NUCLEAR SAFETY QUARTERLY REPORT
MAY, JUNE, JULY, 1968
FOR
NUCLEAR SAFETY BRANCH
OF
USAEC DIVISION OF REACTOR
DEVELOPMENT AND TECHNOLOGY

OCTOBER 1968

**AEC RESEARCH &
DEVELOPMENT REPORT**

BATTELLE

NORTHWEST

BATTELLE MEMORIAL INSTITUTE

PACIFIC NORTHWEST LABORATORY

BATTELLE BOULEVARD, P.O. BOX 509, RICHLAND, WASHINGTON 99352

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

operated by

BATTELLE MEMORIAL INSTITUTE

for the

UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

PRINTED BY/ FOR THE U. S. ATOMIC ENERGY COMMISSION

3 3679 00061 1816

BNWL-894

UC-41, Health and Safety
~~Special Distribution~~

NUCLEAR SAFETY QUARTERLY REPORT
MAY, JUNE, JULY, 1968
FOR
NUCLEAR SAFETY BRANCH
OF
USAEC DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY

By

The Staff of Battelle-Northwest

FIRST UNRESTRICTED
DISTRIBUTION MADE

OCT 25 '68

October 1968

PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON

Printed in the United States of America
Available from
Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia 22151
Price: Printed Copy \$3.00; Microfiche \$0.65

NUCLEAR SAFETY QUARTERLY REPORT
MAY, JUNE, JULY, 1968
FOR
NUCLEAR SAFETY BRANCH
OF
USAEC DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY
By
The Staff of Battelle-Northwest

FOREWORD

This report is the sixth of a series in which Pacific Northwest Laboratory reports its nuclear safety-related studies being performed for the USAEC Division of Reactor Development and Technology.

PREVIOUS REPORTS IN THIS SERIES

BNWL-433	January, February, March	1967
BNWL-537	April, May, June	1967
BNWL-754	July, August, September, October	1967
BNWL-816	November, December, 1967, January 1968	
BNWL-885	February, March, April	1968

TABLE OF CONTENTS

FOREWORD.	iii
SUMMARY	1.1
ENGINEERED SAFEGUARD STUDIES.	2.1
Containment Systems Experiment - G. J. Rogers	2.1
Fission Product Aerosol Control - L. C. Schwendiman and H. H. Van Tuyl.	2.23
PRESSURE BEARING COMPONENT EVALUATION AND MONITORING STUDIES	3.1
Crack Detection in Pressure Piping by Acoustic Emission - P. H. Hutton, R. N. Ord, C. E. Fitch H. N. Pedersen, and J. C. Spanner	3.1
ENVIRONMENTAL STUDIES - R. F. Foster.	4.1
Disposal of Reactor Off-Gas into Soil Systems - L. G. King.	4.1
Simulation Modeling of Thermal Generation in Selected River Systems - R. T. Jaske	4.2
FIXATION OF RADIOACTIVE RESIDUES.	5.1
Phosphate Glass Solidification.	5.1
Spray Solidification	5.5
Advanced Solidification Techniques.	5.12
Development Studies on Solidification Processes	5.15
Heat Transfer Studies	5.20
Waste from Chemical Processing	5.23
Low-Level Waste Treatment	5.29
Evaluation of Solidified Waste Products	5.30

1. SUMMARY

ENGINEERED SAFEGUARD STUDIES

Containment Systems Experiment

A series of leakage rate tests were performed on the CSE containment vessel using room-temperature air, heated-air, and steam-air atmospheres. The tests had the objective of relating the leakage characteristics of a containment system under possible postaccident conditions to the leakage rate determined by standard methods as required by Technical Specifications or operating licenses for nuclear stations.

Base-line tests were performed with room-temperature air at nominal pressure levels of 9, 23, and 35 psig. These tests were followed with heated-air and steam-air atmosphere, leakage-rate tests at nominal test conditions of 9 psig at 175 °F, 23 psig at 225 °F, and 35 psig at 250 °F.

Results of the base-line tests showed good agreement between integrated leakage rates determined by the absolute and reference vessel methods; and the summation of collected leak flows at individual leak points amounting to 85 to 98% of the calculated leakage. Assuming laminar flow, variation of leakage rate with pressure was as predicted.

In both the heated-air and steam-air test series, a substantial decrease in leakage occurred as temperature increased. Additional, deliberately superimposed, leak points were introduced to provide reasonable tests of measurement sensitivity.

A large-scale fission-product transport experiment was performed at the quarter's end. Test conditions were a steam-air containment atmosphere at 250 °F and 48.5 psia, and airborne aerosol removal was by natural processes only. Test results are not yet available.

A mathematical model for removal of elemental iodine by natural processes in a containment vessel was improved by incorporating the effect of the density gradient (resulting from molecular weight gradients) near the wall. Predictions with the model were compared with experimental results from CSE experiments in two, different size vessels, with reasonable agreement found.

Renovation of the CSE blowdown facility was nearing completion, and instrumentation is now being installed on the reactor simulator vessel.

Data from CSE blowdown experiments show significant deviations from predictions of existing mathematical models for foam level and effluent void fraction. A different theoretical approach was developed to predict the void fraction profile in the vessel.

Optional, digital filtering techniques were developed and incorporated into the data processing codes for treating CSE blowdown data. A program for computing time derivatives of the vessel weight-load cell data was developed to eliminate the tedious manual graphical procedure for determining effluent mass flow during blowdown. A computer routine is being programmed to process neutron densitometer data to give effluent void fraction without laborious manual calculation.

Core samples were taken from the CSE reactor simulator vessel wall in and near a large, ultrasonic test-defect indication. To date, results from examination of these core samples show that the sample from just outside the defect-indication region is free from defects and has normal tensile properties. The two, core samples from the defect indication region show numerous small void- or inclusion-defects elongated in the rolling direction at a location agreeing with the ultrasonic test

indications. Tensile tests made on the core sample specimens show that the material around the flaws behaved in a ductile manner and exhibited reasonable strength.

Fission Product Aerosol Control

Absorption of methyl iodide by sprays of aqueous solutions of hydrazine and sodium thiosulfate was measured in a series of once-through spray runs. Separate collection of wall-film liquid and use of a specially designed drop-collector permitted comparative assessment of absorption by wall-film liquid and falling drops. The measured absorption was 1.5 to 3 times faster than predicted from stagnant film and drop theory.

The effect of chamber size on washout rate was estimated based on the washout model. For one set of conditions, nominally typical of those expected for a large scale containment system, the washout half-time using 5 wt% hydrazine was predicted to be about 60 min. This estimate utilized extrapolation from the experimental results using the droplet-wall removal model.

Further reaction kinetics data were obtained for the liquid systems of methyl iodide and 1,1-dimethylhydrazine, monomethylhydrazine, sodium sulfite, and sodium bisulfite. Of these, sodium sulfite gave the most rapid rate.

A conclusion was reached that the radiation sensitivity of hydrazine and the formation of hydrogen as a product will be serious deterrents to its use in reactor sprays, unless selective timing of reactive sprays can be realized following the loss of flow accident, or satisfactory means to mitigate hydrogen explosive potential can be incorporated into the reactor design.

PRESSURE BEARING COMPONENT EVALUATION AND MONITORING STUDIESCrack Detection in Pressure piping by Acoustic Emission

Detailed study of acoustic emission wave propagation in reactor materials has continued. Emphasis is being placed on study in the frequency range of 0.5 to 3.0 MHz and on the use of critical angle transducer orientation. Some success has been achieved in detecting signal variations between brittle and ductile material. There are also indications that what was earlier thought to be a variation in signal frequency content, as observed by lower frequency sensors, is actually a reflection of acoustic emission repetition rate or signal density for increments of deformation and crack growth.

It appears that critical angle detection in the higher frequency range (low MHz) will improve the signal-to-noise ratio and provide a sharper wave front for improved source location resolution. One MHz, PZT-5A transducers have been fabricated by using a ceramic mounting and backing material in place of the previously used epoxy to improve their high temperature longevity. These are currently in the testing process

A low-noise, wide-band preamplifier has been fabricated and satisfactorily field tested for acoustic emission detection. The performance is superior to previously used commercial preamplifiers, and the cost is about one third that of the commercial models.

Background-noise spectra measurements have been made on the Dresden I nuclear power reactor during start-up from cold flow to full power operation. For this reactor, it appears that an acoustic emission monitor system operating in the range of 750 kHz to 3 MHz would be relatively unhampered by reactor system noise interference.

FIXATION OF RADIOACTIVE RESIDUES

During the report period, approximately 2,400,000 Ci of radioactivity were processed in the sixth phosphate glass solidification run, and approximately 360,000 Ci were processed in the third spray solidification run. Phosphate glass Run PG-6 was made by using sulfate-free Purex Waste 1 (PW-1) feed while SS-3 was made with sulfate-containing Purex Waste 2 (PW-2) feed.

The high rare-earth concentration (35 wt% as oxide) of PW-4m waste tends to raise the melting point and increase nonhomogeneity of spray solidifier melts. Satisfactory phosphate melts for the spray solidifier have only been achieved by adding sodium and iron to the point where the final melt resembled PW-1 spray solidifier product. Some borosilicate systems have been investigated, of which several have been prepared that have "stick" temperatures above 700 °C and melting points in the range 925 to 975 °C.

The water-nitric acid separation characteristics of an analog computer model of WSEP were verified by comparison with actual WSEP data. Data from a pot calciner run have also been used to adjust the parameters of the model to match data on solids entrainment and ruthenium volatility.

Three, successful vacuum batch transfers of phosphate glass melts were made in engineering scale equipment to test advanced solidification techniques.

The vacuum distillation flowsheet for treating sulfate condensate resulting from phosphate glass solidification was tested with a simulated solution and tracer ruthenium. The overall decontamination factor was greater than 5×10^3 .

Steady-state cooling capacities of the two, WSEP product pot furnaces were calculated. The highest planned heat generation rate of 25 kW in a stainless steel pot is within the calculated furnace limits.

2. ENGINEERED SAFEGUARD STUDIES

CONTAINMENT SYSTEMS EXPERIMENT - G. J. Rogers

Containment Leakage Studies - M. E. Witherspoon

A series of ambient air, hot air, and steam-air leakage tests were completed late in May. The objective of these tests is to relate the leakage of steel containment vessels as measured in a standard air test to the leakage of the same vessel in postaccident atmospheres. The results and subsequent analyses are important as a means of assessing the adequacy of current requirements for leak tightness as stipulated by the Division of Safety Standards. These tests supplement other leakage tests performed on the same vessel and summarized in previous reports.^(1,2)

Base-line air-leakage rate tests at three nominal pressure levels (9, 23, and 35 psig) were conducted to establish the nominal leakage rate of the vessel and the loss at each individual leakage point. The vessel was then retested with heated-air and steam-air atmospheres at the same nominal pressures and three nominal temperatures (175, 225 and 250 °F).

Calibrated thermocouples, a nickel-alloy resistance wire, a helium-tested reference tubing system, lithium chloride humidity sensors, and a heat exchanger and blower were within the contained volume as shown in Figures 2.1 and 2.2.

Base Line Air Tests

The base-line air tests were performed with the results listed in Table 2.1 (Test No. 1 to 3). The reference method value is the average of 24 single hourly observations. The absolute method is calculated with a perfect gas formula for conditions at the beginning and end of the 24 hour period, and it is also calculated by a linear regression analysis of

FIGURE 2.1. *Equipment for Hot-Air and Steam-Air
Leakage Rate Tests*

*All Points of Contact Insulated
from Containment Shell

FIGURE 2.2. Resistance Wire Routing in
Containment Vessel

TABLE 2.1. Summary of Data; Hot-Air and Steam-Air Leakage Rate Series

Test No.	Nominal Conditions °F, psig	Reference Method	Absolute Method (Wire)	Collected Leakage	Comments
1	75, 35	0.408	0.477 (0.471 ± 0.0098)	0.402	0.477 = base for downward extrapolation
2	75, 23	0.393	0.394 (0.417 ± 0.0156)	0.349	0.415 = laminar extrapolation from 35 psig
3	75, 9	0.259	0.252 (0.226 ± 0.025)	0.228	0.258 = laminar extrapolation from 35 psig
4	175, 9	--- (a)	0.120 (0.108 ± 0.0176)	0.09	Leakage rates at individual points diminished during heat-up
5c	225, 23	0.220	0.269 (0.268 ± 0.01)	0.15	Superimposed additional 0.1% leakage
6	250, 35	0.105	0.124 (0.147 ± 0.012)	0.10	Superimposed leak diminished
1a	75, 35	0.266	0.289 (0.301 ± 0.033)	0.182	Includes original and superimposed leakage after cool-down
7	175, 9 ^(a) (S/A)	---	(-) 1.20	nil	
8a	225, 23 (S/A)	---	(-) 1.39	0.285	Superimposed add. leak
8b	225, 23 (S/A)	---	(-) 1.90	0.05	Superimposed leak diminished
9	250, 35 (S/A)	---	(-) 9.6	0.48	Superimposed ~0.5% additional leakage
10b	75, 35	0.47	0.532	0.255	Ambient air retest

(a) Unstable control conditions put reference system temporarily out of order

all 24 hourly observations. The value in parentheses is the regression value with a 95% confidence interval (2 sigma).

The collected leakage column is the result of measuring individual leakage points with flow meters, summing them, and converting the sum to an equivalent percentage loss per day.

Agreement between the methods was considered reasonable and gave confidence that the instrumentation was performing satisfactorily. The results of a laminar extrapolation of the data at 35 psig to lower pressures agree well with calculated results. Comparison of collected leakage to calculated values of leakage indicate that 85 to 98% of the total leakage was collected.

Hot Air Tests

Test Number 4 shows that heating of the vessel to 175 °F at 9 psig had the immediate effect of reducing the collected leakage from ~0.23%/day to 0.09%/day. A corresponding drop is seen in the calculated leakage. The leakage at original leakage points continued to diminish in Test Numbers 5 and 6 (225 and 250 °F, respectively). The collected leakage in Test Number 5 includes additional, deliberately superimposed leakage points; these leaks also diminished during the 24 hr period of the Number 6 test. A retest of the vessel after a cooldown (Test No. 1a) indicated that most of the leaks had not recovered their original leakage rate, and total leakage (including the superimposed leakage) was less than that at the beginning (Test No. 1).

Steam-Air Tests

Additional testing in steam-air atmospheres (Test No. 7, 8a, 8b, and 9) showed a further flow decrease at the superimposed leakage points. The atmosphere for these tests was

established by beginning with one atmosphere of ambient air in the vessel, and adding steam until the desired condition was reached. Once the nominal test condition was attained, steam feed was halted and the vessel began a pressure decline down the saturation pressure curve. The method for calculation of the air fraction lost is the perfect gas law applied to the partial pressure of air. The ratios of steam to air are determined by letting the measured average temperature define the saturation pressure and by measuring the total pressure with a gage. Test Number 8 was an isothermal test with the same calculational procedure applied.

The collected leakage volume is the fractional percent of initial air contents leaking at the beginning of each test. These leaks were successfully collected by using flowmeters with condensate traps. Generally, all the individual leaks diminished as exposure time to the steam-air atmosphere increased.

The negative values for the steam-air calculations can be partially attributed to several unapplied corrections; accumulation of condensate resulting in a volume decrease, shrinkage of the vessel due to temperature decline (as much as 85 °F in Test No. 9), shrinkage due to pressure drop (27 psi in Test No. 9), and an unknown measurement error caused by a time lag in the thermocouple system. Preliminary analysis indicates that the latter error is the most influential. The Test 10b was a cold air retest of the vessel with all superimposed and original leaks included.

The trend of these data indicates that leakage rate effects caused by actual atmospheric weight and temperature are swamped by physical alterations of leakage paths because of the temperature and composition of the atmosphere. The results suggest that relatively small corrections and

adjustments to aggregate leakage levels to account for changes in molecular weight and viscosity may be an unnecessary refinement. There are more gross effects which give conservatism to the estimate of leakage based on simple ambient air test with no corrections for atmosphere. Integrated leakage tests on steam-and air-atmospheres were unsuccessful because an accurate temperature measurement during a decline is difficult to attain. Further analysis and testing are needed to determine whether it is possible to get this measurement at all.

Fission Product Transport Studies

Large-Scale Experiments - R. K. Hilliard

One large-scale fission product transport test was made near the end of the quarter. In this experiment, CSE Run A-5, no active safeguards were used, and fission product transport within the containment vessel was due to natural processes only.

The primary objectives of this experiment were to obtain experimental data on FP behavior in the containment vessel in an environment similar to those expected after a Loss-of-Coolant Accident to a large PWR, and to compare the observed behavior with predictions based on mathematical models. To achieve the latter objective, several secondary objectives were included:

- Rate of mixing of FP simulant materials in the atmosphere
- Characterization of the fluid dynamics (temperature and velocity gradients, steam-air ratios, and steam condensation rates)
- Measurement of FP aerosol properties in the containment atmosphere
- Determination of the FP leakage rate through selected leak points
- Visual observations of the atmosphere during the test.

Test conditions were similar to those reported⁽³⁾ for an earlier test, Run A-2, except that the vessel was insulated, and the atmosphere was 250 °F, 48.5 psia in Run A-5. Samples are being analyzed and no conclusions can be made at this time except that all test objectives appeared to be accomplished.

The next large-scale experiment will be a containment spray test that will demonstrate the efficiency of sprays for removing fission products from steam-air atmospheres.

Mass Transfer Model for Natural Transport of Elemental Iodine - R. K. Hilliard and J. G. Knudsen

The mathematical model discussed in a previous report⁽⁴⁾ was improved by accounting for the effect on natural convection of the density gradient near walls caused by molecular weight gradients. The theoretical basis for this modification can be found in Jacob.⁽⁵⁾ The model provides a method for predicting rates of transfer of gaseous fission products from steam-air atmospheres to containment vessel surfaces. Its advantage over existing models^(6,7,8) is its ability to predict the gas phase mass transfer coefficient.

Assumptions, in addition to those used in earlier models, are: (1) flow patterns are caused only by natural convection due to heat transfer to vertical walls, and (2) the steam-air mixture is saturated at all locations. The simplification is made that the walls are at thermal equilibrium, though the model can be extended to cover the unsteady state.

The correlation recommended by McAdams⁽⁹⁾ for turbulent heat transfer by natural convection, modified for density gradients caused by mass transfer of steam, is

$$h_i = 0.13 k_f \left(\frac{g}{v^2} \text{Pr Sc} \right)^{1/3} \left(1 - \frac{M_b T_i}{M_i T_b} \right)^{1/3} \quad (1)$$

By analogy,

$$k_c = 0.13 D_v \left(\frac{g}{v^2} \text{Pr Sc} \right)^{1/3} \left(1 - \frac{M_b T_i}{M_i T_b} \right)^{1/3} \quad (2)$$

All the terms are known except T_i , the temperature at the interface. T_i can be estimated by an energy balance:

$$\begin{aligned} q/A &= h_i (T_b - T_i) + \frac{k_c}{RT_b} \Delta H_{\text{vap}} (p_b - p_i) \\ &= h_w (T_i - T_o) = h_o (T_o - T_a) \end{aligned} \quad (3)$$

The value of T_i can be calculated, for stated conditions of bulk gas temperature, T_b , surrounding air temperature, T_a , and wall coefficient, h_w . Equation (2) is then used to calculate the mass transfer coefficient for elemental iodine or any other diffusing species for which the effective diffusivity is known. Assuming gas phase diffusion is limiting, the gas phase concentration calculated for a puff release is

$$\frac{C_g}{C_o} = \exp \left[- \left(k_c \frac{A_T}{V} + k_s \frac{A_W}{V} \right) t \right] \quad (4)$$

Calculations for different cases of h_w , T_b , and T_a showed that k_c and k_s are relatively independent of these parameters over the range of interest. Also, k_c is about ten times greater than k_s . Thus, the concentration half-time depends chiefly on the ratio of total surface area to volume.

The predicted values of heat- and mass-transfer rates are compared in Table 2.2 with experimental values from tests in two sizes of vessels. Agreement is fair, but improvement could be made to the model if either: (1) liquid reaction rates and diffusion were considered, or (2) allowance was made for the portion of surface area that may have laminar boundaries.

TABLE 2.2. Model Predictions Versus Experiments in the CSE

Run Number		$T_b - T_i$ °F	Steam Flux lb hr ⁻¹ ft ⁻²	k_c (I ₂) ft hr ⁻¹	k_s ft hr ⁻¹	$t_{1/2}$ (I ₂) min
D-1 (a)	Prediction	7.2	0.26	16.0	2.2	5.0
	Experiment	(c)	0.375	(c)	(c)	8 ± 1
D-2 (a)	Prediction	7.2	0.25	16.0	2.2	5.0
	Experiment	(c)	0.340	(c)	(c)	10 ± 1
A-1 (b)	Prediction	7.8	0.10	16.2	1.5	8.7
	Experiment	(c)	0.096	(c)	(c)	16 ± 2
A-2 (b)	Prediction	7.8	0.10	16.2	1.5	8.7
	Experiment	5.4 ± 2.0	0.11	(c)	(c)	10 ± 1

2.10

BNWL-894

(a) CSE drywell: $A_T/V = A_W/V = 0.46 \text{ ft}^{-1}$; $h_w = 20 \text{ Btu hr}^{-1} \text{ ft}^{-2} \text{ } ^\circ\text{F}^{-1}$;

$T_b = 250 \text{ } ^\circ\text{F}$, $T_a = 110 \text{ } ^\circ\text{F}$.

(b) Main CSE vessel: $A_T/V = 0.293 \text{ ft}^{-1}$; $A_W/V = 0.19 \text{ ft}^{-1}$; $h_w = 38 \text{ Btu hr}^{-1} \text{ ft}^{-2} \text{ } ^\circ\text{F}^{-1}$; $T_b = 180 \text{ } ^\circ\text{F}$; $T_a = 110 \text{ } ^\circ\text{F}$.

(c) Not measured or not applicable

A prediction for a large PWR with 4-ft thick concrete walls and A_T/V of 0.2 ft^{-1} gave 23 min for the concentration half-life of elemental iodine. This rate would hold until the concentration is about 1.5% of its initial value. Thus, natural transport processes can be an important mechanism for attenuating the consequences of a major accident to a water reactor.

Nomenclature

A_T = total internal surface area
 A_W = heat transfer surface area
 C_g = mass concentration of diffusing material in bulk gas
 C_o = mass concentration at time zero
 D_v = diffusion coefficient
 g = acceleration due to gravity
 h = heat transfer coefficient
 ΔH_{vap} = latent heat of condensation
 k_c = mass transfer coefficient due to diffusion
 k_f = thermal conductivity of gas film
 k_s = mass transfer coefficient due to condensation
 M = molecular weight of gas mixture
 p = partial pressure of steam
 Pr = Prandtl number
 g/A = heat flux
 R = gas constant
 Sc = Schmidt number
 t = time
 T = temperature, absolute
 V = gas volume
 ν = kinematic viscosity

Subscripts:

a refers to outside air
 b refers to bulk gas mixture
 i inside surface
 o outside surface
 w wall

Coolant Blowdown Studies - R. T. Allemann

A reactor simulator vessel with a volume of 150 ft³ and designed for operation at 600 °F and 2750 psig is being used in experiments to study loss-of-coolant accidents. The experiments have the objective of testing the validity of mathematical models used for predicting the consequences of reactor accidents. The vessel is mounted in a steel framework with the discharge nozzles directed horizontally. The blowdown apparatus is instrumented to measure internal pressures, temperatures, liquid level, void fraction of the fluid in the exit duct, mass of fluid remaining in vessel, thrust reaction forces, shell stresses, and forces on reactor dummy core structures.

The main effort this quarter was directed toward the construction of a new, vessel supporting-structure to replace the system that failed last year. Installation of the vessel support system is complete. The instrumentation on the simulator vessel is being installed in preparation for resuming experimental work.

Void Fraction Profile in a Vessel During Coolant Blowdown - N. P. Wilburn

Knowledge of the void fraction profile in a nuclear reactor vessel during a loss-of-coolant blowdown is important in several ways: it determines the foam level in the vessel and, hence, influences the rate of heat transfer from the core; it determines rates of flow through the vessel internals; finally, and most important, it determines the properties of the effluent. These properties (stagnation enthalpy and pressure) are required as input to two-phase critical flow models. Heretofore, two empirical approaches have been used to obtain an approximation to these properties:

1. The fluid in the vessel was assumed to be a homogeneous mixture⁽¹⁰⁾ and, hence, had a constant void fraction profile.

2. The void fraction profile was assumed to be linear in axial distance up the vessel,^(11,12) and effluent properties were computed from the void fraction at the withdrawal plane.

Moody⁽¹³⁾ has obtained a theoretical description on the liquid-vapor action in a vessel. He assumed that:

- The bubbles were small enough so that the gas phase velocity was that which would be present in an infinite liquid.
- The effluent void fraction was a weighted average of that above and below the withdrawal plane.
- A pseudo steady-state integration of the resulting differential equations was possible.

Comparison of his results with data recently obtained from the Containment Systems Experiment (CSE) facility shows that the foam level and effluent void fraction predicted from his model deviate significantly from measured values.

A different theoretical approach, predicting the void fraction profile and then the effluent stagnation enthalpy from mass and energy balances around the withdrawal plane, has been developed. The partial differential equations that were written, described the mass and energy balances inside the vessel but neglected kinetic energy terms. Thermal equilibrium was assumed throughout. The gas-phase velocity was assumed to have reached a terminal value everywhere with respect to the local liquid velocity, and the difference was assumed to be either constant (2 ft/sec) or given by Wilson's correlation with void fraction.⁽¹⁴⁾ Two, first-order nonlinear partial differential equations for the void fraction and the gas phase velocity result. It was found that these equations could not be directly integrated by standard finite difference techniques because of a phenomenon akin to "shocking" in wave equations.

However, a solution can be obtained by integration of the derived ordinary differential equations along the single set of characteristics by a Runge-Kutta or other technique. For the special case of constant pressure after an assumed initial depressurization, the void fraction and slope of the characteristic are constant on each characteristic and function of the initial void fraction. Thus, different initial values will cause intersection of the characteristics which is typical of "shocking".

The predictions of this model as to foam level and effluent void fraction will be compared with experiments using a vessel without internal resistances.

The results of this theoretical study indicate that the assumptions of a simple, linear void-fraction profile or the effluent void fraction being a weighted average of that above and below the withdrawal plane are not valid, or particularly good, when used to predict the effluent stagnation enthalpy.

Processing of Blowdown Data - N. P. Wilburn

Filtering. To reduce the random noise present in data collected from the blowdown facility and to eliminate undesired frequencies in the data (such as the 5 Hz vessel and support resonance), optional digital filtering of the data was incorporated into the 1108 data-processing code MPXDP. A third order recursive sine filter⁽¹⁵⁾ is used which has the following form:

$$y_n = w_0 x_n + \sum_{i=1}^{2m} w_i y_{n-i} \quad (1)$$

where y_n is the filtered result

x_n are the raw data

w_i are weights $\left(\sum_0^{2m} w_i = 1 \right)$

m is the order of the filter.

The Nyquist frequency $f_n = (2\Delta t)^{-1}$, where Δt is time between samples, is the maximum, possible, frequency component present in the raw data x_n . Given f_n , a cutoff frequency f_c is selected to filter the data. From f_c/f_n , the w_i are computed.⁽¹⁵⁾ The amplitude of all frequencies above f_c is greatly reduced in y_n .

As f_c/f_n becomes smaller, w_0 approaches zero; whereas, the other w_i are of order 1 to 10. Eventually, w_0/w_i becomes of order 10^{-6} ; the rounding or truncation error in the digital computer becomes significant; and the output of the filter, y_n , diverges. This point occurs for $f_c/f_n \sim 0.05$ in a third order sine filter.

The problem can be circumvented by a two-step digital filtering process, similar to a two-section passive analog filter. The data are first filtered with $f_c = 0.1 f_n$. Then, since the amplitude of the frequencies above $0.25 f_n$ are negligible, only every fourth output point (y_{4n}) need be retained. These are then filtered with $f_c/f'_n > 0.1$, but since f'_n is $0.25 f_n$, the effective cutoff frequency can be as low as $0.025 f_n$. Cutoff frequencies as low as $0.025 f_n$ are currently being used to filter the load-cell data ($f_n = 50$ Hz) where the 5 Hz resonance signal is 10 times that of the signal of interest.

Derivatives. The effluent mass flowrate from the vessel can be determined from the time derivative of the weight-load cell data (LC-1). To eliminate the tedious graphical procedures of derivative taking, a routine to determine derivatives of the data numerically has been included in the processing code MPXDP. The data are normally filtered first to eliminate high-frequency noise components amplified when derivatives are taken. Then the data are grouped into sets

of N points ($N = 4$ to 20) and a second order orthogonal polynomial fit to each set of data. The derivative is then computed at the midpoint of the set from an analytical differential of the polynomial.

Densitometer Data Reduction - R. T. Allemann

During blowdowns of the CSE reactor simulator vessel, the void fraction of fluid in the discharge duct is measured by use of a neutron densitometer. This system includes a fast-neutron generator and four, neutron detector chambers. The system depends on the moderation and scattering of the fast neutrons by hydrogen atoms in the fluid passing through the blowdown pipe. The neutron detector outputs do not give void fraction directly, and calibration of the system is necessary, as well as corrections to account for differences between calibration conditions and run conditions.

The neutron generator detector system is calibrated by filling the discharge pipe with a matrix of aluminum tubes, some of which are then filled with water to give different void fractions. The detector output is then plotted as a function of the fraction of water in the pipe. Since the detector is most sensitive to "slow" or moderated neutrons, its output is in turn a function of the amount of water or the void fraction in the discharge tube. The detector output is also a function of the neutron energy. Because the moderation is accomplished by the water, the energy of the moderated neutrons will be affected by the water temperature. Therefore, a temperature correction should be made to the detector output, particularly, if the fluid temperature changes during the blowdown. Also, a correction must be made for the fact that steam at high pressure and not air is the void fraction during the actual blowdowns. A final correction is made to account for changes in the neutron output of the generator.

Since calculation of these corrections is quite laborious, a computer routine is being programmed to process the data directly, giving void fraction and quality with time.

Effect of Quality on Mass Velocity - R. T. Allemann

Some preliminary mass velocity and quality data from CSE blowdowns have been compared in Figure 2.3, with predictions given by several mathematical models at similar pressure. The salient point of these data is that the quality of the fluid is very low for bottom blowdowns with either initially saturated or subcooled liquid. It has been suggested that little slip is possible at low quality and that deviations from the homogeneous model result from nonequilibrium effects.⁽¹⁶⁾ The CSE data agree with the trend predicted by the homogeneous model. The slip model equilibrium theories predict values of mass velocity that are too high. Even though these are plotted with quality values at the critical plane (i.e. the outlet), a correction to the inlet conditions for the Moody model still shows mass velocities higher than those measured. The Fauske model would probably not show better agreement in this low quality region.

Therefore, the preliminary data indicate that for the very low quality flows attained during bottom blowdown, a homogeneous model prediction gives better agreement than Moody⁽¹⁷⁾ or Fauske⁽¹⁸⁾ model predictions. The variation of flow with quality, even at these low values of quality, is an important factor in making vessel blowdown calculations. Because of this importance, a more detailed analytical treatment is being made of the vapor-liquid action in the vessel.

Reactor Simulator Vessel Core Sample Analysis -

W. C. Townsend

The CSE reactor simulator vessel was being used for an experiment with a 6.8 in. break diameter and initial coolant

FIGURE 2.3. Effect of Measured Quality on Mass Velocity

conditions of 2400 psig and 595 °F when a structural failure occurred in the thrust support system. This failure subjected the vessel to a relatively severe loading condition. Although no significant damage to the vessel was evident, extensive nondestructive tests including radiography, penetrant, magnetic particle and ultrasonic tests were carried out on all portions of the vessel which had been subjected to large stresses either during the completed blowdown experiments or the thrust support failure accident. Tested portions of the vessel were:

- The supporting trunnions and leg brackets on the vessel
- A previously noted but acceptable UT test defect indication in the top head
- The regions around the 8 in. blowdown nozzles loaded during blowdown experiments or the support failure accident.

Ultrasonic tests showed some enlargement of the known top-head defect indication and a new defect indication below the bottom blowdown nozzle used for previous experiments. Detailed ultrasonic tests were made with smaller crystals to map the defect-indication area. These indicated that the defect was bounded by irregular areas (Figure 2.4) which indicated loss of back reflection. In some, areas an indication of depth to a defect was obtained (cross-hatched areas of Figure 2.4). The depth indications were noted to be approximately constant, which suggested that the defect indication resulted from a laminar type defect.

Three, core samples denoted A, V, and Y were removed from representative areas in the vessel wall. The A sample was removed from apparently sound material, while the V sample was removed from an area indicating defect depth. The Y sample was removed from an area having 100% loss of back reflection and no depth indication. The locations of the core samples are shown in Figure 2.4.

Neg 0680708

FIGURE 2.4. Defect Regions in Reactor Simulator Vessel

A testing program is being carried out on the three, core samples to determine the material characteristics in the defect region and the nature of the defect indication.

The core-sample testing program is approximately 90% complete and includes the following tests:

Metallurgical examination	completed
Ultrasonic holography	completed
Ultrasonic	completed
Radiography	completed
Tensile	completed
Microprobe analysis	in progress
Fracture surface analysis	in progress

Analyses of the results from those tests completed are being compiled for a composite interpretation and report.

Preliminary results from one or more of the tests indicate:

- That the A sample does not have noticeable defects
- Agreement as to the location of the defect in the V and Y samples (This agreement includes radiographs, ultrasonics, photomicrographs, and visual measurements.)
- That the remaining material of each sample is still ductile
- That the fracture surface contains an apparently oxidized surface.

Ultrasonic holography techniques were applied; however, the results are inclusive. Acoustic emission data were taken during the tensile tests, and have not yet been reduced.

Banding of metallic layers was evident from the photomicrographs taken during metallographic examination. The band spacing on the A sample was approximately 1/8 in.; 0.004 in. on the Y sample. The V sample had approximately 50% with the 4 mil spacing, and the remaining section with the 1/8 in. spacing. Banding is not uncommon and is not considered harmful.

Photomicrographs of the A and V samples have been studied and compared with other photomicrographs of welded joints. The comparison suggested that the A and V core samples were removed from a previously welded area. Subsequent interpretation of fabrication reports indicates that A and V samples were possibly removed from a previous weld repaired region. From these reports, two large defect indication areas in the lower hemispherical head were weld-repaired. Although repair areas relative to the nozzles are not known, it is apparent that one of these is too near the welded joint (between the hemispherical head and the cylindrical shell) to be applicable. The other weld-repaired region is at the proper circumferential position so that the A and V core samples could have been removed from this region.

Results of the tensile tests are given in Table 2.3.

TABLE 2.3. *Tensile Tests of CSE Reactor Simulator Vessel Core Samples*

	<u>A</u>	<u>V</u>	<u>Y</u>
Gauge length, in.	0.42	0.65	0.74
Yield load, lb	1550	1560	700
Yield stress (avg), psi	49,400	49,600	22,300
Ult. load, lb	2,000	1940	960
Ult. stress (avg), psi	63,750	61,800	30,600
Extension at ult., in.	0.047	0.040	0.023
Strain at ult. (avg), %	11.2	6.1	3.15
Extension at failure, in.	0.175	0.105	0.0063
Strain at failure (avg), %	42	16	8.5

The reduced diameter of these specimens was uniform and 0.200 in. The axis of the tensile specimens was perpendicular to the surface of the simulator vessel surface; therefore, the plane at the defects is perpendicular to the axis of the specimens.

Stress and strain calculations were based on the nominal test area and not the actual stressed area. The actual stressed area will be smaller because of the defects. Then, the stress and strain in the V and Y samples will be larger than reported in Table 2.3.

During the tensile tests, a small flaw was evident in the V sample shortly after the yield stress had been reached. Undoubtedly, this was a prior defect. This flaw did not propagate but continued to open, and the material at the crack roots behaved in a ductile manner.

A large flaw about 1/8 in. long was evident on sample Y before testing. Another flaw appeared above the first at the yield stress. Considering the net area, the yield and ultimate loads are not unreasonable. Only the metal in the plane of the flaws reached the yield stress so that all deformation occurred there.

FISSION PRODUCT AEROSOL CONTROL - L. C. Schwendiman and
H. H. Van Tuyl

Removal of Organic Iodides with Reactive Sprays -
L. C. Schwendiman and A. K. Postma

Following a serious reactor accident, volatile and gaseous fission products will be released to the containment volume. A possible method for reducing the airborne concentration of molecular radioiodine and methyl iodide is to immobilize these constituents by adding a reactive compound such as hydrazine to the spray.

During the current reporting period, a series of once-through spray runs was completed in which airborne methyl iodide was sprayed with dilute hydrazine. The results were compared with predictions based on an absorption model. The

rate of washout of methyl iodide in a large containment vessel was predicted from the model and earlier experimental measurements.

The goal of the once-through experiments was to provide direct experimental measurements of the washout parameters A and B (amount of solute gas absorbed per unit volume of liquid per unit gas phase concentration, for wall film and drops, respectively). In each experiment, the wall film liquid was collected separately from the drops falling into the bottom of the chamber. Two, drop catchers (modified Erlenmyer flasks) were placed near the bottom of the chamber to collect falling drops with immediate isolation from the chamber atmosphere. These drop catchers were designed to preclude methyl iodide absorption by drop liquid subsequent to impact against any collecting surface. Humidified methyl iodide-free air was continually purged through the flask to prevent airborne methyl iodide from reaching the liquid surface. Ten cubiccentimeters of 10% sodium thiosulfate solution was added to the flask to react rapidly with unreacted methyl iodide carried within drops into the catcher. Drops impacting against an inlet chimney were not collected, but instead drained away at the outside of the flask. Only those drops falling nearly vertically entered the flask. Gas phase concentrations of methyl iodide were measured using beds of activated charcoal operating in series. The results of these runs are shown in Table 2.4.

The drop absorption coefficient, B, (Table 2.4), is the average of the values obtained for the two catchers used in each experiment. Drop sizes were measured from drops collected near the bottom of the 10 ft tall chamber. Thus, the mass median diameter measured would probably represent drops larger than the average in the chamber because coalescence would cause an increase in size with increasing distance from the nozzles.

TABLE 2.4. Results of Wall Film and Drop Absorption Experiments in Once-Through Spray Runs

Run No.	Average Temp, °C	Solution Composition	Wall Flow Rate, cm ³ /sec cm	Drop Flow Rate, cm ³ /sec	A	B	Mass Median Drop Diameter, μ
					Wall Film Absorption Coefficient	Drop Absorption Coefficient	
13	89	2.15 wt% N ₂ H ₄	0.138	47.8	6.27	1.06	320
14	88	1.0 wt% Na ₂ S ₂ O ₃	0.125	42.4	4.78	1.45	320
15	86	4.38 wt% N ₂ H ₄	0.143	52.4	8.55	1.95	320
17	87	1 wt% Na ₂ S ₂ O ₃	0.151	55.2	4.4	1.23	320

The temperature within the chamber decreased several degrees centigrade over the 7 min spray period. The spray solution was heated to the tank temperature before each test to reduce temperature variations.

The results obtained in these once-through runs in which drop catchers were employed are compared in Table 2.5 to the predictions based on stagnant film and drop theory. From Table 2.5, it should be noted that the measured washout rates are 1.5 to 3 times faster than predicted. This ratio of measured to theoretical washout rate is somewhat lower than found in earlier recycle experiments. This difference may be due to absorption into the shallow pool which existed in the bottom of the chamber. In the recycle experiments, bottom pool absorption could not be separately accounted for; whereas, in the once-through runs, wall film and drop absorption were measured directly.

Reactor containment vessels are much larger than those used in the experiments carried out in this study. Hence, application of the small scale tests to large scale systems of practical interest must be based on knowledge of the scale factors that apply. We have attempted to show the effect of

TABLE 2.5. Comparison of Once-Through Runs with Theory

Run No.	Reactant Concentration	Predicted Wall Absorption Coefficient	Predicted Drop Absorption Coefficient	A Experimental	B Experimental	Predicted Drop Absorption Wall Film Absorption	Measured Drop Absorption Wall Film Absorption
		A	B	A Theoretical	B Theoretical		
13	2.15 wt% N_2H_4	2.57	0.580	2.44	1.84	0.20	0.15
14	1.0 wt% $Na_2S_2O_3$	3.39	0.61	1.41	2.38	0.16	0.27
15	4.38 wt% N_2H_4	3.37	0.73	2.54	2.67	0.21	0.22
17	1.0 wt% $Na_2S_2O_3$	2.39	0.67	1.52	1.83	0.22	0.27

spray chamber size on methyl iodide washout rate by calculating the washout rate expected for cylindrical chamber from laminar flow theory.

In Figure 2.5, the washout rate of methyl iodide in cylindrical spray chambers, with height to diameter ratios of 2.5, is shown as a function of chamber size. The spray density was assumed to be $0.0068 \text{ cm}^3/\text{cm}^2 \text{ sec}$, drop diameter was taken as 550μ , and the wall flow rate was chosen as $0.1 \text{ cm}^3/\text{cm sec}$. The first order rate constant was chosen as 6.55 sec^{-1} , the value predicted for 5 wt% hydrazine solution at 120°C .

The increase in washout half-time with increasing chamber size is a reflection of the importance of wall absorption, which is predicted to account for over 50% of the total, even for a cylinder 100 ft in height. The curve shown is not highly sensitive to changes in the assumed conditions. For larger drops, the wall would be relatively more important, and the curve shown in Figure 2.5 would be steeper. For smaller drops or higher reaction rates, the curve would tend to level out, with the vessel size being of less importance.

Based on this calculation, we would expect a washout half-time of about 60 min for a vessel 80 ft in height for 5 wt% hydrazine, for the nominal spray conditions specified in Figure 2.5.

Physical Chemistry of Methyl Iodide Systems - R. A. Hasty and S. L. Sutter

The objective of this research is the determination of the physiochemical factors affecting the rate of spray removal of

FIGURE 2.5. *Predicted Effect of Vessel Size on Washout Rate for Methyl Iodide*

methyl iodide from reactor containment atmospheres. Earlier studies have demonstrated a complex relationship of the rate of removal of methyl iodide from the gas phase. Such a relationship is dependent upon the properties of the reactive solution used as the spray.⁽¹⁹⁾ The rate of reaction of methyl iodide in the spray solution can be determined in the laboratory by the use of the "sparge removal technique", described in our last contribution⁽²⁰⁾ along with its utilization to determine the kinetic expression for the reaction of methyl iodide in hydrazine solutions. Since the technique can be applied to a number of chemical systems, we have determined the reaction rate constants for several reagents that can be used to react with methyl iodide. The reactants that have been studied or are currently being studied are 1,1-dimethylhydrazine, monomethylhydrazine, sodium sulfite, sodium bisulfite, and sodium thiosulfite. Kinetics of the reaction of methyl iodide with the latter reactant have been reported in the literature, but the study was over such a narrow temperature range (10 to 25 °C) that a re-examination of the earlier work and extension of the measurements to higher temperatures were considered worthwhile.

The results of the kinetic investigation of the five reactants examined are given in Table 2.6. The reaction of methyl iodide with sodium sulfite was interesting in that the reaction rate was affected by the ionic strength as well as by pH.

TABLE 2.6. Reaction Rate Studies

Reactant	Bimolecular Reaction Rate Constant, liters moles ⁻¹ sec ⁻¹	Temp, °C	Concentration Range Studied, moles liter ⁻¹	Temp Range Studied, °C	Energy of Activation, kcal moles ⁻¹
1,1-Dimethylhydrazine	7.7×10^{-3}	24.7	0.13-0.53	24.7-34.7	18.4
Monomethylhydrazine	$<1.5 \times 10^{-4}$	35.4	0.076	35.4	--
Sodium sulfite	4.1×10^{-2}	24.7	0.02-0.08	24.7-49.3	18.6
Sodium bisulfite	6.4×10^{-4}	24.7	4.7	24.7-49.3	~19
Sodium thiosulfate	3.1×10^{-2}	24.7	*	*	*

* Studies incomplete

Earlier radiation stability research and studies made recently at ORNL have shown that hydrazine decomposes and releases hydrogen in copious quantities under the high radiation anticipated in a reactor accident. A second disadvantage recently disclosed was the impact that the use of any ammonia derivatives would have on the design of spray cooling heat exchangers. Current specifications call for copper alloy components that would be quickly corroded by ammonia derivatives. These observations have been reviewed and a conclusion reached that hydrazine as a spray additive for methyl iodide will not be acceptable to commercial reactor designers, even though it has many favorable features.

Research to demonstrate the value of hydrazine, per se, as an additive to sprays for removing methyl iodide is terminated. However, reaction rate data made available by these studies and the experience acquired in using hydrazine as a spray additive will be of considerable value in experimental verification of droplet removal models when liquid reaction rates limit the absorption process.

REFERENCES

1. *The Staff of Battelle-Northwest. Nuclear Safety Quarterly Report for January through March 1967, BNWL-433. Pacific Northwest Laboratory, Richland, Washington, July 1967.*
2. *The Staff of Battelle-Northwest. Nuclear Safety Quarterly Report for April through June 1967, BNWL-537. Pacific Northwest Laboratory, Richland, Washington, December 1967.*
3. *The Staff of Battelle-Northwest. Nuclear Safety Quarterly Report for July, August, September, and October, 1967, BNWL-754, p. 2.1. Pacific Northwest Laboratory, Richland, Washington, June, 1968.*
4. *The Staff of Battelle-Northwest. Nuclear Safety Quarterly Report for November, December, 1967, January, 1968, BNWL-816. Pacific Northwest Laboratory, Richland, Washington, September 1968.*

5. M. Jacob. Heat Transfer, John Wiley & Sons, Inc., New York, 1949. vol. I, p. 592.
6. R. K. Hilliard et al. Comparisons of the Containment Behavior of a Simulant with Fission Products Released from Irradiated UO_2 , BNWL-581. Pacific Northwest Laboratory, Richland, Washington, March 1968.
7. R. L. Ritzman, et al. "Analytical Description of Fission Product Transport and Deposition in Containment Vessels," Trans. Am. Nucl. Soc., vol. 10, no. 2. November 1967.
8. G. M. Watson et al. Effects of Containment Size on Fission Product Behavior, ORNL-4033. Oak Ridge National Laboratory, January 1967.
9. W. H. McAdams. Heat Transmission, McGraw-Hill Book Co., Inc., New York, N.Y., 1954. 3rd ed., p 173.
10. S. G. Margolis and J. A. Redfield. Flash: A Program for Digital Simulation of the Loss-of-Coolant Accidents, WAPD-TM-532. Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, May 1966.
11. J. A. Redfield, J. H. Murphy, and V. C. Davis. Flash 2: A Fortran IV Program for the Digital Simulation of a Multinode Reactor Plant During Loss-of-Coolant, WAPD-TM-666. Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, April 1967.
12. Quarterly Technical Report STEP Project, January-March 1967, edited by K. A. Dietz, IDO-17239. Idaho Operation Office, Idaho Falls, August 1967.
13. F. J. Moody. Liquid/Vapor Action in a Vessel During Blow-down, APED-5177. General Electric Company, San Jose, California, June 27, 1966.
14. J. F. Wilson, R. J. Grenda, and J. F. Patterson. "The Velocity of Rising Steam in a Bubble Two-Phase Mixture," Trans. Am. Nucl. Soc., vol. 5, no. 1, pp. 151. 1966.
15. H. Holtz and C. T. Leondes. "The Synthesis of Recursive Digital Filters," Jahrbuck der Chemil/Meyer, vol. 13, no. 2, pp. 262-280. August 1966.
16. R. E. Henry. A Study of One-and-Two-Component Two-Phase Critical Flow at Low Qualities, University of Notre Dame, 1967. (Ph.D. thesis)

17. F. J. Moody. "Maximum Flow Rate of a Single Component, Two-Phase Mixture," J. Heat Transfer, vol. 87-C, p. 134. 1965.
18. H. K. Fauske. Contribution to the Theory of Two-Phase, One-Component Critical Flow, ANL-6633. Argonne National Laboratory, Lemont, Illinois, 1962.
19. R. A. Hasty. "Reaction of Hydrazine and Iodomethane. Removal of Iodomethane from a Gas Phase in Contact with Alkaline Hydrazine Solution," J. Chem. Eng Data, vol 13, pp. 134-138. 1968.
20. R. A. Hasty, "Physical Chemistry of Methyl Iodide Systems," Nuclear Safety Quarterly Report, February-April, 1968, for Nuclear Safety Branch of USAEC Division of Reactor Development & Technology, BNWL-885. Pacific Northwest Laboratory, Richland, Washington, p. 2.49. October 1968.

3. PRESSURE BEARING COMPONENT EVALUATION AND MONITORING STUDIES

CRACK DETECTION IN PRESSURE PIPING BY ACOUSTIC EMISSION -

P. H. Hutton, R. N. Ord, C. E. Fitch, H. N. Pedersen, and
J. C. Spanner

Introduction

Development of new and more discerning techniques for nondestructively testing reactor pressure boundaries to assess their integrity is of general interest to the nuclear industry for further enhancement of nuclear safety. One such technique showing considerable promise is based on detection of "acoustic emission" or elastic waves generated in metal by the energy released as it plastically deforms and fractures. This technique has potential application for integrity surveillance during both hydrostatic testing of a reactor system and on-line operation. It is, however, limited to detection of a dynamic condition--i.e., it will not detect a static flaw or discontinuity.

The USAEC, Division of Reactor Development and Technology has sponsored development work at Battelle's Pacific Northwest Laboratory, starting in February 1966, directed toward development of the technology and equipment needed to apply this new surveillance technique routinely to reactor systems. In the interest of increased overall effectiveness of development work, a coordinated effort has been established between this program and acoustic emission plant application work being done by Phillips Petroleum Company at the Idaho Test Site.

Test work has been concentrated on low-alloy carbon steel and Type 304L SS.

Discussion

Task 1 - Acoustic Emission Signal Analysis and Characterization

It has long been realized that maximum, reliable utilization of acoustic emission signals will require a knowledge of acoustive wave generation, and propagation in the finite media of the various reactor pressure system materials. It has been suspected, but not fully confirmed, that signals of a much higher frequency than those commonly detected in acoustic emission tests form the basic signal generated during the deformation and fracture of metals. Analysis of these high frequency signals can provide more knowledge of metallurgical behavior effects on acoustic emission, and thus enhance flaw detection, location, and description capabilities. A recent reactor background noise study (discussed later in this report) has provided added impetus to investigating the higher frequency acoustic emission.

Experiments were conducted to study ultrasonic wave behavior in a finite plate geometry using wide-band, short-duration ultrasonic pulses. Short-duration, high-frequency, acoustic emission signals should experience behavior comparable to the ultrasonic pulses in a plate, and this technique provides an easy, inexpensive means of generating preliminary guide data. By observing and analyzing the ultrasonic signals, the expected behavior of acoustic emission signals will be more clearly understood.

A 3 ft x 4 ft x 1/2 in. thick plate of A-212B steel was used for these initial studies. It was anticipated that signals produced by grain failure in tension, and in shear could be approximated by introducing the ultrasonic signals normal to the plate edge with longitudinal and shear transducers. To more easily separate and study the response waves that

travel in the plate, a receiver was mounted with a fixture to provide critical-angle adjustments for various propagation modes. Water couplant contained in a latex membrane was used to immerse the receiver, and alcohol was used between the latex membrane and the plate to avoid undesirable oxidation.

The critical angles favoring selective detection of longitudinal, shear, and Rayleigh wave transmission modes were chosen for initial studies. With a longitudinal transmitter mounted on the edge of the plate and the receiver set at the longitudinal critical angle, the first signal to the receiver will be a pure longitudinal wave. This signal should be free of other interference provided that energy from preceding transmitted pulses has completely decayed. The observed signal is the first longitudinal, and the subsequent signals are due to mode conversion from longitudinal to shear at an angle θ_s in the plate, where

$$\theta_s = \sin^{-1} \frac{V_s}{V_1},$$

provided the longitudinal wave mode converts at an angle of 90 degrees.

The time separation (t) of each subsequent pulse is calculated from the relationship

$$t = \frac{T}{V_s} (1 - V_s^2/V_1^2)^{1/2}$$

where T is the plate thickness. For the test situation, T was 1/2 in., $V_s = 1.28 \times 10^5$ in./sec, and $V_1 = 2.34 \times 10^5$ in./sec. With these values, t calculates to be 3.27 μ sec, which agrees with Figure 3.1 (Time scale = 4 μ sec/cm).

The signal observed at the shear critical angle for the above transmitter arrangement is shown in Figure 3.2. Since the transmitter is a longitudinal polarized transducer,

FIGURE 3.1. Longitudinal Critical Angle Signals for 1/2 in. Plate Using Longitudinal Transmitter

FIGURE 3.2. Shear Critical Angle Signals for 1.2 in. Plate Using Longitudinal Transmitter

response at the shear critical angle should not be expected. However, with the transmitter being a lead zirconate titanate (PZT) ceramic element, polarization in one mode of vibration is not entirely possible. Consequently, some shear energy is generated. These signals are evident at the very beginning of the trace (Figure 3.2) and are the very small, high frequency pulses preceding the main energy pulse. The reason for more than one shear mode signal is not clear but may be due to material anisotropies. (The pulse arriving with the fastest time was used for shear wave velocity determinations.)

As the receiver angle was further increased to the Rayleigh angle, the major peak shown in Figure 3.2 increased in amplitude and was essentially the only high frequency pulse remaining. From this, it is implied that the major peak in Figure 3.2 was caused by Rayleigh waves. Also, it is of interest that the main signal observed when the receiver angle was reduced to zero (or normal incidence) was from the Rayleigh wave. This normal incident receiver orientation is the customary way in which acoustic emission signals are detected and generally the indicated propagation velocities are near theoretical Rayleigh velocities. However, in some cases a significant variation has been observed. The question arises as to how signals are even detectable with normal incidence receiver orientation since the main polarization direction is perpendicular to the direction of wave propagation. This is possible because the ceramic transducers have relatively strong low-frequency radial and circumferential modes that respond to waves propagating parallel to the plate surface. Though these radial modes do have advantages for some acoustic emission detection applications, they also tend to complicate analysis of wave disturbances using wide band electronic systems because of sensitivity to more than the thickness mode polarizations (Figure 3.2). This raises a further question which will

hopefully be answered by this work--i.e., might not signals which are marginal when detected with a normal incident receiver be greatly enhanced by critical angle detection?

The next experiment used a shear polarized PZT transmitter also mounted at the plate edge. This arrangement was expected to produce signals similar to those from shear failures. When the receiver transducer was again set at the longitudinal wave critical angle, small amplitude signals very much like those in Figure 3.1 were observed which indicated that some longitudinal waves were generated by the shear poled PZT element.

At the shear critical angle, the signal shown in Figure 3.3 was observed. The signal arriving first in time was the direct transmitted shear. As before, the Rayleigh surface wave signal was also substantially present in addition to contributions from the other unexplained signals discussed for Figure 3.2. As the receiver angle was further increased to the Rayleigh critical angle, the shear mode tended to disappear as the Rayleigh mode dominated. Also, as can be seen in Figure 3.3, low frequency radial and circumferential modes were present, making wide band signal interpretations difficult.

From the previous study, it appeared that information pertaining to whether actual acoustic emission signal originates from tensile or shear failure of crystals, could be determined. If a crack mechanism occurs in tension, it should stimulate strong longitudinal waves and possibly low-amplitude shear. Conversely, if cracking occurs in shear, then this mode should dominate. As can be seen in the preceding figures, the higher frequencies contain most of the pure longitudinal, shear, and Rayleigh mode information. Also it is believed that the higher frequencies may eventually tell more about other subtle failure mechanisms.

FIGURE 3.3. *Shear Critical Angle Signals
for 1/2 Plate Using Shear
Transmitter*

Initial work attempting to detect higher frequency acoustic emission signals using a wide band electronic system was unsuccessful because the high frequency noise tended to mask the desired information. To overcome this, a narrow band electronic system was introduced. Optimum frequency "windows" for acoustic emission signal detection have not yet been defined; however, transmission of frequencies in the 0.5 to 3.0 MHz range over long path lengths is of obvious practical importance when recently acquired reactor noise data are considered. To select optimum, narrow frequency bands for emission detection, a parametric study must be undertaken.

Available amplifiers, tuned in the 5 MHz frequency range, were used for some initial studies; more versatile, tuned amplifiers are on order. To keep the experiments within the capabilities of laboratory tensile equipment, small 1/8 in. thick strips of metal were used. Preliminary work was done with single-edge notch specimens of Type 304L SS and Type A-212B carbon steel.

Small 0.125 in. diam 5 MHz PZT-5 transducers were mounted with special fixtures to facilitate changing the receiver angle.

One transducer with a narrow-band amplifier was positioned near the notch and was used to trigger the oscilloscope. Another, positioned farther down the strip, had its signal amplified and fed to the oscilloscope vertical display. Signals received on the oscilloscope for longitudinal, shear, and Rayleigh critical angles were recorded photographically. Because the transducers were small, their simultaneous response to shear and Rayleigh modes appeared to be equally favorable at either critical angle; therefore, data were taken at only two angles--the longitudinal critical angle and an angle between the shear and Rayleigh.

Typical acoustic emission signals for 304L SS at the longitudinal angle are shown in Figure 3.4, and typical shear-Rayleigh signals are shown in Figure 3.5. As expected, many successive signals arrived at the longitudinal angle; however, contrary to expected results by the earlier plate work, more than one successive signal also appeared in many of the shear-Rayleigh emission pulses. Earlier work did show more than one shear signal, but the spacing between these would be smaller in the strip material because of smaller path lengths and sample thickness. (This spacing would be condensed in the first portions of signals shown in Figure 3.5.) One explanation for these multiple signals is that more than one emission may occur in a very short time period for one metallurgical event. Present state-of-the-art does not adequately describe fracture mechanisms as related to specific acoustic emission, and complete interpretation of observed signals is not yet possible. Some work was done with oil-tempered tool steel, and signal differences (as compared to the more ductile materials) were observed.

The above results are obviously very preliminary. Continued investigation is planned to define the optimum transducer angle, to relate high-frequency signal changes to material characteristics, and to determine the optimum trade-off point between material attenuation of high frequencies versus proximity to reactor noise of lower frequencies.

FIGURE 3.4. *Typical Longitudinal Critical Angle Acoustic Emission Signals for 304L SS Strip*

FIGURE 3.5. Typical Rayleigh-Shear Critical Angle Acoustic Emission Signals for 304L SS Strip

Task 2 - High Temperature Transducer Development

Preliminary laboratory test work has shown that detection of acoustic emission using critical angle transducer orientation and operating in a higher frequency range (low MHz) has the potential for significant improvement in the quality of information obtained. It appears that this will improve the signal to noise ratio, minimize transmission-velocity variations, and sharpen the wave front definition.

Figure 3.6 shows a sample of the results obtained in this preliminary work. The three transducer arrangements were installed on a steel single-edge-notch test specimen, together with a fourth transducer used to trigger the oscilloscopes. Orientation of the trigger transducer and the other transducers was such that source of the observed signal could be positively identified by position on the scope screen. This provided an easy and reliable means of identifying the signal as originating from the flaw rather than from the tensile machine or grips. All three traces were generated by the same acoustic emission event. The top trace is the response of a "standard" 1 MHz transducer arrangement such as has been used to date for test work. The bottom two traces were produced from a 2 and 5 MHz transducer oriented at the shear wave critical angle. Operating at the higher frequencies permits reduction of background noise by filtering or narrow band amplification, with the wave front becoming more sharply defined. This latter effect enhances the accuracy of source location by improving the accuracy of determining time relationships among various transducers. Use of the critical angle transducer orientation is beneficial in that it favors reception of a selected propagation mode which, in turn, tends to standardize the propagation velocity involved.

1 MHz TRANSDUCER
VERTICAL MOUNT
20 kHz HI-PASS FILTER
RTV-116 SILICONE RUBBER COUPLANT

2 MHz TRANSDUCER
SHEAR ANGLE MOUNT
500 kHz HI-PASS FILTER
WATER COUPLANT

5 MHz TRANSDUCER
SHEAR ANGLE MOUNT
4-1/2 - 5-1/2 MHz NARROW BAND
AMPLIFIER
WATER COUPLANT

2 μ SEC/CM SWEEP RATE IN ALL
CASES
PZT-5A TRANSDUCER CRYSTALS

FIGURE 3.6. *Effect of Transducer Frequency and Orientation on Response to the Same Acoustic Emission Signal*

These results are of particular interest because of their compatibility with the results of nuclear reactor noise measurements discussed under "Task 6 - Application Engineering."

Some of the more apparent facets of this technique yet to be resolved include determination of the maximum useable frequency for effective propagation in materials of interest, development of an omnidirectional, critical angle transducer, and determination of a coupling method suitable for field use.

Several 1 MHz, PZT-5A transducers, fabricated by use of a ceramic backing material, are being tested for their ability to withstand extended exposure to high temperature (600 to 650 °F) environment. The epoxy material previously used in fabricating these transducers is not satisfactory for long-term high-temperature applications because it gradually deteriorates with time at temperature. Sensors of this type were used in noise studies on the Dresden I reactor and after about three weeks of service, although they were still functional, the damping effect of the backing had obviously deteriorated.

Task 3 - System Analysis Study

No further work has been done this quarter.

Task 4 - Monitor System Development

A low-noise, solid-state wide-band preamplifier has been fabricated and field tested with satisfactory results. Bandwidth of the current design is 1 KHz to 3 1/2 MHz with a noise level of 16 μ V referred to input, which is only about a factor of two above the noise level of the best commercial preamplifier yet found over a much narrow bandwidth (1 to 300 KHz). The new preamplifiers have been built and tested in both X10 and X100 versions. Figure 3.7 shows the compact size of the instrument.

ACOUSTIC EMISSION

Low Noise, Broadband Preamplifier Developed for Acoustic Emission Detection

FIGURE 3.7. *Low Noise, Broadband, Pre-amplifier Developed for Acoustic Emission Detection*

Task 5 - Cooperative Work With Phillips Petroleum Company

Task 6 - Application Engineering

Tasks 5 and 6 are discussed together because one of the major efforts on the acoustic emission program this quarter was a cooperative study of background noise at the Dresden I nuclear power reactor by Phillips Petroleum Co. (PPCo.) and Battelle-Northwest (BNW). Dresden I is a part of the Commonwealth Edison Co's. electrical power generation network located near Morris, Illinois. Arrangements were made by PNL to install transducers at selected locations on the reactor at selected locations on the reactor system in late May, 1968.

BNW installed sensors and supporting electronics at four locations on the Dresden I pressure system (Figure 3.8). These locations were: 1) 22 in. west recirculation header, 2) a control rod thimble external to the reactor, 3) 6 in. bypass line on the "D" secondary steam generator, and 4) top head of the reactor vessel (two sensors).

In addition to the sensors installed by BNW, PPCo. installed three sensors on the 22 in. west recirculation header to obtain independent information. One of these sensors was provided by BNW; the other two commercial sensors came from PPCo.

One of the BNW sensor channels on the vessel head was nonfunctional, apparently because of a broken wire at the preamplifier. The second sensor at this location provided only limited data because the preamplifier failed after two days of reactor operation due to the environmental temperature exceeding component limits. PPCo. experienced early deterioration (approximately two days at temperature) of their two commercial sensors because of temperature effects.

The BNW measurement system is illustrated in Figure 3.9, with the transducer response characteristics shown in Figure 3.10.

Noise measurements were made on all functional points at various stages from initial start-up (full cold recirculation flow) to about 90% of full power (190 MW electrical). The data obtained are summarized in Figures 3.11 through 3.14. This is a valid representation of the data because with the flow essentially constant, there was no significant change in the noise spectrum on the primary recirculation system from cold conditions to full power level. The total flow through the vessel was approximately 26 million pounds per hour. Each 22 in. header carries essentially half of

Ⓣ = Transducers for Noise Measurement

FIGURE 3.8. Partial Schematic of Dresden I Nuclear Power Reactor System Showing Transducer Locations for Noise Measurements

3.17

BNWL-894

FIGURE 3.9. Schematic of Instrumentation System

FIGURE 3.10. *Response Characteristics of Transducer Used to Measure Background Noise-Dresden I Constant Amplitude-Swept Frequency Input*

this or 13 million pounds per hour for a corresponding flow velocity of 25 ft/sec. As shown in Figure 3.13, there were two different flow conditions, and in turn, two different noise spectrums for the 6 in. by-pass line. Unfortunately, there were no apparent feasible means of determining with any accuracy what the flow values were in either case. The total recirculation flow splits about equally among four secondary steam generator systems. During start-up, flow to the secondary steam generators is controlled by manipulating a main 18 in. valve and a 6 in. by-pass around this valve (where one of the transducers was located) downstream of the heat exchanger, together with a 6 in. line that completely bypasses the heat exchanger. The control parameter is steam generator output, with no direct measure of the specific flow in the various parts of this system. For a brief period early in the start-up cycle, the main valve was throttled; this

FIGURE 3.11. Noise Spectrum 22 in. West Recirculation Header

FIGURE 3.12. Noise Spectrum Control Rod Thimble

FIGURE 3.13. Noise Spectrum 6 in. By-Pass Line

FIGURE 3.14. Noise Spectrum Reactor Vessel Head

increased the flow through the 6 in. by-pass. The upper curve in Figure 3.13 was the resulting noise spectrum out to 1 MHz. No analysis beyond 1 MHz was obtained for this condition; however, the spectrum falls off quite rapidly toward 1 MHz. When the main valve was opened, the noise spectrum assumed a shape very similar to that of the other points.

The noise distribution measured on the head of the vessel, shown in Figure 3.14, was measured only out to 1 MHz because of the unexpected early failure of the preamplifier in this location.

Figure 3.15 shows the information from which the measurement-system electronic-noise spectrum included in the previous curves was derived. These data were generated by using the total measurement system including a transducer cushioned with the face open to air during the measurements. This was done at the Dresden site and again in the laboratory with consistent results.

Typical spectral data traces for the various locations are presented in Figures 3.16 through 3.19. A relative amplitude scale is used in presenting all of the spectral distribution information. Since the information is random in nature, a more accurate measure of absolute amplitude can be derived from the video traces of the actual signals as given in Figures 3.20 and 3.21. As the spectral curves show, the bulk of the noise at all locations is contained in the frequency range from dc to 200 kHz with the exception of the high flow case for the 6 in. by-pass line. In this latter case, significant noise components are evident on out to the 900 kHz to 1 MHz range but this was not reflected in noise data obtained from the other points on the system. The preliminary conclusion drawn from this is that an acoustic-emission monitor system operating in the 750 kHz to 3 MHz range would be relatively

Tektronix Type IL5 Analyzer

Tektronix Type IL10 Analyzer

20 kHz Hi-Pass Filter Used in Both Cases
 10^4 Gain

FIGURE 3.15. *Measurement System Noise Spectrum
 Transducer in Air and Connected
 to Electronic System Used for
 Reactor Noise Measurements*

FIGURE 3.16. Noise Spectrum Sample-22 in. West Recirculation Header, Dresden I Nuclear Power Reactor During Start-up and Normal Operation

Frequency, MHz
20 kHz Hi-Pass Filter

Frequency, MHz
200 kHz Hi-Pass Filter

Tektronix Type IL5 Analyzer
 10^4 Gain

Frequency, MHz
Tektronix Type IL10 Analyzer
 10^4 Gain

FIGURE 3.17. Noise Spectrum Sample-Control Rod Thimble Adjacent to Vessel Exterior Dresden I Nuclear Power Reactor During Start-up and Normal Operation

20 kHz Hi-Pass Filter

200 kHz Hi-Pass Filter

Tektronix Type IL5 Analyzer
Normal Flow-Main Valve Open
 10^4 Gain

Frequency, MHz
20 kHz Hi-Pass Filter

Tektronix Type IL5 Analyzer
Normal Flow-Main Valve Open
 10^4 Gain

Frequency, MHz
20 kHz Hi-Pass Filter

Tektronix Type IL5 Analyzer
High Flow-Main Valve Throttled
 10^4 Gain

FIGURE 3.18. Noise Spectrum Sample-6 in. By-Pass, "D" Secondary Steam Generator Dresden I Nuclear Power Reactor During Start-up and Normal Operation

FIGURE 3.19. *Noise Spectrum Sample-
Reactor Vessel Head
Dresden I Nuclear Power
Reactor During Start-up*

unhampered by noise interference except for possible limited periods of very high local-flow velocities (>25 ft/sec). As pointed out previously, however, such a conclusion must be treated somewhat cautiously because it is based on information obtained from one commercial power reactor plus some earlier experience with the Hanford plutonium production reactors. Even on a qualified basis, the information is of considerable value in continued work toward development of a practical system for utilizing acoustic emission as a means of continuously monitoring for crack growth in the pressure boundaries of an operating reactor. It offers encouragement that success can be achieved, and it points to the frequency region where development work on improved detection techniques can be most productively applied.

0.01 V/cm Vertical, 2.0 μ sec/cm Sweep
 10^3 Gain into the Oscilloscope
 200 kHz Hi-Pass Filter
 Measurement System Noise

0.01 V/cm Vertical, 2.0 μ sec/cm Sweep
 10^3 Gain into the Oscilloscope
 200 kHz Hi-Pass Filter
 22 in. West Recirculation Header

0.02 V/cm Vertical, 2.0 μ sec/cm Sweep
 10^3 Gain into the Oscilloscope
 200 kHz Hi-Pass Filter
 Control Rod Thimble Adjacent to
 Vessel Exterior

0.02 V/cm Vertical, 1 mil-sec/cm Sweep
 10^3 Gain into the Oscilloscope
 200 kHz Hi-Pass Filter
 Reactor Vessel Head

FIGURE 3.20. *Video Traces of Noise Signals Dresden I Nuclear Power Reactor During Start-up and Normal Operation*

0.01 V/cm Vertical, 2.0 μ sec/cm Sweep
 10^3 Gain into the Oscilloscope
200 kHz, Hi-Pass Filter

5.0 V/cm Vertical, 0.1 mil-sec/cm Sweep
 10^3 Gain into the Oscilloscope
20 kHz, Hi-Pass Filter

6 in. By-Pass, "D" Secondary Steam Generator

FIGURE 3.21. *Video Traces of Noise Signals
Dresden I Nuclear Power Reactor
During Start-up and Normal
Operation*

Work Planned for the Next Quarter

Work for next quarter includes continued study of wave-propagation characteristics and high-frequency critical-angle detection toward definition of the optimum frequency and technique for on-line plant application. Also, a series of tests will be continued to study frequency classification, event rate, and pulse amplitude for different material conditions.

Transducer development-omnidirectional, critical angle design, and high temperature sensor development and testing will be continued while a study of coupling techniques for high temperature field application will be initiated.

Acoustic emission produced during one pipe fatigue test at General Electric Company's Vallecitos plant will be monitored, and tests to further define capability of acoustic emission detection in-plant will be performed in cooperation with Phillips Petroleum Company.

•

•

•

•

•

•

•

•

4. ENVIRONMENTAL STUDIES

R. F. Foster

DISPOSAL OF REACTOR OFF-GAS INTO SOIL SYSTEMS - L. G. King

Battelle-Northwest has been conducting research work contributing to the analysis and interpretation of results from a tracer test at a field site within the National Reactor Testing Station (NRTS) in Idaho. During the field test, air containing ^{133}Xe as a tracer was injected into the fractured basalt rock between the depths of 110 and 122.5 ft below ground surface via a cased well. Samples were collected at various points at the ground surface, in the atmosphere a few feet above ground surface, and at various depths in wells constructed especially for that purpose. Battelle-Northwest has developed mathematical models to describe the gas movement during the injection phase of the field test. Two different approaches to modeling the gas flow have been pursued. The first assumes homogeneity with respect to permeability and allows solutions to boundary value problems to be obtained in closed form. The second approach allows permeability to be a function of space and, hence, relies upon finite difference methods of solution.

The homogeneous model was not sufficient for predicting the gas movement in the field. This model adequately represented the total flow rate that could be injected into the ground, but travel times calculated with the model were much greater than measured in the field. The model was extended to include homogeneous, anisotropic media having a horizontal permeability much greater than the vertical permeability. The latter model appears to give realistic explanation for observed gas movement.

The finite-difference model with layers of different permeability (homogeneity was assumed within each layer) was

also not capable of predicting measured travel times but did adequately predict total flow rate of gas during the injection. Details are included in the final reports on this project which terminated at the end of June 1968. The reports are expected to be released in the near future.

SIMULATION MODELING OF THERMAL GENERATION IN SELECTED RIVER SYSTEMS - R. T. Jaske

Operations research with the COL HEAT thermal simulation model to investigate the effects of future nuclear power plants on temperatures of the Upper Mississippi Basin is nearing completion for the trunk stream and several major tributaries. During this report period the following specific accomplishments were made:

- Test simulations of the Minnesota, St. Croix, Chippewa, Wisconsin, Rock, Cedar-Iowa, and Des Moines Rivers were carried out using data collected earlier in the program. The simulations, with the exception of the Chippewa and Wisconsin Rivers, are satisfactory. A paucity of good field data appears to be responsible for the discrepancies in the cases of the Wisconsin and Chippewa Rivers; consequently, a detailed review of the two streams is underway.
- As follow-up report to BNWL-728, "A Test Simulation of the Temperature of the Illinois River and a Prediction of the Effects of Dresden II and Dresden III Reactors," is near completion. The report covers the reach between River Mile 230 and River Mile 0. The report is expected to be issued in October.

Field data have been obtained for a test simulation of the thermal regime of the Phelps County (Nebraska) irrigation canal. The canal extends for approximately 60 miles downstream

of the Canaday Power Plant, a 100 MW steam plant that uses canal water for direct cooling. This study is expected to provide information of value for possible future use of irrigation waters for cooling of power reactors.

5. FIXATION OF RADIOACTIVE RESIDUES

PHOSPHATE GLASS SOLIDIFICATION

Corrosion from Fluoride-Bearing Feed Solutions - R. F. Maness

Corrosion of the denitrator-evaporator (all titanium construction) by fluoride-bearing feed solution was estimated by exposing titanium coupons to 100 gal/tonne PW-1 solution (meq/P = 1.0) containing 0.1M HF. The solution contained 0.5M $\text{Al}(\text{NO}_3)_3$ with an equivalent reduction in the $\text{Fe}(\text{NO}_3)_3$ content. The test was made in all-Teflon[®] equipment with semi-continuous addition of feed to solution concentrated to a boiling point of 133 °C. The exposure time was 19 hr. Titanium corrosion rates in the liquid and in the vapor phases were 5.0 and 1.3 mils/mo, respectively. A titanium coupon exposed to the boiling condensate collected from the above test corroded at a rate of 16 mils/mo. Based upon this value, the fluoride content of the condensate was estimated to be about 5×10^{-3} M.

A phosphate glass run was made with the concentrated feed solution produced above. The run was made at 1100 °C with semi-continuous feed addition to an alumina crucible equipped with an overflow weir. Platinum specimens were exposed to the melt and to the vapor phase, and a Nionel specimen was exposed in the off-gas line immediately above the condenser. The platinum specimen exposed to the melt corroded at a rate of 0.23 mils/mo (50 hr exposure) with no evidence of embrittlement. The specimen exposed to the vapor phase evidenced no weight loss. However, the specimen fractured on bending, which indicated embrittlement. The Nionel specimen exposed in the off-gas line corroded uniformly at a rate of 6.1 mils/mo. Only about 15% of the condensate was collected, while the remainder flashed off during feed addition.

[®] du Pont Co., Wilmington, Delaware

Radioactive Demonstration

Phosphate Glass Run PG-6 - J. L. McElroy, J. N. Hartley,
and M. R. Schwab

During Run PG-6, waste equivalent to 0.7 tonnes of 20,000 MWd/tonne power reactor fuel was processed to produce approximately 174 kg of glass product. The 62 liter of radioactive glass produced a self-generating heat density of approximately 131 W/liter (8100 W total).

A total of 616 liters of sulfate-free PW-1 type feed (889 liters/tonne) containing 2,560,000 Ci of radioactivity, including approximately 8,000 Ci of radoruthenium, were fed to the denitrator in 50 hr at an average feed rate of 12.3 liters/hr. Table 5.1 lists pertinent data for Run PG-6 as well as for all WSEP runs to date.

The sulfate-free PW-1 feed used in PG-6 was made by using radiocerium solution from B-plant as a source of radioactivity. Without Purex 1WW in the feed, no foaming problems occurred in the denitrator-evaporator as had happened in previous runs. The fact that Antifoam B* was not added further supports the previous evidence that small (<100 ppm) quantities of organics (probably dibutyl phosphate, DBP, from degraded tributyl phosphate, TBP) in the Purex 1WW probably caused foaming in previous runs. The sodium and phosphate concentrations in the PW-1 feed were 2.6 and 6.23M, respectively (at 378 liters/tonne).

Several plugging problems occurred throughout the run, but did not cause any delays. Dip tube and airlift line plugging occurred frequently, and required constant observations and frequent correction to avoid any interruption in the process. The main plugs that occurred during the run were:

* Dow Corning Antifoam B

TABLE 5.1. WSEP Interim Run Summary Sheet No. 10 (8/12/68)

OPERATIONAL DATA

Solidifier/WSEP #	1/1	1/2	2/3	3/4	2/5	4/6	5/7	6/8(h)	1/9	2/10	3/11	4/12	5/13(r)	6/14	3/15
Solidifier	PC	SS	PC	PC	SS	PC	PC	PC	PG	PG	PG	PG	PG	PG	SS
Date	11/66	1/67	2/67	2/67	4/67	6/67	7/67	9/67	11/67	12/67	1/68	3/68	4/68	5/68	7/68
Operating Mode	A	A	B	A-Modified	B-Modified	C	C	A	Part 1	Part 2	A	A	Part 1	Part 2	A
Feed Type	PW-2	PW-1	PW-2	PW-2	PW-1	PW-1(e)	PW-1(g)	PW-2	PW-2(j)(k)	PW-2(m)	PW-2(o)	PW-2(q)	PW-1(r)	PW-1	PW-2
Feed #/tonne	400	455	400	420	455	370	410	390	478	565	586	458	540	666	590
Pot Dia. in.	12	12	12	8	8	12	8	8	12	8	8	8	12	8	8
Pot Material	304-L	304-L	304-L	304-L	304-L	304-L	304-L	304-L	Mild Stl.	304-L	Mild Stl.	Mild Stl.	Mild Stl.	304-L	304-L
Total Run Feed Time, Hrs.	43.7	37	42	31	57	42	28	43(i)	61.4	31	48.5	63.1	77	50	35.6
Startup Time (Concentration Period), Hrs.	----	----	----	----	----	----	----	----	9	5	10	13	12	6	----
Steady State Operation, Hrs.	----	----	----	----	----	----	----	----	52.4	26	32	16	65	47	----
Shutdown Time (Dilution Period), Hrs.	----	----	----	----	----	----	----	----	11.7	15.0	14	14.4	13	2	----
Drying + Cooling Time, Hrs.	27	12.8	28	15.5	6.6	33	11	8	----	----	----	----	----	----	----
Avg. Feed Rate, #/hr.	19.5	16.3	22.4	18	13.1	25	18.4	10.7	15.4	13.6	18	11.1	9.4	11.1	10.1
Avg. Feed Rate During Startup, #/hr.	----	----	----	----	----	----	----	----	21.6	19	22	16.4	20	25	----
Avg. Feed Rate During Steady State Operation, #/hr.	----	----	----	----	----	----	----	----	14.3	12.6	14	11.4	7.5	10	----
Total Feed, liters	850	603	940	553	789	1060(f)	515	460	944(l)	422(l)	873(n)	697(p)	725	420	362
Total Feed, tonne equiv.	2.2	1.3	2.3	1.4	1.7	3	1.25	1.2	1.97	0.75	1.5	1.52	1.3	0.63	0.61
Total Feed to Melter, liters	----	----	----	----	----	----	----	----	691	394	520	317	669	271	----
Feed: Solid Vol Ratio	8	16	9	9	16	9.1	9.4	7.7	10.4	13	9.2	8.5	6.1	10.5	9.0

PROBLEMS

None	1. Scaling on Calciner Walls 2. High Filter Δ P 3. Stalagmite in Pot 4. High Air In-Leakage	1. Solids in Feed 2. Sulfate Volatilization During Pot Drying.	None	1. High Filter Δ P 2. Feed Flow Fluctuations	1. Foaming in Pot Calciner 2. Evaporator Diptube Plugging	1. Foaming in Pot Calciner 2. Evaporator Diptube Plugging	1. Feed Pump Failures Resulted in 3-Part Run	1. ← Foaming in Denitrator-Evaporator 2. ← WSEP Evaporator Diptube Plugging 3. Feed Pump Failure Resulted in Two Part Run	3. Stalagmite Formation 4. Glass Clinker in Sample Chamber	3. Foaming Suppressed by Addition of Antifoam-B.	1. Melter Feed Line Plug Resulted in Two Part Run. 2. Plugging Problems in Airlift Pot Recirculation Line 3. Den. - Evap. Diptube Plugging	1. Diptube and Airlift Line Plugging Occurred Frequently	1. Solids Feeder Plugged 2. Broken Calciner Filter
------	--	---	------	---	--	--	--	---	---	--	--	--	---

FRACTION OF ACTIVITY DISTRIBUTED: TOTAL LESS Ru(T-Ru) AND Ru

Thru Melter Cond (T-Ru)	----	----	----	----	----	----	----	----	117 (s)	270 (s)	500 (s)	2	3.3	1.9	----
10-3 (Ru)	----	----	----	----	----	----	----	----	183	410	410	12	104	43	----
Thru 1st Cond (T-Ru)	4	0.53	~4.0	2.9	0.18	NA	NA	12	3.2	5.8	NA	35	16	13	110
10-3 (Ru)	140	710	~300	310	740	54	33	110	39	36	NA	22	61	17	870
Thru Evap Cond (T-Ru)	1	50	90	< 1.5	6.3	7.0	0.32	0.1	1.1	< 0.12	0.86	0.31	2.8	1.5	4.0
10-6 (Ru)	74	830	1420	120	4000	1700	310	350	10,340	1000	390	850	2100	1000	2300
Thru Fract Cond (T-Ru)	< 1.6	630	40	3.2(c)	2.4	9.6	25	< 0.1(c)	1790	11	96	2	35	3.3	31
10-9 (Ru)	< 13	400	3950	120	3000	280	17,000	170	16,600	1300	14,000	2000	670	2900	8000
Off-gas to Scrub (T-Ru)	1.6	290	1400	100	< 1	3.1	NA	10	NA	40	NA	4	0.8	NA	22
10-9 (Ru)	880	330	10,600	1700	250	51	NA	70	NA	9000	9300	750	1300	700	24,900
Off-gas to Stack (T-Ru)	NA	< 10	1.1	< 1.7(c)	3.6	< 1.3	< 0.7	0.6(c)	4.1	5.8	0.54	2	1	1	11
10-12 (Ru)	NA	< 1000	< 2400	< 2300	< 880	< 1000	< 1300	NA	NA	NA	NA	NA	NA	NA	NA
Ratio Activity in Fract Cond to 10CFR20 (T-Ru)	NA	140	350	70	1.5	1800	2200	20	17,250	770	1100	400	1800	550	940
(Ru)	< 16	16	6400	350	45	1000	10,000	1600	14,850	10,000	66,000	7800	1800	3200	37,000

FILLED POT DATA

Total Curies in Feed to Pot	85,000	83,000	430,000	375,000	419,000	650,000	1,150,000	1,300,000	316,000	399,000	1,540,000	1,107,000	2,354,000	2,560,000	333,000
Curies of Ru in Feed to Pot	9200	5700	35,000	32,000	5300	47,000	18,000	36,000	13,000	19,600	10,200	30,912	42,200	8000	25,700
Solid in Pot, liters	105	43	105	60	48	110	55	60	105	41.3	34.5	73.3	113.3	62	40
Solid in Pot, kg	151	133	141	86.4	152	166	86	79.5	290	118	98	213	306	174	102
Solid in Pot, #/te	50	32	45	47	32	41	43	50	48	46.4	50.7	61	108	94	65
Heat in Pot, watts	600 (420)(a)(b)	330	1700 (1500)	1770 (1800)	(2100)	(2600)	(4300)	5100 (4850)	1300	(1700)	3300	(4200)	8800	9800 (8100)	1600
Equiv. Aging Time, yrs.	50	40	6	3.7	3.8	4.7	1.5	1.0	5.8	2.3	1.0	1.6	0.5	0.2	3.8
Temp. in Air, °C	140	170	485(d)	410(d)	315(d)	477	725	940	320(d)	415(d)	655(d)	510(d)	730(d)	780(d)	318(d)
Temp. in Water, °C	NA	NA	NA	NA	NA	370	NA	850	140	NA	430	NA	NA	NA	NA
Temp. to Wall, °C	70	30	235	220	65	312	495	700	90	140	255	175	350	380	81
Total Corrosion, mm	None	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Welded Lid	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pot Self Pressurization, psi	< 1	NA	< 1	< 1	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Pot Leakage, atm cc/sec	15 → 7 psi/90 Days	NA	NA	NA	NA	NA	NA	NA	1.1x10 ⁻⁸	2.4x10 ⁻⁸	4.6x10 ⁻⁸	NA	NA	NA	NA
Radiation Reading, R/hr at 6 in.	2.5x10 ⁻⁴	2.5x10 ⁻³	2.4x10 ⁻⁴	3.9x10 ⁻⁴	1.1x10 ⁻⁴	3.8x10 ⁻⁴	4.9x10 ⁻⁴	5.7x10 ⁻⁴	9.5x10 ⁻³	4.8x10 ⁻⁴	4.1x10 ⁻⁴	3.7x10 ⁻⁴	4.7x10 ⁻⁴	NA	NA

*NA-Not Available

- (a) Values in Brackets are from Pot Calorimetry Data
- (b) Calorimetry Value is After 3 Months Decay
- (c) Corrected Because of Foreign Contamination to give DF Approximately Equal to that for Ru
- (d) Values are with Pot Setting in Cool Furnace
- (e) SO₄ and Na were 0.25 and 0.47M Vice Desired Values of 0.10 and 0.14M Respectively
- (f) 60 Liters Expelled from Pot due to Foaming
- (g) SO₄ and Na were 0.35 and 0.64M Vice Desired Values of 0.10 and 0.14M Respectively
- (h) PC-6 was a 3 part run. All DF Data is for 1st Part Only
- (i) Feeding Extended to Obtain a Maximum Fill
- (j) Al Substituted for 37% of the Nominal 0.445M Fe
- (k) Fe Substituted for 90% of the Ru
- (l) 12% of Total Feed to Denitrator lost due to Foaming
- (m) Al Substituted for 65% of the Nominal 0.445M Fe
- (n) 27% of Total Feed to Denitrator lost due to Foaming
- (o) Al Substituted for 40% of Nominal 0.445M Fe
- (p) 50% of Total Feed to Denitrator lost due to Foaming
- (q) Al Substituted for 59% of Nominal 0.445M Fe
- (r) Rare Earths were 6 Times Nominal 0.12M R. E. U⁺⁶ was 0.05M Versus Desired 0.01M, 0.24M H₃PO₄ Added to Compensate for Excess U⁺⁶
- (s) Activity Includes Feed Losses from Denitrator from Overflow to Melter Condensate Receiver

- The denitrator-evaporator airlift air line plugged twice during startup and once during shutdown.
- The airlift pot-weight factor dip tube plugged shortly after startup, and was not unplugged until after the run.
- The denitrator weight factor and specific-gravity dip tubes were kept clear by hourly steam blowdowns, but would start to plug if more than an hour elapsed between blowdowns.
- The airlift recirculation line required frequent backing out of the plug valve as well as acid flushes to remove solids from this area.

During Run PG-6, 1.7% of the radioactive ruthenium and 1.3% of other radioactive species fed to the denitrator were volatilized or entrained from the denitrator. Likewise, approximately 4.3% of the radioactive ruthenium and 0.2% of all other radioactive species fed to the melter were volatilized or entrained.

The fraction of feed activity in the final condensate stream from the acid fractionator was 2.9×10^{-6} for radioactive ruthenium and 3.3×10^{-9} for all other radioactive species. This represents activity above CFR20 limits* by factors of 3200 and 550, respectively.

With the pot sitting in the cool furnace, the 9800 W of internal heat produced a centerline temperature of 780 °C and a centerline-to-wall temperature difference of 380 °C. These values gave an effective thermal conductivity for the product of 1.0 W/m °C [0.61 Btu/(hr)(ft)(°F)].

* AEC's Rules and Regulations, 10CFR20, Table II, Column II.

SPRAY SOLIDIFICATIONFlowsheets for Processing PW-4m in the Spray Solidifier -

J. E. Mendel

The third high-level radioactive waste composition to be included in the WSEP Program will be PW-4m. It has been defined as the waste resulting from the "clean" reprocessing of fuel with an exposure of 45,000 MWd/tonne at 30 MW/tonne. The principal differences between PW-4m and the two other WSEP waste compositions are:

	<u>PW-4m</u>		<u>PW-1</u>		<u>PW-2</u>	
	<u>wt%</u>	<u>mole %</u>	<u>wt%</u>	<u>mole %</u>	<u>wt%</u>	<u>mole %</u>
FP oxides	85.7	76.7	37.2	36.2	26.4	16.4
Na ₂ O	7.6	18.3	4.8	8.0	15.1	24.4
Fe ₂ O ₃	3.1	2.8	54.8	53.8	18.6	11.7
SO ₃	---	---			36.6	45.7

The composition of the simulated PW-4m being used in laboratory testing is given in Table 5.2.

Indications are that PW-4m can be processed by the phosphate glass process without great difficulty. Melts with M⁺/P ratios of 0.9 and 1.0 were smooth and easy to pour at 775 °C (M⁺ included 1.5M Na which was added to lower the viscosity and melting point).

The optimum combination of additive fluxing and modifying agents was sought for processing PW-4m in the spray solidifier process. The goal was to find compositions which form calcines that are not "sticky" at 700 °C and, thus, will not build deposits on the walls of the spray calciner; yet which melt at approximately 900 °C. Somewhat higher melting points can be accommodated in the spray solidifier process, but experience indicates that a melting point of 900 °C or below permits optimum operation. Nearly 100 PW-4m melts have been prepared.

TABLE 5.2. *Composition of PW-4m at 100 gal/tonne^(a)*

<u>Inert Chemicals</u>	<u>Molarity</u>
Fe	0.05
Cr	0.012
Ni	0.008
Al	0.001
Na	0.322 ^(b)
U	0.010
NO ₃	6.50
P	0.003
<u>Fission Products</u>	
Rb (K)	0.0141
Cs (K)	0.0783
Sr	0.0356
Ba	0.0410
RE	0.274
Zr	0.143
Mo	0.130
Tc (Mo)	0.031
Ru (Mn)	0.0816
Rh (Co)	0.0133
Pd (Ni)	0.0425
Ag (Cu)	0.0016
Cd (Cu)	0.0025
Te (SO ₄)	0.0144

(a) *Substitutions, as shown in parentheses, are being made for some of the fission product elements.*

(b) *Nominal Na is 0.10M. Excess is due to addition of Mo as Na₂MoO₄.*

Phosphorus, boron, and silicon--singly and in combination--were used as fluxing agents. Lithium, sodium, iron, aluminum, zinc, and molybdenum in various combinations were tested to modify the melt characteristics.

The relatively high rare earth concentration (35 wt% as oxide) tends to raise the melting point and increase nonhomogeneity in PW-4m melts. To make phosphate-based PW-4m melts that could be processed in the spray solidifier, it was necessary to add sodium and iron until the final melt resembled PW-1 spray solidifier product.

Investigation of some borosilicate melts was also started. Although further optimization is required, several PW-4m borosilicate melts have been prepared which have "stick" temperatures above 700 °C and which melt in the range of 925 to 975 °C.

Corrosion from Fluoride Bearing Feed Solutions - R. F. Maness

A fluoride-bearing PW-2 solution (0.03M fluoride) was concentrated to a boiling point of 128 °C and fed semicontinuously to a 304L SS pot held at 700 °C. The calcine produced was then fed semicontinuously to a 304L melter pot held at 850 °C. The off-gas, during both operations, was pulled through a 304L sintered stainless steel filter held at 300 °C. The total exposure time was 33 hr. The filter evidenced a weight loss of 0.5%, but no attack was apparent from microscopic examination. Platinum specimens exposed to the melt and to the vapor phase during the melting operation were unattacked. The fluoride content of the condensate collected during the calcination operation was estimated to be $5 \times 10^{-3} \text{M}$.

Platinum Melter Corrosion - R. F. Maness

The small electric furnace used to operate the spray-calciner melter-freeze valve has been subject to frequent failure. Burners operating on a propane-air mixture could be used to operate this valve; therefore, experiments were made to determine if platinum embrittlement occurred under these operating conditions. Aliquots of PW-2 melt were placed periodically on a section of 10 mil thick platinum

sheet heated to about 1000 °C with a torch burning a propane-air mixture. The specimen was allowed to cool to room temperature at 30 min intervals and was then flexed to determine ductility. After 36 cycles, the specimen was cleaned and examined microscopically. One area of the specimen appeared to contain surface cracks. However, examination of this area after sectioning, polishing, and etching revealed no cracking or other preferential attack. With a Meeker burner used rather than a torch, the experiment was continued on the remaining portion of the specimen. The specimen fractured after two additional cycles. An experiment similar to that described above was made by using, in addition to a propane-air burner, a reducing flame from a small torch burning propane with insufficient oxygen. The platinum specimen fractured after 10 cycles.

Ruthenium Volatility Studies - D. W. Brown

The effect of hypophosphorous acid, H_3PO_2 , on ruthenium volatility in the PW-1 spray solidification process was studied on a laboratory scale. Data indicate that 1M hypophosphorous acid lowers the ruthenium volatility to below the detectable limits of the analysis--in this case, less than 3% ruthenium.

Development Spray Calciner Run DSC-18 - J. D. Moore

A spray solidification run (DSC-18) was made with the cold semiworks spray solidifier to fill a special test pot with simulated PW-2 melt. The pot will be used for studies of gas evolution from storage pots with molten cores. The 12 in. diam unheated pot was filled uniformly with about 40 liters of melt by batch-dumping from the melter freeze valve and weir during a 29.5 hr run.

The spray calciner, operating at 700 °C, was fed with simulated 137 gal/tonne high sulfate (PW-2) waste at an average rate of 11 liters/hr (0.5 tonne/day). This rate reached the

capacity of the melter operating at a furnace temperature of 925 °C. Air atomization of feed through an internal-mix nozzle produced a fine homogeneous powder feed to the melter. Operation of the system was satisfactory.

After the run, the calciner internal walls were coated with an apparently steady-state layer of powder between 1/8 and 3/16 in. thick, and the calciner filters were covered with a 1/16 in. coating. The filter pressure drop increased gradually from 1 1/4 to 5 in. of water with a filter blow-back pressure of 35 psig. For the last quarter of the run, the blow-back pressure was increased to 40 psig, and the filter pressure drop gradually decreased from 5 to 4 1/4 in. of water at the end of the run, thereby providing reasonably good control of filter pressure drop. The particulate decontamination factor (DF) across the filters varied from a low of 150, 2 hr after the feed was on, to 2600 after 12 hr when a steady-state coating of powder had formed on the clean filters. The DF remained at 2600 to the end of the run.

A new solids feeder was successfully tested for addition of sodium metaphosphate flux to the melter. The feeder consists of two 1-1/2 in. ball valves spaced about 18 in. apart in a pipe connected to a diagonal nozzle in the calciner cone. The valve handles are coupled together to an air-driven piston which opens and closes the valves. When one valve is open, the other is closed, thus maintaining a seal of the calciner at all times. Sodium metaphosphate chips are fed into the pipe from a vibrating auger feeder. Once every 5 min, a timer actuates the piston which simultaneously opens the bottom valve, closes the top valve, and dumps the accumulated chips into the melter. Total flux addition with this equipment was within 2% of the desired amount. No solids plugging was encountered.

Melt was drained in 4 to 5 liter batches at a batch rate of about 10 liters/hr without the formation of stalagmites in the product pot. When batch-dumping the melt through the melter freeze valve, foamy melt discharging from the freeze valve bridged across to the freeze valve furnace liner, plugged off the outlet, and burned out the furnace. The foamy melt was caused by evolution of oxides of sulfur when the melt temperature was raised excessively to 910 °C to drain through the freeze valve. The tip of the freeze valve was too cold to completely melt the plug so the tip was probed with a rod to start the melt flow. Subsequent batch dumping of melt through the weir also indicated that the weir tip was not adequately heated. Heating the weir tip with an oxyacetylene torch or manually rodding the weir tube initiated draining of the melter. Modifications to the freeze valve and weir tube assemblies will be made accordingly.

All of the ruthenium in the feed to the calciner was volatilized to the off-gas condensate.

Radioactive Demonstration

Run SS-3 - J. L. McElroy, W. R. Bond, M. R. Schwab,
and J. N. Hartley

The fifteenth WSEP pilot plant demonstration of the solidification of high-level radioactive wastes was completed with the third spray solidification Run SS-3. During Run SS-3, 362 liters of high-sulfate waste (at 590 liters/tonne) containing 333,000 Ci of radioactivity including 26,000 Ci radio-ruthenium were fed to the spray solidifier in 35.6 hr at an average rate of 10.1 liters/hr (Table 5.1, page 5.3 for SS-3 data). A total of 40 liters of ceramic solid was produced with an internal heat generation rate of 40 W/liter corresponding to 38% of thermal maximum for the product. The total of 1600 W of fission product heat and the centerline-to-wall

temperature difference of 81 °C gave an effective thermal conductivity of 1.28 W/m °C [0.74 Btu/(hr)(ft)(°F)] for the ceramic product.

An internal-mix spray nozzle was used to spray the PW-2 type feed [additives of 1.61M H_3PO_4 , 1.17 LiOH, 0.6M $\text{Ca}(\text{NO}_3)_2$, 0.25 $\text{Al}(\text{NO}_3)_3$ at 378 liter/tonne] into the solidifier. Feed atomization with unheated air prevented nozzle plugging as in the previous run, SS-2. In addition to the feed additives, solid sodium hexametaphosphate was added to the melter batch-wise to maintain an additive composition of 1.27M NaPO_3 at 378 liter/tonne feed concentration, to reduce the melting point to ~700 °C. A gravity flow, mechanically vibrated solids feeder was used to add the solid flux (sodium metaphosphate). A heated air, purge line was added to the solids feeder during the run to minimize condensation of solidifier vapors and subsequent plugging in the solids feed line where it enters the cone of the calciner.

Run SS-3 was plagued by equipment failures, and after several attempts to maintain a continuous operation, the run was terminated due to equipment failures. Early in the run before the feed was started to the solidifier, the freeze valve and weir heaters failed, the feed tank agitator failed, and the feed pump began leaking, in that order. Without adequate feed agitation, the heavy solids in the feed would have settled, thus the run was shut down until the agitator and feed pump were repaired. The run was started without the weir and freeze valve heaters by using a propane torch on the freeze valve to batch drain the product from the melter to the pot receiver. After about 8 hr of feeding, the melter solids addition line plugged at the calciner and the run was shut down.

Because of a potential for embrittlement in the platinum freeze valve in the presence of a propane flame, the original 12 in. diam pot receiver was replaced with an 8 in. pot, thus shortening the run and requiring only half the original product planned.

During the second part of Run SS-3 (second pot receiver), the feed was shut off intermittently (after 1 to 8 hr of operation) to periodically flush out plugs of entrained calcine in the vent and drain lines from the solidifier condenser (E-111) with nitric acid. Significant quantities of solids in the off-gases (11% of the total calcine produced) indicated a failed solidifier off-gas filter. Failure of one of the 316L sintered stainless steel filters has since been visually verified, and investigations to determine the cause of failure are presently underway. Another shutdown was necessary to repair the solidifier vibrator (used to maintain the heated walls of the calcine free from calcine deposition). The run was terminated after about 32 hr of total feed-on time (including about 8 hr with the first pot receiver) when the solidifier off-gas line plugged at the inlet to the solidifier condenser and could not be cleared while "on-line."

Approximately, 87% of the ruthenium in the feed to the solidifier was volatilized. As yet, no quantitative data are available on sulfate losses from the melt. However, on two occasions when the melt temperature exceeded 890 °C, sulfate fumes were observed evolving from the melt as it drained from the melter into the receiver pot.

ADVANCED SOLIDIFICATION TECHNIQUES

Feasibility of Vacuum Batch Transfer of Melts from Simulated Purex-Type Waste - J. N. Hartley

This investigation is being made to provide information for assessing the feasibility of a promising new concept of a

batch melting solidifier for treating high level radioactive waste. The batch solidifier consists of an evaporator chamber above a batch melter as an integral unit. The waste is concentrated and melted batch-wise, then transferred to the final storage container. Vacuum batch transfer of melt through a heated tube to a final receiving vessel was tested.

Three successful transfers were made by using phosphate glass melts of simulated wastes. A fourth transfer was started by using a sulfate containing ceramic melt from the spray solidifier but due to an equipment failure, was discontinued.

The major features of the vacuum transfer equipment are shown in Figure 5.1. Glass or ceramic is melted by an induction furnace. The molten glass or ceramic is then transferred through a heated tube to a final mild steel receiving pot.

Run VT-1

The first transfer was made by using a sulfate-free (PW-1) phosphate glass melt. A foamup of melt in the receiving pot after transferring 30 kg of melt caused the transfer to be stopped. The cause of the foamup was an excessive vacuum (29 in. mercury) in the receiving pot. The 13.3 liters of glass product was transferred to the mild steel receiving vessel in 0.65 min at an average rate of 20.4 liters/min.

The transfer tube was heated to 850 °C before the transfer was started, and the melt was held at 900 °C. After the transfer, the unheated receiving pot-wall temperature had increased to 350 °C.

Run VT-2

The second vacuum transfer was successfully completed by transferring 44 kg of PW-1 phosphate glass melt. Thirty liters of glass were transferred in 2.4 min at an average rate of

FIGURE 5.1. Vacuum Transfer Equipment Arrangement

12.5 liters/min. Fifteen inches of mercury vacuum were used instead of the 29 in. used in the previous run. No transfer problems occurred with the transfer tube heated to 1000 °C, although this was 150 °C above operating conditions and resulted in a weakening of the 304L transfer tube. The melt was at 935 °C before the transfer started; after the transfer was complete, the receiving pot wall temperature had increased to 445 °C.

Run VT-3

The third transfer was successfully completed by using a phosphate glass melt from a sulfate containing (PW-2) feed. During the transfer, 30 kg of melt were transferred to a mild steel receiving pot. The 13 liters of glass was transferred in 1.2 min at an average rate of 10.8 liters/min. The transfer rate was about 1/2 the rate in previous transfers because of a heater failure on the tip of the transfer tube where the average temperature was only 200 °C as compared to ~850 °C for the rest of the tube. This caused some buildup of glass in the pipe, but no other problems occurred. The melt was at 925 °C when the transfer started, with the vacuum at 15 in. of mercury.

Run VT-4

The fourth transfer was started by using a sulfate-containing ceramic melt from the spray solidifier, but was discontinued due to equipment failure. More experimental work is planned with this type of melt.

DEVELOPMENT STUDIES ON SOLIDIFICATION PROCESSES

Computer Simulation - C. R. Cole, P. C. Owzarski,
and J. L. McElroy

The initial simulation of the WSEP off-gas train and pot calciner has been completed. The various elements included in the simulation are illustrated in Figure 5.2. The present

FIGURE 5.2. Units of the WSEP System Which Are Included in the Present Computer Simulation Model

simulation of the WSEP Plant is capable of any of the three modes of operation (A, B, or C). The present WSEP simulation can be preprogrammed to operate in any desired manner or it can be operated manually as the actual plant might be. The model is now ready to be used to predict the results of any method of operation of the pot calciner.

Heat and material balances are performed in each of the units simulated. Nitric acid separation, where it occurs, is computed from water-nitric acid equilibrium data and enthalpy data on liquid and vapor mixtures of water-nitric acid. Solids entrainment in the units, where it occurs, is assumed to be proportional to a power function of vaporization rate out of that unit and to the solids concentration in that unit. Ruthenium volatility is computed as a power function of the nitric acid concentration in the liquid and is proportional to the ruthenium concentration of the liquid.

The water-nitric acid separation characteristics of the model were verified by comparisons with data from a nonradioactive WSEP run where acid feed of known concentration was fed to the evaporator, and the separation characteristics of the evaporator and fractionator were recorded. Figure 5.3 shows the comparison between the model and the actual WSEP data for this run.

Data from WSEP Run PC-2 were used to adjust the parameters of the model in order to match the observed data on solids entrainment and ruthenium volatility. Figure 5.4 shows a comparison between the actual WSEP data on solids entrainment and ruthenium volatility for Run PC-2 and the simulation's match of this data.

FIGURE 5.3. Comparison Between the Computer Acid Run Simulation and Actual WSEP Data

FIGURE 5.4. Comparison Between Actual WSEP Data for Run PC-2 for Solids and Ruthenium Radioactivity Levels and the Levels Computed by the Model.

HEAT TRANSFER STUDIES

Pot Furnace Cooling Capacities - J. D. Kaser

To assist in planning future WSEP runs at higher levels of radioactivity, the steady-state cooling capacities of the induction heated (5A) pot furnace, and the resistance heated (6A) pot furnace were calculated as functions of pot wall temperature. One of these two furnaces contains the 8-ft-long product-receiving pots during waste solidification in WSEP.

Heat balances between the pot wall and the various surfaces within the furnaces were made by assuming a uniform temperature at each surface and neglecting the heat losses at the top and bottom of the furnaces. Accounting for the effect of two cooling air streams in the furnace (indirect air blasting at the outside of the susceptor wall and direct forced air within the susceptor) resulted in a system of independent nonlinear equations solved by an iterative technique and programmed for the computer.

The results of these calculations are plotted in Figures 5.5 and 5.6. Curves showing calculated wall temperatures for pots hanging in air are included for convenient comparison. Table 5.3 compares relevant WSEP run data with calculated temperatures read from Figure 5.5. When the uncertainties are considered in temperature and heat generation rate measurements and in the values of the heat transfer coefficients used in the calculations, the table shows fair agreement between calculated and observed values. The centerline temperatures were estimated from the calculated pot wall temperatures by using a thermal conductivity value of 0.43 Btu/(ft)(hr)(°F) obtained from preliminary phosphate-glass process run data.

Corrosion experience to date indicates that a stainless steel pot wall probably should not exceed approximately 850 °C during processing and that a carbon steel pot should not exceed

FIGURE 5.5. Steady-State Pot Temperatures During Cooling in the WSEP 5A Furnace

FIGURE 5.6. *Steady-State Pot Temperatures During Cooling in the WSEP 6A Furnace*

TABLE 5.3. *Comparison of Measured and Calculated Pot Temperatures*

WSEP Run No.	Heat Gen. Rate, kW	Wall Temperature, °C		Cooling Condition
		Observed	Calculated	
PC-4	2.6	165	160	Hanging in air outside furnace
PC-5	4.3	230	240	↓
PC-6	5.1	240	250	
PG-5	8.8	345	400	
PG-6	8.1	420	420	In furnace. Outer cooling air only

700 °C. In these cases, the calculations indicate that heat generation limits for a stainless pot is 33 kW in 5A furnace and about 28 kW in 6A furnace. The corresponding limits for a mild steel pot would be 25 and 20 kW. The highest potential heat generation rate now planned for WSEP is 25 kW, using a stainless steel pot in 5A furnace which is within these calculated furnace limits. Since the melting points of the fusible solid wastes range between 700 and 850 °C, the pot cores will remain molten during both furnace cooling and air storage at heat generation rates exceeding 5 to 10 kW.

WASTE FROM CHEMICAL PROCESSING

Waste from Fast Reactor Fuels - E. A. Coppinger

The radioactive waste resulting from chemical processing of fast reactor fuels will be different than wastes from processing of thermal reactor fuels. The higher fuel exposure and specific power in fast reactors will result in much higher fission product contents and heat generation rates in the waste. The fission product spectrum and amounts of neptunium and transplutonium isotopes will also be different in the waste because of the different irradiation conditions, higher Pu/U ratio in

the fuel, and higher neutron energy in a fast reactor. The reprocessing scheme for fast reactor fuels may also result in differences in the chemical composition of the waste. A very short study was undertaken to provide a limited assessment of the possible aqueous waste resulting from processing of fast reactor fuels.

Approximate heat generation rates, fission product concentrations, and changes in fission product spectrum in fast reactor fuel have been determined and are compiled in Tables 4 through 7. The amounts of neptunium and transplutonium isotopes in a conceptual fast reactor design, obtained from available calculations, are also listed.

Conceptual designs usually divide the fast reactor into three major zones, i.e., the core, axial blanket, and radial blanket. The axial blanket fuel and core fuel are in the same fuel pin in several design concepts. In such cases, the irradiated fuel from the core and axial blanket will be processed together and the waste will contain a blend of radioisotopes and stable fission products from these two fuels. Since the burnup of the axial blanket is typically less than 10,000 MWd/tonne, the waste from the blended fuel will be fairly similar to the waste from thermal reactor fuel in terms of total fission product content and heat generation rates.

The relative amounts of ^{106}Ru and the potential by-products, palladium and rhodium, will be significantly increased in wastes from processing of fast reactor fuel. The relative amount of ^{90}Sr will be decreased. The relative amounts of cadmium, tin, antimony, and tellurium will be increased although the total amount of each will still be small. Some of these latter fission products, notably antimony and tellurium tend, like ruthenium to be volatile at high temperatures and could complicate waste solidification processes.

TABLE 5.4. *Approximate Fission Product Content in Irradiated Fuel(a)*

Element	Fission Product Content, g/tonne U + Pu	
	Fast Reactor Core Fuel ^(b)	PWR Fuel ^(c)
Se	166	46
Rb	609	318
Sr	1,310	837
Y	677	432
Zr	7,584	3418
Nb	60	8.8
Mo	8,800	3228
Tc	2,384	789
Ru	8,535	2068
Rh	2,490	400
Pd	6,660	970
Ag	756	44
Cd	440	52
In	42	1.6
Sn	436	48
Sb	160	15
Te	1,666	446
Cs	10,830	2567
Ba	3,540	1350
La	3,058	1160
Ce	7,110	2408
Pr	3,125	1072
Nd	10,168	3638
Pm	1,015	124
Sm	2,470	686
Eu	407	157
Gd	323	64
Tb	30	1.4
Dy	20	0.46

(a) Does not include gases or elements present in quantities less than 10 g/tonne

(b) 100,000 MWd/tonne at 200 MW/tonne, 180 day cooling

(c) 30,000 MWd/tonne at 32.5 MW/tonne, 180 day cooling

TABLE 5.5. *Approximate Amounts of Various Fission Products After 90 Days Cooling*

Nuclide	Fast Reactor Core Fuel ^(a)		PWR Fuel ^(b)	
	kCi/tonne	W/tonne	kCi/tonne	W/tonne
⁸⁹ Sr	816	2,806	195.8	673
⁹⁰ Sr(c)	111.7	754	72	486
⁹¹ Y	1410	5,180	312.3	1147
⁹⁵ Zr	3160	15,790	476	2378
⁹⁵ Nb	5220	25,100	783	3765
¹⁰³ Ru	1920	6,816	225.7	800
¹⁰⁶ Ru(c)	3800	36,110	345.4	3282
^{115m} Cd	31.4	115	0.1	0.4
^{123m} Sn	99	330	0.4	1.3
¹²⁶ Sn	0.004	--	--	--
¹²⁵ Sb	73.2	238	7.2	23.4
^{125m} Te	15	13	1.5	1.3
^{127m} Te(c)	57	108	2.7	5.1
^{129m} Te(c)	72	333	13.8	63.8
¹³⁴ Cs(d)	220	2,230	144.4	1464
Cs- ¹³⁷ Ba(c)	337	1,630	98.5	476
¹⁴¹ Ce	1230	1,820	188.6	279
Ce- ¹⁴⁴ Pr(c)	3790	30,926	787	6422
¹⁴⁷ Pm	1000	360	122.8	44
¹⁵¹ Sm	14.6	1.7	0.35	0.04
¹⁵⁴ Eu(d)	3	26.8	4.3	38.5
¹⁵⁵ Eu	144	130	4.5	4.1

(a) 100,000 MWd/tonne at 200 MW/tonne

(b) 30,000 MWd/tonne at 32.5 MW/tonne

(c) Watts include short-lived daughter

(d) Not fission products. Produced from neutron capture by ¹³³Cs and ¹⁵³Eu.

TABLE 5.6. *Approximate Amounts of Various Fission Products After 180 Days Cooling*

Nuclide	Fast Reactor Core Fuel ^(a)		PWR Fuel ^(b)	
	kCi/tonne	W/tonne	kCi/tonne	W/tonne
⁸⁹ Sr	237	815	56.8	195
⁹⁰ Sr ^(c)	111	749	71.6	483
⁹¹ Y	490	1,800	108.4	398
⁹⁵ Zr	1210	6,046	182.3	911
⁹⁵ Nb	2346	11,280	353	1697
¹⁰³ Ru	404	1,434	47.5	168
¹⁰⁶ Ru ^(c)	3200	30,400	291.2	2767
^{115m} Cd	7.3	27	0.02	0.07
^{123m} Sn	61	203	0.26	0.86
¹²⁶ Sn	0.004	--	--	--
¹²⁵ Sb	68.7	223	6.8	22
^{125m} Te	14.9	13	1.5	1.3
^{127m} Te ^(c)	31.6	60	1.5	2.9
^{129m} Te ^(c)	11.4	53	2.1	9.5
¹³⁴ Cs ^(d)	200	2,027	133.2	1350
Cs- ¹³⁷ Ba ^(c)	335	1,620	97.9	473
¹⁴¹ Ce	180	266	27.6	41
Ce- ¹⁴⁴ Pr ^(c)	3050	24,888	632.4	5160
¹⁴⁷ Pm	950	342	115	41
¹⁵¹ Sm	14.6	1.7	0.35	0.04
¹⁵⁴ Eu ^(d)	3	26.8	4.3	38.5
¹⁵⁵ Eu	131	118	4.1	3.7

(a) 100,000 MWd/tonne at 200 MW/tonne

(b) 30,000 MWd/tonne at 32.5 MW/tonne

(c) Watts include short-lived daughter

(d) Not fission products. Produced from neutron capture by ¹³³Cs and ¹⁵³Eu.

TABLE 5.7. *Neptunium and Transplutonium Isotope Content in Fast Reactor Core Fuel (a)*

Nuclide	Amount, g/tonne U + Pu		
	At Discharge	90 Day Cool	180 Day Cool
²³⁷ Np	332.6	332.6	332.6
²⁴¹ Am	1165.1	1297.7 (b)	1428.4 (b)
²⁴² Am	56.5	56.5	56.5
²⁴³ Am	<u>337.2</u>	<u>337.2</u>	<u>337.2</u>
Total Am	1558.8	1691.4	1822.1
²⁴² Cm (c)	193.9	132.3	90.1
²⁴³ Cm	0.4	0.4	0.4
²⁴⁴ Cm	<u>5.3</u>	<u>5.2</u>	<u>5.2</u>
Total Cm	199.6	137.9	95.7

(a) Based on equilibrium core exposure of 110,000 MWd/tonne at a specific power of 56.5 MW/tonne

(b) The increase in ²⁴¹Am after discharge results from beta decay of ²⁴¹Pu. When plutonium is removed by processing before cooling for 90 or 180 days, the increase in ²⁴¹Am content in process solutions stops.

(c) Any ²⁴²Cm left in the waste stream after processing decays to ²³⁸Pu.

The chemical composition of the waste is difficult to specify quantitatively because of uncertainties concerning fuel composition and the fuel processing scheme. On the basis of a Purex-type process, the aqueous waste could range from relatively "clean" solutions of fission products to solutions containing moderate to high concentrations of sodium, iron, and sulfate. The waste generated during processing of carbide fuels is a possible exception. If carbide fuels are dissolved directly in acid, the waste could contain soluble organic compounds that could complicate waste solidification processes.

LOW-LEVEL WASTE TREATMENTDistillation Demonstration Unit (DDU) - S. J. Dougherty

Assembly of the DDU has been essentially completed, and tests of the apparatus have been conducted to determine any necessary equipment modifications or additions. These tests have been conducted with boiling water at 20 and 50 mm Hg pressure and with a boiling acid mixture at 50 mm Hg. The approximate analysis of the acid mixture was

HNO_3 : 1.0M
 H_2SO_4 : 0.5M
 H_3PO_4 : 0.19M

The results of these runs have clearly indicated the capabilities of the equipment. The results show that:

- The differential pressure (D/P) cells are an entirely unsatisfactory method for liquid-level measurement at the operating pressures of 10 to 100 mm Hg.
- Better contact is required between the heaters and evaporator tank.
- When D/P cell and heater modifications are made, the equipment will perform very well.

Although D/P cells are generally considered to be a satisfactory system for liquid-level measurement at atmospheric pressure, they are unsuitable for use in partially evacuated systems, primarily because of the admission of excessive air to the evacuated system. Because of the unsuitability of the D/P cells in measuring liquid levels, alternate methods of liquid level measurement will be used.

The poor heat transfer between the heaters and evaporator tank will be solved by placing a heat-conducting substance between the heaters and tank.

EVALUATION OF SOLIDIFIED WASTE PRODUCTS

Nonradioactive Lab Studies

Study of Thermal History of Melts - D. W. Brown

Sixteen phosphate-glass melts were prepared from a solution of PW-2 waste such that each melt had a different thermal history. Characterization tests have been run on these melts, and Table 5.8 summarizes the results. The greatest factor affecting the properties of the melt was the storage temperature. Samples stored at 600 °C were much more leachable than those stored at 100 °C. Porosity correlates very well to solubility within a storage temperature bracket.

TABLE 5.8. Summary of Thermal History on Melts and Data

Beaker No.	M.T., (a) °C	T.A.M.T., (b) hr	C.R., (c) degrees/hr	S.T., (d) °C	Surface Appearance	Drip Pt., °C	72 hr Leachability Loss, %	Density, g/cm ³	Porosity of Sample, %
2	1200	1	100	600	dull very few holes	855	41.4	2.843	1.3
3	1200	25	100	600	dull very few holes	865	35.3	2.874	2.06
10	1200	1	30	600	glossy no holes	872	1.9	2.882	0
15	1200	25	30	600	glossy cracked	895	7.4	2.918	0
7	1000	1	30	600	very dull many holes	800	41.4	2.988	2.25
8	1000	1	100	600	very dull many holes	800	33.8	2.981	3.24
9	1000	25	30	600	very dull no holes	795	39.3	2.977	2.4
16	1000	25	100	600	dull pin holes	845	35.9	2.926	2.31
12	1200	1	30	100	glossy smooth	875	1.2	2.906	0
13	1200	25	30	100	glossy smooth	875	1.6	2.897	0
14	1200	1	100	100	glossy smooth	870	1.4	2.916	0
17	1200	25	100	100	glossy cracked	870	2.2	2.909	0
4	1000	1	30	100	dull no holes	845	5.5	2.944	2.93
5	1000	1	100	100	glossy cracked	855	4.5	2.890	~1.0
6	1000	25	30	100	glossy smooth	825	5.1	2.814	2.81
11	1000	25	100	100	glossy smooth	915	1.5	2.897	0

(a) M.T. - melt temperature

(b) T.A.M.T. - time at melt temperature

(c) C.R. - cooling rate

(d) S.T. - storage temperature

Study of High Temperature Storage of Solidified Waste Products Containing Sulfate - D. W. Brown

A vessel 12 in. in diameter by 26 in. high has been built and filled 22 1/2 in. deep with spray solidification sulfate-containing (PW-2) melt. This will be heated at the center of the melt. The pressure buildup, caused by decomposing sulfate, and the temperature across the melt will be measured.

Product Characterization Laboratory Studies - M. Mularz

Phosphate-glass samples prepared in the laboratory from sulfate-containing (PW-2) feed are being evaluated to characterize the physical properties of the glasses.

Thermal diffusivity will be determined by a rapid thermal pulse method. Linear coefficients of thermal expansion are being measured, and preliminary data indicates a high linear coefficient. Some samples are oozing a clear liquid (presumed to be a phosphate) at elevated temperatures (~400 to 500 °C).

A prototype drop weight apparatus has been developed for dispersibility measurements. It consists of a 2 in. diam plexiglass tube, a weight with a 1/8 in. diam blunt-end steel tip, and a steel holder for the sample. The tube allows the weight to fall in a straight line with sufficient clearance allowed to prevent drag. When the glass samples break under impact, the particles are collected and classified by weight and size as: greater than sieve size, sieve size, and sub-sieve size. Analysis of the subsieve sizes will be attempted with a Coulter Counter.

Although the thermal shock apparatus has not been developed as yet, it will consist of a tube furnace capable of 600 °C and a water bath into which the sample can fall. Thermal conductivity will be determined on samples that are 1 in. long with a 1 in. diam.

Some of the samples are porous and cracked; therefore, in the evaluation of most parameters, the values obtained must be interpreted carefully.

Product Measurements, Testing and StorageSolids Storage Engineering Testing - V. P. Kelly

A-Cell modifications for SSET that are being completed include:

- Design and fabrication of a push-pull through-the-wall transfer mechanism.
- Completion of gallery work, which includes correcting problems with manipulator monorail installation.
- Installation of special mounting brackets in cell for waste pot and test pod storage.
- Fabrication of an ion chamber shielding cask for in-cell radiation monitoring.

Completion of these project items is expected by fall.

SSET Equipment and Systems Fabrication and Installation.

Fabrication and installation of SSET equipment is proceeding in the effort to activate SSET late this year. Status of the major items of equipment is:

- Construction of 12 standard test pods is complete.
- Construction of four salt storage pods is complete except for assembly. With the exception of assembly of these pods, the full complement of pods for SSETF is now complete.
- Design and fabrication of in-cell transfer hardware and work trays is started.
- Design and installation of items is well along, including (1) emergency in-cell cooling system; (2) emergency test pod cooling system; and (3) auxiliary viewing for both the airlock and A Cell.
- Fabrication of the core driller and a second generation welding turntable is started.

Remote Handling Tests for SSETF. Remote handling design verification testing is in process to verify operability of in-cell components.

Product Characterization - R. J. Thompson

The following product evaluation data on radioactive WSEP pots were collected during the period:

1. PG-4 Pot Calorimetry--4.2 kW $\pm$ 10% (referenced to the shut-down date of PG-4 run).
2. Pot pressures for filled pots in Rack 4A

<u>Run No. for Pot</u>	<u>Pressure</u>
a. PC-1	No reading
b. SS-1	1 psig
c. PC-2	-10 in. Hg
d. PC-3	-28 in. Hg
e. SS-2	1 psig
f. PC-4	12 psig
g. PC-5	-2 psig
h. PC-6	-11 in. Hg
i. PG-1	No reading
j. PG-2	-30 in. Hg
k. PG-3	-24 in. Hg
l. PG-4	No pressure (leaky jumper)

The pots from PG-5, PG-6 and SS-3 were welded.

The pot wall thickness scanner is still giving erratic results, but repairs are underway.

•

•

•

•

•

•

•

•

Distr-1

DISTRIBUTION

No. of
Copies

2	<u>AEC Chicago Patent Group</u> G. H. Lee R. K. Sharp (Richland)
327	<u>AEC Division of Technical Information Extension</u>
1	<u>AEC Division of Technical Information Extension</u> for retransmittal to: US/UK, England A. R. Edwards
3	<u>AEC, RDT Site Representative - PNL</u> P. G. Holsted (2) A. W. Waterhouse
1	<u>Aerojet - General</u> Idaho Falls W. E. Nyer
1	<u>Aerojet - General</u> Sacramento F. J. Climent
7	<u>Argonne National Laboratory</u> C. E. Dickerman S. Fistedis R. O. Ivins P. Lottes R. C. Vogel LMFBR Program Office A. Amorosi L. Baker
61	<u>AEC Library, Washington</u> Advisory Committee on Reactor Safeguards F. R. Fraley (17) Division of Reactor Development and Technology R. S. Brodsky J. W. Crawford R. L. Ednie D. E. Erb

Distr-2

No. of
Copies

A. Giambusso
H. L. Hamester
E. E. Kintner
J. A. Lieberman (5)
R. R. Newton (5)
R. E. Pahler (2)
A. J. Pressesky
I. C. Roberts
M. A. Rosen
E. E. Sinclair
S. A. Szawlewicz
G. W. Wensch
M. J. Whitman

Division of Compliance

L. Kornblith, Jr.

Division of Operational Safety

H. Gilbert

Division of Production

G. B. Pleat

Division of Reactor Standards

M. Bolotsky
G. Burley
E. G. Case (10)
A. B. Holt
R. Waterfield

Division of Compliance, Region IV

J. W. Flora

1 Atomic Energy Commission, RDT Site Representative
 San Diego

R. H. Ball

1 Atomic Energy Commission
 USAEC Technical Representative
 Toronto

H. J. Reynolds

2 Atomics International

H. Morewitz

Liquid Metals Engr Center

R. W. Dickinson

Distr-3

No. of
Copies

1	<u>Auburn University</u> School of Engineering and Engineering Experiment Station G. H. Nix
2	<u>Babcock & Wilcox Co.</u> Virginia R. Wascher Washington L. R. Weissert
6	<u>Battelle Memorial Institute</u> A. R. Duffy D. L. Morrison/D. L. Ritzman (1) S. Paprocki (2) D. N. Sunderman (2)
1	<u>Battelle Memorial Institute</u> Frankfort, Germany G. Leistner/K. J. Kober
1	<u>Bechtel Corporation</u> R. F. Griffin
1	<u>Brookhaven National Laboratory</u> A. W. Castleman
1	<u>Canoga Park Area Office</u> R. L. Morgan
1	<u>Chalk River Nuclear Laboratories</u> Chalk River, Ontario, Canada Station 3 G. Hake
1	<u>Chicago Operations Office</u> Atomic Energy Commission D. M. Gardiner

Distr-4

No. of
Copies

1	<u>Combustion Engineering</u> M. F. Valerino
1	<u>Consolidated Edison Company</u> J. J. Grob
7	<u>Douglas United Nuclear</u> T. W. Ambrose N. R. Miller W. F. Nechodom J. W. Riches D. L. Renberger J. R. Spink Files
2	<u>General Atomic Division (AEC)</u> A. J. Goodjohn
3	<u>General Electric Company,</u> <u>APED San Jose</u> S. Vandenberg G. E. Wade E. Zebroski
4	<u>General Electric Company,</u> <u>San Jose (Trumbull)</u> P. Bray M. Siegler W. A. Sutherland B. Wolfe
1	<u>Harvard Air Cleaning</u> <u>Laboratory</u>
2	<u>Idaho Nuclear Corporation</u> J. A. Buckham C. M. Slansky
2	<u>Idaho Operations Office</u> <u>Atomic Energy Commission</u> D. Williams

Distr-5

No. of
Copies

2	<u>IIT Research Institute</u> E. V. Gallagher T. A. Zaker
2	<u>Atlantic Richfield Hanford Company</u> File G. R. Kiel
1	<u>Los Alamos Scientific Laboratory</u> J. H. Russel
1	<u>MPR Associates, Inc.</u> T. Rockwell III
1	<u>National Bureau of Standards</u> C. Muehlhause
1	<u>Naval Ordinance Laboratory</u> J. Proctor
1	<u>North Carolina State University</u> M. N. Ozisik
1	<u>Nuclear Fuels Services</u> R. P. Wischow
13	<u>Oak Ridge National Laboratory</u> R. E. Adams R. Blanco J. Buchanan W. B. Cottrell (4) D. Ferguson C. E. Miller G. W. Parker L. F. Parsley D. B. Trauger (2)

Distr-6

No. of
Copies

4	<u>Oak Ridge Operations Office</u> <u>Atomic Energy Commission</u> D. Cope (2) E. Delaney W. L. Smalley
1	<u>Oregon State University</u> James G. Knudsen
1	<u>Pacific Gas and Electric Company</u> W. Nutting
14	<u>Phillips Petroleum Company</u> G. O. Bright G. F. Brockett S. Forbes (2) C. Haire G. B. Matheny W. E. Nyer (2) F. Schroeder (3) N. K. Sowards T. R. Wilson (2)
2	<u>Richland Operations Office</u> <u>Atomic Energy Commission</u> C. L. Robinson C. N. Zangar
2	<u>San Francisco Operations Office</u> <u>Atomic Energy Commission</u> Lt. Col. J. B. Radcliffe C. V. Backlund
1	<u>duPont, Aiken</u> A. H. Peters
2	<u>TRW Incorporated</u> <u>TRW Systems Group</u> D. B. Langmuir S. M. Zivi

Distr-7

No. of
Copies

2

University of California, Berkeley
Institute of Engineering Research

H. A. Johnson
V. E. Schrock

1

University of Washington

Wells Moulton

6

Westinghouse Electric Co.
(APD)

E. Beckjord
D. Fletcher
H. Graves
F. M. Heck
R. A. Wieseemann

Westinghouse Electric Corp.
(HTD)

A. Lohmeier

57

Battelle-Northwest

F. W. Albaugh	D. W. Pearce
J. M. Batch (12)	H. N. Pedersen
J. J. Cadwell	L. T. Pedersen
L. A. Carter	D. L. Reid
D. L. Condotta	W. D. Richmond
F. G. Dawson	G. J. Rogers (5)
D. R. de Halas	L. C. Schwendiman
R. F. Foster	D. E. Simpson
J. C. Fox	J. C. Spanner
J. J. Fuquay	E. E. Voiland
W. A. Haney	R. G. Wheeler
H. Harty	N. G. Wittenbrock
J. F. Honstead	D. C. Worlton
R. T. Jaske	Patent Section (2)
R. L. Junkins	Technical Information
H. V. Larson	Files (5)
J. M. Nielson	Technical
R. E. Nightingale	Publications (2)
H. M. Parker	
R. S. Paul	

•

•

•

•

•

•

•

•