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QUARTERLY REPORT
APRIL, MAY, JUNE, 1966**

H. HARTY, et al.

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ENGINEERING DEVELOPMENT DEPARTMENT
QUARTERLY REPORT
APRIL, MAY, JUNE, 1966

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ENGINEERING DEVELOPMENT DEPARTMENT

QUARTERLY REPORT

APRIL, MAY, JUNE, 1966

DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY

CIVILIAN POWER REACTORSFAST FLUX TEST FACILITYHeat Transfer StudiesCermet Fuel Analysis - J. M. Yatabe and G. M. Hesson

The possible thermal insulation barrier between the fuel particle and the stainless steel matrix of the FFTF cermet fuel could introduce a time delay in the negative reactivity coefficient due to fuel expansion. This could result in serious effects under fast transient power insertions. Preliminary calculations using simple analytic models have indicated that this could be a problem area, and, therefore, more exact and precise analytical models were formulated to properly evaluate this problem.

The current analytic model being used describes a single PuO_2 particle and an associated cell of the stainless steel matrix located at the position of average steady state temperature in the fuel pin. This model assumes that at low fuel particle-to-matrix temperature differentials [less than approximately 400 °F (222 °C)], a gap exists between the particle and the matrix. The mode of heat transfer across the gap was assumed to be principally conduction through the fission gases. At larger fuel pellet-to-matrix temperature differences, the differential thermal expansion causes the particle to contact the matrix. The contact or interface pressure is a linear function of the temperature differential

up to the yield strength of the softer material (stainless steel). By combining this relationship for contact pressure with the experimental measurements of Cohen, Lustman, and Eichenberg for interface heat transfer coefficients as a function of contact pressure, tables have been developed to provide interface heat transfer coefficients for any condition of contact between fuel particle and matrix. Linking the gap conductance and the contact conductance relationships provides a complete history of the fuel-to-matrix heat transfer coefficients under all possible conditions.

The power generation in the particle and unit cell of stainless steel was described by the neutron kinetic equations with reactivity feedback due to the Doppler effect and fuel expansion (the importance of negative feedback term due to sodium expansion is being investigated).

Two approaches have been made to describe the environment surrounding the fuel particle and associated matrix. The first assumes that during transient conditions, the PuO_2 particle and unit cell of stainless steel is completely insulated from its surroundings. This is a good approximation if the transient occurs in a very short period of time (less than 5 msec). However, for longer transients, this model is inaccurate since it permits large temperature rises in the stainless steel.

The second approach assumes that the particle and matrix lose heat through a portion of the fuel pin to the coolant. Preliminary numerical results have been attained for this model for step reactivity insertions from 1.4\$ to 1.5\$. These results indicate that near melting temperatures of the matrix and PuO_2 particle could be reached within a period of 5 msec after the step insertion of 1.5\$. For insertions of 1.1\$ to 1.4\$, the maximum temperature levels attained were well below the melting point. These results also indicate that the temperature levels

attained within a few milliseconds after the step insertion would change the mode of fuel pin expansion from fully elastic to partially plastic deformation. This would reduce fuel expansion, and, therefore, modifications must be performed on present analytical model to account for this plastic deformation. To more accurately simulate the thermal conditions in a complete fuel pin, a model describing a segment of the cermet fuel pin has been formulated. This model divides the PuO_2 and the matrix into 10 concentric rings of equal volume. Additional rings have been added to represent cladding and coolant. Heat transfer between PuO_2 and stainless steel was assumed to occur by the same mechanism described earlier for the particle model. The feedback terms of the neutron kinetics equations operate on the volume average temperatures of the stainless steel matrix and the PuO_2 fuel.

This analytic model is being solved by the TIGER V computer code.

Gas Cooling of Fuel Subassemblies During Inspection -

D. R. Dickinson

An analytic study was conducted to establish the feasibility of inert gas cooling of discharged FFTF fuel subassemblies during examination and disassembly. Consideration was given to driver fuel subassemblies of the reference design generating 20 kW of decay heat during interim inspection without disassembly and kW during disassembly and more complete examination. A maximum permissible cladding temperature of 600 °F was assumed; primary consideration was given to argon as the coolant. Cooling will require axial gas flow before the removal of the liner tube, and transverse flow after its removal. The study indicated that gas cooling will be feasible; however, certain problems must be overcome by proper design and adequacy of cooling demonstrated by tests. The principal problem areas disclosed by this study are:

1. Film requirements - For axial flow with the liner in place, about 300 ft³/sec of argon will be required at 20 kW; 150 ft³/sec at 10 kW. For transverse flow at 10 kW, the flow requirement is difficult to estimate as it will depend strongly on the ability to cool the fuel pins near the support and on leakage around the edges as discussed below but is estimated at about 1000 ft³/sec of argon. If helium were used, the volumetric flow required would be the same as for argon; for nitrogen, 71% as great.
2. Pressure drop - Because of the close spacing of the fuel pins, significant pressure drop will be required to provide adequate gas flow. If argon is used, the pressure drop will be about 7 psi for axial flow at 20 kW, and 0.5 to 1.0 psi for transverse flow at 10 kW. Compression to 7 psi will heat the gas by about 100 °F; it will be desirable to cool the gas before it is used to cool the fuel. These pressures will impose significant forces on the fuel bundle which must be adequately supported (about 100 lb axially and 70 to 140 lb laterally). The use of helium or nitrogen would be better in this respect than argon, as the pressure drop would be only 15 and 41% as great, respectively.
3. Spacers - The spacers in the fuel pin bundle will hinder gas cooling, especially with transverse flow. Either the reference spacer design, or wire wrap on a 10 in. pitch will block transverse flow over a length of 1/2 or 1 in. It is not possible to predict the amount of gas flow and mixing in this region without experimental data. Axial conduction can provide adequate heat removal to better cooled portions of the pin with temperature differentials on the order of 50 to 100 °F for cermet and 100 to 200 °F for oxide fuel.
4. Flow leakage - Cooling in transverse flow will require baffles which will channel the gas flow through the fuel

- bundle and prevent leakage around the sides. These must be flexible enough to permit rotation of the fuel and to function after removal of part of the fuel pins. This will make close tolerances difficult and leakage a major problem. Leakage equal to 100% of the flow through the fuel bundle would be possible with gaps on the order of 0.1 in.
5. Flow transition - During the removal of the liner around the fuel bundle, a transition must be made from axial to transverse flow. The simultaneous imposition of axial and transverse pressure gradients leads to a complex flow pattern, and it was not possible to estimate cooling requirements in this situation in this brief study. A detailed computer program and experimental data will be required.

Core Mockup Facility - K. G. Toyoda and K. D. Hayden

The Core Mockup Facility provides design testing and operational support capabilities for the reactor vessel system, the closed loop, and other critical areas in which available facilities cannot adequately meet the testing needs for the FFTF. The utility of the Core Mockup Facility extends beyond the design and construction period since the facility, closely integrated to the FTR, is essential for operating continuity, safety, and high on-line effectiveness.

As part of the Core Mockup Facility planning and assistance to the FFTF program management, liquid metal test facility capability in the United States was surveyed. Each facility reported by many industry contractors and national laboratories was identified as to its general capabilities and coded for digital computer input in terms of:

- Type of test performed on the equipment
- Coolant
- Coolant purity control

- Thermal capability
- Temperature range
- Flow rate range
- Structural material.

Similarly, each testing activity for the FFTF can be coded to show the general capabilities of facility required to perform the test. A computer program for Univac 1107 has been written to sort on (a) specific facility, listing all activities which may be performed on that facility, and, (b) specific activity, listing all facilities which may perform the particular test activity. Preliminary computer run shows the procedure and application to be satisfactory.

FFTF Work Plans describing the activities leading to conceptual system design description, WP No. 048, and preliminary/final system design description, WP No. 148, were completed. These plans cover the activities through May, 1967, at which time Title II design and the Final System Design Description are scheduled to be completed. As a planned activity of WP-048, design confirmation and proof tests required for the FFTF reactor vessel system are being investigated. Major problems thus far delineated are (1) flow mixing and distribution analyses directed to attaining 1200 °F core coolant outlet temperature without serious damage to the vessel and its internals and (2) mechanical components compatibility related to mechanical fits, retainment of safe core geometry on normal reactor transients, thermal stresses, and vibrations. A hydraulic vessel mockup for water and dry testing using a large scale reactor vessel mockup has been proposed for the preliminary studies of these problems. A document establishing the specific needs and requirements of the hydraulic vessel facility is being prepared. Studies are continuing to evaluate the program and facility requirements of a sodium containing

Core Mockup Facility which will provide the rigorous design confirmation and proof testing capabilities under prototypic coolant and thermal environment. This core mockup facility and the Hydraulic Vessel Mockup will provide complementing capabilities for the FFTF design and construction support tests.

The four industry technical reviews of the FFTF submitted by Atomics International, General Electric, Phillips Petroleum, and Westinghouse are being evaluated. Comments on the proposed Core Mockup Facility vary from recommendation to:

- Evaluate needs for the facility before proceeding past the conceptual design and survey the use of existing facilities to meet the FFTF test needs.
- Immediate implementation of core mockup design and study of increasing proposed capabilities.
- Addition of a scale model hydraulic facility.

Sodium Facilities Study - R. L. Knecht

Scope studies and advanced planning for a centralized Sodium Studies Facility to provide current equipment and technology state-of-the-art working knowledge in support of FFTF have been completed. The facility, to be located in the basement of the 324 Building, has been partitioned to separate the radioactive and nonradioactive work areas and a project proposal is being prepared for the building alterations necessary to provide adequate ventilation, electrical, water, air, fire detection, and radiation monitoring services to the laboratory.

The facility will house and service the following major equipment categories:

- Forced convection sodium flow systems
- Sodium analytical equipment
- Laboratory furnishings and safety equipment.

Laboratory equipment items that are available, on order or in shop fabrication, are sodium analytical equipment including an atomic absorption spectrometer, a gas chromatograph, a flame photo spectrometer and analytical glassware; laboratory furnishings such as radiochemical fume hoods, base cabinets, work benches and lockers; vacuum dry boxes and an associated gas purification train; safety showers and sodium fire extinguishing equipment; and an alpha, beta-gamma hand and shoe contamination checker.

Startup of the Sodium Studies Facility is targeted for late in the second quarter of FY-67. The probable critical path item to startup of the laboratory is the preparation and execution of the project proposal to provide the necessary laboratory environmental services. The time required for this could extend beyond the second quarter of FY-67. The laboratory furnishings and sodium analytical equipment should be on hand ready for installation and checkout early in the second quarter of FY-67. The Radioisotope Transport System is now under construction and is expected to be ready to operate by the end of the second quarter of FY-67.

CONCEPTUAL REACTOR DESIGN STUDIES

Nuclear Electric Plant Capital Cost vs Site Location - F. J. Kempf and D. S. Trent

Site preparation costs for the 1000 MW_e plant at the selected locations of this study can vary by a factor greater than 20. These costs ranged from \$165,000 at the Hanford site to approximately \$3.7 million at the Turkey Point, Malibu, and San Onofre sites. The highly variable site preparation costs are attributed primarily to unique topological, geological and excavation or dredge and fill requirements at the various locations.

Cooling tower costs, based on an induced draft recirculating coolant installation, were estimated at \$9.2 million at Hanford and approximately \$10.1 million at the other river sites. These estimates are direct construction costs which include \$1.5 for cooling tower pumps and piping at each site.

A \$9.1 million estimate was developed for maximum marine line costs on the basis of 16 ft diam, reinforced concrete pipes having 2600 ft discharge and 3200 ft intake line lengths. This estimate includes marine intake and discharge tower costs but excludes power plant intake and discharge structures which are included in the reference plant construction cost estimates.

A draft report on the plant cost versus siting study was completed and forwarded to the AEC for comment. Additional data on cooling tower cost as influenced by design considerations are being reviewed and this section of the report will be revised prior to publication. In addition some cost information on sea water versus fresh water condenser installations is being developed.

USAEC-AECL COOPERATIVE PROGRAM

Cross-Flow Mixing Between Parallel Flow Channels During Boiling - 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 There is presently insufficient information concerning mixing with two-phase flow in multirod bundles. Some information describing mixing in single-phase flow exists, but this information cannot be extrapolated to two-phase flow.

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 subchannel pressure. Both types of inter-channel 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 flow* between adjacent channels may be superimposed.

These equations were derived by considering one subchannel of a rod bundle as influenced by all adjacent subchannels which mix with it both through turbulent cross flow and diversion of flow into or out of the channel. These equations were presented in the October-December, 1965 quarterly report and are again presented for reference.

$$\frac{dm_i}{dx} = - \sum_{j=1}^N W_{ij} \quad i = 1, 2, \dots, N \quad (1)$$

$$m_i \frac{dh_i}{dx} = q_i + \sum_{j=1}^N w'_{ij} (h_j - h_i)$$

$$- \sum_{j=1}^N \left[\begin{array}{ll} 0 & ; \text{ if } W_{ij} \geq 0 \\ W_{ij} (h_j - h_i) & ; \text{ if } W_{ij} < 0 \end{array} \right] \quad i = 1, 2, \dots, N \quad (2)$$

* There is presently no way to compute these cross flows analytically; however, correlations based on experimental data appear possible.

$$- A_i \left[g + (\rho_i u_i)^2 \frac{\partial v'_i}{\partial p} \right] \frac{dp_i}{dx} = \frac{m_i^2}{\rho_i A_i} \left[\frac{f_i K_i \rho_i}{2 D_i} + \rho_i \frac{\partial v'_i}{\partial h} \frac{dh_i}{dx} \right]$$

$$+ \sum_{j=1}^N W'_{ij} f_M (u_i - u_j) + g \rho_i A_i \cos \theta$$

$$+ \sum_{j=1}^N \left[W_{ij} f_D u_i \left(1 - 2 \frac{v'_i}{v_i} \right); \text{ if } W_{ij} \geq 0 \right]$$

$$\left[W_{ij} f_D \left(u_j - 2 u_i \frac{v'_i}{v_i} \right); \text{ if } W_{ij} < 0 \right]$$

$$i = 1, 2, \dots, N \quad (3)$$

$$p_i - p_j = C_{ij} |W_{ij}| W_{ij} \quad (4)$$

Where

- m mass flow rate, $A u_f [\rho_g \alpha + \rho_f (1 - \alpha)]$
- x space variable
- w diversion cross flow between adjacent channels
- w' turbulent (fluctuating) cross flow between adjacent channels
- h enthalpy, $X h_g + (1 - X) h_f$
- h_g, h_f saturated vapor and liquid enthalpy
- q' heat addition per unit length
- g gravitational constant
- A cross-sectional area
- K two-phase friction multiplier
- D hydraulic diameter
- ρ density, $\rho_g \alpha + \rho_f (1 - \alpha)$

ρ_g, ρ_f	saturated vapor and liquid density
f	friction factor based on all liquid flow
v	specific volume, $1/\rho$
v'	effective specific volume for momentum considerations, $(1 - X)^2/\rho_f (1 - \alpha) + X^2/\rho_g \alpha$
u	channel effective velocity, $m/\rho A$
X	quality, $m_g/(m_g + m_f)$
γ	slip ratio, u_g/u_f
α	void fraction, $A_g/(A_g + A_f)$
c	cross flow resistance constant
f_M	turbulent mixing correction factor
f_D	diversion cross flow correction factor
N	number of subchannels
i, j	subchannel designation
ij	double subscript refers to condition from i to j
f, g	saturated fluid and vapor

These equations of continuity, energy, and momentum are complicated nonlinear equations and only for very simple cases, such as two parallel channels and single-phase flow, have closed form analytical solutions been obtained. For a more general case, a numerical solution to the equations is required.

Computer Program Development. Development of the digital computer program which solves the mixing equations was continued.

The mixing equations are currently being treated as an initial value problem. It is more truly a boundary value problem, however, due to the very nearly equal pressures

between rod bundle subchannels, 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. By use of 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 is currently being used. It has given stable solutions for two parallel channels both with and without boiling. By this technique, the mixing equations are integrated in the following manner:

$$m_{i,x+\Delta x} = m_{i,x} + \frac{\Delta x}{2} \left(\frac{dm_{i,x}}{dx} + \frac{dm_{i,x+\Delta x}}{dx} \right) \quad (5)$$

$$h_{i,x+\Delta x} = h_{i,x} + \frac{\Delta x}{2} \left(\frac{dh_{i,x}}{dx} + \frac{dh_{i,x+\Delta x}}{dx} \right) \quad (6)$$

$$p_{i,x+\Delta x} = p_{i,x} + \frac{\Delta x}{2} \left(\frac{dp_{i,x}}{dx} + \frac{dp_{i,x+\Delta x}}{dx} \right) \quad (7)$$

The gradients at x in Equations (5,6,7) are known from the initial conditions. If the step size is small enough to give small changes in flow, the gradients in Equations (5) and (6)

can be estimated from conditions at x . In Equation (7), the pressure gradient at $x+\Delta x$ is greatly influenced by the amount of diversion cross flow, which cannot be estimated from conditions at x . If Equation (3) is substituted into Equation (7), it can be seen that the pressure $p_{i,x+\Delta x}$ is a linear function of the diversion cross flow when the flow and enthalpy estimates at $x+\Delta x$ are made. By use of this relationship and Equation (4), the solution for the diversion cross flow at $x+\Delta x$ can be obtained. This new value can then be used to update the previous values of flow and enthalpy at $x+\Delta x$. The cycle is then repeated until agreement between the new and prior values of diversion cross flow is obtained.

Digital computer calculations are currently in progress to determine how well this technique will apply to more than two channels.

Experimental Program. An experimental program was undertaken to verify the analytical description of mixing and to obtain values of the interchannel mixing. The electrically heated test sections being used for these experiments closely simulate two adjacent subchannels in multirod bundles. A schematic drawing of the experimental apparatus is shown in Figure 1.

At the inlet of the test sections, flow is introduced to the two subchannels through separate inlets. The total flow is extracted from the outlet of each subchannel and condensed and cooled by the subchannel heat exchangers. By measuring the inlet and outlet flow and enthalpy, the amount of mixing can be determined from a heat balance on the heat exchangers. Pressure measurements at the inlet, outlet, and several intermediate points of each subchannel are used to set the proper end conditions. Also, from the pressure drop measurements

FIGURE 1. Parallel Channel Experiment Loop Instrumentation

between the subchannels at the intermediate pressure tap locations, lateral flow resistance can be determined.

During boiling, both flow diversion and turbulent cross flow occur simultaneously. This makes measurement of the diversion cross flow difficult if the turbulent cross flow is large. To more accurately measure the diversion cross flow, two different shapes of test sections are to be used in the experiments (Figure 2). The first is one which simulates two adjacent subchannels of the "square" type as in a fuel bundle with a square pitch array. In this test section, flow diversion should be negligible because of the equal heat input and flow area of the subchannels. To measure the amount of fluctuating cross flow, a small amount of heavy water will be

Sizes Shown are Approximately 2-1/2 Times Actual Size

FIGURE 2. Test Section Shapes for Parallel Channel Experiment

continuously injected into the inlet of one channel. Since heavy water has thermodynamic properties similar to light water, heavy water vapor and liquid would be transported between subchannels during boiling. By measuring the heavy water concentrations at the outlet of the two streams, the amount of turbulent cross flow during boiling can be determined. This test section is about 30% complete and will be run later in the experimental program.

The second test section shape will simulate two adjacent subchannels of the "square" and "triangular" types as in a 19 rod bundle. In this test section, both types of cross flow will occur with diversion being significant during boiling. From the values of turbulent cross flow determined from tests using the first test section shape, the amount of diversion flow can be determined more accurately. This test section has been completed and installed in the experimental facility. Experiments are currently in progress.

Parallel Channel Reactor Boiling Stability Study

Neutron Kinetics - Coupled Reactor Theory - J. R. Worden

Results obtained to date from the analog computer indicate the need for an improved nuclear kinetics model. Hydraulic instability typically occurs at power levels of about twice the normal operating power at which point the light water coolant boils over nearly the entire length of the core. The present 4-node model⁽¹⁾ fails under this condition since the boundary between the boiling and subcooled regions defines the axial

(1) H. Harty, et al. Engineering Section Quarterly Report, Reactor and Materials Technology Department, BNWL-219, Pacific Northwest Laboratory, Richland, Washington, October, November, December 1965.

nodes. An alternative model, based on coupled reactor theory,⁽¹⁾ has been examined and, in addition to eliminating the problem described, offers the further advantages of simpler kinetics equations and a more rigorous formulation.

The digital computer code, HFN,⁽²⁾ has been modified to provide the parameters needed for a 2-node kinetics model based on coupled reactor theory. The kinetics equations for the 2-node model, assuming a single group of delayed neutron, are

$$(\dot{P}L)_i = (1 - \beta) \sum_j k_{ij} \frac{(PL)_j}{l_j} - \frac{(PL)_i}{l_i} + \lambda \sum_j k_{ij} C_j$$

and

$$\dot{C}_i = \beta \frac{(PL)_i}{l_i} - \lambda C_i \quad i, j = 1, 2. \quad (1)$$

where

$(PL)_i$ = power level in the i th node

$(\dot{P}L)_i$ = $d(PL)_i/dt$

and

l_i = mean lifetime of neutrons absorbed in i th node.

- (1) R. Avery. "Theory of Coupled Reactors," Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva 1958, P/1858, vol. 12, p. 182, United Nations, Geneva, 1958.
- (2) J. R. Lilley. Computer Code HFN - Multigroup Multiregion Neutron Diffusion Theory in One Space Dimension, HW-71545. General Electric Company, Richland, November 17, 1961.

The coupling coefficients, k_{ij} , and the lifetimes, l_i , are obtained from a calculation of neutron spatial distributions in the static core, with appropriate direct and adjoint flux weighting of nuclear cross sections. The parameter, k_{ii} , is a measure of the criticality of the i th node without the contribution of the other node.

To test the modifications to HFN, calculations were made for a 2-zone (UO_2 and PuAl) fuel loading⁽¹⁾ in the Plutonium Recycle Test Reactor (PRTR). The calculated neutron production rates, used as weighting functions in obtaining the k_{ij} and l_i , are shown in Figure 3 and the adjoint fluxes are shown in Figure 4. Identifying the UO_2 zone with the subscript, 1, and the PuAl zone with the subscript, 2, the following coupling coefficients are calculated:

$$k_{11} = 0.828 \text{ (UO}_2\text{)}$$

$$k_{22} = 0.659 \text{ (PuAl)}$$

$$k_{12} = 0.367$$

$$k_{21} = 0.170$$

These coefficients can be combined to give an effective multiplication factor, k , for the critical ($k = 1.000$) reactor of $k = 1.007$, indicating adequacy of the calculational methods used in HFN.

(1) PRTR Physics Subcouncil. Plutonium Recycle Test Reactor Critical Test Results, HW-61900 BA. General Electric Company, Richland, Washington, December 1961.

FIGURE 3. Rate of Fission Neutron Production: PRTR 2-Zone Loading, UO_2 and PuAl

FIGURE 4. Thermal Neutron Importance: PRTR 2-Zone Loading, UO_2 and PuAl

Development of Hydrodynamic Equations - D. S. Rowe

In the previous Quarterly Reports,^(1,2) a description of the basic mathematical model being used for this study was presented. This included the equations for the system hydrodynamics and neutron-kinetics. Also included was a description of the method being used to compute fuel temperature and steam drum pressures. During this quarter, changes have been made to the neutron-kinetics model, and an improved method of defining the constants and functions appearing in the hydrodynamic equations has also been made.

Limitations on the amount of analog computing equipment require that a coarse segmentation of the system be made. For the chosen segmentation, it is difficult to choose average constants and functions for the finite difference equations. To define these more accurately, an integration of the partial differential equations is made by assuming a linear variation in the variables, stepwise changes in areas and lengths, and steady state distribution of functions which are dependent upon enthalpy. The resulting equations are of the same form as the previous finite difference equations but with the constants and functions more accurately defined.

The hydrodynamic equations have been programmed on the analog computer and debugged. Calculations to check the steady state and transient response were performed to compare with previous calculations made by AECL personnel using their modified version of HYDNA. The calculations were performed for a BLW-250 central channel by using friction constants, areas and lengths which produce pressure drops established

- (1) H. Harty, et al. Engineering Section Quarterly Report, Reactor and Materials Technology Department, October November December, BNWL-219, Pacific Northwest Laboratory Richland, Washington. 1965.
- (2) H. Harty, et al. Engineering Development Department Quarterly Report, January February March, BNWL-300, Pacific Northwest Laboratory, Richland, Washington. 1966.

for the central channel reference design (Table I). Analog calculations were made to determine the response of the channel flow rate to small step increases in channel heat flux (Figure 5). A comparison of the results from the analog and digital computations shows reasonable agreement between the two calculations, especially near the nominal pressure drop of 178 psi. Oscillations in the inlet flow rate appeared only when very close to the unstable point and with a period of about 0.3 sec. The digital calculations gave some flow oscillations farther from the unstable point and with a period of about 0.6 sec.

Table I. *BLW-250 Cental Channel Reference Conditions for Analog Computer Study*

Average Mass Velocity	3.41×10^6 lb/hr ft ²
Average Heat Flux	0.174×10^6 Btu/hr ft ²
Inlet Enthalpy	504 Btu/lb
Steam Drum Pressure	765 psia
Average Exit Quality	16%
Pressure Drop:	
Inlet Header to Fuel	45.2 psi
Fuel to Outlet Riser	103.9 psi
Outlet Riser to Steam Drum	29.4 psi
Total	178.5 psi

Some of the basic differences between the analog and digital calculations can help explain some of the differences seen in Figure 5. The frictional pressure drop in the analog calculations are based on the square of the flow rate whereas the digital calculations are based on a flow raised to a power less than 2.0. The effect of this is to cause the slope of the flow rate versus heat flux curve at constant pressure drop to be less for the analog simulation. Also, the two-phase friction multiplier being used for the digital calculations (Martinelli-Nelson) is about twice that of the Armand multiplier being used for the analog simulation. Analog

calculations were performed to compare this effect by using adjusted friction to give the same reference pressure drops when the Martinelli-Nelson multiplier was used. The effect was to cause only a small increase in the slope of the flow rate versus heat flux curve. The threshold flow was about the same and the threshold power a little less.

The differences found in the period of oscillation can probably be attributed to the size of the channel segments used in the difference equations. The period of oscillation is a function of the transit time through the two-phase piping. With only two nodes with two-phase flow, the effective transit time is faster than if a large number of segments were used. This could produce a shorter period of oscillation.

FIGURE 5. *Threshold of Instability Comparison Using Analog Computer Model and Digital Computer Program, HYDNA*

Evaluation of Zirconium 2.5% Niobium Pressure Tubing -
P. J. Pankaskie

The general objective of this program is to evaluate Zr-2.5 wt% Nb alloy tubing as a pressure tube material with reference to tests and reactor experience that has been obtained on Zry-2 pressure tubes.

Continuation of crack propagation tests has produced the following results:

Specimen Identity	Speciman Condition	Test Temperature, °C	Slot Length, in.	Hoop Stress at Failure, psi	Crack Length at Failure in.
30-12	30% CW Hydrided (250 to 300 ppm)	300	1 1/2	39,000	Brittle, propagated full length of test piece
60-15	60 CW Hydrided (250 to 300 ppm)	150	1/2	88,000	Brittle, propagated full length of test piece
30-23	30% CW	-30	1/2	115,000	4 in. from each end of slot - shear
30-24	30% CW	-30	1 1/2	51,000	1 to 2 in. from each end of slot - shear
30-25	30% CW	0	1/2	111,000	3 in. from each end of slot - shear
30-26	30% CW	0	1 1/2	50,500	1 1/2 in. each end of slot - shear
30-27*	30% CW	0	2 1/2	55,000	3 in. from each end of slot - shear
30-28*	30%	0	4	44,000	3 in. from each end of slot - shear
60-24	60% CW	-30	2 1/2	34,000	1 1/2 in. each end of slot - shear
60-21	60% CW	-30	4	17,500	1 in. each end of slot - shear
60-26	60% CW	-30	4	20,500	1 in. each end of slot - shear

Tests 30 to 27 and 30 to 28 employed a thin "bladder" to retain pressure in the test specimen after the web of the slot had failed. In test 30 to 28 a 30 mil wall aluminum tube was first tried. In this trial the aluminum "bladder" failed in a nonductile manner and at a pressure below which crack propagation occurs. This test was repeated by using a 30 mil wall, Zry-2 "bladder" with the test results as shown. Test 30 to 27 employed the 30 mil wall Zry-2 bladder which was inserted in the test assembly prior to failure of

the slot web. Comparison of these data with data obtained by different techniques show little difference in the hoop stress to cause failure or to initiate the propagation of the crack in the web of the milled slot. If the bladder technique is used, ductile shear type cracks can be forced to propagate so long as a sufficiently large hoop stress can be maintained by the bladder. This same technique has been applied successfully to tests of hydride test pieces at 150 °C and 300 °C.

Drop weight tests⁽¹⁾ were made on specimens cut from both the 30% CW and 60% CW tubing. The purpose of these tests was to determine the nil ductility temperature (NDT). From these tests, the NDT for the 30% CW temper occurs at about room temperature while for the 60% CW temper the NDT appears to be somewhat below -60 °F. Previous metallographic examinations revealed that the 60% CW materials were low in hydride content (est ~10 ppm or less) while the 30% CW contained more hydride (est ~40 ppm). Further metallographic examinations are in progress.

Two PRTR tests for irradiation of Zr-Nb components have been initiated. PRTR test 125 involves the exposure of a special Zr-2.5 Nb alloy pressure tube. This tube was fabricated by AECL, Chalk River, to PRTR process tube specifications and has been installed in Process Channel 1449. In the location it will "see" service comparable to the standard process tubes except that its nominal hoop stress during reactor operation will be 19,000 psi as compared to 12,500 psi for standard tubes. Special diameter measurements will be

(1) R. C. Aungst. Additional Crack Propagation Tests on Zircaloy-2 Reactor Pressure Tubes, BNWL-SA-179, Pacific Northwest Laboratory, Richland, Washington, Sept. 20, 1965.

obtained by AECL personnel at appropriate exposure intervals to determine the effects of neutron flux and hoop stress on the creep behavior of 2.5 Nb alloy.

PRTR test 127 is designed to obtain data on the effects of neutron exposure on the crack propagation characteristics of Zr-2.5 Nb alloy tubing. Four test specimens will be irradiated as an assembly in a nonfueled process channel. Two test specimens have been hydrided to about 250 to 300 ppm, and two comparable specimens are in the as received cold worked condition.

Further evaluation of use of venturis for measuring diametral creep in the 350 °C corrosion loop is continuing. Flow fluctuations continue to be a major source of error in attempting to determine creep deformation. Pressure snubbers have been obtained and have been installed in an attempt to damp-out flow fluctuations. The use of commercial "snubbers" and "needle valves" do reduce the fluctuation in the DP cells by substantial amounts. At flow rates of about 40 gpm through the test apparatus, random fluctuations of as much as 2 in. of water are still observed. Without "snubbers" fluctuations of 10 in. or more would have been observed at similar flow rates.

APPLIED AND REACTOR PHYSICS

HIGH TEMPERATURE LATTICE TEST REACTOR

Safety Rods - M. R. Kreiter

As mentioned in the previous quarterly report,⁽¹⁾ a problem was encountered with excessive boron transport from the B₄C poison material used in the safety blade during run Number 4.

(1) H. Harty, et al. Engineering Development Department Quarterly Report, BNWL-300, Pacific Northwest Laboratory Richland, Washington, January-March 1966.

Apparently the water in the cooling gas combined with the boron in the poison material to form boric acid. The boric acid then vaporized and was carried to the hotter regions of the mockup where it was disassociated to form boron oxide which precipitated on the metal surface of the rod. Figure 6 shows droplets of boron glass on the surface of the rod.

There are also corrosion problems associated with this phenomena. Figure 7 is a photograph of the corrosion present between the TD nickel sheath and B_4C graphite poison material. No further analysis of this problem was made since the boron transport problem dictated a change of poison material.

The material selected as a replacement material was co-precipitated rare earth oxides (gadolinia and europia) in nickel. Samples of this material were procured. These performed satisfactorily under laboratory testing. Quantities of this material were ordered for use in the safety blade, and the blade design was modified to allow use of this material.

Oscillator Development - D. H. White

As mentioned in the previous quarterly report,⁽¹⁾ selection of an adequate drive system for the oscillators was an important step in the oscillator development program. This drive system evaluation was completed. A total of eight drive system concepts were evaluated from a technical and economic point of view. This evaluation resulted in the selection of an electro-hydraulic servo type system for use on the oscillators.

(1) H. Harty, et al. Engineering Development Department
Quarterly Report, BNWL-300, Pacific Northwest
Laboratory, Richland, Washington, January-March 1966.

Neg 0653399-6

FIGURE 6. *Droplets of Boron Glass on the Rod*

Neg 0653456-2

FIGURE 7. Corrosion Between the TD Nickel Sheath
and B_4C

The majority of the design modifications were completed and fabrication of all components was started. Procurement of all major and/or long-term delivery items was started. The design of a cold testing mockup for the oscillators was completed. Fabrication of this mockup was started and procurement of the electronic equipment, used to operate the oscillators while under test, was completed.

Phoenix Fuel Program - D. S. Trent

Heat transfer and hydraulic analysis in support of the MTR-Phoenix fuel experiment was continued on the symmetric 3 x 9 partial plutonium core. Reactor power capability, as governed by thermal limits, was estimated for different power distributions over a range of horizontal and axial power peaking factors. These calculations were based on the fuel box assembly considered in previous work which has 16, 80 mil thick fuel plates, including 20 mil aluminum cladding, and otherwise standard MTR fuel box dimensions.

Engineering Physics has estimated three-dimensional power distributions for two different control rod positions at reactor startup. One power distribution was for control rods inserted halfway into the core, and the other for a case where the core is about 60% rodded. No fuel shading was employed to reduce localized power peaking, nor were flux suppressors considered in either case.

Standard MTR thermal hydraulic calculations were performed with the aid of a computer program, written especially for this purpose, to establish reactor power capability for a range of axial and horizontal peaking factors. In previous work, the position and magnitude of the axial power peaks was assumed since this information had not yet been calculated. The latest nuclear calculations

have provided this information, and it has been determined that the most severe peaking conditions occur at the bottom of the core. In the case with the control rods inserted halfway into the core, the maximum power peak occurs in a plutonium fueled control rod near the top of the fueled section or about halfway through the core.

Hydraulic conditions at this point in a control rod are similar to those at the top of the core for stationary fuel assemblies; therefore, the coolant pressure and saturation temperature is significantly higher than at corresponding axial points in stationary fuel assemblies. As a consequence, thermal hydraulic estimates indicate that this peak does not limit the reactor power as severely as the power peaks occurring at bottom of the stationary fuel assemblies.

Figure 1 (Page 16) illustrates the reactor power capability as a function of horizontal peaking factor for various axial peaking factors. In all cases the fuel element nominal surface temperature, which was determined to be the factor governing reactor power, was limited to 245 °F (118 °C). For the half-rodded and 60% rodded cores, the axial peaking factors were about 2.0 and 3.1, respectively. In both cases the horizontal factor was about 1.8, resulting in overall peaking factors of 3.6 and 5.6. As indicated by the calculations graphically presented in Figure 8, the maximum reactor power for the 50% and 60% rodded cases would be about 28 and 19 MW, respectively.

A study is presently underway to investigate possible power reactor core designs for Phoenix fuels and the applicability of the MTR experiment to Phoenix fueled power reactors in the 100 to 300 MW_t range. Several fuel types and designs are being analyzed in a number of different core configurations.

Of primary interest is a comparison of pin and plate type fuel elements from both a physics and engineering standpoint,

FIGURE 8. Reactor Power Versus Horizontal Peak Factor for Plutonium Fuel Box

since the pin geometry is customary in power reactors, whereas plate geometry will be used in the MTR experiment. Engineering Physics is presently performing nuclear calculations for several preliminary core configurations having different metal-to-water ratios, pin sizes, and fuel compositions. The design analysis to establish the thermal hydraulic limits and heat transfer capabilities of these cores is in progress.

REACTOR FUELS AND MATERIALS

IRRADIATION DAMAGE TO REACTOR METALS

ATR Gas Loop Support

Model Gas Loop - P. M. Jackson

The installation of the new model loop heater was completed on April 6, 1966. Since this time the system has been operated on a three shift per day, five days per week basis except for a short operator training period and maintenance periods. This extensive testing of the model heater is being done to determine whether or not the design parameters used on the ATR heater will meet the stringent ATR gas loop requirements. The testing program for the model heater includes:

1. Operation at various flows and temperatures to determine the heater's pressure drop and heat transfer characteristics. At several flow levels various power ratios between the nichrome and molybdenum heater sections will be run while maintaining a constant outlet temperature.
2. Step power settings to determine the short time transient characteristics of the heater.
3. Accelerated thermal cycling to evaluate the structural integrity during shut down cycles which are equal to or

surpass most severe temperature shocks that are anticipated during the ATR loop operation, such as reactor scram and loss of all power.

4. Extended operation at maximum temperature to insure long term reliability.

The model heater has been operated for a total 975 hr with 25 thermal cycles between room temperature and various heater outlet temperatures. The heater has operated about 65% of the time with an outlet gas temperature of 2100 °F (1150 °C). The original heater has been disassembled and specimens are being removed for examination. This heater operated about 690 total hours during its 1 1/2 years in the Model loop. The gas bearing compressor has operated faultlessly for a period of about 1660 total hours and has undergone about 165 total starts.

Figure 9 is representative of a complete power failure in which all heaters, the gas circulator and associated equipment would fail simultaneously. The curves show the System Temperature versus time after shutting off the electrical power.

Figure 10 is similar to Figure 9 except this represents the most severe thermal excursion possible with the model loop. In this test the heater power was shut off, but the gas circulator was left operating. After a period of one hour the circulator was shot down.

ATR Gas Loop Heater - G. L. Fox

The ATR gas loop heater is being designed to heat helium flowing at a rate of 4140 lb/hr to a temperature of 2100 °F (1150 °C). The heating is accomplished in two stages with electric resistance heating elements. The gas first passes through an annular ring of nickel-chromium

FIGURE 9. Temperature of Model Gas Loop Heater as a Function of Time After All Power Was Shut Off

FIGURE 10. Temperature of Model Heater as a Function of Time After Heater Was Turned Off with the Circulator Continuing to Run

tubes, makes a 180° turn, and then passes through a center core of molybdenum tubes. Both tube banks are supported with aluminum oxide ceramic. The heater internals are insulated from the outside conditions with metallic foil insulation.

The performance of the ATR gas loop is being simulated with an analog computer by an offsite group. Various constants, which are necessary for this analysis have been calculated for the heater and transmitted to this group. Among these constants were the thermal response time of the thermocouple wells. Because of the tapered geometry, it was necessary to use a transient heat transfer program, Tiger V, to obtain the time constant. Results were obtained for both high and low gas flow rates.

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 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.

Two groups of laboratory experiments were performed on a uniformly electrically heated test section 0.435 in. ID and 12 ft long heated. The first group of tests was run with water at a mass velocity of 3×10^6 lb/hr-ft² and a downstream pressure of 1500 psig. As the inlet quality was increased from 0 to 50%, the burnout heat flux decreased from 360,000 to 250,000 Btu/hr-ft². The burnout location moved

progressively from the downstream end to a point 3 ft from the upstream end as the inlet quality was increased to 30%, but did not move any closer to the upstream end with further increases in quality. This behavior is generally similar to that previously observed with a 6 ft long test section. Of particular significance, upstream burnout was not accompanied by a rise in burnout heat flux with increasing quality, as had been found at higher mass velocities.

Upon attainment of boiling burnout conditions, further increases in heat flux resulted in only moderate wall temperature rises; therefore, it was possible to continue the test beyond the first indication of burnout. When the heat flux was increased over the burnout value, the region of burnout spread until, had the test continued far enough, it would have extended over most of the length of the tube. In several cases, the point of strongest burnout indication (highest wall temperature) shifted upstream. The test section had rings of thermocouples 90° apart at the midpoint and at the downstream end. These showed a very large circumferential variation in the burnout heat flux (as great as a factor of 1.7). Earlier, similar variations had been found at the downstream end of the 6 ft test section. This is a major unexpected and unexplained phenomenon encountered in these tests. The last test (50% inlet quality) was continued to a heat flux of 700,000 Btu/hr-ft², or 2.8 times that at the first burnout indication. In this test, 100% outlet quality was attained. At the conclusion of this test, the inlet quality was gradually reduced from 50 to 10% with the heat flux held constant at 540,000 Btu/hr-ft². Initially, almost the entire test section was in a post-burnout state. In general, the downstream wall temperatures fell somewhat, while very large increases were noted near the upstream end.

The second group of experiments was run with water at a mass velocity of 7×10^6 lb/hr-ft², downstream pressure of 1500 psig, and inlet conditions ranging from slightly sub-cooled to 15% quality. In these tests, the burnout heat flux remained constant with increasing enthalpy. The burnout location moved upstream until it reached a point 3 ft from the upstream end. Then, on further increases in enthalpy, the burnout location remained stationary, but the burnout heat flux rose from 1.07 to 1.28×10^6 Btu/hr-ft². This behavior is in general agreement with previous results with a 6 ft heated length.

PLUTONIUM RECYCLE PROGRAM

REACTOR ENGINEERING DEVELOPMENT

Heat Transfer Calculations Concerning FERTF Fuel Element Basket Basket - G. M. Hesson

It has been proposed that a ribbed "basket" be placed around the fuel element in FERTF to prevent direct contact between the pressure tube and any molten fuel which is ejected from the fuel elements. Calculations were performed to predict the effectiveness of such a basket in limiting the maximum temperature to which the pressure tube is subjected.

These calculations assumed that a strip of molten fuel approximately 0.5 in. thick, 0.5 in. wide, and indefinitely long is in perfect thermal contact with the basket. Four possible fuel-basket configurations have been studied:

- Fuel contacts the 1/8 in. thick basket at a location where it has no supports between it and the pressure tube, and the basket is deformed until it contacts the pressure tube.
- Fuel contacts the 1/8 in. thick basket at a location opposite a 1/8 in. thick rib between the basket and the pressure tube.

- Fuel contacts the 1/8 in. thick basket at a location opposite a 1/16 in. thick rib between it and the pressure tube.
- Fuel contacts a basket which is 1/4 in. thick with 1/8 in. deep axial grooves 1/8 in. apart around the circumference of the basket.

The calculations for the first case show that if perfect thermal contact between the basket and pressure tube exists, the maximum average temperature across the tube would be about 2100 °F (1150 °C). However, more realistic values of the thermal contact gave temperatures of 1200 to 1500 °F (650 to 815 °C). At these temperatures, a Zircaloy pressure tube would be expected to fail; a Hastelloy pressure tube would be expected to have a time-to-failure of several to an indefinite number of hours.

The second case gave maximum basket temperatures of 735 °F (380 °C). At this temperature, either a Zircaloy or Hastelloy pressure tube would be expected to have an indefinite life.

The third case gave maximum basket temperatures of 890 °F (480 °C). At this temperature, a Zircaloy pressure tube would have a time-to-failure on the order of 10,000 hr and a Hastelloy pressure tube would have an indefinite life.

The fourth case gave maximum temperatures of 738 °F (380 °C) if coolant is maintained in the grooves. Either a Hastelloy or Zircaloy pressure tube would be expected to have an indefinite life at this temperature. However, if one of the grooves should become blocked so that the coolant flow were stopped, the pressure tube maximum temperature would be expected to increase to 1370 °F (740 °C). At this temperature, a Zircaloy pressure tube would be expected to promptly fail and a Hastelloy pressure tube would have a time-to-failure on the order of 1,000 hr.

PRTR Moderator-Reflector System Modifications -

M. R. Kreiter

The design of modifications required to isolate the PRTR moderator and reflector systems was completed and approved. The associated design change was approved and issued. All material and equipment required to complete the modifications was procured and the installation of equipment is estimated 80% complete.

PRTR Shroud Tube Removal Machine - D. E. Rasmussen

Fabrication of the shroud tube removal machine and assembly of the machine over the PRTR mockup was completed during the quarter. Figures 11, 12, and 13 show various parts of the assembled machine. The drive motor on the right and its associated gear train provide vertical translation of the spindle; the motor at the center and its associated gearing provide rotation of the spindle (Figure 11). The rotational drive motor will also provide translation of the spindle, when the proper gearing is used, to allow a spiral cut in the tube through the roll joint area at the bottom of the calandria. In the upper part of Figure 11 is the shroud which covers the slip ring assembly. This assembly is used to carry power and control signals to the spindle motors shown in Figure 12. The upper motor is the cutter drive motor; the two smaller motors directly below it control cutter depth and lock the tool head when the bottom weld or bearing pad cuts are made. These two motors are digital stepping motors, thus, their position may be accurately controlled. The three motors (Figure 12) transmit their motion through shafts down the center of the spindle (lower part of photograph) to the cutter head attached to the lower end of the spindle, (Figure 13). This photograph shows the 6 in. weld cutter head at completion of a test run. The test shown here was a simulated top weld cut. Other heads

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FIGURE 11. Main Drive Section of Shroud Tube
Removal Machine

Neg 0653292-4

FIGURE 12. *Spindle Motors in Shroud Tube Removal Machine*

Neg 0653292-6

FIGURE 13. *Cutter Head Attached to Lower
End of Spindle*

which are available for the machine are shown in Figure 14. From left to right they are the tube puller ring and head, the bottom weld fly cutter, the bearing pad cutter, and the spiral tube cutter. These four heads are all of the 4 in. size. These heads are also available in the 6 in. size except for the bearing pad cutter.

The machine has successfully cut the top and bottom welds, and has spirally cut a 6 in. tube in the mockup.

A rough draft of the operating manual for the machine was completed.

MATERIALS DEVELOPMENT

PRTR Pressure Tubes - M. C. Fraser

The PRTR pressure tubes were fabricated from Zircaloy-2 ingots by a forging-hot extrusion-tube reducing process. These tubes have a unique shape, one end is flanged and the other tapers to a reduced diameter. They also have a unique micro-structure distribution, there is an annealed region between the taper and about tube mid-length, the remainder is cold worked. The ability of these tubes to sustain PRTR system loads with ever increasing neutron exposure has been a subject of study and testing since reactor start up. The following paragraphs describe the study and testing done during the last quarter year.

Crack propagation tests have been performed over the temperature range 21 to 149 °C to obtain data applicable to crack length of 0.75, 2.0, and 2.8 in. The cracks were established in the wall of the tube specimens by pressurization which induced fracture at the base of a machined slot. The machined slots were 80% of the wall deep and were the desired crack length.

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FIGURE 14. Various Heads Available for Shroud Tube Removal Machine

Lines plotted on the graph in Figure 15 show the trends of the data obtained from these tests. The trend line of previously obtained data for approximately half-inch cracks is also shown. They show that the stress at crack propagation of a given length crack decreases with increasing test temperature. The rate of decrease is insensitive to crack length because the trend slopes are minus 10,500 psi stress per 100 °C plus or minus 5%. The values of stress at propagation are normalized values of the pressure load at propagation. They were calculated with Barlow's mid-wall formula which, for the PRTR tube geometry yields about the same values as the maximum Lamé formula.

FIGURE 15. *Trend of the Effect of Temperature Upon Hoop Stress at the Onset of Crack Propagation*

Properties of Irradiated PRTR Pressure Tubes - M. C. Fraser

PRTR pressure tube 6596 was discharged out of channel 1948 for the purpose of postirradiation destructive examination in accordance with PRTR reactor safeguards policy. Prior to cutting the tube into test specimens, it was ultrasonically examined and its average diameter was measured at several locations along its length. Five 20 in. lengths were cut from the tube for burst or crack propagation testing. Five 2 in. lengths were cut from the tube for metallography and hydrogen analyses. A short piece of the taper section was obtained for examination of a suspect area on its inside surface. This suspect area had been discovered during in-reactor monitoring.

The taper section was moved to Radiometallurgy where the suspect area was studied. A positive replica of the suspect area was made and the wall of the taper section in the region of the suspect area was measured. These two pieces of information were convincing indications that the suspect area, a mark which appeared to have been made by a chiseling action and which extended nearly around the inside circumference, was produced by a component from a process tube alignment tool. The examination of this suspect area is not complete.

NUCLEAR SAFETYCONTAINMENT SYSTEMS EXPERIMENTFacilities and Equipment - C. E. Linderoth and G. J. Rogers

Construction activities were directed primarily at readying the containment vessel for leak tests and initial aerosol transport tests. In addition, in order to accelerate the overall CSE program, construction was started on a separate facility for blowdown and hydraulic studies employing only the reactor simulator vessel.

Significant construction features completed during the quarter were the liquid sampling system, the external portion of the aerosol sampling system, compressed and liquid gas storage dock, and the aerosol injection line. Significant progress was made on the fabrication of Maypack clusters, the high pressure loop piping, and the Dowtherm heat exchanger system.

The scope and design criteria for the Phase I hydraulic core for the simulator were completed.

Aerosol Behavior - Effect of Cladding Type - R. K. Hilliard

The fission product aerosol which will be used in CSE research studying fission product transport in containment will be generated by volatilizing radioactively labeled stable fission product elements or compounds and passing these through the hot zone of a secondary furnace containing molten clad unirradiated UO_2 . This second furnace has two purposes: to add a source of cladding and fuel particles, and to provide a high temperature region for ionization, decomposition or other reactions of the aerosol. All previous tests in the ADF have used stainless steel clad UO_2 . Because many reactors use or propose to use Zircaloy-clad fuel, a determination is needed on whether the type of cladding used in the CSE aerosol mixture is important. Therefore, a containment test (Run 64) was made in the ADF using Zircaloy-2 cladding. Airborne concentration as a function of time is compared in Figure 16 with that of Run 61, which was a duplicate except that it used stainless steel cladding. Table II lists the conditions and pertinent measurements for these studies. It was concluded that cladding type did not affect iodine behavior in containment. This does not imply that cladding type is unimportant to mechanisms of fission product release from irradiated fuel, where irradiation history and metal oxidation reactions may be of importance.

Aerosol Injection Method - R. K. Hilliard

The study of the effect of Aerosol injection method continued. ADF experiments in which an iodine plus a stainless steel- UO_2 aerosol were injected into a vessel by a steam jet showed that it is desirable to prevent liquid entrainment from entering the jet with the motive steam. Figure 16 displays airborne concentrations versus time for Runs 59, 60, and 61; and Table II lists the conditions and measurements. Run 59 used an uninsulated steam jet with casual insulation and trapping of the motive steam line. In Run 60, the motive steam line was preheated slightly by wrapping with electrical heating tape and the jet body was likewise heated to prevent liquid entrainment. Improvement was noticed in percent of injection mass airborne, in solubility in the condensate and in iodine form. Injection in Run 61 was accomplished by allowing a few inches vacuum in the containment vessel to induce flow through the generation equipment. This has been the method used in the majority of previous ADF tests. The delivery line in this case was 1 in. tubing for the full length, thus eliminating turbulence. Figure 16 shows there was essentially no difference in the behavior of the two methods when liquid entrainment was prevented. Conclusive evidence cannot be shown however, until similar tests are performed in a vessel sufficiently large that the steam jet does not perturb the vessel pressure.

Iodine Generation - L. F. Coleman

Methods for radioactive labeling of large masses of stable iodine were evaluated. The iodide-dichromate fusion method is suitable for generating quantities up to about 1 g. Larger amounts cause difficulties because of an increase in fusion temperature due to reduced chromic oxides. An investigation of palladium iodine decomposition was also abandoned due to the

FIGURE 16. Effect of Aerosol Injection Method on Iodine Airborne Concentration

Table II. Summary of Aerosol Tests in Stainless Steel Vessel

Run No.	59	60	61	63	64
Aerosol Type	I ₂ , SS-UO ₂	I ₂ , SS-UO ₂	I ₂ , SS-UO ₂	I ₂ , SS-UO ₂	I ₂ -Zr-2-UO ₂
Initial Iodine Conc. (mg/m ³)	1.24	1.05	2.00	120	1.43
Delivery Method	Unheated Jet	Heated Jet	Vacuum ^(a)	Vacuum ^(a)	Vacuum ^(a)
Containment Temp. (°C)	80 ± 5	80 ± 5	81 ± 2	81 ± 2	81 ± 2
Containment Pressure (psig)	0 ± 0.2	0 ± 0.2	0	0	0
Steam Condensation Rate (g/min)	80 for 5 min, then 17	100 for 5 min, then 10	13	13	16
Iodine Recovery (%)	92	80	98	95	95
Iodine Form Injected (%)					
Filter	5	2	56	0.8	27
Ag Screens	91	95.5	43	98.8	70
AC Paper	4	2.5	1.3	0.4	2.8
Charcoal	0.005	0.005	0.01	0.01	0.03
Iodine Form after 10 min (%)					
Filter	40	98	99	5.5	96
Ag Screens	40	1.4	0.15	93.6	0.3
AC Paper	18	0.6	0.9	0.7	3.2
Charcoal	4	0.1	0.15	0.2	0.5
Half Life, early, (min)					
Filter	7	10	14	20	15
Ag Screens	7	7	15	5	30
AC Paper	60	42	45	95	25
Charcoal	∞	∞	∞	∞	275
Total	12	10	14	5	15
Iodine Solubility ^(b) in					
Condensate, (%)	50	99.5	99.8	99.2	99.8
Dep. Veloc at 20 min (cm/sec)					
Ag plated Cu	0.27	0.03	0.04	0.33	0.03
Stainless Steel	0.17	0.03	0.02	0.09	0.02
Carbon Steel	0.17	0.02	0.04	0.30	0.02
Phenoline 302	0.20	0.05	0.04	0.28	0.04
Amercoat 66	-	-	0.03	-	0.04

(a) 5 in. Hg vacuum initially. Aerosol flow raised pressure to 0 psig.

(b) Filtered through Gelman type A glass fiber filter, washed with 100 ml H₂O.

complex materials handling requirements. The system showing the most promise is the equilibration of solid KI with liquid I_2 at 130 to 150 °C. The exchange half life was 12 to 15 min at 130 ± 2 °C when 0.25 g of spiked KI was reacted with 2 g of elemental iodine. Iodine prepared by this method was volatilized into an air carrier stream and various fractions analyzed for specific activity. The first 5 wt% and the last 10 wt% had identical specific activities within the 3% experimental error. Both fractions were >99% elemental iodine, based on Maypack discrimination.

Iodine Form in Steam Condensate - J. D. McCormack

Determination of the chemical form of iodine in steam condensate is complicated by the possibility of reaction and exchange soon after absorption. Some information on form after about hours was obtained on ADF samples by three procedures. First, condensate samples were filtered through Gelman type A glass fiber filters to determine the fraction associated with particles. In almost every case, greater than 99% passed through the filters. Second, samples were extracted with benzene, then back extracted with caustic. Typically, 99% remained in the water phase, <0.15% in the benzene fraction. Third, samples were sparged with air while boiling. No iodine was released. The combined information showed that the iodine existed (at least several hours after taking the sample) in a nonvolatile, highly soluble ionic form.

Maypack Behavior - J. D. McCormack

Considerable information is available in the literature on Maypack efficiency for discriminating various iodine forms. Differences in Maypack components and usage necessitate local evaluation, and such a study has been in progress in the ADF.

Two Maypacks heated to 96 °C were used in parallel. After loading, one was removed while the other was purged by a continued flow of clean air or steam and air. The effects of mass loadings, flow rate, and carrier gas type were studied for both I_2 and HI. The I_2 was generated by iodide-dichromate fusion, collected in a cold trap at -70 °C and then released into the Maypacks. Commercial HI was labeled by exchange with NaI-131 in water. Release was achieved by heating gently to dryness and evaporating into either air or a mixture of 8% argon in a hydrogen carrier gas. The composition of HI prepared in this manner was about 80% HI, 20% I_2 as shown by extraction with water and carbon tetrachloride.

Results to date indicate that:

- I_2 mass loadings above about 5 mg may cause breakthrough of 6 silver plated screens (100 mesh) at 0.5 SCFM.
- HI was partially retained by glass fiber filters, and the fraction retained decreased with an increase in mass loading.
- Clean air flow through a loaded Maypack eluted some of the I_2 from the filters but did not cause loss from the Maypack.
- I_2 in a steam-air carrier behaved similarly to HI in air; i.e., a significant fraction was retained by the fiber filter.

In all these tests, the Maypack compounds were 2 Gelman type A glass fiber filters, 6 silver plated screens, 1 Gelman type AC charcoal impregnated cellulose acetate paper, and 2 in. of coconut charcoal (unimpregnated), respectively.

Proposed conditions for aerosol sampling in the CSE are such that Maypack usage is justified, but further work on HI behavior is warranted.

Visual Observations of Steam-Air Atmospheres - R. K. Hilliard

Pyrex windows were installed on top and side nozzles in the stainless steel aerosol tank. The atmosphere was observed during both steady-state isothermal tests with continual steam makeup and with one-shot steam pressurization followed by natural pressure decay. When steam first entered a cold tank, a billowing dense fog formed. This soon coalesced to form a mist through which the tank walls could be observed. Drop condensation appeared to be with the drops growing into rivulets which either fell from top nozzles or ran down the sides in irregular paths. During most of the time in both pressure decay and isothermal tests, the atmosphere was absolutely clear and the walls appeared dry except for the rivulets and condensation near poorly insulated nozzle areas. Maypack sampling during misty and clear conditions showed no significant differences.

Gamma Energy Analytical Program - L. F. Coleman and

N. P. Wilburn

The on-line computerized gamma energy analytical system⁽¹⁾ was put into operation. This system interfaces the outputs of four 256-channel analog to digital converters to a small digital computer. Inputs to the analog to digital converters are from four NaI(Tl) scintillation detectors. Outputs can be simultaneously and independently stored. Processing of multicomponent gamma energy spectra is accomplished by a modified least squares method.^(1,2)

(1) N. P. Wilburn. The Combination of On-Line Analysis with Collection of Multicomponent Spectra in a PDP-7, BNWL-CC-700, Pacific Northwest Laboratory, Richland, Washington, July 15, 1966.

(2) W. L. Nicholson, J. E. Schlosser, and F. P. Brauer. The Quantitative Analysis of Sets of Multicomponent Time Dependent Spectra from Decay of Radionuclides, HW-75806, General Electric Company, Richland, Washington, December, 1962.

At present, the analytical program is designed to solve for a maximum mixture of 8 isotopes.

The channel grouping of the 255-channel spectra is limited to 20 groups. These limits are imposed by the available memory and magnetic tape storage area, which could be expanded if necessary. However, this system is entirely adequate for the presently visualized CSE program.

Gamma energy analysis techniques used by the CSE are based on storage of individual standard isotope spectra for each isotope, each detector, each geometry position, and each specific sample type. This minimized interpolation and extrapolation to assumed sample types and geometries, but requires a large number of standards. About 1000 standard spectra are filed on addressable magnetic tape and are called into the computational program as needed.

Standard spectra for two isotopes, I-131 and Ba, La-140, have been filed. Ru-103, Ru-106, Cs-137, and possibly others will be filed shortly. Procedures for instrument checking and calibration are in process.

Documentation of the code written for the digital computer to cause it to operate as a multichannel analyzer is now complete and will be issued in two formats:

- As BNWL-CC-700, which includes detailed description and listings of the code, and
- As BNWL-SA-684, which will be published in the Proceedings of the Spring 1966 Digital Equipment Users Society Meeting.

Decontamination of Coatings - Lyle Perrigo

Corrosion Engineering Unit began an experimental study of decontamination methods for use in CSE vessels. Coupons contaminated in the ADF will be decontaminated by vapor phase

techniques. A method for decontaminating Phenoline 302 coated carbon steel vessels without adversely affecting future tests is being sought.

Iodine Sampler Development - R. K. Hilliard

The Battelle-Columbus Laboratory began the second phase of their study of a new concept for sampling atmospheric iodine. The objective is to develop a sampler which will unequivocally distinguish between various forms of iodine. The proposed sampler consists of a particulate filter followed by a column of a matrix of solid particles (probably Na_2SO_3 on Chromasorb W) coated with a thin film of dimethylsilicone fluid. Preliminary tests with steam showed that a very viscous organic such as SE-30 silicone gum was necessary to prevent it from being eluted from the Chromasorb by the steam. Current effort is directed toward increasing capacity for methyl iodide.

Instrumentation - R. T. Allemann and M. E. Witherspoon

Investigations on a scoping study basis served to define possible techniques useful for the measurement of mass flow of two-phase water. Such measurements will be required during the blowdown tests of the planned containment system experiments. For the planned blowdown tests, approximately 150 ft^3 of water, at about 600°F and 2500 psi, will be discharged through a Sch 160 pipe in a time period of about 5 sec. The laboratory tests indicated the technical feasibility of using thermal and/or fast neutron techniques, along with either ultrasonic or gamma attenuation methods, to provide the required determinations.

A prototype of a high speed liquid level measuring system for use in the reactor vessel blowdown studies was designed

and fabricated. The system, which uses time domain reflectometry was tested in laboratory equipment and worked satisfactorily. Pilot studies are planned for the SM-2 vessel before starting the design of the probe for the CSE reactor simulator vessel.

The response of a heated thermopile type anemometer in a steam-air mixture at 250 °F was compared with the response of the anemometer in air only. No obvious differences were noted. Calibration of the instrument remains a problem.

Data Acquisition System - N. P. Wilburn

In addition to the radiochemical analyses function, the PDP-7 is to be used as a high-speed data logger to monitor the thermodynamic transient which occurs during blowdown. The computer has been interfaced to a relay matrix capable of selecting any one of 160 analog voltages in the following manner:

The analog voltage from the primary instrument is connected to a patch panel on the back of the matrix housing by means of about 150 ft of twisted pair shielded conductor and a five-pin connector. Each signal then passes through a conditioning circuit where it is attenuated and filtered (low pass, high pass, or T), if necessary. The three-wire output of the circuit (plus, minus, shield) is then connected to three "reed" type relays which have an actuate time of about 0.5 msec. When the computer selects a point to monitor, it actuates 12 relays or 4 points simultaneously. This action connects the 4 conditioned signals to 4 individual amplifiers whose outputs range between 0 and -10 Vdc. About 1.0 msec is required for switching transients to decay away. Thus, after a hardware delay of 1.5 msec, the computer causes

one of the 4 amplifier outputs to be connected to an analog to digital converter by means of a solid state switch and converted which takes 16 μ sec.

A computer code to operate the "multiplexer" described above during the thermodynamic transient and record the data collected onto magnetic tape has been written and debugged. The code, which can be described as a multithread-multiprogrammed code, consists of routines to drive the multiplexer, magnetic tapes, teletype for operator input and output, and routines for startup and termination.

The multiplexer code has been checked out with preliminary dynamic voltage tests. These tests consisted of monitoring the sinusoidal line voltage and a ramp voltage originating from an oscilloscope sweep. From these tests and others, it was determined that the 4 analog amplifiers were badly out of adjustment, so corrections were made. Because of this, some long-term reproducibility and stability checks are planned for the system using a precision voltage source.

The multiplexer code uses a series of tables to control the scanning sequence. These tables are packed binary to conserve storage; hence, are quite inconvenient for a user to generate. Thus, an auxiliary code has been written and debugged. It types out queries to the user for typed-in information which the code uses to generate the required tables. The result is a punched tape which can later be easily read into the computer with the optical reader.

REACTOR SAFETY ANALYSIS AND EVALUATION

Testing Small Vessels - D. R. Ireland

An experimental investigation of failure in low alloy steel pressure vessels is in progress. This program is combining the transition temperature and the fracture mechanics

approaches to reactor safety analyses. The program is primarily concerned with analyzing the failure of irradiated low alloy steel pressure vessel models.

The pressure vessel model currently being used is called a subsize-vessel specimen and has been described in a previous report.⁽¹⁾ The elastic-strain energy release rate function (G) of longitudinal cracks through the wall of subsize-vessel specimens is being determined by compliance testing procedures.⁽¹⁾ The cracks through the wall have presented several problems. These cracks are formed by milling a 0.06 in. wide slot 80% through the wall thickness and internally pressurizing the specimens to failure of the web under the slot. The opened webs provide through-wall-thickness cracks with sharp crack tips. The sealing of the cracks during the loading of the specimens can be accomplished by several techniques. Unfortunately, most of these techniques are not satisfactory for the compliance tests.

The compliance testing depends on the accurate measurement of the stored elastic energy of the specimen. It was found that slight variations in the patching techniques caused unacceptable reproducibility in stored elastic energy measurements. A series of strain gage tests were run to evaluate the effects of the various patching techniques on the reproducibility of strain versus internal-pressure curves. The patching technique that appears to be best suited to the compliance testing is a long rubber "mouse." The "mouse" is simply a rubber cylinder, with a steel "through bolt" placed inside the specimen. Once the mouse is slipped inside the specimen, the rubber cylinder is loaded in compression by tightening a nut on the end of the bolt. This loading of the mouse squeezes

(1) H. Harty, et al. Engineering Development Department Quarterly Report, BNWL 300, Pacific Northwest Laboratory, Richland, Washington, January, February, March, 1966.

the rubber against the inside surface of the specimen and makes a water tight seal. A piece of steel shim stock placed between the mouse and the crack prevents the rubber from being extruded out through the crack. During the testing of a specimen, the hydrostatic pressure of the water is transmitted to the specimen by the mouse. In a 14 in. long subsize-vessel specimen, the mouse occupies the central 12 in. of the specimen.

During this period the major accomplishment was the destructive testing of two irradiated A212B subsize-vessel specimens. The results of the tests have not been completely analyzed. The cylindrical specimens acted as a basket to contain fuel irradiated in the M3 position of ETR for one reactor cycle. Preliminary dosimetry studies indicate the specimens received an exposure of approximately 2.5×10^{18} nvt ($E > 1$ Mev). The temperature of the specimens during the reactor exposure was between 400 and 500 °F (204.4 and 260 °C). The specimens were 18 in. long with a 2.0 in. ID, and both specimens had longitudinal slots 0.06 in. wide by 1.0 in. long milled 0.2 in. through the 0.25 in. thick specimen wall. The specimens were internally pressurized to failure of the 0.05 in. thick web under the slot. The opened webs provided through-wall-thickness cracks with sharp crack tips. A rubber mouse patch was used to seal the cracks during the bursting of the specimens. The method of testing the specimens—internally pressurizing under closed end conditions—produced an approximate axial stress to hoop (circumferential) stress ratio of 1:2.

The specimens were arbitrarily identified as DRI-1 and DRI-2. DRI-1 was burst tested at 86 °F (30 °C) and DRI-2 was burst tested at 66 °F (19 °C). The failure appearances of DRI-1 and DRI-2 are shown in Figures 17 and 18, respectively. As indicated in these figures, DRI-1 failed in a ductile manner and DRI-2 failed in a brittle manner. The failure appearances

of these specimens indicate the irradiated material's NDTT is approximately 76 °F (24.5 °C). The unirradiated A212B has an NDTT of -22 °F (-30 °C).⁽¹⁾ This apparent NDTT increase of approximately 100 °F (37.8 °C) is in agreement with the upper boundary of drop-weight and Charpy V-notch Δ NDTT data.⁽²⁾

The ductile failure of DRI-1 occurred at a hoop stress of 60,000 psi, while the brittle failure of DRI-2 occurred at a hoop stress of 55,400 psi. An unirradiated A212B specimen with a similar 1.0 in. long crack failed ductility at a hoop stress of 47,500 psi when tested at 80 °F (26.7 °C). The burst testing of these two irradiated specimens was successfully monitored by nondestructive testing equipment. The results of this monitoring are discussed in the Ultrasonic Monitoring of Bursts Tests section of this quarterly report.

The testing of these two specimens has not been completed. The fracture surfaces will be metallographically examined and additional dosimetry tests will be made.

The second set of two irradiated A212B subsize-vessel specimens has been removed from the M3 position of the ETR, and the specimens are being prepared for testing during the next quarter. These specimens have been exposed for two reactor cycles.

(1) H. Harty, et al. Engineering Development Department Quarterly Report, BNWL 300, Pacific Northwest Laboratory, Richland, Washington, January, February, March, 1966.

(2) L. E. Steele and J. R. Hawthorne. New Information on Neutron Embrittlement and Embrittlement Relief of Reactor Pressure Vessel Steels, NRL Report 6160, U. S. Naval Research Laboratory, Washington, D.C. August 7, 1964.

Neg 661533-3

FIGURE 17. *Two Views of DRI-1 Burst Tested at 86 °F
(30 °C)*

Neg 661533-4

FIGURE 17. *Second View*

Neg 661533-5

FIGURE 18. *Two Views of DRI-2 Burst Tested
at 66 °F (19 °C)*

Neg 661533-6

FIGURE 18. *Second View*

Ultrasonic Monitoring of Burst Tests - T. E. Michaels

Ultrasonic monitoring techniques were applied to unirradiated burst tests of A106B pressure vessel steel and irradiated burst tests of A212B pressure vessel steel. Acoustic emission failure patterns have been established for each case. Over a bandwidth of 50 kHz to 1 MHz, acoustic emission was received on the A106B burst test specimens on only those tested below -10 °C. The irradiated A212B specimens emitted acoustic energy over a bandwidth of 50 kHz to 1 MHz at 30 °C. Thus a shift was observed in the upper temperature limit (UTL) for signal received over the particular bandwidth in frequency studied.

This shift of the UTL due to a neutron exposure of 2.5×10^{18} nvt (>1 MeV) corresponds to a shift in NDT from approximately -30 to 25 °C. Since the UTL can be determined without carrying a test to failure, this has promise for aiding in determination of the NDT for irradiated specimens.

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