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HEAVY WATER ORGANIC COOLED REACTOR

**QUARTERLY TECHNICAL PROGRESS REPORT
APRIL THROUGH JUNE 1966**

MASTER

AEC RESEARCH AND DEVELOPMENT REPORT

ATOMICS INTERNATIONAL—COMBUSTION ENGINEERING

A Joint Venture for Heavy Water Organic Cooled Reactors

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HWO CR QUARTERLY REPORT

TITLE: Demonstration Plant Engineering

WBS NO: 14000

REPORTING PERIOD: April Through June 1966

I. PROJECT OBJECTIVES

The overall objective of this activity is the design and construction of an HWO CR power plant having a rating of 500 Mwe. Subsequent operation of this plant will demonstrate the inherent high fuel utilization and low fuel cycle costs of the HWO CR concept. Basic performance characteristics of the HWO CR will also be verified with particular emphasis upon the following essential plant features: fuel elements, pressure tubes and pigtailed, refueling machine, calandria and end shields, primary loop components, and the reactivity control system. A further objective is to provide guidance to the research and development efforts performed under other activities by identifying functional requirements and design limitations and by preparing performance specifications for all systems and components.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. GENERAL STUDIES

1. Safety Studiesa. Maximum Credible Accident (MCA)

Further studies relative to the postulated MCA assumed a guillotine type severance of an upper reactor inlet header and melting of the fuel in the one-quarter of the core supplied by the header. In this conservative preliminary analysis, it was assumed, for lack of contrary evidence, that gross penetration of the calandria by molten fuel and organic occurs.

The containment building pressure and temperature conditions following an MCA were determined through the use of equations of state and energy balances with the following assumptions:

1) The D₂O is vaporized instantaneously and the quantity is just the amount that would be converted to saturated steam by the following heat sources: (a) the sensible heat of the organic coolant, (b) the latent and sensible heat from molten fuel in one-quarter of the core, and (c) chemical heat from the D₂O-UC reaction of 50% of one-quarter of the core. (Other metal-water reactions were not considered.)

2) The pressure increase is instantaneous so that heat transfer through the building structure or to heat sinks within the building is not significant. (Following the peak pressure, the building water spray system reduces the pressure and effectively suppresses the pressure below the peak value thereafter.)

3) No organic fire occurs in the building. No explosions occur from organic vapors, organic dust cloud, or the deuterium generated by the UC-D₂O reaction.

4) The building free volume is 1.5×10^6 ft³ and containment atmosphere prior to the incident is dry air at 80°F.

The MCA, based on the assumptions described, results in a containment building peak pressure of 32 psig at a corresponding temperature of 240°F. Based on this preliminary evaluation, the building must be designed to contain a gross release of fission products under the burden of this peak pressure.

Analytical results indicate that the total energy release is approximately 90×10^6 Btu and the percentage contribution from various sources is as follows: (a) 16% fuel-thermal, (b) 21% fuel-chemical, and (c) 63% coolant-thermal.

As indicated, the coolant represents the largest heat source. A coolant spill of 240,000 lb, or approximately 30% of the building inventory, was assumed.

The radiological aspects of the MCA are reported in AI-CE-TDR-1037, "Radiological Consequences of Postulated Maximum Credible Accident for the HWOOCR Demonstration and Dual Purpose Plants," April 21, 1966, currently in preparation. This document includes a regulatory recommended assumption that 5% of all available iodines are unfilterable organic compounds. In the initial analysis, radiological consequences to the public were determined for the MCA in accordance with the release fraction of TID-14844. The results showed that HWOOCR plants can be safely located relatively close (~1 mile) to large population centers provided that: (a) low leak-rate containment buildings (0.1%/day or less) are employed, and (b) rapid high efficiency (90%) internal cleanup of the containment building atmosphere takes place, without limitation on iodine cleanup, immediately following the accident. However, if 5% of the iodines are not amenable to such cleanup techniques, the required population center distance could increase by a factor of three or four.

b. UC-Penetration of Calandria

In view of the uncertainty associated with the possible penetration of the calandria tube by molten UC, an analytical and experimental program was initiated in an endeavor to obtain a better insight into this potential. The analytical work was published in AI-CE-TDR-1065, "Thermal Physics Technical Note No. 70, HWOOCR Molten Fuel Accident Analysis." The TAP-3 program was used for this analysis.

It was concluded that because of the rapid (0.25 to 0.5 sec) melting of the cladding there would be no spewing of molten fuel. The pressure tube would melt in 100 sec if none of the fuel were touching it. Some pressure tube melting would occur in as little as 3 sec if there were intimate fuel contact. The central fuel element melts at 450 sec. Assuming that all the fuel is then molten and in intimate contact with the calandria wall, film boiling begins at 455 sec. At 530 sec, the median fuel temperature is at 750°F and the median calandria wall temperature is decreasing from a maximum of 418°F. At no time does the calandria wall experience melting. Neither the thermal gradient across the calandria tube wall nor the weight of the fuel column pose a stress problem that even approaches rupture. Thus, penetration of the calandria tube could only occur if additional force is applied by some other means.

Experimental testing to corroborate these analytical conclusions has not been initiated as yet; however, preliminary experiments on UC-water reactions were initiated. In these experiments, molten drops of about one-half gram of UC were dropped in about 60 grams of ice water with no visible reaction. This work is continuing with higher temperature water and steam in an endeavor to determine reaction rates of molten or very high temperature UC.

c. Containment Mathematical Model

In an endeavor to correct overconservatism in the assumption of instantaneous conversion of all available energy to steam for pressurizing the containment building, a complex, time-dependent, accident model is being developed. A preliminary block-flow diagram was prepared for orientation of group effort and will function as the basic interface coordination tool. Since it is not definitely known whether the MCA or a combined pressure tube-calandria tube rupture will establish the design pressure, it was decided to include the latter accident in the computer model.

An equation for reactor power as a function of time after system rupture was formulated. Enthalpy balance equations were written for the system in the inlet header ruptured state. Relationships for external enthalpy gain and loss as a function of time are being prepared.

In formulating a mathematical model for D_2O and organic relief and disposition in the event of penetration between the organic system and the calandria D_2O system, it was found that the relief system design had not yet been developed in sufficient detail for the model. Two proposed design schemes were then prepared and submitted to process design. When design agreement is reached, development of the mathematical model will continue.

d. Table of Accidents

The HWO CR Table of Accidents was revised and expanded to show the complete and current status of safety studies.

e. Safety Criteria and Philosophy

A safety criteria and philosophy document was prepared for the HWO CR based on the "27" criteria, protective system suggestions originating in the ANS-4 subcommittee, and HWO CR special consideration developed through design considerations. In addition, the work on pressurized water reactors sponsored by the N-6 committee was reviewed for applicability to the HWO CR since this effort represents the first concerted attempt to establish general safety criteria for a specific reactor concept. The report makes extensive use of the pressurized water reactor work modified for the HWO CR. This document has been released for comment, but it is recognized that considerable expansion will be required over the next year.

f. Potential Tritium Hazards Under Accident Conditions

This analysis was performed for the Reference Plant and is reported in Section 15000.

2. Reliability Studies

A reliability evaluation of the HWO CR shutdown control rods was performed and was based on studies comparing component allocations. The purpose of the evaluation was to define a mathematical model which could be manipulated to optimize the operation of the shutdown control system. Several preliminary

models aided in determining the initial component allocations which were subsequently refined. Once a rough order of magnitude was established, component allocations were varied to arrive at the best value of reliability within a reasonable demonstrated test program.

In order to better evaluate the shutdown control rod components, each period of operation was analyzed separately. For each period, only those components actually in operation, or depended upon, were considered in the system reliability.

The reliability calculations for the startup phase allowed no failures associated with the rod systems. In order to compare pump effects on the system, calculations were run for system reliability when at least one or more pumps are working and when at least two or more pumps are working. The difference between the two pump systems alone is a factor of 25; however, the effect on the total system is less than a factor of 2. The greatest reliability advantage during startup is gained by increasing an individual rod assembly reliability. For normal operation, the double coil latch solenoid affords the greatest advantage. During test operations, which are carried out once a week, two or less failures are permitted without a shutdown penalty. The fact that shutdown failures are allowable during test operations greatly increases this reliability over the no-failure case. For shutdown, the system reliability is significantly increased by increasing the rod assembly reliability and by allowing two or less failures.

This study concluded with the following system recommendations:

System Criteria	Recommendation			
	Pump System	Latch Solenoid	Rod Assembly	
			Reliability	Number of Tests at 90% Confidence
Highest reliability	One out of four working	Two-coil solenoid	0.9999	23,000
Optimum operational	Two out of four working	Two-coil solenoid	0.999	2,000

The figures shown for the highest reliability criteria require that a large number of tests (23,000) be performed to demonstrate that a level of at least 0.9999 has been achieved. In this case, however, the performance of this many tests of the rod assembly would be impractical.

B. REACTOR DESIGN

1. Reactor Structures

a. Calandria Structure Studies

Preliminary structural analyses of the integral reactor vessel were completed for the following designs:

- 1) Stainless steel vessel with a shell expansion joint, top support only.
- 2) Stainless steel vessel with a shell expansion joint, top and bottom supports.
- 3) Inconel-carbon steel vessel with no expansion joint, top support only.

These analyses have been performed for both the normal operating and accident conditions.

Based on preliminary analysis, a single support design was considered to be superior to the double support design from a structural standpoint. In general, the single support stainless steel design results in the most favorable stress conditions.

An analysis of the heat loads and the temperature conditions for the upper and lower end shields of the reactor vessel is in progress with the aim of minimizing the thermal gradient through the end shields and consequently reducing stresses and deflections due to thermal bowing of the shield grid plates.

The use of a separate cooling system containing high purity light water for each of the reactor vessel end shields with close control of the temperatures is required.

A preliminary erection sequence for the reactor complex area has been prepared. This sequence is based on erecting the reactor vessel in three sections in order to reduce the lifting requirements and to simplify alignment. An effort is now in progress to assign reasonable time durations to the individual steps in the sequence.

A trade study was conducted which evaluated the separate vessel concept vs the integral vessel for the reactor. The integral vessel was determined to be the superior design.

b. Calandria Pressure Relief Analysis

An analysis was performed to predict the transient pressure which would occur in the calandria after a rupture occurred in both a pressure tube and its

associated calandria tube. As a result of a rupture, organic coolant would contact the heavy water and form a steam cavity, the growth of which would displace the adjacent liquid heavy water. The rapid displacement of the water is accompanied by high transient pressures both near the rupture location and at the calandria wall. Flow of the liquid water out of the calandria relief lines is expected to limit the pressure rise at the calandria wall.

A physical model was assumed for the analysis. The model was put into a mathematical form consisting of a set of simultaneous ordinary nonlinear differential equations. Solutions were obtained on a digital computer using a Runge-Kutta technique for numerical solution.

(1) Physical Model

It is assumed that a complete and instantaneous circumferential rupture occurs in both the pressure and calandria tubes, and that the organic coolant then flows from both ends of the ruptured pressure tube. The organic flowrate is determined by the instantaneous differential between the pressure of the steam in the cavity which is formed in the heavy water at the rupture location and the constant pressures in the inlet and outlet headers. Inertia and flow resistance of the organic in the pigtails and pressure tube leading to the rupture location are included. The steam in the cavity is assumed to be saturated and to be in constant thermal equilibrium with all the liquid organic coolant which has flowed into the cavity.

Spherical symmetry is assumed for the cavity, for the liquid heavy water displacement by the growing cavity, and for the calandria.

This displacement of heavy water from the calandria is influenced by five factors:

- a) The volume growth rate of the spherical steam cavity.
- b) The rate of compression of the liquid heavy water in the calandria.
- c) The rate of a perfectly elastic circumferential expansion of the cylindrical calandria.
- d) The rate of decrease of the annular gas gap between the pressure and calandria tubes as the calandria tubes collapse onto the pressure tubes.
- e) The acceleration and flow resistance of a slug of liquid heavy water in the relief lines.

(2) Results of Numerical Solutions

A typical reference set of parameters was assumed, comparable to those for the Demonstration Plant presented in AI-CE-Memo-25. These parameters were

varied one or more at a time and solutions are obtained. A solution can be expressed in the form of a table with the eight dependent variables as a function of time which runs from zero to 0.25 sec. These variables are:

- a) Pressure inside the cavity
- b) Radial velocity of cavity wall
- c) Cavity radius
- d) Pressure at the calandria wall
- e) Length of liquid heavy water slug in relief lines
- f) Velocity of liquid heavy water in relief lines
- g) Velocity of organic coolant in the inlet pigtail
- h) Velocity of organic coolant in the outlet pigtail.

Figure 14000-1 shows for several cases the absolute pressure calculated at the spherical calandria wall vs time, which is comparable to the pressure at the top of the real calandria. These curves are for the rupture location at the coolant inlet end of the fuel region, since the final pressures at the calandria wall are higher for this case than for the rupture at the outlet end.

Figure 14000-1. Absolute Pressure at Calandria Wall vs Time

CASE 1 — Reference

This curve is based on the normal operating parameters. It is assumed that the calandria tube collapses from a round tube to an elliptical tube of minor diameter equal to pressure tube diameter as the critical collapse pressure is reached in the calandria. Rapid, divergent oscillations in pressure occur at least up to

0.25 sec. It is uncertain whether or not these pressure oscillations would continue, since the buoyancy of the steam-filled cavity would cause it to rise, and another cavity would probably start to form. The cavity radius at the maximum time shown is 2.13 ft.

CASE 2 – Rigid Calandria Tube

In Case 2, the calandria tube does not collapse. Both the magnitude and frequency of the pressure oscillations increase considerably over those of Case 1.

CASE 3 – 20% Loss of Thermal Energy of Organic

It is assumed in this case that 20% of the thermal energy available in the organic for making steam is lost either through conduction into the heavy water liquid or by inefficient mixing within the cavity. The peak pressures are reduced from those of the reference case for the three peaks calculated. This pressure decrease is larger for the later occurring peaks.

CASE 4 – Relief Line Diameter Doubled

The reference case has eight relief lines from the calandria. In Case 4, the diameter of these lines is increased from 6 in. to 1 ft. This decreases the first pressure peak by only 0.9 psi. It decreases the third pressure peak by 13.1 psi. Inertia of the water in the calandria and in the relief lines, therefore, governs the maximum initial pressure peak. An increased relief area affects only the later occurring pressure peaks and also the relatively constant pressure which might exist at times greater than the maximum calculated value of 0.25 sec.

c. Pressure Tube Assembly

Design details for the preliminary pressure tube assembly design were completed. The pressure tube extension material has been changed to SA 213 T2 low alloy steel. This material does not require a post-weld heat treatment. This allows the final assembly field weld to be made without the use of a 304 SS extension piece.

Preliminary evaluation of early tube rupture test results has indicated that the original choice of design allowable stresses was reasonable. All points obtained thus far have fallen above the minimum expected transverse strength. Some stress sensitivity at notches is indicated but, when evaluated on the basis of stress in the thickness remaining below the notch, this does not appear nearly as severe as other sources have indicated when applied to HWOCR design.

2. On-Power Refueling System

Because of uncertainties associated with the operation of mechanisms in hot organic coolant, it was decided that all mechanisms located in the pressure tube assemblies must be removable at any time by the refueling machine for inspection or maintenance while the reactor is operating. Accordingly, a design study was performed to devise and analyze alternatives to the refueling concept described in AI-CE-Memo-25 that would meet this requirement. Three of the concepts generated were judged to be practical and are being analyzed in further detail for design feasibility and for effects upon other reactor components. These concepts are described in the following paragraphs.

a. Lock Ring Scheme

During steady-state, the fuel column is supported between two lock-rings latched in the pressure tube. The distinctive feature of this concept is that during refueling the lock-rings are not carried beyond the coolant connections. Each lock-ring is designed to permit the passage of the refueling machine's inner ram through it, as required, to move the column. A gate is used in each machine to hold the column as individual pieces are removed or added to the column. The column is first moved into the upstream machine where a new fuel assembly is added, and then into the downstream machine where a spent fuel assembly is removed. Refueling spacers carried in each machine are added to the column to minimize the machine ram stroke.

The sequence of operations as depicted in Figure 14000-2 is:

- 1) Establishing steady-state configuration.
- 2) Removing end seals.
- 3) Removing downstream lock-ring.
- 4) Adding two refueling spacers to downstream end of column. Removing upstream lock-ring.
- 5) Removing upstream permanent spacer and shield plugs. Adding new fuel assembly to upstream end of column.
- 6) Adding shield plugs, permanent spacer, and two refueling spacers to upstream end of column. Removing refueling spacers, permanent spacer, shield plugs and spent fuel assembly from downstream end of column.
- 7) Installing downstream shield plugs, permanent spacer, and lock-ring.
- 8) Removing refueling spacers from upstream end of column.
- 9) Adding downstream lock-ring to column.
- 10) Centering column and fastening lock-rings in pressure tube.
- 11) Installing end seals.

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Figure 14000-2. Lock Ring Scheme

b. Latching Shield Plug Scheme

This concept is similar to the preceding one except that the fuel column is held in place by jaws in the outer shield plugs. These shield plugs are released by the machine rams and pushed into the core as the column is moved first into the upstream machine for addition of a new fuel assembly and then into the other machine for removal of a spent fuel assembly.

The sequence of operations as depicted in Figure 14000-3 is:

- 1) Establishing a steady-state configuration.
- 2) Removing end seals.
- 3) Removing upstream shield plugs. Adding new fuel assembly to upstream end of column.
- 4) Installing upstream shield plugs. Removing downstream shield plugs and spent fuel assembly.
- 5) Installing downstream shield plugs. Centering column and locking in place.
- 6) End seals replaced.

c. Latch Plug Scheme

In this scheme the fuel column is also held in place by latching shield plugs. During refueling a latch plug and a guide sleeve are fastened in each pressure

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Figure 14000-3. Latch Plug Scheme

tube extension. Latches in the latch plug hold the column in place as components are removed or installed one at a time, first by the upstream machine and finally by the downstream machine. After the shield plugs are reinstalled, the latch plugs and guide sleeves are removed. This scheme incorporates the feature of grappling all components, including fuel assemblies, and special ram adapters are provided for this purpose.

The sequence of operations as depicted in Figure 14000-4 is:

- 1) Establishing a steady-state configuration.
- 2) Removing end seals.
- 3) Installing latch plugs. Returning latch plug adapters to magazines.
- 4) Installing guide sleeves. Returning guide sleeve adapters to magazines.
- 5) Removing upstream locking shield plug.
- 6) Removing upstream inner shield plug.
- 7) Adding new fuel assembly to upstream end of column. Returning fuel adapter to magazine.
- 8) Installing upstream shield plugs. Removing downstream locking shield plug.
- 9) Removing downstream inner shield plug and spent fuel assembly.
- 10) Installing downstream shield plugs. Both guide sleeves removed.
- 11) Removing latch plugs.
- 12) Installing end seals.

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Figure 14000-4. Latching Shield Plug Scheme

3. Core Design Studies

a. Trade Studies

During this report period five trade studies were issued on various design decisions which are pertinent to the HWO CR design. A brief description of the results of these studies is given below.

(1) Zircaloy-4 and SAP Cladding Materials

A study was performed to determine the better cladding material for a fuel element with UC fuel. Based on present-day technology, two cladding materials, Zircaloy-4 and SAP, offer the greatest promise for the HWO CR. The results of the study indicated that a SAP-UC fuel system is superior, principally because of the greater thermal conductivity of SAP. The fins on a Zircaloy-4 fuel rod effectively reduce the heat flux to only 93% of an unfinned design compared to 66% for a finned SAP design. The relatively poor thermal conductivity of Zircaloy would seriously limit the power producing capability (max. kw/ft) of the UC fuel.

(2) SAP Cladding and Pressure Tube Materials

SAP which contains either 7 or 10% Al_2O_3 could be used for fuel cladding and pressure tube material. A comparison was made of fuel element designs using each of these SAP materials to determine which was best for the HWOCR. With the 10% oxide material, smaller cladding and pressure tube thicknesses are possible due to the greater strength of the material. This results in lower fuel enrichments and lower power costs; however, the economic incentive for using 10% material is small.

Fuel element irradiation tests will be run using both materials. If a need for slightly higher cladding ductility results from the fuel element development program, the SAP-7% oxide material would be adopted for the cladding.

(3) Demonstration Plant Fuel Discharge Exposure Study

A study was made to determine the optimum fuel discharge exposure for the Demonstration Plant. Capital costs and fuel costs were determined for discharge exposures of 14,000, 18,000, and 22,000 Mwd/MTU. The reference average discharge exposure was established at 18,000 Mwd/MTU, based on the following conclusions:

- a) Reactor capital, organic makeup, and heavy water costs are lower at a fuel exposure of 18,000 Mwd/MTU.
- b) Total annual power costs (mills/kw-hr), excluding plant capital costs, are insignificantly lower at 22,000 Mwd/MTU compared to the 18,000 Mwd/MTU condition. An expected increase in the engineering hot channel factor related to slug eccentricity at the higher exposures could eliminate any indicated cost advantage at the 22,000 Mwd/MTU exposure.
- c) The maximum heat release (kw/ft) for the 22,000 Mwd/MTU exposure condition is very near the performance limit with respect to fuel growth.

(4) Bi-Directional and Uni-Directional Flow and Refueling Schemes

An evaluation was made of the effect on reactor performance of uni-directional and bi-directional refueling systems with either uni-directional or bi-directional coolant flow. Results of the evaluation indicate that uni-directional refueling is unattractive because of the associated increase in axial power peaking. Uni-directional flow results in higher cladding surface temperatures when coupled with bi-directional refueling. In order to maintain the same maximum surface temperature, 47% more pressure tubes would be required. Based on total power cost evaluation for each system, the bi-directional flow and refueling system offers the greatest advantage.

(5) UO₂ and UC-SAP Fuel Designs

An analysis was made to determine the relative merits of a UO₂-Zr fuel system as compared to the reference HWOCR fuel systems of UC-SAP. The analysis was performed for each system based on identical thermal limitations and comparable nuclear data. Based on equal fuel fabrication costs (\$/kg U) for the two fuel systems, the results of the analysis indicated that the UC-SAP combination has 6.9% lower power costs and 4.2% higher dynamic fuel utilization than the UO₂-Zr fuel system. This is primarily the result of the higher fuel enrichment and larger reactor necessary with the UO₂-Zr system.

b. Fuel Assembly End Plates

During this report period, studies were made to determine the effect on power cost of substituting stainless steel end plates for the Zircaloy-4 end plates of the reference fuel element design. It has been estimated that the 0.125-in.-thick Zircaloy-4 end plates could be replaced by 0.093-in.-thick stainless steel end plates. This would result in 0.0035 $\Delta K/k$ change in reactivity worth with a corresponding 1.1% increase in fuel enrichment and 0.3% increase in total power cost.

Calculations on the Demonstration Plant indicate that the hydrogen solubility limit of Zircaloy end plates will be exceeded in the fuel elements in the periphery of the core, which have a long core residence time. Based on the small cost differential and the uncertainty of material integrity as affected by hydriding, the end plate material has been changed from Zircaloy-4 to stainless steel.

c. Fuel Element Growth Performance Code

A fuel element growth and fission gas release performance code has been completed and is now operational. This code will predict the centerline temperatures, gap clearances, clad strain, and gap heat conductance based on experimental data for the fuel growth and fission gas release. Additional calculations are made for the fission gas pressure and end gap clearance. These quantities are calculated for both the beginning and end of cycle condition at any designated axial position.

At present, the code is limited to fuels with relatively constant thermal conductivities and small fission gas release rates. An advanced code which eliminates these restrictions is being developed.

4. Nuclear Analysis

a. Reactivity Coefficients

(1) Revised Scattering Kernels

Thermal energy group calculations employing the THERMOS⁽¹⁾ code have previously used Brown-St. John-type scattering kernels. Scattering kernels for several light nuclides were revised to be more consistent with experimental data. These were:

- a) H (H₂O) – transport corrected Nelkin kernel⁽¹⁾
- b) D (D₂O) – transport corrected Nelkin kernel⁽²⁾
- c) H (Organic) – transport corrected Nelkin kernel⁽³⁾
- d) C (UC) – Brown-St. John kernel with mass 40

Use of the above kernels resulted in a 53% increase in the fuel temperature coefficient and a 10% decrease in the coolant temperature coefficient for the point burnup representation of the Demonstration Plant equilibrium core. The net profit was a change in the power coefficient, $\Delta\rho/\Delta$ (% P), from $+3.05 \times 10^{-5}$ to 1.58×10^{-5} , a decrease of 48% for the UC fueled core.

The large increase in the fuel coefficient is due to the heavier mass representation of the carbon in UC, which decreases the total scattering cross section at low energies and also reduces the temperature derivative of the scattering cross section. The reduction in the coolant coefficient is due to the use of a Nelkin type benzene kernel.

To illustrate further the dependence of the fuel temperature coefficient on the choice of scattering kernel assumed for the light elements contained in the fuel material, the temperature variation of $(\eta f)_{th}$ was evaluated as a function of the effective mass used in a Brown-St. John scattering kernel representation for the oxygen in UO₂. As the effective mass changes from 16 to 50, $(\eta f)_{th}$ decreases by less than 0.3% while the temperature coefficient of $(\eta f)_{th}$ decreases by 72%. These results show the importance of using the most realistic scattering representation possible for the evaluation of temperature coefficients.

(2) Analysis of Fuel and Coolant Temperature Coefficients

Fuel and coolant temperature coefficients were evaluated for a range of lattice cell conditions using different calculational models with the following objectives in mind:

- a) To test the effects of varying portions of the calculational model and thus obtain some estimate of the uncertainties in the coefficients
- b) To point up those processes which are most important in determining the magnitude of the coefficients
- c) To evaluate the dependence of the coefficients on such parameters as feed enrichment (burnup), lattice pitch, fuel area, number of rods per cluster and volume fraction of organic within the pressure tube.

The basic calculational model employs the THERMOS code in the thermal energy calculation and the zero-dimensional FORM⁽⁴⁾ code in the epithermal energy groups corrected to include heterogeneous fast fission effects. In addition, the HAMMER⁽⁵⁾ code which performs a one-dimensional multigroup spectral calculation over the full energy range has been employed for selected calculations. The thermal energy calculation by both models is basically the same; consequently any differences are due to epithermal effects. The following paragraphs summarize the results of calculations carried out to date.

The thermal energy calculation has a significant effect on the magnitude of the coefficients when upscattering is taken into account. The change in $(\eta f)_{th}$ with a change in coolant or fuel temperature is a function of the relative amount of uranium and plutonium present in the lattice. For the fresh fuel condition the change in $(\eta f)_{th}$ with an increase in fuel temperature is negative but as plutonium builds up the change reverses in sign and continues to increase with burnup. The change in $(\eta f)_{th}$ with an increase in coolant temperature is positive; the magnitude of the change increases with burnup.

An increase in $(\eta f)_{th}$ with an increase in fuel temperature leads to a fuel temperature coefficient which is less negative than the value predicted from the Doppler effect alone. The FORM-THERMOS model predicts a decrease in fuel temperature coefficient with burnup due to an increase in the magnitude of $\Delta(\eta f)_{th}/\Delta T_f$ with burnup. The HAMMER model predicts a slightly higher fuel temperature coefficient for fresh fuel conditions and a smaller change with burnup since Doppler broadening of Pu²⁴⁰ resonances is also included. The fuel temperature coefficient increases as the D₂O/U ratio decreases and to a lesser degree as the volume fraction of organic increases.

The behavior of the temperature coefficients with feed enrichment (and discharge burnup) is as follows. The fuel temperature coefficient, for the fresh fuel condition, increases with decreasing enrichment. The coolant coefficient

for the fresh fuel condition is small and insensitive to the feed enrichment but the rate of change of the coefficient with burnup increases rapidly with decreasing enrichment, the net effect being that the relative magnitudes of the fuel and coolant coefficients appear to be relatively insensitive to the feed enrichment.

(3) Power Coefficient

The power coefficient involves not only the fuel and coolant temperature coefficients but also the magnitude of the fuel and coolant temperature change per percent power increase. For the equilibrium core of the reference Demonstration Plant reactor, the ratio of the fuel to coolant average temperature rise per percent increase in power is 5.5 whereas the coolant temperature coefficient is approximately nine times larger than the fuel temperature coefficient; consequently, the net power coefficient is positive. Power coefficients were estimated for several different types of lattices and the following trends were observed:

- a) For a given fuel channel configuration, a reduction in the D_2O/U ratio leads to a reduction in the power coefficient due to an increase in the fuel coefficient and a reduction in the coolant coefficient.
- b) For fixed values of the D_2O/U ratio and rod cluster configuration, an increase in the volume fraction of organic generally leads to a reduction in the power coefficient since the coolant temperature rise per percent increase in power decreases faster than the increase in coolant temperature coefficient.
- c) For constant values of the lattice pitch, volume fraction of organic, and number of rods per cluster, an increase in the fuel area leads to a reduction in the power coefficient due to an increase in the fuel coefficient, a reduction in the coolant coefficient, and a reduction in the coolant temperature rise per percent increase in power.
- d) To first order the power coefficient in a fixed lattice cell geometry is proportional to the kw/ft rating of the fuel; hence a reduction in the power output of the reactor by a given factor leads to a similar reduction in the power coefficient.

b. Fuel Bundle End-Effects

Local axial peaking factors and reactivity effects associated with fuel bundle end fittings have been calculated for a stepped end-plate design and a modified flat end-plate design. The lattice cell geometry as defined by a cut through the middle of the bundle was similar to that defined in AI-CE-Memo-25. All calculations were carried out for fresh fuel of 1.3 wt % enrichment.

(1) Stepped End-Plate Design

The stepped end-plate design was the same as described in AI-CE-Memo-25; i. e., the fuel pins in the outer ring of the cluster have a 0.4-in.-long natural UC pellet at both ends of the fuel column and the fuel columns are 1.63 in. shorter than for the other fuel pins. Thus, for tandem bundles, the fuel columns in the central pins are separated by 1.63 in. and in the outer fuel pins by 3.26 in., including a 1-in. fission gas and growth gap allowance in one end of the fuel pins. Calculations were carried out for 0.125-in.-thick end-plates of Zircaloy, stainless steel, and half density stainless steel. The loss in reactivity resulting from the replacement of fuel by coolant, end-plates, etc., was 0.41, 1.22, and 0.76% $\Delta\lambda/\lambda$, respectively. The local axial peaking factor for the fuel pins in the outer ring (gas gap end of bundle) was 1.19, 1.14, and 1.16 for the cases defined above.

(2) Modified Flat End-Plate

In this design all fuel pins are of the same length and attached to a flat plate; the opposite face of the plate has an attachment which performs the same function as the stepped end-plate design. The end-plates are 0.093-in.-thick stainless steel. The fuel-to-fuel space between tandem bundles is 2.50 in. including the 1-in. fission gas and growth gap. Five conditions were analyzed: (a) all fuel of 1.3 wt % enrichment; (b) all fuel pins having 1-cm-long natural UC pellets at both ends; (c) fuel columns in the outer ring of rods had natural UC pellets of 0.4 in., 1 in., and 2 in. in length at both ends of the fuel column. The loss in reactivity resulting from the replacement of fuel by coolant, end-plates, etc., was 0.83, 1.19, 1.0, 1.27, and 1.66% $\Delta\lambda/\lambda$, respectively. The local axial peaking factor for the outer fuel pins (gas gap end of bundle) were, in order, 1.35, 1.23, 1.22, 1.13, and 1.10. For the core with natural UC pellets in all pins there was substantial reduction in the local axial peaking in all fuel pins; e.g., in the center pin there was a 30% reduction.

(3) Stepped End-Plate Design Adjacent to Shield Plug

The local axial peaking factor for the end of a fresh fuel bundle adjacent to a shield plug, when the combination is inserted into the core during a refueling operation, was found to be approximately the same as for two tandem fuel bundles.

5. Preliminary Safety Analysis

a. General

During this quarter, the major effort was placed on (1) investigating the space-time kinetics problem associated with the reactor and (2) determining the general response characteristics of the reactor to various perturbations using a single point and a multi-node reactor model.

b. Space-Time Kinetics Problem

Previous effort on this problem resulted in an extension of the point kinetics model to include axial variation of spatially dependent properties of the core. The result was a multi-node analog model using five axial nodes with appropriate thermal-hydraulic and neutronic point reactor characteristics defined for each node. This extension of the point kinetics model to include axial variation of spatially dependent core properties was the first step toward determining if the simplified point kinetic reactor model could be conservatively applied to a large-size HWOCR having a positive power coefficient.

Emphasis was next placed on developing a calculational model which would predict the transient response of the core resulting from the introduction of localized perturbations in a plane transverse to the axis of the core. Several digital programs were reviewed and the WIGL2 code was selected for exploratory investigations into the space-time kinetic behavior of the HWOCR. WIGL2 is a one-dimensional, two-group, space-time diffusion theory program with delayed neutron, thermal-hydraulic, and xenon feedback capabilities. Since this program was written for pressurized light water reactors, certain modifications were required in the thermal hydraulic feedback segment to incorporate organic coolant properties, spatially varying flowrates, and a uniform coolant temperature rise across the reactor in all coolant channels at steady-state condition. Test problems employing a slab representation of the Demonstration Plant reactor are currently being run on the Philco 212 to gain familiarity with the code and to evaluate the space-time dependence of the power distribution to perturbations of varying magnitude and radial location in the core. A modified version of the code is currently being debugged on the CDC-3600 computer. This version will have expanded edit routines but the major benefit will be to greatly reduce the turn-around time to the computer.

c. Preliminary Kinetics Analysis

A preliminary analog computer study was completed to determine the general effects on reactor behavior of various transients affecting core reactivity. The transients included step and ramp reactivity changes, ramp increases in coolant temperatures, and exponential decreases in coolant flow. Two reactor models were used in the study: a 5-node model with time independent spatial flux shape, and a single-point model patterned after the 5-node model. The value of the power coefficient was also varied to study its effect on the transient behavior.

The typical results of the analog study are shown on Figures 14000-5 through -13. Because of the long transit time (>60 sec) for the reactor outlet coolant to reach the reactor inlet and because of the long time constant associated with the coolant heat exchangers, the transient response data obtained up to a period of 30 sec are very representative of the reactor behavior even though the coolant outlet temperature feedback and other plant feedbacks were not simulated.

Figures 14000-5 through -7 show the typical uncontrolled power response of the reactor to step and ramp changes in reactivity for several values of power coefficients. It can be noted that for small step reactivity perturbations ($<0.02 \times 10^{-3} \Delta k/k$) and small ramp reactivity perturbations ($<0.005 \times 10^{-3} \Delta k/k/\text{sec}$), many seconds must pass before the reactor power starts to rise rapidly. For these small perturbations, which are very seldom exceeded under normal steady-state plant operation, an automatic controller can easily maintain constant power, or the operator can readily control the reactor manually as long as he pays constant attention to the reactor power level.

The main advantage of the negative power coefficient, as shown on Curve 2 of Figure 14000-6 is the inherent stabilizing effect of the negative power coefficient without operator action for the small reactivity perturbations normally encountered during steady-state power operation.

Figure 14000-8 shows the typical response of the reactor to ramp increases in coolant inlet temperature for two values of power coefficient. Since the plant generated perturbations under normal operating conditions are generally smaller than $6^\circ\text{F}/\text{min}$, there appears to be sufficient time for operator corrective action, assuming the operator is constantly monitoring reactor power. As shown on Figure 14000-9, with a simple automatic control system having a control rate of $\pm 0.05 \times 10^{-3} \Delta k/k/\text{sec}$, inlet coolant temperature changes as high as $50^\circ\text{F}/\text{min}$ can be readily controlled.

Figure 14000-5. 750 Mwe Reactor Power Response to Step Reactivity Additions (5-node model — $\alpha_p = +3.1 \times 10^{-5} \Delta k/k/\% P$)

Figure 14000-6. 750 Mwe Reactor Power Response to Step Reactivity Additions (single point model — $\alpha_p = +1.58 \times 10^{-5} \Delta k/k/\% P$; and $-1.24 \times 10^{-5} \Delta k/k/\% P$)

Figure 14000-7. 750 Mwe Reactor Power Response to Ramp Additions of Reactivity (single point model - $\alpha_p = 1.58 \times 10^{-5} \Delta k/k/\% P$; and $-1.24 \times 10^{-5} \Delta k/k/\% P$)

Figure 14000-8. 750 Mwe Reactor Power Response to Ramp Increases in Inlet Coolant Temperature (single point model - $\alpha_p = 1.58 \times 10^{-5} \Delta k/k/\% P$; and $-1.24 \times 10^{-5} \Delta k/k/\% P$)

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Figure 14000-9. 750 Mwe Reactor Power Response with a Reactivity Controller to Ramp Increases in Inlet Coolant Temperatures (single point model - $\alpha_p = +3.1 \times 10^{-5} \Delta k/k/\% P$)

The most severe predictable power transients in the reactor are those due to a decrease in coolant flow associated with coolant pump stoppages or valve closures. Figures 14000-10 and -11 show the uncontrolled reactor power response to a decrease in coolant flow from 100 to 75% of full flow on various exponential periods and for several power coefficients. As shown on these figures, the general effect of decreasing the power coefficient is to decrease the rate of power rise in the initial stages of the transient.

The complete loss of flow accident which can be caused by the closure of all coolant valves or by the loss of coolant pumping power represents a limiting transient condition which must not result in intolerable core damage. Figure 14000-12 shows the uncontrolled and unterminated temperature transients associated with a complete loss of coolant flow on an exponential period of 15 sec. It can be readily noted on these figures that decreasing the power coefficient from $+1.58 \times 10^{-5} \Delta k/k/\text{sec}$ to $-0.39 \times 10^{-5} \Delta k/k/\text{sec}$ has only a small effect on the transient responses. Because of this, protection of the reactor must be based on operation of the reactor shutdown system.

Figure 14000-10. 750 Mwe Reactor Power Response to Exponential Decrease in Coolant Flow from 100 to 75% (single point model – $\alpha_p = +1.58 \times 10^{-5} \Delta k/k/\% P$)

Figure 14000-11. 750 Mwe Reactor Power Response to Exponential Decrease in Coolant Flow from 100 to 75% (single point model – $\alpha_p = -0.39 \times 10^{-5} \Delta k/k/\% P$)

Figure 14000-12. 750 Mwe Reactor Coolant, Fuel, and Fuel Clad Temperatures Response to Exponential Decrease in Coolant Flow from 100 to 0% (single point model — $\alpha_p = +1.58 \times 10^{-5} \Delta k/k / \% P$; and $-0.39 \times 10^{-5} \Delta k/k / \% P$)

Figure 14000-13 shows the response of the reactor power and fuel clad temperature to the loss of coolant flow which is terminated by the shutdown system. Although the figure shows that the reactor trip signal came at 130% of full power, the reactor would normally be tripped at 110% power. The transient resulting from tripping the reactor at the 130% point is approximately representative of initiating the loss of coolant flow with the reactor at 80% power and tripping the reactor at 110% power. It should be noted that the peak clad temperature reached becomes higher as the power coefficient is decreased; this is due to the larger integrated power seen by the fuel for the slightly slower transients.

The study also indicated that the most economical outlet header pressure is 100 psia, which agrees with the initial design work. The latter conclusion means that the advantages of locating the main coolant pumps in the cold leg can be obtained with no changes in the present reactor design. Figures 14000-14 and -15 show the pressures and temperatures in the main heat transfer system for both pump locations with the present reactor outlet pressure.

Figure 14000-13. 750 Mwe Reactor Power and Fuel Clad Temperature Response to Exponential Decrease in Coolant Flow from 100 to 0% (single point model – $\alpha_p = +1.58 \times 10^{-5} \Delta k/k/\% P$; $+1.17 \times 10^{-5} \Delta k/k/\% P$; and $-0.39 \times 10^{-5} \Delta k/k/\% P$)

Figure 14000-14. Coolant System Operating Pressures (pump in hot leg, 100 psia outlet header pressure)

Figure 14000-15. Coolant System Operating Pressures (pump in cold leg, 100 psia outlet header pressure)

For the trade study, a margin on the net positive suction head (NPSH) was incorporated by assuming that the coolant contained the maximum operating water content of 500 ppm instead of the nominal 200 ppm specified for normal operation. For additional margin on NPSH, the pump speed was set about 18% below present commercial limitations for the available suction pressure.

The principal advantages in pump reliability and plant safety gained by location of the pump in the cold leg are:

1) Pump reliability

- a) Lower thermal stresses (transient and steady-state) in the pump casing
- b) Lower heat transfer rate to shaft seals and bearings
- c) Lower pressure inside pump casing and across seals

2) Plant Safety

- a) Lower organic pressure in superheater shell
- b) Lower steam-organic pressure difference in reheat
- c) Reheat steam pressure greater than organic pressure above 60% plant load
- d) Lower temperature of coolant leakage in the event of seal failure
- e) Additional means of detecting excess water in coolant by abnormal pump operation.

Location of the main coolant pumps in the cold leg also produces economic advantages because of the smaller pump size and lower power requirements. These results are due to the higher fluid density in the cold leg.

C. PLANT ENGINEERING

The major effort in HWO CR Demonstration Plant engineering during this report period was the completion of trade studies and the preparation of trade study reports. This work was undertaken to provide detailed evaluations of the alternative designs in the principal design areas of the plant and to document justification for the designs selected. The plant size basis selected for these studies varied according to the design area covered by the study. Design areas peculiar to the HWO CR's and which the Demonstration Plant is required to prove were studied on the basis of Reference Plant size (1000 Mwe). Design areas of a more conventional nature were studied on the basis of Demonstration Plant size (500 Mwe rated, 750 Mwe design).

1. Main Heat Transfer System

a. Main Coolant Pump Location and Reactor Outlet Pressure

Previously prepared designs* for the HWO CR main heat transfer system located the main coolant pumps in the hot leg of the system, between the reactor outlet header and the superheater inlet. The pressure levels in the main heat transfer system were controlled by maintaining a pressure of 100 psia in the reactor outlet header. This arrangement was selected because the pressure tubes in the reactor were maintained at a minimum thickness while system pressure levels were sufficient to prevent cavitation at the reactor outlet and the coolant pump suction. Any increase in outlet header pressure would require an increase in pressure tube wall thickness and a consequent decrease in neutron economy. For the system conditions, the main coolant pumps were originally estimated to require a minimum of 110 ft NPSH. Location of the pumps in the cold leg with this NPSH available would have required an increase in outlet header pressure resulting in a greater thickness of pressure tubes and hence higher fuel costs.

*AI-CE-Memo-6, "HWO CR Reference Plant Conceptual Design"
AI-CE-Memo-25, "HWO CR Demonstration Plant Conceptual Design"

Since the initial design work, there have been sufficient changes in coolant flowrate and system pressure drop to justify a more detailed evaluation of pump location and outlet header pressure. Accordingly, a trade study was performed comparing plant designs with the pumps located in the hot leg and in the cold leg over a range of outlet header pressures. This evaluation was based on the 1000-Mwe Reference Plant size. However, the study results and recommendations are also applicable to the 500-Mwe Demonstration Plant.

The conclusions of the study indicated that the main coolant pumps should be located in the cold leg of the main heat transfer system. This conclusion was based on the lower equipment and operating costs, and on improved plant safety and component reliability which would result from the change in pump location.

The principal factors that affect the choice of outlet header pressure are increased pump costs at lower pressures (due to lower allowable speeds and larger pump sizes) and the increased operating costs at higher outlet header pressures (due to reduced neutron economy and increased fuel enrichment). Table 14000-1 lists some of the pertinent operating conditions and resulting cost differences calculated for the 1000-Mwe Reference Plant.

b. Main Coolant and Main Feedwater Pump Drives

A trade study was completed comparing two different electric motor drive arrangements and a steam turbine drive arrangement for the main coolant and main feedwater pumps. For this study, the same type of drive arrangement was used for both the coolant and feedwater pumps. The evaluation was made for the 1000-Mwe Reference Plant size. The conclusions are valid for the 500-Mwe Demonstration Plant, but with slight differences in the numerical values of calculated cost and performance.

The following three drive arrangements and corresponding methods of flow control were studied:

- 1) Direct synchronous motor drive with flow control by throttling.
- 2) Synchronous motor drive through a fluid coupling with flow control by variation of coupling output speed.
- 3) Steam turbine drive through a gear reducer with flow control by variation of turbine speed.

TABLE 14000-1
PLANT OPERATING CONDITIONS AND
COST DIFFERENCES

Pump Location	Hot Leg			Cold Leg		
Outlet Header Pressure (psia)	100	125	150	100	125	150
Pump						
Flowrate per pump (gpm)	78,500	78,500	78,500	71,600	71,600	71,600
Total head (ft)	945	945	945	860	860	860
Shaft power requirement (hp)	17,400	17,400	17,400	15,810	15,810	15,810
Shaft speed (rpm)	1345	1960	2500	1350	1940	2465
Pump suction pressure (psia)	106.8	131.8	156.8	63.3	88.3	113.3
Fluid temperature (°F)	750	750	750	574	574	574
Fluid vapor pressure (psia)	70	70	70	24	24	24
Pump discharge pressure (psia)	440.8	465.8	490.8	397.1	422.1	447.1
Main Coolant System						
Superheater inlet pressure (psia)	438.2	463.2	488.2	106.8	131.8	156.8
Evaporator inlet pressure (psia)	422.1	447.1	472.1	90.7	115.7	140.7
Reactor inlet header pressure (psia)	334	359	384	334	359	384
Plant Economics						
Net plant heat rate (Btu/kw-hr)	9813	9813	9813	9777	9777	9777
Differential fuel cost ($\text{\$/10}^6$ Btu)	Base	0.0754	0.1453	Base	0.0754	0.1453
Capital fuel cost difference (\$)	167,000	633,000	1,067,000	Base	467,000	900,000
Capital equipment cost difference (\$)	255,500	-197,800	-402,700	Base	-399,600	-587,400

The results of the trade study confirm that the present design which uses steam turbine drives is the best arrangement. This conclusion was based principally on economic considerations, although other factors also support the use of turbine drives instead of synchronous motors.

The evaluation was based on maintaining the same gross output from the turbine-generator (1088 Mwe) for each pump drive arrangement. Under this condition, the turbine drive results in a higher net electrical output from the plant and an improved net plant heat rate. Although the use of electric motors would result in a lower cost for capital equipment, the deciding economic factor in favor of turbine drives is the credit for the excess capability when using the demand charge of \$75.00 per kilowatt. Table 14000-2 summarizes the major economic differences between the three pump drive arrangements.

TABLE 14000-2
PUMP DRIVE TRADE STUDY RESULTS

Pump Drive Arrangement	Direct Motor Drive	Motor Drive with Fluid Coupling	Turbine Drive
Net plant output (Mwe)	1025	1016	1076
Net plant heat rate (Btu/kw-hr)	9920	9980	9810
Fuel plus organic unit ($\$/10^6$ Btu)	7.57	7.57	7.57
Capital fuel plus organic cost difference at 1076 Mwe net (\$)	516,000	809,000	Base
Capital equipment cost difference at net plant output (\$)	Base	988,000	1,782,000
Excess capability cost to reach 1076 Mwe at 75/kw (\$)	3,825,000	4,500,000	Base
Evaluated cost difference (\$)	2,559,000	4,515,000	Base
Justified incremental investment (\$)	968,000	Base	5,309,000

In addition to the economic advantages to be gained at maximum flowrates, turbine driven pumps are also more economical to operate at reduced flowrates. This economy at part-load operation is achieved by reducing the turbine speed. Although variable speed operation can also be achieved with fluid couplings, the slip losses of the coupling are greater than the turbine losses. In the event of emergency, dual-inlet turbines (which were selected for startup operation) can be quickly switched from the normal reheat steam supply to the main steam supply. In order to switch large synchronous motors to an alternate power supply, a delay period is required to prevent motor damage. During this delay (approximately 1 sec minimum), the pump speed and flowrate may drop sufficiently to cause plant control problems.

There has been little operating experience with large synchronous motors suitable for driving the main coolant pumps (approximately 18,000 hp). Steam turbines with the required power rating are relatively common and sufficient operating experience has been obtained to verify their reliability.

c. Steam Generating Equipment

The several design concepts of HWOCR steam generating equipment that have been considered were re-evaluated on the basis of the system requirements which are reported in AI-CE-Memo-25. A preliminary analysis justified the selection of the central manifold header design (CMHD) concept for the evaporator, superheater, and reheater units for the Demonstration Plant. The advantages of this design include lower manufacturing cost and minimized thermal stresses in the cylindrical tubesheet. The design provides essentially two heat exchangers in one shell.

A more detailed design and cost analysis of the Demonstration Plant steam generator based upon the CMHD is currently being conducted. The objectives of this analysis are (1) to delineate by means of a detailed design study any performance or fabrication problems that were not discernible in the preliminary studies performed to date, and (2) to provide steam generator designs of sufficient detail to enable accurate final cost estimates. The steam generator manufacturing and design experience of Combustion Engineering are being utilized in this study.

The first phase of this detailed analysis is the CMHD evaporator, superheater, and reheater as shown in AI-CE-Memo-25. The initial evaporator design being studied does not include the preheater. Later studies will be made of the evaporator with preheater. In a second phase effort, other steam generator designs, such as once-through units and vertical U-tube units, will be studied to ensure that the final selection incorporates the requisite high performance reliability and lowest evaluated costs.

The design study of the evaporator with no preheat zone and the superheater and reheater has been completed. The heat transfer surface area requirements compare closely with the preliminary studies. A comparison of the surface areas is shown in Table 14000-3.

Steam generators based upon the central manifold header design concept have been in use at the Hanford New Production Reactor for several years. Operational experience has been acquired with ten units. Tube material problems have been reported but they were found to be unique to the operation at Hanford. Additional minor problems of vibration of tubes in the outer region of the tubes bundle have been corrected by the use of additional tube bundle supports.

TABLE 14000-3
HWO CR DEMONSTRATION PLANT STEAM GENERATOR
SURFACE AREA

Unit	Heat Transfer Surface Area Requirements	
	Preliminary Studies (ft ²)	Revised Results (ft ²)
Evaporator (no preheat zone)	65,250	64,500
Superheater	33,800	29,300
Reheater	44,000	42,400

2. Reactor Shutdown Cooling System

A comprehensive report on this system was initiated during this quarter. The basic system and its design characteristics have been previously described in AI-CE-Memo-25. However, the report currently in preparation is additionally providing the technical backup and design basis for the system, the basis of design for the system and its major equipment, and a separate discussion of the operation of the decay heat removal system for each of the following assumed plant conditions:

- 1) Normal operation with the reactor at power
- 2) Operation with the reactor shut down
 - a) Main loop operating
 - b) Main loop isolated
- 3) Operation with a loss of system pressure
- 4) Operation with a coolant leak
 - a) Leaks less than 50 gpm
 - b) Leaks between 50 and 2000 gpm
 - c) Leaks greater than 2000 gpm
- 5) Operation with a steam leak
 - a) Small leaks
 - b) Large leaks

From the work accomplished to date, it is apparent that the interrelated hydraulics of the system and the fuel element heat transfer under the several operating modes need more analysis. Also, the passive operation of the shutdown cooling system is based on the satisfactory operation of many large check valves, and the reliability of the valves requires evaluation.

5. Coolant Purification System

A trade study report was completed recommending the use of a single-stage flash distillation column with a stripping section, as used at PNPf and OMRE, in conjunction with an adsorption system to maintain coolant composition and quality. This system is low in annual costs, proven in operation, and offers flexibility of operating parameters.

a. Coolant Processing Rate

The total coolant processing rate was estimated by extrapolation of data from operation of the OMRE Core III purification systems and the resulting coolant conditions. For the Demonstration Plant design conditions, a coolant purification rate of 57,000 lb/hr was calculated. This flowrate is higher than that reported in AI-CE-Memo-25, "500-Mwe Nuclear Demonstration Plant Conceptual Design," which shows a processing rate of 31,000 lb/hr. The lower rate was obtained by scaling up the OMRE processing rate on the basis of total energy absorbed by the coolant. Later consideration indicated that the coolant radiation damage should be a more valid scale factor than total energy. Use of this scale factor results in the higher rate of 57,000 lb/hr.

b. Alternate Purification Methods Evaluated

The methods evaluated, both individually and in combination, were as follows.

(1) Coolant Composition Control

a) High process rate "deep cut" distillation in which a portion of the high boilers or HB (quaterphenyls and some hexaphenyls) is distilled and returned to the main loop with the distilled terphenyl. Return of distilled HB enables a high throughput to maintain coolant quality without the concomitant reduction in HB level.

b) Conventional OMRE or PNPf-type simple flash distillation in which only material of vapor pressures equal to or higher than para-terphenyl are distilled overhead and returned to the main loop. All lower vapor pressure fractions are removed from the bottom of the column.

(2) Coolant Quality Control

a) High process rate "deep cut" distillation in which the distillation rate is established by the estimated impurities generation rate and the maximum allowable concentration of impurities in the coolant.

b) Removal of impurities by adsorption on activated clay.

c) Removal of impurities by filtration.

TABLE 14000-4
EFFECTIVENESS OF ALTERNATE COOLANT
PURIFICATION SYSTEMS

Criteria	System				
	Deep Cut Distillation	Conventional PNPf-OMRE Distillation	Adsorption	Full Flow Filtration	Conventional Distillation Plus Adsorption
Control of Coolant Composition (HB Control)	Acceptable	Acceptable	Unacceptable (Does not control HB)	Unacceptable (Does not control HB)	Acceptable
Control of Coolant Quality	Acceptable	Unacceptable (Flow rate too low)	Acceptable	Unacceptable (Does not remove colloidal material)	Acceptable
Operating Experience Available	No	Yes	Yes	Yes	Yes
Processing Rate Adaptable to Change	No	No	Yes	No	Yes

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Figure 14000-16. Annual Coolant
Purification Costs

The effectiveness of the alternates, either individually or in combination, in achieving the dual objectives of coolant composition and quality control is summarized in Table 14000-4 and shows that only the "deep cut" distillation process, or a simple flash distillation system combined with adsorption, satisfies the criteria for both coolant composition and quality control. These two alternates were the only ones which were given further analysis.

In the "deep cut" distillation system, the total purification flow is processed by the distillation system. The total flow is much higher than that required to maintain the HB content at 10% at full power operation. The HB level is maintained at the desired concentration by distilling a portion of the quarterphenyls and hexaphenyls (currently considered as HB fractions) and returning it to the main coolant stream with the distilled terphenyl. Assuming the distillation process is 100% effective in the removal of potential film-forming impurities, this approach offers the potential of combining all coolant processing requirements into one system. However, this system has a high capital cost as well as a high operating cost due to the high heat requirements to provide the latent heat to the high coolant throughput.

The conventional distillation system and an adsorption system, when employed together, form a very satisfactory purification process. The flowrate to the distillation column can be adjusted to control the HB level in the coolant and the remainder of the coolant can be processed in the adsorption system to maintain coolant purity. The major uncertainty in the adsorption process involves the capacity of the clay to hold impurities. Variation in clay capacities affects the total amount of clay required and thus the cost of the process.

Figure 14000-16 presents the annual charges for a "deep cut" distillation system and for a combined simple distillation-adsorption for three different values of E/A (pounds coolant effluent per pound adsorbent) for the clay. This figure shows that over a wide range of total purification rates and for a range of E/A values considered reasonable for the HWO CR, the combined simple distillation-adsorption system results in the lowest purification cost.

4. Coolant Recovery System

A study was initiated to evaluate alternate methods of generating H_2 for use in the hydrocracker portion of the coolant recovery system and for possible

recovery of H_2 from the plant waste gas stream. Methods being evaluated for H_2 supply are (1) purchased liquid H_2 , (2) production by electrolysis of H_2O , (3) production by NH_3 dissociation, and (4) production by steam-hydrocarbon reforming. Methods of possible H_2 recovery from the waste gas stream are (1) cryogenic recovery, and (2) diffusion through a palladium barrier.

5. Coolant Pressurization System

Alternate means of coolant pressurization were evaluated. The system recommended by the study is one which combines the degasification and pressurization functions in a single system, and utilizes pumps to provide hydraulic pressurization. This process has been proven in operation at PNPf and is the most economical of the alternates considered.

The following paragraphs present a brief discussion of the alternate systems evaluated.

a. Hydraulic Pressurization Combined with Degasification

Hydraulic pressurization can be provided either by pumps or by a static fluid head. When the pressurization system is combined with the atmosphere pressure degasifier, pressurization by static fluid head is very costly due to the elevation required (300 ft) to achieve the necessary hydraulic pressure. Pressurization with pumps was shown to be much more economical.

b. Nitrogen Pressurization with Separate Degasifier

Pressurization with N_2 requires maintaining and controlling the gas pressure above a free surface of the liquid. As the coolant volume changes due to changing coolant temperature, the complete surges are contained by the surge tank with subsequent venting and repressurizing with N_2 . The quantity of N_2 consumed is a function of temperature-volume cycles of the coolant. In addition to the N_2 consumption costs, the costs of a separate degasifier must be included.

c. Pressurization with Coolant Vapor Pressure with Separate Degasifier

Coolant pressure control can be achieved by controlling the coolant temperature within a surge tank and utilizing the vapor pressure of the coolant to provide the system pressure. The major disadvantage of this method of pressurization is the thermal decomposition of the coolant associated with maintaining a tank of coolant at elevated temperatures. In addition, a separate degasification system would be required.

Table 14000-5 summarizes the estimated annual charges for the alternates considered. These costs illustrate the large economic advantage of combining the degasification and pressurization functions and utilizing pumps to supply the hydraulic pressure.

TABLE 14000-5
ANNUAL COSTS OF ALTERNATE PRESSURIZATION METHODS

Cost Items	Systems					
	Hydraulic Pressurization with Degasification		N ₂ Pressurization plus Degasification		Vapor Pressure Pressurization plus Degasification	
	With Pumps	With Static Head	Pressurization	Degasification	Pressurization	Degasification
Capital Equipment (\$)	71,000	414,000	25,000	71,000	250,000	71,000
Annual Charges at 0.12/yr (\$/yr)	8,500	50,000	3,000	8,500	30,000	8,500
Operating Cost (\$/yr)	3,700	-	17,000	3,700	2,000	3,700
Maintenance Cost (\$/yr)	7,000	1,000	1,000	7,000	1,000	7,000
			21,000	+ 19,200	33,000	+ 19,200
Total	19,200	51,000	40,200		52,200	

6. Selection of Reactor Coolant

A trade study was conducted to select the best polyphenyl organic coolant for the HWO CR plant. The comparisons included thermal and radiation stability, melting point, safety characteristics, cost differentials based on heat transfer, pumping, pressure level, and coolant makeup costs. A commercial mixture of ortho- and meta-terphenyls was recommended as the HWO CR coolant. The preferred mixture is a meta-rich terphenyl, Santowax WR, with Santowax OM as a close contender. The difference between these coolants is small so that tests carried out with Santowax OM coolant should yield applicable information.

7. Moderator Systems

A trade study was performed to evaluate methods for removing moderator heat from the HWOCR. The study was based on the 1000-Mwe Reference Plant size; however, the results are applicable to the Demonstration Plant.

Three methods of disposing of the moderator heat were considered: (1) dumping it directly to cooling water from the river, (2) dumping it indirectly to river cooling water by means of an intermediate heat transfer circuit, and (3) utilizing it to heat condensate in the lowest pressure feedwater heater. It was found that the economic balance points among these three methods were very sensitive to the temperature of the moderator exiting from the reactor. Figure 14000-17 shows the relative annual cost of the three moderator heat disposal systems as a function of moderator outlet temperature. It can be seen that above 190°F outlet temperature the feed heating method is economically superior to direct dumping to the river; however, if indirect dumping to the river is used, the feed heating method is more economical only at outlet temperatures above 180°F. A final choice of a method of disposing of moderator heat in the Demonstration Plant must await further work in two areas: (1) a determination of the

Figure 14000-17.
Moderator Cooling System
Annual Cost vs Moderator
Outlet Temperature

maximum gross outlet temperature of the moderator with a proper margin against local boiling anywhere in the calandria tank, and (2) an evaluation of the problems inherent in dumping moderator heat directly to the river, such as contamination of the river and loss of moderator inventory through heat exchanger leakage.

8. Turbine Plant

a. Turbine Cycle Arrangement (Non-reheat vs reheat)

A trade study was performed to determine whether a non-reheat or a reheat steam turbine cycle arrangement is the better selection for the HWOCR. The study was based on the 1000-Mwe Reference Plant size. The non-reheat cycle uses slightly superheated steam at the

inlet to the low pressure turbine section, whereas the reheat cycle employs an organic coolant heat exchanger which reheats the steam between the turbine high and low pressure sections. Capital and fuel operating cost information for the two arrangements was determined and economic comparisons were made. Other aspects that influence the cycle selection, such as reliability, plant safety, and operation were also evaluated. The results of the study indicated that the steam cycle employing reheat steam provides an improvement in net plant cycle efficiency of approximately 1.7% and a net plant heat rate gain of 520 Btu/kw-hr when the cross-over steam is reheated to 725°F. This improvement in heat rate justifies an incremental investment as high as \$2,500,000 for an investor-owned plant at a 12% fixed charge rate. The estimated capital cost of the reheat cycle arrangement was only \$520,000 higher than for the non-reheat cycle. Based on the comparison of differential fuel and capital costs, a total annual savings of \$238,000/year is possible with the reheat steam cycle arrangement for an investor-owned plant.

In view of the apparent cost savings, improved heat rate, improved turbine operation, reduced turbine maintenance, and improved overall plant performance, it was concluded that the reheat steam cycle is the better arrangement for the HWO CR. This more detailed evaluation of turbine cycles justified the reheat cycle arrangement selected in the previously developed conceptual design of the Reference and Demonstration Plants (AI-CE-Memo-6 and -25).

Table 14000-6 shows a summary of the evaluated costs of the two cycle arrangements. A summary of the advantages and disadvantages of each cycle arrangement is listed in Table 14000-7.

b. Turbine Design

The conceptual design of the HWO CR Demonstration Plant as described in AI-CE-Memo-25 utilizes a single 750-Mwe TC4F-52 (tandem-compound, four-flow low pressure section, 52-in. last-stage buckets) turbine. During this report period, this unit was compared on technical and economic bases with five other single full-size turbines and with three pairs of half-size turbines. The study considered the capital cost of the turbines and the turbine cycle auxiliary equipment and condensers, and the associated fuel cycle costs and heat rate. The analysis for the single full-size units indicated that the lower capital cost

TABLE 14000-6
TURBINE CYCLE TRADE STUDY
(Summary of Evaluated Costs)

Cost Item	Units	Non-Reheat Cycle	Reheat Cycle
Net station heat rate	(Btu/kw-hr)	10320	9800
Net station heat rate differential	(Btu/kw-hr)	520	Base
Net plant output	(Mwe)	1076	1076
Fuel and organic charge per Btu/kw-hr	(\$/yr)	575	575
Fuel and organic makeup cost	(\$/yr)	300,000	Base
Capital worth of heat rate penalty at 12% fixed charges - investor owned	(\$)	2,500,000	Base
at 7% fixed charges - public owned	(\$)	4,280,000	Base
Differential capital cost of equipment	(\$)	Base	520,000
Subtotal of fuel cost and capital cost evaluated at 12%	(\$)	2,500,000	520,000
at 7%	(\$)	4,280,000	520,000
Differential evaluated cost - fuel and capital cost at 12%	(\$)	1,980,000	Base
at 7%	(\$)	3,760,000	Base
Justified incremental investment at 12%	(\$)	Base	2,500,000
at 7%	(\$)	Base	4,280,000
Total differential annual savings using reheat at 12%	(\$/yr)	—	238,000
at 7%	(\$/yr)	—	262,000

TABLE 14000-7
TURBINE CYCLE TRADE STUDY
(Summary of Advantages and Disadvantages)

Cycle	Advantages	Disadvantages
Non-Reheat	<ol style="list-style-type: none"> 1. Arrangement is simplified for equipment and piping layout. 2. No possibility of organic coolant leaking into steam turbines. 3. Lower overall capital costs. 4. Turbine intercept valves not required. 	<ol style="list-style-type: none"> 1. Lower thermal efficiency. 2. Higher moisture content in turbine last-stage buckets. 3. Larger turbine auxiliary equipment. 4. Larger reactor complex. 5. Higher fuel cycle costs.
Reheat	<ol style="list-style-type: none"> 1. Better fuel economy. 2. Lower annual fuel costs. 3. Low moisture content in turbine low pressure sections. 4. Turbine plant auxiliaries are smaller. 5. Reactor is smaller with fewer pressure tubes. 6. Heavy water inventory is less. 	<ol style="list-style-type: none"> 1. Higher overall capital cost. 2. Reheater rupture leaks organic into steam turbines. 3. Main coolant system more complex. 4. Reheater steam piping and intercept valves cause a layout problem. 5. If containment changes to install main coolant loops inside containment vessel, reheater installation capital cost increases.

of a 750 Mwe, TC-4F-43 unit more than offset the economic advantages of the turbine cycle for the same unit with 52-in. last-stage buckets. Also, it was determined that fuel cycle costs for the Demonstration Plant would have to increase by approximately 50% (to $10.5¢/10^6$ Btu) before the TC-4F-52 unit would be economically justified for a publically-owned station. To justify the 52-in. bucket turbine for an investor-owned station, the fuel cycle costs would have to be considerably greater than $10.5¢/10^6$ Btu.

9. Plant Control System

Analytical tasks deriving data for the development, evaluation, and selection of a plant steady-state control program were initiated. Preliminary part-load operational data have been calculated for the following potential control programs.

Program 1: Constant coolant flow is maintained over the full power range. Reactor inlet and outlet coolant temperatures increase with increasing power level. Steam pressure increases slightly.

Program 2: Coolant flow is varied (50 to 100%) over the power range of 0 to design power level. Reactor inlet and outlet coolant temperatures increase with increasing power level. Steam pressure is maintained constant.

Program 3: Coolant flow is varied (50 to 100%) over the power range of 0 to rated power (500 Mw) and maintained constant between rated and design power level (750 Mw). Reactor inlet and outlet coolant temperatures increase with increasing power level. Steam pressure is maintained constant.

Program 4: Coolant flow is varied (50 to 100%) between the power range of 30% to design power level. The average coolant temperature is maintained constant. Steam pressure decreases with increasing power level.

Program 5: Constant flow is maintained through the core. Boiler organic flow is varied by means of a bypass system. The average coolant temperature and steam pressure are maintained constant.

Program 6: Over the power range of 50 to 100%, coolant flow is varied (50 to 100%); constant coolant ΔT is maintained across the reactor. Below 50% of design power level, constant core flow is maintained at 50% of rated flow; the reactor inlet and outlet coolant temperatures decrease and increase respectively with increasing power level; and boiler bypass is employed. Steam pressure is constant over the full power range.

Program 7: Variable coolant flow (50 to 100%) and boiler bypass are employed over the full power range. The reactor inlet and outlet coolant temperatures are constant from 50 to 100% of design power level. The average coolant temperature and steam pressure are constant over the full power range.

A mathematical model of the primary coolant loop was formulated. The model includes the following system components:

- a) Reactor (kinetics and heat transfer)
- b) Reactor control (including control rods)
- c) Coolant pump
- d) Coolant pump turbine driver
- e) Coolant isolation valves and actuators
- f) Superheater
- g) Reheater
- h) Evaporator
- i) Piping, headers, mixing tanks.

The mathematical model was programmed on the analog computer. Three analog computer consoles were combined to allow simulation of the complete organic closed loop. Steady-state and dynamic checks were made of the various system components and the total primary coolant system. Steady-state values checked to within 0.5 to 1.0°F.

Derivation of a simulation model of the steam plant is partially completed. When completed, the steam plant simulation will be added to the primary system model; thus the total model will comprise a complete simulation of reactor and main heat transfer system.

10. Plant Layout and Piping

a. Reactor Building Interface Control Drawing

An interface control drawing for the reactor building complex was developed. This drawing is the initial step in establishing a method of communicating design layout information among the participants in the Demonstration Plant design effort.

b. Piping Design Criteria

Preliminary criteria were established for the design and stress analysis of the main heat transfer piping system. The preliminary criteria considered, and are consistent with, the latest draft of the proposed ASA Nuclear Piping Code.

c. Water Hammer Study

A study of the potential water hammer problems associated with rapid closing of valves in the coolant piping systems was initiated. The first part of the analysis investigated the case where both primary coolant pumps are shut down simultaneously and the containment isolation valves in each loop are closed at a constant rate within 10 sec. For this case, the calculated generated pressure surge is only 13.5% higher than normal operating pressure.

11. Structures

a. Containment Building Concepts

Conceptual structural design and cost evaluation studies of the following hemispherical containment building concepts were conducted:

- 1) A hemispherical steel pressure vessel to handle internal pressure with a reinforced-concrete hemispherical dome external to the steel vessel to provide shielding and handle external loads, and a reversed-dish steel-lined reinforced-concrete base slab and thrust ring.

- 2) A steel-lined reinforced-concrete hemispherical dome over the same reversed-dish base, the reinforced concrete to handle both internal and external loads.

3) A BONUS-type concept consisting of a heavy steel-lined reinforced-concrete hemisphere, with a steel-lined flat slab base. Internal and external loads handled by the concrete cylinder and hemisphere, with uplift forces due to internal pressure resisted by dead weight of the cylinder and hemisphere.

The study showed that of the three hemispherical concepts considered, the BONUS-type is the least costly. However, all three hemispherical concepts are more costly than the cylindrical concept developed for the Demonstration Plant conceptual design and described in AI-CE-Memo-25. Pertinent design features and cost elements for the various concepts are summarized in Table 14000-8.

TABLE 14000-8
SUMMARY OF HEMISPHERICAL CONTAINMENT CONCEPTS

Item	Cylindrical Concept (AI-CE Memo-25)	(1)	(2)	(3)
		Steel Pressure Vessel Concrete Shielding Dished Base	Reinforced Concrete Steel Liner, Dished Base	Reinforced Concrete Steel Liner, BONUS-Type
Diameter (ft)	140	375	375	332
Height of cylinder (ft)	152	-	-	22
Height of hemisphere (ft)	70	187	187	166
Total height (ft)	222	187	187	188
Excavation (\$)	267,000	562,000		
Steel plate liner (\$)	836,000	3,144,000	2,418,000	1,992,000
Concrete cylinder (\$)	1,008,000	-	-	1,047,000
Concrete hemisphere (\$)	376,000	1,500,000	2,062,000	1,752,000
Concrete thrust ring (\$)	280,000	2,310,000	2,310,000	-
Concrete Base Slab (\$)	72,000	540,000	540,000	372,000
Total \$	2,839,000	8,056,000	7,892,000	5,836,000

12. Miscellaneous Activities

a. Specifications

Following a realignment of the specification tree, a review of the previously prepared end item specifications was made, and modifications and new specifications were initiated as required. The revised tree lists end item specification requirements for the main coolant pumps and drives, main coolant block valves, turbine-generator, and steam generators. In addition, the overall system specification was subdivided into a system specification for the entire plant and separate system segment specifications for the principal plant component segments.

b. Plant Design Characteristics

The 500-Mwe HWO CR Demonstration Plant design operating characteristics published in AI-CE-Memo-25 were reviewed and updated in conjunction with the trade study evaluations (not including the primary coolant pump relocation). The basic data for the rated and the design conditions of the Demonstration Plant are given in Figures 14000-18 and -19.

c. Architect-Engineer Services

In the design of the HWO CR Demonstration Plant, the services of an architect-engineer will be used to supplement the design efforts of Atomics International and Combustion Engineering. To define the scope of work for the architect-engineer, a detailed listing of the division of project effort between the architect-engineer, Atomics International, and Combustion Engineering was prepared for the entire project, including Title I (preliminary engineering), Title II (final engineering), Title III (construction engineering), and construction management. Also, a complete description of the expected architect-engineer and construction management services was prepared.

d. Plant Scheduled Maintenance

A preliminary study of the Demonstration Plant scheduled maintenance requirements was completed. The estimated maintenance data used in the study were based on typical requirements for conventional equipment in similar service. As the design progresses and more definitive information on equipment and systems is available, these estimates will be updated to establish the maintenance requirements for the final plant design.

Figure 14000-18. Reactor Plant Heat Balance - Rated Load

Figure 14000-19. Reactor Plant Heat Balance - Design Load

Maintenance requirements were estimated for the following systems and subsystems:

- 1) Plant Control System
- 2) Plant Protective System
- 3) Aqueous and Gaseous Waste Disposal System
- 4) Component Cooling System
- 5) Moderator System
- 6) Coolant Conditioning Subsystem
- 7) Feedwater Conditioning Subsystem
- 8) Reactor Shutdown Cooling Subsystem
- 9) Primary Heat Transfer Subsystem
- 10) Superheater, Reheater, and Steam Generator Subsystem
- 11) Steam Supply Subsystem
- 12) Turbine-Generator Subsystem
- 13) Electrical Subsystem
- 14) Leakage Detection Subsystem
- 15) Radiation Detection and Monitoring Subsystem

III. NEXT REPORT PERIOD ACTIVITIES

A. GENERAL STUDIES

For the next quarter the analog of the organic coolant system will be checked for stability of operation and the cases delineated in the table of accidents will be operated. The core rupture model for the loss of coolant accident will be placed on an analog system if the above operation is stable. If not, programming for digital solution will be initiated. Pressure buildup associated with the in-core rupture of the pressure calandria tube will be developed as a computer model similar to models developed for boiling and pressurized water reactors.

B. REACTOR DESIGN

1. Reactor Structures

a. Calandria Structural Studies

The selection of the materials for the reactor vessel and the method of support will be made on the basis of the cost estimates and the results of the structural analysis.

Design of the light water cooling systems for the end shields and the thermal shield will be developed.

Detailed development of the construction schedule and the erection sequence for the reactor area will be performed.

b. Calandria Pressure Relief

The results of the analysis of the tube rupture in the calandria will be interpreted in terms of the calandria design.

c. Pressure Tube Assembly

- 1) Completion of all preliminary component design specifications.
- 2) Continued evaluation of results of development program.

2. On-Power Refueling

On-power refueling design activities will cover the following areas:

- a) Completing the comparative evaluation of alternative concepts including the selection of one as the new reference.
- b) Developing a tool post design suitable for the selected concept.
- c) Developing latching mechanism designs required for manipulation of the fuel column components during refueling.
- d) Developing a concept for alignment of the refueling machine with the pressure tube.
- e) Developing a refueling machine environmental control system design.
- f) Developing a machine closure plug design.

3. Core Design Studies

- a) Continued documentation and evaluation of various trade-offs associated with the reference design.
- b) More detailed studies of thermal hydraulic performance.

4. Nuclear Design

- a) Continued investigation of those factors affecting the power coefficient.
- b) Continued evaluation of trade-offs related to the reference design fuel bundle.

C. PLANT ENGINEERING

A principal effort in the next quarter will be the preparation of a document covering the overall design aspects of the turbine plant and the justification bases for the design selected. This document will consist of information previously generated in the turbine steam cycle and other trade studies, augmented with hitherto undocumented data (as on main condenser sizing and feedwater heating train design).

Other items of planned effort for the next report period are:

- 1) Completion of the MCA containment mathematical model.
- 2) Completion of preliminary gross-effect experiments on UC-H₂O reactions.
- 3) Re-evaluation of siting analyses with respect to organic non-filterable iodines.
- 4) Completion of the plant analog simulation model; analysis of postulated accidents and of the performance characteristics of the plant, systems, and components.
- 5) Specification of the plant control system criteria, selection of the plant steady-state control program, and initiation of the synthesis of the plant control system.
- 6) Completion of trade study reports and end item specifications.

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R. A. Blaine, "Modifications and Extensions of the FORM Code," NAA-SR-Memo-9174 (1963)
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HWO CR QUARTERLY REPORT

TITLE: 1000-Mw Reference Plant Engineering

WBS NO: 15000

REPORTING PERIOD: April through June 1966

I. PROJECT OBJECTIVES

The overall objective of this activity is the design and construction of a 1000-Mwe HWO CR plant so that subsequent operation of this plant will:

- a) Successfully demonstrate economic operation of the HWO CR concept in a plant of this size.
- b) Confirm that the essential HWO CR features demonstrated by the 500-Mwe plant may be scaled up successfully to larger plants.
- c) Provide cost and design data for use in defining large dual-purpose power and desalination plants.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. SAFETY STUDIES

Tritium exposure to the public, which would result from spillage of the entire heavy-water inventory within the containment building, has been determined. The accident was considered occurring both as a separate incident and in conjunction with the major-loss-of-coolant accidents. In both instances, the tritium exposure was within 10 CFR 100 guidelines at nominal siting distances of interest (exclusion distances of 0.25 to 0.50 miles; low population zone distances of 1 to 10 miles) for a containment building leak rate as high as 1.0%/day. The maximum heavy-water spill in the auxiliary building was also estimated and was determined to be capable of causing tritium exposures in excess of 10 CFR 100 limits unless the affected rooms were promptly isolated following the spill.

B. PLANT ENGINEERING

The principal effort during this report period was the revision of the preliminary conceptual design of the 1000-Mwe HWO CR Reference Plant, as described in AI-CE-Memo-6, to incorporate improvements made during the design of the Demonstration Plant, as described in AI-CE-Memo-25. A report describing

the differences between the original (AI-CE-Memo-6) Reference Plant design and the revised design is being prepared.*

The principal difference between the original and revised Reference Plant design is in the coolant temperature at the inlet to the reactor (595°F original, 574°F revised). The reactor outlet temperature remains unchanged at 750°F; thus, the revised reactor ΔT is 176°F. This parameter change resulted from additional optimization studies subsequent to the original conceptual design. In turn, this change induces corresponding changes in coolant flow rate, system pressure drop, coolant pumping power, and the sizing of steam generating units. These basic changes are illustrated in Figure 15000-1, which is a simplified flow diagram and reactor plant heat balance for both the original and revised Reference Plants.

III. NEXT REPORT PERIOD ACTIVITIES

Design of the 1000-Mwe Reference Plant will continue to be reviewed in the light of new information as it becomes available from the Demonstration Plant design efforts. Problem areas unique to the Reference Plant will be investigated also, as each is identified and evaluated.

*AI-CE-Memo-53, 1000-Mwe HWOCR Reference Plant—Revisions to Preliminary Conceptual Design Report (August 1966)

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Figure 15000-1. Simplified Flow Diagram and Reactor Plant Heat Balance

HWO CR QUARTERLY REPORT

TITLE: Dual-Purpose Plant Engineering

WBS NO: 16000

REPORTING PERIOD: April through June 1966

I. PROJECT OBJECTIVES

The dual-purpose (electric power production and water desalting) plant differs from the power-only plant in three major respects. First, the dual-purpose plant can utilize low temperature, high entropy heat in the evaporator sections of the plant, instead of in the low-pressure sections of the turbine. Second, in many areas of the country the requirements for water and power may become such that very large dual-purpose reactors of the order of 8,000 to 10,000 Mwt will be needed. The heavy water pressure tube reactor lends itself to extrapolation to these large sizes in single units. Third, the financing associated with water production or distribution plants differs from that associated with the investor-owned utility power plants. For investor-owned utility power plants, the fixed charge rates range from 10 to 14%. The fixed charge rate for water plants ranges from 4 to 7% in the large public water systems. This difference in fixed charge rates means that for the desalination of water, a high capital cost-low energy cost plant appears more advantageous than a high energy cost-low capital cost unit.

In the light of the above incentives, the objectives of this activity are as follows:

- a) Determine the feasibility of a 10,000-Mwt HWO CR plant.
- b) Determine the near-optimum parameters of fuel systems composed of UC, UO_2 , and U-metal. Determine the economic advantages of each of these fuel systems at its optimum operating parameters.
- c) Determine optimum coupling schemes for a HWO CR type reactor as a function of typical operating parameters associated with the different fuel materials mentioned above.
- d) Perform conceptual designs of an optimized 3,000 Mwt and of a 10,000-Mwt Dual-Purpose HWO CR Plant.
- e) Define areas and show incentives for incremental research and development for desalting applications.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. SELECTION OF FUEL ELEMENTS FOR STUDY

Those fuel element combinations eligible for consideration in the optimization studies in Task B include:

- 1) U-metal fuel material with Al or Zr cladding in the rod, annular, and inverse cluster geometries.
- 2) UO_2 with Zr cladding in the rod, annular, and inverse cluster geometries.
- 3) UC with SAP cladding in the rod, annular and inverse cluster geometries.

Since certain of the possible combinations listed above appear much more promising from a performance, fabrication, or economic standpoint, attention has been focused on these, as shown in Table 16000-1. These elements have been selected on the bases of anticipated performance, reasonable assurance of economical fabrication feasibility, reasonable extrapolation of present day metallurgical technology, and previously gained design experience.

TABLE 16000-1
FUEL ELEMENTS FOR DUAL-PURPOSE PLANT

Number	Fuel Element Configuration	Fuel Material	Cladding Material
1	Rods	UC	SAP
2	Rods	UO_2	Zr
3	Rings	UO_2	Zr
4	Rings	U - Metal	Zr
5	Rings	U - Metal	Al
6	Cluster	UO_2	Zr
7	Cluster	U - Metal	Zr

B. METHOD FOR CONDUCTING OPTIMIZATION

In order to maintain the scope of this study within practical limitations the following parameters have been established:

- 1) Power for all designs will be 1000 Mwe.
- 2) The coolant will be Santowax WR, 10% HB, except at low temperature, where other coolants will be investigated if it becomes desirable.
- 3) No consideration will be given to fuel element fouling.
- 4) The reactor will be orificed to the basic radial power distribution.
- 5) Only vertical, one-coolant-pass, pressure tube type reactors will be investigated.
- 6) Maximum fuel string pressure drop, excluding end plate contribution, will be limited to 120 psi.
- 7) Active fuel length for all designs will be 17.3 ft.
- 8) The refueling will be on-line bidirectional.
- 9) Bidirectional flow will be used except for U-metal designs, where unidirectional flow will be considered.
- 10) Annular fuel element designs will be limited to a maximum of three fuel rings with a center nonfueled support end.
- 11) Rod type fuel element designs will be composed of 37 rods similar to the reference design.
- 12) A D₂O/U ratio of 17.6 will be maintained. Confirmation of this selection will be made.
- 13) The established temperature limits for the various materials are:

Fuel	Temperature (°F)	Cladding	Temperature (°F)	Pressure Tube	Temperature (°F)
UC	2200	SAP	850	Zr-4	700
UO ₂	4500	Zr	800	SAP	800
U - Metal	850	Al	700 to 750		

- 14) The following established fuel exposure limits will require confirmation with respect to optimization:

Fuel	Burnup (Mwd/MTU)
UC	15,000
UO ₂	18,000
U - metal	4,000 - 6,000 - 8,000

Using the above parameters as the basis, each configuration is being optimized for reactor ΔT , fuel element geometry, organic volume fraction, and fuel area, using the criterion of specific cost of heat generation. A compatible nuclear analysis will be performed for each condition. The parameters of each

promising configuration will be studied for various reactor outlet temperatures and plant efficiencies. This information will be combined with parametric plant studies to determine the near optimization of each configuration. Figure 16000-1 shows the logic sequence used in this optimization for a hypothetical case.

Nuclear analysis during the past quarter has been associated primarily with the evaluation of local radial peaking factors (fresh fuel conditions) for preliminary designs of annular and inverse cluster type elements. This effort is part of the iterative procedure required between thermal-hydraulic and nuclear design to arrive at an optimized fuel element design.

C. THERMAL AND HYDRAULIC CODES FOR FUEL ELEMENT OPTIMIZATION

1. Code for Rodded Elements

The computer code THEME-1, developed previously, will be used for the rodded element phase of this study. With minor modifications this code may also be applied to the inverse cluster geometry. This program performs a steady-state thermal and hydraulic evaluation of a multi-rodded fuel element subchannel in a single-pass organic-cooled reactor design. Given the basic reactor parameters of power, exit temperature, temperature rise, fuel length, power distribution, and fuel element geometry, the program will calculate the various thermal and hydraulic parameters for the limiting fuel channel. In addition, coolant, surface temperatures, local velocities, and critical heat flux ratios for various axial positions are calculated for the subchannel. The organic properties are internal to the program in order to maintain a simplified input. The flow distribution for the subchannel is determined by using fuel element pressure drop as a forcing function. An internal subroutine will integrate any axial power distribution at each axial position for determination of the coolant temperature. The Dittus-Boelter correlation for heat transfer coefficient is corrected at each axial position for temperature. Critical heat flux ratio is based on the Core and Sato correlation. An additional feature of the program is a provision for mixing which is performed by reducing the difference in enthalpy rise between the subchannel and the average subchannel of the fuel element.

The following options are available:

- a) ΔT and NPT* search given T_{sm} ,[†] T_{exit} , Orificing, Max Velocity and Core Power.
- b) Velocity and NPT search given T_{sm} , T_{exit} , Orificing, ΔT and Core Power.
- c) T_{sm} and Velocity search given NPT, T_{exit} , Orificing, ΔT and Core Power.
- d) Orificing and NPT search given T_{sm} , T_{exit} , Max Velocity, ΔT and Core Power.

2. Code for Optimization of Annular Fuel Elements

The computer code being developed for optimization of annular fuel elements is similar in performance to the code for rodded elements, with the following exceptions:

- a) A geometry description of all annular channels and fuel rings is an input to the code.
- b) The flow distribution to the annular channels is determined by an iterative pressure-drop balancing technique.
- c) Calculation of coolant temperatures and cladding surface temperatures is performed for every channel and cladding surface.
- d) No provision is presently provided for coolant mixing. A provision will be added to provide for plenum mixing at fuel bundle interfaces.

3. Code for Parameter Study of Heat Transfer System

Detailed computer programs PESO I and II were developed for a parameter study of the primary organic coolant heat transfer system. In general, for a given power generation configuration, the programs perform a steady-state thermal and hydraulic evaluation calculating the heat load, flow, temperatures in the loops, designing heat exchangers and loop piping, estimating the primary loop capital cost and plant capital cost, the energy generation cost in the primary loop, and the total energy generation cost, which includes the reactor primary loop and the turbine plant.

Major inputs include:

- a) Plant gross power generation
- b) Generating efficiency
- c) Steam conditions at turbine throttle
- d) Reactor inlet and outlet temperatures

*NPT - Number of pressure tubes

[†] T_{sm} - Maximum surface temperature of fuel

- e) Number of loops
- f) Turbine plant costs
- g) Secondary system unit cost data.

Major outputs include:

- a) Primary loop capital cost
- b) Total plant capital cost
- c) Power generating cost in the primary loop
- d) Total power generating cost
- e) Temperatures, flows, and heat loads throughout the system
- f) Design of the heat exchange equipment at optimum coolant velocity
- g) Loop piping size
- h) Number of boilers and superheaters required per loop
- i) Main coolant pump size for either a motor or turbine driven unit
- j) Coolant damage rate expressed as organic makeup cost
- k) The optimum coolant flow split between superheater and reheater for a given heat duty.

Estimated output costs can be plotted on a cathode ray tube display against coolant temperature difference across the reactor.

D. PROPOSED BASIS FOR FUEL COST CALCULATIONS

The procedure for determining the fuel cycle cost has been formulated for use in optimizing the fuel system concepts which may be employed in HWOCR dual purpose desalting plants with thermal ratings equivalent to a net electrical output of 1000 Mw. Minimum available energy costs will be determined for each of the following fuel materials: (a) uranium metal, (b) uranium carbide, and (c) uranium dioxide. Costs are based on 1985 desalination ground rules for public financing using "equilibrium" fuel cycles to obtain approximate fuel costs. A more detailed cost investigation will follow, comparing the selected concept for each fuel material based on a plant lifetime of 30 years. In the latter calculations a value is assigned to fuel discharged from the reactor at the end of plant life, including linearly prorated credits for fabrication and debits for processing of partially irradiated fuel. In accordance with the ground rules, interest payments and collection of revenue are to be made every six months.

All systems to be evaluated are based on a plant operating factor of 0.90 with continuous on-line refueling. A summary of economic data for 1975 and 1985 ground rules is shown in Table 16000-2.

TABLE 16000-2
GROUND RULES AND COST BASES

Parameters	1975 Rules	1985 Rules
Plant Net Electrical output (Mw)	1000	1000
Plant operating factor	0.90	0.90
Natural uranium as U_3O_8 (\$/lb)	6.00	8.00
Conversion of U_3O_8 to UF_6 (\$/kgm U)	2.70	1.35
Diffusion plant separative work cost (\$/kgm U)	30.00	30.00
Diffusion plant tails enrichment (%)	0.28	0.26
Value of fissile Pu (\$/gm)	9.04	9.76
Fuel material loss during fabrication (%)	0.20	0.20
Fuel material loss during reprocessing (%)	1.0	0.25
Pre-irradiation storage at reactor site (days)	60	60
Shipping cask utilization factor	0.80	0.80
Shipping time one-way (days)	8	8
Convert fuel material and fabricate (days)	*	*
Delay storage at reactor site (days)	120	120
Reprocessing holdup time (days)	*	*

*To be determined by ORNL

The fuel cycle cost is resolved into the following components: (1) burnup and losses, (2) fabrication, (3) processing, (4) shipping, (5) fixed charges on fissile and fertile inventories, and (6) interest charges on operating capital invested in fabrication, processing, and shipping. ORNL will supply fuel fabrication and processing costs from fuel element designs which will be submitted from AI-CE. The recipes associated with the calculation of the fuel cycle costs have been formulated and forwarded to ORNL for review.

Figure 16000-2 shows the model which has been programmed into the fuel cycle costing procedure.

Figure 16000-2. Cost or Value as a Function of Time

Figure 16000-3. Task B Dual-Purpose-Plant Study Gross Turbine Cycle Efficiency and Steam Generator Thermal Duty

E. TURBINE PLANT EFFICIENCY PARAMETRIC STUDIES

In order to conduct the fuel system optimizations, the gross turbine cycle effectiveness must be determined as a function of throttle steam pressure and temperature. This is located as step No.4 in the logic sequence presented in Figure 16000-1. Figure 16000-3 is the results obtained for use in this task.

F. PRIMARY HEAT TRANSFER LOOP PARAMETRIC STUDIES

Primary loop efficiencies are defined to be the ratio of thermal power at the steam generator outlet to thermal fission power. The temperature of this efficiency is necessary to make a summation of all efficiencies for determining cost of heat at the steam generator outlet. This determination is located as step No.4 in the logic sequence described in Figure 16000-1. In addition, costs of the primary loop which are reported in the form of cents per million Btu as a function of reactor $\Delta T(^{\circ}\text{F})$ and reactor ΔP are determined as a part of determining overall plant costs. This determination is part of step No.5 in the logic sequence described in Figure 16000-1. Figure 16000-4 shows typical curves of the results obtained for a reactor outlet temperature of 750°F making steam at 900 psig/ 725°F with no reheat. Curves have been obtained for the cases described in Table 16000-3.

Figure 16000-4. Primary Loop Costs and Efficiency as Functions of Reactor $\Delta T(^{\circ}\text{F})$ and $\Delta P(\text{psi})$

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TABLE 16000-3
CASES COMPLETED FOR COOLANT SYSTEM AND
TURBINE PLANT STUDIES

Case Number	Reactor Outlet Temperature (°F)	Steam Temperature (°F)	Steam Pressure at TT (psig)	Reactor ΔP (psi)	Reactor ΔT (°F)
1 A	750	725	900	150	100 to 200
1 B	750	725	900	250	100 to 200
2 A	750	725	600	150	100 to 200
2 B	750	725	600	250	100 to 200
3 A	750	725	400	150	100 to 200
3 B	750	725	400	250	100 to 200
4 A	650	625	900	150	60 to 100
4 B	650	625	900	250	60 to 100
5 A	650	625	600	150	60 to 140
5 B	650	625	600	250	60 to 140
6 A	650	625	400	150	80 to 180
6 B	650	625	400	250	80 to 180

III. NEXT REPORT PERIOD ACTIVITIES

During the next period, in the scope of Phase B, optimum parameters and design characteristics for UC, UO_2 , and U-metal fuel systems will be determined. The economic advantages based on minimum cost of available heat to a condensing turbine will be determined for comparison of the fuel systems under their respective operating conditions. The UC fuel system will be compared economically to the other fuel systems at their operating conditions. Phase C will conduct economic studies of coupling arrangements for a HWOCR plant under various operating parameters. The results of Phase B and Phase C will be combined to define the operating parameters of an optimum HWOCR dual purpose plant. Reports containing the results of Phases B and C will be issued.

HWO CR QUARTERLY REPORT

TITLE: Dual Stream Reactor Concept

WBS NO: 16000A*

REPORTING PERIOD: April through June, 1966

I. PROJECT OBJECTIVES

The dual stream reactor concept involves two physically separated coolant streams operating in parallel, both drawing heat from a single reactor and utilizing the energy to produce steam at two different temperatures. This concept was developed for application in a dual purpose plant—the high temperature stream to be used for electric power production as well as to produce process steam for desalination; the low temperature stream to be used for desalination only. The present study is intended to determine the feasibility of a dual stream HWO CR and to determine its major features.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

The heat conduction problem for some dual stream fuel elements has been solved and dual stream fuel elements found feasible within thermal and hydraulic limitations. Tools have been developed and put into use to survey a large range of multi-tube fuel elements. These techniques have led directly to dual stream fuel element designs.

Some experience has been obtained with treatment of thermal neutrons similar to methods coming into use lately in Europe.

A. FUEL ELEMENT DESIGN LIMITATIONS

Thermal and hydraulic study of fuel elements, as described in the last progress report, continues to be pursued. In addition to the points specified in

*Work performed in part at Rensselaer Polytechnic Institute under Subcontract No. CE-HW-2.

the last progress report, the following are the major limitations on the thermal and hydraulic design:

- 1) Maximum length of fuel equals 12 ft.
- 2) Spacing of coolant tubes of approximately 0.15 in. with occasional spaces near perimeters of 0.10 in.
- 3) Minimum ID of coolant tubes equals 1/4 in.
- 4) Pressure drop through straight sections of coolant tubes ≈ 200 psi.
- 5) Outer clad must withstand external pressure of 100 psia, internal ≈ 300 psi.
- 6) Same criterion as 4 for inner coolant tubes.
- 7) Maximum fuel temperature of $\leq 5,000^\circ\text{F}$.
- 8) Maximum cladding-organic surface temperature $\leq 850^\circ\text{F}$ for steady state.
- 9) Chugging limitations on D_2O .
- 10) Performance requirements based on reference HWO CR (Presumes optimization of primary and secondary components)— $T_{\text{in}} = 550^\circ\text{F}$, $T_{\text{out}} = 750^\circ\text{F}$, $\Delta T = 200^\circ\text{F}$.
- 11) Considerations of burnout at the fuel element-heavy water interface and at the fuel element-organic interface is, of course, a limit on design. However, since the other limitations appear to be more restrictive than the burnout limitations, burnout is of no concern in the design of these fuel elements.

B. DESIGN PROBLEMS AND METHODS

The thermal hydraulic design involves the setting of the following variables:

- 1) number of organic coolant tubes
- 2) organic tube diameter and spacing
- 3) cladding thicknesses
- 4) diameter of fuel element
- 5) coolant velocity
- 6) temperatures at all important positions
- 7) heat transfer to organic
- 8) D_2O and specific power.

Three computer codes for solving the heat conduction problem are the major tools developed to handle these aspects of the dual stream fuel element.

The axial heat transfer problem is treated with a code, ORTUTH, which has been written and made operational. Given the heat generation in the fuel as a function of axial position, the inlet temperature and velocity, and the necessary mechanical and thermal parameters, the code computes the axial variation of the parameters (in one or two passes). The code has been used mainly to determine pressure drop, ΔT across the core, and temperature.

A code developed at Los Alamos, ROB, has been made operational here. This code solves the heat conduction problem in a cross section of a cylinder pierced by a set of holes on a fixed ring and one central hole. This program has been used to provide a complete analysis of the transverse heat conduction problem in "one-ring" (plus central hole) multi-tube elements and to analyze the outer ring of multi-ring elements.

The heat conduction problem in a plane perpendicular to the axis of a regular infinite lattice of cylindrical tubes has been coded and made operational. This code, LATHOL, has been used in analyzing multi-ring elements in regions where edge effects are small.

C. RESULTS OF ANALYSIS

1. Design of Coolant Tube

ORTUTH, the program for axial heat transfer solutions, was used to survey a number of coolant tube sizes and core lengths. The purpose of the analysis was to find combinations of coolant tube diameters, core lengths, and heat fluxes to the organic which would together provide a temperature rise across the core (ΔT) of 200°F or more, subject to the many design limitations described above. The following general conclusions arise from the analysis:

- a) The surface temperature limitation is the one which applies in all cases, rather than the maximum fuel temperature limitation.
- b) The optimum tube size within the limitations of ΔP , T_{surf} , and ΔT occurs below the minimum 1/4-in. ID tube size which is reasonable for use in transporting organic coolant through a heat producing fuel element. This sets the fuel element design at this minimum tube size diameter.
- c) With the tube size selected, locating the tubes with a minimum separation, in order to achieve a high specific power, will not result in excessive organic volume fractions from a nuclear point of view.
- d) With the tube size fixed at its minimum feasible value, it was found that the tube length should be as large as possible, to obtain a high ΔT . This makes use of the so-called "roof-topping" effect using surface temperature limits.
- e) The tube design was such as to provide a core ΔT near 200°F in the hottest tube. This presumes orificing of the colder fuel elements to provide a core average ΔT of 200°F.

On the basis of the ORTUTH analysis, the tube design described in Table 16000A-1 was developed. It will be noted that this design makes no provision for power peaking between the first and second organic pass through

TABLE 16000A-1
REFERENCE ORGANIC COOLANT TUBE
DESIGN PARAMETERS

	Average Tube	Hottest Tube
Tube ID (in.)	0.250	0.250
Tube OD (in.)	0.270	0.270
Tube length (ft)	12	12
Tube pitch (in.)	0.420	0.420
Edge to edge spacing (in.)	0.150	0.150
Average specific power (kw/kgU)	18.7	24.9
Maximum specific power (kw/kgU)	26.2	34.9
Average heat flux (Btu/hr-ft ²)	150,000	200,000
Maximum heat flux (Btu/hr-ft ²)	210,000	280,000
Coolant inlet velocity (ft/sec)	28.2	37.5
Coolant inlet temperature (°F)	550	550
Coolant outlet temperature (°F)	756	756
Core ΔT (°F)	206	206
Tube pressure drop (psi)	100	167
Max. clad surface temperature (°F)	835	851
Max. fuel temperature (°F)	1420	1634
Power peaking factors		
Core radial	1.33	1.33
Core axial	1.40	1.40
Among passes	1.0	1.0

the tubes. In addition, there is some margin on ΔT in the sense that the organic flow rate can be increased slightly to lower the surface temperature and still provide an acceptable ΔT value. These two effects provide some conservatism in the design.

2. Single-Ring Fuel Element Design

With both the design of the coolant tube and its spacing within the fuel element fixed, the next step was to consider arrays of these tubes within a multi-tube fuel element. Here the ROB code was especially useful in providing

temperatures and heat fluxes in a multi-tube fuel element with a single ring of coolant tubes. Since the spacing of the tubes within the fuel element was essentially determined earlier, the purpose of the present analysis is to determine the location of the outer surface and the amount of heat transferred to the D_2O . The design limitation on this part of the analysis is that this outer surface be so located that the heat flow to the outer ring of tubes not be excessive and, in fact, not be much more than used in the single-tube analysis.

If the regular triangular array is maintained, the one-ring fuel element consists of a center hole surrounded by six others in a hexagonal pattern. For this arrangement, it was found that there was considerable azimuthal variation in temperature and heat flux around the outer tubes. This so-called azimuthal peaking was so severe that the outer surface had to be within 0.09 in. of the outermost tube, and the average heat flux to the outer tube was reduced by 20% relative to the single tube analysis. It was found that a much more uniform distribution of temperature around the outer holes was achieved by adding one tube to the ring, making a total of eight tubes in the fuel element. This also facilitates dividing the organic flow within the fuel element to two equal passes. Figure 16000A-1 shows the final design of the eight-ring fuel element, with the dimensions given applying to the fuel region. Table 16000A-2, intended as a supplement, summarizes the thermal performance characteristics of this fuel element as a whole.

TABLE 16000A-2
PERFORMANCE OF EIGHT-TUBE FUEL ELEMENT

	Average Element	Hottest Element
Total Power (kw)	497	639
Fraction of Power - to Organic	0.55	0.57
- to D_2O	0.45	0.43
Heat Flux to D_2O (Btu/hr-ft ²) - Maximum	203,000	265,000
- Average	148,000	193,000
Heat Flux to Organic (Btu/hr-ft ²) - Maximum	207,000	280,000
- Average	144,000	196,000

3. Determination of Cladding Wall Thicknesses

In the dual stream design, the D_2O is to be operated at a pressure of 100 psi. Maximum pressure of the organic flowing through the fuel element will depend on the detailed pressure drops in the system, but will be somewhere between 300 and 400 psia. Therefore, the design criteria selected for the cladding, both on the outside of the fuel element and for the coolant tube, were that the material be self-standing against an external pressure of 100 psi and an internal pressure of about 400 psi; the former criterion will be limiting. The analysis of thermal stresses within the fuel element during operation was considered outside of the scope of the present design, but it is expected that by appropriate design of end fittings, allowance can be made for thermal expansion without creating excessive stresses. On the basis of the extreme pressure criterion, and using

an elastic pressure collapse analysis, it was determined that the thickness-to-radius ratio of the cladding must be no less than 4%. In addition, it was felt that the clad thickness for the organic coolant tubes should be no less than 10 mils, which is more than 4% of the radius.

FUEL OD = 1.51 in.
 COOLANT TUBES: 0.25 in. ID, 0.01 in. WALL
 OUTER CLADDING: 1.51 in. ID, 0.03 in. WALL

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Figure 16000A-1. Eight-Tube
 Dual-Stream Fuel Element

III. NEXT REPORT PERIOD ACTIVITIES

The analytical tools for the thermal analysis of dual stream elements have been developed and the analysis carried to the point where a typical dual stream element has been designed. Design of multi-ring fuel elements is in progress. In the next three months, design of all fuel elements to be considered will be completed, the hydraulics of the D_2O will be more fully studied, and more definitive nuclear analyses will be begun.

HWO CR QUARTERLY REPORT

TITLE: Reactor Physics

WBS NO: 13200

REPORTING PERIOD: April through June, 1966

I. PROJECT OBJECTIVES

The objective of the reactor physics task is to provide nuclear design methods incorporating digital computer codes, nuclear cross section data and analytical techniques which will predict the nuclear characteristics of HWO CR plants with sufficient accuracy. The uncertainties in such nuclear characteristics as fuel enrichment, life-time power distribution, control requirements, plutonium production, and reactivity coefficients must be small enough so as not to limit the achievement of the lowest possible energy cost.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. ANALYSIS OF EXPERIMENTS

In the preceding quarterly progress report,⁽¹⁾ the effect of thermal scattering kernels on HWO CR lattice calculations employing the THERMOS program was discussed. Based on this analysis, a selection of the more appropriate thermal scattering kernels was made and these kernels were employed in the analysis of several 19-rod UO_2 cluster experiments with the HAMMER code. The results of this analysis indicated that the HAMMER-derived eigenvalues, for an annular representation of the lattice cells employing D_2O and air coolants, showed a systematic trend with increasing lattice pitch.

Resonance self-shielding, as treated by the Chernick-Nordheim theory, for the resolved and unresolved resonance energy regions are calculated by evaluating the collision probabilities in the fuel. The analyses have now been extended to include the ZED-2 seven-rod uranium oxide⁽²⁾ and the seven-rod uranium-carbide experiments⁽³⁾ using the same representation of the lattice cell. In this

representation, the mean diameter of each fuel ring is set equal to that of the corresponding pin circle with cladding, void, and coolant homogenized over the remaining volume inside the pressure tube. For several of the uranium-oxide experiments, comparisons have been made between calculated and experimental values of neutron density ratios, fast fission ratios, and initial conversion ratios. The thermal scattering kernels employed were the Honeck-Nelkin kernels for H_2O ⁽⁴⁾ and D_2O ,⁽⁵⁾ Parks kernel for carbon⁽⁶⁾ and both the Brown-St. John kernel⁽⁷⁾ and Ardente-Nelkin kernel⁽⁸⁾ for hydrogen in the organic coolant. The lattices analyzed employed UO_2 and UC fuel of natural enrichment and D_2O moderator at room temperature.

Figure 13200-1 shows the material bucklings predicted by HAMMER for the seven- and nineteen-rod UO_2 cluster experiments. For D_2O and air coolants, this HAMMER model underestimates the bucklings and eigenvalues at tight macroscopic lattice pitch and overestimates them slightly at larger pitches. Preliminary analysis indicates that the systematic trend in eigenvalues with increasing macroscopic lattice pitch is also evidenced for the seven-rod uranium-carbide cluster experiments employing D_2O and air coolant; however, the eigenvalues for the seven-rod UC clusters are approximately 1% lower than those in the corresponding seven-rod UO_2 cluster experiments at equivalent unit cell areas.

The eigenvalues for the 19-rod and 7-rod UO_2 clusters with HB-40 coolant do not exhibit any systematic trend with lattice pitch, but are about 0.8% low. Preliminary analysis shows that the THERMOS-HAMMER calculations by the cylindricized cluster model overestimate the neutron densities in coolant-cladding, pressure-calandria tube and moderator as compared with the corresponding experimentally deduced values. For the 19-rod UO_2 cluster experiments with D_2O and air coolant, the calculated ratios of neutron density in coolant-cladding regions to fuel regions are higher than the experimental values, but are still within the experimental errors ($\pm 2\%$). The calculated ratios of moderator-to-fuel neutron densities are 3% too large. This good agreement does not seem to be maintained with HB-40 coolant in the 19-rod cluster or in the 7-rod UO_2 and UC clusters with D_2O , air, and organic coolant.

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Figure 13200-1. Experimental and Calculated Bucklings for 7-Rod and 19-Rod UO_2 Clusters as a Function of Lattice Pitch

Figure 13200-2 shows the eigenvalues for the 7-rod and 19-rod UO_2 cluster experiments as a function of lattice pitch. The systematic trend in the eigenvalues with lattice pitch is evident. The points indicated are calculated values.

Table 13200-1 shows a comparison of eigenvalues obtained by employing the transport corrected Ardente-Nelkin and the isotropic Brown-St. John thermal scattering kernels for hydrogen in the organic coolant. This comparison was

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Figure 13200-2. Eigenvalue vs Lattice Pitch

TABLE 13200-1
COMPARISON OF EIGENVALUES EMPLOYING TRANSPORT
CORRECTED ARDENTE-NELKIN KERNEL AND ISOTROPIC
BROWN-ST. JOHN KERNEL FOR HYDROGEN IN THE
ORGANIC COOLANT

Coolant	Lattice Pitch (cm)	B^2 (m^{-2}) EXPERIMENTAL	B^2 (m^{-2}) BSJ* HAMMER	B^2 (m^{-2}) A-N† HAMMER	k_{∞} BSJ HAMMER	k_{∞} A-N HAMMER	k_{eff} BSJ HAMMER	k_{eff} A-N HAMMER
HB-40	Hexagonal	19-ROD UO_2						
	18	-0.365 ± 0.062	-0.869	-0.920	0.9856	0.9842	0.9916	0.9904
	21	1.214 ± 0.035	0.700	0.646	1.0144	1.0138	0.9896	0.9881
	24	1.568 ± 0.019	1.229	1.200	1.0312	1.0311	0.9916	0.9907
	28	1.528 ± 0.012	1.258	1.261	1.0411	1.0417	0.9915	0.9915

*Brown-St. John kernel (isotropic) for hydrogen in organic coolant

†Ardente-Nelkin kernel (transport corrected) for hydrogen in organic coolant

made for 19-rod uranium-oxide clusters. As shown, the differential effect of the two kernels on eigenvalues is lattice pitch dependent, but is, in general, very small. Note that the eigenvalues with the Ardente-Nelkin kernel are slightly lower than with the Brown-St. John kernel.

Table 13200-2 shows a comparison between calculated and reported experimental values of conversion ratios (CR) and fast fission ratios (δ^{28}). There is good agreement between the calculated and the reported experimental conversion ratios. However, the reported experimental conversion ratios depend on cross section values in the data reduction, and these cross sections are not necessarily consistent with those employed in HAMMER. The calculated values for the fast fission ratio, δ^{28} , are also in good agreement with the experimental values.

B. URANIUM-CARBIDE EXPERIMENTS AT SAVANNAH RIVER LABORATORY

A program of physics experiments employing 37-rod uranium-carbide clusters of 1.0 wt % enrichment were proposed and authorized. The experiments will be performed at the Savannah River Laboratory. The UC clusters will contain 31 rods having a fuel pin diameter of 0.471 in. and six rods on the periphery having a fuel pin diameter of 0.272 in. The cladding will be aluminum. Sufficient fuel has been authorized to fabricate 31 clusters four

TABLE 13200-2
CALCULATED AND REPORTED EXPERIMENTAL INITIAL CONVERSION
RATIOS, CR, AND FAST FISSION RATIOS, δ^{28}

Coolant	7-ROD UO_2 (Fuel Pin Diameter = 2.40 cm; Density = 10.20 g/cm ³)					19-ROD UO_2 (Fuel Pin Diameter = 1.421 cm; Density = 10.45 g/cm ³)				
	Hex. Lattice Pitch (cm)	δ^{28} Experimental Values	δ^{28} Calculated with HAMMER	CR (Reported Experimental Values)	CR Calculated with HAMMER	Hex. Lattice Pitch (cm)	δ^{28} Experimental Values	δ^{28} Calculated with HAMMER	CR (Reported Experimental Values)	CR Calculated with HAMMER
D_2O	18.0	0.058±0.002	0.058	0.916±0.005	0.926	18.0	0.056±0.001	0.055	0.965±0.005	0.986
	22.0	0.056±0.002	0.055	0.810±0.004	0.817	21.0	0.054±0.001	0.052	0.866±0.004	0.868
	24.56	0.053±0.002	0.054	-	-	24.0	0.053±0.001	0.050	0.804±0.004	0.800
	28.0	0.056±0.002	0.053	0.730±0.004	0.743	28.0	0.048±0.001	0.049	0.745±0.004	0.749
	36.0	0.052±0.002	0.053	0.699±0.003	0.702	36.0	0.048±0.001	0.048	0.705±0.004	0.699
Air	19.0	0.058±0.002	0.060	0.873±0.004	0.886	18.0	0.066±0.001	0.062	0.995±0.005	0.979
	22.0	0.058±0.002	0.057	0.794±0.004	0.807	21.0	0.062±0.001	0.057	0.844±0.004	0.854
	24.56	0.056±0.002	0.056	-	-	24.0	0.057±0.001	0.054	0.788±0.004	0.786
						28.0	0.055±0.001	0.053	0.732±0.004	0.734
						36.0	0.053±0.001	0.052	0.684±0.003	0.684
HB-40	19.0	0.059±0.002	0.056	0.824±0.004	0.832	18.0	0.058±0.001	0.055	0.916±0.005	0.902
	22.0	0.054±0.002	0.054	0.784±0.004	0.786	21.0	0.054±0.001	0.051	0.827±0.004	0.829
	24.56	0.053±0.002	0.053	-	-	24.0	0.052±0.001	0.049	0.784±0.004	0.783
	28.0	-	-	0.737±0.004	0.742	28.0	0.049±0.001	0.048	0.758±0.004	0.746

$$\delta^{28} = \frac{\int_0^\infty \sum_f^{28} (E) \phi(E) dE}{\int_0^\infty \sum_f^{25} (E) \phi(E) dE}; \quad CR = \frac{\int_0^\infty \sum_c^{28} (E) \phi(E) dE}{\int_0^\infty \sum_a^{25} (E) \phi(E) dE} \quad \text{Where } \sum_a^{25} (E) = \sum_c^{25} (E) + \sum_f^{25} (E)$$

feet long. The lattices will be moderated by D_2O ; D_2O , air, and the organics HB-40, Dowtherm-A and Terphenyl will alternately occupy the coolant regions. The measurements will include buckling determinations both at room temperature ($20^\circ C$) and elevated temperature ($200+^\circ C$), activation foil ratios such as Lu^{176}/Cu and $Pu^{239}_{fiss}/U^{235}_{fiss}$ and the microscopic parameters ρ^{28} , δ^{25} , the fast fission ratio δ^{28} , and the modified conversion ratio CR.* The experiments will be performed in the Process Development Pile (PDP), a 16 foot-dia, zero-power, D_2O reactor and the Subcritical Experiment (SE), a 5-foot-dia, D_2O exponential. A comparison is made of the areas of fuel, coolant, and structure for the HWOCR lattice⁽⁹⁾ and experiments:

<u>Description</u>	<u>UC Experiments at SRL</u>	<u>HWOCR Lattice (AI-CE-Memo-25)</u>
$\frac{\text{Volume organic}}{\text{Volume cladding}}$	0.843	0.843
$\frac{\text{Volume cladding}}{\text{Volume fuel}}$	0.313	0.345
$\frac{\text{Volume cladding}}{\text{Volume organic}}$	0.371	0.410

C. THERMAL SPECTRUM

In the previous quarterly progress report, it was pointed out that the ratio of the bound to free-atom scattering cross section for hydrogen, based on a Brown-St. John representation of the scattering kernel for hydrogen in Santowax, showed deviations from the experimental data for benzene at low energies, whereas much better agreement was obtained with the data based on Nelkin-type kernels for hydrogen bound in polyethylene and benzene. Figure 13200-3 shows a comparison between the calculated and experimentally derived data.

D. CLUSTER REPRESENTATION FOR RESONANCE ENERGIES

The proper representation of the rod cluster is especially important for resonance energies. In particular, the fuel and coolant temperature coefficients of reactivity can be quite sensitive to the model employed.

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Figure 13200-3. Comparison of $\sigma_s/\sigma_{\text{free}}$ vs Energy for Hydrogen Kernels in Organic Coolant

1. General

A few of the possible approaches for calculating the escape probability for a rod cluster are: (1) Monte Carlo calculations over the appropriate energy region with an explicit rod cluster geometry, as with REPLICA⁽¹⁰⁾ which is, however, presently limited to cluster configurations of 19 rods; (2) same as in (1) but with an annular representation of the rod cluster, again by REPLICA using the annular geometry option; and (3) by defining an equivalent rod having the same effective surface, S_{eff} , as the rod cluster and evaluating the resonance integral by a program such as ZUT.⁽¹¹⁾ The latter approach has been examined by two methods; one based on the so-called rubber-band model and the second is based on finding the radius of an isolated rod which has the same first collision probability as the rod cluster or annular representation thereof. First collision probabilities were evaluated by the TRAM⁽¹²⁾ program for a single energy group and a range of fuel cross sections using only hydrogen and deuterium scattering cross sections in the coolant and moderator. For several cluster geometries and coolant conditions, the effective rod radius based on the TRAM calculations is 0.3% to 5.0% higher for the annular representation than the explicit representation of the cluster.

Table 13200-3 summarizes the resonance escape probabilities derived for the two geometric representations of several rod clusters, using: (1) the detailed REPLICA calculations; (2) the TRAM-derived effective radius, combined with resonance integrals evaluated by ZUT, with cross sections matched to those employed in REPLICA; and (3) the survey design rubber band method which uses either all scatterers or H and D scattering only.

The spread in resonance escape probability for all these methods is in general less than one percent. The choice of scatterer to use in calculating collision probabilities introduces an uncertainty of half a percent for the organic cooled cases. Void coefficients are sensitive to the model, but part of the differences can arise from the statistical uncertainty ($\sim \pm 0.15\%$) in the Monte Carlo REPLICA resonance escape probabilities.

2. Ispra Experiment Analysis

An analysis was made of Ispra⁽¹³⁾ resonance experiments on 7-rod UC clusters and 19-rod U-metal clusters having air, diphenyl, and H_2O coolants. The

TABLE 13200-3

COMPARISON OF RESONANCE ESCAPE PROBABILITIES

	Case						
	A	B	C	D	E	F	G
Number of Rods	19	19	19	19	19	37	37
Rod Radius (cm)	0.81	0.81	0.81	0.93	0.81	(0.61 & 0.36)*	
Micro Pitch (cm, Hex)	1.79	1.79	1.79	2.06	2.13	1.63	1.63
Macro Pitch (cm, Sq)	26.2	26.2	23.4	30.2	26.6	28.0	28.0
Coolant	Organic	Void	Organic	Organic	Organic	Organic	Void
<u>Escape Probability</u>							
REPLICA - Explicit	0.8980	0.8953	0.8704	0.9013	0.8953	-	-
REPLICA - Annular	0.8962	0.8970	0.8634	0.9015	0.8918	0.8991	0.9082
TRAM - Explicit	0.8899	0.8896	0.8637	0.8933	0.8928	0.8946	0.9005
TRAM - Annular	0.8918	0.8918	0.8661	0.8936	0.8929	0.8948	0.9010
Rubber Band							
All Scattering	0.8888	0.8886	0.8631	0.8919	0.8920	0.8948	0.8992
H and D Scattering	0.8935	0.8906	0.8694	0.8969	0.8957	0.8993	0.9012

*31 rods of 0.61 cm radius and six rods, at perimeter, of 0.37 cm radius.

effective rod radii were obtained from both TRAM explicit cluster collision probabilities and from the CEND survey rubber band method. Linear $\sqrt{S/M}$ resonance integrals for isolated rods were used where the metal values were derived by E. Hellstrand and G. Lundgren⁽¹⁴⁾ and the values for UC were obtained from measurements in reference 12. It can be seen in Table 13200-4 that the calculated effective resonance integrals are 0.3 to 0.9 barns low for UC and high by 0.3 to 0.6 barns for U-metal rods, which is within the quoted experimental uncertainty of $\pm 6\%$. The effect of choice of scatterers varies between 0.1 and 0.6 barns and is shown in the last two columns. These last calculations were made at Ispra using SHOCK which calculates a Dancoff coefficient for rod clusters.

TABLE 13200-4
ISPRA EXPERIMENTS

No. of Rods	Fuel	Coolant	Effective Resonance Integrals				
			TRAM-R _{eff}	Rubber Band	Expt.	Calculations	
						All* Scattering	H&D Scattering
7	UC*	Air	9.30	9.43	10.20 ± 0.6	9.79	9.69
7	UC	Diphenyl	10.72	10.98	11.10 ± 0.7	11.07	10.83
7	UC	H ₂ O	11.26	11.35	10.95 ± 0.7	11.36	11.27
19	U-Metal*	Air	8.91	9.04	8.35 ± 0.5	9.02	8.43
19	U-Metal	Diphenyl	10.58	10.72	10.30 ± 0.6	10.54	10.19
19	U-Metal	H ₂ O	11.32	10.96	10.90 ± 0.7	10.96	10.81

$$\left. \begin{array}{l} \text{using } RI_{(UC)} = 3.14 + 26.95 \sqrt{S/M} \\ RI_{(U-Metal)} = 2.95 + 25.8 \sqrt{S/M} \end{array} \right\} \text{ibid Ref. 4}$$

3. HAMMER Comparison with REPLICA

The 27 explicit resonances, $1/v$ cross section and unresolved resonances to 5530 ev used in REPLICA were placed in a HAMMER library.⁽¹⁵⁾ Preliminary results show that for the same annular representation and for the range of cases listed in Table 13200-3, the HAMMER U^{238} resonance escape probabilities range from 1.0% to 1.7% lower than those from REPLICA. This discrepancy is being investigated further.

E. Pu^{240} RESONANCE SELF-SHIELDING

The amount of Pu^{240} calculated at any stage of burnup is greatly dependent upon its cross section, which changes because of varying resonance self-shielding. Using 11 explicit resonances for Pu^{240} , ZUT calculations were made for effective uranium carbide rod radii ranging from 0.8 cm to 3.2 cm, with Pu^{240} -to- U^{238} ratios of from 1×10^{-5} to 5×10^{-3} at temperatures of 68°F, 800°F, 1100°F and 1400°F. These indicate that for an equivalent rod radius of 1.35 cm and a Pu^{240} -to- U^{238} ratio of 1.1×10^{-3} (typical of 10,000 Mwd per ton HWO CR fuel exposure), the geometrical self-shielding factor by itself would be about one-third. The U^{238} diluent was treated here by the integral method in ZUT. Using the asymptotic treatment of removing the U^{238} entirely yields significantly larger resonance integrals. The integral method was used to analyze Nichols⁽¹⁶⁾ Pu^{240} resonance measurements in plutonium-aluminum alloy rods and gave results well within the experimental error.

F. HAMMER MODIFICATIONS

A series of modifications or options to HAMMER have been initiated in order to make it a more useful design tool. The first phase, which is complete, edits effective four-group microscopic cross sections as suitable for input to the point burnup code, CRUMB.⁽¹⁷⁾ The next set of options will calculate region-wise microscopic cross sections suitable for explicit representation in diffusion codes, and will include the individual resonance contribution from each ring of rods. Planned modifications include both region-wise burnup and a version which will edit various parameters to compare with those obtained in critical or exponential experiments.

G. DEVELOPMENT OF MONTE CARLO CODE - SAFE

SAFE, the Monte Carlo code, originally developed for criticality hazards analysis, has sufficiently general geometry routines to perform cell calculations for thermal reactors. This ability has been demonstrated by carrying out a calculation for a HWOCR cell. The analysis of such calculations, however, calls for considerably more detailed information than the mere value of k_{∞} required for criticality hazards studies, or even the values of the four factors η , ϵ , p , and f . The extensive monitoring desired for cell calculations requires more storage capacity than was originally available in SAFE; on the other hand, there are features in the original SAFE geometry provisions that are more general than needed for cell calculations. A revised version of SAFE has been compiled from which excessive geometrical generality has been eliminated, thus leaving sufficient storage for considerable extension of the monitoring.

Adaption of the Monte Carlo code SAFE to the HWOCR cell problem has been completed. This was accomplished by inserting the capabilities listed below into a version of SAFE which had been especially adapted to the HWOCR cell problem.

- 1) Treatment of a single rod type was replaced by the ability to treat up to six fuel rod types which can differ in both composition and size. This treatment is limited to two dimensions with cell boundary conditions for a square macro pitch, and with limitations of 20 zones, including the pressure and calandria tube walls. This limitation in no way compromises the HWOCR cell calculation, and permits streamlining the code; it should result in reduced computing costs.

2) In order to calculate Σ_f from $\bar{v} \Sigma_f$, $\bar{v}$ will be input to the code as a linear function of energy.

3) The number of isotopes per material was increased to eight and the number of isotopes per given cell problem was increased to fifteen.

4) Monitoring of the following quantities was programmed: the power per rod type; the fission and capture for each of six isotopes in four energy bands in the fuel; the capture and scattering events for other cell zones in each of four groups; four-group effective D_2O and non- D_2O region cross sections [$\bar{v} \Sigma_f$, Σ_f , Σ_c , Σ_s , $(1 - \bar{\mu}) \Sigma_s$, and Σ transfer]; four thermal-spectrum velocity moments for each cell zone; and the energy deposition in carbon and hydrogen.

These modifications and additions have been successfully compiled on the IBM 7094. A test run of the same calculation previously run using R. B. U. is now in preparation and will subsequently be analyzed to assure correctness of all phases of the revisions and monitoring incorporated into SAFE.

H. THERMAL SPECTRUM ANALYSES*

A method for understanding and analyzing thermal neutron spectral effects has been developed. The method uses escape probabilities and describes scattering kernels by means of an expansion in a series of a Maxwellian and a remainder sensitive enough to show the effects of model-dependent scattering kernels. This method yields equations for the thermal flux which are a natural extension into the thermal range of the NR1A resonance treatment of Chernick and Vernon.⁽¹⁸⁾

The method has been tested against some light water lattices because of the availability of certain test cases. In Table 13200-5, values are shown for slab lattices in which the new method (labeled M-T) is compared with exact results to demonstrate that the M-T method sensitively picks up kernel effects. The fuel slabs were composed of approximately 1 wt % U^{235} . In Table 13200-6, a uranium-plutonium system is examined to determine the adequacy of the method in predicting f and η in strongly non- $1/v$ lattices. The U^{235} was depleted to approximately 0.33 wt % and the Pu^{239} concentration was taken to be approximately 0.50 wt %.

The test cases show the lowest order method capable of predicting thermal f and η values to about 0.5%, even in difficult space-energy situations.

*Work performed in full at Rensselaer Polytechnic Institute under Sub Contract No. CE-HW-2.

TABLE 13200-5
THERMAL UTILIZATIONS IN URANIUM
SLAB LATTICES

VW/VM	M-T Free Gas	M-T Nelkin	Exact Free Gas	Exact Nelkin
1.5	0.9115	0.9070	0.9129	0.9099
1.25	0.9259	0.9220	0.9272	0.9235
1.0	0.9406	0.9374	0.9418	0.9407
0.50	0.9706	0.9686	0.9709	0.9710

TABLE 13200-6
COMPARISON OF F AND ETA FOR
U-PU SYSTEM

VW-VM	F		ETA235		ETA239	
	M-T	Exact	M-T	Exact	M-T	Exact
1.5	0.9262	0.9304	2.0781	2.0774	1.9124	1.9014
1.25	0.9400	0.9439	2.0774	2.0771	1.8994	1.8900
1.00	0.9538	0.9570	2.0767	2.0768	1.8994	1.8751
0.5	0.9805	0.9812	2.0747	2.0766	1.8124	1.8255

I. ADDITIONAL ACTIVITIES

The ANL version of THERMOS,⁽¹⁹⁾ which allows 50 energy groups compared to the present 30 and has library preparation routines entirely in FORTRAN was made operational on the CDC-3600, as was MACH-I,⁽²⁰⁾ a one-dimensional multigroup diffusion and perturbation code which will be particularly useful in obtaining neutron lifetimes and β_{eff} for kinetic calculations.

III. NEXT REPORT PERIOD ACTIVITIES

Effort will continue on the analysis of critical experiments. HAMMER will continue to be modified to provide a better design tool. The resonance

representation of clusters, especially in HAMMER, will be further evaluated. The self-shielding of Pu^{240} will be analyzed by a THERMOS code which would have a thermal cutoff of about 2 ev instead of the present 0.625 ev, and would allow Doppler broadening of the 1.054 ev resonance.

Final debugging of the HWOCR version of SAFE for the IBM 7094 will be completed. A larger version of SAFE, for the IBM system 360, will then be started. In this version, efforts will be made to improve the physics methods wherever possible.

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HWOCR QUARTERLY REPORT	
TITLE: Pressure Tube Development	WBS NO: 21000
REPORTING PERIOD: April through June, 1966	

I. PROJECT OBJECTIVES

The overall objective of this project is to develop a pressure tube assembly and associated components for use in the HWOCR. Included in the work is the testing of components and representative assemblies under simulated environmental conditions, to ensure and demonstrate that the goals of structural adequacy, reliability, maintainability and service lifetime can be met.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. SUMMARY

The following were the major areas of progress during this report period:

- 1) Manufacture of two SAP pressure tube assembly test sections for TF-20 was initiated.
- 2) The TS-10-2 test program was completed.
- 3) Short-time SAP-pressure-tube rupture testing was completed.
- 4) TF-50 was completed and put into operation.
- 5) End seal and mechanical coupling testing was initiated in TF-50.
- 6) TF-20 primary piping fabrication was completed.
- 7) The preliminary design study for the SAP pressure tube irradiation experiment in WR-1 was completed.

B. PRESSURE TUBE DEVELOPMENT

Two pressure tube assembly test section designs (TS 20-1A and TS 20-4) were released for manufacture. Figure 21000-1 depicts a generalized arrangement of these test assemblies, including shield plug and mock up fuel elements.

TS 20-1A will utilize a 4.000-in. ID, 0.100-in.-thick wall, SAP 895 pressure tube. The pressure tube will be connected to low carbon steel extensions by SAP 895 to stainless steel sandwich rolled transition joints. To contain the CO₂

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Figure 21000-1. General Arrangement, Pressure Tube Assembly Test Section

gas within the 0.30-in. gas gap, a 1/8-in.-thick asbestos gasket and a 6-in. OD Avica bellows will be provided at the top flange and lower shield tank, respectively. Both the upper flange and lower bearing sleeve are equipped with anti-torque devices to prevent excessive pipe loads from being transmitted to the thin walled pressure tube. All materials for the fabrication of this test section have been received. Fabrication is approximately 20% complete.

The design of TS 20-4 incorporates a 0.103-in.-thick, 4.000-in. ID, SAP 895 pressure tube attached to hot pressed transition joints by flash welding. The SAP 895 to stainless steel transition joints will be connected in turn to stainless steel-low carbon steel extensions. The reactor reference design stainless steel calandria and end shields are simulated. Removable guide bearings with a 0.003 to 0.005 in. coating of Cr_2O_3 for wear and galling resistance are incorporated in the assembly. For CO_2 containment, an asbestos gasket and bellows will be provided at the extremities of the upper and lower end shields, respectively. Manufacture is expected to start July 15, with an estimated 29 week delivery. Efforts to expedite this delivery are in progress.

Design of TS 20-5, which will have provisions for incorporation of reactor pressure tube internal components (e.g. fuel latch mechanisms and shield plugs), has been delayed pending final decisions on the refueling scheme and resulting reference pressure tube configuration. Final design work is expected to be completed in the next reporting period.

Mock-up fuel element design and preliminary shield plug design for use in TF-20 test sections have been completed, based on the current refueling scheme.

Instrumentation and equipment requirements for pressure tube assembly testing have been established. Appropriate temperature, pressure, displacement, force and vibration apparatus were ordered.

Seven Zircaloy-4 tubes of 4.000-in. ID and 0.075-in. wall thickness, originally planned for test section use, were received from Harvey Aluminum. This tubing was inspected and accepted, based on specification requirements, and is now in storage.

Non-destructive testing techniques for inspection of large diameter SAP tubing are being reviewed. Several surface flaws in the existing SAP tubing, which were not shown by original ultrasonic inspection performed by an outside

vendor, were detected visually. More refined non-destructive techniques for SAP inspection are being investigated. These include eddy-current, X-ray and more sensitive ultrasonic processes. Ultrasonic inspection of the second SAP tube for TF-10 testing (TF-10-4) was completed. A 0.004-in.-deep, 1/8-in.-long defect on the outside and inside diameters was used as a standard. Several minor defects estimated to be 0.006 to 0.008 in. in depth, were detected, but are not large enough to preclude use of this tubing.

Longitudinal and loop properties of the seven SAP pressure tubes were obtained. Table 21000-1 lists the tensile and yield strengths and the percent elongations for these tubes, designated P-6 through P-12. Table 21000-2 lists the loop properties.

TABLE 21000-1
LONGITUDINAL DATA ON SAP-895 USING FLAT STRIPS
(Sheet 1 of 2)

Sample from Tube	Tensile Strength (psi)	Yield Strength (psi)	Elongation (%)	Load Rate at Room Temperature in Air (in. /min)
P4	41,500	28,600	14	0.05
P4	41,100	28,600	13	0.05
P4	41,600	28,000	8	0.05
P4	42,100	28,900	11	0.05
P4	42,000	28,500	13	0.05
P4	42,200	29,400	11	0.05
P6	42,300	28,300	11	0.05
P6	42,400	29,800	13	0.05
P6	41,900	29,000	10	0.05
P7	41,100	27,800	12	0.05
P7	41,300	28,000	12	0.05
P7	41,300	28,000	12	0.05
P8	40,500	27,000	12	0.05
P8	40,400	27,000	9	0.05
P8	40,800	27,900	12	0.05
P8	41,500	28,700	14	0.05

TABLE 21000-1
LONGITUDINAL DATA ON SAP-895 USING FLAT STRIPS
(Sheet 2 of 2)

Sample from Tube	Tensile Strength (psi)	Yield Strength (psi)	Elongation (%)	Load Rate at Room Temperature in Air (in. /min)
P8	41,715	29,000	13	0.05
P8	41,300	29,200	11	0.05
P8	40,300	28,600	14	0.025
P8	41,600	29,000	13	0.1
P8	41,100	29,500	14	0.025
P9	42,200	29,600	13	0.025
P9	41,800	29,400	11	0.025
P9	42,000	28,900	12	0.1
P9	42,600	29,600	8	0.05
P9	42,100	28,100	8	0.05
P9	41,600	28,900	12	0.05
P9	41,700	28,900	12	0.05
P9	41,700	28,800	12	0.05
P10	41,900	28,300	12	0.05
P10	41,600	28,200	12	0.05
P10	41,700	28,000	12	0.05
P10	42,600	29,400	11	0.05
P10	42,700	28,800	7	0.05
P10	42,300	28,500	8	0.05
P10	42,600	29,000	8	0.05
P11	42,100	28,900	15	0.05
P11	41,900	28,500	14	0.05
P11	42,300	29,800	14	0.05
P11	42,000	28,900	12	0.1
P12	41,900	29,600	13	0.1
P12	41,300	29,200	7	0.025
P12	41,600	28,900	14	0.1

TABLE 21000-2
HOOP DATA ON SAP-895 USING HOOP RINGS

Sample from Tube	Group*	Tensile Strength (psi)	Elongation (%)	Load Rate at Room Temperature in Air (in. /min)
P4	B	43,400	4	0.05
P4	B	42,200	4	0.05
P4	B	42,800	4	0.05
P5	A	39,700	8	0.05
P5	A	39,700	8	0.05
P5	A	39,200	7	0.05
P5	A	38,650	7	0.05
P5	A	39,150	6.5	0.05
P5	A	39,850	7	0.05
P5	A	39,650	7	0.05
P8	A	39,650	8	0.05
P8	A	39,100	8	0.05
P8	A	40,000	10	0.05
P8	A	39,650	10	0.05
P8	A	39,000	8	0.05
P10	A	39,500	10	0.05
P11	A	39,350	8	0.05
P11	A	38,150	7	0.05
P11	A	39,800	9	0.05
P11	C	40,850	8	0.05
P11	C	40,300	6	0.05

*Group "A" - As received tube

Group "B" - Hydraulically stretched 5% initially

Group "C" - Explosively stretched 5% initially

C. SINGLE-CHANNEL TEST LOOP (TF-10) TEST PROGRAM

The TS-10-2 test program was completed. Figure 21000-2 shows the general layout of TS 10-2. The program was run in three phases as shown in Table 21000-3.

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Figure 21000-2. General Arrangement,
TS 10-2

TABLE 21000-3
OPERATION OF TS 10-2

Phase	Running Time (hr)	Fuel Preload (lb)	Fuel Bundle Configuration
I	20	0	None
Ia	500	0	Aligned
II	500	426	Aligned
III	500	426	2nd and 4th bundles mis- aligned 20°

The following parameters were investigated during this test program:

- 1) Rolled joint integrity
- 2) Pressure tube behavior
- 3) Pressure tube internal wear
- 4) Fuel bundle vibration
- 5) Fuel rod fin wear
- 6) Fuel bundle pressure drop

Final compilation of the data will be completed during the next report period; the results obtained through Phase II are presented in the following paragraphs:

1. Rolled Joint Integrity

Several periods of slight leakage, listed in Table 21000-4, were observed.

2. Pressure Tube Behavior

TS-10-2 exhibited the same trends as the S.S. TS-10-1A; that is,

- a) No pressure tube vibration was detected.
- b) Displacement of the pressure tube with respect to the calandria was dependent on temperature and piping loads only; flow had no effect.

3. Pressure Tube Internal Wear

At the conclusion of Phase Ia, several wear marks up to 0.010 in. in depth were found on the tube inside diameter at the elevation of the top end of the uppermost fuel bundle. A preload of 425 lb was added for Phase II, with the bundles still aligned. Subsequent inspection showed additional wear marks up to 0.005 in. deep, this time near the middle of the uppermost bundle. Vibration data support

TABLE 21000-4
TS-10-2 ROLLED JOINT LEAKAGE

Phase	Hour Leakage Began	Temp (°F)	Hour Leakage Ceased	Temp (°F)	Average Rate (gm/hr)
I	61	550	64	550	0.057×10^{-3}
I	170	550	216	650	0.009×10^{-3}
I	239	550	245	550	0.043×10^{-3}
I	249	550	264	600	0.013×10^{-3}
I	266	650	268	650	0.073×10^{-3}
I	284	550	286	550	0.140×10^{-3}
I	287	550	295	550	0.036×10^{-3}
I	296	550	310	550	0.020×10^{-3}
I	364	550	374	500	0.041×10^{-3}
I	382	600	396	600	0.032×10^{-3}

Note: No leakage was detected during Phases II and III. However, Phases II and III were restricted to 33% attended operation and data were taken only during that period.

both these areas as having had significant movement during the test runs. Therefore, the wear found on the pressure tube is attributed to the rubbing of the fuel rod fins on the pressure tube inside surface under vibration. It should be noted that these end conditions were not representative, in that the organic lines were close-coupled to the test section and no flow straighteners were used.

4. Fuel Bundle Vibration

Fuel bundle vibration was found to occur at a specific axial location within a given bundle, rather than along the entire string, and was 0.001 in. to 0.005 in. in amplitude and 2 to 6 cps in frequency. Vibration was significantly higher in amplitude at the center of the fuel bundles than at the end plates. Figure 21000-3 shows this trend at a temperature of 600°F for Phase II.

5. Fuel Rod Fin Wear

Fin wear up to 0.005 in. was measured at elevations and azimuths corresponding to the pressure tube wear marks. Wear of a smaller magnitude (no

21000-10

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Figure 21000-3. Fuel Bundle Vibration Amplitude vs Distance from Top of Fuel String

detectable depth) was found at fin locations corresponding to the rub marks on the pressure tube.

Figures 21000-4 and 21000-5 show the wear of a finned rod and a solid bearing rod, and a 7X blowup of the wear of the fin, respectively.

6. Fuel Bundle Pressure Drop

Pressure drops measured for the end plate aligned and misaligned cases closely duplicated the results obtained during the TS 10-1A test program (Figure 21000-3, page 21000-8, AI-CE-31).

D. FIVE-CHANNEL TEST LOOP (TF-20)

The major organic piping was installed, the shroud wall was constructed, all major equipment was placed, the electric contract work was started, the instruments were being mounted on the control panels, and the insulation contract was begun. A portion of the secondary system and all of the test section auxiliary system remain to be installed. Construction is expected to be completed in September.

E. COMPONENT TEST LOOP (TF-50)

Shakedown of the facility was completed, and three classes of components were tested:

- 1) Pressure tube end seals
- 2) Inner sleeve rolled joint
- 3) Mechanical couplings.

The rolled joint test program consists of the three phases listed in Table 21000-5. For information regarding pressure tube end seals, see section 81300. Test data on mechanical couplings are presented in Table 21000-6. As of the end of the report period 50-2-01 had undergone the first 100 cycles of Phase I and had exhibited no leakage.

F. PRESSURE TUBE ASSEMBLY IRRADIATIONS TESTS

An experiment is being planned which involves the irradiation of a SAP pressure tube assembly in the WR-1. This initial pressure tube test assembly will be made of SAP-895; and will be enclosed in a shroud tube made of stainless steel to protect the WR-1 calandria tube in the event a failure occurs in the

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Figure 21000-4. Wear on Finned Rod and Bearing Rod

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Figure 21000-5. Wear on Finned Rod, Top Bundle (7X)

TABLE 21000-5
INNER SLEEVE ROLLED JOINT TEST INFORMATION

Phase	Temperature (°F)	Mode	Pressure (psig)	Flow (gpm)	No. of Cycles	Temp. Ramp (°F/sec)
I	400-750-400	Automatic	300	3	200	30
II	400-750-400	Thermal	300	5	200	50
III	400-750-400	Cycle	300	7	200	90

pressure tube or transition joints. A core position in the inner ring of the reactor will be used for the pressure tube assembly irradiation.

A preliminary design of the irradiation test assembly was discussed with cognizant WR-1 operating personnel. Minor modifications in the design resulted from these talks. The proposed pressure tube irradiation assembly is shown in Figure 21000-6. This pressure tube assembly is similar in design to the existing WR-1 assemblies. The assembly is designed to operate in a 4.755 in. ID large calandria site, with minimal modification to the reactor mechanical components. Shown in phantom in the figure are the existing WR-1 reactor structure and mechanical components of the large calandria sites. The experiment will be designed to use either WR-1 driver fuel or HWOCR experiment fuel.

TABLE 21000-6
THERMAL CYCLING OF MECHANICAL COUPLINGS

Test Section	Description	Hr at 400°F	Hr at 750°F	No. Thermal Cycles 400°/750°/400°F
TS-50-1-03	Graylo coupling (coated seal ring)	80	11	9
TS-50-1-04	Graylo coupling (coated seal ring)	80	11	9
TS-50-1-05	Graylo coupling (coated seal ring)	80	11	9
TS-50-1-06	Conseal coupling	80	11	9

The overall pressure tube assembly is approximately 31 ft - 8 in. in length. It will consist of a 3.246 in. ID 8 ft - 6 in. long, (cold) SAP pressure tube. Two hot pressed SAP to 347 stainless steel transition joints will connect the in-core portion to the steel extension. The SAP pressure tube will be dimensioned for a 100,000 hr design life. This conservative approach will result in a tube wall near 0.150 in. in thickness.

The shroud tube will be of 347 stainless steel and will be at least 0.035 in. thick. It will be dimensioned to protect the WR-1 calandria tube in the event of a pressure tube failure. Differential expansion between the shroud and the pressure tube will be absorbed by a two-ply stainless steel bellows. The gas annulus enclosed by the shroud will be monitored for organic coolant leakage during operation of the experiment. Inlet and outlet purge lines will circulate CO₂ from a separate system for this purpose.

For the in-reactor guide bearing surfaces, a 0.003 in. to 0.005 in flame sprayed coating of Cr₂O₃ will be applied to the sliding surface to prevent galling or excessive friction. Both upper and lower extensions will have controlled guidance to center the in-core pressure tube under conditions of feeder pipe reaction loads and thermal expansion of the reactor system.

Detailed design of the irradiation assembly is in progress. Procurement of long-delivery-time items will begin early in the next quarter.

Preparations are being made for the out-of-core testing of the irradiation assembly, as required by WR-1 operations. Modifications required in TF-10 to accommodate the out-of-core tests are being designed. The test requires a minimum of 150 hr loop operation of the assembly at conditions approximating those anticipated in-reactor.

G. PRESSURE TUBE BURST TESTS

The pressure tube burst and stress rupture tests initiated in the last quarter were continued. A series of short-time burst tests, the results of which are presented in Table 21000-7, indicate that the tubing tested has a slightly lower transverse ultimate strength than reported for SAP 895 in the literature (see Figure 21000-7).

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Figure 21000-7. Uniaxial Transverse Ultimate-Strength SAP-895

TABLE 21000-7
SHORT-TIME BURST TEST RESULTS SAP-895 4 IN. ID TUBING

Specimen No.	Material	Temp (°F)	Stress Ratio	Hoop Stress
3-1	SAP-895	70	2/1	43,700
1-1	SAP-895	700	2/1	15,250
2-1	SAP-895	800	2/1	15,150
2-2	SAP-895	800	2/1	12,950
2-1 (a)	SAP-895	800	2/1	12,800
4-1	SAP-895	900	2/1	9,650
4-2	SAP-895	800	2/1	13,250

Upon initiation of the 100-hr test, it was discovered that the end closure device employed in the short-time tests imposed high localized strains which caused premature hairline cracks in the specimens at the ends. Specimens reduced in wall thickness at the center suffered the same type of failure. Several specimens failed in this manner before an effective seal redesign could be employed. Additional 4-in.-diameter SAP tubing is being obtained to replace the lost specimens.

The successful end seal redesign incorporates an inflatable metal seal on the OD of the specimen, with a match-machined bar stock center piece. This device was used on the last four specimens listed below and appears to have solved the problem.

Table 21000-8 contains a tabulation of the stress rupture specimens tested to date and the results of the tests.

Metallurgical examination and evaluation of the failed specimens is now under way.

Although no long-time tests have yet been completed, a preliminary comparison between these stress rupture data and the literature data, correlated using a Larson-Miller Constant of 40, indicates that these test points fall within the predicted range of transverse strength. There is an apparent notch sensitivity in the material; i.e., the presence of a notch decreases stress rupture life.

TABLE 21000-8
SAP TUBE SPECIMENS STRESS RUPTURE TESTS

Specimen No.	Material	Temp (°F)	Notched	Stress Ratio	Hoop Stress	Failure Time (hr)
1-2	SAP-895	800		2/1	8260	(2)
2-3	SAP-895	800		2/1	8260	(2)
3-2	SAP-895	800	(1)	2/1	8260	(1)
4-3	SAP-895	700		2/1	12525	2.5 ⁽³⁾
4-10	SAP-895	800		2/1	8230	2.7
4-11	SAP-895	800		2/1	7810	1.3 ⁽²⁾
3-3	SAP-895	800	(4)	2/1	7150	16+ ⁽⁵⁾
4-5	SAP-895	800		2/1	7000	166.8
2-4	SAP-895	800	(6)	2.1	7000	7.9
1-3	SAP-895	700		2/1	8200	(7)
2-5	SAP-895	700	(6)	2/1	8200	20

Notes:

1) Notch was unintentionally 14 mils deep in a 0.0638 in. wall. Stress level in thickness under notch was 10580 psi. Failure occurred in 30 sec.

2) Premature failure occurred due to end closure strains. Failure mode was hairline cracking at the ends.

3) Failure occurred at a lap type defect. Failure mode was a small leak.

4) Notch 6 mils deep in 0.101 in.-thick-wall. Stress based on thickness under notch.

5) Failure occurred by hairline cracking at ends. Pressure leaked off slowly.

6) Notched depth = 0.009 in. in 0.0975 in. wall. Failure occurred at notch.

7) Specimen still running at time of this writing.

However, with an inside diameter notch, and with stress level based on the remaining thickness, even the notched specimens have fallen above the predicted minimum transverse rupture strength.

III. NEXT REPORT PERIOD ACTIVITIES

As listed below, a variety of activities will be in work during the coming report period.

- 1) Fabrication of TS 20-1A and TS 20-4 for TF-20 will continue.
- 2) Efforts to improve non-destructive testing for SAP pressure tubing using ultrasonic and eddy-current testing techniques will continue.
- 3) The TS 10-4 test section will be assembled, installed in TF-10 and testing will begin.
- 4) TF-50 thermal cycle testing on end seals, mechanical couplings, transition joints and ball valves will continue.
- 5) TF-20 fabrication will be completed.
- 6) The WR-1 in-reactor test section design will be completed and parts will be ordered for fabrication.
- 7) Metallurgical evaluation of burst test specimens will be accomplished.
- 8) 1/1 biaxial stress ratio burst tests will be initiated.
- 9) 2/1 biaxial stress ratio burst testing will continue with long-time specimens.

HWO CR QUARTERLY REPORT

TITLE: Transition Joints

WBS NO: 21200

REPORTING PERIOD: April through June 1966

I. PROJECT OBJECTIVES

The objective of this task is to develop a transition joint between the in-core and out-of-core sections of the HWO CR pressure tube assemblies. These joints are required to maintain strength and leak-tight integrity under all anticipated conditions of reactor operation. The reference pressure tube material for the HWO CR is SAP, with the possibility of using zirconium alloys as a back-up. The SAP transition joints being developed and/or evaluated for these purposes include:

- a) SAP to stainless steel hot pressed joints
- b) SAP to stainless steel rolled joints

SAP stainless steel explosively formed joints have been tried, and since these show the least potential for success, they have been discontinued in favor of the other approaches. Limited work is also being performed in development of the following Zircaloy transition joints:

- a) Zircaloy to stainless steel hot pressed joints
- b) Zircaloy to stainless or carbon steel tandem extruded joints
- c) Zircaloy to stainless or carbon steel rolled joints

The operating environment of the pressure tube includes internal pressure and flow of organic coolant at temperatures up to 765°F, thermal cycling, pressure cycling, and vibration and radiation. The pressure tube will also be subjected to piping reactions and other mechanical loads. The principal potential problem areas associated with transition joints are: (1) multiaxial thermal and mechanical stresses, (2) formation of brittle intermetallics at the joint interface by diffusion, and (3) reactions, such as hydriding, with the organic coolant.

An intensive test and evaluation program for joints which appear most promising is being conducted, including mechanical and metallurgical evaluations. The most promising joint types will be tested as parts of pressure tube

assemblies, both out-of-pile and in-pile. The development of a suitable diffusion barrier between the steel and the SAP or zirconium pressure tube material is an integral part of this project.

II. PROGRESS DURING THE REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. SAP-STEEL HOT PRESSED TRANSITION JOINTS

1. Fabrication of Experimental Joints

Hot pressing of experimental test joints continued during the quarter. Four additional SAP-stainless steel transition joints were fabricated in accordance with Figure 21200-1. Joints No. H-31 and H-32 were of the SAP 895 - 347 SS type. The other two joints, H-33 and H-34, were of the SAP 895 - 304 SS type. Buckling was observed in the type 304 stainless steel joint components as a result of the hot pressing operation (50 tons load at 1025°F). The indication is that the higher yield strength of the 347 SS at pressing temperatures is required to withstand the pressing loads. The somewhat more stringent welding control requirements between the transition joint and the type 304 stainless steel are considered acceptable.

A process specification for fabrication of hot pressed HWOCR size transition joints of the SAP 895 - 347 SS type with a Ni-W diffusion barrier has been written and released. The equipment used for hot pressing SAP-steel transition joints is illustrated in Figure 21200-2.

2. Testing and Evaluation of Joints

Test data for SAP-steel transition joints fabricated in accordance with Figure 21200-1 are summarized in Table 21200-1. Joints were tensile tested to determine their strength in the as-bonded condition and after various thermal treatments.

Joints employing A-106B carbon steel (H-25 and H-27) as the steel component gave an as-bonded tensile strength at 750°F about the same as joints fabricated with type 347 stainless steel. The fractured joint specimens typically revealed a considerable amount of SAP clinging to the face of the steel component (60 degree included angle).

Figure 21200-1. SAP-Steel Transition Joint Tensile Test Specimen

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The effect of thermal cycling was determined on two transition joints. The joints were alternately heated and cooled by lowering into the heated organic bath and raising into a cooled argon cover gas using the test rig shown in Figure 21200-3. One joint, No. H-24 was subjected to 1100 complete thermal cycles between 550 and 750°F and remained helium leak tight with no apparent loss in tensile strength. Similar results were obtained for Joint No. H-17 which was cycled 10,000 times between 650 and 750°F.

Thermal stability tests of bond strength were determined for Joints H-16, H-29, and H-32 by heating in air and subsequently He leak testing and tensile testing. An isothermal heat treatment of 312 hr at 1000°F led to leakage and complete loss of bond strength in Joint H-16 due to extensive tungsten-aluminum intermetallic formation. Joints H-29 (XAP 001 - A-106B) and H-32 (SAP 895 - 347 SS) were heat treated 1000 hr at 950°F and remained He leak tight. H-32 revealed an apparent loss of tensile strength of about 40% from previous room temperature values for as-bonded joints. Joint H-29 was tensile tested at 750°F and showed a tensile strength of about 20% less than the expected value for an as-bonded joint. After tensile testing, practically no SAP remained on the steel component of H-32 while a considerable amount remained on the steel component of H-29. Both joints were examined metallographically. A small portion of SAP remaining on H-32 revealed no measurable tungsten-aluminum intermetallic

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a. Front View

Figure 21200-2. Equipment for Fabrication of Hot Pressed
SAP-Steel Transition Joints
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b. Rear View

Figure 21200-2. Equipment for Fabrication of Hot Pressed
SAP-Steel Transition Joints
(Sheet 2 of 3)

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c. Internal View of Retort

Figure 21200-2. Equipment for Fabrication of Hot Pressed SAP-Steel Transition Joints
(Sheet 3 of 3)

TABLE 21200-1
FABRICATION AND TEST DATA FOR SAP-STEEL HOT PRESSED
TRANSITION JOINTS

Bonding Temperature (°F)	1025 ± 15
Bonding Time (min)	5 ± 1
Bonding Load (tons)	50
Diffusion Barrier	Nickel-Tungsten
Joint Geometry	4.000 in. ID by 0.187 in. wall, 60 degree double knife edge on steel component

Joint Number	Material	Steel Material	Tensile Test Data			Remarks
			Temperature Test (°F)	Ultimate Load, (lb)	Ultimate Tensile Strength (psi)	
H-16	SAP 895	347 SS	RT [†]	NIL	—	Heated at 1000°F (air atm) 314 hr, not He leak tight
H-17	SAP 895	347 SS	750	25,000	10,100	Thermal cycled 10,000 times 650 to 750°F, He leak tight
H-24	SAP 895	347 SS	750	26,200	10,500	Thermal cycled 1100 times 550 to 750°F, He leak tight
H-25	XAP 001	A-106B	750	28,900	11,700	As-bonded
H-27	XAP 001	A-106B	750	25,750	10,400	As-bonded
H-29	XAP 001	A-106B	750	20,450	8,270	Heated at 950°F (air atm) 1000 hr, He leak tight
H-32	SAP 895*	347 SS	RT [†]	45,500	18,400	Heated at 950°F (air atm) 1000 hr, He leak tight
H-33	SAP 895*	304 SS	750	31,300	12,600	As-bonded, 304 SS component buckled
H-34	SAP 895*	304 SS	—	NT [†]	—	304 SS component buckled

*New lot of SAP 895

[†]RT = room temperature

NT = not tested

formation which suggests that the joint might have been defective originally, possibly due to a poor nickel or tungsten coating or a lack of total bonding during hot pressing. H-29 revealed an intermetallic thickness range of 0.000058 - 0.000155 in. and an average of 0.000102 in. The microstructure of H-29 is illustrated in Figure 21200-4. It is noteworthy that this amount of intermetallic did not produce a catastrophic loss in tensile strength.

B. TEST SECTION FOR THE TF-10 LOOP FACILITY

Fabrication of the TS-10-4 pressure tube test section was completed. The 20-ft-long test section consists of a SAP 895 tube (4.000 in. ID by 0.100 in. wall)

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Figure 21200-3. Test Rig for Thermal Cycling
in Santowax R - Argon Media

Figure 21200-4. Microstructure of Joint No.H-29 Showing W-Al Intermetallic Formation After 1000 Hr at 950°F

with transition joint assemblies flash welded to each end of the tube. The transition joint assemblies consist of hot pressed SAP 895 - 347 SS (nickel-tungsten diffusion barrier) joints with A-106B carbon steel extensions welded to the stainless steel. The test section is illustrated in Figure 21200-5 showing the lower carbon steel extension in the foreground. A sectioned transition joint assembly typical of those used in TS-10-4 is shown in Figure 21200-6.

Final checkout of the test section prior to shipment to Windsor included a hydrostatic pressure test of 745 psig for 5 min followed by a helium leak test which showed no leakage at a leak rate sensitivity of $<5 \times 10^{-10}$ scc/sec.

C. NONDESTRUCTIVE TEST (NDT) DEVELOPMENT

Preliminary studies have been completed applying nondestructive eddy current techniques for thickness measurement of the nickel and tungsten coating on the steel transition joint component. Measurements for control purposes must be made on the 60° knife edge of the stainless steel section of the joint before hot pressing. This preliminary work was performed using nickel and tungsten foils on a type 347 stainless steel joint section. Results have demonstrated that tungsten thickness values can be measured in the range up to 0.0040 in 0.0001 in. increments with available commercial instrumentation. The nickel substrate

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Figure 21200-5. TS 10-4 Pressure Tube Test Section
Assembly

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Figure 21200-6. Sectioned Transition Joint Assembly Typical
of Those Used in the TS 10-4 Test Section

introduces a small bias which must be standardized out by the fabrication, qualification, and use of appropriate calibration standards. The nickel layer thickness alone has not been successfully measured by this method because of its relatively strong ferro-magnetism. Results of measurements of the nickel thickness in a saturating magnetic field have been erratic, probably because of probe design and the difficulty of obtaining a uniform field in the restricted space available. Part configuration has required that special probe fixturing and miniature probe coils be designed and procured.

Primary emphasis of NDT development is currently being placed on the tungsten thickness measurements. Test specimens are being prepared having maximum and minimum thickness tungsten layers on maximum and minimum nickel substrates. These specimens will be qualified as calibration standards by uniformity of eddy current response and correlation with tungsten thickness values obtained metallographically.

D. EXPLOSIVE BONDING

The significant results of six explosive bonding tests run with 6063 aluminum tubing (substituted for SAP) and stainless steel tubing were described in the previous quarterly. Niobium foil was used in a portion of the interface in the sixth run.

Three additional explosive bonding tests, runs 7, 8, and 9, were made during this quarter. The stainless steel tubes for these runs were coated beforehand with approximately 0.0005 in. of tungsten metal vapor deposited by the hydrogen reduction of tungsten hexafluoride. SAP tubing was substituted for the type 6063 aluminum tubing used for the first six runs. Run No.7 was the most successful run of the total of nine runs. Approximately 1/2 in. of Gelex explosive thickness was used for this test. After bonding, the joint was cut in half and x-rayed. Approximately 80% of the joint was continuous and free of voids. Break-up of the tungsten coating was observed in two areas. This is believed due to the tubes not being precisely concentric when bonding occurred.

Six segments of the joint were tested for shear strength. Values from 1760 to 5700 psi were obtained with an average of 3760 psi. Two sections of the joint were thermal cycled between 550 and 750°F at a rate of 10°/min, or faster. No deleterious effects were found after 250 of these cycles.

Test runs No.5, 8, and 9, were unsuccessful. Insufficient explosive was used in run No.8 resulting in very little bonding. A thin sheet of rubber was placed adjacent to the OD of the stainless tube in run No.9 in an effort to obtain better distribution of the detonation forces, and the same amount of explosive was used as in run No.7; but unfortunately the Cerrobend inner mandrel was not cast properly. Voids in the cerrobend caused severe distortion of the tubes and very localized bonding.

Explosive bonding test run No.7 using a tungsten metal diffusion barrier film between the SAP and stainless tubes resulted in good bonding around approximately 80% of the interface with little or no formation of intermetallic phases. This result demonstrated a potential for a successful explosive bonding process. However, it appears that such parameters as size of explosive charge, location of charge, and gap configuration will require extensive investigation in order to obtain satisfactory bonds.

The explosive bonding effort was concluded during this quarter in order to concentrate efforts on the more promising hot-pressed and rolled joints.

E. DIFFUSION BARRIER STUDY

A basic problem associated with SAP-steel transition joints involves the formation of brittle intermetallics at the joint interface by interdiffusion at reactor service temperatures. This phase of the program is directed towards inhibition of this diffusion by the application of candidate barrier materials in thin layers at the interface. Tungsten, rhenium, tantalum, niobium, molybdenum, vanadium, and vanadium-silicon duplex coatings were selected for investigation.

Diffusion couples are being heat treated at various times and temperatures in an effort to define an expected relationship of the type

$$x^n = kt \quad ,$$

where

$$k = k_0 e^{-Q/RT} \quad ,$$

and

x = thickness of diffusion zone (cm)

t = time (hr)

Q = activation energy (cal/mole)

T = absolute temperature ($^{\circ}$ K)

R = universal gas constant = 2 cal/ $^{\circ}$ K-mole.

Since the pressure tube transition joints are expected to operate at temperatures to 765 $^{\circ}$ F for 30 years, it is imperative to know the extent of diffusion at these conditions and how much diffusion can be tolerated. This information must be obtained by accelerating the diffusion rate at temperatures above 765 $^{\circ}$ F. Of primary importance is the determination of the activation energy (Q) for diffusion of the barrier material with SAP. It can be shown that the required time (t_2) at a higher temperature (T_2) to give an equivalent diffusion thickness for 30 years (t_1) at 765 $^{\circ}$ F ($T_1 = 680^{\circ}$ K) is related to Q by the relationship

$$\ln \frac{t_2}{t_1} = \frac{Q}{R} \left(\frac{1}{T_2} - \frac{1}{T_1} \right)$$

Diffusion data have been obtained by metallographic examination of several couples in the SAP-W and SAP-Re systems at temperatures of $952 \pm 12^\circ\text{F}$ and $1062 \pm 7^\circ\text{F}$. These data are presented in Table 21200-2. For the relationship

$$x^n = kt$$

it was determined that $n = 3/2$ for SAP-W at 1062°F and for SAP-Re at 952°F . The data for SAP-W are plotted in Figure 21200-7. The 952°F data line for SAP-W is represented by a dashed line, since a value for n cannot be realistically computed from a single test result.

TABLE 21200-2
DIFFUSION DATA FOR SAP-W AND SAP-Re SYSTEMS

Couple Number	Couple Type	Annealing Temperature ($^\circ\text{F}$)	Annealing Time (hr)	Diffusion Layer Thickness (cm)
W-28	SAP-W	1062	8	3.2×10^{-3}
W-26	SAP-W	1062	66	1.28×10^{-2}
W-27	SAP-W	1062	250	3.2×10^{-2}
W-22	SAP-W	952	1000	8.7×10^{-3}
R-19	SAP-Re	952	100	1.62×10^{-2}
R-20	SAP-Re	952	500	3.63×10^{-2}
R-21	SAP-Re	952	1000	7.50×10^{-2}

Insufficient data are available for accurately calculating the activation energy for the SAP-W system at present. As previously reported, it is suspected that a possible phase transformation exists in the temperature range of 950 to 1000°F .*

*AI-CE-31, HWO CR Quarterly Technical Progress Report, January through March 1966, p 21200-26

Figure 21200-7. Plot of Diffusion Data for SAP-Tungsten

If this is so, the value of the diffusion constants will probably differ below and above the transformation temperature. Assuming that the transformation does not exist and that $n = 3/2$ at 952°F, the data in Table 21200-2 lead to a calculated minimum activation energy (Q) for the SAP-W system of about 60,000 cal/mole. If these assumptions are correct, an important significance of the 60,000 cal/mole Q value is that a thermal exposure of 770 hr at 950°F would give an equivalent diffusion thickness after 30 years at 765°F.

Niobium continues to appear attractive as a diffusion barrier to SAP. No intermetallics were observed in couples with thermal exposures of 5000 hr at 950°F and 1000 hr at 1050°F. The potential degradation of the joint by hydriding damage to the niobium will be pursued in the test coupons associated with the WR-1 953 and 954 irradiation experiments.

F. SAP-STAINLESS STEEL ROLLED JOINTS

1. Separate Inner Sleeve Configuration

The SAP pressure tube of TS-10-2, which was equipped with the separate inner sleeve type joints, completed over 1000 hours of operation under simulated HWO CR thermal and mechanical stress conditions. There is no evidence of organic coolant leakage through the joints. A final helium leak check, dimensional, and a visual inspection of the joints and tube will be made.

Two short double ended specimens of the inner sleeve configuration have been fabricated. Both assemblies showed no measureable leakage on helium leak tests before and after tensile cycling. The tensile cycle schedule was as follows:

Test Number	Number of Cycles	Tensile Load on Joint (lb)	Internal Tube Pressure (psi)
1	500	2000	200
2	500	2000	15
3	500	3000	100
4	500	3000	150
5	500	3000	200

One section is currently undergoing thermal cycle testing in the TF-50 loop and the other will be thermal cycled in TF-50 during July and August 1966.

2. Sandwich Type Rolled Joints

The sandwich type rolled joint specimen configuration is shown in Figure 21200-8. An initial fabrication difficulty with splitting of the SAP tubing during the flaring operation was overcome by applying a compressive axial load to the tube during flaring. SAP tubing was flared for joint fabrication by mechanical mandrels, hydraulic expanders and by explosive forming. All three methods proved to be satisfactory. Rolled joints were fabricated in pairs, one at each end of a short test section of SAP tubing, approximately 18 in. long. This arrangement permitted the welding of end plates to the stainless steel sleeves to effect end closures. The end plates also served for attachment of a hanger rod and air line for internal gas pressurization of the joints during thermal cycling and also for attachment of pull test fixtures for mechanical strength tests of the joints.

The first pair of sandwich type (Model 13) rolled joints were fabricated using 0.082 in. wall thickness SAP tubing flared hydraulically to fit the annulus between the two stainless collars. Because the stainless sleeves were fabricated to accommodate a tube wall thickness of 0.110 in. with an assembly radial clearance

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Figure 21200-8. Double Ended Sandwich Type Rolled Joint Specimen (Model 13)

of 0.020 in., it was necessary to expand the SAP to 9% in the hoop direction. Some reduction of wall thickness of the SAP and the stainless sleeves during rolling increased the hoop expansion of the SAP to approximately 10%. The joints were helium leak checked, inserted into a salt bath, gas pressurized internally to 100 psi and thermal cycled 1000 times between 500 and 760°F.

No measurable helium leakage was observed before or after the test at an instrument sensitivity of 10^{-10} cc/sec. The joints were subsequently tensile tested at room temperature. Failure occurred at 32,000 lb in the SAP tubing at the base of one of the joints. The specimen after the tensile test is shown in Figure 21200-9. Sectioning of the joints revealed that the shearing action of the grooves in the steel collars caused a circumferential crack in the SAP during rolling approximately 0.750 in. from the SAP tube steel sleeve interface. The cross-sectioned views of the joint are shown in Figures 21200-10 and 21200-11. It was apparent that the leak tightness and strength of the joint was provided by a 3/4-in. length of SAP tubing sandwiched between the stainless collars.

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Figure 21200-9. First Model 13 Specimen After
Tensile Test

A second pair of joints identical to the above was fabricated, helium leak checked, and subjected to 1000 thermal cycles between 500 and 800°F. A helium leak check after thermal cycling showed no measurable leakage. This pair was then inserted into the salt bath for a 500-hr thermal soak, the upper joint at 960°F and the lower at 850°F. A helium leak check after the thermal soak test showed no measurable leakage until the joint temperature was reduced below 275°F. This pair of joints will be reheated to 750°F by reinsertion into the salt bath and helium leak checked again to determine if the joint becomes self sealing above some minimal temperature. This test will be followed by a mechanical tensile test and metallographic sectioning to determine whether any intermetallic diffusion had taken place at the 960 or the 850°F temperatures. It is suspected that in spite of the joint's excellent leak tightness, the joint may have a circumferential crack in the SAP since all of its fabrication parameters were similar to the first pair.

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Figure 21200-10. Cross Section of One End of
First Model 13 Specimen

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Figure 21200-11. Cross Section of Other End
of First Model 13 Specimen

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Figure 21200-12. Cross Section of Model 15 Sandwich Type
Rolled Joint Specimen

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Figure 21200-13. Autoclaved Explosively
Bonded Zr-2 - SS Transition Joint

Based on the above experience, the stainless sleeve configuration was modified to reduce the assembly clearance and, therefore, the hoop elongation required in the SAP, to simplify the use of hydraulic expansion mandrels for flaring the inner steel sleeve prior to rolling, and to permit easier removal of the rolling tool. These changes are reflected in the Model 14 joint.

To determine whether the reduced amount of hoop elongation eliminated the circumferential cracking of the SAP, two Model 14 joints were rolled and sectioned; both had circumferential cracks in the SAP. A third joint in which the groove taper was increased (Model 15) was fabricated and sectioned. As shown in Figure 21200-12, no cracking of the SAP tubing occurred. To determine whether grooves are necessary, a grooveless specimen was fabricated, helium leak checked and thermal cycled for 1000 cycles between 500 and 800°F with 100 psi internal pressure. No visible leakage could be observed. The joint will be helium leak checked and, if leak tight, subjected to a 500-hr thermal soak at 950°F followed by a tensile test and metallographic examination for intermetallic diffusion.

G. EXPLOSIVE BONDED ZIRCALOY-STAINLESS STEEL TRANSITION JOINTS

Explosive bonding is being investigated as a potential fabrication method for a zircaloy to stainless steel transition joint for the case of a zircaloy pressure tube. Explosively bonded plates of zircaloy 2 to Type 304 stainless steel as well as a transition joint machined from the bonded plates were obtained through a courtesy of E. I. duPont de Nemours and Company. Autoclaving in 750°F steam produced a continuous black oxide film on the zircaloy with no evidence of accelerated corrosion at the bond interface (Figure 21200-13). Welding tests on 0.075 in. thick specimens had no detrimental effects on bond strength nor on the corrosion behavior of the zircaloy. Diffusion heat treatments for 500 hours in inert gas lowered the tensile breaking strength by approximately 50%, from 60,000 psi for the as-bonded metals to 33,000 and 28,000 psi following the 750 and 850°F heat treatments. All specimens failed in a brittle manner at the bond interface without measurable ductility in either base metal. The significant decrease in breaking strength is due to the formation of a continuous intermetallic layer, 3 to 4 μ in thickness, at the bond interface (Figure 21200-14). The as-bonded metals were free from intermetallics except for discontinuous pockets of

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Figure 21200-14. Intermetallic Diffusion Layer Formed
During 750°F Heat Treatment

Figure 21200-15. Autoclaved Coextruded
Transition Joint of Zr-4 to
Stainless Steel

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of solidified jet. The transition joint will be thermal cycled to investigate the effect on mechanical properties and intermetallic formation.

H. TANDEM EXTRUDED TRANSITION JOINTS

Zircaloy-4 to Type 304 stainless steel transition joints, tandem extruded by two commercial vendors, NMI and Torrance, have been evaluated for integrity of bond (Figure 21200-15). Nondestructive tests including visual examinations to 40x, fluorescent penetrant inspection and ultrasonic inspection (performed on NMI joints only) revealed no evidence of joint separation. Autoclaving in 750°F steam developed a uniformly black oxide film on the zircaloy portion of joints from the two vendors. Room temperature tensile tests of longitudinal strips showed failures in the all-zircaloy portion adjacent to the transition region. The results also indicated that the Torrance joints were in a cold worked condition (80,300 psi yield strength for zircaloy), as compared to the NMI joints which showed a yield strength of 53,500 psi, typical for annealed zircaloy. The 9-in. long strip specimens machined from the Torrance joints bowed 3/8 in. due to high compressive stresses in the stainless steel within the bi-metallic portion of the joints. Chisel and notch bend tests readily separated the zircaloy from the stainless steel along the bond interface of the Torrance joints. The NMI joints successfully passed these tests. Poor bond quality of the Torrance joints was attributed to the presence of oxides or other impurities at the bond interface (Figure 21200-16). These impurities may be the result of improper surface preparation or to oxidation during extrusion due to failure of the evacuated mild steel jacket. The NMI joint exhibited a continuous metallurgical bond with a 1-micron thick layer of intermetallics at the bond interface (Figure 21200-17). The effect of thermal cycling of an NMI joint on mechanical behavior and integrity of bond will be evaluated.

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Figure 21200-16. Impurities at Bond Interface of
Torrance Joints

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Figure 21200-17. Bond Interface of NMI Zr-4 -
SS Joints

III. NEXT REPORT PERIOD ACTIVITIES

A. SAP-STAINLESS STEEL HOT PRESSED JOINTS

Activities planned for the next report period include the following:

- 1) Fabricate a test section for the TF-20 loop.
- 2) Fabricate test articles for the TF-50 loop.
- 3) Continue thermal cycling tests and initiate pressure tests of transition joints.
- 4) Fabricate rod type transition joint specimens for irradiation in the WR-1 reactor.
- 5) Continue diffusion barrier studies.
- 6) Continue NDT development for control of nickel-tungsten coating thicknesses on steel joint components.
- 7) Continue evaluation of flash welded SAP pressure tube material.
- 8) Initiate development of nondestructive testing methods for flaw detection in the hot pressed joint and the flash weld.
- 9) Pursue methods to form SAP tubes with an increased wall thickness at the ends which will permit the hot pressed joint to be made directly on the SAP pressure tube.

B. SAP-STAINLESS STEEL ROLLED JOINTS

The following activities are planned:

- 1) An evaluation of the grooveless joint will continue to determine its mechanical strength and leak tightness after 1000 thermal cycles between 500 and 800°F and after a 500-hr, 950°F thermal soak.
- 2) The evaluation of interdiffusion at the SAP-steel interface will be continued.
- 3) Rolled joints for a TF-20 test section will be fabricated.
- 4) Testing of the separate inner sleeve type of rolled joint will be completed.

C. ZIRCALOY-STAINLESS STEEL METALLURGICALLY BONDED JOINTS

Evaluations of metallurgically bonded zirconium stainless steel transition joints from three commercial suppliers will be completed.

HWO CR QUARTERLY REPORT	
TITLE: Fuel	WBS NO: 28000
REPORTING PERIOD: April Through June, 1966	

I. PROJECT OBJECTIVES

The objective of the fuel development program is to provide a fuel element system which meets HWO CR requirements. This program includes design, fabrication, development, testing, and evaluation of fuel element assemblies which demonstrate high neutron economy and low fuel cycle costs.

The present HWO CR reference fuel string involves rod bundle type assemblies using hyperstoichiometric uranium carbide clad with SAP.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. FUEL MATERIALS

The principal objective of this project is to advance UC fabrication technology to produce high quality, low cost hyperstoichiometric UC fuel within the dimensional and compositional range specified for the HWO CR concept.

1. Feed Material Process Development

A series of 11 small-scale (10 kg) UC synthesis experiments was completed during this report period. The experiments were conducted to evaluate the effect of an improved vacuum environment, attained by the addition of a booster-diffusion pump, on feed material impurity content, and to determine if a more efficient sintering cycle, based upon reduced reaction temperatures ($<2050^{\circ}\text{C}$) and pressures, is feasible. Experiment variables included three sintering temperatures (1850, 1950, and 2050°C), two nominal finishing pressures (1.5 to 2.0μ , the limit attainable with the booster pump vs 15 to 30μ attained with a mechanical forepump), and two nominal hold times (3-3/4 and 4-3/4 hr at sintering temperature with the forepump operating, or 1/4 vs 1-1/4 hr after the

ultimate pressure was reached with the booster pump operating). The results obtained are shown in Table 28000-1.

TABLE 28000-1
RESULTS OF UC SYNTHESIS EXPERIMENT

Run No.	Booster Pump Used	Operating Temperature (°C)	Total Cycle(1) Time (hr)	Final Pressure (μ)	Chemical Analyses		
					C(2, 3) (wt %)	O ₂ (ppm)(3)	O ₂ (ppm)(4)
1	Yes	2050	5	1.65	4.77 $\pm$ 0.03	52 $\pm$ 32	157 $\pm$ 153
2	Yes	1850	6	2.05	4.76 $\pm$ 0.03		626 $\pm$ 53
3	No	1850	5-1/4	13-1/2	4.85 $\pm$ 0.02	570 $\pm$ 201	
4	Yes	1250	6-1/4	1.60	4.79 $\pm$ 0.06	231	293 $\pm$ 120
5	Yes	2050	4-1/4	1.57	4.85 $\pm$ 0.03	345 $\pm$ 83	715 $\pm$ 213
6	Yes	1850	6-3/4	1.25	4.76 $\pm$ 0.07	278 $\pm$ 125	339 $\pm$ 84
7	No	1950	5-1/4	12	4.74 $\pm$ 0.02	810 $\pm$ 297	817 $\pm$ 271
8	Yes	1950	5-1/4	1.30	4.84 $\pm$ 0.07	703 $\pm$ 58	693 $\pm$ 97
9	No	2050	4-1/4	12	4.73 $\pm$ 0.06	650 $\pm$ 56	637 $\pm$ 93
10	No	1950	4-1/4	9	4.86 $\pm$ 0.07	-	-
11	No	2050	5-1/4	9	4.81 $\pm$ 0.04	-	-

- Notes: (1) Total cycle time includes approximately 30 to 40 min to reach operating temperature.
 (2) Tests up to and including Run 6 were made with a batch containing 11.60% carbon; all succeeding batches prepared with 11.58% carbon.
 (3) Material sampled in argon dry box.
 (4) Material sampled in normal procedure; i. e., lab. air environment.

The results indicate that a satisfactory product (i. e., UC with low oxygen content) may be obtained at temperatures as low as 1850°C, but apparently at the expense of increasing the total cycle time. Completion of the reaction, required to achieve a low oxygen product, seems to be influenced more by higher sintering temperatures (by virtue of increasing reaction rate and diffusion of CO out of the agglomerate) than by lower system pressures. However, the use of low pressures in conjunction with reference operating temperatures will assure continuous production of high quality UC feed material.

2. Feed Material Process Scaleup

Parameters were established for manufacturing operations associated with the production of UC feed material to existing specifications, both in sizes of up to

100 kg for HWOCR and 60 kg for PNPf Core II. A process specification for HWOCR feed material production was proposed.

It was concluded that further scaleup to approximately 120 kg agglomerate batches is feasible in the process equipment being installed in the new UC-SAP Pilot Facility. This scaleup would be expected to reduce the production cost and increase UC capacity in the Pilot Facility by approximately 10% with no increase in equipment requirements.

3. UC Skull Casting Scaleup

During the last report period, two approaches toward eliminating the mold cracking problem were either in the process of being evaluated or scheduled for evaluation during the current report period. The first approach using end plugs with increased diametral clearance of 0.010 in. in place of the usual 0.002 in. between mold wall and plug did not prevent the mold from fracturing at the web. The molds for this initial series of scaleup experiments were composed of ATJ grade graphite, cylindrically-shaped, and having 91-cavities in a hexagonal close packed array.

The other approach to minimizing the cracking problem was to use four 25-cavity square-shaped modular or multi-section molds in place of the single 91-cavity mold, on the theory that mold fracture would be confined to the module in which it initiates and not propagate to other sections of the mold system. Thus, only a quarter of a new mold would have to be replaced, assuming only one module failed, as opposed to the entire mold. Four melts were completed using a combination of new and used graphite mold modules to produce a variety of as-cast slug diameters. Module life, in terms of the number of melts in which the module could be used, is given in Table 28000-2. Based upon the limited results shown, the average module life was determined to be 3.75 melts. The immediate goal is to achieve a minimum average life of 6 melts per module.

4. Material Finishing

Scaleup centerless grinding development was initiated during the report period following installation and qualification testing of a Cincinnati grinding machine. A total of seven preliminary production process test runs were completed on batches of UC slugs having as-cast diameters pertinent to both HWOCR and PNPf Core II designs. The runs were made for the purpose of determining the maximum diametral stock removal per pass as a function of minimum slug breakage. The results are summarized in Table 28000-3.

TABLE 28000-2
MODULE LIFE TEST RESULTS

Module No.	Total Number of Melts Module Used	As-Cast Slug Diameter (in.)	Observations	Disposition
1	3	0.484	Cracked on 1st and 2nd melts	Reusable
2	5	0.491	Cracked on 1st, 4th and 5th melts	Not reusable
3	6	0.489	Cracked on 1st, 3rd, 4th and 5th melts	Not reusable
4	3	0.369	Module did not crack during melt cycle; cracked during removal of slugs.	Not reusable
5	1	0.284	Did not crack	Reusable
6	1	0.487	Cracked on 1st melt	Not reusable
7	1	0.491	Did not crack	Reusable

TABLE 28000-3
MAXIMUM DIAMETRAL STOCK REMOVAL PER GRINDING PASS
VS SLUG DIAMETER

Test Run	Slug Batch Size (kg)	Average Slug Diameter (in.)		Total Number of Passes	Maximum Diametral Stock Removal Per Pass (in.)				
		Before Grinding	After Grinding		Pass 1	Pass 2	Pass 3	Pass 4	Pass 5
1	20	0.493	0.475	5	0.002	0.003	0.005	0.005	0.002
2	11	0.493	0.475	5	0.002	0.003	0.007*	0.005	0.002
3	8	0.473	0.461	4	0.002	0.003	0.005	0.002	-
4	12	0.473	0.462	4	0.002	0.003	0.004	0.002	-
5	3.5	0.364	0.354	4	0.002	0.003	0.003	0.002	-
6	8	0.364	0.354	4	0.002	0.003	0.004	0.001	-
7	4	0.281	0.272	4	0.002	0.002	0.003	0.002	-

*0.007 in. removal per pass was eventually reduced to 0.005 in. to minimize slug breakage

On the basis of the data available from the seven test runs completed to date, a preliminary centerless grinding production throughput rate for UC slugs correlated to slug diameter has been prepared and is given in Table 28000-4.

TABLE 28000-4
MASS RATE OF FINISH GRINDING* UC FUEL SLUGS
AS A FUNCTION OF SLUG DIAMETER

Slug Diameter (in.)	Mass Rate of Slugs Processed (kg/hr)	Total Length of Slug Processed (in. /hr)
0.475	9	240
0.462	8	240
0.354	6	240
0.272	4	240

*Using a Cincinnati centerless grinder with a diamond wheel

Two series of small-scale carbothermic reduction tests utilizing recovered sludge (almost completely oxidized UC) were completed. In the first series of experiments, four individual batches of dried sludge (300 to 500 grams each) plus a control batch of ceramic grade UO_2 powder were reduced to carbide simultaneously. In a second series of four experiments, ceramic grade UO_2 with additions of 0, 5 wt %, 10 wt % and 20 wt % recovered sludge were reduced, with graphite, to carbide in 5-kg lots. The products from each series of experiments are currently being analyzed and the data tabulated. Preliminary results to date indicate that it may be feasible to recycle the sludge directly into feed stock production without a major purification cycle.

B. SAP DEVELOPMENT

The objectives of the SAP development activity are to:

- 1) Develop reliable sources for SAP alloys; evaluate and establish quality and uniformity of SAP raw materials.
- 2) Develop fabrication processes and product specifications for semi-finished and finished HWO CR SAP shapes.

- 3) Develop mechanical properties data on SAP alloys directly applicable to HWOOCR design.
- 4) Develop nondestructive and destructive inspection techniques for raw materials and semi-finished SAP products.
- 5) Maintain technical cognizance of ORNL SAP development program and other programs relating to the application of SAP to HWOOCR.

1. SAP Procurement

An AI-CE representative visited Montecatini in Italy, and the Danish Research Establishment, Riso, at Roskilde, Denmark to discuss problems associated with SAP production, general SAP technology, and irradiation effects on SAP. A trip report is being prepared.

Montecatini has recently installed a vacuum sintering furnace to allow 8 to 12 SAP compacts to be sintered at one time rather than the previous limit of one at a time. The previous delay in delivery schedule is attributed to the installation of this new furnace. The furnace had only been in operation on an experimental checkout basis before the HWOOCR order.

The quality control procedures were fully agreed upon between HWOOCR and Montecatini representatives. Such items as the exact test specimen configuration, methods of controlling strain rates for tensile testing, specimen location, orientation, and specific methods of analysis were resolved. Montecatini can successfully produce hollow billets (for tube blanks) and will ship 80% of the 7-in. -diameter order as hollows. Further experiments will be required to produce 7-in. -diameter solid billets without some surface roughening problems. Montecatini also stated a preference for producing a finished product to specification rather than the semi-shape for HWOOCR. The schedule for shipment of SAP by Montecatini for HWOOCR is presented in Table 28000-5.

The first shipment of 2-in. -diameter billets (XAP 005) was received from ALCOA. The certified test results from the supplier's tests have been reviewed. The results of ALCOA mechanical tests on the 2-in. -stock are shown in Table 28000-6.

Metallographic specimens of SAP stock, supplied with the order, are being examined at AI for comparison with the results of nondestructive tests. Similar comparisons on the chemical compositions of the material are in progress.

TABLE 28000-5
DELIVERY SCHEDULE FOR PRODUCTION OF SAP-ISML
FOR HWOCR DEVELOPMENT NEEDS

Scheduled Shipping Date	No. of Billets	Diameter* (in.)	Alumina (%)	Remarks †
6-30-66	31	7	10	With hole
7-31-66	6	7	10	With hole
7-31-66	28	2	10	Without hole
8-31-66	27	7	10	With hole
9-30-66	34	2	7	Without hole
10-31-66	7	2	7	Without hole
10-31-66	4	7	7	With hole
10-31-66	23	7	10	Without hole

* The 7-in. -diameter billets are 2 ft long
The 2-in. -diameter billets are 18 ft long.

† "With hole" indicates that these billets contain a 60-mm-diameter hole oriented longitudinally to the extrusion direction and concentric with the outer surface. These billets will be used as tube blanks for direct pressure tube extrusion.

TABLE 28000-6
SHORT-TIME TENSILE TEST RESULTS ON SAP 005 BILLET MATERIAL
(2-in. -Diameter rods)

Compact No.	Room Temperature			900°F		
	Yield Strength (0.2%) (ksi)	Ultimate Tensile Strength (ksi)	Elongation (%)	Yield Strength (0.2%) (ksi)	Ultimate Tensile Strength (ksi)	Elongation (%)
23	26.4	34.1	20.5	8.2	9.1	3.5*
51	26.8	34.3	20.0	8.5	9.1	5.0
29	26.1	34.4	23.0	8.2	8.9	7.5
56	27.0	34.8	20.5	8.3	9.1	4.5
31	27.9	34.9	24.0	8.4	8.8	9.5
40	27.6	35.1	22.0	8.7	9.3	4.5*
33	28.1	35.1	23.0	8.5	9.1	9.5
28	27.3	35.2	20.5	8.9	9.6	6.5
43	27.5	35.2	23.0	8.8	9.5	8.5
5	27.6	35.2	18.5	8.2	9.0	4.0
54	27.1	35.3	19.0	8.6	9.2	6.0
48	28.1	35.3	21.0	8.7	9.3	7.0
67	28.1	35.4	18.5	8.5	9.0	4.0
41	28.6	35.4	22.0	8.6	9.1	4.5*
36	28.4	35.6	22.0	8.9	9.3	6.0
47	28.0	35.7	20.5	8.8	9.3	5.0
70	28.7	36.2	18.5	8.5	9.2	3.5*
74	28.6	36.3	19.0	8.8	9.7	2.5*
59	28.2	36.7	17.0	9.3	10.0	4.0
751	28.2	36.7	19.5	9.3	9.9	4.0*
60	29.0	36.7	17.0	9.1	9.7	3.5
64	29.0	36.9	18.0	9.2	9.8	6.0
65	29.6	37.6	18.0	9.6	10.2	4.5*
81	29.0	37.8	15.5	9.4	10.2	3.0*

*Specimen broke at gage mark.

2. Fabrication and Mechanical Properties

Tube burst stress-rupture tests of fuel cladding tubes planned for use in fuel irradiations have been initiated; the results of tests completed to date are presented in Table 28000-7. Further tests are in progress and will be reported in subsequent reports along with the Carson-Miller plots so derived.

3. Irradiation Tests

A series of four SAP irradiation tests has been performed at Riso in the DR2 and DR3 reactors. These irradiations were concluded when the specimens reached an integrated dose of 4×10^{20} nvt (> 2.9 Mev).^{*} Irradiations at 35°C showed a slight increase in strength, decrease in elongation, and a decrease in impact strength over unirradiated samples in post-irradiation tensile and impact testing.⁽¹⁾ Tests on material irradiated at 270°C showed no detectable effect of irradiation on the short-time properties.⁽²⁾ Tests performed on specimens irradiated at 400 to 420°C also showed no effect of irradiation on the short-term tensile and impact properties of SAP over the post-irradiation test temperature range of room temperature to 600°C.

Riso does not presently plan to do further studies on the effects of irradiation on the properties of SAP. A number of samples which have been irradiated to 4×10^{20} nvt (> 2.9 Mev) but not tested are presently in storage at Riso. Indications are that these specimens could be made available to the HWOCR program for further study (such as additional irradiation in the DR3, or physical, chemical, and mechanical studies). A list of the specimens is presented in Table 28000-8. A rough estimate of possible future irradiation time in DR3 was that (1) the specimens could be back in the DR3 as soon as two months after a contractual agreement could be reached, and (2) doubling the present dose would take another six months.

A trade study is now being performed to determine how these specimens can best be utilized for the HWOCR SAP Development program.

A preliminary design package for a SAP tensile specimen irradiation test was prepared. This test, tentatively set for irradiation in the ETR reactor, will

^{*}This value has been calculated to be equivalent to 1.2×10^{21} nvt ($E > 1.0$ Mev)

TABLE 28000-7
TUBE BURST STRESS-RUPTURE TEST RESULTS -
SAP CLADDING

Alloy	Test Temperature (°F)	Hoop Stress (psi)	Rupture Time (hr)
895	750	6940	1525
		6710	1178
	850	5620	543
		5620	396
		5440	788
		5440	1407
001	750	6430	1118
		6075	1503
	850	4650	516
		4650	465
		5290	22
		4950	371
		5290	12
		4950	83
		4650	876
		5010	800
		4650	634
		4650	879
005	750	6430	14
		6075	38
		6075	1257
		6430	1643
	850	4650	0.5
		5010	24
		4950	129
		4950	4
930	850	4400	140
		4650	551

TABLE 28000-8
IMPACT AND TENSILE SPECIMENS OF SAP-AIAG*

Irradiation Temperature (°C)	Alloy Designation	Al ₂ O ₃ (%)	No. of Specimens
35	SAP 865	14	22
35	SAP 895	10	51
35	SAP 930	7	53
270	SAP 865	14	22
270	SAP 895	10	22
270	SAP 930	7	22
400	SAP 865	14	49
400	SAP 895	10	46
400	SAP 930	7	50

*These specimens have been irradiated to $4 \times 10^{20} > 2.9$ Mev nvt at Riso but have not yet been tested. They could be made available for HWOCR Development needs.

contain 172 tensile specimens to be irradiated to a peak fast flux (> 1 Mev) exposure of 5×10^{21} nvt. The preliminary design operating conditions for the SAP tensile specimen irradiations are shown in Table 28000-9.

The technical approach to the study of irradiation effects on SAP is to conduct creep-rupture tests on as-fabricated and heat-treated control specimens,* for comparison with post-irradiation tests of irradiated specimens. The selection of test conditions and irradiation conditions is to be carried out to provide statistically reliable evaluation of irradiation effects on SAP.

C. SAP JOINING DEVELOPMENT

The objective of this task is to develop the machining, assembly and welding techniques for SAP-clad fuel bundles required for the HWOCR program. The principal efforts are directed at the development of magnetic force welded

*The control specimens will be subjected to a thermal history simulating the in-pile thermal conditions.

TABLE 28000-9
EXPERIMENT DATA SHEET – SAP TENSILE
SPECIMENS IRRADIATION TEST

General

Irradiation Facility.	ETR
Number of test assemblies	4
Number of usable specimens per assembly	43
Specimen material	5-7 wt % and 10 wt % SAP alloys

Nuclear Data

Peak fast (> 1 mev) flux (nv)	3×10^{14}
Peak thermal (< 0.4 ev) flux (nv)	5×10^{14}
Design peak integrated fast flux dosage (nvt)	5×10^{21}
Estimated residence time in ETR core (yr)	1

Heat Transfer and Hydraulic Data

Coolant flowrate (gpm)	16.5
Coolant velocity past test section(ft/sec)	~12
Coolant temperature rise (°F) at 10 w/gm	~6
Film coefficient (Btu/hr-ft ² -°F)	1838
Equivalent diameter, De (ft)	0.021
Re (based on De)	3.53×10^4
Peak design gamma heat rate (w/gm)	10-12
Peak linear heat rate at 10 w/gm (kw/ft)	6.25
Test assembly power output at 10 w/gm (kw)	12.0
Specimen peak design temperature (°F)	800

Stress Data

Peak thermal stresses (at 16 w/gm) (psi)	
Outer shroud tube	8230
Inner shroud tube	4850
Maximum pressure stress (psi)	2670

(MFW) end closures for the fuel rod. In addition, this task is responsible for the development of the magnetic force welding process for the fabrication of end closures for the Piqua Core II fuel elements.

The 150 kva magnetic force welder had been installed prior to the reporting period. A number of welds were made to become familiar with the operating characteristics of the machine. During the reporting period, an effort was made to improve the quality and reliability of the magnetic force welds in the 0.487-in. -ID x 0.025-in. -wall Piqua cladding. Room temperature tensile tests were used as a screening method for initial process parameter development. This resulted in a process which made end closures that were stronger than the clad tubing in room temperature tensile tests. A group of twenty welds was prepared and tensile tested at room temperature and at 850°F. Table 28000-10 compares these results with those obtained on the cladding and the welds produced at Battelle Northwest Laboratories (BNWL). Since the samples failed in the weld during elevated temperature tensile testing, an effort was made to improve the weld quality by further process optimization. An "evolutionary operation" (EVOP) technique is being used to investigate the effect of variations in the weld current, the phase between the magnetic force and weld current, and the magnet air gap. The results are being evaluated using 850°F tensile testing. EVOP is a technique that can be employed to investigate the effect of three independent process variables by making small systematic changes to these input variables. After each set of changes the results are reviewed. The review results in further changes and subsequent evaluations, seeking through gradual alteration to optimize the process. Figure 28000-1 is an example of a typical EVOP cube with the results obtained. While the results of the first EVOP cube did not produce samples that failed in the tube during elevated temperature tensile testing, an improvement can be seen in hot tensile strength over welds produced using previous settings.

Some preliminary welding development was also done on the small-size Piqua cladding (0.363-in. -ID by 0.025-in. -wall). A set of parameters was found that would reliably produce samples that fail in the tube during room temperature tensile testing. Table 28000-11 summarizes an EVOP run on the small diameter cladding using room temperature tensile testing as a means of evaluating the weld quality.

FACTORS STUDIED	X	A	B	C	AB	AC	BC	ABC	0
WELD DELAY (%)	15	25	15	15	25	25	15	25	20
WELD CURRENT (kiloamps)	52	52	56	52	56	52	56	56	54
MAGNET GAP (mil)	140	140	140	190	140	190	190	190	165

HOT TENSILE RESULTS (850°F)									
X	lb	A	lb	B	lb	C	lb	AB	lb
19*	440	13	435	32	472	25	427	41	430
20	464	14	485	33	455	27	430	42	435
22	402	16	455	34	483	29	NO TEST	43	455
23	465	17	290	36	452	30	442	44	442

AC	lb	BC	ABC	lb	+	lb
7	446	†	46	450	1	510
8	422		49	448	3	425
11	340		50	429	4	375
12	NO TEST		54	435	6	470

*WELD NUMBER
†NO TEST ARC ERUPTIONS

Figure 28000-1. Typical Evolutionary Operation Cube

TABLE 28000-10
RESULTS OF TENSILE STRENGTH TESTS OF
MAGNETIC FORCE WELDS

Sample Group	No. of Samples	Ultimate Tensile Strength*			Fracture Location		
		(psi)			Weld	Tube	Grip
Avg	Max	Min					
Room Temperature Tests							
BNWL	10	39,900	41,850	34,600	5	5	--
AI	10	36,900	38,500	35,900	--	1	9
Tubing QC	5	35,500	36,300	34,500	--	--	--
850°F Tests							
BNWL	10	10,950	11,900	8,900	10	--	--
AI	9	10,500	11,000	9,500	9	--	--
Tubing QC†	5	10,700	11,000	10,500	--	--	--

*Ultimate strength of the welds calculated on the cross sectional area of the thin wall of the tubing adjacent to the welds.

†The tubing data were obtained at 900°F and are extrapolated to 850°F.

TABLE 28000-11
SUMMARY OF EVOP RUN IN ROOM TEMPERATURE
TENSILE TESTING

Factors Studied	X	A	B	C	AB	AC	BC	ABC	O
Weld Delay (%)	20	35	20	20	35	35	20	35	30
Weld Current (k-amp)	38	38	42	38	42	38	42	42	40
Magnet Gap (mil)	160	160	160	190	160	190	190	190	175
Failure Location (T=Tube, W=Weld)	T	W	T	T	T	W	No Test	T	50% T 50% W

Figure 28000-2. 850°F Uniaxial Tensile Stress Rupture Curve for GNWL Magnetic Force Welded End Closures

After the EVOP investigations it appeared that a more extensive investigation of additional parameters affecting the welds would be required. Statistical analysis was used to determine that a designed experiment based on a two replicate fractional factorial would provide the best method of determining the effects and interactions of the following variables: welding current, magnet air gap, clapper size, preload, and the delay or phase relationship between the magnetic force and the weld current. This experiment is currently in progress.

Several of the MFW closures made at Battelle Northwest have been stress rupture tested at 850°F in uniaxial tension. The results plotted in Figure 28000-2 indicate that the long-time rupture strength of these welds is considerably less than the strength of the tubes. The scatter in the data further demonstrates the need for additional process development. The following least squares regression equation has been developed for the BNWL data:

$$\ln (\text{Rupture Time}) = 125.90 - 14.469 [\ln (\text{stress})].$$

Three BNWL welds selected at random were examined every 0.050 in. on planes parallel to the end closure axis. The results indicate that there were fairly extensive weld defects which usually could be traced through several layers. The 20 tensile specimens tested during the previous quarter were examined metallographically during this period. All of them showed some evidence of weld defects. Weld defects were also observed in the fracture surfaces of the stress-rupture specimens.

The high resolution ultrasonic test being developed for the MFW joint is providing promising results. Test development to date has been performed on sections of actual joints containing 0.014-in. -diameter drilled holes and 0.004-in. -wide by 0.030-in. -long electrical discharge machined slots. These were machined into the joint zone from both the inside and outside surfaces of the weld samples. It has been found possible to visually discriminate on the oscilloscope screen between indications from the artificial flaws and indications from the inside surface weld flash. Since both types of indications appear at the same position on the oscilloscope screen, flaw detection becomes a problem in pattern recognition. This type of pattern recognition is difficult to resolve and record by instruments, whereas it is a relatively easy task for an operator

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Figure 28000-3. U-3 Loop Coolant Activity Between U-305 Rupture and Termination

to visually distinguish these differences in pattern. Work is continuing with the objective of eliminating the signal from the flash so that conventional ultrasonic instrument gating and recording systems can be used.

D. IRRADIATION TESTING

1. U-305 Irradiations

a. Initial U-305 Irradiation

Irradiation of the U-305 experiment was terminated on April 16, 1966, due to the apparent progressive deterioration defect which had occurred on April 8. The initial failure was detected during steady reactor operation by a sharp increase in coolant radioactivity. Samples of coolant and cover gas were examined on a pulse height analyzer and the presence of gaseous fission products (Xenon and Krypton) and associated daughter products in quantities much larger than normal was confirmed. No radio iodine was observed. Following the initial failure, operation of the U-305 experiment continued in a normal manner except for several additional coolant activity bursts on April 12, 13, and 14, all of which decayed to normal levels. On April 15, the reactor was shut down for reasons not associated with the U-3 loop and, upon startup 1/2 hour later, a sharp increase in coolant activity was noted. This activity again decayed to a normal level but was followed by a sharp rise to 45,000 cpm (coolant gamma activity). For the next 12 hr, the coolant activity remained above 13,200 cpm and several spikes towards 45,000 cpm were observed. (Refer to Figure 28000-3 for a complete record of the coolant gamma activity.) These later activity spikes were different than those which occurred prior to the reactor outage, in that the activity rose to the maximum level within 1 to 5 min as compared with 1/2 to 1 hr for the previous bursts. The behavior of the fuel string was closely evaluated at the time; it was determined that sufficient information had been obtained from operating the defected element and that further operation might jeopardize the materials information from the fuel string and the condition of the loop. Consequently, the U-305 irradiation was terminated by shutting down the reactor on April 16. The experiment remained in the loop for a 40-hr cooling period and was then transferred to an HB-40-filled thimble in the NRU storage block. Removal of the experiment from the process tube posed no problems and there were no indications of interference or binding.

During the operation of the experiment in the defected state, the following observations were made: (a) there were no deviations from normal in the experimental data; i. e., the pressure drop and cladding temperatures remained normal. (b) A correlation between reactor power cycles and coolant activity spikes was not apparent; i. e., cycling the reactor power did not necessarily produce an increase in the coolant activity.

The U-305 experiment had resided in the U-3 loop of the NRU reactor for a period of nine months and had accumulated a peak UC burnup of approximately 10,000 Mwd/MTU, exclusive of end peaking. Considering end peaking, the maximum burnup could be as high as 14,000 Mwd/MTU. During the operation there were periods in which film formation had apparently occurred and periods in which the film thickness appeared to decrease, as indicated by a decrease in the normalized temperature parameters. In some cases, there appeared to be a correlation of surface temperature changes with the coolant chemistry parameters (particularly the water content); however, the results were not conclusive. Over the lifetime of the experiment, there was an increase of approximately 50°F in the cladding temperatures, most of which occurred during the first five days of operation. The initial pressure drop across the test section was 71 psi, and over the duration of the test the pressure drop increased to about 104 psi. This increase showed no apparent correlation with the cladding temperature increases. Measurement difficulties experienced with the differential pressure instrumentation system were such that there may be an appreciable uncertainty in the actual value of the observed pressure drop and pressure drop change. In general, the U-305 test met its design objectives by successfully operating at and above HWOCR design conditions to a peak UC burnup of 10,000 Mwd/MTU.

Following discharge, the U-305 fuel string was held in a storage block for approximately seven weeks to reduce the decay heat generation rate. The U-305 experiment was moved from the storage block to the universal cell at Chalk River on June 7 where the initial visual examination was performed. Areas of bare cladding were observed on all bundles. A thin brown-colored translucent layer was observed adjacent to the cladding on the SAP-UC bundles; a somewhat thicker black flaky film was observed on top of this layer. The film on the upper Zr- UO_2 bundle was thin and black in appearance, and could be loosened in large flakes. The lower Zr- UO_2 bundle had a very thin, powder-like black

film which was easily removed. In general, the film formation can be assessed as nominal. Subsequent film thickness measurements indicated a maximum thickness of 1.5 mils, which is reasonably consistent with the observed film temperature increase, assuming a film thermal conductivity of 0.5 Btu/hr-ft-°F, and a surface heat flux of 326,000 Btu/hr-ft².

A detailed visual examination of all exterior surfaces of the complete string revealed two bulges in the cladding of one rod in the upper Zr-UO₂ bundle (JRA). One bulge was a circumferential ridge at the weld between the upper end cap and cladding; the other was a bulge under one fin approximately 2-1/2 in. above the lower end of the same rod. Circumferential ridging in one rod of the lower Zr-UO₂ bundle (JRA) was also observed and photographed. All of the stainless steel parts (end hardware and spacer rings) were very clean and free of film deposits. Based upon the initial examination, no obvious defects were observed in any of the rods and no anomalies were observed on any of the SAP-UC bundles (LTA, LSA, and LRA). The hot cell examination was interrupted on June 10th and the string returned to storage.

The U-305 fuel string was returned to the hot cell on June 16. The surface film noted during the previous examination in the hot cell was now very loose, with large strips detached. On June 17, the bottom two Zr-UO₂ bundles were removed. The removal was easily accomplished and all hardware was very clean. The bottom end plate of the lower SAP-UC bundle was in excellent condition, as evidenced by the visibility of machining marks.

The upper Zr clad bundle was placed under the stereomicroscope for observation at 20X. Several cracks and defects were found on the rod facing the center of the core. The most severe defect was found 4 to 5 in. from the top of this rod. In this area, severe cracking of the cladding extending approximately 1/2 in. along the rod between and including two adjacent fins was observed. In the center of this area a piece of cladding was missing. The cladding void was about 0.1 to 0.2 in. square and the fuel had cracked and bulged out into this area. Other areas of extensive cracking were observed on this rod in the region of the end cap weld, although not so severe as that described above. Defects on two and possibly three rods in addition to those above have been observed. Similar but less severe defects were also observed on the lower Zr-clad bundle.

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Figure 28000-4. U-305 AI Beta Parameters

No defects were observed on the SAP-UC bundles confirming the earlier observations.

The first phase of the post-irradiation examination of the U-305 experiment was completed at Chalk River on June 24, 1966. The bottom SAP-UC bundle and the upper Zr- UO_2 bundle (the two high flux bundles) were shipped to Canoga Park for detailed hot cell examination. The balance of the string is in storage at Chalk River pending future disposition.

b. Alternate I U-305 Irradiation

Following the defect and subsequent removal of the U-305 fuel string, the U-305 Alt. I assembly was inserted in the U-3 Loop. This fuel string is being operated as a film formation test, and is designed to evaluate the effect on film formation characteristics of temperature, coolant water content, coolant purity, (as controlled by clay bed adsorption) and low boiler concentration. In general, when a new coolant condition is attained it will be maintained for a few weeks to evaluate the effect it produces.

The U-305 Alt. I experiment was brought to full power on May 12. The initial operating conditions were a 572°F inlet temperature and a 300 ppm coolant water content. These conditions were specified in an attempt to duplicate the initial operating period of the U-305 experiment with the exception of the coolant water content. For the U-305 experiment, the initial water content was less than 50 ppm; however, during subsequent operation it was found that 300 ppm appeared to be helpful in inhibiting surface temperature rises. In the U-305 test, the first five days of operation resulted in film formation which increased the cladding temperatures by about 50°F. For the U-305 Alt. I test, no such cladding temperature increase was observed during the initial period of operation. Figure 28000-4 shows the beta parameters (ratio of the coolant film ΔT to the coolant channel ΔT) for the U-305 Alt. I operation to June 24, 1966. The first several weeks are very steady. The scatter in the data for June is attributable to unsteady reactor operation and instrumentation calibration problems. The initial operation of the U-305 Alt. I test has shown that it is possible to define a system such that no significant film formation will occur when a fuel string is initially operated. Another possible indication that there has not been any appreciable film formation is the presence in the coolant of the Mg^{27} recoil

product from the $\text{Al}^{27}(\text{n}, \text{p})\text{Mg}^{27}$ interaction. The target aluminum in this case is the SAP cladding and if it were covered with sufficient film, the recoil atoms of Mg^{27} would not be able to penetrate the film. Pulse height analysis of the coolant from the previous U-305 test which was known to have had film deposition did not indicate the presence of this isotope.

The calibration and operation of the pressure drop instrumentation for the U-305 Alt. I test has received additional attention. During the initial startup, the dynamic pressure drop (which does not include the 14 psi static head) was indicated to be 74 psi, which was almost identical to that for the initial U-305 run. After one shutdown and repeated calibrations, it became apparent that the instrumentation used to measure the pressure drop was unreliable. A portable precision pressure gauge was moved to the site and the high and low pressure leg readings were used to determine the dynamic pressure drop. With this instrumentation arrangement, the dynamic pressure drop was 88 psi on June 6 (the corresponding ΔP cell reading was 74 psi prior to the calibration adjustment). During the shutdown between June 12 and 14, the dynamic pressure drop increased to 94 psi and during the following week's operation it gradually increased to 96 psi. It is planned to install a new pressure drop measurement system which incorporates three separate differential pressure transmitters, each with separate pressure legs. The piping for these pressure legs is being rerouted so that the distance involved is halved as compared to the existing route. The installation of this system is scheduled for early in the next report period.

During the shutdown in the first week of June, a leak was noted in the upper seal region of the fuel element. This leak released about a pound of coolant per day and was not considered serious. However, on June 25 this leak suddenly increased and by the time the reactor was shut down and the loop depressurized and cooled, four to five drums or about one-third of the total loop coolant inventory had been lost onto the top of the reactor. Subsequent investigation showed that the graphite thermocouple lead-through seal had failed. The seal has been repacked and is currently operating without significant leakage.

The operating conditions as of June 24 for the carbide fueled section of the U-305 Alt. I experiment are listed in Table 28000-12.

TABLE 28000-12
U-305 ALT. I OPERATING CONDITIONS
(June 24, 1966)

Power level (Mwt)	1645
Maximum Measured Cladding (T/C) temperature (°F)	799
Dynamic pressure drop (psi)	96
Coolant water content (ppm)	289
Coolant inlet temperature (°F)	572
Estimated peak UC burnup (Mwd/MTU)	1718

2. WR-1 Irradiation Tests

a. EXP-WR1-951

After several delays due to problems with both the test assembly and the reactor, nuclear operation of the first in a series of SAP-UC irradiation experiments in the WR-1 Reactor, the WR1-951 Experiment, was initiated on May 30, 1966. The WR1-951 experiment had initially been inserted in the WR-1 reactor on April 6, 1966. A spill of HB-40 coolant occurred on April 7, during change of a reactor pressure tube. As a result of the spill, the WR1-951 experiment was removed from the reactor on April 7, visually inspected, washed in HB-40, and stored in its handling fixture.

On April 27, the experiment was reinserted into position E-6 and full flow established. A coolant leak developed in the graphite end closure; operating conditions were 100°C coolant and 50 psig system pressure. Corrective action was unsuccessful in stopping the leakage and the experiment was removed from the reactor on April 29. The experiment was then shipped to Canoga Park where the graphite seal was replaced with a brazed seal following successful completion of qualification tests on this new seal at both Whiteshell and Canoga Park. The experiment was then shipped back to WNRE and inserted in position E-6 of the reactor on May 25. Prior to initiating nuclear operation, the thermocouples were cross calibrated at three different coolant temperatures and the flow - ΔP characteristics were determined for the WR-1-951 test assembly.

As indicated above, EXP-WR1-951 began nuclear heat generation on May 30. The first period of operation ended on June 3, after five days of WR-1 operation at 50% power, when the reactor was shut down to evaluate a leaking pressure tube

problem. This shutdown, during which three pressure tubes were removed and examined, lasted until June 16. Nuclear operation then continued until June 21 when a broken bolt on a seal in the "B loop" degasification system forced a shutdown. Operation was again resumed on June 26 and has continued through the end of June with only minor power outages. During the latter operational period, EXP-WR1-951 has run in the 70 to 85% of design power range. A typical set of operating conditions is listed in Table 28000-13.

TABLE 28000-13
EXP-WR1-951 OPERATING CONDITIONS
(June 29, 1966)

Power level (Mwt)	0.885 (81.5% of design)
Coolant flow (gpm)	307
Coolant inlet temperature (°F)	635
Coolant ΔT (°F)	35.1
Peak measured cladding (T/C) temperature (°F)	762
Peak measured fuel (T/C) temperature (°F)	1560
Moderator level (in.)	82.1

Due to the wide variation in moderator level (66 to 83 in.) and the resulting shift of the peak axial power, the use of a film formation parametric analysis has involved considerable uncertainties during this initial operating period. It does appear however that no significant film formation has occurred to date.

b. EXP-WR1-952

As result of the seal leakage problems with the WR1-951, the graphite seal was replaced on the WR1-952 experiment. At the close of the report period, the WR-1-952 test assembly was prepared for shipment to WNRE. All approvals have been obtained from AECL on the WR-1-952 irradiation proposal.

c. EXP-WR1-953 and-954

At the close of the report period, the WR-1-954 end item specification and experiment layout drawing were in final approval stages. It was decided not to use the available 10% SAP alloy in the WR-1-954 experiment. This decision resulted

from the fact that the 10% SAP tubing, as extruded from the available alloy, showed a tendency to split during twisting operations and no definitive cause could be associated with this problem. Furthermore, since a three to four month delay would result from waiting for 10% SAP alloy material from Montecatini, it was decided to use all 7% material in the WR-1-954 in order to achieve the other test objectives at the earliest date. The WR-1-953 test will, however, incorporate extruded finned SAP cladding of the 10% alloy from Montecatini.

E. OUT-OF-PILE TESTING

1. Hydraulic Testing

The objectives of the hydraulic test program are to determine the hydraulic characteristics of typical HWO CR fuel bundles and associated hardware. The areas of specific interest are the determination of the velocity profiles, pressure losses, coolant mixing, and vibration characteristics of HWO CR fuel element assemblies. The following hydraulic tests (low temperature water tests) were completed during the report period.

a. TF-3 Tests

Four potential HWO CR end plate designs (2 flat and 2 stepped) were compared on the basis of pressure drop characteristics at fluid velocities in the 15 to 35 fps range. The flat end plates exhibited substantially lower pressure drops in the aligned position but equal or slightly higher pressure drops in the misaligned condition. The stepped end plates tested, on the other hand, while exhibiting a higher pressure drop in the aligned position, were less sensitive to misalignment. The changes in pressure drop with misalignment for the four sets of end plates at 30 fps are presented in Figure 28000-5.

The pressure-drop, flowrate characteristics of a full-scale, 2-string, 37-rod fuel bundle assembly were determined in the water facility. The complete mockup is shown in Figure 28000-6. Each bundle has a stepped Zircaloy end plate at one end and a flat end plate at the other. The tests were conducted "back-to-back" with matching end plates to compare their hydraulic characteristics. The rods in the fuel assembly closely simulated those in the Demonstration Plant Reference Design with the exception of fin height (0.076 in. vs 0.056 in. for the Demonstration Plant.)

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Figure 28000-5. Comparison of Pressure Drop Characteristics for Stepped and Flat End Plates

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Figure 28000-6. TS 3-1, Hydraulic Test Model

The tests were conducted with 168°F water under conditions geometrically and dynamically similar to HWOCR coolant conditions. Pressure losses associated with the following components were obtained:

- a) Entrance and exit losses across stepped (reference design) and flat end plates
- b) Friction losses across the finned fuel bundles
- c) Fuel bundle junction losses with stepped end plates back-to-back and flat end plates back-to-back for varying degrees of misalignment ranging from 0 to 60 deg.

The water data were converted to reactor coolant conditions (Santowax OM + 10% HB between 575 and 750°F) by applying the principles of model theory.

Pressure drop data from these tests have been utilized to determine the hydraulic characteristics of a full-length, 5-string prototype channel. This information is presented in Figure 28000-7 for a string of 5 fuel bundles with stepped end plates in the aligned and worst case misaligned positions for 575°F inlet and 750°F outlet coolant conditions, assuming a linear rise in coolant temperature through the core. For the reference design case of a 5-string fuel bundle with stepped end plates and a coolant velocity within the fuel bundle of 30 ft/sec, the pressure drop can vary between 129 and 145 psi, depending upon the degree of alignment. For flat end plates the corresponding ΔP range is 123 to 156 psi.

Comparison of entrance and exit losses at the design flowrate shows the stepped end plate losses to be 15% lower than those for the flat end plates.

The fuel bundle junction pressure losses showed the following:

- a) Flat end plates exhibited approximately a 2.2/1.0 change in ΔP with misalignment, whereas stepped plates exhibited about 1.5/1.0 change.
- b) The losses appear to be symmetrical about the 30 deg misalignment case.

Friction factors computed from the experimental data on frictional losses across the finned fuel bundles are plotted in Figure 28000-8 and compared with the Moody smooth tube curve.⁽³⁾ These results are in agreement with those of other investigators; namely, friction factors of finned rod bundles are lower than smooth tube friction factors.

Figure 28000-7. Pressure Drop Across 5-String Fuel Assembly with Stepped End Plates at Reference Design Temperatures (575°F Inlet, 750°F Outlet)

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Figure 28000-8. Fuel Bundle Friction Factors

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b. Coolant Mixing Tests

Work continued on the development of the noise analysis method for determining coolant mixing within HWOCR fuel assemblies. Hydraulic tests were conducted with a 61-channel test section to determine the crossflow between two adjacent finned SAP cladding tubes. Figure 28000-9 is a schematic representation of the crossflow test section. Analysis of the experimental data indicated a mean value for the crossflow ratio of 0.380 with a standard deviation of ± 0.177 .

Dye injection mixing test work was initiated this report period. The purpose of the dye-injection tests is to provide a relatively inexpensive and quick means for measuring mixing in the 37-rod finned fuel bundles. The results of the dye injection tests will also focus the effort on the salt injection tests in the specific areas which need more detailed investigation.

The two test fuel bundles which were built for the pressure drop tests in the TF-3-Loop are being modified for the dye-injection mixing tests. Small-diameter stainless steel tubing was fitted inside of five fuel tubes of each bundle for injecting dye and for extracting water samples. During testing, the diluted dye will be injected into one subchannel and downstream of the injection point. Water samples will be taken from the injection subchannel as well as from the surrounding subchannels. The dye concentration of the samples will be determined by measuring the dye coloration by means of a spectrophotometer. Knowing the axial change in dye concentration, the crossflow rate can be determined and the mixing ratio can be calculated.

The main variables which will be investigated during the mixing tests are: flowrate (15 to 40 ft/sec), bundle alignment (0 to 40 deg misalignment), and water temperature (70 to 200°F).

CROSS FLOW TEST SECTION

TYPICAL ELECTRODE AND PRESSURE TAP DETAIL

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Figure 28000-9. Dual Channel HWO CR Cross Flow Test Section

III. NEXT REPORT PERIOD ACTIVITIES

A. FUEL MATERIALS

Scaleup UC skull casting will be continued with major emphasis on reducing the mold cracking problem and improving the overall quality of as-cast hyperstoichiometric UC slugs ranging in diameter from 0.290 in. to 0.490 in. Preliminary process specifications based upon current casting scaleup development will be prepared.

The effort on centerless grinding parameter development for the HWOCR prototype fuel slugs will be directed toward optimization of the process at its current level of development. Process reliability will be established and preliminary process specifications will be prepared.

B. SAP DEVELOPMENT

Mechanical tests of cladding tube material, with known fabrication and non-destructive test results will be continued.

Material from current SAP production will be fabricated to study the effects of fabrication variables on material properties.

Design of the SAP irradiation experiment will be initiated.

Material from current SAP production will be used to prepare specimens for the irradiation experiment.

C. SAP-JOINING DEVELOPMENT

During the next report period, the process development for the large-diameter Piqua fuel rods will be completed. Welding development will continue on the small Piqua and on the WR-1 rod sizes. Welding parameter optimization for all three cladding sizes is expected. The qualification and reproducibility tests will be initiated. Efforts to develop nondestructive testing techniques will also continue. The magnetic force welder will be moved from Building 042 to Building 005 (the new UC pilot fabrication facility) Santa Susana during the month of August.

D. IRRADIATION TESTING

1. U-305 Irradiations

The U-305 Alt I film formation test will be continued. The first effect to be investigated will be the effect of coolant temperature. The coolant inlet temperature will be increased to increase the maximum surface temperature to approximately 845°F. The experiment will be operated in this manner for a period of time to indicate whether or not film formation occurs at this higher temperature. The next portion of the test will determine the effect of low water content. Coolant water content of the order of 50 ppm will be maintained while the coolant inlet temperature is varied between 510 and 620°F. These tests are to determine if film formation occurs as a result of low water content and, if so, whether or not a temperature threshold exists. The next portion of the test is to determine how long the system can be operated without coolant purification by operating the loop with the purification system valved out for a minimum period of four weeks.

The detailed hot cell exam of the U-305 experiment will be initiated.

2. WR-1 Irradiation Tests

a. EXP-WR1-951 and-952

The -951 test will be continued in the WR-1 reactor. The -952 test will probably be inserted into the reactor by the end of August 1966.

b. EXP-WR1-953 and-954

The design, analysis, and fabrication efforts on the WR-1-954 and 953 tests will be continued.

c. WR-1 Test Evaluation

The evaluation of the -951 performance, including data collection, analysis and interpretation will be continued. The computer program, designed to handle the tape punch data obtained from the WR-1 reactor, will be expanded to include the -952 test assembly.

E. OUT-OF-PILE TESTS

1. Component Compatibility

The investigation of the SAP-UC compatibility will be continued.

2. Hydraulic Testing

Additional end plate pressure drop tests will be conducted. Preparations will be initiated for fuel and vibration tests.

3. Coolant Mixing

Coolant mixing tests will be conducted on multi-rod models which are representative segments of the fuel bundle assembly. Preliminary results from the dye-injection mixing tests will be obtained and evaluated.

4. Mechanical Testing

End plate load-deflection tests will be initiated. Preliminary fuel bundle and fuel rod axial compression tests (without magnetic force weld end closures) will be initiated.

REFERENCES

1. RISO Staff, "Metallurgy Section, Annual Progress Report for the Period Ending March 31, 1964," Danish Atomic Energy Commission Research Establishment (RISO), RISO Report No. 90 (August 1964).
2. RISO Staff, "Metallurgy Section, Annual Progress Report for Period Ending March 31, 1965," Danish Atomic Energy Commission Research Establishment (RISO), RISO Report No. 110 (July 1965).
3. L. F. Moody, "Friction Factors for Pipe Flow," ASME Trans. 66, (1944) pp. 671-684.

HWOCR QUARTERLY REPORT

TITLE: Large Components Testing

WBS NO: 31000

REPORTING PERIOD: April through June 1966

I. PROJECT OBJECTIVES

The overall objective of this effort is to proof-test certain main-coolant-loop components in order to assure adequacy of design and fabrication, and to assure reliability after installation in the Demonstration Plant. The principal items of equipment requiring such proof-testing are the main coolant loop pumps and valves.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A preliminary design for a large pump and valve seal test facility was initiated. A P&I diagram of the process systems was prepared (Figure 31000-1). Equipment for the facility was sized; flows, temperatures, and pressures defined; and control and operational procedures established. A site at the Atomics International Santa Susana facility was selected as a possible location for the test facility. Preliminary equipment arrangement drawings and piping drawings were prepared showing the adaptability of the loop to the existing site. Cost estimates, schedules, and manpower estimates were prepared.

Other work performed under this task includes the preparation of components and systems descriptions and test plans.

III. NEXT REPORT PERIOD ACTIVITIES

The principal effort during the next report period will be directed toward firming the design for the seal test facility.

Figure 31000-1. P&I Diagram, Seal Test Loop

HWOCR QUARTERLY REPORT

TITLE: Organic Coolant System Safety

WBS NO: 31800

REPORTING PERIOD: April through June, 1966

I. PROJECT OBJECTIVES

The objective of this effort is to provide reliable information in the area of potential organic coolant accidents and confident accident consequence prediction techniques in a form which is useful for safeguard analyses and evaluation, engineering safeguards design, and the development of plant engineering design criteria for the HWOCR Demonstration Plant.

Accident analysis for organic reactor design is based on either core melting accidents, fuel handling accidents, or organic-coolant spills. Of major importance is the study of the effect of a coolant spill and fire upon the release of fission products and other radioactivity contained in the reactor coolant. Normal design assumes that the reactor would operate with a limited number of defective fuel rods, resulting in the coolant containing fission products. The activity level of the coolant will depend upon the degree to which coolant purification removes the fission products. A major uncertainty is the extent to which the fission products such as I^{131} become chemically bound to the organic and the method by which these are removed by coolant purification and released to the environment. Another uncertainty is the fate of the fission products associated with an organic spill. The release of fission products during a coolant fire is also of importance.

The organic spill accident may strongly influence the criteria for containment design and, therefore, can have a significant effect on the cost of the HWOCR plant. A major part of the uncertainty in this accident arises from the lack of detailed information on the pressure rise due to the thermal energy release resulting from burning organic and the remedial action required in extinguishing an organic fire. The use of water sprays or other extinguishing methods requires analysis and experimental study to evaluate behavior.

An accident which strongly influences the design of the organic pressure relief system and may influence system design is the rupture of a tube in the steam generator. Some developmental testing has been conducted on this problem. Evaluation of the available data will be made to determine whether further testing is required.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A draft of the test plan for characterization of organic fires and fission products was prepared. The general objective of the plan is to ascertain the thermal and radiological consequences of organic fires in organic-cooled nuclear power plants. Specific objectives are the determination of:

- 1) Limits of flammability of hot organic pools and sprays in atmospheres of less than 21% oxygen
- 2) Combustion rate of organic pools as a function of pool and gas temperatures and oxygen concentration
- 3) Vaporization rate of organic liquid as a function of gas and pool temperatures and gas flow rate
- 4) Application of models developed, such as by analytical prediction of temperatures and pressures in a typical HWO CR building as result of an organic fire
- 5) Effect of mitigating water sprays on thermal consequences of an organic fire and on fission product release
- 6) Quantitative release of iodine and other volatile fission products to the atmosphere from pools of organic, with and without combustion
- 7) Transport, fallout, deposition, and filter collection of fission products released from organic pools, both with and without combustion.

Application of expected test results to the safety analysis of an HWO CR is discussed on the basis of potential primary coolant spill accidents in several plant areas. A key factor in the overall program approach will be the utilization of applicable experimental and analytical technology available from sodium fires research at AI.

Preliminary bench-scale experiments were undertaken during this report period to determine the vaporization rate of Santowax OM as a function of sweep gas velocity and temperature, and the minimum oxygen concentration required to render a mixture of organic coolant vapor, oxygen, and nitrogen flammable. In the vaporization experiment, at various temperatures and flows, the organic vapor was collected on a glass-fiber filter and the vapor concentration determined by change in weight of the filter. The results of this experiment are shown in Figure 31800-1.

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Figure 31800-1. Evaporation Rate vs Flow

During the flammability test, mixtures of oxygen, nitrogen, and vapor were allowed to reach equilibrium, after which an electric current was applied to an ignition wire. The results of the test are shown in Figure 31800-2. Upper limits of flammability were not determined, because the initial experimental apparatus was unable to produce higher concentrations of vapor.

A review of available test information on steam generator tube ruptures was completed.

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Figure 31800-2. Percent O₂ in O₂-Na Mixture vs Volume Percent Coolant Vapor

III. NEXT REPORT PERIOD ACTIVITIES

Within limitations of available funding, experimental and analytical work will proceed in accordance with the prepared test plan.

HWO CR QUARTERLY REPORT

TITLE: Evaluation and Analytical Techniques

WBS NO: 35110

REPORTING PERIOD: April 1966 through June 1966

I. PROJECT OBJECTIVES

The objective of this activity is to provide information on the properties and chemistry of organic coolants as required by the HWO CR program. This will be accomplished by evaluating available data and by new experimental work where required. Specific fields to be covered are:

- a) Physical properties
- b) Decomposition and makeup rates
- c) Chemical analysis and testing
- d) Coolant oxidation
- e) Halogen chemistry.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. PHYSICAL PROPERTIES

1. Differential Scanning Calorimeter Measurements

Many measurements were made to study the effect of high-boiler and of p-terphenyl (para) on the melting behavior of synthetic o-m-terphenyl mixtures, simulating proposed HWO CR coolant compositions. The results are being evaluated. Interpretation is complicated by incomplete crystallization, shown by heats of fusion lower than for the pure crystalline compounds. The degree of crystallinity gradually increases over a period of weeks. Addition of HB (from the U-305 experiment), up to the nominal HWO CR concentration of 10%, has little effect on the melting range, but a large effect in decreasing crystallinity.

A typical series of measurements with 10% HB and a 2:1 o-m ratio (similar to Santowax OM) gave the following final melting points as a function of p-terphenyl content:

p-terphenyl %	1.5	4.0	6.5	11.5
Final m. p. (°C)	77	96	103	132

The thermograms (plots of heat absorbed against temperature) for all except the first mixture show two peaks. The second peak is barely visible at 4.0% para, and increases in area with increasing para content. This behavior is in agreement with published phase diagrams for o-p and m-p mixtures. The first peak is due to melting of a eutectic, probably at a little below 4% para, and the second peak corresponds to the melting of the remaining para. The area of the second peak could not be measured accurately because of a slight vaporization which begins at about 100°C. Tightly sealed capsules will be used in later runs to prevent the vaporization.

To determine whether any kinetic effects were involved, two runs were made on the 11.5% para mixture in which the sample was held isothermal for 15 min and 1 hr, respectively, between the first and second peaks. The second peak was approximately the same as when the scan was made without interruption, indicating that a high-melting phase (p-terphenyl) did exist at equilibrium.

Two practical conclusions are:

- a) 10% HB does not prevent separation of p-terphenyl from mixtures similar to Santowax OM; therefore, there would be no advantage in using a different coolant for startup.
- b) A specification of 4% maximum p-terphenyl should be adequate to avoid separation of a significant high-melting phase.

The specific heats of solid o- and p-terphenyl were also measured with the differential scanning calorimeter. At 27°C, the values were 0.28 cal/g-°C for ortho and 0.29 cal/g-°C for para.

2. General

A topical report* giving a compilation of data on physical properties important in reactor design, as a function of temperature and HB content, was issued.

* AI-CE-15, "Physical Properties of Some Polyphenyl Coolants"

B. DECOMPOSITION AND MAKEUP RATES

1. MIT Organic Loop Experiment

The runs on Santowax WR at a fast neutron fraction of 0.06 (fuel position 20) were completed with the termination of Run 18 on June 10, 1966.⁽¹⁾ Runs with a fast neutron fraction of 0.40 in fuel position 1 are expected to begin in August, following dosimetry in both positions, loop maintenance, and installation of a continuous feed-and-bleed system.

Analyses and calculations for Run 17 (572°F) were completed. The average terphenyl-concentration was 89.7% and the HB concentration 6.2%. $G(-omp)$ (molecules of terphenyl decomposed per 100 ev absorbed radiation) was 0.181, and $G(\rightarrow HB)$ (molecules of terphenyl converted to HB per 100 ev) was 0.168, with a standard deviation of 0.006. Analyses on Run 18 are in progress.

2. Effect of Ortho-Meta Ratio

In choosing the HWOCR coolant, it is important to determine whether there is any difference in makeup rate between a high-o-terphenyl coolant, such as Santowax OM, and a high-m-terphenyl coolant such as Santowax WR. Examination and discussion of the available data have led to the following conclusions:

a) Up to 610°F, the average temperature in the U-305 experiment, o- and m-terphenyl have essentially the same value of $G(\rightarrow HB)$, hence makeup rates would be the same for OM and WR.

b) At high temperatures, 700 to 750°F, predicted makeup rates are less certain because of such complications as dose rate effects, radiopyrolysis, and increased activation energy. Available evidence indicates no very large difference in $G(\rightarrow HB)$, which determines makeup rate. However, $G(-omp)$ will be higher for the high-ortho coolant, leading to a higher concentration of intermediate boilers, IB (non-terphenyl coolant).

c) It is recommended that two direct comparisons of the decomposition rates of Santowax OM and WR should be made in the MIT Loop, by means of steady-state runs at 700 and 750°F with 10% HB. A direct comparison of radiopyrolysis with irradiated OM and WR should also be made.

C. CHEMICAL ANALYSIS AND TESTING

1. Water Analyzer

The integrating electrolytic moisture monitor on the U-3 loop dump tank off-gas has continued to operate satisfactorily. During periods of steady reactor operation, the ratio of the instrument reading to the Karl Fischer water content

of the coolant agreed with that observed during the last report period.⁽²⁾ However, during reactor shutdown the readings were erratic. During the period between removal of the U-305 experiment and insertion of Alternate 1, while operating without a fuel element, the water injection pump was turned off. The Karl Fischer water content dropped from 174 to 49 ppm in one day, while the instrument reading changed from 5931 to 3440 μg (from 403 to 244 ppm, using the factor of 14.7 $\mu\text{g}/\text{ppm}$ observed previously.⁽²⁾) Thus, the instrument readings were not only high, but lagged behind in following the decrease in water content. Changing the sampling point to the degasifier instead of the dump tank might eliminate the lag, but probably would not eliminate fluctuations due to shutdown. However, the degasifier instrumentation, giving temperatures, pressures and flow rates, should permit a correct interpretation of such fluctuations. Undoubtedly the best gas sample for water determination would be vapor in equilibrium with the coolant. An in-line vapor pressure meter would be an ideal source of gas samples for water determination.

Evidence was obtained that the Karl Fischer analysis itself may be subject to errors due to loss of water vapor from coolant samples. On June 22, six samples were taken. Three were analyzed immediately, giving 291, 284, and 301 ppm. The following day, two samples gave 250 and 256 ppm, and on June 24, the remaining sample gave 269 ppm. These results are insufficient to prove that a loss occurred, but they do indicate some uncertainty in the Karl Fischer analysis.

2. Chlorine Analysis

In a further test of the neutron-activation coincidence-counting method, analyses were made on samples of Santowax OMP and PNPf coolant to which weighed amounts of triphenyl tin chloride had been added. A good correlation was obtained between chlorine added and found, as shown in Figure 35110-1. The results confirm the accuracy of using aqueous standards to analyze organic samples.

A paper on the coincidence counting method was presented at the Spring Meeting of the American Chemical Society. A topical report describing the method will be released.*

* E. Bramblett, "AI-CE-35 Gamma-Gamma Coincidence Counting Applied to Chlorine Analysis by Neutron Activation"

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Figure 35110-1. Chlorine Analysis of Terphenyls
with Terphenyl Tin Chloride Added

D. COOLANT OXIDATION

The pump for the oxygen absorption loop⁽³⁾ was received. The system was completed and installed in the cobalt-60 facility. To comply with safety requirements, the organic part of the system was modified to permit nearly complete enclosure and provision for flooding with inert gas in case of fire.

E. HALOGEN CHEMISTRY

No experimental work was done in this field during the report period.

III. NEXT REPORT PERIOD ACTIVITIES

Additional melting measurements will be made on o-m-p-terphenyl mixtures using a sealed capsule. A paper on the differential scanning calorimeter measurements will be presented at the fall meeting of the American Chemical Society.

A meeting will be held at MIT to decide the course of the decomposition rate experiments for the next three years.

An effort will be initiated on development of an in-line vapor pressure meter with provision for sampling for water analysis.

The water analyzer in the U-3 loop will be connected to the degasifier instead of the dump tank, and additional data will be obtained.

The topical report on the coincidence counting method for chlorine determination will be issued.

Dose rates in the oxygen absorption loop will be measured with Fricke dosimeters. The performance of the system will first be checked out with HB-40. Oxidation experiments will then be carried out with Santowax OM. The first objective will be an estimate of activation energy.

Some simple experiments will be carried out on the deposition of chlorine compounds from coolant on SAP surfaces, with and without radiation.

REFERENCES

1. AI-CE-4, "HWOCR Quarterly Technical Progress Report, July through September 1965," Table 0550-4.
2. AI-CE-31, "HWOCR Quarterly Technical Progress Report, January through March 1966," Table 35110-6.
3. AI-CE-17, "HWOCR Quarterly Technical Progress Report, October through December 1965," Figure 35110-3.

HWO CR QUARTERLY REPORT

TITLE: Parameters Affecting Film Formation

WBS NO: 35120

REPORTING PERIOD: April through June 1966

I. PROJECT OBJECTIVES

The objectives of this activity are to develop a detailed, quantitative understanding of film formation processes which take place in reactors of the HWO CR design, explore possible methods of reducing film formation rates, and complete the design and test of a film formation rate meter intended to measure in situ the film-forming tendency of organic coolants. A forced circulation loop (Film Formation Loop) will be operated to evaluate the effect on film formation rates of such parameters as temperature, radiation type and intensity, coolant type and composition (including impurities), and coolant flow rate. By developing an understanding of the film formation micro-processes, direct engineering evaluation will be possible of the macro-processes taking place in a nuclear reactor. Some in-pile tests using a fuel element will be conducted in the U-3 loop to supplement the basic work with an accelerator. Film formation rate meters, developed as part of this project, will be tested and evaluated.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. FILM FORMATION LOOP

1. Components and Subassemblies

Purchase orders have been placed for all major components and subassemblies of the Film Formation Loop, with the exception of the heat rejection system. The only significant item to be purchased for this system is a secondary organic circulating pump. It was anticipated that a suitable pump and spare would be available from the Experimental Organic Cooled Reactor (EOCR); however, AI-CE has been informed that these pumps are in use at the National Reactor Test Site (NRTS).

Delivery of the temperature differential and pressure differential systems are at present six months behind schedule because of the inability of the manufacturer to meet performance specifications. Vendors are taking corrective action. Because of delays in the selection of the irradiation site, no overall program delay is currently anticipated because of these late deliveries.

2. Loop Construction

The loop cabinet has been assembled. Installation of tubing and components is being delayed until more items have been received.

3. Water Test Loop

The water test loop, to be used to test various design concepts, has been operated in three successive runs. The purpose of the first run was to acquaint personnel with the loop; during this run, a heater section was not installed in the system. The purpose of the second run was to test flow and electrical control instrumentation, exclusive of the heater section. During the third test, a simple annular heater section was operated at a heat flux of $250,000 \text{ Btu/hr-ft}^2$ for 2 hr. No problems in design, control, or instrumentation have been encountered.

4. Selection of Irradiation Site

Operational difficulties have been encountered with the possible use of the Linac at the University of Saskatchewan. It is unlikely the difficulties will be resolved prior to the scheduled start date for loop irradiations. Requests for Proposal relative to the provision of irradiation services have been forwarded to General Atomic and Rensselaer Polytechnic Institute, the two Linac facilities in the United States capable of providing the required irradiation dose rates.

5. Laboratory Equipment

In support of the Loop activity at the irradiation site, a number of analytical instruments will be necessary. A visible-ultraviolet and an infrared spectrometer have been purchased and tested. Both a low and high temperature gas chromatograph, obtained from other programs, have been renovated and modernized. The high temperature gas chromatograph has been temporarily forwarded to CRNL to support the analytical effort relating to the U-3 film formation experiments.

B. U-3 LOOP FILM FORMATION TESTS

A series of film formation tests were planned for the U-3 loop using the U-305 Alternate 1 fuel string. These tests are designed to provide a better understanding of the cladding temperature behavior observed during the operation of the U-305 experiment. The tests planned are shown in Table 35120-1.

TABLE 35120-1
FILM FORMATION TESTS WITH ALTERNATE 1 FUEL STRING

Tests	Purpose
1) At least 3 weeks as above	Determine whether initial temperature increase on U-305 was due to poor coolant quality and/or low water content or is inherent with a fresh fuel element.
2) Raise coolant inlet to 620°F	Determine temperature effect on film formation rate.
3) Reduce water to 15-20 ppm	Determine if film formation accelerated by low water content in clean coolant.
4) Reduce water to 300 ppm, turn off purification	Determine effect of coolant purity at high water.
5) Reduce water to 15-20 ppm	Determine effect of low water in dirty coolant.
6) Degasifier Tests:	Determine effect of low boiler removal.
a) No condensation	
b) Degas at 50 psig	
c) Degas at 200 psig	

The experiment began on May 11, 1966 and surface temperatures had shown no increase by the end of the reporting period. Thus, the initial temperature increase observed with U-305 did not occur and it is clear that there is nothing inherent in the initial operation of a fresh fuel element with clean coolant which leads to a surface temperature increase. The subsequent tests are expected to allow a separation of the effects of coolant purity, water content, degasification and purification conditions on film formation rate. A substantial portion of these tests is expected to be completed during the next reporting period.

C. FILM FORMATION RATE METER

Three runs have been completed in which U-3 coolant was irradiated using a 100-kv x-ray fluorescence machine. Runs were made at room temperatures, with an accelerating potential of 90 kv and a current of 20 ma. The duration of each run was approximately 30 hr. The dose rate at the metal sample's surface is estimated at 0.01 w/gm. In the first run, a polished tantalum coupon was used. In the second run, both polished and abraded stainless steel coupons were used. In the third run, a polished stainless steel coupon was irradiated in coolant in which the benzene insoluble fraction was concentrated 3:1. A comparison of the irradiated coupon with unirradiated specimens showed no change in surface color or texture, indicating no film deposit. Within the accuracy of weighing ($\pm 5 \mu\text{g}$), no weight changes were noted. The indicated film formation rate, if any, was therefore less than $0.2 \mu\text{g}/\text{cm}^2\text{-hr}$.

III. NEXT REPORT PERIOD ACTIVITIES

A. FILM FORMATION LOOP

Testing of loop components and design concepts in the water test loop will continue.

Assembly of the Film Formation Loop will start.

Evaluation of proposals from the irradiation sites will be completed.

B. U-3 FILM FORMATION TESTS

The test series will continue with the effect of higher surface temperature, water content, and purification being determined.

C. FILM FORMATION RATE METER

Evaluation of various methods for measuring film thicknesses will continue.

HWO CR QUARTERLY REPORT

TITLE: Coolant Purification and Recovery

WBS NO: 67100

REPORTING PERIOD: April through June 1966

I. PROJECT OBJECTIVES

The overall objective of this effort is to establish the optimum organic process systems and operating conditions necessary to achieve high coolant quality at a minimum of equipment and coolant makeup cost in HWO CR plants. The processes to be studied are filtration and adsorption of coolant, and hydrocracking of high boilers (HB).

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. HYDROCRACKER (BENCH-SCALE) SITE SELECTION

A survey of available irradiation sites was made to select a suitable site for installation of an irradiation loop and accompanying bench-scale hydrocracker unit. The following site selection criteria were established:

- 1) The coolant dose rate shall be around 1.0 w/gm.
- 2) The fast neutron fraction shall be in the range of 50 to 75%.
- 3) The cost for the bench scale hydrocracker test program shall be minimized.
- 4) The ratio of out-of-core volume to in-core volume shall be minimized.
- 5) There shall be existing organic coolant laboratory and technology work at the facility.
- 6) The HB generation rate must be at least 20 gm/hr.

The results of the survey are summarized below:

<u>Reactor</u>	<u>Conclusions</u>
NRX	Available E-2 loop too small; dose rate not sufficient.
NRU	Recommended for detailed evaluation.
WR-1	Recommended for detailed evaluation.
MTR	Location is remote from organic test sites. Not recommended.
GETR	Location is remote from organic test sites. Not recommended.
SRE	Major test loop design problems. Not recommended.

<u>Reactor</u>	<u>Conclusions</u>
Hanford	No operating priority; classified operation. Not recommended.
EBR-II	Can irradiate static capsules only. Not suitable.
MITR	In-core volume too small. HB generation rate too small for adequately sized bench scale hydrocracker.
STIR	Recommended for detailed evaluation.

As a result of this survey, a further and more detailed evaluation was made concerning the relative merits of the NRU, STIR, and WR-1 facilities.

The WR-1 was recommended as the first choice, primarily because the estimated cost for utilizing that facility was substantially less than for the other two. STIR was selected as the second choice because its proximity to the existing organic technology laboratories and trained personnel would minimize staffing and communications problems.

B. HYDROCRACKER (BENCH-SCALE) DESIGN

Design of the hydrocracker process system was initiated. The basic design criterion was that the out-of-pile/in-pile volume ratio of the loop would be a minimum so that steady-state coolant conditions could be achieved in less than one month's operating time. The process conditions, i.e., temperature, pressure, and flowrates, were selected so as to duplicate those expected in an HWO CR plant. Provisions were included for multiplate distillation of the coolant so that the HB content in the hydrocracker feed stream can be varied. Provisions for coolant degasification and adsorption processing were also made so that the coolant chemistry and irradiating conditions will closely simulate those of HWO CR. Figure 67000-1 is a preliminary piping and instrumentation diagram of the bench-scale hydrocracker.

III. NEXT REPORT PERIOD ACTIVITIES

If approval to install the bench-scale hydrocracker WR-1 is received, an experimental proposal will be prepared for submittal to AECL at Whiteshell. The system design and program plan will be completed.

PROCESS STREAM DESCRIPTION

STREAM NO.	①	②	③	④	⑤	⑥	⑦	⑧	⑨	⑩	⑪	⑫
PRODUCTION	PRIMARY COOLANT	ADSORBER FEED	RECOVERY LOOP FEED	RADIOLYTIC GAS	DEGASSED PRIMARY COOLANT	DEGASSED PRIMARY COOLANT	STILL FEED	RECLAIMED COOLANT	HYDRO-CRACKER FEED	HYDRO-CRACKER GAS	HYDRO-CRACKER PRODUCT	RECOVERY LOOP RETURN
FLOW (lb/hr)	500	10	5	0.3 SCFH	3	2	3	2	1	15SCFH	1	5
TEMPERATURE (°F)	750	750	750	300	300	300	750	650	900	300	300	300
PRESSURE (psia)	100	100	100	15	15	15	100	15	1000	15	15	100
COMPOSITION (wt%)												
TERPHENYL	90	90	90	TRACE	90	90	80	99	47	TRACE	65	94
H. B.	10	10	10	-	10	10	20	1	47	-	35	6
H ₂	TRACE	TRACE	TRACE	100	-	-	-	-	6	100	-	-

7-13-66

Figure 67100-1. Bench Scale Hydrocracker

66-7680-08-1

HWOCR QUARTERLY REPORT

TITLE: On-Power Refueling

WBS NO: 81000

REPORTING PERIOD: April through June 1966

I. PROJECT OBJECTIVES

The overall objective of this task is the design and evaluation testing of the on-power refueling machine and its auxiliary systems under all conditions of operating environment except radiation. Testing will be performed in a logical, step-by-step manner with initial emphasis on (a) concept evaluation testing in critical interface areas between the refueling machine and reactor pressure tubes, (b) fuel and other components to be manipulated, and (c) auxiliary fuel handling systems. The results of these tests will form a basis for the prototype design or selection of major machine components. Selected components will then be fabricated and design proof-tested under environmental conditions. The final phase of the development program will be the prototype testing of the assembled refueling machine under operating conditions.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. FLEXIBLE HOSE TEST

Because of delays in the startup of TF-50 and a change in the method of hose flexing, the tests originally scheduled for completion this quarter were not carried out. Also, a teflon-lined hose is now planned for comparative testing with the flexible metal hose. The test plan and the test rig design have been modified accordingly.

B. SNOUT SEAL TEST

The test plan was revised to include consideration of out-of-parallelism between flange faces and temperature gradients between flange faces, in addition to make/break cycling.

C. FUEL CHANNEL LATCHES

A two-dimensional working model of the fuel channel latch was designed and fabricated. This model demonstrates the feasibility of the mechanical linkage design principle.

A design layout of the fuel channel latch was completed. The detail fabrication drawings are being prepared. Fabrication has been deferred pending the results of the current refueling concept studies.

D. FUEL LATCH CAN

The design layout and detail fabrication drawings of the fuel latch can were completed. Fabrication of a test model has been deferred pending the results of the current refueling concept studies.

E. MECHANICAL COUPLING TEST

Two Grayloc and two Resistoflex couplings were thermal cycled 80 times from 400 to 700°F during TF-50 Loop shakedown with no visual leakage. Coupling testing will resume after preliminary testing is completed on higher priority items.

F. CANDU END SEAL MECHANISM TEST

A preliminary test section design was completed.

G. END CLOSURE SEAL

Four radial seals and a test housing were received from the supplier. Two seals had been tested by the supplier in a water environment. The testing had included short-time thermal soaking, mild thermal cycling with 500°F water, and mechanical cycling at room temperature in an air environment.

During the above acceptance tests, one seal experienced 5 thermal cycles and 50 mechanical cycles. Post-test examination revealed that this seal had been overstressed. The seal was reground and mechanically cycled an additional 66 cycles with no change in diameter. The seal was assembled in the test housing (Figure 81000-1) and subjected to a series of tests in TF-50 (Table 81000-1). The seal surfaces are shown, after testing, on Figure 81000-2.

Figure 81000-1. Pressure Tube End Seal Test Fixture

7680-25768

TABLE 81000-1
END CLOSURE SEAL TESTS

Tests	Temperature (°F)	Time (hr)	Coolant Pressure (psig)	Coolant Flowrate (gpm)	Average Seal Leakage (cc/hr)
400°F Thermal Soak	400	64	300	3.85	0.7
Heatup from 400 to 750°F	400-750	9	300	4.8	6.3
750°F Thermal Soak	750	61	300	4.6	9.5
25 Thermal Cycles*	400-675 and reverse	20	300	1.0	3.2

*A surge of 340 cc was observed in the initial cooldown from 750 to 400°F.
Post-test inspection showed that the seal diameter had increased by 0.0045 in.

The coolant residue was removed from the test housing sealing surface, a new seal was installed and testing has resumed. Leak rates during the 400°F soak, which was in progress at the end of this report period, were considerably less than 0.1 cc/hr.

7680-25769

Figure 81000-2. End Seal After Test

H. BEARING TESTS

Ball bearing proposals have been received from a number of vendors. The proposals are based on various race and ball material combinations.

I. BALL VALVE TEST

A test section was designed for ball valve testing in the component test loops at 200 and 400°F. The ball valve has been received. The test has been temporarily postponed to permit evaluation of a new valve seat material, which was the result of the vendor's inability to furnish the original valve seat material in the sizes required for refueling machine operation.

III. NEXT REPORT PERIOD ACTIVITIES

The following activities are planned for the next report period:

- a) The end closure seal testing will continue.
- b) A test fixture will be designed and fabricated for the CANDU end seal coking test. Testing will commence during the report period.
- c) A test requirement will be written for a ball screw material compatibility test. The test fixtures will be designed and fabricated. A ball screw assembly will be procured.
- d) Test fixtures for the snout seal tests will be detailed and built.

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