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HEAVY WATER ORGANIC COOLED REACTOR

QUARTERLY TECHNICAL PROGRESS REPORT

JANUARY through MARCH 1966

AEC RESEARCH
AND DEVELOPMENT REPORT

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

ATOMICS INTERNATIONAL—COMBUSTION ENGINEERING

A Joint Venture for Heavy Water Organic Cooled Reactors

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FOR THE
U.S. ATOMIC ENERGY COMMISSION

May 15, 1966

ATOMICS INTERNATIONAL—COMBUSTION ENGINEERING

A Joint Venture for Heavy Water Organic Cooled Reactors

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HWOCR PROGRAM WORK BREAKDOWN STRUCTURE

4-1-66

*INCLUDES FOREIGN TECHNICAL REPRESENTATIVES

7680-1966 B

HWOCR Program Work Breakdown Structure

HIGHLIGHTS SUMMARY

Following are the technical highlights for the January to March 1966 quarterly report period:

1) The conceptual design of the HWO CR 500 Mwe Nuclear Demonstration Plant (AI-CE-MEMO 25) was completed and submitted to AEC.

2) The Long Range Program Plan (AI-CE-MEMO-26), a revision and updating of the original plan issued in June 1965, including detail for GFY 1967 and 1968, was published and distributed.

3) The Components Test Loop, TF-10, was placed in operation. Primary functions of TF-10 include pressure tube conceptual design investigations, conducted through pressure drop, heat balance, and structural performance data analysis.

4) The NRU U-305 experiment, in-pile since July 1965, achieved its design burnup of 10,000 Mwd/MTU, and was then removed due to an apparent cladding defect.

5) The first of the WR-1 fuel strings (951) were completed and shipped to the site for insertion during the next quarter.

6) The magnetic force welder was installed and placed in operation for development of SAP-SAP fuel rod end closures.

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PROGRAM PLAN

ACTIVITY: Demonstration Plant Engineering

ACCOUNT: 14000

I. PROJECT OBJECTIVES

The overall objective of this activity is the design and construction of an HWOCR 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 HWOCR concept. Basic performance characteristics of the HWOCR will also be verified with particular emphasis upon the following essential plant features: fuel elements, pressure tubes and pigtails, refueling machine, ca-landria 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

The conceptual design of an HWOCR Demonstration Plant was completed during this period and a report, AI-CE-Memo-25, issued. The design and construction of this plant will be the initial step in achieving the ultimate objective of the HWOCR Program; namely, the capability of an HWOCR plant to produce reliable economic power while also efficiently utilizing nuclear fuel.

Such a plant must be large enough to be of current interest to utilities and to provide proper demonstration of the concept, but not so large as to produce an unwarranted capital risk. Studies have shown that a 500-Mwe HWOCR plant will best satisfy these various criteria.

This plant, however, is faced with contradictory objectives. First, it must be a reliable plant which operates smoothly, efficiently, and with a minimum of down time. Yet at the same time it must be capable of demonstrating some of the features and basic performance characteristics mentioned above where these performance characteristics are related to the ultimate capability of this type of plant. To meet these contradictory objectives, this plant is being designed for a rated power level of 500 Mwe and a capability of ultimately producing as much as 750 Mwe. The power density and all other associated core parameters in this plant, when operating at 750 Mwe, are identical to those in the reference plant as described in Task 15000. This power level is referred to as the design power level whereas the nominal 500-Mwe power level of this plant is referred to as the rated power level.

The design output of the Demonstration Reactor Plant is 2174 Mwt. Operating with an organic coolant temperature of 750°F at the reactor outlet, the plant generates superheated steam which is delivered to the turbine at 725°F and 900 psig. A nuclear reheat steam cycle is employed, yielding a net plant efficiency of 34.6% at design conditions.

The reactor consists of 376 fuel channels, each containing a stack of five SAP-clad slightly enriched UC fuel assemblies in the pressure tube. These assemblies are arranged vertically in a calandria vessel which contains the heavy-water moderator. "Pigtail" piping connects the process tubes to the coolant-inlet and -outlet headers at the top and bottom of the reactor. Reactivity shutdown control is provided by horizontal neutron-absorbing elements which are held out of the core region by D₂O hydraulic pressure during reactor operation and are inserted by compression springs for shutdown; control of xenon and power level is provided by adjustable absorber rods supported by stainless steel tapes.

The heat generated in the heavy-water moderator is used to heat the boiler feedwater. The heavy-water inventory is maintained at a high purity level by filtration and ion-exchange facilities. The deuterium and oxygen released by decomposition are collected by an inert sweep gas system and are delivered to a recombiner for recovery as heavy water.

The reactor coolant is a mixture of terphenyls which is noncorrosive and which has several desirable properties. Among these are low cost, low vapor pressure, and low induced radioactivity. Two coolant loops are provided, each consisting of a circulating pump, evaporator, superheater, and reheater with interconnecting piping.

Coolant composition control is maintained by several process systems. Decomposition gases are removed by degasification while particulate impurities are removed by filtration and film-forming impurities are removed by adsorption on clay. High boilers, which are high-molecular-weight coolant decomposition products, are removed by coolant distillation. These are then converted to usable coolant by catalytic hydrocracking.

The steam turbine is a tandem-compound, four-flow, 1800-rpm, four-casing unit with 52-in. last stage blades, and is designed for a reheat cycle. Steam from the superheaters enters the double-flow high-pressure casing at 900 psig and 725°F. The discharge passes through the reheaters, returning to the three double-flow low-pressure casings at 152 psig and 725°F. Final exhaust to the condensers is at a pressure of 1.5 in. Hg. Feedwater is heated, using moderator heat and extraction steam from the turbine, to a final temperature of 420°F.

The reactor and moderator systems and the refueling machine are housed in a cylindrical pressure containment building. All other plant systems and components, including the steam generators, are located outdoors or are housed in conventional buildings which are appropriately grouped around the reactor building. The main coolant piping is equipped with double shut-off valves where it penetrates the reactor building.

A set of preliminary specifications for the Demonstration Plant was completed during the report period. These included an overall specification for the plant system, as well as the following basic system (end item) specifications:

- Neutron Level Control Devices
- Reactor Shutdown Cooling Subsystem
- Fuel Assembly
- Pressure Tube Assembly
- Fuel Transfer, Handling and Storage Subsystem

On-Line Refueling Subsystem
Reactor Vessel Complex
Primary Heat Transfer Subsystem
Superheater, Reheater and Steam Generator
Steam Supply Subsystem
Turbine Generator
Plant Control Subsystem
Plant Protective Subsystem
Aqueous and Gaseous Waste Disposal Subsystem
Component Cooling Subsystem
Moderator Subsystem
Coolant Conditioning Subsystem
Feedwater Conditioning Subsystem
In-Plant Electrical Supply Subsystem
Net Power Transformation and Transmission Subsystem
Structure and Site Subsystem
Reactor Shielding

Work was started on a series of design justification documents during this report period. These "trade study" reports are intended to support design decisions in critical areas of the plant. The following reports have been initiated:

Reactor Orientation (horizontal or vertical)
Refueling Approach (bi-directional or uni-directional)
Type of Core Fuel Material
Type of Fuel Element Cladding
Type of Pressure Tube Material
Fuel Discharge Burnup Level
Pressure Tube Lattice Spacing
One vs Two Refueling Machines
Type of Material for Calandria
Primary Pump Location (hot leg or cold leg)
Primary Pump Type of Drive (steam or electric)
Part-Load Control Mode
Maximum Number of Operating Failed Fuel Elements

Shutdown Cooling Method

Type of Steam Generator, Reheater, and Superheater

Organic Purification Method

Organic Pressurization Method

Moderator Cooling System (regenerative or non-regenerative)

Turbine Cycle (reheat or non-reheat)

Type of Turbine

Feedwater Heating Arrangement (number of stages, feedpump location, deareator)

Main Condenser Sizing

B. SAFEGUARDS STUDIES

Preliminary calculations of the radiological effects of the maximum credible accident (MCA) considered melting of one-fourth of the core with release to the building atmosphere of 100% of the noble gases, 50% of the halogens, and 1% of the solids. Deposition of 50% of the iodines on surfaces within the containment building was assumed. Internal recirculating air filtration was required to obtain desirable siting distances. Calculated siting distances are shown in Table 14000-1 for building leak rates of 0.1%/day and 1.0%/day.

TABLE 14000-1
SITING DISTANCES (miles)

Filtration Rate (scfm)	Filtration Efficiency (%)	Equivalent 100% Cleanup (building volumes/hr)	Building Leak Rate (%/day)			
			0.1		1.0	
			Exclusion	Low Pop.	Exclusion	Low Pop.
300,000	95	11.4	0.16	0.35	0.54	1.13
150,000	95	5.7	0.20	0.38	0.70	1.18
*150,000	*10	0.6	0.46	0.62	1.63	2.07

*Equivalent to 15,780 scfm at 95% Efficiency.

Whenever possible, studies conducted for the 1000-Mwe reference plant were extrapolated for Demonstration Plant safety analysis or engineered safeguards specifications. Also, an extensive development of five fault trees for safety analysis and protective system evaluation was accomplished. The results of these efforts were included in the Demonstration Plant conceptual design report, AI-CE-Memo-25.

A study was made of the possible consequences of water leakage into the organic coolant. The coolant degasification system is designed to remove 100 lb/hr of water when the water content of the coolant is 200 ppm. This low water concentration results in a water partial pressure of about 6 psi at operating coolant temperatures and poses no problems.

A large water leak, as from a complete steam generating unit tube failure, would introduce water into the coolant at a rate which exceeds the degasifier capacity. When the water concentration exceeds about 2000 ppm, the total vapor pressure of the coolant exceeds the pressure at the reactor pressure tube outlet. However, should the volume rate of vapor released from the coolant be within the liquid handling capacity of the coolant pressurizing system, the main coolant system pressure will not increase. Therefore, large quantities of water vapor could be generated in the reactor pressure tubes without external indications of this abnormal condition.

Two possible ways of providing such indication are described below:

- 1) Install a sampling line between the steam generating unit outlets and the degasifier tanks. Install a pressure reducer in this line and set it at a pressure just above the coolant vapor pressure. Measurements downstream of this point of a reduction in coolant density and/or the formation of voids will provide indications of an abnormally high water content and indicate the necessity for main loop isolation.

- 2) Locate the main coolant pumps in the cold legs of the loops. In this case, coolant high water content due to inleakage at the steam generating units would result in pump malfunction as the result of a reduction in net positive suction head at the pump suction (steam generating unit outlet), and require pump shutdown and main loop isolation.

C. REACTOR DESIGN

1. Demonstration Plant Conceptual Design

The reactor designed for the Demonstration Plant consists of moderator, calandria tank, shielding tanks, pressure tubes, fuel elements, and refueling machines. The pressure tubes contain the fuel element assemblies in tandem. The fuel is hyperstoichiometric uranium carbide clad in SAP, a dispersion-strengthened aluminum material. The column of fuel is cooled by an organic terphenyl mixture, Santowax OM* containing 10% high boilers (HB).

The orientation of the reactor is vertical and the pressure tubes which contain the organic coolant are at higher pressures and temperatures than the surrounding structures. The heavy-water moderator is contained in a cylindrical calandria vessel provided with external cooling circuits. Axial shielding from radiation sources within the core is obtained by providing light-water cooled compartments above and below the D₂O calandria, resulting in an integral design of the end shields and calandria. The concrete radial biological shield supports the reactor and end shields. Refueling is performed with the reactor at power by two refueling machines which are located above and below the reactor structure. The machines are supported vertically from rail structures by carriages which position them opposite the ends of any fuel channel.

The core has one-pass bidirectional coolant flow with bidirectional on-power refueling. The coolant flows in opposite directions in adjacent process tubes to maintain flow in the same direction as the fuel movement. This arrangement results in lower coolant temperatures in the higher power regions of the core thus improving the thermal performance. Process tubes are orificed to match coolant flowrates to the basic radial power distribution, resulting in a uniform coolant temperature rise across the process tubes.

The reactor is designed to produce a total fission power of 2174 Mwt, of which 2054 Mwt is transferred directly to the coolant. The organic inlet temperature is 574°F and the outlet is 750°F, with a maximum cladding surface temperature of 870°F. The core is made up of a total of 376 pressure tubes. Each pressure tube contains five fuel assemblies, each of which consists of 37 fuel rods.

*Monsanto Chemical Company trade name.

Uranium carbide (UC) is used as the fuel on the basis of its high uranium density, high thermal conductivity, and potentially stable irradiation performance at the thermal conditions used as a basis for this design. The selection of SAP as the fuel cladding material is based upon its compatibility with coolant and fuel at design conditions as well as its low thermal neutron absorption cross section and high thermal conductivity. The high thermal conductivity of the SAP permits designing with finned cladding to utilize more fully the thermal capabilities of the UC fuel.

2. Conceptual Design Followup Activities

a. Reactor Structures

Preliminary structural analyses of the integral vessel conceptual designs for (1) an Inconel-carbon steel design with top support only and (2) a stainless steel design with a shell expansion joint and top and bottom support were completed and are under review.

Moderator circulation within the calandria was studied and preliminary consideration given to using existing experimental equipment at Savannah River Laboratories for determining flow patterns.

A component specification for the shield plug was completed and those for the end fitting assembly and the bellows assembly are in preparation. The shield plug assembly drawings have been completed and a preliminary assessment made of the effect of relocating the fuel channel latches to reduce the number of shield plugs required and the total fuel motion during refueling.

b. Nuclear Analysis

An evaluation of the uncertainties related to the nuclear design techniques utilized on the Demonstration Plant Conceptual Design was completed.

A two-dimensional study of xenon oscillations as well as a study to determine the effect of various physics and design parameters on the power coefficient have been initiated.

3. Fuel Element Design

A fuel element performance digital computer program to assess more accurately the growth of uranium carbide as a function of time and temperature

has been written. The design of a 37-rod bundle utilizing six smaller rods as bearing surfaces has been laid out. Analytical studies of the thermal-hydraulic performance of this alternate fuel element design have been initiated.

4. Safety

A computer program for the evaluation of a postulated accident involving the simultaneous rupture of both the pressure tube and the calandria tube has been written.

Studies have been conducted on the analog computer to establish the effect of the positive power coefficient on reactor behavior. Preliminary analyses indicate that a simple controller would be adequate for normal operating conditions.

D. PLANT ENGINEERING

In developing the design of the Demonstration Plant, the 1000-Mwe Reference Plant preliminary conceptual design^{*} was reviewed in detail and, where possible, improvements were incorporated to reduce capital and operating costs, increase plant reliability, and ensure safe operation. Also, the additional design technology developed since completion of the Reference Plant preliminary conceptual design enabled the Demonstration Plant conceptual design to be prepared in greater detail and on a firmer basis. The principal points of departure from the older design are indicated in Table 14000-2. The basic data for the rated and the design conditions for the Demonstration Plant are listed in Table 14000-3.

The following paragraphs summarize the highlights of the Demonstration Plant conceptual design effort.

1. Main Heat Transfer System

Studies were conducted to establish the reactor coolant temperature rise and flow requirements for the plant. The studies determined the overall plant characteristics at reactor coolant temperature rises of 200 and 176°F. This work was done in conjunction with separate plant optimization studies. The analyses were based on maintaining a main coolant flow that would produce a maximum velocity of 30 ft/sec through the core and a reactor outlet temperature of 750°F.

^{*}AI-CE-MEMO-6: "Preliminary Conceptual Design of 1000-Mwe HWOCR Power Plant"

TABLE 14000-2
COMPARISON OF HWO CR REFERENCE AND DEMONSTRATION PLANTS

Item	Original Reference Plant	Current Demonstration Plant
Net plant design output (Mwe)	1076	752
Number of main coolant loops	3	2
Reactor coolant design ΔT ($^{\circ}F$)	155	176
Reactor shutdown cooling system	Water-cooled heat exchanger, connected across reactor inlet and outlet mixing tanks.	Dowtherm-cooled heat exchanger, connected across reactor inlet and outlet headers.
Moderator system	Calandria drained during emergency while using spray cooling	Calandria not drained during emergency. Condenser added to return vaporized heavy water in event of pressure tube rupture.
Turbine-generator	TC6F-52-1800 3 condensers	TC4F-52-1800 2 condensers
Feedwater heating	3 trains	2 trains
Reactor building	15 psig design pressure; steel structure with concrete shielding.	34 psig design pressure; reinforced concrete structure with steel membrane

TABLE 14000-3
DEMONSTRATION PLANT DATA FOR RATED AND
DESIGN LOAD CONDITIONS

Item	Rated	Design
Net output (Mwe)	505	752
Total fission power (Mwt)	1592	2174
Reactor outlet coolant temperature ($^{\circ}F$)	694	750
Reactor coolant ΔT ($^{\circ}F$)	129	176
Steam temperature at turbine throttle ($^{\circ}F$)	675	725
Steam pressure at turbine throttle (psig)	900	900
Reheat steam temperature at turbine ($^{\circ}F$)	675	725
Net plant efficiency (%)	31.7	34.6

After evaluating the effect of these temperature conditions on the reactor design and the main coolant system equipment, a reactor coolant temperature rise of 176°F was selected for the design load condition. The sizes of the main coolant system components were established for a reactor coolant outlet temperature of 750°F, reactor coolant inlet temperature of 574°F, and total coolant flowrate of 68.0×10^6 lb/hr through the reactor.

The important plant data for temperature rises of 200 and 176°F are compared in Table 14000-4 based on using two main coolant heat removal loops.

TABLE 14000-4
DEMONSTRATION PLANT DATA FOR REACTOR COOLANT ΔT 'S OF
200°F and 176°F (Design Load Conditions)

Reactor Coolant ΔT (°F)	200	176
Reactor thermal power to coolant (Mwt)	2054	2054
Total reactor coolant flow (lb/hr)	60.1×10^6	68.0×10^6
Reactor inlet coolant temperature (°F)	550	574
Reactor outlet coolant temperature (°F)	750	750
Total system pressure drop (psi)	300	334
Flowrate, each pump (gpm)	73,500	83,200
Shaft horsepower, each pump (hp)	15,000	18,700

Two main coolant loops were selected for the Demonstration Plant because the size and requirements of the major equipment (main coolant pumps, steam generators, main coolant valves, and piping) are nearly the same as required for the three-loop 1000-Mwe plant. Thus, the use of two loops enables the Demonstration Plant to fulfill the requirement that it demonstrate the technical feasibility of the large components required for the 1000-Mwe Reference Plant.

2. Reactor Shutdown Cooling System

The reactor shutdown cooling system was designed for a maximum of availability and reliability. To attain this, the following characteristics were designed into the system:

- a) Redundancy of mechanical components
- b) Continuous "on-line" availability under reduced flow conditions
- c) Automatic step-up of flowrate as required, without reliance on mechanical or electrical devices
- d) Normal pumping power source backed up by emergency electric power system.

The first three of the above characteristics were obtained by: (a) employing two parallel full-capacity pumps, (b) maintaining a continuous core bypass flow of a fraction (approximately 5%) of design flow during normal reactor operation, and (c) utilizing the reduction in core pressure (which accompanies loss of normal coolant flow through the core) to cause the shutdown cooling system flow to increase to design value and to be diverted through the core.

3. Coolant Process Systems

The basic coolant processing systems of the HWOCR are the degasification, adsorption, and coolant recovery systems. The design of the Demonstration Plant coolant processing systems involved principally a scale-down of the 1000-Mwe Reference Plant Systems and the incorporation of minor improvements.

A departure from the Reference Plant coolant processing provisions was the inclusion of full-flow coolant filters in the Demonstration Plant design. These filters are placed in the main heat transfer loops. The addition of these filters represents a step in the direction of more conservative design and tends to alleviate dependence on the adsorption system.

4. Moderator Systems

The D₂O moderator cooling system for the Demonstration Plant was scaled down from the 1000-Mwe Reference Plant design. The system is designed to remove 112 Mw of heat for the plant design load of 752 Mwe. This heat is transferred to boiler feedwater in the first state low-temperature heaters. Calandria inlet and outlet temperatures are 140 and 200°F, respectively. The moderator auxiliary systems for the Demonstration Plant are also scaled-down versions of the Reference Plant systems.

5. Reactor Service Systems

The basic reactor service systems are the waste gas, aqueous waste, spent fuel pool cooling, and component cooling water systems. With few exceptions, the design of these systems for the Demonstration Plant involves scaling down the 1000-Mwe Reference Plant systems and incorporating minor improvements.

A significant improvement in the waste gas system design for the Demonstration Plant over the Reference Plant was the replacement of the static gas decay tanks with series-connected continuous-flow decay tanks. These tanks can be used at high pressure (400 psig) or at a reduced pressure. Calculations indicated that the reactor could operate with up to 50 failed fuel pins without requiring pressurization of these tanks. However, by allowing the pressure in the storage tanks to build up to 400 psig and passing the waste gas through the tanks at this pressure, a decay period of about 24 hr is obtained. By this continuous flow-decay process, the number of failed fuel pins which can be permitted in the reactor without exceeding allowable fission product concentration in the exhaust gas is increased by an order of magnitude.

6. Turbine, Main Steam, and Feedwater Systems

For the Demonstration Plant conceptual design, the same plant steam and feedwater flow arrangement selected for the 1000-Mwe Reference Plant was used, with the following modifications:

- a) The number of turbine ends was reduced from 6 to 4.
- b) The valve leak-off was rerouted since it was no longer needed to supply steam to the low pressure gland seals.
- c) The reactor coolant ΔT and weight flowrate were slightly modified.

With these modifications, it was possible to use essentially the same computational sequence and techniques as were used in analyzing the Reference Plant. The resultant design load heat balance for the turbine plant is shown in Figure 14000-1.

The part-load techniques suggested in the General Electric 1965 Nuclear Method were used to determine the plant operating conditions at the rated load. The 1965 Nuclear Method assumes that the turbine is so large that throttling control is the only method that can be used with these units. This method of

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Figure 14000-1. HWOCR Steam and Feedwater System Heat Balance - Design Load

control, which reduces pressure without obtaining energy from the steam as is the case with admission control in a control stage, adversely affects the part-load performance of the plant to a small degree. However, in this study it was assumed that any control problems or peculiar design requirements that are imposed by any of the assumptions or operating conditions could be resolved during the plant design phase. The reactor temperature and the superheater and reheater outlet temperatures were assumed to vary as shown in Figure 14000-2. The assumption of superheater and reheater temperature was subsequently verified by evaluating the part-load heat transfer characteristics of the evaporator, superheater, and reheater. With these assumptions, the heat balance for the plant rated load condition was completed (Figure 14000-3).

Figure 14000-2. HWOCR Reactor, Superheater, and Reheater Temperature as a Function of Reactor Thermal Load

III. NEXT REPORT PERIOD ACTIVITIES

Engineering effort on the HWOCR Demonstration Plant will consist of conceptual design followup, pre-Title I activities. Emphasis will be placed on completion of trade study documentation and revision of existing end item specifications.

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Figure 14000-3. HWOCR Heat Balance - Rated Load

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PROGRAM PLAN	
ACTIVITY: 1000-Mwe Plant Engineering	ACCOUNT: 15000

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 provide:

- a) Successful demonstration of economic operation of the HWO CR concept in a plant of this size
- b) Confirmation that the essential HWO CR features demonstrated by the 500-Mwe plant may be scaled up successfully to larger plants
- c) Cost and design data which will be used in defining large dual-purpose power and desalination plants.

An additional objective is the identification of areas requiring research and development effort.

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

A. GENERAL STUDIES

The maximum credible accident (MCA) was established as the rupture of an inlet header to one-fourth of the core. The following events could ensue from such an accident:

- 1) Organic coolant spills into coolant header cell.
- 2) Because of loss of coolant and consequent temperature rise, the fuel melts and pressure and calandria tubes develop leaks in the affected quarter sector of the core.
- 3) Organic coolant and molten UC penetrate into calandria.
- 4) UC - D₂O reaction occurs in calandria.
- 5) D₂O steam is generated in calandria by hot organic.

Design features incorporated in the HWO CR would counteract the above potential consequences of an MCA in the following manner:

a) Spilled organic is collected, cooled, and returned to the coolant system by the reactor shutdown cooling system.

b) Vapors in the calandria rise to an overflow tank, then to a condenser. The condensate is returned to the calandria cooling sprays.

c) An overpressure in the calandria causes a rupture disc located high on the calandria side walls to rupture, relieving vapors to the cell surrounding the calandria.

d) In the event of an overpressure in the cell surrounding the calandria, a rupture disc ruptures allowing the vapors in the cell to follow the same path as in (b) above.

e) In the event of a larger overpressure in the calandria or surrounding cell, a rupture disc ruptures, releasing the vapors to the heavy-water cooling areas. The vapors then vent via a trap to the main reactor building galleries, where they are condensed by a water spray or fog system. An emergency recirculating heating and ventilating filtration system reduces fission product contamination within the containment atmosphere.

Additional design features include the isolation of all organic-containing components from the main containment region, with the pressure relief of the cells containing these components routed to the main containment region. Condensing water spray or fog (see (b) above) does not enter into the organic component cells. Thus, this measure prevents the generation of high pressure by the condensing water spray or fog due to contact of the water with hot coolant leakage and the consequent generation of steam. The organic component cells are provided with non-steam-generating fire smothering systems.

Preliminary, highly conservative analyses have indicated that the MCA results in less than 30 psig pressure in the containment and that a 0.1%/day leak rate (at peak pressure) building is acceptable for the AEC reference site. A more exacting analysis is expected to reduce the peak pressure to 20 psig or less.

An analysis of the tritium hazards associated with the MCA was made with the assumption that, at most, all of the D_2O would be heated to steam by contact with the hot organic coolant. Under these conditions, a model was formulated which takes credit for D_2O condensation on the inner surfaces of the containment building but not for condensation by water spray or fog. The results indicated that both the 2-hr and 30-day doses at a half-mile radius are negligible.

B. ENGINEERING

Due to the concentration of effort on the conceptual design of the HWOCR Demonstration Plant (WBS No. 14000), engineering effort on the 1000-Mwe plant was reduced during this report period.

The following progress was made with respect to optimization computer codes:

1. Primary Loop Code

A subroutine for reheat steam applications was added to provide for optimization of the coolant split between the superheater and reheater and to provide design data for the reheater if desired. In addition, a modified version of the code was developed for use in desalination-power dual-purpose applications.

2. Steam Cycle Code

The capability of this code to work out a case in which all of the regenerative heating stage pressures are defined was confirmed. In view of the desirability of having the additional capability to determine the approximate pressures in the preliminary cycle studies, a modification was incorporated to allow the code to work either case at the user's option.

Numerous studies and cost analyses were completed to evaluate types of containment and containment construction features. Containment schemes considered include the following:

- a) Containment of reactor only
- b) Containment of reactor and primary coolant loops
- c) Low-pressure containment (less than 5 psig) of reactor, primary coolant loops, and auxiliary systems.

The types of containment construction considered includes the following:

- 1) Steel plate lined reinforced concrete construction
- 2) Steel pressure vessel construction.

The following conclusions were obtained from these studies and analyses:

- a) Pressure containment of the reactor only is less costly than containment of the reactor and primary coolant loops, and less costly than low-pressure containment of the reactor, primary coolant loops, and auxiliary systems.

b) Steel plate lined reinforced concrete construction is less costly than a steel pressure vessel for both the reactor only and the reactor plus primary coolant loops containment schemes.

III. NEXT REPORT PERIOD ACTIVITIES

During the next report period a revised version of the 1000-Mwe Conceptual Design Report (AI-CE-Memo-6) will be prepared in summary form to bring the 1000-Mwe plant to the same stage of design as the 500-Mwe Demonstration Plant. Continued effort will also be applied to the following three areas:

- a) Dynamic analyses of plant operation
- b) Plant optimization studies and development of the optimization computer codes
- c) Safety studies, with emphasis on containment requirements and designs.

PROGRAM PLAN	
ACTIVITY: 10,000-Mwt Dual-Purpose Plant Engineering and Development	ACCOUNT: 16000

I. PROJECT OBJECTIVES

The dual-purpose 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. This makes it feasible for this plant to operate at lower coolant temperature than the power-only plant. Second, in many areas of the country, the requirements for water are such that very large dual-purpose reactors of the order of 8,000 to 10,000 Mwt can be used to evaporate a billion gallons per day while producing 1000 to 2000 Mwe as a by-product. 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 rates for water plants range 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 light of the above considerations, the objectives of this effort are to:

- a) Determine the feasibility of a 10,000-Mwt HWOCR plant.
- b) Determine near-optimum parameters for a large HWOCR dual-purpose plant, and examine the effect of economic ground rules on these parameters.
- c) Perform a conceptual design of an "optimum" of 10,000-Mwt dual-purpose plant.
- d) Define areas and show incentives for incremental research and development for desalting applications.

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

Phase A of the dual-purpose work, which was to demonstrate the practicability of extrapolating the reference power-only plant to the 10,000-Mwt size, was completed on March 31, 1966. The conclusion was reached during Phase A that it is not only practicable but economically desirable to design and build a HWO CR nuclear power plant in the 10,000-Mwt range. The Phase A study uncovered no technical problems that were peculiar to large-size extrapolations which could not be solved by standard engineering design techniques.

The 10,000-Mwt plant was extrapolated from the 1000-Mwe reference design power-only plant with modifications reflecting the engineering improvements which were developed in the course of doing the 500-Mwe demonstration plant conceptual design. These modifications were reflected in changing the reactor inlet temperature to 574°F, the reactor temperature rise to 176°F, the primary piping coolant velocity to 22 ft/sec, and the containment construction concept to a steel-lined reinforced concrete structure.

The reactor was extrapolated by figuratively splitting the circular reference plant design and inserting therein modules (which were reported upon in the last Quarterly Report) until the reactor provided the desired thermal power rating. A total of 1728 pressure tubes with 452 MTU were required.

The calandria tank design was changed to provide better design of the flat sides which developed in the course of extending the core length. Inconel was selected as the calandria tank material instead of stainless steel because of several economic and structural considerations. The thermal expansion coefficient of Inconel more closely resembles the other materials used in the reactor construction. Use of a stainless steel calandria would have required the inclusion of an expansion joint for adequate allowance for differential thermal expansion. It was determined that a savings of \$83,000 would be realized in the use of Inconel over stainless steel in material costs alone, with no consideration of the additional expense which would be incurred if a stainless steel expansion joint were utilized.

Extrapolation of this size caused no change in the refueling concept. Location of the service area for the refueling machine was reoriented 90 degrees to minimize the diameter of the containment vessel.

The module concept which was described in the Quarterly Report of October through December 1965 (AI-CE-17) was utilized for the extrapolation. Consequently, no basic change in fuel design, pressure tube design, or pigtail arrangement was experienced in the extrapolation studies.

Control of the reactor was investigated by a two-directional xenon stability code. It was determined that the period of xenon fluctuation was of such magnitude that normal control for power level selection was more than adequate to keep the reactor in balance. Further work will be done in detail design to determine the exact number and location of control points for this reactor.

Four basic headers are utilized along either side of the reactor for the purpose of manifolding the primary coolant flow into two inlet and outlet mix tanks. Eight heat transfer loops (four from each inlet and outlet header) are utilized to transfer the heat to the secondary coolant. Eight loops were selected since equipment sizes required for this plant size were similar to those which will have been demonstrated in the demonstration plant. Therefore, no effect is realized on practicability by the consideration of eight loops. Cost savings will be realized by the duplication of auxiliary systems for the components within the eight loops. Investigations were conducted to determine the practicability of extrapolating the size of heat transfer components so that four heat transfer loops would be utilized instead of eight. Assurance was received from suppliers of heat exchangers, main coolant pumps, piping, and valves, that extrapolations of this magnitude will be within the state-of-the-art in 1975 with no additional development required for this program. Testing the hydraulic characteristics of such large pumps would be required as would proof testing of the specific design. The trade-offs required for this evaluation (number of loops) will be conducted during the optimization studies.

Safety aspects of the 10,000-Mwt plant have been analyzed with emphasis on possible changes and hazards resulting from extrapolation of the 1000-Mwe plant. In general, the safety considerations for the 10,000-Mwt plant are identical to those for the 1000-Mwe plant when located at the AEC reference site, and additional safeguards are not required. However, the size or capacity of engineering safeguards frequently requires enlargement.

The number of defective fuel pins consistent with the MPC and off-site integrated dose criteria of 10 CFR 20 was determined. Assuming that degasification processes are 100% effective in removing xenon and krypton from the coolant and 10% effective in removing iodine, approximately 400 defective fuel pins can be tolerated.

Containment of the extrapolated plant is identical in its scope to that utilized in the power-only plants which were used as the design reference. Specifically, the containment has within its boundaries the reactor and refueling machines. The steam generators and pumps are located outside the containment boundaries and are connected to the inside of the containment by piping which is capable of isolation by means of double valving (in and outside of the containment). This concept of containment is no less justifiable for the 10,000-Mwt plant than it is for the reference plant.

An upper limit type of MCA similar to that considered in the 1000-Mwe plant was postulated and its consequences estimated in order to determine any significant effects on plant feasibility from a safety standpoint. The results indicate that for a location at the AEC reference site, no change in basic safety requirements were necessary. The MCA was defined to be a single gross rupture in any header, feeder, or coolant line.

III. NEXT REPORT PERIOD ACTIVITIES

During the next period, Phase B of the overall HWOCR Dual-Purpose Plant Engineering and Development effort will be initiated. The objective of Phase B is the preliminary economic optimization of a 3000-Mwt HWOCR plant (nuclear island only) suitable for dual-purpose applications. In this phase, emphasis will be placed on investigation of the effects of core materials with lower temperature limits on plant design and optimization; on the evaluation of incentives for the development of materials for lower temperature design conditions; and on the identification of points of step change due to the temperature limitations of materials.

PROGRAM PLAN**ACTIVITY:** Dual Stream Reactor Concept**ACCOUNT:** 16000A**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. The 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 HWOCR and to determine its major features.

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

Activity on the dual stream reactor concept this quarter has centered around the heat transfer problems and the analysis of possible fuel elements as well as the preliminary fixing of nuclear parameters. This is preparatory to dynamical feasibility considerations. There has also been progress this quarter in the development of methods to treat some specific problems in heat transfer and some general problems involving the geometric and spectrum dependence of η and the thermal utilization.

A. FUEL ELEMENT DESCRIPTION

The dual stream concept basically involves appreciable heat transfer to the D_2O as well as the organic. This necessitates the use of UO_2 rather than UC since the good heat conduction path between the D_2O and U (i. e. , only clad, no separating insulating gas gap) precludes the use of UC because of a possible resultant water carbide reaction in the event of clad failure.

In order to secure another basic advantage of the dual stream concept, the fuel must be essentially in direct thermal contact with the D_2O as well as the organic to allow for decay heat removal by either stream. This and the above consideration lead directly to a cylindrical fuel element of UO_2 clad with Zr-4 immersed in D_2O and pierced by an array of Zr-4 (or SAP) tubes carrying the organic coolant.

With U-shaped organic coolant tubes within each fuel element, organic connections at the top of the core only are necessary, and some measure of reduction of the effect of power peaking within an element may be achieved.

The density and conductivity of fuel will probably be fixed as those appropriate for vibratory compacted UO_2 . Without going into details of the mechanical arrangement, it appears that the dual stream concept may be in conflict with on-line refueling. However, the final analysis of fabrication problems and consideration of mechanical details, stress, and similar problems are outside the scope of this project.

B. PRELIMINARY HEAT TRANSFER SCOPING

An idealized annular fuel element consisting of concentric annuli from the center outward composed of (Org) (Clad 1) (UO_2) (Clad 1) (Org) (Clad 1) (UO_2) (Clad 1) ... (Clad 1) (UO_2) (Clad 1) (D_2O) has been analyzed. Some of the results of this model are as follows:

- 1) There is enough flexibility to provide any ratio of heat to D_2O to heat to organic (Figure 16000A-1). An increase in the heat-to-organic stream may be achieved by adding interior fuel annuli.

- 2) At power densities of the order of 20-60 kw/kg, equivalent annuli of fuel should be less than 0.6 to 0.25 in. thick to avoid central melting of the UO_2 .

- 3) Similar limits to (2) above are found necessary to avoid organic temperatures greater than 850°F.

- 4) Heat flux through inner surfaces is somewhat sensitive to curvature.

- 5) Maximum heat flux on outer surfaces and maximum fuel temperature are not sensitive to radius.

An additional model for a fuel element pierced by a regular array of 0.2 in. diameter tubes was considered. This model was predicated on a pressure drop along an organic tube of 150 psi in 24 ft. The heat flow per unit area of interface

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Figure 16000A-1. Heat Transfer in a Single Annulus With Organic Inside

was assumed to be the same to both the organic and D₂O streams. These results and those for an annular calculation are shown in Figure 16000A-2. This figure indicates that fairly low volume fractions of organic are involved.

C. PRELIMINARY NUCLEAR SCOPING

Six cells were studied for k_{eff} vs burnup. The fuel element was a model formed of concentric annuli of (UO₂) (Org and Clad) (UO₂) (Org and Clad) ... (UO₂) (Clad) (D₂O) (Reflecting Boundary). The basic geometry is described in Table 16000A-1. Cases A and B differ mainly in the ratio of organic volume to UO₂ volume. Cases C and D differ mainly in the volume ratio of organic to UO₂. Cases A and B differ from cases C and D in the radius of the fuel element. Case E is similar to B except for the size of cell in that there is more D₂O around the element. They have the same relative volume of clad, UO₂, and organic. Case F bears the same relationship to case D as does E to B. Further the volume ratio of D₂O to UO₂ is 10.8 for cases A, B, C, and D; it is 21.60 for cases E and F. The enrichment is 1.25 wt %. The hydrogen content in the D₂O was taken to be 1/4%. Other parameters are essentially the same as those used in the thermal studies above.

Figure 16000A-2. Organic Volume Fraction in Fuel Element

The nuclear parameters were fixed using the codes, CEPTR, THERMOS, CHAFE, FORM, and CRUMB. Some of the results are shown in Table 16000A-2 and Table 16000A-3. Some of the results may be categorized as follows:

- 1) Effect of fuel rod size is essentially a reduction of $(\sigma a)_{\text{eff}}$ in larger rods because of spectral hardening.
- 2) Increasing the amount of D_2O seems to influence the results mainly through the increased absorptions due to the greater amount of H_2O brought in as well as more thoroughly thermalizing the spectrum.
- 3) The fast fission effect is large and increases with the increased size of fuel lump.
- 4) Pu^{239} production decreases with lumping and increasing amounts of D_2O .

D. METHODS DEVELOPMENT

A method based on escape probabilities is being developed to account for geometrical effects of thermal η and f values. It may be combined with a simplified representation of thermal energy exchanges which results in an energy treatment which can extend naturally to a fairly high thermal cutoff. It is being tested against "exact" transport calculations for slab lattices.

E. EVALUATION

The effort to date has demonstrated that, in principle, a fuel element can be designed to accommodate any ratio of relative heat transfer to the two streams. By its nature, the multitube fuel element in feasible sizes will result in a majority of the heat going to the organic. The nuclear analysis of the multitube fuel element has shown that a higher reactivity can be obtained for any given enrichment and D_2O to uranium ratio than in the reference HWOCR design. All or part of this reactivity will be needed to compensate for burnup since there is presently no reasonable way for on-line refueling in this concept.

TABLE 16000A-1
FUEL ELEMENT GEOMETRY PARAMETERS

Case	No. of Coolant Tubes	Tube Pitch (cm)	No. of Equivalent Annuli	Outer Rod Radius (cm)	D ₂ O Radius (cm)	Volume Fractions			
						UO ₂	Org.	Clad	D ₂ O
A	7	0.968	2	2.03	6.30	0.0830	0.00921	0.0118	0.896
B	19	0.683	3	2.03	5.60	0.0804	0.0317	0.197	0.868
C	127	0.968	7	7.12	21.4	0.0824	0.0145	0.0137	0.890
D	271	0.683	10	7.12	18.9	0.0795	0.0395	0.0224	0.859
E	19	0.683	3	2.03	7.65	0.0430	0.0170	0.0105	0.929
F	271	0.683	10	7.12	25.8	0.0428	0.0212	0.0120	0.924

TABLE 16000A-2
 k_{eff} AS A FUNCTION OF BURNUP

Mwd/MTU	A	B	C	D	E	F
0	1.2570	1.2131	1.3117	1.2587	1.3247	1.3381
50	1.2140	1.1725	1.2674	1.2172	1.2796	1.2938
1,000	1.1995	1.1613	1.2525	1.2053	1.2613	1.2767
10,000	1.0401	1.0167	1.0971	1.0590	1.0744	1.0979
20,000	0.9178	0.9044	0.9643	0.9334	0.8967	0.9168

TABLE 16000A-3
 NUCLEAR PARAMETERS OF FUEL ELEMENTS

	$k_{\infty}(\text{BOL})$	$(\eta f)_{th}$	P_3	E_1	$\text{Pu}^{239} \text{ at } 20 \text{ Mwd/kg U (gm)}$
A	1.257	1.533	0.7820	1.034	6.22
B	1.213	1.506	0.7720	1.027	6.55
C	1.312	1.504	0.8119	1.094	5.62
D	1.259	1.481	0.8010	1.070	5.90
E	1.325	1.500	0.8650	1.024	5.45
F	1.338	1.458	0.8821	1.070	5.05

III. NEXT REPORT PERIOD ACTIVITIES

The effort on the dual stream will continue with detailed thermal and hydraulic analysis of the two streams and final conceptual design of the fuel element. In particular, computer codes will be sought or developed which will describe the heat transfer to the organic containing tubes and to boiling D_2O on the outside of the fuel element with the completed fuel element design. The nuclear analysis will concentrate on determining enrichments and material inventories and, in particular, reactivity coefficients for the different designs.

PROGRAM PLAN	
ACTIVITY: Reactor Physics	ACCOUNT: 13200

I. PROJECT OBJECTIVES

The objectives of the reactor physics task are to provide nuclear design methods incorporating digital computer codes, nuclear cross section data and analysis 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. RESONANCE ABSORPTION

To test the cylindrical representation of rod cluster geometries for calculating the resonance captures in U-238, first collision probabilities for a flat source distribution in fuel were calculated for explicit clustered rods and for annular representations, in one energy group, by means of the Monte Carlo code TRAM.⁽¹⁾ Collision probabilities in annular and cluster representations were within a few percent of agreement for a range of absorption cross sections. REPLICA⁽²⁾ calculations show that the annular representation yields resolved resonance integrals that can vary from 1.4% lower to 5% higher for various 19-rod clusters. Equivalent single rod average radii obtained from the explicit TRAM calculations and used with ZUT⁽³⁾ resolved resonances yield somewhat lower resonance integrals by from 3% to 5% than obtained with REPLICA for the explicit cluster geometry.

B. THERMAL SPECTRUM

Comparisons of thermal scattering kernels for the HWO CR system were continued. The SUMMIT code,⁽⁴⁾ which generates the bound Parks' kernel for carbon in graphite and can be employed for carbon in the organic coolant, was made operational.

Comparisons between the calculated scattering cross sections for hydrogen in Santowax (Brown-St. John kernel)⁽⁵⁾, as a function of energy, and experimental for benzene^(6,7) showed deviations at lower energies. Some other highlights obtained from this analysis are:

1) The calculated and experimental values of σ_s/σ_{free} (the ratio of the hydrogen bound scattering cross section to the hydrogen free scattering cross section) versus energy were found to be in fair agreement. The calculated values resulted from benzene and polyethylene⁽⁸⁾ kernels generated by GAKER⁽⁹⁾ and the experimental data published by Ardente⁽⁷⁾ were for benzene and diphenyl.

2) With the number densities held constant and only the temperature changed, the temperature coefficient, $\partial \ln \eta_f / \partial T$, is relatively insensitive to the scattering kernel representation used for the D₂O moderator. However, changing the scattering kernel representation for hydrogen in the organic coolant from the Brown-St. John kernel to the bound atom scattering kernel for polyethylene reduced $\partial \ln \eta_f / \partial T$ by 12%.

3) Placing a scattering ring around the cluster unit cell, for various scattering kernel combinations, indicated an average reduction of Σ_a by 0.09%, $\xi \Sigma_f$ by 0.1%, η_f by 0.01% and σ_a (U-238) by 0.08% while the unit $1/v$ absorption cross section integrated over the unit cell increased by 0.03%.

Transport correcting the Honeck-Nelkin kernels for D₂O and the organic, with scattering ring, increased Σ_a by 3.29%, $\xi \Sigma_f$ by 3.22% while η_f decreased by 0.07% as compared to the Brown-St. John kernels for all the isotopes without scattering ring. Comparisons of the Honeck-Nelkin transport corrected kernels to the isotropic Honeck-Nelkin scattering kernels increased η_f by 0.009%.

C. ANALYSIS OF EXPERIMENTS

Several 19-rod UO₂ cluster experiments were analyzed with the multigroup HAMMER code which was recently developed at Brookhaven National Laboratory and Savannah River Laboratory for calculating neutronic properties. The HAMMER code⁽¹⁰⁾ performs a multigroup calculation of thermal ($E \leq 0.625$ eV) and epithermal ($E > 0.625$ eV) flux distributions based on integral transport

theory, followed by a multigroup Fourier transform leakage spectrum correction based on homogenized cell parameters. Resonance self-shielding calculations for the resolved and unresolved resonance energy regions, based on the Chernick-Nordheim theory,⁽³⁾ are incorporated in the code together with a one-dimensional reactor core diffusion theory calculation. The purpose of the analysis is to test the applicability of the HAMMER code for HWO CR design calculations.

The ZED-2 experiments analyzed consist of 19-rod clusters fueled with neutral UO_2 encased in Zircaloy-2.⁽¹¹⁾ The calandria tube used in these experiments was of aluminum while both aluminum and Zircaloy-2 were used as pressure tube material. The coolant was either D_2O , void, or HB-40.

The eigenvalues calculated by the HAMMER code for the 19-rod ZED-2 experiments are indicated in Table 13200-1. The eigenvalues for the D_2O coolant cases show a systematic trend with increasing lattice pitch which may be due to uncertainties in the experimental bucklings or in the HAMMER theoretical model and cross sections. It has been assumed in the HAMMER calculation that the fuel rods in the cluster can be synthesized into cylindrical fuel rings.

D. BIDIRECTIONAL BURNUP

AI-CE-Memo-14, describing the results of a study on bidirectional burnup and verifying the adequacy of the relatively simple model used for survey calculations, was released for printing.

E. RBU MONTE CARLO CALCULATIONS

Some changes and additions have been made in the summary, presented in the July-September 1965 quarterly report, of the results of the RBU calculation. The value of ρ obtained for the actual geometry and the homogenized geometry is 0.76 and 0.78; the fraction of neutrons produced by fissions above 0.821 Mev is 0.040 and 0.0377 respectively, and the fraction of neutron slowing down from above 0.821 Mev is 0.56 and 0.53.

TABLE 13200-1
EIGENVALUES FOR 19-ROD UO₂ CLUSTERS

Coolant	Case	Triangular Lattice Pitch (cm)	Exp. B^2_{Al} (m ⁻²)	k_{eff}^*	Comment
D ₂ O	ZED-2-D ₂ O-19-1	18	1.407 ± 0.047	0.9808	† Critical Systems
	-4	21	3.479 ± 0.029	0.9867	
	-6	24	4.065 ± 0.014	0.9919	
	-8	28	3.952 ± 0.013	0.9954	
	-12	36	2.754 ± 0.012	1.0055	
Air	ZED-2-AIR-19-1	18	0.718 ± 0.086	1.0017	
	-3	21	3.351 ± 0.055	0.9935	
	-5	24	4.115 ± 0.036	0.9954	
	-7	28	4.070 ± 0.018	0.9994	
	-11	36	2.970 ± 0.006	1.0077	
HB-40	ZED-2-ORG-19-1	18	-0.365 ± 0.062	0.9916	
	-2	21	1.214 ± 0.035	0.9897	
	-3	24	1.568 ± 0.019	0.9918	
	-4	28	1.528 ± 0.012	0.9916	
			B^2_{Zr}		
D ₂ O	ZED-2	18	1.19 ± 0.07	0.9880	§
		21	3.30 ± 0.05	0.9952	
		24	3.91 ± 0.05	1.0009	
		28	3.80 ± 0.03	1.0054	
		36	2.64 ± 0.03	1.0154	

*0.713 wt % U²³⁵O₂ in UO₂

†Zr-2 clad UO₂ fuel, Al pressure tube and Al calandria tube

§Zr-2 clad UO₂ fuel, Zr-2 pressure tube and Al calandria tube

F. DEVELOPMENT OF THE MONTE CARLO CODE - SAFE

The adaptation of SAFE to the HWOCR cell problem requires a fairly extensive tabulation of fluxes, reaction rates, spectra, etc. At the same time, those features of SAFE enabling it to solve three-dimensional, finite (non-cell) configurations with partial flooding are not needed for HWOCR calculations. During the past quarter the SAFE code was therefore reviewed and a version was prepared in which storage for the above-mentioned tabulation was made available by stripping SAFE of those features not applicable to the HWOCR program.

Work is proceeding on SAFE modifications which will enable HWOCR problems to be solved using the SAFE code. These modifications include extension of the geometric and monitoring capability, and capability of handling fuel rods of varying size and composition and increased number of isotopes.

G. REACTOR BURNUP PERTURBATION

A perturbation theory for calculations of reactor burnup has been developed for the purpose of estimating the uncertainties in burnup properties arising from uncertainties in cross sections and in calculational methods. The burnup property investigated is the limiting lifetime, beyond which criticality cannot be achieved. The principal difficulty is to define an adjoint for the non-linear problem which arises because the reaction rates that determine the changing isotopic concentrations contain products of these concentrations by the neutron fluxes. This difficulty is met by basing the treatment on the linear equations that approximately determine small departures of the solution from a known solution. The perturbation of the limiting lifetime by a small change in a cross section from the value used in calculating the burnup history is then expressed in terms of that known history and the functions adjoint to the above-mentioned small departures. The departures themselves need not be calculated, but only their adjoints, which are also determined by linear equations. A FORTRAN-IV program has been written to implement the theory in the case of a simple two-group point-burnup model, and some representative calculations have been made with this program.

H. FISSION PRODUCTS

Preliminary calculations with CINDER,⁽¹²⁾ using different yields for the fissile isotopes and explicit U-236, indicate increases for fission product poisoning of about 8% over presently used cross sections in the range of 10,000 Mwd/T to 20,000 Mwd/T. Fast cross sections for the extended Nd-176 to Sm-152 chain have been obtained and are being tested.

I. ADDITIONAL ACTIVITIES

The one-dimensional transport code DTF-2⁽³⁾ was made operational and should prove useful for control rod studies. New high energy cross sections for the plutonium isotopes were obtained.

III. NEXT REPORT PERIOD ACTIVITIES

U-238 resonance absorption measurements from Ispra will be analyzed and further evaluations will be made of resonance calculations. Studies of fission product chains will be continued. The suitability of HAMMER to HWOCR will continue to be evaluated and modifications will be initiated. Modifications to the Monte Carlo Code SAFE will continue. Additional critical experiments including UC fuel will be analyzed.

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PROGRAM PLAN

ACTIVITY: Pressure Tube Development

ACCOUNT: 21000

I. PROJECT OBJECTIVES

The overall project objectives are development of a reliable pressure tube and associated components for reactor use. Objectives for this quarter, in support of the overall objectives, were to:

- a) Proceed with rupture testing of SAP pressure tube materials.
- b) Complete stainless steel test section (TS-10-1A) for TF-10 operation.
- c) Fabricate the SAP 895 section (TS-10-2), install, and commence testing.
- d) Fabricate the SAP 895 hot pressed transition joint test section (TS 10-4).

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) Shakedown of the single-channel test facility (TF-10) with a stainless steel pressure tube test section (TS-10-1A) was completed.
- 2) Fabrication of the SAP 895 inner-sleeve rolled-joint test section (T.S. 10-2) was completed and the test program initiated.
- 3) The pressure tube burst test program was initiated.
- 4) Detailed design for one ZR-4 and two SAP 895 pressure tube test sections for testing in TF-20 was completed.
- 5) Major fabrication work on the five-channel test loop (TF-20) was initiated.
- 6) A TV borescope system was purchased for pressure tube inspection.
- 7) An order was placed for four B. F. Goodrich Radial Seals and one test section housing for use in end seal development.

B. PRESSURE TUBE DEVELOPMENT

Major emphasis was placed on finalizing test section designs and test planning during this quarter and ensuring that these reflected Demonstration Plant requirements. Two SAP 895 pressure tube test section designs, TS 20-4 and TS 20-1A, were completed. TS 20-1A contains sandwich rolled transition joints and a bellows gas annulus seal and is depicted in Figure 21000-1. TS 20-4 contains hot pressed/flash welded transition joints, a bellows gas annulus seal, and an integral shield-calandria tank design (Figure 21000-2). One additional 10% SAP test section (TS 20-5) is being designed and is expected to be ready by November 1966. Future test sections for testing in TF-20 consisting of 7% SAP alloy tubes are now being investigated. Table 21000-1 summarizes current plans and status as related to the first developmental pressure tube assemblies to be tested.

TABLE 21000-1
FIRST DEVELOPMENTAL PRESSURE TUBE ASSEMBLY
TEST SECTIONS

Designation No.	Pressure Tube	Basic Variation of Assembly	Status
TS 20-1A	SAP 895, Sandwich Rolled Joint	Bolt hold down. Separate shield-calandria tanks. Gas seals and bellows expansion seal. Removable bearing sleeves. Out of shield latch steps.	Basic design complete. Design modifications to be completed by 4-21-66
TS 20-4	SAP 895, Hot Pressed Joint	Spanner nut hold down. Integral shield-calandria tanks. Gas seals and bellows expansion seal. Adjustable torque sleeve. Cr_2O_3 to stainless steel bearing surfaces. Out of shield latch steps.	Design complete. Procurement in progress.
TS 20-5	SAP 895, Hot Pressed Joint	Integral shield-calandria tank. Bellows expansion seal. Provisions for reference fuel support system and end seal mechanism. Fuel bundle and pressure tube vibration sensors. Flexible hose piping connections.	Design layout near completion. Awaiting reactor design input. Latch steps not defined.
TS 20-6	SAP (7%)	Similar to TS 20-4. Provisions for reference internal components. Flexible hose piping connections.	Not started.
TS 20-7	SAP (7%)	Similar to TS 20-4. Provisions for reference internal components.	Not started.

SECTION A-A
ENLARGED SECTION B-B
ENLARGED SECTION C-C
ENLARGED SECTION D-D
ENLARGED SECTION E-E
ENLARGED SECTION F-F
ENLARGEDNOTE: MOCKUP FUEL AND
SHIELD SPACERS
NOT SHOWN

SECTION A-A
ENLARGED

SECTION B-B
ENLARGED

SECTION C-C
ENLARGED

SECTION D-D
ENLARGED

SECTION E-E
ENLARGED

SECTION F-F
ENLARGED

NOTE: MOCKUP FUEL AND
SHIELD SPACERS
NOT SHOWN

Test plans for the above test sections are being prepared. Final test plans for the first series of test sections to be tested in TF-20 will be available during the next report period. Quotes for fabricating the 20-1A test section were requested from outside vendors; only one vendor replied, with an 18 to 20-week delivery date. CE facilities for fabricating the test sections are being investigated. An order for fabricating TS 20-4 will be placed during the next quarter.

The seven pressure tube extrusions received from NMI during the last report period were allocated for tube rupture specimens, pressure tube test sections, rolled joint sections and flash welding and hot pressing evaluations.

C. HB-40 LOOP (TF-1)

TF-1 has been connected to the TF-50 primary pumps and will be used as a secondary cooling loop for the pumps. TF-1 will not be reported in a separate section after this quarter.

D. SINGLE-CHANNEL TEST FACILITY (TF-10)

The TS 10-1A tests were completed on January 13, 1966. The test report was written and will be issued during the early part of the next report period.

The principal results of the TS 10-1A tests were:

a) Pressure tube heat loss was found to depend only on organic temperature and moderator inlet temperature. CO₂ flow, organic flow, and moderator flow were found to have no effect within experimental accuracy. The following table presents the overall heat balance data for TS 10-1A:

Primary Coolant Temperature (average °F)	Moderator H ₂ O Inlet Temperature (average °F)	Pressure Tube Heat Loss (Btu/hr)
450	50	20,436
550	51	29,044
650	52	39,137
650	116	34,194

b) No pressure tube vibration was detected and pressure tube displacement within the calandria was found to be dependent on temperature and piping loads only; flow had no effect. At no time did the pressure tube make contact with the calandria tube.

c) Usable strain gauge data were not obtained because the drift due to temperature was several orders of magnitude greater than the actual strain signals.

d) The ΔP vs flow studies indicated a higher pressure drop than had been expected. The results are shown in Figure 21000-3. The reference velocity of 30 ft/sec corresponds to 335 gpm.

e) There was no detectable wear on the fuel rod fins resulting from operation for 33 hr in the pressure drop tests.

The 10-2 test section, consisting of a 4.00-in. ID SAP pressure tube with a rolled joint at each end, was installed in TF-10 and the test program was initiated on March 7, 1966. The fuel string used in the 10-1A tests was installed in 10-2. The principal objectives of this test are the determination and evaluation of the following:

- 1) Integrity of the rolled joints
- 2) Pressure tube deformation and vibration
- 3) Fuel bundle vibration
- 4) Fuel bundle pressure drop
- 5) Fuel rod fin wear
- 6) Pressure tube internal wear.

Figure 21000-3. TS-10-1A Pressure Drop vs Organic Flow at 650°F (HB-40)

This information will be obtained at conditions simulating the reference design reactor conditions, within the limits of TF-10 capabilities. All components necessary for the assembly of the hot pressed joint test section (TS 10-4) were procured during the last report period.

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

Major construction for TF-20 was initiated. The primary piping, auxiliary piping, and loop structure contracts were placed and work begun. Major loop components, such as the pumps and heat exchangers, were received. Instrumentation was purchased and electrical design work is approximately 80% completed. Ventilation, electrical, and insulation contracts will be placed during the next quarter.

F. THERMAL CYCLE LOOP (TF-50)

As reported in the last quarter, the primary pumps were disassembled shortly after TF-50 shakedown had begun. A thorough investigation revealed that the pump stator cans were circumferentially slit open and that only a complete rework of the pumps could rectify the situation. The problems were found to be due to basic flaws in design of the stator. The pumps were repaired by the supplier and re-installed in the loop, and shakedown was resumed.

Test sections being fabricated for testing in the thermal cycle loop include:

- 1) Rolled Transition Joints (see Section 212000)
- 2) Flexible Hose (see Section 81000)
- 3) B. F. Goodrich End Seal
- 4) Mechanical Couplings (see Section 81000)
- 5) Static Seals (see Section 81000)

An order was placed for four B. F. Goodrich radial seals and one seal test housing. The test section is expected to be delivered during the next quarter.

G. PRESSURE TUBE BURST TESTS

A burst test program on 7% and 10% SAP pressure tubes was initiated during this report period to obtain design data on the behavior of the tubes at

temperatures and pressures of interest to the HWOCR design. Material property information that will be obtained include:

- 1) Stress rupture life
- 2) Notch sensitivity
- 3) Ultimate tensile strength

The tests are being conducted at CE's Nuclear Components Division Metallurgical Labs in Chattanooga, Tennessee. A list of tests to be conducted is included in Table 21000-2.

TABLE 21000-2
TABULATION OF SAP RUPTURE TESTS

Material	Test Type	Number	Temperature (°F)	Hoop to Longitudinal Stress Ratio	Notched On ID
SAP 895	Rupture	1	700	2/1	
		4	800	2/1	
		1	900	2/1	
SAP 895	100-hr	2	700	2/1	
		1	800	2/1	
		1	800	2/1	X
		1	900	2/1	
		1	900	1/1	
SAP 895	1000-hr	1	700	2/1	X
		1	700	1/1	
		2	800	2/1	
		1	800	2/1	X
		1	800	1/1	
		1	900	2/1	
		1	900	2/1	X
SAP 895	5000-hr	1	700	2/1	
		1	700	1/1	
		1	800	2/1	
		1	800	2/1	X
		1	900	2/1	
		1	900	1/1	
XAP	1000-hr	1	800	2/1	
		1	800	2/1	X
		2	800	1/1	
		1	800	1/1	X
		1	900	2/1	

To date, one room temperature specimen of 4-in. ID SAP 895 tubing has been ruptured at 1310 psi, which corresponds to 43,700 psi stress in the tube wall.

H. OTHER ASSOCIATED WORK

1) A TV and borescope system was purchased for internal inspection of pressure tube assemblies. Other inspection items required are being investigated.

III. NEXT REPORT PERIOD ACTIVITIES

A. PRESSURE TUBE DEVELOPMENT

- 1) Fabrication of two SAP 895 pressure tube test sections will be initiated.
- 2) Design of a test assembly (TS 20-5) which will ultimately incorporate reactor pressure tube internal components will be completed.
- 3) Tube burst testing of six short-time specimens, six 100-hour specimens, and several 1000-hour specimens will be completed.
- 4) Testing of the B. F. Goodrich end seal will be initiated.
- 5) A program proposal for in-reactor testing of a pressure tube assembly will be written.

PROGRAM PLAN

ACTIVITY: Transition Joints

ACCOUNT: 21200

I. PROJECT OBJECTIVES

The objective of this task is to develop a transition joint between the out-of-core steel piping and the pressure tubes which will be a SAP or a zirconium-based alloy. These joints are required to maintain strength and leak-tight integrity under all anticipated conditions of reactor operation. The transition joints under evaluation during this quarter include:

- a) SAP-to-stainless steel hot pressed joints
- b) SAP-to-stainless steel explosively formed joints
- c) SAP-to-stainless steel rolled joints
- d) Zircaloy-to-stainless steel hot pressed joints.

The operating environment of the pressure tube includes internal pressure and flow of organic coolant at a temperature of 750°F, thermal cycling, pressure cycling, and vibration. 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, (3) reactions with the organic coolant such as corrosion and hydriding, and (4) leakage of the organic into the gas annulus.

An intensive test and evaluation program for joints which appear most promising is being conducted, including mechanical tests, metallurgical examination, thermal cycling tests, and thermal stability tests. 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 REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. SAP-STEEL HOT PRESSED TRANSITION JOINTS

1. Fabrication of Experimental Joints

A total of 27 helium leak tight SAP-steel transition joints, of the configuration required for the HWO CR, were hot pressed during this quarter. The joint components were machined as shown in Figure 1. Four-inch, schedule-80 pipe was used as raw material for the steel components, while specially extruded heavy-wall tubing was used for the SAP components. The knife-edge surfaces of the steel components were electroplated with nickel (~ 0.2 -mil thick) using a Wood's nickel striking bath. The nickel-plated surfaces were subsequently vapor plated with tungsten (~ 0.5 to 2.0 -mil thick) by locally heating in the presence of gaseous WF_6 and hydrogen causing direct reduction of the fluoride. The joint fabrication data are summarized in Table 1.

The SAP components were machined in two different lengths ("L" dimension of Figure 1). Three-inch lengths of SAP were used for fabricating joints for tensile testing. The longer $6\frac{5}{8}$ -in. length is needed when flash welding transition joints to SAP tubing to provide space for the clamping dies of the flash welding machine. The shorter hot pressed joints were finish-machined for tensile testing; a joint of this type is illustrated in Figure 2.

Difficulties were experienced during initial hot pressing trials due to a tendency of the SAP component to gall on the ID during removal of the hot pressing mandrel. "Aquadag" (colloidal graphite) was previously used successfully as a mandrel lubricant for hot pressing smaller diameter transition joints, but was found to be inadequate for the larger joints. The application of boron nitride to the mandrel did not significantly improve the lubrication problem. Excellent lubrication was provided and SAP galling was eliminated by the use of Molykote (MoS_2).

2. Testing and Evaluation of Joints

Tensile test data for several full sized transition joints are included in Table 1. The low tensile strength of joint H-12 was possibly caused by boron

Figure 21200-1. SAP-Steel Transition Joint Components

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

Bonding Temperature: 1025 ± 15°F							
Bonding Load: 50 Tons							
Bonding Time: 5 ± 1 min. except where noted							
Mandrel Lubrication: Molykote except where noted							
Diffusion Barrier: Nickel-Tungsten							
Joint No.	SAP Material	Steel Material	SAP Length, L, (in.)	Tensile Test Data			Notes*
				Test Temperature (°F)	Ultimate Load (lb)	Ultimate Tensile Strength (psi)	
H-4	XAP001	347 SS	6-5/8	RT	73,800	29,800	1, 6
H-5	SAP895	347 SS	3				1, 2, 6
H-6	SAP895	347 SS	3				6
H-7	SAP895	347 SS	6-5/8				6
H-8	SAP895	347 SS	6-5/8	RT	83,400	33,700	2, 3
H-9	SAP895	347 SS	3				
H-10	SAP895	347 SS	3				
H-11	SAP895	347 SS	3				
H-12	SAP895	347 SS	3	750	47,800	19,300	3a
H-13	SAP895	347 SS	3				
H-14	SAP895	347 SS	3				
H-15	SAP895	347 SS	3				
H-16	SAP895	347 SS	3	750	26,300	10,600	4
H-17	SAP895	347 SS	3				
H-18	SAP895	347 SS	6-5/8				
H-19	SAP895	347 SS	6-5/8				
H-20	SAP895	347 SS	6-5/8	750	27,700	11,200	5
H-21	SAP895	347 SS	6-5/8				
H-22	SAP895	347 SS	6-5/8				
H-23	SAP895	347 SS	6-5/8				
H-24	SAP895	347 SS	3	750	27,500	11,100	5
H-25	XAP001	A-106B	3				
H-26	XAP001	A-106B	3				
H-27	XAP001	A-106B	3				
H-28	XAP001	A-106B	3	750	27,500	11,100	5
H-29	XAP001	A-106B	3				
H-30	XAP001	A-106B	3				

*Notes: 1. Aquadag Mandrel Lubrication
 2. ID Galling
 3. Boron Nitride Mandrel Lubrication
 3a. Boron Nitride Mandrel Lubrication, possibly contaminated joint
 4. Heated at 950°F (air atm), 504 hr, helium leak tight
 5. Heated at 1000°F (air atm), 312 hr, not helium leak tight
 6. Bonding time ≤ 1 min

3-2-66

7680-4035

Figure 21200-2. Hot Pressed SAP 895-347 Stainless Steel Transition Joint for Tensile Testing

nitride contaminating the joint interface during hot pressing. This is suggested by poor adherence of SAP on the ID side of the steel knife-edge surfaces, as illustrated in Figure 3. The fracture of H-14 after tensile testing at 750°F is shown in Figure 4 and is generally typical of the fractures of H-9 to H-15 (excluding H-12).

Joints H-15 and H-16 were subjected to thermal treatments of 950 and 1000°F, respectively, in an air atmosphere. Both joints were overheated by 50°F for approximately 30 min. due to an instrument malfunction. Joint H-15 was tensile tested at 750°F after heating 504 hr at 950°F. The tensile strength was not significantly different from a similar joint (H-14) in the asbonded condition. Joint H-16 was found to leak severely in a helium leak test after heating 312 hr at 1000°F. Subsequent metallographic examination revealed gross degradation of the tungsten layer and reaction of SAP with both the tungsten and the steel component.

Figure 21200-3. Steel Component
From Joint No. H-12 (SAP 895-
347 SS) After Tensile Test at
Room Temperature

H-12

2-4-66

7680-4029

Figure 21200-4. Joint No. H-14
(SAP 895-347 SS) After
Tensile Test at 750°F

H-14

7680-4023

3. SAP to Carbon Steel Hot Pressed Transition Joints

The feasibility of hot pressing SAP-carbon steel transition joints was demonstrated with the fabrication of six joints (H-25 through H-30). A-106B pipe was used as the steel component with XAP 001 as the SAP material. (XAP 001 was used to conserve the limited supply of SAP 895 alloy.)

B. FLASH WELDING DEVELOPMENT

Flash welding trials with 4-in. ID SAP 895 tubing were conducted to establish optimum welding parameters. Both 0.090- and 0.100-in. wall SAP materials were used. A total of nine joints were flash welded and tensile specimens were machined from the joints. The tensile test data are summarized in Table 2. The maximum joint efficiency was somewhat lower than that previously obtained with XAP 001 material. Metallographic examination of the first six joints generally showed an absence of defects. The microstructure of F-4 is typical of these and is shown in Figure 5.

TABLE 21200-2
TENSILE TEST DATA FOR FLASH WELDED
4-in. ID SAP TUBES

Joint No.	No. Specimens	Ultimate Tensile Strength (ksi)		0.2% Yield (ksi)		Joint** Efficiency (%)	Remarks
		Range	Avg.	Range	Avg.		
F-1 *	10	38.8-41.7	40.4	26.6-29.8	28.2	-	Failure in weld
F-2	3	36.9-37.5	37.2	26.0-26.9	26.4	86.8	Failure through weld
F-3	3	37.3-37.3	37.3	27.2-29.0	28.0	87.1	Failure in weld
F-4	3	37.3-37.7	37.4	27.8-29.5	28.5	87.3	Failure in weld
F-5	3	32.4-36.9	35.4	26.6-27.3	27.0	82.6	Failure in weld
F-6	4	36.3-36.8	36.5	26.6-28.3	27.4	85.2	Ductile failure through weld
F-7	3	38.2-39.5	38.7	27.3-28.7	28.0	90.4	Ductile failure through weld
F-8	3	38.2-40.2	39.3	27.2-27.4	27.3	91.7	Ductile failure through weld
F-9	3	-	-	-	-	-	Not tested

*Upset partially removed from F-1. Upset completely removed from F-2-F-8

**Comparative Ultimate Tensile Strength, based on 42,800 psi for parent metal: Joints F-1 through F-6 are 0.090-in. wall, joints F-7-8 and 9 are 0.100-in. wall

Figure 21200-5. Microstructure of
Flash Welded Joint No. F-4

5-2-66

100X
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C. TEST SECTION FOR TF-10

A test section for TF-10, designated TS-10-4, is approximately 95% complete. This assembly consists of a A-106B carbon steel - 347 SS - SAP 895 double transition, flash welded to each end of a SAP 895 pressure tube.

Six hot pressed joints were fabricated as candidate joints for TS-10-4 (H-18 through H-23 in Table 1). An automatic TIG welding procedure was developed for joining A-106B carbon steel to 347 stainless steel which utilized Inco 82 consumable inserts and filler wire. Carbon steel extensions were subsequently welded to the six hot pressed joints and machined to final dimensions. The upper transition joint assembly (H-19) used in TS-10-4 is shown in Figure 6.

The transition joints were flash welded to the SAP 895 pressure tube section (tube P-8) using previously established weld parameters for joints F-7 and F-8.

4-6-66

7680-4034 CN A

Figure 21200-6. Upper Transition Joint Assembly
(No. H-19) Used in TS-10-4
Test Section

The internal flash and upset was removed from TS-10-4 by preliminary boring in a lathe followed by final machining with a specially designed milling head. The milling head is illustrated in Figure 7.

The TS-10-4 test section will be complete after welding a carbon steel extension to the lower transition joint assembly followed by a final machining operation on the extension.

D. NONDESTRUCTIVE TEST DEVELOPMENT

Nondestructive testing activities were performed to establish radiographic techniques for the carbon steel-to-stainless steel fusion weld, and dye penetrant testing and exploratory ultrasonic scans of the transition joint were also performed.

The carbon steel-to-stainless steel fusion welded joint configuration is particularly well suited to radiographic testing for internal soundness.

7680-51158
Figure 21200-7. Milling Head Used for Removal of Internal Upset Material from Flash
Welded Joints in TS-10-4 Test Section

21200-10

Radiography was used to aid in establishing the welding process, and to permit repair of undesirable joints. Stereo radiography was used to determine flaw locations, to permit rapid and effective repair welding.

Ultrasonic testing of the hot pressed transition joint is being investigated as the primary method of nondestructively determining internal joint soundness. All testing to date has been performed using the Delta Scanner. This scanner is a separate transmit-receive transducer array that is substantially less sensitive to flaw orientation than more conventional ultrasonic test systems. In tests performed to date, the joint assembly has exhibited signals from the transition joint zone. Signals from the joint would normally indicate flaws, but signals probably originated from the complex joint interface (SAP-W-Ni-SS) as the result of sound velocity differences. Subsequent tensile tests of the assembly indicated a sound joint. A flat interface specimen, 3/8 in. diameter by 2-1/4 in. long, made of 1100 aluminum alloy diffusion bonded to stainless steel through nickel-tungsten, has been prepared to determine more accurately the source of the indications noted from the transition joint.

High resolution dye penetrant testing of the transition joint, after inside and outside surface machining, has proved very effective for detecting surface flaws. The Hi-Rez method was used for this work, since replicas of the surface being tested could be obtained as a permanent record for correlation with destructive test results. Only one small surface indication was found on any of the joints tested. This was a 1/8 in. long flaw on the inside surface of H-13, which was found to be a small disbond zone after destructive testing.

E. ROD-TYPE TRANSITION JOINTS FOR WR-1 IRRADIATIONS

Rod type SAP-steel transition joints will be irradiated in the WR-1 reactor to determine the irradiation stability of the joints. The specimens will be enclosed in a specimen holder and attached to the hanger rods of the WR-1-953 and -954 irradiation experiments. A specimen holder has been designed which will provide for irradiation of 16-rod specimens 0.350 in. in diameter by 2.50 in. long. Both Ni-W and Ni-Nb will be used as diffusion barrier materials in the SAP-steel rod specimens. Evaluation of irradiated specimens will be based primarily on tensile testing and metallographic examination.

F. SAP-TO-STEEL ROLLED JOINTS

Design investigation for SAP-to-steel transitions has shown that the use of rolled joints in the assembly of the pressure tube offers the following advantages.

- 1) Low cost
- 2) Ease of assembly
- 3) Elimination of SAP-to-SAP flash welds

The major disadvantage is the uncertainty of leakage of rolled joints under cyclic thermal stress environments. The objective of this work is to develop fabrication techniques which eliminate the leaks.

A program to develop and test SAP to steel rolled joints which was initiated during the last report period is continuing. Those joints which appear most promising are being tested in component form and as part of pressure tube assemblies. Table 3 lists the articles fabricated and/or tested during this report period. Hardware components for 4-in. ID sandwich-type joints have been received and fabrication is in progress.

During this report period, a primary effort was devoted to preparation and fabrication of inner sleeve rolled joints for the TS 10-2 test section which was accomplished in the following steps:

- a) The SAP tube was inserted into the calandria tube test section with the cleaned end at the top of the test section.
- b) A support fixture suspended from the bottom of the test section provided a supporting platform for the SAP tube.
- c) A 3-1/2-in. -diameter steel pipe approximately 18 ft long with rubber O-rings spaced at 2-ft intervals was inserted into the SAP tube to provide a tube bending restraint (Figure 8).
- d) The outer steel collar was placed in position and secured to the top of the test section and the SAP tube height was adjusted to locate the SAP tube properly with respect to the outer collar.
- e) A steel mandrel was inserted into the end of the SAP tube and hydraulically driven to expand the SAP tube until it came into contact with the outer collar in the grooved area of the collar (Figure 9). The mandrel was removed and the SAP tube cleaned with water and acetone rinses to remove the lubricant.
- f) The inner steel sleeve was placed in position and a hardened steel mandrel was hydraulically driven through the inner sleeve to expand the steel sleeve until it came into contact with the SAP tube.
- g) The hydraulic jack was removed and the rolling tool (Figure 10) was used to complete fabrication of the joint.

TABLE 21200-3
ROLLED JOINT FABRICATION AND PRELIMINARY TEST DATA

Joint Size	Joint Type	Testing	Results
4 in.	2-groove, not line-on-line	<ol style="list-style-type: none"> 1. Helium leak tested 2. Tensile tested 3. Thermal cycled 	<ol style="list-style-type: none"> 1. No measurable helium leakage through either path. 2. When tensile testing, tube broke before joint pulled out at 41,000 psi. 3. No measurable helium leakage after 500 thermal cycles (550 to 750°F).
3.24 in.	8-in. -long double ended specimen; one end 2-groove line-on-line, other end 5-groove not line-on-line	<ol style="list-style-type: none"> 1. Helium leak tested 2. Tensile tested 3. Thermal cycled 	<ol style="list-style-type: none"> 1. Initially no measurable helium leakage through either leakage path. 2. When tensile tested, tube broke before joint pulled out at 16,300 psi. 3. No measurable helium leakage during 1000 thermal cycles between 550 and 750°F; however, on cooldown, joint started leaking at 400°F. The leakage increased exponentially on further cooldown with maximum leakage at room temperature.

5-2-66

7680-25646

Figure 21200-8. Calandria Test
Section and Inner Support Pipe

Figure 21200-9. Hydraulic Jack
and Fixturing for Driving Mandrel

5-2-66

7680-25647

5-2-66

7680-25648

Figure 21200-10. Rolling Tool Operation

h) After completion of the first joint the SAP tube was removed from the test section, inverted and then reinserted from below into the test section. The procedure for assembling the upper SAP tube joint was the same as for the bottom joint.

i) After rolling was completed, the 3-1/2-in. -diameter support pipe was removed and the upper joint assembly was bolted to the test section.

A helium leak check of the joints at room temperature showed no measurable leakage between the outer steel collars and the SAP tube at either joint with an instrument sensitivity of 10^{-6} cc/sec.

G. EXPLOSIVE BONDING

The SAP-to-stainless steel explosive bonding feasibility study initiated in the last quarter has continued. A total of six preliminary bonding attempts have been conducted to date employing 6063 aluminum tubes to conserve the SAP material. Only the fifth and sixth attempts were bonded sufficiently to enable sectioning for metallographic and peel test samples. Figure 11 shows the intermetallic formed in the fifth attempt. This section was taken at the midpoint of the 6-in. -long lap joint assembly. The intermetallic is 0.0025 in. thick, indicating excessive energy input for this material combination. Inter-metallic formation was expected, but not to this extent. Peel testing reflected the brittle behavior of the intermetallic; the joint separated easily along the bond with no observable adhesion between the 6063 aluminum and the type 321 stainless steel.

On the sixth bonding attempt, the explosive charge was reduced by 25%, and 0.002-in. niobium foil was placed in approximately one third of the interface, in an effort to reduce the intermetallic formation. In areas without the niobium foil, the intermetallic formation is similar to that of the fifth attempt, as can be seen in Figure 12. The results of the peel testing and metallography of the niobium containing interface are encouraging. Peel testing indicated adhesion of the niobium to both the stainless steel and the 6063 aluminum, and some resistance to separation was observed. Metallography disclosed the "wave like" bonding between the niobium and stainless steel in some areas, Figure 13, indicating an unstable jet effect. Figure 14 shows a continuous intermetallic formation at the niobium-stainless and the 6063 niobium-aluminum interface; and Figure 15 shows a similar intermetallic formation at the niobium aluminum

Stainless
Steel

6063 Aluminum

5-2-66

100X

7680-25649

Figure 21200-11. Photomicrograph of Explosive Bonded 6063 Aluminum to Type 321 Stainless Steel Showing the Inter-metallic Formed on Bonding. Thickness is Approximately 0.0025 in.

Stainless
Steel

6063 Aluminum

5-2-66

100X

7680-25650

Figure 21200-12. Photomicrograph of Explosive Bonded 6063 Aluminum to Type 321 Stainless Steel Showing Similar Intermetallic Formation with Reduced Charge Weight. Thickness is Approximately 0.005 in.

Stainless
Steel

Niobium

6063 Aluminum

5-2-66

100X

7680-25651

Figure 21200-13. Photomicrograph of Explosive Bonded 6063 Aluminum to Type 321 Stainless Steel with Niobium Interleaf Showing Results of Unstable Jet Effect

Stainless
Steel

Niobium

6063 Aluminum

5-2-66

100X

7680-25652

Figure 21200-14. Photomicrograph of Explosive Bonded 6063 Aluminum to Type 321 Stainless Steel With Niobium Interleaf Showing Intermetallic Formed on Both Interfaces

Figure 21200-15. Photomicrograph of Intermetallic Formed
Between Niobium and 6063 Aluminum Upon
Explosive Bonding

interface. Generally, intermetallic formation was observed at either the niobium-stainless or the niobium-aluminum interface, and only occasionally at both interfaces.

The next bonding attempt will be made between tungsten-coated stainless steel and SAP. This bonding attempt is waiting for tungsten coating, which requires minor modification of the coating equipment and is presently in progress.

H. ZIRCONIUM ALLOY-STEEL HOT PRESSED TRANSITION JOINTS

During the quarter, bonding parameters for fabricating diffusion couples were investigated. The bonding temperature was varied from 1100 to 1600°F in approximately 100°F increments, with pressing performed after 5 min at temperature, and cooling immediately after pressing. Metallographic examination disclosed no intermetallic formation, and only local contact between the Zircaloy-4 and the stainless steel. Two bonding attempts were performed at 1600°F and the time before and after pressing was varied. Figure 16 shows the intermetallic formed with 30 minutes at 1600°F before and after pressing. The second phase in the equiaxed alpha matrix shows a decreasing concentration

a. Bright Field

500X

b. Polarized Light

500X

5-2-66

7680-25654

Figure 21200-16. Photomicrographs of Zircaloy 4 to
Type 321 Stainless Steel Bond (1600°F for
1 hour total)

from the interface, and is thought to be the result of diffusion of a beta stabilizer (perhaps chromium) from the stainless steel. Figure 17 shows a similar result with 15 min at 1600°F before pressing and 5 min afterward. The white particles in the polarized light photos are oxygen-rich areas and demonstrate surface cleaning by diffusion of the oxides into the base metal. These bonds are satisfactory for the diffusion studies of the Zr-steel system, but refinement of the bonding technique will be required for hardware items.

I. COMMERCIALLY AVAILABLE ZIRCALOY-TO-STAINLESS STEEL JOINTS

During this report period the following was accomplished.

- 1) Five tandem extruded 4 in. -ID Zircaloy-4 to 304 stainless steel joints were ordered from both NMI and Torrance Brass.
- 2) Five friction welded 4 in. -ID Zircaloy-4 to 304 stainless steel joints were ordered from AMF.
- 3) Two explosively bonded 4 in. -ID Zircaloy-2 to 304 L stainless steel joints were supplied by Dupont at no cost, together with several solid sections for test evaluation. Preliminary metallographic examination revealed a continuous bond without the formation of a diffusion zone, as shown in Figure 18. An infrequently observed brittle constituent at the bond interface (Figure 19) may be either entrapped foreign material on the plate surface or locally melted metal, as hypothesized by Dupont personnel. Further evaluation will be performed during the next report period.

J. DIFFUSION BARRIER STUDY

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

During the quarter, the tungsten, rhenium, tantalum, and niobium short time screening diffusion anneals were completed and evaluated, as indicated in the following table:

Temperature (°F)		Time (hr)	
1050	100 complete	600 complete	1,000 complete
900	500 complete	3,000 in progress	5,000 in progress
750	1,000 complete	6,000 in progress	10,000 in progress

a. Bright Field

500X

b. Polarized Light

500X

5-2-66

7680-25655

Figure 21200-17. Photomicrograph of Zircaloy 4
to Type 321 Stainless Steel Bond (1600°F for 20
minutes total)

5-2-66

Unetched Mag 50X

7680-25656

Figure 21200-18. Bond Interface Between Zircaloy-2 (left)
and 304L Stainless Steel (right)

5-2-66

Unetched Mag 500X

7680-25657

Figure 21200-19. Brittle Constituent at Bond Interface

From the completed diffusion anneals, it has been determined that:

- a) No observable diffusion occurred between the SAP and the niobium in any of the couples.
- b) No observable diffusion occurred between SAP and tantalum in 100 hr at 1050°F, 500 hr at 900°F and 1000 hr at 750°F. Thin (0.0001 to 0.0002) intermetallic layers were detected after the 600-hr and 1000-hr anneals at 1050°F.
- c) No observable diffusion occurred between SAP and tungsten and also SAP and rhenium after 500 hr at 900°F and 1000 hr at 750°F. After 100 hr at 1050°F, the aluminum had reacted completely with tungsten and rhenium coatings and the formation of intermetallics with the stainless steel substrate had begun.

The rapid reaction at 1050°F in 100 hr and the lack of reaction at 900°F in 500 hr suggests either a very high activation energy or the occurrence of a rapid diffusion high phase between 1050 and 900°F. Hansen and Anderko* tentatively indicate the existence of a phase transformation at 1076°F in the high aluminum portion of an Al-W phase diagram. To check the possibility of a high temperature phase occurring at 1050°F, additional diffusion couples were fabricated from solid tungsten and rhenium rod stock and annealed at 1025 and 1000°F. Annealing times were 16 and 113 hr.

In the SAP-tungsten couples, fracture of the tungsten disks provided diffusion interfaces in both the transverse and longitudinal rod axis. Anisotropy of diffusion was observed. With a diffusion direction parallel to the rod axis, no intermetallic was formed at 1000°F in 16 and 113 hr, or at 1025°F for 16 hr. In 113 hr at 1025°F, an intermetallic layer 0.0035 in. thick was formed. In a diffusion direction perpendicular to the rod axis, thin intermetallic layers (<0.0005 in.) were formed at 1000°F for both times. At 1025°F, reaction layers $\approx$ 0.0009 in. and 0.005 in. were produced in 16 and 113 hr, respectively. Figure 20 shows the pronounced difference in diffusion direction observed in the 1025°F, 113-hr couple. The 1000 and 1025°F diffusion couples indicate the existence of a phase change in the SAP-W system with the phase boundary between 1000 and 1025°F. The occurrence of this phase change dictates that thermal treatments for accelerated life testing must be performed at or below 1000°F. The anisotropy of diffusion has prompted an investigation presently in progress of the preferred orientation of the chemical vapor deposits of tungsten of the reference joint compared to the rod stock.

*Hansen and Anderko, "Constitution of Binary Alloys", 1958.

Figure 21200-20. Photomicrograph of Tungsten-SAP Diffusion Couple
After 113 hr at 1025°F Showing the Dependence of Diffusion
Upon the Crystallographic Orientation of the Tungsten

In the SAP-rhenium system, diffusion was observed at both the 1000°F and 1025°F annealing temperatures. The thickness of the intermetallic zone is as follows:

Temperature (°F)	Thickness (inch)	
	16 hr	113 hr
1000	0.003	0.0035
1025	0.0045	0.0155

At 1000°F, the diffusion penetration is along the grain boundaries as shown in Figure 21. At 1025°F, the intermetallic-rhenium interface occurs along the grain boundaries, but volume diffusion is readily apparent as shown in Figure 22.

A chemical analysis to determine the impurity level is being performed. If the rod stock is of high purity, the grain boundary diffusion of aluminum will eliminate rhenium as a candidate diffusion barrier material.

Rhenium

SAP

5-2-66

250X

7680-25658

Figure 21200-21. Photomicrograph of Rhenium-SAP Diffusion Couple After 113 hr at 1000°F Showing Grain Boundary Diffusion

Rhenium

Intermetallic

5-2-66

250X

7680-25659

Figure 21200-22. Photomicrograph of Rhenium-SAP Diffusion Couple After 113 hr at 1025°F Showing Gross Volume Diffusion

Molybdenum coated diffusion couple parts have been procured and diffusion couples fabricated and electron-beam seal welded. Short-term high temperature ($\geq 1000^\circ\text{F}$, $< 100\text{-hr}$) diffusion anneals are presently in progress to determine if a high temperature phase with accelerated diffusion exists as has been observed in the SAP-tungsten system.

Vanadium coating of Type 321 stainless steel by chemical vapor deposition has been evaluated* and found unacceptable due to excessive diffusion at the deposition temperature. Because of this difficulty and the promising results obtained with the other barrier materials, work on the vanadium and vanadium-silicon duplex barriers has been postponed.

III. NEXT REPORT PERIOD ACTIVITIES

Activities planned during the next report period include the following:

- a) Complete fabrication of the TS-10-4 test section.
- b) Fabricate a SAP-steel test section for the TF-20 loop.
- c) Issue a preliminary process specification for flash welding SAP pressure tube material.
- d) Initiate thermal-pressure cycling tests of hot pressed, flash welded subassemblies.
- e) Continue non-destructive test development for hot pressed SAP-steel and flash welded SAP joints.
- f) Continue diffusion studies.
- g) Fabricate SAP-steel hot pressed rod specimens for attachment to hanger rods of WR-1-953 and -954 irradiation experiments.
- h) Complete preparations for the field fabrication of sandwich type SAP to steel rolled joints for TS 20-3.
- i) Complete testing and report on the inner sleeve SAP-steel rolled joints.
- j) Initiate thermal cycle testing of inner sleeve SAP-steel rolled joints in TF-50.
- k) Perform preliminary evaluations of metallurgically bonded zirconium-to-steel transition joints purchased from AMF, Torrance Brass, and NMI.

*J.G. McArdle, "Evaluation of Vanadium Coatings on Stainless Steel", AI-CE-TDR-1031, March, 1966

PROGRAM PLAN	
ACTIVITY: Reactivity Control Systems	ACCOUNT: 22000

I. PROJECT OBJECTIVES

The objective of this task is the development of the reactivity control systems necessary for the safe operation of the reactor. In general, control systems must be provided to compensate for changes in reactivity associated with (a) fuel temperature changes, (b) coolant temperature changes, (c) buildup and decay of xenon and samarium, (d) moderator temperature changes, (e) void effects in the coolant and the moderator, and (f) fuel burnup. In addition, an adequate shutdown margin must be provided.

The HWO CR Program goal of high fuel utilization and economic power costs requires the design of the reactivity control devices to minimize:

- 1) The excess reactivity needed for control purposes
- 2) The parasitic material remaining in the core at full power
- 3) In-core power peaking
- 4) Initial capital costs and replacement costs.

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

A. GENERAL

During this quarter conceptual designs of the power control rod actuators and shutdown control rod actuators were completed. Fabrication was initiated on a shutdown control rod actuator test device to determine feasibility of the design. Two-dimensional analysis of the xenon oscillation problems was initiated.

B. SHUTDOWN CONTROL SYSTEM

1. Development Progress

The shutdown control rod conceptual design selected for detailed study consists of a horizontally oriented, hydraulically operated, spring return type actuator assembly. This concept is shown schematically on Figure 22000-1.

The shutdown control rods penetrate horizontally the biological shield, the thermal shield, gas space, and calandria tank. The control rod actuator consists of a D_2O -operated hydraulic cylinder with a spring-loaded return. The cylinder has a 3.0-in. bore and a stroke of 130 in. The control rod is an integral part of the piston rod and is supported at the front end by a roller bearing assembly located at the moderator tank wall.

The calandria tank, shield tank, and CO_2 cooling gas are sealed by a series of concentric steel tubes extending from the various tanks to the area outside the biological shield. The tube diameters are stepped to prevent radiation streaming. All D_2O seals are static gasket type to minimize leakage. Except for the moderator tank sleeve which is welded to the tank, all the components are removable for maintenance or replacement after the moderator has been drained below the level of the particular control rod. However, it is possible to replace the latching mechanism, latching solenoid, and scram spring without draining the moderator tank.

The shutdown rods are in the fully inserted position with the reactor shut down, or in the fully withdrawn position with the reactor operating. Magnetic reed switches, located on the outside of the cylinder, actuate position indicating lights in the control room.

To withdraw a shutdown control rod, a 3-way solenoid valve controlling the flow of D_2O to the piston is energized. This pressurizes the head end of the piston cylinder and forces the piston into the retracted position; the scram spring is compressed at the same time. When the rod is fully withdrawn, it is automatically latched by the latching mechanism. Hydraulic pressure is then removed by de-energizing the 3-way solenoid valve. This action automatically connects all D_2O hydraulic lines from the cylinder to the D_2O in the calandria tank; pressure equalization to the moderator tank pressure on both sides of

22000-3

3-15-66C

7680-15224

Figure 22000-1. Shutdown Control Rod Actuator Schematic

the piston is thus obtained. De-energizing the latch solenoid causes an automatic scram of the rod. The rod is driven into the core by the scram spring with an initial force of about 60 lb and a fully extended force of about 10 lb.

Low density $\text{Al-B}_4\text{C}$ material is being investigated for use as the neutron absorber for the control rod. A low density aluminum material, manufactured under the trade name Foamalum, appears to be suitable as a carrier for B_4C and can be fabricated with a density of less than one.

2. Test Program

The test program outlined in the previous quarterly report, AI-CE-17, was initiated during this quarter. Fabrication of the Shutdown Control Test Device, shown on Figure 22000-2, has been started.

The test device will be operated in a test facility which is about 60% complete. This test facility consists of a 630-gal., 14-ft-long stainless steel tank equipped with seven viewing windows; the facility is shown on Figure 22000-3. The three windows on each side of the tank will permit visual observance of the rod during operation; the end window will be used to photograph the rod behavior.

C. XENON OSCILLATION AND POWER LEVEL CONTROL

The conceptual design of the xenon and power level control rod actuator selected for detailed study is a modification of the horizontal rack-and-pinion actuator described previously in AI-CE-17; Figure 22000-4 shows the conceptual design of the 3-actuator assembly selected. The major modifications to the actuator previously described in AI-CE-17 are given below.

1. In-Core Guide Structure

The in-core guide structure has been changed to a trifoil arrangement of 1-1/4-in. OD Zircaloy tubes welded together to form an integral assembly. Each guide tube assembly extends to the center of the reactor where it is supported by a vertical support tube.

2. Neutron Absorber Element

The design of the absorber element has been modified to include roller bearings on the poison element to minimize the wear on the guide tubes.

4-28-66

Figure 22000-2. Shutdown Control Test Device

7680-25619

4-28-66

7680-25620

Figure 22000-3. Shutdown Control Test Facility

Figure 22000-2. Shutdown Control Test Device

3-15-66

Figur

3. Actuator Mechanisms

A conceptual design was completed of an arrangement of the rack-and-pinion drive mechanisms to incorporate three actuators per assembly.

4. Analysis of Xenon Induced Spatial Oscillations

A two-dimensional study of xenon oscillations using the XENOSC code has been initiated.

III. NEXT REPORT PERIOD ACTIVITIES

During the next quarter it is planned that the following items will be accomplished.

a) A low density ($< 1 \text{ gm/cm}^3$) Al-B₄C poison element will be fabricated for evaluation purposes.

b) Design efforts on the power and xenon oscillation control systems will continue.

c) Testing of the Hilborn in-core detectors will begin at the Savannah River Laboratories.

d) Analysis of the method of compensating for xenon induced spatial oscillations will be continued using the XENOSC code.

PROGRAM PLAN

ACTIVITY: Evaluation and Analytical Techniques

ACCOUNT: 35110

I. PROJECT OBJECTIVES

The object 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) Coolant evaluation (availability, properties, cost)
- 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. COOLANT EVALUATION

1. Physical Properties

Many measurements were made with the differential scanning calorimeter. Table 35110-1 shows melting points and heats of fusion measured on the best samples of polyphenyls available, 98 to 99% pure. These are in good agreement with values in the literature. The heat of fusion of p-terphenyl had not been measured directly before, but was estimated at 35 cal/g from PVT data.

Strong evidence was obtained for the existence of an unstable crystalline form of m-terphenyl, melting at 79°C. This phase is formed when liquid m-terphenyl supercools to about 60°C, and converts to the normal form spontaneously, usually within a day.

TABLE 35110-1
MELTING POINTS AND HEATS OF FUSION
OF POLYPHENYLS

Compound	Melting Point (°C)	Heat of Fusion (cal/g)
Biphenyl	69 ± 1	28 ± 1
o-Terphenyl	57 ± 1	18 ± 1
m-Terphenyl	87 ± 1	25 ± 1
p-Terphenyl	212 ± 1	35 ± 1

Measurements on Santowax OM with 10% HB, after one month of storage, still showed a low heat of fusion, indicating incomplete crystallization. Santowax OM without HB showed the same effect to a lesser degree. Melting points fluctuated somewhat from sample to sample. Because of these fluctuations, no difference in melting range could be observed between OM and OM + 10% HB, as shown in Table 35110-2.

TABLE 35110-2
MELTING PROPERTIES OF SANTOWAX OM

Sample	Initial Melting Point (°C)	Final Melting Point (°C)	Heat of Fusion (cal/g)
OM	22 ± 5	72 ± 5	12 ± 2
OM + 10% HB	22 ± 5	72 ± 5	5 ± 2

The differential scanning calorimeter can also be used for specific heat measurements, by means of an accessory kit with sapphire discs as standards. Table 35110-3 shows the results of measurements on solid biphenyl, compared with two earlier measurements. Specific heat data may be useful in characterizing deposits such as coke, sludge, and film. To measure specific heats on liquid coolants at high temperatures by this method, it will be necessary to design sealed containers (with good thermal contact) to prevent evaporation.

TABLE 35110-3
SPECIFIC HEAT OF BIPHENYL

Temperature (°K)	Specific Heat (cal/g)		
	AI	Ref. 1	Ref. 2
300	0.300	0.303	0.299
320	0.332	0.329	0.331
330	0.341	0.341	0.347
340	0.354	0.354	0.362

Ref. 1: M. E. Spaght, et al., J. Phys. Chem. 36, 882 (1932)

Ref. 2: K. Ueberreiter and H. J. Orthmann, Z. Naturforsch.
5a, 101 (1950)

B. DECOMPOSITION AND MAKEUP RATES

1. MIT Organic Loop Experiment (Santowax WR)

Steady-State Run 17, at 90% terphenyl and 300°C (572°F), was started on February 28, 1966. Run 16, at 62% terphenyl and 300°C, gave a preliminary value of $G(-omp) = 0.098$ (molecules of terphenyl decomposed per 100 ev absorbed). Using second-order kinetics, this gives a value for pure terphenyl of $G_O(-omp) = 0.25$. With the current fast neutron fraction of 0.08, and assuming $G_n/G_\gamma = 4.5$, the corresponding value of G_γ is 0.20, in good agreement with estimates for Santowax OMP.⁽¹⁾

2. U-305 Experiment (Santowax OM)

Makeup rate information has been derived from the current experiment in the U-3 Loop by comparing HB production with thermal energy production. The loop is operated on a feed-and-bleed system, with a continuous bleed through pump seals and other leaks, compensated by occasional additions of coolant to maintain a constant level in the dump tank. Since it is difficult to measure the leakage directly (though an effort is being made to do this), the coolant additions are assumed to equal the coolant removed (the total coolant volume

has been kept constant within 3% except for one spill in which 40% of the coolant was lost). The weight of coolant added is multiplied by the fraction of HB in the coolant (before the addition) to give the weight of HB removed. At steady state, this should equal the HB produced.

Figure 35110-1 is a plot of cumulative HB production against integrated loop power (thermal energy production). Except for the period following the spill (a large deviation from steady state), the points fall on a straight line with a slope of 5.7 lb/Mwd, the indicated makeup rate. The HB analysis is the principal source of uncertainty in this number. The correct value could be as high as 6.7 or as low as 4.7 lb/Mwd, depending on whether the distillation or the sublimation method for HB is correct.

The U-305 experiment is potentially a good source of radiolytic makeup data for HWO CR, for the following reasons:

- a) The fuel element geometry (coolant-fuel ratio) is similar to HWO CR, and the power density is comparable. Therefore the gamma and fast neutron energies absorbed in the coolant, even though not accurately known, should be about the same as in HWO CR. This permits bypassing dose rate and yield calculations to obtain makeup rate directly.

Figure 35110-1. HB Production Rate in the U-3 Loop

b) The coolant, Santowax OM, is the reference coolant for HWOOCR. The following differences between U-305 and HWOOCR do exist at present:

1) The present average in-core temperature is lower in U-305. This is not important, since radiolysis is not very temperature dependent below 750°F, and a correction can be applied.

2) There is no large volume of coolant at high temperature. Therefore radiolysis, which may be important in HWOOCR (see below) is negligible in U-305.

3) The present HB level is $18 \pm 3\%$ instead of the nominal 10% in HWOOCR. The correction for this is discussed below.

4) The feed-and-bleed system in U-3 removes IB (Intermediate Boilers) as well as HB; hence the concentration of IB is lower (and the terphenyl concentration higher) than in HWOOCR. It is not certain whether this affects the rate of HB formation.

Any effect of IB would be much more important with a high-ortho coolant such as Santowax OM than with a high-meta coolant like Santowax WR, since more IB is produced. In U-305, the IB concentration is approximately equal to the HB concentration. It would be still higher in HWOOCR, since no IB will be removed.

If we make the common assumption that the HB production rate is proportional to the square of the non-HB concentration, then the HWOOCR radiolytic makeup rate indicated by the U-305 results is

$$(5.7 \pm 1.0) \left(\frac{90}{82 \pm 3} \right)^2 = 6.9 \pm 1.7 \text{ lb/Mwd}$$

This is in good agreement with current estimates.⁽²⁾ The precision of this estimate would obviously be improved by better HB analyses, but even as it stands it is perhaps as good as estimates from radiolysis experiments, in which the absolute accuracy of the dosimetry is usually about $\pm 25\%$. The assumption used (which amounts to assuming that IB behaves like terphenyl) is consistent with the results of radiolysis experiments, but most of these have not involved high HB concentrations. It should be noted that there is no conflict between this assumption and the generally accepted view that terphenyl radiolysis is second order in terphenyl concentration. At steady state, the terphenyl concentration automatically adjusts so that terphenyl consumption equals HB

production. This would largely eliminate any difference in makeup rates between OM and WR, since ortho- and meta-terphenyl have approximately the same $G(\rightarrow \text{HB})$ value, differing only in $G(-\text{omp})$.

3. Radiopyrolysis and Dose Rate Effect

Pyrolysis and radiopyrolysis (accelerated pyrolysis due to radiolysis products) are relatively important in HWO CR because of the large out-of-pile volume of coolant at high temperatures. Radiopyrolysis has been observed in both loop and capsule experiments in Europe and Canada and at MIT. The rate of pyrolysis is increased by radiolysis products by about a factor of 5 (the exact factor is no doubt a function of irradiation history and coolant composition).

Another high temperature effect which must be considered is the decrease in radiolysis yield of o-terphenyl with increasing dose rate. As shown in Table 35110-4, radiolysis yields do not increase with temperature as rapidly at 1 w/g (and 55% fast neutron fraction) as at 0.1 to 0.3 g. Since the dose rate to the coolant in HWO CR will exceed 1 w/g, the lower radiolysis yields apply.

TABLE 35110-4
DOSE RATE DEPENDENCE OF ORTHO-TERPHENYL
DECOMPOSITION RATES

Temperature (°F)	$G^{\circ}(\rightarrow \text{HB})^*$		$G(-\text{terphenyl})$	
	0.1 to 0.3 w/g [†]	1 w/g [§]	0.1 to 0.3 w/g [†]	1 w/g [§]
660	0.45	0.41	0.8	0.75
720	0.55	0.41	1.1	0.68
750	0.65	0.48	1.5	0.85
800	0.9	0.57	2.2	1.1

*Initial yields calculated from capsule data assuming second order kinetics.

[†]Data from AECL-2258, "Radiolysis of Ortho and Meta-terphenyl", A. W. Boyd and W. J. Connor (1965).

[§]Data from PR-CM-42, Progress Report, April-June, 1965.

Table 35110-5 is an illustrative calculation for various hypothetical loops. The calculation assumes 10% HB, $G(\rightarrow \text{HB}) = 0.58$ at 0.1 w/g and 750°F, $G(\rightarrow \text{HB}) = 0.43$ at 1 w/g and 750°F, and Euratom radiopyrolysis data. Cases 11 and 12 are closest to HWOOCR conditions. This table is intended only to illustrate the possible importance of radiopyrolysis and should not be construed as a prediction. It seems clear that radiopyrolysis may be important in makeup rates and should be studied further.

MIT plans to measure radiopyrolysis by capsule pyrolysis experiments on coolant removed from the organic loop. The experiments performed during the report period were on unirradiated coolant, to establish a control pyrolysis rate.

C. CHEMICAL ANALYSIS AND TESTING

1. Water Analyzer

The off-gas sampling unit⁽³⁾ was installed in the U-3 Loop. Off-gas was taken from the dump tank rather than the degasifier, because the location was more convenient and it was believed that dump tank off-gas would be less affected by fluctuations in degasifier operation and would give an equally good indication of coolant water content. The dump tank is purged with nitrogen and acts as a second degasifier stage, removing about half of the water.

The continuously indicating instrument proved to be less satisfactory in this application than the method of analyzing a measured gas sample with the integrating instrument, because of the difficulty of keeping the low off-gas flowrate constant. The continuous instrument was eventually incapacitated when the supply of nitrogen for dilution failed and the cell was flooded.

The integrating instrument has continued to function smoothly. Improvements on technique have reduced the time required for a measurement to 15 min. During February and March, 1966, 32 comparisons were obtained between coolant water content (by the Karl Fischer analysis) and the water content of the standard 50-ml gas sample as shown by the instrument. The results are shown in Table 35110-6. More points below 150 and above 300 ppm would be desirable for a good statistical analysis, but there is evidently a correlation between gas water content and coolant water content. The ratio shows no

TABLE 35110-5

HYPOTHETICAL RADIOPYROLYSIS AND DOSE RATE EFFECTS

Case	Dose Rate (w/g)	Reactor Temp. (°F)	High Loop Temp.	% of Loop at High Temp.	Out-of-pile /In-pile	Radiation* Decomposition	Radiopyrolysis*† Decomposition	% of Total† Decomposition due to Radiopyrolysis
1	0.1	750	650	30	2.2	0.0013	0.00014	9
2	0.1	750	650	5	20	0.0002	0.00002	9
3	0.1	750	750	100	2.2	0.0013	0.00045	25
4	0.1	750	750	50	20	0.0002	0.00022	54
5	0.1	750	750	100	20	0.0002	0.00045	70
6	1.0	750	650	50	2.2	0.0094	0.00014	1
7	1.0	750	650	5	20	0.0015	0.00002	1
8	1.0	750	750	100	2.2	0.0094	0.00045	5
9	1.0	750	750	50	20	0.0015	0.00022	13
10	1.0	750	750	100	20	0.0015	0.00045	23
11	1.0	700	700	50	100	0.0003	0.00004	12
12	1.0	750	750	50	100	0.0003	0.00022	42

*Weight fraction per hour

†Radiopyrolysis data of Leveque, et al., A/Conf. 28/p/53 (1964)

35110-8

TABLE 35110-6
U-3 LOOP WATER ANALYZER DATA

Date	Off-Gas H ₂ O, μ g	Coolant H ₂ O, ppm	Ratio μ g/ppm
2-1-66	3845	144	26.7
2	2880	229	12.6
3	3239	181	17.9
4	3013	179	17.3
8	2893	176	16.4
10	2883	186	15.5
14	3215	202	15.9
15	2812	195	14.4
16	3106	227	13.7
17	2909	236	12.3
18	2842	186	15.3
21	2817	209	13.5
22	2994	190	15.8
23	3226	240	13.4
24	2342	170	13.8
25	2996	212	14.1
28	2645	195	13.6
3-1-66	3069	196	15.7
2	3060	191	16.0
3	3293	218	15.1
4	3083	217	14.2
5	3533	274	12.8
6	3806	283	13.4
7	3190	250	12.4
8	3190	257	12.4
9	3430	353	9.7
10	3339	315	10.6
14	2307	104	22.2

TABLE 35110-6
U-3 LOOP WATER ANALYZER DATA (Cont.)

Date	Off-Gas H ₂ O, μ g	Coolant H ₂ O, ppm	Ratio μ g/ppm
18	4439	399	11.1
26	5153	392	13.1
28	6225	437	14.2
29	5386	339	15.9
		Mean	14.7
		Standard Deviation	2.22

apparent trend with either time or water content. The standard deviation is 15% of the mean, corresponding to ± 30 ppm at 200 ppm. The occasional large deviations (Feb. 1, March 14) may be due to irregular degasifier operation.

2. Chlorine Analysis

Development was concluded on the neutron activation and coincidence counting method. In addition to coolant analyses already reported,⁽⁴⁾ three samples of coolant from the U-305 experiment were analyzed, giving 0.99, 0.92, and 0.98 ppm chlorine. This is considerably higher than in PNPf coolant.

Additional tests were carried out to check the possibility of chlorine loss due to heating during irradiation for activation analysis. Samples were re-irradiated and re-analyzed repeatedly. The results are shown in Table 35110-7. The first three samples were from one batch of Santowax OMP, the next two were irradiated Santowax OM from the U-3 loop, and the last two were from another batch of OMP. The errors shown are standard deviations based on the variance of the aqueous standards irradiated with each sample. It appears that no significant loss occurred from either solid or liquid samples. Losses did occur, however, when Santowax OMP was melted and poured into the irradiation container, rather than packing it in mechanically. The loss was as high as 40% when the sample was melted for 5 min.

TABLE 35110-7
CHLORINE CONCENTRATION (PPM) OF TERPHENYL SAMPLES
ANALYZED REPETITIVELY

Designation	First Analysis	First Re-analysis	Second Re-analysis	Third Re-analysis	Sample Form
AB99-1	2.02 ± 0.11	1.94 ± 0.10	1.94 ± 0.09	1.81 ± 0.09	solid
AB99-2	1.84 ± 0.10	1.72 ± 0.09	1.94 ± 0.14	1.77 ± 0.08	solid
AB99-3	2.16 ± 0.17	1.85 ± 0.12	-	-	solid
21-1	1.07 ± 0.04	0.91 ± 0.07	-	-	liquid
21-2	0.92 ± 0.04	0.96 ± 0.07	-	-	liquid
AB125-1	2.48 ± 0.18	2.87 ± 0.12	2.79 ± 0.20	-	solid
AB125-2	1.94 ± 0.15	1.92 ± 0.10	-	-	solid

Development of the x-ray fluorescence method for chlorine was also completed during the period. A new method of sample preparation was developed; a thin layer of coolant is spread between Mylar films. It is estimated that 0.6 ppm chlorine could be detected with a 10-min. counting period.

D. COOLANT OXIDATION

All parts of the oxygen absorption loop⁽⁵⁾ were completed and assembled, except for the remote diaphragm pump, which has not yet been received. The cobalt-60 facility was not completed during the period, but is expected to be available soon. The residual gas analyzer mass spectrometer and the associated vacuum and gas sampling systems were tested and performed successfully with air. The analyzer was also used successfully as a helium leak detector.

Bendix time-of-flight mass spectrometer has been completely overhauled and modernized and now operates satisfactorily. Base-line resolution was obtained with the mercury isotopes. This instrument will be used in conjunction with gas chromatography to identify oxidation products.

Several organic samples were analyzed for oxygen by fast neutron activation. The procedure had been developed for the analysis of SAP. This should be very useful in the coolant oxidation studies.

E. HALOGEN CHEMISTRY

No experiments were performed in the period. Activity in this field was confined to literature study and the chlorine analytical methods discussed above.

III. NEXT REPORT PERIOD ACTIVITIES

The topical report on coolant physical properties, AI-CE-15, will be issued. The effect of p-terphenyl on the melting properties of ortho-meta mixtures will be investigated.

Progress on coolant decomposition rates at MIT and in Canada will be followed closely. The many factors influencing decomposition rates will be evaluated further with a view to compiling an up-to-date summary report on this subject.

Operation of the water analyzer in the U-3 loop will continue, if possible with a wider range of water content.

It is expected that the pump for the oxygen absorption loop will be received and that some preliminary experiments can be carried out in the period.

No new experimental work on analytical methods or on halogen chemistry is planned for the next quarter.

References

1. HWO CR Quarterly Technical Progress Report, July through September 1965, AI-CE-4, Table 0550-4, p. 0550-10, 11
2. HWO CR 500 Mwe Nuclear Demonstration Plant Conceptual Design, AI-CE-Memo-25, March 28, 1966, p. III-4
3. HWO CR Quarterly Technical Progress Report, October through December 1965, AI-CE-17, Figure 35110-2, p. 35110-7
4. HWO CR Quarterly Technical Progress Report, October through December 1965, AI-CE-17, Table 35110-3, p. 35110-10
5. HWO CR Quarterly Technical Progress Report, October through December 1965, AI-CE-17, Figure 35110-3, p. 35110-11

PROGRAM PLAN

ACTIVITY: Parameters Affecting Film Formation ACCOUNT: 35120

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 HWOCR 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 propensity of organic coolants. A forced circulation loop (Film Formation Loop) will be designed and operated to evaluate the effect on film formation rates of such parameters as radiation type and intensity, coolant type and composition (including impurities), and coolant flowrate. By developing an understanding of the film formation micro-processes, direct engineering evaluation of the macro-processes taking place in a nuclear reactor will be possible. 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. Preliminary Layout

A preliminary layout of the Film Formation Loop has been completed. The loop has been sized, pre-heat requirements determined, and hardware location established.

2. Components

Specifications for all loop component parts have been completed. Purchase orders have been placed with the various vendors on all long lead items except for the main circulating pump. A revised proposal for the pump has been received and only a few relatively minor discrepancies have to be resolved.

Two simple annular heater sections were fabricated. The DC power supply and pressure control system have been received and tested.

3. Water Test Loop

Assembly of a low pressure, low temperature water loop has been completed. This loop will be used to test various design concepts for the Film Formation Loop. A simple annular heater section has been installed in the water test loop and electrical bus connections to the DC power supply completed. One problem associated with the test section is reduction of mechanical loading to the relatively thin wall of the heater. A spring balance approach is being tested.

Various film thickness measurement techniques are being investigated. A promising method involves the use of a commercially available ellipsometer. This tool is widely used in measuring films of this thickness deposited on metal surfaces.

4. Selection of Irradiation Site

General Atomic has reevaluated the cost of irradiation at their facility and submitted a preliminary proposal for the Film Formation Loop program. A comparative evaluation of the technical and cost factors associated with performing this irradiation work at General Atomic or at the University of Saskatchewan accelerator facility is in progress and a decision recommendation will be made early next quarter.

B. FILM FORMATION RATE METER

A radiation cell is being designed to permit the irradiation of terphenyl in a standard 50 or 100 kv x-ray fluorescence machine. The deposition rate and character of the deposition can then be assessed to determine if this approach might be taken in the design of a film formation rate meter.

III. NEXT REPORT PERIOD ACTIVITIES

A. FILM FORMATION LOOP

Test of the heater section, heat exchanger and control systems will be completed in the water test loop. Assembly of the Film Formation Loop will start. Evaluation of film thickness measurement techniques will continue.

B. FILM FORMATION RATE METER

Tests will start on the character and rate of deposition of film in a 50 or 100 kv x-ray fluorescence machine.

PROGRAM PLAN

ACTIVITY: Fuel

ACCOUNT: 28000

I. PROJECT OBJECTIVES

The objective of the fuel element 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 fuel assembly design involves rod and bundle type elements 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 Refinement

The experimental effort to study the effect of improved vacuum on the UC carbothermic reduction reaction was delayed due to a defective booster-diffusion pump. The pump was replaced and subsequently subjected to a series of qualification tests. The new unit operated satisfactorily and an ultimate pressure of 1×10^{-3} torr was attained with the furnace operating in the range of 1850 to 2050°C. The system, comprised of furnace and pump, is now ready for a series of carbothermic reduction experiments to be conducted at two or more finishing pressures (including lowest attainable) and at several combinations of temperature (1850 to 2050°C) and time.

2. Feed Material Process Scaleup

Scaleup of the reference UO_2 carbothermic reduction process for synthesizing UC feed material, initiated during the previous report period, continued. A total of eleven batches of $\text{UO}_2 + \text{C}$ agglomerate were prepared in the 3 ft^3 (capacity) V-blender and subsequently vacuum sintered. Batch sizes of 60 and 100 kg were used and the agglomerates were blended from a mixture of 88.45 wt % UO_2 and 11.55 wt % (-200 mesh) graphite powder.

The results of carbon and oxygen analyses for the eleven batches are given in Table 28000-1.

TABLE 28000-1
CARBON AND OXYGEN ANALYSES OF SCALEUP BATCHES
OF UC FEED MATERIAL

Heat No.	Batch Size (kg)	Sintering Temp. ($^{\circ}\text{C}$)	Carbon (wt %)		O_2 (ppm)
			$\bar{X}$	σ	
1	100	2082	4.80	-	231
2	100	2081	4.88	-	318
3	100	2100	4.73	0.020	321
4	60	2088	4.72	0.042	100
5	60	2101	4.66	0.012	95
6	100	2103	4.80	0.032	154
7	100	2088	4.71	0.017	119
8	100	2108	4.76	0.062	251
9	100	2098	4.71	0.033	181
10	100	2109	4.74	0.030	118
11	100	2091	4.65	0.021	247

The above eleven heats yielded 824 kg of UC with a total uranium process loss of 3.41 kg or $\sim 0.4\%$. This loss covers blending, agglomeration, and sintering. All of the UC produced conformed to the oxygen requirement of the product specification ($< 500 \text{ ppm } \text{O}_2$) and all but one heat (No. 2) conformed to the carbon requirement.

A preliminary process specification for the production of UC feed material for use in the fabrication of fuel slugs for the HWOCR demonstration plant was completed.

3. UC Skull Casting Scaleup

Major effort was spent in developing process parameters and mold designs for casting small diameter UC slugs in large quantities. Casting efforts during the previous report period were limited to 91-cavity molds, having a 0.490-in.-diameter 7-1/2-in.-long cavity, and a theoretical slug yield of 28.0 kg.

Two new graphite molds with increased slug capacity (~33 kg/heat theoretical yield) were designed for casting 169, 0.374-in.-diameter, and 271, 0.295-in.-diameter slugs, respectively. A series of six 70-kg skull casting heats were made using the new mold designs and a 91-cavity mold modified in length from 7-1/2 in. to 8-1/2 in. to give an increase in theoretical slug yield from 28.0 to 32.1 kg. The results obtained for these six heats are given in Table 28000-2.

TABLE 28000-2
RESULTS OF UC SKULL CASTING SCALEUP HEATS

Heat No.	Charge Weight (kg)	Total UC Poured (kg)	Mold Size (No. of Cavities)	Mold Capacity (kg)	Total Slug Weight (kg)	Slug Diameter (in.)	Wt % C
4	71.9	39.0	169	33	29.2	0.374	5.10
5	72.2	41.7	169	33	28.2	0.374	4.82
6	70.4	37.2	271	33	28.4	0.295	4.95
7	71.9	36.2	271	33	20.1	0.295	4.96
8	69.6	43.8	91	32.1	31.4	0.490	4.84
9	67.0	42.1	91	32.1	23.5	0.490	5.04

The results indicate that large slug yields may be obtained for the three slug diameters evaluated. However, there is a mold cracking problem with the current multi-cavity, sleeveless mold design. One approach taken to eliminate this problem was to increase the mold cavity end plug clearance from 0.002 in. to 0.010 in. on the assumption that the plugs expand rapidly before

other parts of the mold heat up, thus initiating a fracture at the area of contact between plug and mold. Heat No. 9 was made using a 91-cavity mold having a 10-mil diametral clearance. This mold was observed to have developed hair-line cracks at the webs extending across the diameter of the mold. The cracks were less severe than previously encountered and the mold was usable for one additional heat. However, the 10-mil clearance is considered excessive because UC "flash" formed between the plug and mold wall. New designs are being planned with a 6-mil clearance to avoid flash formation and still provide sufficient room for thermal expansion. Another approach which is being considered as a means of minimizing the cracking problem, is the use of modular or multi-section molds. This is based on the theory that mold fracture initiates at the weakest point within the mold body and propagates through the mold to relieve induced stresses. In a modular mold, such crack propagation would be confined to a single module, minimizing mold replacement cost.

4. Surface Finish Casting Development

The reference mold material, ATJ graphite, used in casting UC slugs, is known to have sufficient porosity to retain small residual amounts of UC which upon exposure to air eventually convert to the oxide. The mechanism responsible for rough and defective surfaces on UC slugs is believed to be due to liberation of gas (CO) by the reduction of the UO_x during the pouring of molten UC. On the theory that eliminating the porosity in the mold would improve the as-cast surface of UC slugs, several ATJ molds were coated on the ID with a 3-mil-thick coating of pyrolytic graphite. The initial casting made in the coated mold had a smooth, almost glass-like surface over most of the slug. However, some rough areas were found where the pyrolytic coating had flaked off the mold. Subsequent castings from the same mold had rough surfaces since most of the coating had flaked off during the initial casting.

Free standing pyrolytic graphite tubes were substituted for coated ATJ molds. The castings produced exhibited relatively poor surfaces primarily due to excessive flaking or delamination of the pyrolytic graphite.

5. Material Finishing

An additional centerless grinder was acquired for scaleup development, and was installed and will be operational upon receipt of a diamond grinding wheel.

A program was initiated to determine the technical and economical feasibility of recovering and recycling uranium material collected as sludge during the machining of UC slugs.

A collection of sludge (~2.03 kg) was removed from a scrubber unit attached to one of the centerless grinders. The sludge — consisting of fuel material, impurities, and water — was dried (yielding ~2 kg of powder) and analyzed for chemical composition and particle size distribution. Results of the chemical analysis are given in Table 28000-3.

TABLE 28000-3
COMPOSITION OF UC SLUDGE COLLECTED IN SCRUBBER

Wt % U	Wt % C	O/U Ratio	Major Impurities (ppm)	
84.7	0.53	2.45	Fe 3000	Cu 150
			Al 250	Na 375
			B 5	Si 250
			Ca 200	Zn 250

Most of the impurities found are believed to have been picked up from the water in the scrubber. Based upon the O/U ratio, a minimum of two-thirds of the uranium is in the form U_3O_8 with the balance being UO_2 . The disposition of carbon was determined to be as follows:

Total carbon = 0.53 wt %

Free carbon = 0.33 wt %

Carbon as carbonate (CO_3) = 0.07 wt %

By difference, the amount of carbon combined as UC was calculated to be 0.13 wt %, which indicates that there is ~2.6 wt % of unreacted UC in the sludge.

A physical analysis of the dried sludge produced the results given in Table 28000-4.

TABLE 28000-4
PARTICLE SIZE DISTRIBUTION AND DENSITY OF
DRIED SLUDGE

Mesh Size	% of Total Analyzed	Density Bulk	Tap (gm/cc)	Particle Size* (μ)
+200	1.0	-	-	-
-200 +325	63.5	1.6	2.3	2.1
-400	35.5	1.7	2.5	1.9

*Fisher Sub-Sieve Sizer

About 99% of the powder passed through the 200 mesh screen. The residual 1% was found to consist mainly of glass and iron chips most likely picked up in the scrubber. The above data also indicates that 99% of the sludge is suitable from the standpoint of particle-size for recycle to the carbothermic reduction process.

Samples taken from mesh size fractions -200, +325, and -400 were fired in air at several temperatures to study material characteristics. No appreciable weight change occurred at temperatures up to 200°C, indicating that the sludge is definitely not pyrophoric and therefore will not require special handling. A 1% weight gain was observed at 400°C and further reheating to 1000°C resulted in an overall 0.5% weight loss, attributed to removal of residual carbon as CO₂. In general these results indicate that different mesh size fractions may be blended together and handled as a single mixture, and that the fuel is mostly oxide, as indicated by the chemical analysis in Table 28000-3.

A number of gram scale experiments were carried out to convert the sludge into UC. Results are given in Table 28000-5 where they are compared with results for synthesizing UC from reference UO₂. On the basis of carbon and oxygen content it does appear that good quality UC may be produced from sludge. A spectrographic analysis of the UC products for trace impurities indicates that all impurities were reduced to an acceptable level with the exception of boron (10 ppm vs allowable limit of 3 ppm) and titanium (200 ppm vs allowable limit of 75 ppm). Arc melting of the UC products may be effective as a means of reducing these impurities to within acceptable limits.

TABLE 28000-5

PREPARATION OF URANIUM CARBIDE ON A GRAM
SCALE FROM RECOVERED GRINDING SLUDGE

Run No.	Input		Reaction*		Weight Loss		Product Composition [§]	
	Oxide Identity	Carbon (Wt %)	Total Time (hr)	Time at Temp. (hr)	Exp. (Wt %)	Theo.† (Wt %)	Carbon (Wt %)	Oxygen (ppm)
SL-1B	Sludge	12.26	5-3/4	3/4	22.0		5.41	-
SL-1D	Sludge	12.04	5-3/4	3/4	22.1		5.09	-
SL-2C	Sludge	11.81	5-1/2	1-1/2	22.1		4.82	80
SL-2D	Sludge	11.71	5-1/2	1-1/2	22.1		4.70	565
SL-3A	Sludge	11.66	4-1/2	1-1/2	22.3		4.66	240
SL-2B	Ref. U ₃ O ₈	12.48	5-1/2	1-1/2	22.0	22.0	4.74	340
SL-3B	Ref. UO _{2.17}	11.96	4-1/2	1-1/2	19.2	19.3	4.86	340
SL-3C	Ref. UO _{2.07}	11.85	4-1/2	1-1/2	18.9	18.7	4.79	390

*Maximum temperature 1750°C, final pressure <0.2μ

†Based on oxide composition and assuming product is pure UC_{1.0}

§ Values are an average of three carbon analyses and two oxygen analyses

- No analysis

B. CLADDING AND PRESSURE TUBE MATERIALS DEVELOPMENT

The development of basic materials information for the fuel element cladding and the pressure tubes is the overall objective of this phase of the program. SAP is being considered for both cladding and pressure tubes while zirconium-based alloys are a backup for the pressure tube application. The work associated with each of these alloy types is discussed separately below.

1. SAP Alloys

The objectives of the SAP development activity are:

- a) Develop reliable sources for SAP alloys and fabrication capabilities for SAP tubing.
- b) Develop fabrication processes and product specifications for SAP tubing.
- c) Develop mechanical properties data on SAP alloys and define the effects of fabrication, strain, thermal, and irradiation history on the mechanical properties of SAP.
- d) Develop nondestructive and destructive inspection techniques for SAP billets, smooth tubing, and finned tubing.
- e) Maintain cognizance of the development and evaluation of improved SAP alloys in terms of HWO CR applications.

a. SAP Procurement

During this quarter technical liaison with the suppliers continued. Evaluation of a sample hollow billet in the 7-in. -diameter size provided by Montecatini is underway, and discussions on the possibility of a portion of their 7-in. -diameter billet order being provided in hollow rather than solid form are being conducted. Montecatini has indicated that delivery of the 2-in. billets will begin in June 1966. Shipment of the full order will be completed in October 1966.

The other supplier of billet material, Alcoa, will begin delivery on their order in April 1966, with completion in September 1966. Discussions with this supplier are continuing on the quality assurance requirements relating to the frequency and size of permissible inclusions determined by metallographic examination.

b. Fabrication and Mechanical Properties

Mechanical properties evaluations on cladding specimens extruded with varying reduction ratios are in progress. Ratios of 1000:1 and 30:1 were used to establish whether an effect due to extrusion ratio exists. Extrusion ratios within this range will be used to further define the effect of extrusion ratio on properties. The tubing configuration is shown in Figure 28000-1 (a and b). Tube extrusion blanks were cut from 7-3/4-in. -diameter. SAP 930 and XAP 001 billets, and the tubing produced represents a reduction of 30:1 from the powder compact stage. Blanks were taken from the billets so that final extrusion was both parallel and transverse to the billet extrusion axis in order to determine whether "cross extrusion" would affect the mechanical properties of the tubing (Figure 28000-2). Approximately 25 ft of tube representing each orientation was obtained for each alloy. The metallographic structures produced were compared with that of material produced under normal fabrication procedures. It was found that the homogeneity of the SAP 930 alloy with the light reduction shows greater variation from previously examined material than does the XAP 001 tube with the light reduction (Figures 28000-3a, b, and c and Figures 28000-4a, b, and c). Transmission electron microscopy, short-time tensile tests, and creep-rupture tests have been planned. The results of these tests will be used to guide further work aimed at the determination of optimum fabrication techniques for cladding and pressure tubes.

Mechanical test specimens of cladding tubes indicating defects have been selected from the nondestructive test results. A common cause for indications of defects in the tubing is gross segregation, (see Figure 28000-5). The effect of indicated defects on the short-term burst strength of cladding tube has been negligible; the tubing properties have been within the normal scatter of data.

Stress-rupture test specimens have been machined from SAP cladding produced for fuel element irradiation experiments, and their fabrication history has been recorded. The creep-rupture tests planned for the material, and the application of the material to fuel irradiation tests, are listed in Table 28000-6. Uniaxial tests will utilize specimens with fins removed so the test results will represent the tube wall. Biaxial tests will be internally pressurized tubes comprising specimens with and without fins removed.

Figure 28000-1. Cladding Tube Configuration of SAP Tubing

Figure 28000-2. Orientation of Test Specimens Relative to Starting Billet Extrusion Axis

Mag: 50X

Lab No. 6593-1-1

Figure 28000-3a. Cross Section of SAP 930 Cladding Tube Showing Structure with 30:1 Reduction Ratio and Extrusion from Blank Transverse to Billet Axis

Mag: 50X

Lab No. 6592-6-1

Figure 28000-3b. Cross Section of SAP 930 Cladding Tube Showing Structure with 30:1 Reduction Ratio and Extrusion from Blank Parallel to Billet Axis

Mag: 50X

Figure 28000-3c. Cross Section of SAP 930 Cladding Tube Showing Structure with 1000:1 Reduction Ratio and Extrusion from Blank Parallel to Billet Axis

Mag: 50X

Lab No. 6594-1-1

Figure 28000-4a. Cross Section of XAP 001 Cladding Tube Showing Structure with 30:1 Reduction Ratio and Extrusion from Blank Transverse to Billet Axis

Mag: 50X

Lab No. 6593-7-1

Figure 28000-4b. Cross Section of XAP 001 Cladding Tube Showing Structure with 30:1 Reduction Ratio and Extrusion from Blank Parallel to Billet Axis

Mag: 50X

Lab No. 6422-4-1

Figure 28000-4c. Cross Section of XAP 001 Cladding Tube Showing Structure with 1000:1 Reduction Ratio and Extrusion from Blank Parallel to Billet Axis

1. The first part of the report is a general introduction to the project. It describes the objectives of the study and the methods used to collect and analyze the data. The introduction also provides a brief overview of the results of the study.

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The billet size ordered from SAP suppliers for cladding tube extrusion is 2-in. diameter. Normal practice has been to extrude billets to a 1-1/2 in. diameter before machining extrusion blanks. It is desirable to avoid this step, to reduce fabrication costs. Consequently, the cladding tube was extruded directly from 2-in. blanks of both 7% and 10% Al_2O_3 alloys. The tubing configuration is shown in Figure 28000-1b. No major problems were encountered during extrusion; surface finish and dimensional tolerances were as specified.

c. Nondestructive Testing

This activity has emphasized development of eddy-current methods for testing finned tubing. Developments are underway for eddy-current procedures to determine discontinuity location in the tubing cross section; i. e., in the fins, near the tube outside surface, or near the inside diameter surface. The physical size and eddy-current response of natural discontinuity conditions are being correlated with standards containing artificial flaws.

The method being developed for locating discontinuities in the tube cross section involves combinations of coil type, operating frequencies, and sensitivity levels. Work to date has been performed using low eddy-current noise level tubing containing 1/2-in. -long electrical discharge machined (EDM) notches in the fins, on the outside surface of the tube wall, and on the inside surface of the tube wall. The depth of the notches has been 10, 15, 20, 30, 50 and 80% of the wall thickness. Results to date have shown that notches in the fins can be identified by using an encircling coil, operating at 64 MHz, and standardizing for a 10% notch at about 25% of recorder saturation. Operating at the same conditions but with a 4 MHz test frequency permits the detection of notches in all regions of the cross section. A comparison of the two traces then isolates those defects in the fins.

Studies have been initiated to correlate eddy-current indications from tubes standardized against EDM notch standards with metallographic results. Two XAP 005 alloy tubes have been evaluated in this manner. The indications in these tubes were found to have originated from two general types of discontinuities — inside diameter surface breaks and inclusions. The inside surface condition was indicated on the eddy-current trace by numerous closely-spaced indications having a response equal to EDM flows in the order of

10 to 25% of the wall. Metallographic studies showed the actual condition to be laps and folds up to 20% of the 0.027-in. wall thickness. The isolated indications on the eddy-current trace were found to be associated with single large or closely grouped small inclusions. Prior electron microscope and particle tests had shown this type of inclusion to be a ferromagnetic Fe-Al compound. Ferromagnetic particles give an eddy-current response greater than nonmagnetic inclusions or voids of equal size. Tests will be performed using a saturation coil to obtain a more realistic response from this type of inclusion.

A nondestructive test is desired to detect gross segregation of the aluminum oxide dispersion. Neither the radiographic nor the eddy-current tests used to date appear to offer a means of detecting this condition. Fast neutron activation is being considered for this application.

2. Zirconium-Based Alloys

The primary objective of this project is to define the conditions under which zirconium-based alloys can be reliably used for HWO CR pressure tubes. The principal uncertainty regarding the use of these alloys for pressure tubes is the effect of hydriding on the properties of the candidate zirconium-based alloys. The specific objectives for this activity are:

- a) To select an alloy composition with optimum corrosion and hydriding characteristics.
- b) To develop suitable protective coatings for inhibiting hydrogen absorption in the selected zirconium alloy.
- c) To establish the maximum hydrogen level permissible for the reliable performance of the selected zirconium alloy.
- d) To determine the effect of additives and impurities in the HWO CR coolant in contact with the candidate zirconium-based alloys.

a. Hydrogen Barrier Development

Screening tests of oxidized, nitrided, and aluminized coatings on Zircaloy-2 in gaseous hydrogen at 750°F were completed during the previous (October to December 1965) report period. Results of these screening tests have eliminated all brittle coatings except oxide films. The screening tests indicate that oxide coatings applied in air are better hydrogen barriers than coatings applied in pure oxygen. This may indicate the formation of an oxy-nitride coating that is superior to a straight oxide coating. Due to the extreme severity of the

gaseous hydrogen (increased hydrogen activity) retort testing and because of the susceptibility of the coatings to failure at defects (which cannot be repaired in pure hydrogen) it was not possible to determine differences in coating effectiveness as a function of the temperature of application or thickness of coating. Therefore, the gaseous hydrogen retort testing has not been pursued during this period.

b. Oxide Coatings

The behavior of oxidized specimens in pure hydrogen, at initial hydrogen pressures of 600 and 100 mm Hg, can be characterized by the pressure decay curves which indicate the gradual absorption of hydrogen prior to the typical catastrophic failure at a defect. This apparent indication of gradual oxide degradation confirms the observation of Shannon⁽¹⁾ that zirconium oxide exhibits a tendency to degrade, unless an oxidant is present. One Zircaloy sample oxidized in air for 8 hr at 750°F to a weight gain 0.78 mg/dm² was exposed to gaseous hydrogen at 750°F and a pressure of 100 mm Hg. The initial pressure decay was similar to that observed previously, 26 hours to drop from 100 mm to 10 mm Hg. The retort was refilled to 100 mm Hg pressure and the pressure again allowed to decay. In previous tests the second volume of hydrogen was absorbed in a few hours; however, the absorption rate for this sample was reduced so that 84 hr were required to absorb the second hydrogen charge (equivalent to approximately 250 ppm hydrogen in the sample). After the second volume of hydrogen was absorbed, the retort was opened to the atmosphere for 2 hr with the sample maintained at 750°F to reoxidize the sample and repair any defects in the oxide coatings. This was indeed the case, as 184 and 384 hr were required to absorb the third and fourth hydrogen charges, respectively. Visual examination of the sample after absorbing approximately 1000 ppm of hydrogen did not reveal signs of catastrophic failure. Metallographic examination of the specimen has not been completed. However, the results of this test indicate that defects in oxide coatings can be repaired in the presence of an oxidant. The results also indicate that thermal cycling from the oxide application temperature to ambient and back to test temperature may induce microcracks in the coatings resulting in ultimate failure of the material.

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1. D. W. Shannon, "Role of Oxidation Rate on the Hydriding of Zirconium Alloys in Gas Atmospheres Containing Hydrogen," HW-76562 Rev. Feb. 1963.

In subsequent experiments the major emphasis in the evaluation of zirconium oxide coating efficiency must be placed on the relationship of coolant composition, including water additions, to the rate of oxide degradation.

c. Aluminum Coating

It is believed that a ductile coating with a low hydrogen permeability will provide the most reliable protection for zirconium alloys exposed to an organic environment at HWOCR operating conditions. Aluminized coatings previously tested, while exhibiting acceptable hydrogen barrier characteristics, have resulted in the formation of brittle zirconium-aluminum intermetallic compounds which are susceptible to local defects. A coating (cladding) of unalloyed aluminum, however, would be expected to be less subject to defects and to the progressive failure observed for the typically brittle coatings previously evaluated. For this reason, experiments to provide and expose pressure bonded aluminum-Zircaloy-4 specimens were initiated.

Preliminary pressure bonding experiments performed on flat strip specimens were unsatisfactory due to inadequate sealing of the bonding capsules and improper fit up of the cladding to the sample. The bonding capsule has been modified to a 1/2-inch-diameter Zircaloy-2 cylinder in a close fitting aluminum tube. Aluminum end caps will be seal-welded to the tubes and evacuated to enhance the bonding characteristics.

The investigation of the aluminum coatings will involve the determination of the thermal stability of the coatings (growth rate of intermetallic diffusion layers) at HWOCR operating temperatures; the determination of the relationship between cladding thickness and the rate of hydrogen pick-up; and the effect of oxide and nitride films on the aluminum cladding.

d. Autoclave Tests

Autoclave tests of oxidized, nitrided and aluminized Zircaloy-2 specimens, duplicating those in the U-305 experiment, are continuing. The specimens are being exposed in Santowax OM at 750°F for a scheduled 5000 hours. Comparative ratios of hydrogen absorption will be measured. The source of hydrogen for these autoclave tests is pyrolytic decomposition of coolant.

A hydrogen probe has been constructed in an attempt to measure the hydrogen activity of organic coolants. The apparatus consists of a thin walled nickel tube connected to a vacuum system with the necessary pressure measuring equipment. Operation of the apparatus is relatively simple; the nickel tube inserted in the organic is evacuated to a known pressure and isolated. The hydrogen pressure in equilibrium with the organic readily diffuses through the nickel tube, thus increasing the pressure in the system. Some initial measurements will be made with the device during the next quarter.

e. Experimental Circulating Organic Loops

Two small loops are being constructed for use in the testing of zirconium alloys and zirconium alloy components under simulated HWOCR conditions.

The construction phase of the two loops has been nearly completed. The duct work has been completed and the electrical and plumbing work is in the final stages. The two loops are shown in Figure 28000-6.

C. SAP JOINING DEVELOPMENT

The objective of this task is to develop joining techniques for SAP fuel element components as required for the HWOCR program.

Installation of the 150 KVA magnetic force welder, used in development of welds for fuel rod end closures, was completed in late January (Figure 28000-7). During the checkout and preliminary operation of the welder it was found that some equipment modifications were desirable. This included redesign of one of the electrodes and other minor changes. The modifications were completed in mid-March.

To shake down the welder and become familiar with the machine operating characteristics, 83 welds were made using XAP 001 cladding of the WR1-951 irradiation test design (0.455 in. ID by 0.030 in. wall). Most of the 83 samples were used in arriving at rough machine settings and practice operation of the welder. At the conclusion of the shakedown phase the welds were room-temperature tensile tested. These tests resulted in weld failure in about half of the samples. Most of the weld failures occurred at about 80% of the strength of the specimens which failed in the cladding (33,000 psi avg. for end cap weld). At this point, an extensive study was made to determine the relationship

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Figure 28000-6. Organic Loops

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Figure 28000-7. Magnetic Force Welder

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between weld quality and welder settings. Tests were made at various settings to determine (1) at what interval during the welding cycle the magnetic force was being applied, and (2) how much force was being applied at different magnet gaps and current settings.

Using the information gained from the above studies and those made during shakedown of the welder, work was initiated on one of the cladding sizes used in the Piqua Core II design (0.486 in. by 0.030 in. wall). To date, about 170 welds have been made in defining welding parameters and in establishing component setup procedures. Room temperature tensile tests were performed in evaluating these closures. Failure in the tube wall vs failure in the weld was used as the criteria in defining the best operating parameters. The welded closures produced near the end of this exploratory series were generally failing in the tube wall, indicating 100% efficiency in short-term tensile tests. A series of welded closures is now being made using fixed setup procedures and welding parameters. These closures will be given both room and elevated temperature tensile tests to verify process reproducibility. Qualification testing of magnetic force welding (MFW) closures cannot proceed until process reproducibility has been demonstrated.

During the report period, evaluation of the Battelle Northwest Laboratories MFW end closures continued, with stress rupture and tensile testing. All welds were subjected to nondestructive inspection (dye penetrant, radiograph for weld quality, and resonance ultrasonics for wall thickness measurements). Ten welds were tensile tested at room temperature and at 850°F with the following results:

Test Temperature	Number of Samples	Ultimate Strength (psi)			Fracture Location	
		Average	Max.	Min.	Tube*	Weld
Room Temp	10	39,900	41,850	34,600	5	5
850°F	10	10,950	11,950	8,870	0	10

*Fracture location in the tube requires that no portion of the fracture passes through the weld zone.

An additional 42 stress rupture tests were completed this quarter bringing the total number of axial stress rupture tests to 55. Two tests are in progress, one specimen has accumulated over 1600 hr at 850°F and 4500 psi. The axial stress rupture data is summarized in Figure 28000-8. A preliminary regression analysis was performed to determine the relationship between stress level and rupture time for 42 of these MFW end closures.

Figure 28000-8. Axial Stress-Rupture Data for Magnetic Force Welded End Closures and a Comparison to Cladding Tube Values

The correlation coefficient and the coefficient of determination for this regression analysis are -0.564 and 0.319 respectively. Correlation coefficients greater than 0.900 are normally observed for stress rupture data on SAP tubing. The scatter in this stress-rupture-time data, is considerably greater than what would normally be expected for this type of testing. The regression equation will be recomputed at the completion of the two remaining stress rupture tests. Curves representing average values for XAP 001 tubing are plotted along with the data points from the tests on MFW end closures to indicate the general magnitude of the strength penalty which would be realized if the closure techniques did not improve. It is noteworthy, however, that these data are

from tensile stress-rupture tests and the loading of the fuel elements will be compressive when the difference is not expected to be as great. Testing in compression will be carried out when the fabrication parameters are established.

D. TESTING

The testing project objectives are:

- 1) To provide operational surveillance and data analysis for irradiation experiments in both the NRU and WR-1 reactors.
- 2) To design and fabricate SAP-UC irradiation experiments for the WR-1 reactor.
- 3) To establish the irradiation performance characteristics of UC fuels and SAP-UC fuel element assemblies.
- 4) To conduct and evaluate out-of-pile hydraulic tests on HWOCR fuel bundle concepts.

1. NRU Irradiation Tests

a. SAP Clad-UC Fuel Assemblies

During the report period, the U-305 experiment continued to operate satisfactorily in the U-3 loop of the NRU reactor. Since insertion in the NRU in late July 1965, the experiment has achieved an estimated peak burnup of 9250 Mwd/MTU through the end of March (exclusive of end peaking).*

During this report period, the computerized data reduction and plotting program became fully operational and data backlog was processed up to date. Because of time lags associated with data transmissions, analysis and display, results are obtained about one month following acquisition of the data by the data logger. Figures 28000-9 through 28000-16 are representative of the information being collected and analyzed. They provide insight into the experiment operation during the months of January and February 1966. Figures 28000-9 and 28000-10 are histograms showing the number of hours the experiment has operated at various power levels. During January and February the experiment accumulated 1165 hr above 80% design power and 248 hr below 80% design power (design power is 1660 kw).

*Values for the peak burnup have been corrected to account for the variation in power generation between the oxide and carbide strings.

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Figure 28000-9. Hours Operated in January 1966
vs Percent Design Power (1660 kw)

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Figure 28000-10. Hours Operated in February 1966
vs Percent Design Power (1660 kw)

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Figure 28000-11. Cladding Temperature History for January 1966

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Figure 28000-12. Cladding Temperature History for February 1966

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Figure 28000-13. January 1966 Beta Parameters for Cladding TC-23

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Figure 28000-14. February 1966 Beta Parameters for Cladding TC-23

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Figure 28000-15. January 1966 U-305 Coolant Water Content

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Figure 28000-16. February 1966 U-305 Coolant Water Content

The three SAP-clad carbide bundles on the upper half of the experiment are instrumented with a total of 12 thermocouples. Seven of these thermocouples are inserted in the cladding fins and five of these are still providing meaningful temperature signals. The history of these measured cladding temperatures for January and February are shown in Figures 28000-11 and 28000-12. The only significant operational difficulty occurred in February when the peak (calculated) cladding temperature gradually approached the maximum limit of 860°F. As seen in Figure 28000-13, this maximum was reached while the experiment was operating near the 1.660 Mw design power level. This increase was also detected by the gradual increase in the beta parameters as shown in Figure 28000-14. Changes in the operational procedure were initiated on March 5, 1966, which halted the cladding temperature increase. These changes included (1) restricting the experiment power production by limiting the loop ΔT to 43°C (43°C is the nominal value) and (2) increasing the organic coolant water content to 300 ppm. Subsequent to this action the temperature increase stopped, and during the following two weeks the peak temperature decreased by 15°F. The coolant water content during January and February is shown in Figures 28000-15 and 28000-16.

At the close of this report period (March 31, 1966), the status of the SAP-clad carbide fueled section of the U-305 experiment was as follows:

Estimated peak UC Burnup	9250 Mwd/MTU
Total experiment power production	1.710 Mwt
Maximum measured cladding temperature	782°F
Estimated Peak Cladding Temperature	837°F
Estimated Peak fuel center temperature	1900 - 2000°F
Coolant Inlet temperature	300°C (572°F)

b. Zr Clad-UO₂ Fuel Assemblies

The performance of the Zircaloy-clad oxide fuel elements has continued to be satisfactory. At the close of the report period, the Zircaloy-clad fuel bundles had achieved an estimated peak burnup of 7500 Mwd/MTU (exclusive of end peaking).

2. WR-1 Irradiation Tests

a. EXP-WR1-951 and -952

The design operating characteristics of these tests were listed in the previous quarterly report (AI-CE-17). During the report period, fabrication of the -951 test assembly was completed and the assembly was readied for insertion in the WR-1 reactor. A photograph of the completed assembly is shown in Figure 28000-17. Insertion of -951 test assembly is scheduled for April 1966.

Installation of the instrumentation system at the WR-1 for monitoring the HWO CR fuels tests was nearly completed. The major item yet to be installed is the 400-channel data logger system which is expected to be shipped to the WR-1 in late April 1966. Analog instrumentation is at the site and will be used for data retrieval during the initial phases of the -951 test.

Fabrication of the -952 test assembly proceeded at a pace somewhat slower than that projected in the previous quarterly report as a result of a reduction in the number of irradiation tests at WR-1. At the close of the report period the -952 fuel rod fabrication was near completion with only those rods containing in-fuel thermocouples remaining to be completed.

b. EXP-WR1-953 and -954

The design operating characteristics of these tests are listed in Table 28000-7. The external configuration of the test assemblies is identical to that of the -951 and -952 assemblies. The most notable difference between the -951, -952 tests and the -953, -954 tests (with the exception of the design operating characteristics) is the fact that only 7 rods in each fuel bundle of the -953, -954 tests are loaded with fuel (Figure 28000-18). The remaining 12 rods in the bundles are filled with helium. This fuel loading schedule is dictated primarily by the excessive reactivity load which would be imposed on the WR-1 core if all 19 rods in each bundle were to be loaded with fuel and operated at peak rod power levels of 25 kw/ft (excluding end peaking).

The -953 test assembly differs from the -954 test assembly primarily in the degree of the slug surface cleanup required to achieve the desired fuel diameter. The cleanup is considerably less in the -953 test than in the -954 test. Only 2 to 3 mils of fuel (radially) was removed for the -953 test assembly,

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Figure 28000-17. Completed EXP-WR1-951 Test Assembly

TABLE 28000-7

EXP-WR1-953 and -954 DESIGN AND OPERATING CHARACTERISTICS

Parameter	EXP-WR1-953	EXP-WR1-954
1) Fuel		
Enrichment (wt % U 235)		
Active stack	4.9	4.9
End slugs	1.5	1.5
Diameter (in.)	0.4535 ± 0.0005	0.4535 ± 0.0005
Fuel removal to obtain diameter radially (in.)	0.002 - 0.003	0.008 - 0.010
Carbon content (wt % C)	5.15 ± 0.3	5.15 ± 0.3
Stack length (in.)		
4.9 wt % fuel	not set	39.5
1.5 wt % fuel	1 in. both ends	1 in. both ends
Total	not set	41.5 in.
2) Cladding		
Material loading schedule	Figure 28000-18	Figure 28000-18
Wall thickness (in.)	0.022 ± 0.002	0.022 ± 0.002
Inside diameter (in.)	0.4585 ± 0.0005	0.4585 ± 0.0005
Nominal fin height (in.)	0.062	0.062
Overall fuel rod length (in.)	43.75	43.75
3) Design Performance Data		
Peak rod power (kw/ft) not including end peaking	25	25
Peak sheath temperature (°F) - for a coolant inlet temperature of $\sim 645^{\circ}\text{F}$	850 - 870	850 - 870
Peak fuel center temperature (°F) - not including end peaking	2100 - 2200	2100 - 2200
Coolant velocity (ft/sec) - based on a header to header ΔP of ~ 150 psi	27	27
Peak fuel specific power (kw/kg)	59	59
Target peak burnup (Mwd/MTU) - not including end peaking	To failure	20,000 to 25,000
4) Specific features	Instrumented to obtain a measure of coolant mixing in bundle interior sub-channels	Instrumented to obtain data on end peaking to check physics model for demonstration plant fuel element design

5-8 wt% Al_2O_3 RODSP, J, K, Z,
X, W, L, C, S, H10 wt% Al_2O_3 RODSN, F, E,
Y, R, T, D, B, M

whereas approximately 8 to 10 mils of fuel (radially) was removed for the -954 test assembly. In addition, the -953 test incorporates a flux couple for monitoring the temperature effects of minimum fuel slug surface cleanup in addition to determining the effects of flux/temperature peaking at the ends of the fuel stack.

During the report period, the fuel and cladding fabrication for the -953 and -954 tests was completed.

c. WR-1 Test Evaluation

During this reporting period, plans and techniques were developed for the acquisition, analysis, and interpretation of data to be obtained from the EXP-WR1-951 test.

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Figure 28000-18. Fuel Loading
and SAP Position Scheme for
EXP-WR1-953 and -954

This data handling system is separated into two distinct functions: (1) on-site data interpretation and (2) computer analysis and display of the data acquired by the data logger.

On-site data interpretation is required to allow site personnel to evaluate the performance of the experiment on a daily basis, observe trends, and initiate the required control action. Procedures have been developed and transmitted to the site personnel for this purpose. The procedure involves plotting all experiment data and parametric information derived from the data on graphs with a common time scale so that various parameters can be compared for cause and effect type relationships. Because of the variable moderator height in WR-1, unique analytical techniques are required to assess properly experiment power production, peak power ratio, and the fouling parameters. Since the total experiment power output at a constant peak power varies with the moderator height, as shown in Figure 28000-19, the channel power or channel ΔT cannot be used to normalize data for power fluctuations. The normalizing factor is the peak power ratio, which is the actual channel power divided by the design channel power at the same moderator height. Experience with the U-305

Figure 28000-19. EXP-WR1-951
Experiment Power Output vs
Moderator Height

experiment has demonstrated the efficiency of a normalization parameter concept as an operational tool. In the case of the WR-1 experiments, this parameter is also affected by moderator height, and the technique for determining this parameter necessarily differs from that being used with the U-305 experiment. The film formation parameter for the WR-1 experiments is defined as the film ΔT (cladding temperature - bulk coolant temperature at the same elevation) divided by the peak power ratio.

In addition, plans were finalized for the computer analysis and display of the data acquired by the on-site data logger. The data logger will record a set of data every hour which will be subsequently analyzed

in detail to give a complete picture of the operation of an experiment.

3. Out-of-Pile Testing

a. Fuel System Evaluation

The results of a comprehensive evaluation of all available in-pile data on unalloyed uranium carbide were published in AI-CE-Memo-9, "The Irradiation Behavior of Unalloyed Uranium Carbide." Uranium carbide fuel specifications were written to cover material and fabrication processes for the EXP-WR-1-953 and -954 irradiation experiments.

b. Component Compatibility

A program is underway to ascertain the extent of the compatibility in UC-SAP contact bonded and UC-SAP-Pb bonded configurations. During this report period, fabrication and assembly of all UC-SAP contact bonded capsules scheduled to date were completed. Testing was initiated on 35 additional capsules during the quarter, bringing the total tests in progress to 62. These tests cover a range of temperature and extended times as shown in Figure 28000-20.

Figure 28000-20. Hyperstoichiometric UC-SAP Contact-Bonded Compatibility Studies

Presently five UC-SAP-Pb bonded capsules are undergoing heat treatment for extended times at 800 and 700°F.

Evaluation of compatibility capsules during this quarter yielded highly encouraging results. No reactions were observed between the hyperstoichiometric uranium monocarbide and SAP specimens which were heat treated for 6500 hr at 700 and 800°F. However, the capsules which were tested at 1100°F for various times produced somewhat inconsistent data. No reaction was noted after 500 hr at temperature and only a localized reaction (3.2 x 1.6 mils) was observed after 4000 hr. However, a test at an intermediate time of 2000 hr resulted in an extensive reaction between the fuel and SAP. This apparent inconsistency is thought to be caused by variations in contact pressure between the specimens.

Lead-bonded specimens, which were tested for 2000 hr at 700 and 800°F revealed no interactions.

c. Thermal Contact Conductance*

The heat transfer characteristics between UC fuels and SAP cladding have a very pronounced effect on the centerline fuel operating temperatures, and thus have a strong influence on specific power limitations.

During the reporting period a series of measurements to determine the effective thermal contact conductance of the UC-SAP interface were performed as a function of gas pressure up to 75 psi. Interfacial temperature was approximately 950°F, with contact pressures of 40 to 300 psi.

The results of these determinations are summarized in Figures 28000-21 and 28000-22. Figure 28000-21 shows surface conductivity, S , as a function of argon pressure, P . Over the range of pressures tested, the data can be approximated by a linear relation,

$$S = 1.03 \times 10^{-1} + 3.89 \times 10^{-4} P \text{ watts/cm}^2 \text{ } ^\circ\text{C.}$$

During these measurements, the UC-SAP mechanical load at the interface was held constant at approximately 40 psi.

Figure 28000-21. UC-SAP Surface Conductivity as a Function of Gas Pressure

Figure 28000-22. UC-SAP Surface Conductivity as a Function of Interface Pressure

*Argon and xenon were used for these tests to simulate end-of-life conditions.

Figure 28000-22 shows surface conductivity, S , as a function of the mechanical load (L) at the interface. These data were taken in xenon at 17 psia. Again the data can be expressed by a linear equation,

$$S = 6.50 \times 10^{-2} + 9.17 \times 10^{-5} L \text{ watts/cm}^2 \text{ } ^\circ\text{C},$$

valid between $L = 40$ psi and $L = 300$ psi.

Measurements taken in vacuum tended to be erratic, yielding values of surface conductivity ranging from 0.043 to 0.237 watts/cm² °C.

Profilometer traces of the UC and SAP surfaces used in these studies indicated a finish of 150μ/in. rms for the UC surface and 50μ/in. rms for the SAP surface.

d. Hydraulic Testing

The following hydraulic tests (low temperature water tests) were completed during the report period.

(1) Hydraulic Tests on Irradiation Test Mockups

During this report period, hydraulic tests were completed on a mockup of the EXP-WR1-951 irradiation experiment. The basic components of this model are shown in Figure 28000-23. The ball and socket joint, located above the exit hardware, was for hydraulic test purposes only and is not a component of the EXP-WR1-951 experiment. A general description of the hydraulic test models which have been tested is presented in Table 28000-8. The test results and evaluation on the first three models were reported in the previous quarterly report.

Figure 28000-23. WR1-951 Hydraulic Test Model

TABLE 28000-8
HYDRAULIC TEST MODELS - GENERAL DESCRIPTION

Model	Clad Design	Pack*	Array	Length (ft)	No. of Bundles	Rods per Bundle	Outer Tube ID (in.)	Spacers
1	EXP-NRU-305	Close	Circular	4	1	19	3.104	Mod. EXP-NRU-305
2	EXP-WR1-951	Close	Hexagonal	3-2/3	1	19	3.250	EXP-WR1-951
3	EXP-NRU-305	Open	Circular	3	2	19	3.250	Dev. WR-1
4	EXP-WR1-951	Close	Hexagonal	9	2	19	3.250	EXP-WR1-951

*Close pack indicates fin-to-fin contact. Open pack, as used here, indicates 0.026-in. nominal clearance between the fins of adjacent tubes.

Friction factors for spiral finned SAP cladding were calculated based on the pressure drop data from these experiments. These friction factor data were correlated by the equation:

$$f = 0.201 N_{Re}^{-0.22385} \quad (\text{Eq. 1})$$

where:

f = Darcy friction factor

N_{Re} = Reynolds number

This summary correlation is plotted in Figure 28000-24 along with results of other investigators, using Moody's smooth tube correlation as a reference.⁽¹⁻⁷⁾ Based on the equivalent diameter concept, several investigators concluded that friction factors of complex geometries, such as rod bundles, agree within $\pm 15\%$ (at $N_{Re} > 5000$) of Moody's smooth tube correlation.^(1,2,3) These conclusions are consistent with the data shown in Figure 28000-24. Of significance is the fact that friction factors of finned rod bundles are consistently lower than smooth tube friction factors.

Contraction and expansion coefficients were calculated for the various fuel rod spacers used in the hydraulic test models. Equations used for these calculations are presented below:

$$\Delta P_s = \frac{K_1 V_1^2 \rho}{2 gc} \quad (\text{Eq. 2})$$

where:

ΔP_s = pressure drop across the spacer

K_1 = geometrically dependent loss coefficient (contraction)

V_1 = velocity at the smaller flow area,

and

$$\Delta P_s = \frac{K_2 (V_1 - V_2)^2}{2 gc} \rho \quad (\text{Eq. 3})$$

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Figure 28000-24. Comparison of Darvy Friction Factor vs Reynolds Number

where:

V_2 = velocity at the larger flow area

K_2 = geometrically dependent loss coefficient (expansion).

Results of these calculations are listed in Table 28000-9.

TABLE 28000-9
COEFFICIENTS OF CONTRACTION AND EXPANSION

Spacer Design	Average K_1		Average K_2
	Entrance	Center	Exit
Modified U-305	0.95	-	0.17
EXP-WR1-951	1.34	0.40	-
WR1-Developmental	1.35	0.53	-

Pressure drops across the entrance, center, and exit sections of the EXP-WR1-951 model are shown in Figure 28000-25. Each of these sections encompassed one (entrance - exit) or two (center) fuel rod spacers. Pressure drops across the entrance hardware were obtained by subtracting previously measured pressure drops across an EXP-WR1-951 entrance spacer, from the measured pressure drops across the entrance section of the EXP-WR1-951 model. The results, included with Figure 28000-25, are identified as entrance hardware pressure drops. Pressure drops across the exit hardware could not be similarly ascertained because pressure drops across an EXP-WR1-951 exit spacer have not been measured. This is of more academic interest, however, since the exit section pressure drop was only about 5% of the total pressure drop across the entire model.

Using the experimental data, pressure drops were calculated for an EXP-WR1-951 fuel element, operating in HB-40 coolant at 570°F. The results of these calculations are shown by Figure 28000-26.

Figure 28000-25. Pressure Drop Across Entrance, Center, and Exit Sections of WR1-951 Hydraulic Model

Figure 28000-26. Coolant Pressure Drop Across an EXP-WR1-951 Fuel Element (HB-40 at 570°F)

During testing, the four models were viewed through the plexiglass tubes. At all flowrates investigated, none of the cladding tubes were observed to vibrate. At maximum flowrates, the flux monitoring tubes of the EXP-WR1-951 model were observed to vibrate with an amplitude of about 1/16-in. These tubes were supported at each fuel rod spacer, and due to their size (1/8-in. by 0.010-in. wall) were extremely flexible between supports. The flux monitoring tubes were removed from the EXP-WR1-951 irradiation experiment as a result of these results.

The ball and socket joint, attached to the hanger tube of the EXP-WR1-951 model, permitted easy manual rotation of the model. However, water flow-rates up to 355 gpm (~30 ft/sec velocity) failed to cause any discernible rotation of the model. Consequently, it is felt that there will be no significant torque on the hanger tube of the EXP-WR1-951 fuel element during in-pile tests.

4. TF-3 Tests

The TF-3 loop was completed and shakedown accomplished by January 21, 1966.

The first portion of an end plate ΔP testing program was planned and completed during this report period. Flat end plates of the type shown in the 1000 Mwe preliminary conceptual design report (AI-CE-Memo-6) and stepped end plates of the type shown in the Demonstration Plant Conceptual Design Report (AI-CE-Memo-25) are being tested. Pictures of each end plate type are shown in Figure 28000-27. Six-inch-long rod stubs are attached to the end plates to simulate fuel rods. Two end plates are tested back-to-back and pressure loss is determined over a range of degrees of misalignment and velocity. Tests have been completed on the 4.26 in. flat and stepped end plates. Table 28000-10 lists the empirical equations which have been found to best fit the reduced data, and are applicable for Santowax OM (10% HB) at 594°F where ΔP is in psid and V is in fps. A plot of pressure drops corresponding to a velocity of 30 ft/sec in the rodless portion of the specimen for flat and stepped end plates of various degrees of misalignment are shown in Figure 28000-28.

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Figure 28000-27. TF-3 End Plate Test Sections

TABLE 28000-10
END PLATE PRESSURE DROP TEST RESULTS

Run No.	End Plate Type	Degrees Misaligned	Best Fit Equation	Experimental Data Standard Deviation	Best V^2 Fit	Experimental Data Standard Deviation
1	Flat with Unfinned Rod Stubs	0°	$\Delta P = 0.01057V^{2.07375}$	±1.5%	$\Delta P = 0.01317V^{2.0}$	±2.3%
2	Flat with Unfinned Rod Stubs	12°	$\Delta P = 0.012228V^{2.25734}$	±1.8%	$\Delta P = 0.02697V^{2.0}$	±4.2%
3	Flat with Unfinned Rod Stubs	20°	$\Delta P = 0.01452V^{2.15421}$	±1.7%	$\Delta P = 0.02375V^{2.0}$	±3.0%
4	Flat with Unfinned Rod Stubs	30°	$\Delta P = 0.01514V^{2.15763}$	±1.1%	$\Delta P = 0.02473V^{2.0}$	±2.7%
9	Stepped with Unfinned Rod Stubs	0°	$\Delta P = 0.01334V^{2.06636}$	±1.1%	$\Delta P = 0.01634V^{2.0}$	±1.3%
10	Stepped with Unfinned Rod Stubs	12°	$\Delta P = 0.01625V^{2.10127}$	±0.8%	$\Delta P = 0.02224V^{2.0}$	±2.1%
11	Stepped with Unfinned Rod Stubs	20°	$\Delta P = 0.01743V^{2.10576}$	±2.1%	$\Delta P = 0.02415V^{2.0}$	±2.1%
12	Stepped with Unfinned Rod Stubs	30°	$\Delta P = 0.01725V^{2.09061}$	±0.5%	$\Delta P = 0.02279V^{2.0}$	±1.9%
13	Stepped with Unfinned Rod Stubs	45°	$\Delta P = 0.01507V^{2.15178}$	±1.1%	$\Delta P = 0.02415V^{2.0}$	±2.4%
14	Stepped with Unfinned Rod Stubs	60°	$\Delta P = 0.01543V^{2.02129}$	±2.6%	$\Delta P = 0.01652V^{2.0}$	±2.6%
21	Stepped with Finned Rod Stubs	0°	$\Delta P = 0.00999V^{2.04224}$	±2.8%	$\Delta P = 0.01144V^{2.0}$	±4.0%

5. TF-3 Fuel Assemblies

Two fuel bundle mockup assemblies for installation in the TF-3 test loop were completed. Each bundle is 43-1/4 in. long and consists of 37 rods of finned aluminum tubing fastened to Zircaloy end plates. Thirty-one of the rods have a 0.669-in. OD, measured over the fins, and are held in a hexagonal array. The fin height is 0.080 in. Six additional rods having a 0.566-in. OD over the fins are machined to fit the outside corner positions to form a cylindrical bundle. Final machining of the bundle to a 4.23-in. OD was accomplished by casting each assembly in wax to form a

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Figure 28000-28. End Plate Pressure Drops for a Velocity of 30 fps

rigid cylinder and turning on a lathe. The two bundles fit inside a 4.26-in. ID pressure tube test section, with a nominal diametral clearance of 0.030 in. Each bundle has a stepped Zircaloy end plate at one end and flat end plate at the other.

Tests will be conducted with the two bundles orientated so that the matching end plates are back-to-back to compare their hydraulic characteristics. Pressure taps located on the test section will be used to measure the pressure drop across the end plates and along the finned tubes. Additional pressure losses resulting from misalignment of the two bundles will be measured for each type of end plate.

6. TF-10 Testing

Results of hydraulic tests for the fuel string described in the last quarterly (see page 2800-51) are given in Section 21000. The bundles from this test have been examined and measurements taken at specific locations. These bundles were installed in the TF-10 facility and a 500 to 1000-hr continuous run was initiated with the objective of determining pressure tube and bundle wear. In addition, fuel bundle vibration will be monitored with Linear Displacement Sensing Equipment.

7. Coolant Mixing Tests

Several experimental methods for determining the degree of coolant mixing in multi-rod fuel bundles were investigated. Methods investigated included:

- a) The electrically heated bundle method using cartridge type electrical heaters fitted inside finned tubes with thermocouples embedded in the fins to measure enthalpy rise and temperature differences between sub-channels. Other measured quantities include the frictional head loss per unit length and the power generated by the heaters. Mass flowrate is computed from the indirect pressure drop measurements.
- b) The counter flow heat exchanger method using preheated diphenal coolant which is circulated through the finned tube. Measurement techniques are similar to those employed for the electrically heated bundle method with additional thermocouples to measure enthalpy change of the organic coolant.
- c) The dye injection method followed by photometric analysis of downstream samples.
- d) The salt injection method which is similar to the dye injection. Distribution of salt is detected by conductivity cells embedded in the fins at various stations downstream of the injection point. Random fluctuations

in conductivity which propagate with the coolant are detected, continuously monitored, and analyzed by noise analysis techniques to measure axial flow distribution.

The salt injection method was selected over other methods primarily because of the successful development of noise analysis techniques for measuring flow directly. Preliminary tests were initiated to ascertain the reliability of this method to determine cross flow.

III. NEXT REPORT PERIOD ACTIVITIES

A. FUEL MATERIALS

1. Feed Material Process Refinement

During the next report period, refinement of the UC carbothermic reduction reaction will be carried out through a series of high vacuum sintering treatments at temperatures in the 1850 to 2050°C range.

2. Feed Material Process Scaleup

A summary report on the UC feed material scaleup development will be completed during the next report period.

3. UC Skull Casting Scaleup

During the next report period, scaleup of the UC skull casting process will be continued with major emphasis on eliminating the mold cracking problem and casting large yields of high quality hyperstoichiometric UC slugs ranging in diameter from 0.490 in. to 0.290 in. Modular molds will be fabricated and evaluated. A preliminary process specification based upon current casting scaleup development will be issued.

4. Surface Finish Casting Development

During the next report period, cast UC slug surface improvement studies will be continued.

5. Material Finishing

Scaleup centerless grinding development for the HWOCR prototype fuel will be continued during the next report period. Parameter development will be

carried out on the recently installed Cincinnati grinder using a 3-in. -wide by 14-in. -diameter diamond wheel. Efforts will be directed toward producing required surface finish on UC slugs with minimum surface removal.

B. CLADDING AND PRESSURE TUBE MATERIALS DEVELOPMENT

1. SAP Alloys

The project will continue technical liaison with SAP suppliers. The known areas requiring attention are the establishment of acceptability of hollow 7-in. -diameter billets from Montecatini, and negotiation of metallographic examination standards for Alcoa billet material.

Mechanical tests of cladding tube material with known fabrication histories and nondestructive test results will be continued. Mechanical tests of tubing produced by direct extrusion from 2-in. blanks will be initiated.

2. Zirconium-Based Alloys

A decision was made near the end of the report period to reduce the effort on the development of zirconium alloys for HWO CR applications. As a result of this decision, an orderly termination of activities by AI-CE under this sub-account is planned. Work on the loops will be suspended following completion of the electrical and plumbing work that is in progress and the loops will be placed in standby condition. The companion zirconium samples from the U-305 experiment will be removed from the autoclave, after the scheduled 5000-hr exposure, and evaluated. The initial measurements to evaluate the proposed technique for determining the hydrogen activity in the organic will be completed.

C. FUEL ASSEMBLY

1. SAP Joining Techniques

During the next report period, welding development on the Piqua Core II size cladding will continue and a preliminary process specification will be written. An auxiliary chamber for the magnetic force welder will be designed. This chamber will allow several rods to be backfilled and welded at one time. The tensile testing program will be completed.

D. TESTING

1. NRU Irradiation Tests

The U-305 experiment was removed from the NRU on April 18, 1966, as a result of evidence of a defected fuel element. Burnup achieved at the time of removal is estimated to be 10,030 Mwd/MTU. Hot cell examination will be started late in the next quarter. Alternate 1 will be inserted and operated for coolant technology studies.

2. WR-1 Irradiation Tests

a. EXP-WR1-951 and -952

The -951 test is presently scheduled to be inserted in the WR-1 reactor in early April 1966. Barring unforeseen problems with either the test assembly or the reactor operation, and assuming a 50% plant factor for the WR-1 reactor during the initial phases of operation, the -951 fuel assembly should achieve a peak burnup of 1600 Mwd/MTU during the next report period.

The -952 test fabrication effort will be completed and the assembly shipped to WR-1. Although the schedule for insertion of the -952 has not been firmly established, insertion of the -952 irradiation may occur late in the next report period.

b. EXP-WR-1-953 and -954

Design and fabrication efforts on these tests will continue. It is presently planned to complete the -954 test before the -953 test. This scheduling is a result of the inclusion of additional instrumentation in the -953 test. Present plans call for shipment of the -954 test to the reactor in late June 1966, with the -953 test following in mid-September 1966.

c. EXP-WR-1-955 and -956

Test plans for these experiments will be prepared during the next report period. Preliminary design and analysis efforts will be initiated.

d. WR-1 Test Evaluation

With the insertion of EXP-WR-1-951 test, data collection, analysis and interpretation will be initiated.

3. Out-of-Pile Tests

a. Fuels Prepractice Evaluation

During the next report period, the extension and verification of UC-SAP compatibility relationships will be continued with the evaluation of UC-SAP contact bonded capsules and the fabrication and testing of UC-SAP-Pb bonded capsules.

b. Hydraulic Testing

During the next report period, end plate pressure drop tests will be completed. Hydraulic tests will be performed on the two 37-rod fuel bundle mockup in the TF-3 water loop. Pressure drop variation with the degree of misalignment between fuel bundles will be measured. Information on fuel bundle wear and vibration will be obtained in the FF-10 loop.

c. Coolant Mixing

During the next report period, work will be continued on the development of the noise analysis method for determining coolant mixing. Hydraulic tests of a bi-channel model will be conducted to determine the cross-flow between two adjacent SAP cladding tubes.

After the bi-channel model tests are completed, analytical work will be initiated to formulate a mathematical model for predicting coolant mixing in a multichannel model. These predictions will be verified by experimental work.

Design of the fuel bundle mixing test section for the TF-3 loop will be initiated.

d. Mechanical Testing

During the next report period, efforts will be initiated to design axial loading tests on fuel rods and fuel assemblies. End plate strain and deflection tests will be initiated.

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PROGRAM PLAN**ACTIVITY:** Core Component Handling and Storage
On Power Refueling**ACCOUNT:** 81000**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 WORK 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.

Preliminary room temperature load/deflection tests have been run on a number of representative metallic face seals. The preliminary tests show that the seals in most cases were extremely sensitive and some seals permanently set when compressed to the upper limit of the deflection range recommended by the manufacturers. As a result, discussions have been held with a number of suppliers aimed at improving the performance through minor changes in design.

A proposal has been received from a manufacturer for development of two snout seal designs. This is being evaluated for its value in augmenting the current program of testing commercially available designs.

C. FUEL CHANNEL LATCHES

Design detailing was initiated for the fabrication of a fuel latch assembly for load testing. Design work was also initiated for the fabrication of a two-dimensional model.

D. FUEL LATCH CAN

Design detailing was initiated for the fabrication of a fuel latch can assembly for flow testing and latch load testing.

E. MECHANICAL COUPLING TEST

Test sections have been designed and fabricated for testing four mechanical couplings which are presently on hand. The mechanical couplings will be evaluated for application as disassembly joints for pressure housings and as possible backup to the present refueling machine coupling concept.

F. "CANDU" END SEAL MECHANISM TEST

The purpose of this test is to determine the effect of organic deposits on the operation of a complex mechanism. A specification listing the test requirements has been written and a "CANDU" end seal minus the seal was received from AECL, Toronto, for testing.

G. END CLOSURE SEAL

An end closure seal and a test fixture were ordered for organic loop testing.

H. BEARING TESTS

Discussions have been held with a number of vendors for the purpose of ordering ball bearings to simulate operation of the ball screw in organic coolant.

III. NEXT REPORT PERIOD ACTIVITIES

The following activities are planned for the next report period:

- a) Flexure tests on the flexible metal hose will be completed.
- b) Ball valve sealing tests will be completed.
- c) The mechanical coupling test will be completed.
- d) Test of the "CANDU" end seal mechanism in hot organic will be completed.
- e) A full-scale two-dimensional working model of the fuel channel latch will be fabricated.
- f) A fuel channel latch and a latch can will be placed on order.
- g) Test fixtures for the snout seal tests will be designed and built.
- h) Additional metallic seals will be procured for "Snout Seal Testing."
- i) Preliminary test requirements will be established for the following:
 - 1) Feasibility test for removal and replacement of fuel latch retainers.
 - 2) Feasibility test for operation of fuel latches under static and dynamic loading.
 - 3) Tests of ball and sleeve type bearings for compatibility with organic environment and load capacity.
 - 4) Leak testing of rotary joints for coolant connections.
 - 5) Water flow tests of refueling system components in a simulated fuel channel to determine pressure drop variation and flow patterns during a refueling cycle.
 - 6) Mechanical coupling leak tests for coolant connections under cyclic pressure conditions.

PROGRAM PLAN	
ACTIVITY: Large Components	ACCOUNT: 31000

I. PROJECT OBJECTIVES

The overall objective of this effort is to qualify some of the major components in the main coolant loop of the HWO CR demonstration plant that will be designed and supplied by the manufacturers prior to their installation in the Demonstration Plant. The principal items of equipment requiring such qualification, or proof-testing, are the main coolant loop pumps and valves. The proof-testing will provide assurance of the adequacy of the design and fabrication of this equipment, and assurance of the reliable performance of these components in the Demonstration Plant.

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

During this report period, the HWO CR Program Plan was revised and updated, with the objectives and work scope of this effort altered to reflect current program philosophy and anticipated budget levels. An important part of this revision was the shift of emphasis from a development approach to that of proof-testing of components.

A preliminary specification was prepared for the HWO CR main coolant pumps. It is planned that this specification will be adapted as required for use as the main coolant pump specification for the HWO CR plants. It will also be used for the purchase of an extra pump seal (along with the Demonstration Plant pumps) for proof-testing.

A preliminary specification was written for the HWO CR main coolant blocking valve stem seal specification and a design specification drawing were prepared during this report period. These documents define in detail the requirements of the seal and will subsequently be issued for procurement of seals for proof-testing.

III. NEXT REPORT PERIOD ACTIVITIES

The principal effort during the last quarter of FY 66 will be that of initiating the design of a Seal Test Facility (STF). This facility will be used to proof-test the main coolant pump and valve seals selected for the Demonstration Plant. Such testing will precede the proof-testing of the complete pumps and valves in a Large Organic Components Test Facility (LOCTF) to be designed and constructed at a later date. Completion of the STF design is scheduled to occur in the first quarter of FY 67.

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