.

SIZE-REDUCTION EVALUATION - G. A. Beitel

The size-reduction scheme based upon a sweeping electric arc (carbon electrode) described in the last quarterly report^[2] has been shelved following detailed discussions with electric furnace design engineers. The two major obstacles were the presence of nonconductors in the equipment to be size-reduced, and the problem of breaking electrodes if a heavy section of a vessel should fall against them.

The discovery of high-powered (megawatt) plasma arc furnace torches has given new impetus to the size-reduction task. These torches, which operate at several megawatts with flame lengths of 15 to 150 cm and temperatures above 6000° K, only consume about 100 liters/sec of gas which can be argon, nitrogen, carbon dioxide, or ambient air. These

torches are relatively simple in design and appear readily adaptable to remote control.

Proposals for performing a demolition test on a 2.5-m long by 1.2-m diameter stainless steel vessel have been requested. The results of this test, if successful, will be used for purchase specifications. The demolition test will be started during the coming quarter and completed during November 1975.

A preliminary Equipment Reduction Facility study report was received from Norman Engineering Company. The report was reviewed by ARHCO personnel and comments forwarded to NECO for incorporation in the final report, which is to be completed by September 30, 1975.

MELTDOWN EVALUATION - G. A. Beitel

A single-step size-reduction and meltdown method utilizing a modified Thermite process was proposed earlier. A field test to demonstrate the feasibility of using powdered silicon and gaseous oxygen was planned. The test was to be conducted in a basalt hole 1.8 m deep and 0.6 m in diameter. The parameters to be studied were (a) experimental technical problems, (b) slag-metal behavior, (c) compatibility of molten steel with a natural basalt formation, (d) heat loss to the basalt, (e) fume generation, and (f) physical properties of the final ingot.

Tests using powdered silicon and gaseous oxygen were totally unsuccessful. The inability to add powdered silicon separately and independently of the oxygen terminated the dual delivery system. When powdered silicon was added in bulk, a slag crust prevented mixing with the molten steel. The failure of this system resulted in a redirection using a solid oxidant-- NaNO_3 ; in this manner, experimentally, mixing

could occur prior to the high-temperature reaction. On a long-term basis, if an NaNO_3 -based reaction were to work, it could be a convenient use for the high-level salt cake, which is mostly NaNO_3 .

Powdered silicon and NaNO_3 reactions had previously been used to ignite the silicon-gaseous oxygen reaction and batches up to 36 kg had burned successfully and safely. A 450-kg mixture was placed in the basalt hole along with approximately 450 kg of steel, and ignited. The result was an explosion (no injuries). Since any workable system to melt down 20- to 40-ton vessels would have had to consume multi-thousand-kilogram batches of the silicon- NaNO_3 fuel, there was no point in pursuing this approach any further.

Now that the plasma arc torch size-reduction scheme appears practical, there is no major drawback to a more conventional meltdown system--preferably electric arc or vacuum induction furnace. This is now the direction of the program. During the coming quarter effort will be made to locate commercial furnace facilities in which to conduct experimental scrap melts.

In hot laboratory tests approximately 350 grams of contaminated (thought to be largely ^{90}Sr) stainless steel pipe contained in a clay-graphite crucible were inductively melted in air at 1500°C . Radionuclide content of the resulting metal ingot has not yet been precisely determined. Qualitatively, however, radiation survey instrument readings of the ingot and the crucible indicate such melting reduced the radionuclide content of the metal by about a factor of 10. These hot laboratory tests will be expanded and continued through fiscal year 1976.

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