ENGINEERING DEVELOPMENT DEPARTMENT**QUARTERLY REPORT****JULY, AUGUST, SEPTEMBER, 1966**

H. HARTY, et al.

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ENGINEERING DEVELOPMENT DEPARTMENT
QUARTERLY REPORT
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CIVILIAN POWER REACTORS

FAST FLUX TEST FACILITY

Heat Transfer Studies

Cermet Fuel Analysis

G. M. Hesson

In an FTR, incidents could occur which could introduce reactivity in the low superprompt region. This reactivity insertion could lead to gross fuel melting unless the power rise is terminated within a small fraction of a second. The principal mechanism for the turnaround of the power excursion for a cermet fuel is the negative reactivity resulting from the thermal expansion of the stainless steel matrix. However, the thermal expansion of the stainless steel matrix is a complicated function of the heat transport within the fuel pin.

An analytic model was developed to describe the heat transport in a cermet fuel and to study the course of events during the short transients following reactivity insertions of 1.0 to 1.5 dollars. This analytic model uses a modified version of the TIGER V transient heat transfer code. Two primary modifications were made to the basic code. The first modification causes the code to calculate a thermal contract coefficient between the PuO_2 particles and the stainless steel matrix from the temperature differential between the two. At temperature differences less than 360 °F (183 °C), a thermal contact coefficient of 500

Btu/hr-ft²-°F was used. Between the temperature differentials of 360 °F (182 °C) and 400 °F (204 °C), the thermal contact coefficient was assumed to increase linearly to 100,000 Btu/hr-ft²-°F. Above a 400 °F differential, the contact coefficient was assumed to be constant at 100,000 Btu/hr-ft²-°F. The second modification causes the code to calculate a power generation from neutron kinetics equations. The neutron kinetics equations incorporate a reactivity insertion term, either a step or a ramp, and reactivity feedback terms based on the Doppler effect and fuel expansion.

Calculations were made with this model using reactivity step insertions of 1.1 to 1.5 dollars and ramp insertions from 5 to 200 dollars/sec. The ramps were continued until a total of 1.5 dollars was inserted, after which the reactivity insertion was held constant.

The calculations show that step insertions greater than approximately 1.4 dollars would cause some fuel element melting. For ramp reactivity insertions, the calculations show that ramp rates up to about 100 dollars/sec would be required for fuel element melting.

The elastic model used to determine the fuel rod expansion becomes increasingly approximate as the matrix temperature increases beyond about

1100 °F (593 °C) because, at these temperatures, the stress behavior becomes partially plastic. Further analysis will be necessary with a transient stress analysis computer code to determine the extent of the error. A topical report is being written describing the model and presenting the results for representative reactivity insertions.

Gas Cooling of FFTF Fuel

D. R. Dickinson

It is currently planned to inspect and disassemble discharged FFTF fuel assemblies in an inert gas atmosphere. During this inspection and disassembly, it is necessary to maintain cooling by a forced gas flow. An analytic study was undertaken to determine the specific cooling requirements.

In this study, the use of an oxide fuel assembly was assumed (earlier, similar calculations were performed for cermet fuel). The assumed fuel assembly had 331 pins, 0.200 in. OD, spaced on a 0.252 in. triangular pitch by a wire wrap on each pin. The heated (fuel) and unheated (fission gas plenum) lengths of the fuel pin were each taken as 3.06 ft. Argon was assumed as the coolant and the maximum allowable surface temperature was taken as 600 °F (316 °C).

During interim inspection with a decay power level of 20 kW, cooling would be accomplished by an axial flow of gas through the bundle in the tightly fitting liner. Under these conditions, a flow of 240 ft³/min with a pressure drop of 3.2 psi will be required to maintain cooling. During complete examination and disassembly,

when the maximum power will be 10 kW, an axial flow of 120 ft³/min with an accompanying pressure drop of 1.0 psi will be required. During cooling by transverse flow after removal of the liner, a flow of 1400 ft³/min with an accompanying pressure drop of 0.7 psi will be required to remove 10 kW. The maximum rate of temperature rise of the fuel if cooling is lost will be 65 °F/min (36 °C) at 10 kW or 130 °F/min (72 °C) at 20 kW.

Also investigated was the possibility of using an organic liquid to provide backup cooling if the gas cooling system were to fail. It was determined that a flow of 1/2 gpm of Dowtherm flowing in the form of a film over the heat transfer surfaces would remove 10 kW with the effluent liquid still well below the boiling point. If criticality is a problem, a compound containing chlorine could be used. One possible compound is α -chloronaphthalene, which is liquid at room temperature and boils at 598 °F (314 °C). The flow required for this compound would be comparable to that for Dowtherm. The principal problem foreseen in using emergency liquid film cooling is to ensure that the liquid flow remains uniformly distributed over the heat transfer surfaces as it flows down the bundle of fuel pins. Testing will be required to determine how to do this.

The thermal hydraulic experimental program needed to support the design and operation of the gas cooled fuel examination and disassembly facility was scoped, and a report (BNWL-CC-817) was written which describes the proposed program in detail. This program is a combination of basic parametric

tests and tests of full scale models of the FFTF fuel assembly. It will consist of the following five phases to be performed over a period of five years:

- Preliminary experiments will provide general scoping and problem area identification for all subsequent tests; large scale heated models of the FFTF fuel assembly will be used, and all phases of fuel inspection will be investigated.
- Parametric tests will determine the effect of varying specific operating conditions and fuel assembly configurations. These tests will be performed with several small and specialized test sections. The results of these tests will permit the extension and generalization of the results to large scale model tests.
- Large scale tests will be performed to evaluate and solve major problem areas encountered in the normal operation of the cooling systems. Specific tests will consider liner removal, flow baffle configuration, and interim inspection.
- Safeguards studies will evaluate operation of the coolant systems under off-normal conditions. Under this program, partial and complete loss-of-coolant accidents and methods of backup cooling will be evaluated.
- Final tests will determine the detailed thermal hydraulic capabilities of the final cooling system design. These tests will also provide a final demonstration of the coolant systems under all major phases of fuel inspection and disassembly.

Temperatures of Fuel Pin Segmenting Spacers

G. M. Hesson

It is anticipated that segmentation devices may be required in the FFTF ceramic fuel pins to restrict fuel slumping. The segmentation devices will probably be metal discs whose diameter is equal to the inside diameter of the fuel jacket. Each disc will have a small hole drilled through its center to permit escape of fission gases to the gas plenums at the ends of the fuel pins. The design of these discs must be determined, in part, by the temperatures of the disc and the cladding in the immediate region of a disc during operating conditions. To supply this information, a section of a reference ceramic fuel pin containing a segmentation disc 1/8 in. thick with a central hole 0.040 in. in diameter was analyzed by the STHTP heat conduction computer code.

Thermal contact coefficients of 500 Btu/hr-ft² between the fuel and jacket, fuel and disc, and disc and jacket were assumed. Disc materials of stainless steel and tantalum were considered. Power generations of 14, 16, and 18 kW/ft were investigated. Preliminary results of the calculations show that the maximum disc temperatures would be about 1600 °F (871 °C). These temperatures are probably high because of the pessimistic values of thermal contact coefficients used in the calculations. At these temperatures, discs of stainless steel appear potentially feasible.

Hydraulic Vessel Mockup

D. L. Hovorka

The Hydraulic Vessel Mockup (HVM) provides capabilities for prestartup design verification and evaluation of reactor vessel hydraulics and mechanical characteristics in water and hot gas environments.

The potentially large requirements for water pumping and electrical power suggest that scale model and/or segment of full size vessel be given serious consideration. Scale models were used for FERMI (upper plenum) and EBR-II to establish mixing and distribution characteristics. SEFOR reactor hydraulics are also being studied with a scale flow model. However the skewed tube core of the FTR presents more complex flow paths than any of the foregoing reactors, and data evaluation would be more difficult. A program to provide the basis for technical and economic evaluation of the HVM facility was initiated. The program included study estimates of various alternate assemblies of the HVM, survey of offsite hydraulic and modeling experience in reactor vessel systems, identification of detailed program content, and study of conductivity measurement technique for mixing evaluation.

For scoping purposes, study estimates were developed for 21 alternative cases of the test facility as affected by the following:

- a. Location - 100 Area and 300 Area
- b. Flow capacity - 6000 to 60,000 gal/min
- c. Vessel size - 1/4 scale to full size model
- d. Mode of operation - short term single pass and recirculation

- e. Techniques for mixing effectiveness, determination, conductivity, and measurement.

The study results provide comparative cost data and other pertinent information required to finalize the HVM concept. Generally, the preliminary indications are that the Hydraulic Vessel Mockup may be favorably located in the 300 Area based on consideration of economics and existing supporting facilities for the following conditions:

- a. All flow rates with recirculation coolant, either isothermal temperature operation or utilizing conductivity measuring technique for mixing determination.
- b. All flow rates on short term flow circulation (once through) under isothermal operation, or using conductivity or temperature distribution technique for mixing evaluation.

The 100 Area site is advantageous where startup power demands are high, makeup water requirements are extremely high, or tests require a large heating load to raise the coolant temperature continuously.

From the initial cost study of 21 cases, three were selected for more detailed evaluation. These were:

- a. Full size isothermal recirculating system with electrolyte injection for mixing studies.
- b. One-half scale hydraulic model coupled with one-quarter sector of full sized vessel suitable for both hydraulic and mechanical testing with electrolyte.
- c. One-half scale hydraulic model coupled with one quarter sector of full size vessel with temperature

distribution technique for mixing determination.

A survey of knowledgeable users of modeling techniques to determine the engineering adequacy of model testing in complex systems has been initiated.

Core Mockup Facility

The Core Mockup Facility provides the capabilities to test mechanical components such as reactor tubes, grid plates, control assemblies, and instruments and to test FTR experimental assemblies under prototypic but non-radioactive operating conditions. This facility will be designed to provide the closely coupled engineering support in terms of proof test results, feasibility, and efficiency of maintenance procedures and training necessary to startup and operate the FFTF at maximum plant availability.

A study outlining the program and facility requirements for a sodium-containing core mockup facility has been initiated. Study of test support needs indicate that a facility is required for design confirmation and proof testing during pre-startup, and for operational support to maintain high plant availability. The pre-startup support is needed largely in the areas where thermal effects (shock and cycling) and coolant environment (materials transport, oxide deposition, sodium frosting) are important to evaluation, since many basic studies can be made on a hydraulic (water) vessel mockup.

Although the important tests to be performed and the support provided should not be measured in terms of dollars saved, a cost evaluation was

made to provide some measure of the value of such a facility. If the FTR time operating efficiency can be increased from 70 to 75% as a result of mockup support, a \$3.5 million dollar facility can be justified. An increase from 55 to 60% warrants a \$5.4 million facility. This analysis indicates that a mockup closely supporting the reactor operation will be economically as well as technically justified.

Sodium Studies Facility

The FFTF program will require a limited number of small scale sodium flow systems to provide state-of-the-art demonstrations and familiarization of personnel with sodium operating procedures and techniques. The facility will be designed to safely house both radioactive contaminated and nonradioactive contaminated flow systems and the necessary supporting equipment required for safe and reliable operation.

Preliminary engineering and scope studies required to support a project proposal and to prepare the design criteria have been completed. Project Proposal BCP-015 and the associated design criteria have been transmitted to the Atomic Energy Commission for review and approval.

Hallam Reactor Equipment

PNL received notification that the Hallam Nuclear Power Facility equipment is to be dismantled and excessed. Listings of equipment to be made available have been received and review for utilization in the FFTF program has been initiated.

CONCEPTUAL REACTOR DESIGN STUDIES

Thermal Sink Limitations Study

J. F. Fletcher, D. S. Trent

Preliminary work has begun on a study to estimate the limitations of the capability of United States river systems to serve as thermal sinks for electrical power generating facilities expected to be installed in the future. The estimates of thermal limitations will be based on use of the water as a once-through condenser coolant. Once such a thermal capacity limit was reached for a given stream, the use of alternate cooling means--evaporative cooling systems, air-dump heat exchangers, or sea water cooling--would presumably become mandatory for any further power plant installations (or other industrial plants) located in the region of that stream.

Thermal capacity calculations will be based primarily on stream flow and temperature data published by the United States Geological Survey, and will be expressed as the heat input required to increase the bulk water temperature to an assumed limit, or range of limits, based on minimum stream flows at high stream temperatures (normally late summer flow conditions will represent thermal capacity minima). Normally, monthly average flows and temperatures will be used for these calculations. However, if minimum monthly average flows or maximum average temperatures differ significantly from annual minima or maxima, day-to-day flow and temperature patterns in the periods of minimum thermal capacity will be scanned and the minimum capacity figure adjusted if warranted. In this study, only streams having a minimum monthly

average flow of $1000 \text{ ft}^3/\text{sec}$ or greater will be considered, since streams smaller than this normally would not have significant thermal capacity during the minimum period. The smaller streams could, if desired, be considered separately in a subsequent study. However, the thermal capacity additions from these streams, it is believed, would not significantly alter the results of the present study.

Estimating the upper limit to which the temperature of a given stream may be raised will require establishing some reasonable basis for the limit. Temperature limits are expected to differ for various streams, governed primarily by the ecologies of the particular streams and of other streams or bays into which they flow. Temperature limits used for this study will be formulated by personnel of the Biology and Environmental Health and Engineering Departments of Battelle-Northwest because of their experience in studying thermal and radiological effects in migratory fish species in the Columbia River.

The limits of thermal capacity as estimated by this study are expected to be conservative, based on constant year-round heat input and on an assumed adiabatic condition at the water-air interface.

The United States Geological Survey publication of stream flow and temperature data is based on a division of the conterminous United States into 14 areas corresponding generally to major river basins. Alaska and Hawaii are reported as separate areas. Data are currently at hand at Battelle-Northwest only for those areas consti-

tuting the Pacific watershed. These areas include:

- Area 9 - Colorado River Basin
- Area 10 - Great Basin (Nevada, Utah, Eastern California and Southeastern Oregon) (incomplete data)
- Area 11 - Coastal streams in California
- Area 12 - Coastal streams in Washington, and upper Columbia River Basin
- Area 13 - Snake River Basin
- Area 14 - Coastal streams in Oregon, and lower Columbia River Basin

Reduction of data for these areas is proceeding pending receipt of data for the remainder of the nation. It is, therefore, planned that the portion of the survey concerned with the Pacific watershed will be completed first, and that other areas will be scheduled for subsequent completion as information is received.

USAEC-AECL COOPERATIVE PROGRAM

Cross-Flow Mixing Between Parallel Flow

D. S. Rowe

Accurate prediction of boiling burnout in rod bundle fuel elements has not been possible except by using the results of laboratory tests which closely simulate the bundle geometry, power distribution, and flow conditions. One of the primary reasons for this is the difficulty in calculating the local flow and enthalpy throughout the bundle. Calculating local flow and enthalpy is a problem in multirod fuel elements because of the different sized flow channels formed

by the arrangement of fuel rods. The shapes of some of these flow passages are such that the ratio of heat transfer surface area to flow area is not equal for all passages, and this causes the coolant enthalpy rise in some flow channels to be much greater than in others. This imbalance in coolant enthalpy affects the channel flow distribution, especially during boiling conditions where the pressure drop for two-phase mixtures is much greater than for liquid flow. Since the pressure drop is nearly the same in each flow channel in the rod bundle, there is an interchange of flow between any of the channels in which the gain in coolant enthalpy is not equal. This interchange of flow, the resulting values of coolant enthalpy and flow rate in the parallel channels, and the effects on the critical heat flux are being studied at the Pacific Northwest Laboratory.

The purpose of this study is to obtain a method of predicting local flow and enthalpy conditions in each subchannel of rod bundle fuel elements. The study consists of an analytical development of calculational techniques and an experimental program to verify the analytical work and supply key information which can only be obtained from experiments.

Analytical Study

Mixing during single-phase flow can usually be described in terms of a turbulent (fluctuating) cross flow between adjacent flow channels. This cross flow is a function of both the channel geometry and the Reynolds

number. For two-phase flow, a diversion of flow from one channel to another occurs in addition to the turbulent mixing. To characterize this behavior, it was assumed that there exists a turbulent cross flow plus a superimposed diversion cross flow caused by the differences in sub-channel pressure. Both types of interchannel flow affect the equations of continuity, momentum, and energy; therefore, these equations were derived using two basic assumptions:

- 1) Steady, one-dimensional, two-phase separated slip flow exists in each subchannel of the bundle.
- 2) Turbulent and diversion cross-flows exist between adjacent channels.

These equations of continuity, energy, and momentum, which were presented in an earlier quarterly report, are complicated nonlinear equations, and solutions can be obtained for only very simple cases such as two parallel channels and single-phase flow. Since the physical situation of interest is much more complicated, it was necessary to employ numerical techniques for solution.

For the numerical solution, the mixing equations were treated as an initial value problem. It is more truly a boundary value problem; however, because of very nearly equal pressures between rod bundle sub-channels, treatment as an initial value problem is justified. Initial conditions at the inlet, and then at each successive increment, are used to extrapolate ahead to the next increment. Pressure drops are computed in the positive sense to permit the proper direction of flow diversion.

A modified Euler integration procedure was used initially to solve the equations, but was found to produce an unstable solution. Using a fourth order Runge-Kutta integration formula, solutions were obtained only when high values of cross-flow resistance and very small step size were considered. The problem with both of these integration methods was the determination of the proper cross-flow. Very small differences in the extrapolated channel pressures caused very large values for diversion cross-flow. This caused successively larger errors in the extrapolated pressures and led to a divergent solution.

An iterative technique similar to the modified Euler integration procedure proved to be successful earlier for a two-parallel channel geometry both with and without boiling. This method was described in detail in the previous quarterly report. Successful application was also attained for the case of a five-parallel channel section of symmetry from a 19-rod bundle. These calculations were performed for both the boiling and the nonboiling cases and for several lengths. Stable solutions were obtained in all cases.

Experimental Program

The purpose of the experimental portions of this program is to verify the analytical description of mixing and to obtain numerical values for interchannel mixing. These experiments are being performed with electrically heated test sections simulating various types of adjacent flow channel configurations that occur in a 19-rod cluster. Initially, a test section simulating a flow channel

formed by fuel pins in a square pitch array adjacent to a channel formed by pins on a triangular pitch array was considered. In this test section, the simulated rods were spaced 0.080 in. apart. Data were taken at inlet temperatures of 330 °F (165 °C) and 510 °F (265 °C), total mass velocities of 1.0, 2.0, and 3.0 x 10⁶ lb/hr-ft², and at a pressure of 900 psia. Preliminary comparisons of the experimental data with the results of the analytical model developed earlier indicated that the amount of turbulent cross-flow during boiling in the channels was not much different than turbulent exchange during nonboiling conditions. This observation was considered rather surprising, because it was originally expected that the boiling would induce additional agitation in the coolant and would, therefore, cause a much greater cross-flow between channels. It is hoped that data obtained with a second test section will provide more information to help explain this apparently illogical behavior.

The second test section is essentially a modification of the first one with a rod spacing of 0.020 in. instead of 0.080 in. This modification was accomplished by placing shims behind the test section support blocks so that the simulated rods in the test section were deflected inward. The resulting flow channel had the simulated rod spacing of 0.020 in. over most of its length with the exception of the 2 in. at each end. Here, the spacing went through a gradual transition from 0.020 in. to the original 0.080 in. Experiments are currently

being performed with this test section. The flow conditions being considered are essentially the same as were used earlier with the 0.080 in. spaced test section.

Evaluation of Zirconium-2.5 wt% Nb Pressure Tubing

P. J. Pankaskie

The general objective of this program is to evaluate Zr-2.5 Nb as a pressure tube material with reference to the test and reactor experience that has been obtained on Zircaloy-2 pressure tubes.

Crack propagation tests were performed at room temperature on four heat-treated test specimens. The heat treatment consisted of: solution treatment at 880 °C, water quench, and aging in vacuum for eight hours at 500 °C. Cracks in all specimens had propagated full length at failure. The results are tabulated in Table I.

Two series of brittle fracture tests were performed with Zr-2 tubing. This tubing is of about the same dimensions as the Zr-2.5 wt% Nb tubing used in this program to date. One series of tests used test specimens with slots 1-1/2 in. long and 1/16 in. wide which penetrated through about 80% of the tube wall from the outer surface. These slots were oriented

TABLE I. Crack Propagation Data for Zr-2.5 wt% Nb Alloy (Test temperature, 25 °C)

<u>Specimen Identification</u>	<u>Specimen Condition</u>	<u>Slot Length (in.)</u>	<u>Hoop Stress at Failure (psi)</u>
30-29	Heat treated	1/2	114,000
30-30	Heat treated	1 1/2	59,600
30-31	Heat treated	2 1/2	41,000
30-32	Heat treated	4	33,400

with respect to the tube axis at angles varying from 0° through 90°. The second series of test pieces was made so that the length of the slot projected onto the tube axis was always 1-1/2 inches; the angle between the slot and tube axis varied from 15° through 75°. The results are tabulated in Table II.

Sealing problems were encountered in the test pieces with the 75° and 9° slots and brittle crack propagation could not be induced. In both series of tests the brittle crack tended to turn towards a line parallel to the axis of the tube and through the midpoint of the slot.

Drop weight tests⁽¹⁾ were used to determine the Nil Ductility Temperature (NDT). The results are tabulated below:

- A) "As received" Condition
 - 30% cold worked (CW)-NDT, Room
 - 60% CW-NDT is below -62 °F.
- B) Heat Treated
 - 1) Solution treated @ 830 °C and water quenched.
 - 30% CW-NDT, Room
 - 60% CW-NDT, 35 °F

(1) R. C. Aungst, "Additional Crack Propagation Tests on Zircaloy-2 Reactor Pressure Tube" BNWL-SA-179, September 20, 1965.

TABLE II. Effects of Crack Orientation on Crack Propagation Behavior of Zircaloy-2 Tubing (Test temperature, 25 °C)

Specimen Identification	Angle Degrees	Actual Slot Length (in.)	Projected Slot Length (in.)	Hoop Stress at Failure (psi)	Crack Length at Failure
Series 1					
Zr-1A	0	1 1/2	1 1/2	93,000	Brittle-Partial
Zr-2A	15	1 1/2	1.44	111,000	Brittle-full length
Zr-3A	30	1 1/2	1.30	128,200	Brittle-full length
Zr-4A	45	1 1/2	1.06	143,900	Brittle-full length
Zr-5A	60	1 1/2	0.79	172,500	Brittle-full length
Zr-6A	75	1 1/2	0.39	-	Could not seal after web of
Zr-7A	90	1 1/2	0	-	slot failed
Series 2					
Zr-1B	15	1.55	1 1/2	107,400	Brittle-full length
Zr-2B	30	1.73	1 1/2	111,000	Brittle-full length
Zr-3B	45	2.12	1 1/2	114,600	Brittle-full length
Zr-4B	60	3.00	1 1/2	96,700	Brittle-full length
Zr-5B	75	5.79	1 1/2	-	Could not seal after web of slot failed

- 2) Solution treated as above plus aged at 500 °C for 8 hr in vacuum.

30% CW-NDT, Room

60% CW-NDT, -40 to -43 °F

For the cold-worked condition, metallographic examinations suggest that the hydride content of the 60% CW is somewhat lower than the 30% CW material. Vacuum extraction analysis indicates that both cold-worked tempers contain about the same level (20 ppm) of hydrogen. Further metallography shows that in the 60% CW the hydride platelets are relatively large, few in number, and to a large extent parallel to the tube surfaces. In contrast, the 30% CW material exhibits relatively large numbers of small platelets which appear to be randomly oriented. For the 30% CW material there is no observable difference in NDT between the cold-worked and the heat-treated conditions. For the 60% CW material, heat treatments changed the NDT 97 °F in one case and 20 °F in the other. Since the 30% CW material was entirely unaffected by heat treatments, it is suspected that heat treatment and

water quenching caused a redistribution of hydrides in the 60% CW material and that the vacuum aging may have resulted in some small amount of vacuum extraction of hydrogen. Had there been other microstructural changes resulting from heat treatments, the NDT for the 30% CW material would also have been changed since both materials were heat treated simultaneously.

Crack Propagation of Zr-2.5 wt% Nb

P. J. Pankaskie

Crack propagation data have been obtained at room temperature for Zr-2.5 wt% Nb alloy tubing in a heat-treated condition. A series of tests was made at room temperature to determine the effects of crack orientation on the crack propagation behavior of Zr-2 tubing. Crack orientations ranging from 0° to 90° to the longitudinal axis of the tubular test specimens were used.

Drop-weight tests were made to determine the Nil Ductility Temperature for Zr-2.5 Nb alloy in the 30% and 60% cold-worked and heat-treated condition.

APPLIED AND REACTOR PHYSICS

PHOENIX FUEL PROGRAM

D. S. Trent

Thermal hydraulic analysis in support of the MTR-Phoenix Fuel and PRCF Experiments was continued.

Previous MTR core design analysis efforts were concerned primarily with

a symmetric 3 x 9 partial plutonium loading with eight control rod positions. The analysis underway at present is concerned with the same basic configuration except a full plutonium loading will be considered. In addition, the possibility of using a

symmetric 3 x 7 arrangement with full plutonium loading will be investigated. Initial three-dimensional power distributions have been supplied by Engineering Physics for the 3 x 9 fuel plutonium core but thermal hydraulic analysis has not been completed for these cases.

Transient heat transfer and steam generation equations were derived for the analog simulation of a postulated power excursion during the PRCF experiment. These equations include transient models for the fuel temperature, clad surface temperature, and void formation from boiling. The equations were formulated in the same manner as presented in HW-69168 SUP, "Plutonium Recycle Critical Facility Safeguards Analysis Supplements," except for a case of flat plates instead of rods and different definition of the fuel-clad contact coefficient.

The PRCF MTR fuel plates are fabricated with unbonded cladding and, as a result of this and manufacturing tolerances, a small amount of air is trapped within the assembled fuel plate. During a power excursion the air will heat and expand, tending to bow the cladding away from the fuel

meat. The net result is severe deterioration of the heat transfer contact coefficient which could possibly lead to melting of the PuAl fuel. To describe the contact coefficient as a function of the entrapped gas temperature, the cladding was treated as a simple fixed-end beam uniformly loaded by the gas pressure which is given by the perfect gas law. Deflection may then be calculated as a function of the gas temperature and used to define the contact coefficient.

High Temperature Lattice Test Reactor Oscillator Development

D. H. White and E. F. Antal

Fabrication of major components continued during the quarter. Procurement and/or fabrication of the oscillator main frames and carriages was completed. A carriage assembly is shown in Figure 1. Shown in Figure 2 is the light duty oscillator frame assembly mounted on the carriage.

An order was placed for the electrohydraulic servo drive systems for both oscillators. At the end of the quarter the project is estimated to be 45% complete.

ENGINEERING DEVELOPMENT

UPSTREAM BOILING BURNOUT

D. R. Dickinson

The burnout heat flux in heat transfer to boiling water usually decreases with increasing enthalpy, and boiling

burnout, therefore, usually occurs first at the downstream end of a uniformly heated channel. However, recent tests at Hanford have demonstrated that under certain conditions

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FIGURE 1. Oscillator Carriage

Neg 0662604-1

FIGURE 2. *Oscillator Frame on Carriage*

burnout occurs at upstream locations. The present research program is being conducted to study further this phenomenon of upstream burnout to determine the conditions under which it occurs and the reason why it occurs.

Detailed analysis was performed on experimental data obtained earlier with an electrically heated test section 0.435 in. ID and 12 ft long. Included in these data was information on postburnout heat transfer and the spread of burnout to new locations as heat flux is increased (at constant inlet enthalpy) beyond the initial burnout indication. These data showed large variations of burnout heat flux at various axial and angular locations. For example, the following behavior was observed in an experimental run with a coolant flow rate 3.0×10^6 lb/hr-ft², inlet quality of 54%, and outlet pressure of 1500 psig. The first burnout was observed to occur at a heat flux of 230,000 Btu/hr-ft² at a point 60 in. from the downstream end. A heat flux increase to 340,000 Btu/hr-ft² resulted in burnout indications at several other thermocouple locations scattered over most of the length. However, further increase to 680,000 Btu/hr-ft² was required to induce burnout indications at all locations. The burnout heat flux at a constant angular position appeared to pass through three maxima and three minima over the 12 ft length, with large variations (as much as three-fold)

occurring between these maxima and minima. The variation of burnout heat flux with angular position was from 330,000 to 400,000 at the downstream end and 300,000 to 680,000 at the midpoint. Similar behavior was observed in other tests at lower inlet qualities, but in these tests, the postburnout rate of temperature rise was sufficiently high to prompt the discontinuation of the tests before all thermocouples indicated burnout.

It appears that either the flow pattern is such as to produce alternate zones of susceptibility and resistance to burnout, or that the onset of burnout in one region tends to inhibit it in neighboring regions. A spiral pattern would explain both the axial and circumferential variations. This type of flow pattern is further supported by the fact that the angular positions with the lowest burnout heat flux at the downstream end had the highest burnout heat flux at the midpoint and vice-versa. Firm conclusions are not possible because of the limited number of locations at which thermocouples were placed. Also, the small postburnout temperature rise and large temperature drop across the tube wall made identification of burnout difficult. It is expected that a new test section now under construction will provide more extensive and precise data about this phenomenon.

PLUTONIUM UTILIZATION PROGRAM

PuO₂-UO₂ FUELING STUDY IN BWR'S AND PWR'S

R. M. Hiatt and J. R. Worden

Previous BNW studies of plutonium fueling schemes for thermal reactors have emphasized parametric surveys to identify optimum strategies. The methods used necessarily involved some simplification in representation of the reactor core. In the present phase of activity, it is desired to examine core designs more specifically for the present generation of commercial reactors. This requires improved methods for sizing, designing, and optimizing nuclear reactor cores within specified thermal and hydraulic limits. This need is important also because of the demand for larger reactors with capabilities of 1000 Mw(e) within the near future. Therefore, the following information to be developed in these studies should be of prime consideration in commercial reactor design studies:

- An economic history over the life-time of the reactor, including such items as fabrication, maintenance, and operational costs.
- Expected reactor life (mechanical endurance).
- Core lifetime and refueling requirements.
- Reactor design within thermal and hydraulic limitations.
- Safe operating limits.

Owing to the complexity of achieving the necessary data, a chain of computer codes is being linked together for this purpose. This lends a degree of flexibility to the method

since various alternative codes may be linked, thus allowing some latitude in problem analyses.

Core size affects mechanical safety and endurance limits at various fuel irradiation levels. A computer code called "REPP" has therefore been written which limits thermal and hydraulic parameters. The code affords the following contributions:

Input Data

- Model and hardware assumptions which may be specified from manufacturing capabilities and techniques consisting of:
 - a) Type of core lattice
 - b) Dimension of unit cell (cross section only)
 - c) Spacing between cells
 - d) Pin diameter and clad thickness
 - e) Total core void fraction defined as $(1 - \text{Fuel Fraction})$
 - f) Fuel density
 - g) Estimated core volume
 - h) Type and number of fuel pin support grids per unit cell and their axial location.
- Entrance coolant specifications:
 - a) Entrance coefficient as a function of velocity head
 - b) Entrance coolant temperature and pressure
 - c) Coolant mass flow rate
 - d) Entrance length (unheated).
- Exit coefficient as a function of velocity head.
- Power parameters
 - a) Axial power density
 - b) A specified safety factor which includes the product of all

peaking factors, divided by the average power peaking factor.

Output Data

- Pressure drop considerations for both homogeneous and nonhomogeneous fluid flow, but not interchannel mixing.
- Designs for a useful core within specified thermal hydraulic limits for two fuel geometries, either of which may contain annular pins, namely:
 - a) Square pin array
 - b) Equilateral triangular array.
- An equivalent core volume defined from specified thermal and hydraulic limitations and total core power (the volume is defined in terms of an equivalent core radius and a length).
- Pin pitch, number of pins per core, and hydraulic diameter.
- Calculation of coolant property values for 10 to 100 mesh points.
- Fuel centerline temperature, and heat flux at a minimum of 10 axial locations.
- Average power density, or heat generation rate.
- Coolant-to-fuel ratio.
- Total effective heat transfer area.
- Core fractions contributed by various components.

- Average Reynolds number.
- Other various input items for succeeding links.

Data from the Connecticut Yankee Reactor were used to check REPP for errors. Below are input data and results of the comparison.

Input data taken from the Connecticut Yankee Reactor design:

Coolant inlet temperature (°F)	546
Coolant inlet pressure (psi)	2,050
Clad wall thickness including 0.00225 in. diametric gap	0.01825
Coolant flow rate (lb/hr)	84.24×10^6
Fuel pellet diameter (in.)	0.373
Active core length (ft)	10
Equivalent core radius (ft) ^(a)	4.96
Total core volumetric void fraction ^(a) defined as (1-fuel fraction) (%)	68

(a) *Best estimated value from given data.*

Results:

	Connecticut Yankee	REPP
Total heat output (Mw_t)	1,473	1473.59
Ave. heat generation rate. (Btu/hr-ft ³)	0.20256×10^8	0.20635×10^8
Coolant mass flow rate (lb/hr-ft ²) ^(a)	0.208×10^7	0.201×10^7
Number coolant channels ^(a)	33805	33803
Number of pins ^(a)	33805	33803
Pin OD (in.)	0.41150	0.41148
Pin pitch (in.)	0.55300	0.5581
Core vol (ft ³)	772.5	761.6
ΔT Coolant (°F)	46	44.8
Exit coolant temperature (°F)	592.0	590.8
Average coolant velocity (ft/sec)	12.1	12.5
Average heat flux (Btu/hr-ft ²)	1.345×10^5	1.3662×10^5
Maximum heat flux (Btu/hr-ft ²)	4.372×10^5	4.440×10^5
Hot channel peaking factor (Fq)	3.25	3.25
Departure from Nuclear Boiling (DNB) ratio	2	1.3965
Average maximum fuel temperature (°F)	?	1316
Total heat transfer area (ft ²)	36,400	36,414
Total pressure drop across the core (psi)	?	34

Differences indicated in the table may arise from the following conditions:

- 1) Lack of information on the Connecticut Yankee reactor necessi-

(a) *Best estimated value from given data.*

tated "a best calculated value" for input data.

- 2) Space occupied by the control rods was not taken into consideration.
- 3) A chopped cosine was assumed to represent the axial power density curve.

- 4) Voids within the core were not clearly defined in the Hazards Report.
 - 5) The revised Tong correlation was used to predict DNB conditions. This may account for the smaller DNB ratio as calculated for REPP.
- The pressure drop across the core was based on a seven-grid honeycomb arrangement. Since pressure drop data and grid arrangement in the Connecticut Yankee Reactor core were not available, the value predicted by REPP cannot be substantiated. It should be emphasized that all values are for conditions in the core and not the pressure vessel. Figure 3 makes a comparison of the core heat flux indicating the sizing process used by REPP.

FIGURE 3. Comparison of DNB Heat Flux with an Allowable Hot Channel Flux and an Average Heat Flux as Calculated by REPP

PRTR AND FERTF FUEL ELEMENT TESTING

D. E. Rasmussen

Testing was started on a newly designed basket type element for use in the FERTF. The first test performed was determination of the forces encountered when charging and discharging the element. In this test, the diameter and straightness of the pressure tube installed in the EDEL-I loop was measured and the nominal diametric clearance (0.096 in.) between the tube and fuel element basket determined. Equipment was installed to bow the pressure tube a measured amount. No change in force was noted until the tube bow was 0.250 in. (centerline displacement at the midpoint of the tube). When the bow was increased to 0.325 in., the fuel would not move into the tube under its own weight. Based on this test, it appears that the maximum tube bow allowable for successful charge-discharge of the fuel is about 0.300 in. if the basket has a nominal diametric clearance of 0.096 in.

A new type of wire-bound short core element (Figure 4) was installed in EDEL-I subsequent to completion of the above test and testing of this element is underway at 1075 psig, 530 F, and 150 gpm.

PRTR EXPERIMENTAL ADDITIONS

P. C. Walkup and D. E. Rasmussen

Planned operation of the PRTR, following recovery from the Rupture Loop incident, entails several modifications to the reactor complex intended to avoid or alleviate problems encountered subsequent to the incident. These modifications, to be funded

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FIGURE 4. PRTR High Power Density
Nineteen-Rod Cluster Fuel Element

from both General Plant Construction funds and operating expense, are:

- 1) Increased Waste Handling Capability - Additional waste pumping capacity, provisions for measuring critical primary loop conditions during PRTR ventilation contain-

ment, and provisions for performing venting and makeup of liquid streams without containment vessel entry. These items, some 14 in total, were combined in a single construction project estimated to cost \$82,000 and to

be completed by April, 1967. Detail design for these items has been completed.

- 2) PRTR Contamination and Waste Control - Additional activity measuring and waste volume accounting capabilities. The scoping of this construction project is in progress.
- 3) Rupture Loop Particulate Removal - Studies indicate that an axial vortex separator, in the Rupture Loop effluent piping, would be effective in removing and retaining fuel debris and not have the disadvantages of a filter or hydroclone system (high pressure drop and high cost, respectively). The device consists of a swirler sec-

tion which creates a vortex in the stream, two collector sections which withdraw the particulate matter, and a deswirler section which straightens the flow. Procurement of materials was completed and fabrication of the device was started. A test loop was designed in which the separator will be tested. The loop consists of a pump, a particle injection device, and a test section. Provision is made for measuring and controlling pressure and flow in the various parts of the system. It is planned to have the separator in service before reactivation of the Rupture Loop.

NUCLEAR SAFETY

CONTAINMENT SYSTEMS EXPERIMENT

Iodine Generation

L. F. Coleman

The aerosol generation facilities for CSE must provide for release of multi-gram masses of each aerosol component at both high rates for short times and low rates for extended times.

Elemental iodine was volatilized into carrier air streams and passed through a pipe 21 ft long to study generation rate, deposition in the pipe, and iodine form at the pipe outlet. The delivery line was type 304 SS with a 2 in. ID. It was heated to 150 °C. Table III lists the results of some of the tests. Release rates up to 11 g/min were obtained.

Deposition in the pipe averaged ~0.1%. Maypack discrimination showed about 0.45% on Gelman Type A filters, 99.5% on six silver plated screens, 0.03% on Gelman Type AC charcoal paper, 0.02% on the charcoal bed. Analysis of individual silver screens indicated that elemental iodine breakthrough of six screens occurred when total screen set loading exceeded 10 mg.

Iodine Formation in Steam Condensate

J. D. McCormack

An effort was made during Aerosol Development Facility Run 65 to determine the changes occurring in the chemical form of iodine after absorption

into steam condensate. Sample C-1 was a condensate sample taken five minutes after iodine was injected into the gas phase. It was split and the first portion was extracted as soon as possible (13 minutes later). Several hours later the second portion was extracted. This was repeated with Sample C-7, taken 90 minutes after injection. The pH of the condensate was 6.7. Although not conclusive, the data in Table IV suggest that iodine form in condensate does not change with time.

Mass Transfer Models

R. K. Hilliard

A mathematical model of transport of fission product vapors in a containment system in the absence of condensing steam was developed for the simple case of a vessel whose contents are well mixed, which has perfect absorbing walls, and a transfer rate limited by molecular diffusion through a boundary layer film. The mass transport coefficient was expressed in terms of containment system fluid properties (temperature,

TABLE III. Elemental Iodine Generation

Total Release (g)	Release Rate at 1 CFM (mg/min)	Deposition in Pipes (%)	Maypack Distribution (%)			
			Filter	6 Ag Screens	Charcoal Paper	Charcoal
1.09	908	0.062	0.940	99.03	0.031	0.001
1.00	388	0.055	0.446	99.52	0.029	0.005
10.07	3,870	0.170	0.219	99.72	0.027	0.038
10.08	670	0.149	0.540	88.60 ^(a)	11.14 ^(a)	<0.005
100.00	11,040	<0.012	0.137	99.79	0.022	0.047
Average		0.090	0.436	99.51	0.027	0.019

(a) Overloaded silver screens; averages exclude these values.

TABLE IV. Condensate Extractions at Various Storage Times

	C-1a	C-1b	C-7a	C-7b
Iodine Conc., mg/ml	0.53	0.53	0.27	0.27
Time sampled, min	5.	5.	90.	90.
Time extracted, min ^(a)	18.	175.	103.	196.
Aqueous phase (ionic), %	99.78	99.76	98.11	98.15
Benzene phase ^(b) (organic), %	0.024	0.020	0.19	0.18
Caustic (elemental), %	0.20	0.22	0.7	1.67

(a) After injection of iodine into containment atmosphere.

(b) After caustic back extraction.

pressure, gas velocity, area-to-volume ratio) and diffusivity of the transported species. For turbulent conditions, the following equation resulted:

$$\frac{C}{C_0} = \exp \left[- \frac{0.0047 T^{0.4} U^{0.77} (D_V)_0^{0.67} A}{P^{0.23} V} \right] t$$

Where C = gasborne conc. at time t
(g/cm³)

C₀ = gasborne conc. at time 0
(g/cm³)

T = temperature (°K)

P = total absolute pressure
(atmos)

U = average bulk gas velocity
parallel to surfaces
(cm/sec)

A = surface area (cm²)

V = volume (cm³)

(D_V)₀ = molecular diffusivity
in gas at 0 °C, 1
atmos (cm²/sec)

t = time (sec)

This equation states that the variables affecting containment gasborne half life are, in order of decreasing importance:

Area/volume ratio	(A/V) ⁻¹
Bulk gas velocity	(U) ^{-0.77}
Molecular diffusivity	(D _V) ^{-0.67}
Temperature	(T) ^{-0.4}
Pressure	(P) ^{0.23}

In order to determine whether bulk gas velocity is a major variable, a run was made in the ADF (No. 66) in which the agitator was run for the first 27 minutes, then stopped, then started again at 180 minutes containment time. Figure 5 shows the rate of

change of the dimensionless gasborne elemental iodine concentration at two different velocities. The air velocity while the agitator was running was measured with a heated thermopile anemometer* as 50 ft/min near the vessel bottom, and ~10 ft/min near the top. An average of 15 ± 15 ft/min was assumed for calculation purposes. Measurements were not possible at zero agitation, but 0.1 ft/min was assumed. The solid curve in Figure 5 is the solution of the equation for these conditions. Several conclusions can be made from this figure:

- 1) gas velocity is an important variable,
- 2) inertial effects prevented sharp time changes in gas velocity,
- 3) departure of experimental points from the model late in the run was probably caused by the walls no longer behaving as perfect absorbers (i.e., a desorption coefficient should be included in the models), and gas velocity measurements in the range 1 ft/min and lower are needed in order to evaluate the model adequately.

Uranium Analysis of Fibrous Filters

J. D. McCormack

The CSE fission product aerosol will include UO_x particles produced by condensation of UO₂ vapor from molten unirradiated fuel. Aerosol concentration will be measured by analyzing the Maypack filter for uranium. Analysis for U²³⁸ by direct alpha counting of the filter is attractive because of its relative rapidity and economy.

* *Hastings Precision Air Meter, Hastings-Raydist, Inc., Hampton, Virginia.*

FIGURE 5. Effect of Air Velocity on Elemental Iodine Concentration

Attenuation must be accounted for, however, since the Maypack filter (Gelman type A), a mat of glass fibers, will permit particles to penetrate to various depths. Figure 6 shows that the relation between direct alpha count and fluorimetric analysis (of extracts from the filters) is constant for the range studied, and can be expressed as:

$$\mu\text{g uranium} = (\alpha \text{ c/m})^{1.2}$$

This relationship should be confirmed for higher and lower mass loadings. The lower detection limit would correspond to about one microgram per cubic meter for CSE sampling conditions.

Coating Service Life in Steam

J. D. McCormack

Autoclave testing of three types of coating on carbon steel coupons was

FIGURE 6. Uranium Alpha Count as a Function of Uranium Mass on Glass-fiber Filters

continued with saturated steam at 130 °C and 30 psig. Three coupons of each type were exposed to the steam and removed periodically for inspection. Table V shows that Coating C, though superior to the other paints, is subject to blistering at 100 hr and cracking at about 1000 hr exposure. The blisters were water-filled, but did not enlarge with exposure time.

TABLE V. Coating Behavior in Steam

Coating	Hours Exposure to Steam at 130 °C		
	Blistered	Cracked	Lifted
A	30	100	1700
B	30	30	100
C	100	1300 ^(a)	>3300 ^(b)

(a) One of three coupons

(b) Test terminated

Decontamination

L. D. Perrigo

Decontamination procedures are being evaluated for painted surfaces to aid in the containment system cleanup after experiments. Work during the past quarter has centered on both laboratory and pilot plant studies.

A 10% H_3PO_4 -0.25% HBr solution reduced iodine activity on contaminated surfaces in the laboratory by a factor of eight in one hour at 100 °C. Other solutions tried included 10% H_2SO_4 , 10% HNO_3 , 10% NaOH-3% KMnO_4 , 10% H_2SO_4 -0.25% HBr, 10% NaS_2O_3 , 2% NaOH-2% gluconic acid, 1M FeCl_2 -0.1M HCl, and 100% dimethyl sulfoxide. At 100 °C these were all ineffective. Decontamination factors ranged from 1.2 to 3 in one hour.

Emphasis in pilot plant work has been on applying vapor phase cleaning techniques. Vapor phase cleaning could be readily applied to large containment system structures with only very minor changes. In addition, only relatively small volumes of waste are generated. Iodine decontamination factors of 1.2 to 2.6 were obtained with steam and NH_4OH , NaOH, KBr, H_3PO_4 , H_3PO_4 -HBr, and alkaline permanganate cycled with HNO_3 . Although ineffective on painted surfaces, vapor phase processes generally produced outstanding stainless steel decontamination with decontamination factors ranging up to 64.

An autoradiographic study of contamination mechanisms was started to aid in the study of containment system safety parameters. Autoradiographs of painted surfaces contaminated with

radioactive iodine showed three possible contamination mechanisms: (1) general adsorption or uniform deposition of extremely fine particles, (2) deposition with subsequent reaction with the painted surface, and (3) deposition of relatively large agglomerates.

Xenon Diffusion Experiments

J. G. Knudsen*

Diffusion tests in the CSE containment vessel using xenon as the tracer element will provide a means of determining diffusion coefficients and fluid mechanical behavior. These data can then be used to predict the distribution of other released fission products.

Transport measurements require a knowledge of:

- 1) manner in which mass is transported throughout the gas space, and
- 2) manner in which the material reacts at the walls of the container.

Neglecting gravitational setting, fission products are probably transported through the containment atmosphere by a diffusional process which is not necessarily molecular but may be turbulent. Superimposed on this diffusional motion may be bulk motion due to convection. If one knows the diffusion coefficient (D) and the convective terms (u, v, w), it is possible to predict the concentration (C) of fission products as a function of time (t) and spatial coordinates (x, y, z) by solution of the equation:

* Consultant, and Assistant Dean of Engineering, Oregon State University.

$$\frac{\partial C}{\partial t} + u \frac{\partial C}{\partial x} + v \frac{\partial C}{\partial y} + w \frac{\partial C}{\partial z} =$$

$$D \left[\frac{\partial^2 C}{\partial x^2} + \frac{\partial^2 C}{\partial y^2} + \frac{\partial^2 C}{\partial z^2} \right]$$

Analytical solution of this equation is limited to a few simple geometries and boundary conditions. For any given containment vessel, numerical solution would be possible but such a numerical solution would require knowledge of the diffusion coefficient (D), bulk motion, and boundary conditions.

For most containment conditions a certain amount of agitation will exist because of introduction of steam and temperature gradients. Therefore, D will be greater than the molecular diffusion coefficient but not likely as large as that of highly turbulent systems such as in turbulent pipe flow.

The use of xenon as a tracer element for diffusion tests in the CSE containment vessel offers several advantages:

- 1) xenon is not absorbed by the walls so the boundary conditions are known exactly,
- 2) xenon does not contaminate the system, hence decontamination time is greatly reduced and more tests may be conducted, and
- 3) sampling apparatus is simple and may be placed at any desired location.

The diffusion coefficients thus obtained for xenon may be used to determine diffusion coefficients for other fission products released into

the containment vessel. In a system where some turbulence exists and the diffusion coefficient (D) is considerably larger than the molecular diffusion coefficient (D_m), the ratio D/D_m is a function of the fluid mechanical behavior of the system and not strongly dependent upon the material diffusing. Therefore, for a given fluid mechanical condition in the vessel,

$$\left(\frac{D}{D_m} \right)_{\text{xenon}} \approx \left(\frac{D}{D_m} \right)_{\text{iodine}}$$

A recent test in the ADF laboratory (Run 68) indicated a value for D of $0.83 \text{ cm}^2/\text{sec}$ for xenon in "quiescent" room temperature air. At the conditions of the experiment the molecular diffusion coefficient is $0.11 \text{ cm}^2/\text{sec}$, indicating a ratio of 7.5 for D/D_m . This suggests considerable motion in the three-ft-diameter vessel even under supposedly still conditions.

Experimentation with xenon in the CSE containment vessel should be conducted at a number of conditions to cover the range of temperature levels, steam-to-air ratios and steam addition rates.

Iodine Paint Reactions

E. C. Martin and J. A. Partridge

Laboratory study of the interaction of iodine with paint, undertaken in support of CSE, was continued. During this period some additional work was done on perfecting the analytical techniques for quantitative work using a gas chromatograph.

Work thus far on the reaction of I_2 with the paint indicates that as

the paint ages or dries the rate of production of volatile iodides decreases markedly. It was observed that I_2 under the influence of ultraviolet radiation reacted with painted stainless steel coupons, which had been aging naturally for two to three months to produce methyl and/or hydrogen iodide at a rate proportional to the concentration of I_2 vapor. In similar runs using coupons which had been in a vacuum oven at 90 °C for 44 days, no volatile iodides were observed. Also, runs were made in which I_2 vapor was allowed to react with Coating C paint filings which had been prepared and allowed to age and dry naturally nine weeks previously. In these runs no volatile iodide was detected. This paint should have been drier than that of the same age on a coupon since any volatile components should evaporate from the small pieces of filings faster than from a smooth painted surface on an impervious substrate.

Work has been initiated with radioiodine, and methods and techniques for handling and sampling are being developed in preparation for experiments involving the reaction of iodine-131 with paint.

Blowdown Experiments on SM-2 Vessel

M. E. Witherspoon

Experimental blowdown work has been carried out on an 88-ft³, low-pressure (200 psig), low-temperature (250 °F) vessel. A total of 20 runs, including seven blowdowns at a four inch nozzle and 13 at an eight inch nozzle, were carried out. The purpose of these tests was to evaluate instrumentation and techniques proposed for the reactor simulator vessel.

The instruments involved in these tests were: a Pace differential pressure transducer used for liquid level measurement, a conductivity probe prototype, also for liquid level, two Norwood strain gage pressure cells, and two Microdot weldable strain gages. All of these devices are types ordered or proposed for use on the simulator vessel. Readout equipment included a Brush two-channel recorder used for the Pace DP cell and the conductivity probe, and Textronics Model 564 memory scopes used for the Norwood and Microdot transducers.

Satisfactory performance was obtained from all sensors at temperatures of 250 °F (121 °C), and with steam, saturated air, and water exposure on the transducers. The conductivity probe leaked, was repaired, and returned for successful testing. Norwood transducers were successfully used to measure stagnation pressure of water impinging on a deflector plate, and the pressure of vapor above the liquid level in the vessel.

The rupture discs used for these tests were burst by two methods. First, the back pressure between the two 150-lb discs was vented so that the inboard disc failed first. Second, the back pressure was increased until the outboard disc failed. The discs ruptured efficiently by both methods. In two tests of the second method, disc material was plastered on the deflector cone and had the effect of shielding the stagnation pressure sensor from the pressure transient at blowdown. As a result, no stagnation data were obtained.

The memory scopes used to trace pressure transients indicated by the

transducers require electronic triggering. This was accomplished successfully by two methods. Conductivity of the water jet on some wire ends in the nozzle allowed a small battery to trigger the scope as soon as the water jet started down the nozzle. The attachment of small gage wire to the inboard rupture disc caused triggering of the scope as soon as the disc ruptured and the wire parted.

Two motion pictures were taken of eight-inch blowdowns. The first was a normal speed film of the entire vessel during blowdown for general observation purposes. The second was a high speed (1000 frames/sec) film of the deflector plate assembly to determine the effectiveness of the assembly in removing the horizontal thrust vector

Blowdown Analysis

R. T. Allemann

The computer code written for the containment system experiment⁽¹⁾ was found unsatisfactory for the SM-2

- (1) B. M. Johnson. "Containment Systems Experiment Part III Mathematical Models of Pressure-Temperature Transients," BNW-233, May 1966.

blowdown because the pressure and temperature of the SM-2 vessel are so low that the straight line approximations used for high pressures flow do not apply.

Step-by-step calculations based on the constant velocity bubble rise model of Margolis & Redfield⁽²⁾ were made to predict the time of transition from liquid to steam flow.

The flow rate based on a nozzle flow coefficient⁽³⁾ gave transition time shown in Table VI.

The Margolis-Redfield model calculation shows excellent agreement with the experimental transition times. Preliminary calculations show that the Margolis-Redfield model can predict the amount of liquid remaining in the vessel after blowdown within a few percent. Further refinements of this approach are being made to cover the blowdown subsequent to water-steam transition.

- (2) S. G. Margolis & J. A. Redfield. *An Analytical Model for Blowdown Experiments*, ANS Trans. 8, 1, 309, 1965.

- (3) N. Uchida and H. Nariai. *Discharge of Saturated Water Through Orifices & Pipes*, Proc. 3rd Int. Ht Trans. Conf., paper 152, p. 1. August 1966.

TABLE VI. Time of SM-2 Vessel Blowdown to Liquid-Steam Flow Transition

	Nozzle Size	
	8 in.	4 in.
A. Time (sec) calculated by nozzle flow coefficient method ⁽³⁾		
1. Top of Nozzle	6.85	27.1
2. Bottom of Nozzle	7.95	29.6
B. Time (sec) calculated by Margolis & Redfield froth model ⁽²⁾		
1. Top of nozzle	8.65	28.5
2. Bottom of nozzle	9.4	30.0
C. Experimental time (sec)	9.0	27.0-30.0

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