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APPLICATION-REACTOR EXPERIMENTS

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LCRE AND SNAP 50-DR-1 PROGRAMS
ENGINEERING PROGRESS REPORT
January 1, 1963 - March 31, 1963

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I. SUMMARY

Prepared under the Direction of the
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I. SUMMARY

This report covers work accomplished during the first quarter of 1963 on the Lithium-Cooled Reactor Experiment and the SNAP-50 powerplant development program. It is submitted to the U. S. Atomic Energy Commission in partial fulfillment of Contract AT(30-1)2789.

The report is summarized below and presented in detail in the body of the report which follows.

A. LITHIUM-COOLED REACTOR EXPERIMENT

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A soak test of 906 fuel pins in lithium was completed as part of a study of fuel pin reliability. Variables in the test were end-cap material and annealing temperatures. No annealed pins have been found which leaked lithium. The program to improve the strength of Cb-1 Zr alloy fuel cladding continued. It has been found, in short time tests, that the addition of 500 to 800 ppm of nitrogen gives a 30 percent improvement in strength and 1000 ppm of carbon gives a 100 percent gain.

The problem of helium generation in beryllium by the n, 2n and n,α reactions is being investigated analytically and experimentally. New cross section information indicates that previous estimates may be low by a factor of two. Samples of beryllium are being irradiated in the Battelle Research Reactor. Tests are being planned for the ETR.

A four-inch pressure vessel completed a 10,000 hour test at 2000F and a ten-inch vessel completed 7800 hours at 1900F. The full scale pressure vessel test failed at startup due to a rupture in the inert gas containment jacket. The vessel was undamaged and repairs to the test are in progress.

The Non-Nuclear System Test was terminated, after 32 hours of operating time on the primary coolant system, by a short circuit between an electrical lug in the heating system and the containment vessel. A power level of 3 Mw and temperature of 1460F had been reached. No damage was done to the lithium system and repairs are in progress.

Improved fuel element tubing containing carbon has been placed on order for possible use in the LCRE core.

The first reactor core pressure vessel, upper head assembly, has been completed except for final machining. The development program for flow-turning the type 316 stainless steel reflector case was successfully completed. To date, attempts to flow-turn the inner case have been unsuccessful. Fabrication of the first lithium pump double sump has proceeded slowly because of welding difficulties.

Major construction for the large, inert gas welding chamber and clean room in the CANEL machine shop was completed. The clean areas are operational.

Considerable planning has been done on the fabrication and installation of the Auxiliary Systems in Idaho. Schedules were prepared and a preliminary specification was written for a single vendor contract, for the construction and installation of these systems.

Title II design of the Idaho test facility is approximately 70 percent complete. The preliminary scheduling of facility construction by the construction contractor shows a slippage of several months beyond what is required to meet the operating date.

B. SNAP-50

Preliminary design criteria have been prepared for the experimental powerplant proposed for the first SNAP-50 flight test. A nominal 300 net Kwe power output has been selected as compatible with both the SPUR requirement and a fixed radiator for main heat rejection. Preliminary requirements have been prepared for both the non-nuclear powerplant test facility at CANEL and the nuclear powerplant test facility in Idaho.

Analysis and preliminary design of a uranium carbide fueled reactor continued. The first configuration of a SNAP-50 type critical experiment was brought to criticality.

Three inpile capsules containing uranium carbide fuel and one containing uranium nitride were examined following irradiation. The results for hyperstoichiometric uranium carbide and for uranium nitride appear promising.

A canned electric motor, driving a direct coupled centrifugal pump, has been adopted in concept for the powerplant primary system. The motor is being studied on subcontract by two companies.

The first test of a lithium hydrodynamic radial bearing at 600F completed 225 hours at the end of the quarter. Two phase operation of a stainless steel, boiling potassium loop was started. The test operated approximately 100 hours at 1580F potassium vapor temperature during the quarter. Preliminary designs of prototype boilers are in process.

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C. ADVANCED APPLICATIONS STUDIES

The major effort of the Advanced Applications group during the quarter was devoted to completing and reporting the SNAP-50 studies which have been in progress since the beginning of the program. The results of these studies are presented in the following reports:

PWAC-387	SNAP-50 Arrangement Studies
PWAC-391	SNAP-50 System Optimization Studies
CNLM-4448	SNAP-50 Preliminary Data for Application Studies

II. LITHIUM-COOLED REACTOR EXPERIMENT

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A. ANALYSIS AND DESIGN

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1. Reactor

A graphical method of predicting fuel pin temperatures at any location in the core was developed from a correlation of all IBM-7090 computer cases calculated for the LCRE fuel pins. For each mass flow rate in the core, depending on location within the core, the method requires an R-Z computer analysis of a fuel pin with an average heat generation rate. By this method the results can then be extrapolated to any fuel pin with the same mass flow rate. All variations in heat generation and thermal conductivities of the fuel pin components are compensated for by means of correction factors. The method is generally accurate to within 2.0 percent except at points of extreme flux peaking where accuracy is within 5.0 percent of computer predicted results. With four basic mass flow rates prevalent in the LCRE core, this method can predict temperatures of all the fuel pins from only four R-Z computer cases.

Minor design changes were made to the pressure vessel to accommodate warpage created by the final pressure vessel weld. Stress analysis calculations for the revised flow-turned reflector case were completed, based on most recent estimates of liquid metal pressure and differential temperature in the reflector. The structural integrity of the reflector case under transient conditions was also investigated. Design of a welded reflector case was completed as an alternate to the flow-turned case utilized in the present design.

Work continued on the design and analysis of control drive cooling. This is being based on revised estimates of radiation heating resulting from analysis of the control drive gamma radiation environment. A mock-up of proposed control drive cooling details, including control drive wiring and gas turbine details, is being prepared to verify assembly procedures.

A heat transfer analysis of the pile oscillator bearings was completed, indicating that insulation on the reflector case and utilization of some of the control drive nitrogen coolant are required to maintain bearing temperatures within tolerable limits.

In a continuing attempt to correlate the analytical and experimental predictions of LCRE critical loading and drum worth, extensive multi-group calculations paralleling CCA-6 critical experiment determinations of critical masses and reactivity coefficients and drum worth evaluation procedures have been performed. It has been found that CANEL - one-dimensional and AIM-6 calculations of coefficients give poor agreement with experimental results while CANEL - two-dimensional calculations yield much better agreement.

The CANEL - two-dimensional calculation of critical masses are, in general, in fairly good agreement with experiment but are still sufficiently different to lead to an uncertainty of about two percent in reactivity for the calculated LCRE drum worth. The drum worth appears sufficient to cover remaining uncertainties in drum requirements and in setting the nominal fuel loading. A further refinement of the predictions will be made to substantiate the operating life and to minimize requirements for fuel shimming operations at initial loading. A series of TDC calculations have been initiated to parallel the CANEL - two dimensional calculations.

Further investigations of the problem of helium production in the fuel pin from the (n, 2n) reaction in beryllium have indicated that the cross sections used in past calculations should be revised. New cross section information found in the literature and an analysis of helium production in MTR-irradiated LCRE fuel samples both tend to confirm a higher production rate by a factor approaching two in some localities. Analysis of the effects on fuel pin cladding has been started.

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The reactivity worth per fuel element, as a function of radial position, was recalculated to include results of reactivity coefficient measurements in the CCA-6 critical assemblies. From this, the reactivity effect of small radial displacements of fuel elements has been updated for use in a continuing examination of bowing coefficients and for use in the fuel loading experiments.

Additional computer codes have been made available for use at CANEL. A DSN code will calculate gamma ray transport in one-dimensional slab, spherical, or indefinite cylinder geometries. Neutron fluxes on binary punched cards, obtained from a DSN neutron flux calculation, are used, together with libraries of gamma ray production cross sections, as input to the FORTRAN BRIDGE code. This is programmed to calculate the gamma energy source distributions for thirteen gamma energy groups and six, sixteen or eighteen neutron energy groups. The BRIDGE code, in turn, provides binary punched card output suitable for use as gamma energy source input for the DSN gamma transport calculation. A thirteen group gamma ray cross section library was compiled with first order anisotropic scattering cross sections.

Past experience in the use of buckling iteration procedure, programmed in the BIP code which is based on the AIM-6 code,has been unsuccessful. AIM-6 is a one-dimensional diffusion theory code for calculation of cylindrical reflector-moderated reactors. An investigation to discover why such calculations would not converge disclosed that, for the low energy neutron groups in which transport was predominantly inward from the reflector to the core,the leakage was not properly accounted for by a term proportional to the group flux. For example, the axial distribution of well moderated neutrons diffusing from the side reflector into the core corresponds to the distribution of these low energy neutrons in the side reflector and is influenced little by leakage between the core and end reflector.

When nonhomogeneous leakage terms were substituted for the terms proportional to the fluxes in those energy groups for which the core leakage was calculated as negative, a modified BIP code gave rapid convergence to an eigenvalue which was in fair agreement with a TDC (Los Alamos Two-Dimensional Code) result obtained with the same cross sections.

2. Reactor Shield

A two-dimensional mapping of (n, α) and elastic slowing-down heating in the LCRE shield was prepared from TDC fluxes. A corollary investigation of TDC flux convergence for deep penetration indicated variations approximating a factor of two, which is, therefore, reflected in the accuracy of the heating maps. Calculations for the graphite region of the shield, using the one-dimensional, DSN code and 16 and 18 group Los Alamos cross sections, have indicated large discrepancies in calculated thermal fluxes when using the two cross section sets. In order to resolve this problem, thermal neutron age calculations are being carried out in various materials for which experimental determinations have been made.

Calculations were completed to determine the neutron and gamma dose and heating rates seen by the nuclear sensors which are located in the LCRE shield, in order to

establish their cooling requirements as well as criteria for the interim inpile testing of the sensors. Results showed flux levels to be higher than desired and that design changes were required in the sensor ducts. These changes have been started. A revised layout of the startup fission counter and intermediate range ion chamber is now in the process of detail design.

Work continued on the design of seals to control the loss of coolant gas at shield penetrations. Gamma and neutron heating in the control drive wiring, located in the region between the reactor and the shield, was calculated to be 0.3 watts/cc. The resulting temperature in this wiring is sufficiently high to warrant the use of silver or nickel conductors with stainless steel sheathing and magnesium oxide insulation. These requirements are being incorporated in design drawings.

Preliminary shield construction assembly drawings were made available to potential vendors. Final drawings are being held to incorporate control drive wire routing, nuclear sensor installation changes and to provide the required sealing of gas coolant passages and shield penetrations.

Design Layouts in Preparation

a. Revision of reactor and inner shield support structure to incorporate control drive wiring.

b. Revision of shield penetrations to incorporate control drive wiring.

c. Design of shield coolant passage seals.

d. Design of pile oscillator wiring.

e. Design of fuel loading experiment source cans.

Drawings in Detail Design and Checking

a. Revisions of startup and intermediate range nuclear sensors.

b. Design of power range nuclear sensor check source drive.

c. Design of control drive gas system and associated shield penetrations.

3. Primary and Reflector Coolant Systems

Studies of the LCRE approach-to-power and design point operating temperature levels were finalized and published. Included in the study were the effects of variations between predicted and actual heat exchanger performance, the effects of reactor lithium exit temperature variation and the effects of liquid metal flow rate variations. The analysis also defines the predicted permissible system operating envelope of the LCRE for the complete operating spectrum based on the following specified thermal limitations of the system:

a. Maximum lithium temperature at primary pumps - 1000F.

b. Maximum NaK temperature at radiator inlet - 1400F.

c. Minimum lithium temperature in primary system - 700F

d. Maximum heat sink air blower capacity - 45 lb/sec/blower.

e. Maximum reactor lithium outlet temperature - 2000F.

The permissible range of variation of reactor lithium temperatures is shown in Fig 1 as a function of reactor power level with the above limitations shown parametrically, where applicable, to define the predicted operating envelope.

The maximum primary pump sump level, as dictated by heat exchanger performance, was determined. A redesign of the primary sump level probes to incorporate larger diameter probes with thicker cladding to improve oxidation resistance was completed. Detail design of the primary fill and drain system, including the freeze plug requirements, was completed.

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Work was started to give better definition of predicted radiation levels in the test cell from activated coolant in the primary and reflector piping systems and to leakage from the shield. A simplified piping diagram was laid out and information was prepared for a code for calculating dose from a straight pipe section. Calculations of fission product decay gamma energy release from the lithium pipes in case of fuel pin rupture were begun.

Required cooling for reflector pipe penetrations through the shield was determined and the orifice size to furnish this coolant flow was determined.

Design layouts of vapor traps, fill and drain lines and gas lines for Li-6 and Li-7 tanks were completed. The design layout of the electrical heating for the primary pump sump was also completed.

The primary system instrumentation layout was completed and detail designs were started. Details for incorporation of welded fittings on lines to the sump were completed. The jacket bellows and piping redesigns to provide more flexibility were also completed.

The primary pumping unit, primary fill and drain tank and auxiliary fill tank electrical drawings were completed. A redesign to eliminate heaters in the area of the flowmeter pole faces was initiated. A study layout to determine the feasibility of adding lithium detectors to the primary jacket was also started.

Data requirements for the final water flow tests of the LCRE coolant addition system mock-up were prepared. Engineering specifications for a quick-connect device for the Li-6 punch valve operating gas system were prepared and tank pressure requirements were determined.

The neutron flux environment at the control drum drive motors and at the outlet thermocouples was determined by TDC calculations. The flux at the latter location appears too high because of possible nuclear transmutations in the thermocouple materials. Flux maps were prepared at other locations along the outlet pipe to assist in determining other possible thermocouple locations.

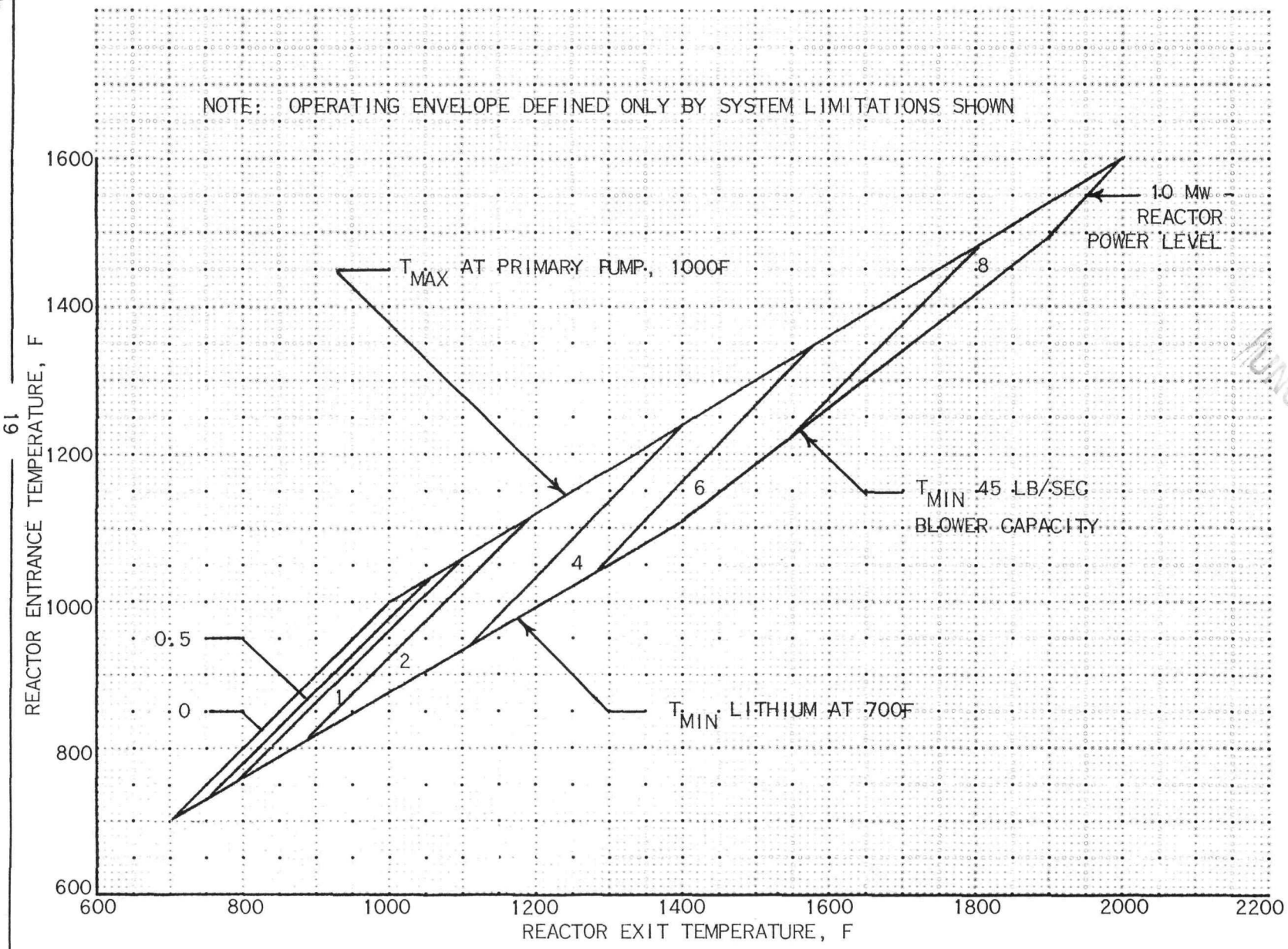
The design work currently being performed on the Primary and Reflector Coolant System is summarized below.

Design Layouts in Preparation

a. Design of lithium leak detectors for addition to the primary coolant system jacket.

b. Revision of heater installation and wiring on the primary pump gas lines and portions of primary piping jacket.

PRIMARY COOLANT SYSTEM OPERATING ENVELOPE



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c. Design of wiring for the reflector coolant loop, including heater installation revisions.

d. Revision to the primary coolant system pipe hanger installation.

e. Revisions to the evacuation gas system valves to incorporate redesigned valves.

Drawings in Detail Design and Checking

a. Revisions to the primary pumping unit gas evacuation lines.

b. Revisions to operators of fill and drain and coolant addition system valves.

c. Revisions to liquid metal level probes.

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4. Secondary Coolant System

An analysis of the test facility exit duct for the heat sink air system was completed and a report defining LCRE requirements was completed for publication. Design layouts of the heat sink air system were completed, including the radiators, blowers, control valves and instrumentation. Design layout of several possible oxygen hot trap arrangements was completed. A configuration utilizing spiral columbium sheet was selected as the trapping element.

The results of a piping deflection analysis, conducted to determine motion due to thermal expansion, has indicated greater vertical deflections than did preliminary estimates. These deflections are within the design capability of the test cell wall pass-throughs and will be accommodated by selection of pipe hangers to provide adequate travel. The heat transfer analysis of the proposed design for the secondary piping pass-through was completed. Investigation is continuing to assure that the test facility concrete temperature limitations are realistic and will be met. The piping of the entire secondary system has been re-examined to ensure that adequate flexibility is available and that acceptable stress levels are not exceeded.

Oxide control station, plugging indicator unit, and pump lube oil and sweep gas system design details were completed.

The design work now in progress is summarized below.

Design Layout in Preparation

a. Design of wiring for the oxide control station cold trap and plugging indicator.

b. Revisions to the secondary coolant system pipe hanger installation.

c. Design of the hot trap installation in the secondary loop.

d. Design of the secondary piping system test cell wall pass-through.

e. Design of shielding for the secondary piping system test cell wall pass-through.

f. Revisions to the secondary pump assembly.

g. Revisions to the secondary electrical heating system wiring.

Drawings in Detail Design and Checking

a. Completion of secondary system assembly drawings.

b. Design of heat sink air system ducting, and blower and valve installation.

c. Revisions to operators of fill and drain and isolation valves.

5. Auxiliary Systems

The detailed designs of control drum shim and scram actuating motor facility circuits were completed. The first circuit requires minor changes to the servo amplifier chassis and the schematic has been revised accordingly. A compensating amplifier has been designed for use in the temperature servo mode of reactor operation. The determination of firm values for components for the compensating network circuit are delayed pending further closed loop tests of the control system mock-up analog computer combination.

The S and P Control area floor plans (Fig 2) were finalized and are now in agreement with the latest flow diagram information. The Safety System flow diagram was brought up to date with all preceding system flow diagrams and issued for final review and approval.

The heat removal assumptions and conditions used in arriving at the test cell post-accident conditions were published and presented to the architect-engineer for use in the test cell and shield coolant systems analysis. Design specifications for the shield coolant return ducting have been established. The routing of the return duct will be such that it will relieve the maximum heat load requirements for the test cell cooling system by mixing its flow with the test cell coolant flow.

Methods for isolating the gas, oil and liquid metal supply lines penetrating the test cell in the event of a major accident were investigated. The possibility of using a minimum number of isolation valves on the helium lines was studied from the standpoint of minimizing activity levels outside of the test cell.

The nitrogen cooling requirements for the liquid metal system vapor traps were determined from the operation of prototype traps on development liquid metal pump tests and the supply and control system has been defined.

Two groups of auxiliary systems required for the LCRE are shown in Fig 3. The CANEL systems are shown in Instrument Flow Diagrams. These are 95 percent complete. The test facility systems are shown in the Title II drawings. All designs will be reviewed during the next quarter, considering LCRE operations as a whole. The control, instrumentation and mechanical portions of these systems are located around the Test Cell and Heat Sink Rooms (Fig 2). The LCRE Heat Sink System is shown in Fig 4.

The designs of liquid metal pump lubricating oil units, helium purification system and liquid metal handling transfer tanks are approximately 80 percent completed. The designs of temporary structures in the test cell and fuel element and reactor core handling equipment for the Fuel Loading Experiment were started. The detail design of test cell pass-throughs is 90 percent complete. A typical pass-through for gas service is shown in Fig 5. All pass-throughs will be leak and pressure tested during the course of installation work.

The production of schematic drawings for all CANEL designed auxiliary systems is now approximately 60 percent completed and face layouts for all S area panels are eighty percent complete.

Consistent with the general plan for fabrication, assembly and installation of auxiliary systems, it is intended that the majority of installation drawings, including wiring diagrams, gas line routing, control plumbing, location of equipment, and interconnection of liquid metal, auxiliary and control systems, will be made by an Auxiliary and Control System subcontractor. Layout drawings showing the desired location of equip-

LOCATION OF AUXILIARY AND CONTROL SYSTEMS

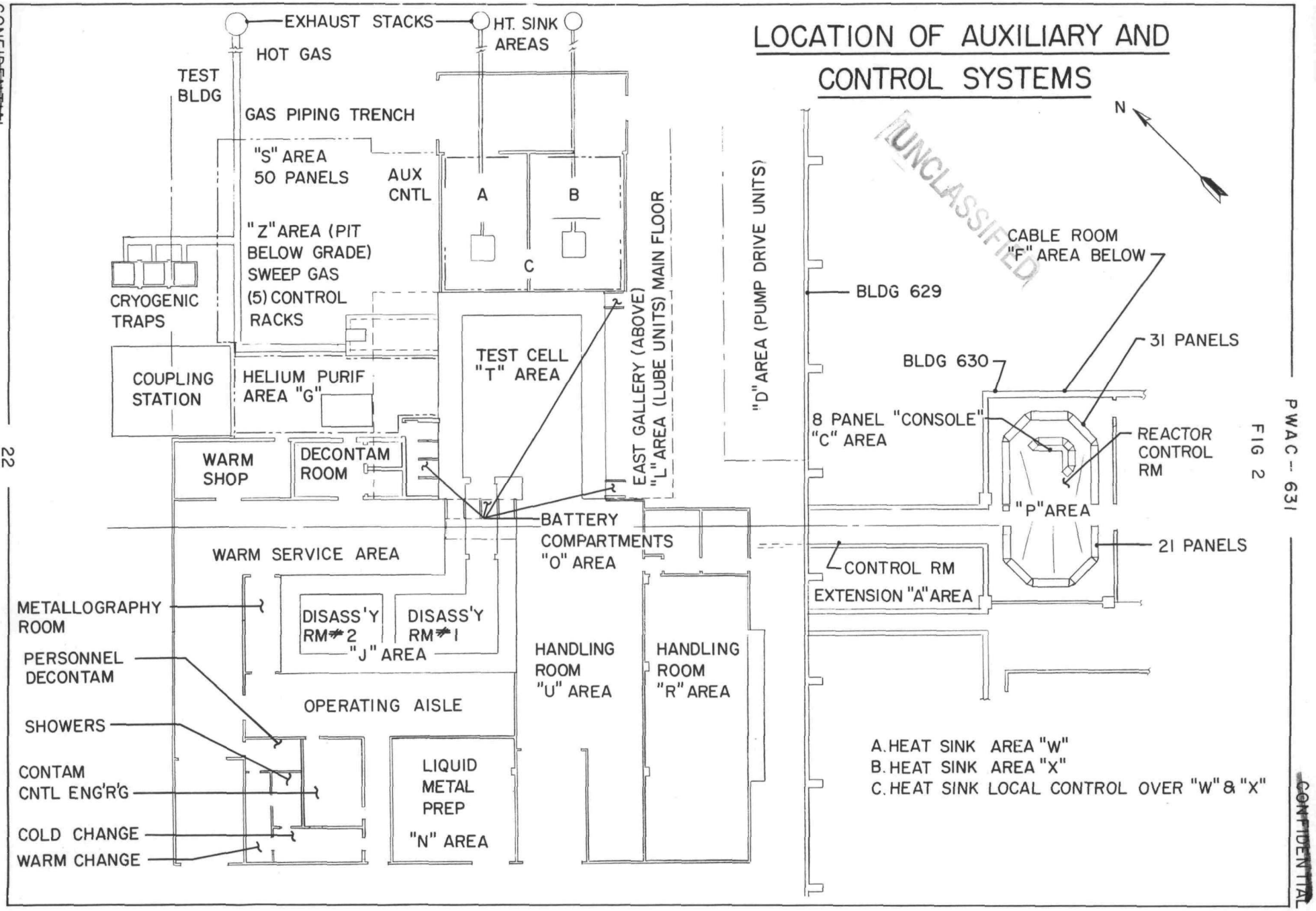

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FIG 2

FIG 3

LCRE SYSTEMS

Auxiliary Systems-Designed by CANEL

1. Primary Circuit Instrumentation
2. Reflector Circuit Instrumentation
3. Secondary Circuit Instrumentation
4. Primary Fill and Drain
5. Primary Coolant Addition
6. Reflector Fill and Drain
7. Secondary Fill and Drain
8. Primary Seal and Sweep Gas
9. Reflector Seal and Sweep Gas
10. Secondary Seal and Sweep Gas
11. Jacket Gas
12. Helium Supply and Purification (includes bulk supply)
13. Pretest Gas Analysis (portable gas analysis and evacuation pump dollies)
14. Secondary Oxide Control
15. Primary Pump Drive (includes test station for pump unit green tests)
16. Reflector Pump Drive
17. Secondary Pump Drive
18. Primary Electric Heating
19. Secondary Electric Heating
20. Primary Pump Lubricating Oil
21. Reflector Pump Lubricating Oil
22. Secondary Pump Lubricating Oil
23. Heat Sink
24. Gear Box Gas
25. Pile Oscillator
26. Gas Line Heating (1)
27. Chamber Check Source
28. Chamber Drive Source
29. Startup Source Drive
30. Cryogenic Trap - Nuclear Instrumentation
31. Cryogenic Trap - Non-Nuclear Process (2)
32. Liquid Metal Handling
33. Shield Cooling - CANEL Instrumentation

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(1) To be completed next quarter after design of gas line routing is completed
(2) To be completed next quarter in collaboration with Cryogenic Trap System subcontractor

Control and Safety Systems-Designed by CANEL

34. Reactor Control
35. Reactor Safety
36. Reactor Flux Monitoring
37. Reactor Gamma Monitoring
38. Battery Power Supply

Fuel Loading Experiment (FLE) Systems-Designed by CANEL

39. Fuel Loading Experiment - Control and Safety
40. Fuel Loading Experiment - Nuclear Instrumentation
41. Core Installation

Test Facility Systems Which Affect Design and Operation of the LCRE-Designed by Burns and Roe

42. Shield Cooling
43. Test Cell Cooling
44. Pressure Relief
45. Test Cell Gas Purification
46. HVAC 103 Heat Sink Rooms
47. HVAC 105 Warm Areas
48. Domestic Water and Fire Protection
49. Cooling Tower Water
50. Shop Air
51. Instrument Air
52. Nitrogen Supply
53. Electric Power

FIG 4

LCRE HEAT SINK AIR SYSTEM

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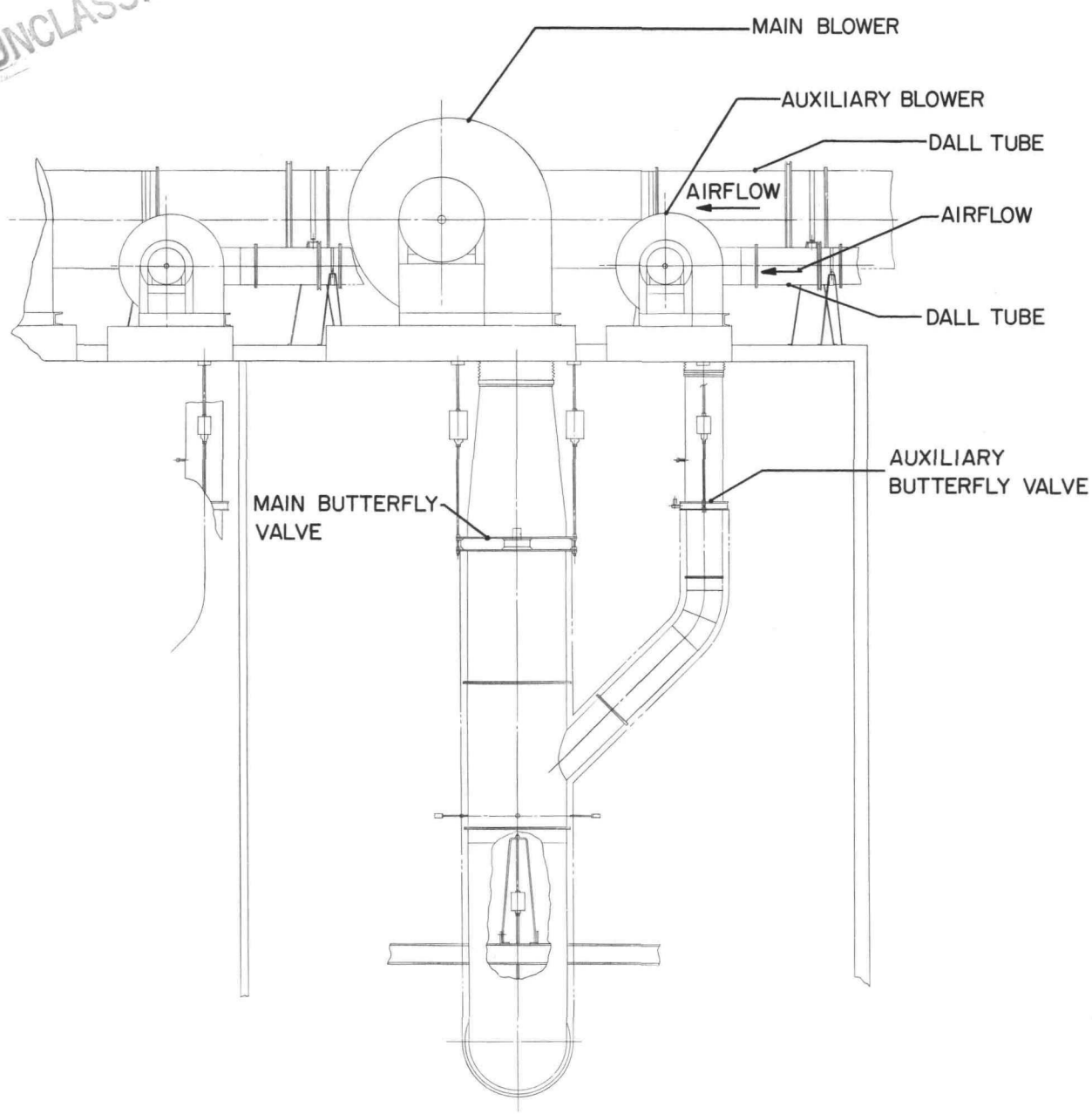
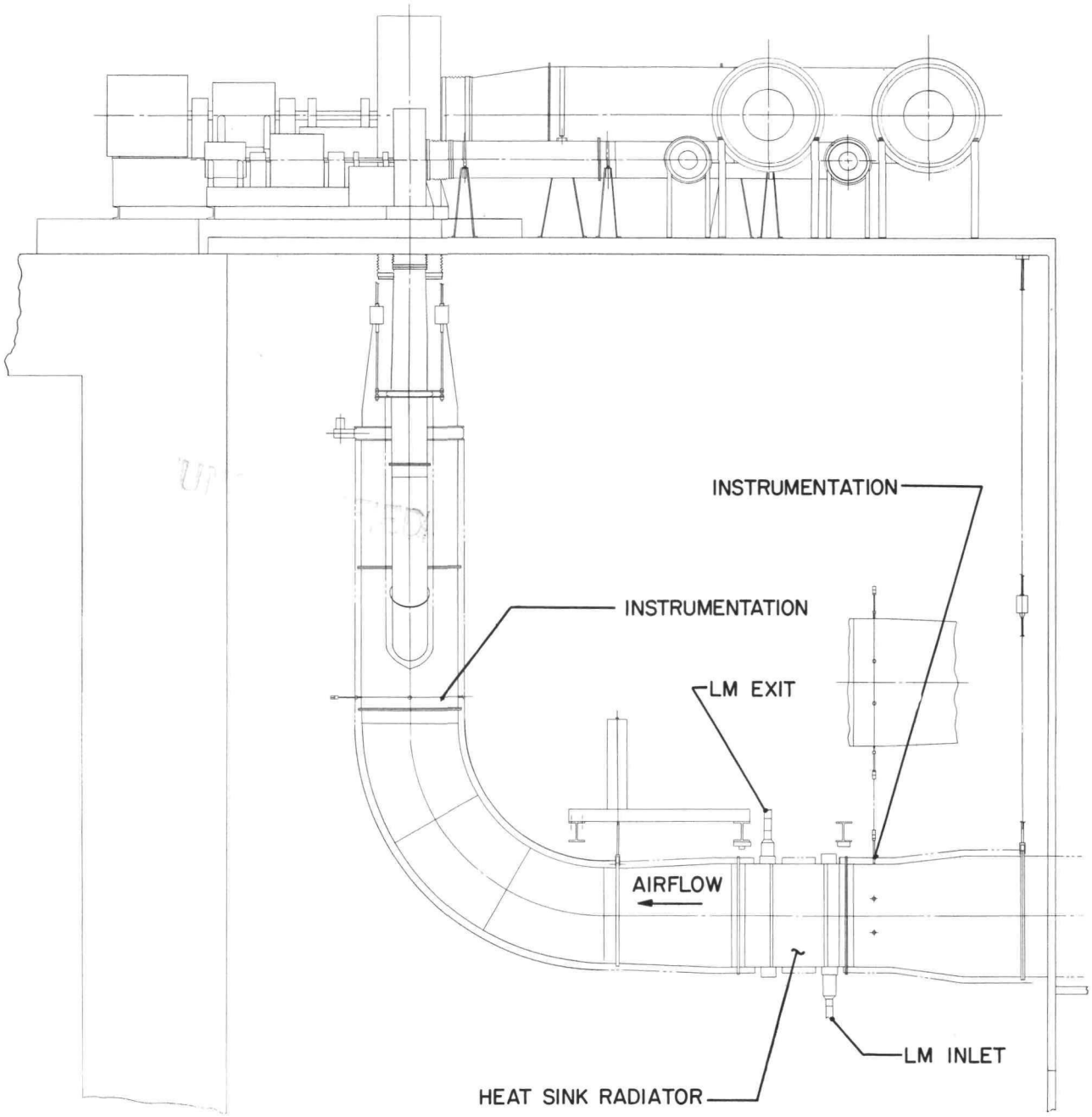


FIG 4

LCRE HEAT SINK AIR SYSTEM

(CONTINUED)



TYPICAL TEST CELL PASS THRU FOR LCRE GAS LINES

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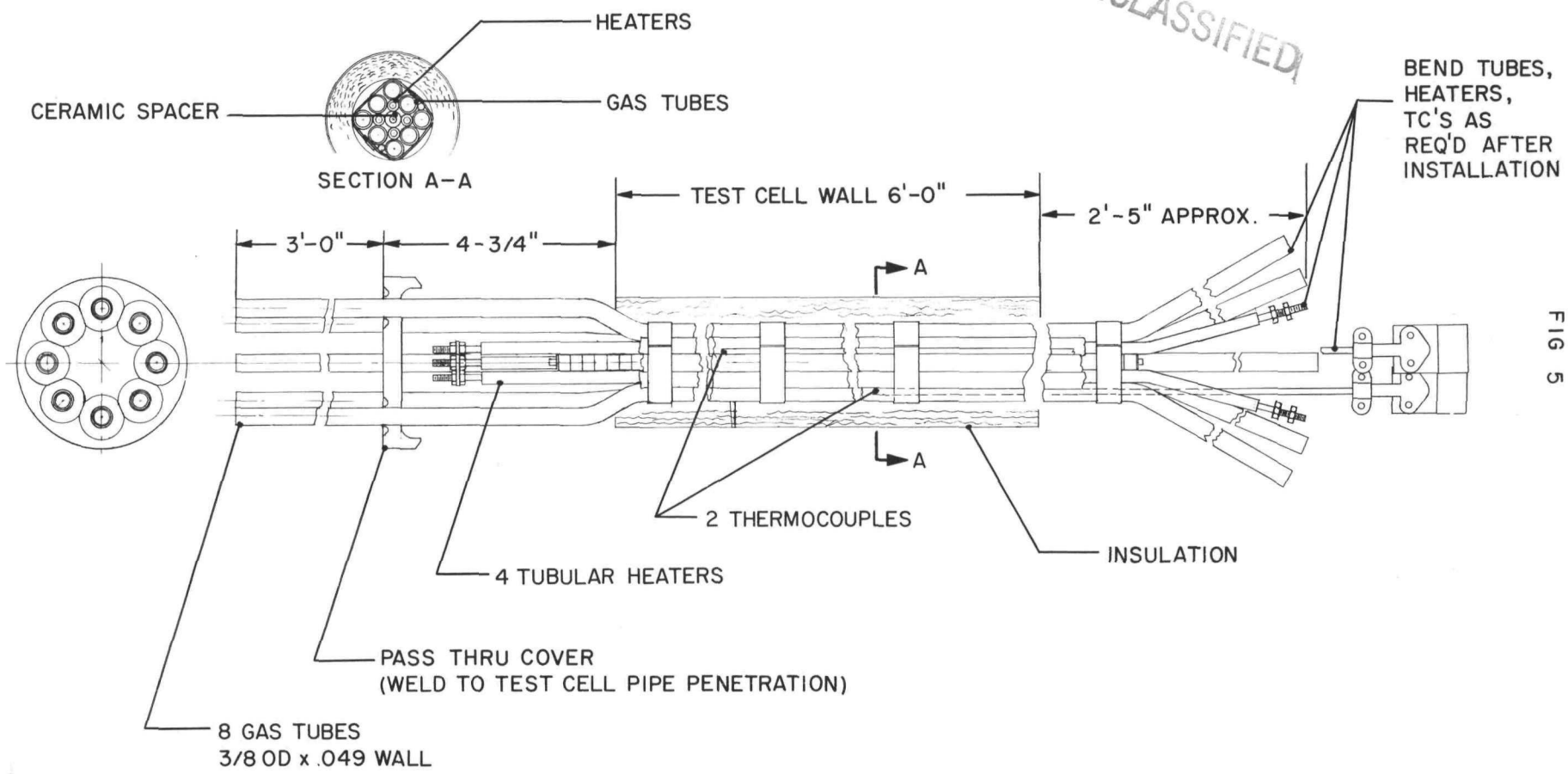

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FIG 5

ment and panels in the L, D, G, P, S, and C areas were completed. These drawings, together with installation drawings for the reactor and liquid metal systems to be provided by CANEL, are required to produce the Auxiliary and Control System installation drawings. Plans for making these installation drawings will be completed during the next quarter in conjunction with planning and negotiations to subcontract fabrication, assembly and installation of the Auxiliary and Control Systems.

6. Hazards

Preparations were completed for the shipment of LCRE fuel pins to ORNL for criticality studies. The results of these determinations will be used to assist in hazards evaluation of the LCRE fuel handling and fabrication procedures. ORNL will determine critical masses of fuel pins in light water moderated and water reflected lattices. These determinations will include the zero spacing and most reactive spacing. In addition, measurements will be made of changes in reactivity obtained by replacing fuel pin BeO end reflectors by light water or Zyglo, and replacing fuel material with BeO.

Hazards resulting from core flooding during LCRE welding operations, using water and chlorinated hydrocarbons as welding chamber coolants, were investigated. Complete core flooding and reflection of the B_4C shrouded core resulted in a negative reactivity of six percent with CCl_4 and a postive reactivity (supercritical) with water.

Additional hazards studies were completed in which various core meltdown models were considered. These models assumed an accident initiated by loss of primary coolant flow, the meltdown occurring in the presence of stagnant lithium. The consequences of fuel pin rupture and fuel-lithium reactions were also considered. For the models which led to an excursion, the energy release was calculated to be less than the 1.6×10^6 Btu reported in PWAC-370 for the loss of coolant accident.

A preliminary format for the LCRE Final Hazards Summary Report has been completed. An outline of the Operations Manual was prepared as a guide for the preparation of operation abstracts. Approximately 87 percent of the abstracts have been completed.

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B. LCRE REACTOR SPECIFICATIONS

General Reactor Data	
Reactor power, Mw	10
Design lifetime, hr	10,000
Critical mass, kg-U^{235}	86.1
Core coolant	Lithium (99.99% Li^7)
Reactor coolant inlet temperature, F	1600
Reactor coolant outlet temperature, F	2000
Type of fuel	UO_2-BeO ceramic
Nominal volume percent UO_2 in matrix	47
Reactor structural material, fuel element cladding and core filler pieces	Cb-1 Zr alloy
Side reflector material	Beryllium
End reflector material	Beryllium oxide
Reflector structural material	Type 316 stainless steel
Reflector coolant	NaK (78 w% K)
Method of reactor control	Rotating poison drums
Poison material	B_4C (natural B)
Weight of reactor package, lb	3300
Reactor Configuration	
Equivalent active core diameter, in.	13.5
Active fueled core length, in.	13.5
End reflector thickness, in.	6.9
Side reflector thickness (nominal), in.	6.9
Core filler OD (nominal), in.	14.476
Pressure vessel ID (nominal), in.	14.716
Pressure vessel OD (nominal), in.	15.136
Reactor jacket ID (nominal), in.	15.381
Reactor jacket OD (nominal), in.	15.557
Reflector outer shell diameter, in.	30.12
Core filler material	Cb-1 Zr alloy
Number of core filler pieces	6
Core Operating Conditions	
Average power density, kw/cc	0.469
Average heat flux, Btu/hr-ft^2	2.30×10^5
Average ΔT film, F	8
Average ΔT clad, F	10
Center pin maximum fuel matrix temperature, drums 60°, F	2745
Center pin total temperature rise, bulk coolant to fuel centerline, at maximum temperature plane, F	610
Maximum fuel matrix temperature-corner pin, edge can, F	2250 (0.75 power)
Coolant inlet pressure, psia*	55
Coolant outlet pressure, psia*	45
Coolant flow direction	Upward
Power from core, Mw	10
Coolant flow rate in fueled core, lb/sec	22.8
Mixed mean core outlet temperature, F	2006
Core bypass coolant flow rate, lb/sec	1.14

*This item changed from previous report

Mixed mean temperature of bypass coolant outlet, F	1920
Average coolant velocity in core, fps	8.25
Average Reynolds number in core	3710
Average film coefficient in core, Btu/hr ft^2 F	32,000

Maximum to average power ratio for thermal design

	Center Pin	Edge Peaking
Axial	1.11	1.78 (R=0 to 6.65 in.) 0.95 (R=6.65 to 6.75 in.)
Radial	1.42 at 60°	2.8 at 160°

Fuel Element Specifications

Type of fuel	UO_2-BeO ceramic
Type of fuel element	Cylindrical pins in hexagonal array
Fuel pin OD, in.	0.305
Fuel pin cladding thickness, in.	0.015
Helium thermal bond thickness, in.	0.001
Fuel matrix diameter, in.	0.273
Active fuel matrix length, in.	13.5
Length of top and bottom end reflectors, in.	6.9
Fuel pin fission gas void length, in.	8.0
Number of fuel pins in hexagonal can, full	91
Number of fuel pins in hexagonal can, partial	78
Equivalent number of hexagonal cans in core	18.14
Distance across flats, nominal fuel pin bundle, full can, in.	2.946
Fuel can wall thickness, in.	0.030
Distance across flats of hexagonal can (maximum), in.	3.021
Diameter of can perimeter flow blockage rods, in.	0.070
Fuel pin cladding material	Cb-1 Zr alloy
Fuel can material	Cb-1 Zr alloy

Material Volume Fractions	Core	Side Reflector	End Reflector
UO_2	0.3167		
BeO	0.3571		0.6738
Li	0.1011		0.1011
Cb	0.2153		0.2153
He	0.0098	0.0012	0.0098
B_4C		0.1293	
Be		0.7435	
NaK (78 w% K)		0.0753	
Type 316 stainless steel		0.0507	

Side Reflector Configuration

Control drum

Number of control drums	10
Control drum arrangement	Close packed
Control drum diameter, in.	6.478
Diameter of control drum circle, in.	22.632
Poison material	B_4C (natural B)
Poison arrangement	Canned assembly of solid segments
Poison segment angle, degrees	120
Poison length above core midplane, in.	12.2
Poison length below core midplane, in.	13.7

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Poison segment gas containment volume, in.^{3}	83.7
Beryllium length, in.	27.5
Bearing materials	
Journal (inner) and bearing (outer)*	90 w/o WC-8 Mo-2 CbC
Reflector material	Beryllium
Stationary Reflector and Vessel	
Number of stationary reflector pieces	
Inner	10
Outer	10
Length of stationary reflector pieces, in.	27.5
Inner reflector vessel ID, in.	15.381
Inner reflector vessel OD, in.	15.557
Inner stationary reflector piece ID, in.	15.755
Outer stationary reflector piece OD, in.	29.750
Reflector jacket ID, in.	29.870
Reflector jacket OD, in.	30.120
Control drum well ID, in.	6.746
Control drum well OD, in.	6.836
Clearance between control drum well and fixed reflector pieces, minimum, in.	0.002
Reflector material	Beryllium
Reflector structural material	Type 316 stainless steel
Reflector coolant	NaK (78 w% K)
Side Reflector Operating Conditions	
Coolant flow direction	Upward
Coolant inlet temperature, F	710
Coolant temperature rise, mixed mean, F	25
Coolant flow rate, total lb/sec	45.2
Coolant inlet pressure, psia*	58
Coolant pressure drop, inlet feeder to exit collector	3.0
Stationary Region	
Heating per reflector piece	
Inner, kw	2.9
Outer, kw	1.12
Heat leakage from reactor, kw	80
Coolant temperature, rise	
Inner passage, F	50
Outer passage, F	4
Maximum temperature rise in beryllium, F	6
Maximum temperature in beryllium, F	760
Coolant flow rate external to drum well, lb/sec	23.2
Average coolant velocity, ft/sec	2.8
Pressure drop through stationary piece length, psi	0.3
Control Drum	
Heating per drum	
Beryllium region, kw	8.5
Poison region, kw	4.2
Coolant temperature rise drum annulus, F	30

*This item changed from previous report

Maximum temperature rise in beryllium, F	73
Maximum beryllium temperature, F	800
Maximum temperature rise in B_4C, F	52
Maximum B_4C temperature, F	1000
Coolant flow rate in drum annulus, lb/sec	2.2
Average coolant velocity, ft/sec	2.44
Pressure drop through drum length, psi	0.3
Helium pressure in poison cans	
Initial fill, psia	0.1
After 10,000 Mw Hr (10% release), psia	74
Nuclear Specifications	
Fission burnup, percent mass (100,000 MwH)	5.05
Total burnup, percent	6.57
B^{10} burnup, percent	1.1
Total control drum worth, Δk	0.09
Operational reactivity losses, Δk	
Fuel burnup	0.0300
Fuel growth (estimated)	0.0053
Xe^{135} equilibrium	0.0005
Xe^{135} buildup, after shutdown	0.0000
Sm^{149}	0.0044
Permanent poison buildup	0.0089
Li^6 buildup	0.0031
Total	0.0522
Temperature - reactivity losses, Δk	
70F to 1000F, wet reflector	0.0067
1000F to 1800F, (1) wet core and reflector	0.0033
NaK worth in reflector, 70F, Δk	0.0030
Lithium worth in core, 1000F, Δk	0.0075
Unusable portion of shim control worth, Δk	0.0100
Maximum excess reactivity required, 1000F, Δk	0.06
Minimum shutdown margin, 1000F, Δk	0.03
Helium generation, atoms (100,000 MwH)	
B_4C in all control drums*	5×10^{24}
BeO in all fuel pins*	8×10^{24}
Be in side reflector	7×10^{24}
Spectrum of neutrons causing fission	
Percent thermal	
Drums in	0.1
Drums out	2.0
Median energy, kev	
Drums in	170
Drums out	100
Flux leakage into shield, percent of total neutrons	
Side leakage	10
End leakage	2
Total	12

(1) Reflector attains maximum temperature of 750F approximately

* This item changed from previous report

Maximum total flux, core center, n/cm^2-sec 10^{15}

Power distribution ratios

Axial (averaged radially)	Calculated	Experimental
Center to average	1.18	1.08
Minimum to average	0.71	0.90
Maximum to average	1.18	1.68

Radial (average axially)	Drums In	Drums Out		
	Diffusion Theory	Diffusion Theory	Transport Theory	Experimental
Center to average	1.45	1.20	1.23	1.12
Minimum to average	0.55	0.85	0.84	0.875
Edge to average	0.55	1.65	1.95	3.0

Prompt neutron generation time, seconds

Drums in	1.5×10^{-7}
Drums out	1.8×10^{-6}

Effective delayed neutron fraction, $\epsilon\beta$ 0.007

Temperature coefficients, Δk per degree F x 10^6

Coolant expansion	
Core lithium	-0.8
Reflector NaK	-0.8
Axial expansion	
Fuel	-2.6
Reflector	-1.0
Core radial expansion	
Top	-0.3
Center	-0.9
Bottom	-0.3
Reflector radial expansion	
Top	-1.1
Bottom	-1.1
Bulk	-3.3
Doppler (U^{235}), 2000F	+0.85

Bowing coefficient, Δk per degree F (of core ΔT)x 10^6 +1.2

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C. RESEARCH AND DEVELOPMENT

1. Fuel

A lithium soak test of 906 pins with different combinations of welding and annealing was completed and the pins have been decontaminated. The fabrication variables were as follows:

Cb-1 Zr alloy End Caps - Not Annealed	151 pins
Cb-1 Zr alloy End Caps - Annealed at 2200F	151 pins
Cb-1 Zr alloy End Caps - Annealed at 2400F	151 pins
Cb-5 Zr alloy End Caps - Not Annealed	151 pins
Cb-5 Zr alloy End Caps - Annealed at 2200F	151 pins
Cb-5 Zr alloy End Caps - Annealed at 2400F	151 pins

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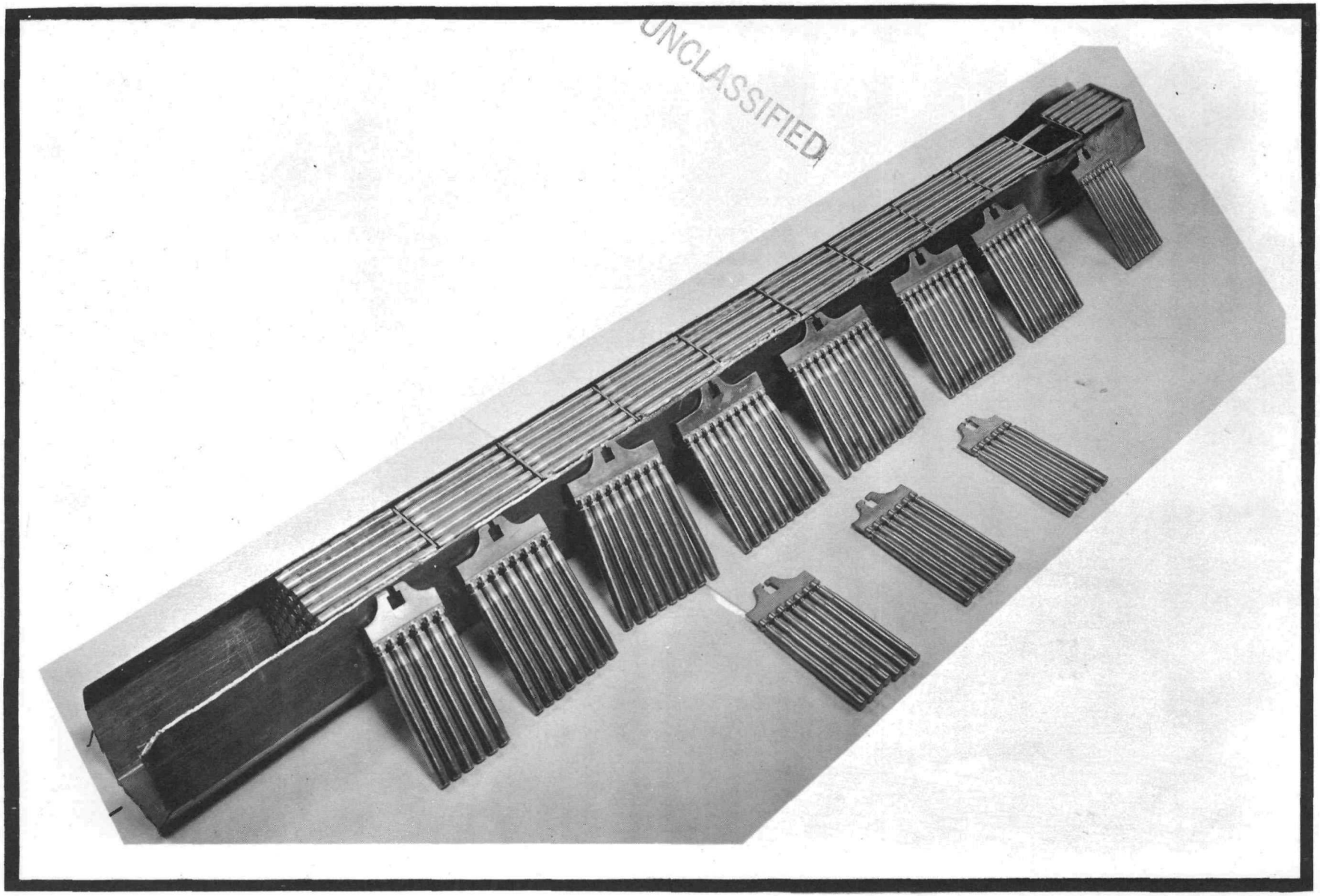
All pins used Cb-1 Zr alloy cladding. Random quantities of each type pin were welded at the same time. The soak test was operated at 1600F for 100 hours, then 2000F for 100 hours and then cycled thermally 15 times within the 2000F to 900F range. Preliminary examination by X-ray methods has shown 12 leaks out of 302 welds of Cb-1 Zr-to-Cb-1 Zr alloy unannealed pins and two leaks out of 320 welds of Cb-1 Zr-to-Cb-5 Zr alloy unannealed pins. No annealed samples have been found which leaked lithium. A more detailed examination, involving pin disassembly, is continuing. Fig 6 shows the test element partially disassembled.

Fabrication of seven LCRE hex cans assemblies, containing pins made with depleted UO_2-BeO, was started. These cans will be used to check out dimensional variances in individual loaded cans and dimensional build-up in a nested bundle of cans. These tests are required because of the dimensional variances found in the explosively formed hex cans. It is necessary to relate these variances to the dimensions of filled cans and to core assembly tolerances.

The preliminary evaluation, performed during the past 18 months on the strength properties of cold-drawn Cb-1 Zr alloy fuel pin tubing for the LCRE (0.312 inch outside diameter x 0.013 inch wall) was made by extrapolating short time tests (less than 1000 hours) at temperatures of 1800F and 2000F. Subsequently,work has indicated that these tests were generally invalidated by nitrogen contamination. Evaluation of fuel pin tubing at 2200F and 2400F under closely controlled conditions to minimize contamination was initiated in October, 1962. Preliminary rupture data obtained at higher temperatures on as-received tubing and extrapolated by the Manson-Haferd parameter indicated that the 10,000 hour 2200F rupture strength was approximately 700 to 800 psi instead of the design strength of 2300 psi. For this reason an intensive program was initiated to develop means of strengthening Cb-1 Zr alloy tubing to be equal to or better than forged material.

Six high temperature biaxial stress rigs were available and operable. Since this was clearly insufficient to handle a large program, 24 additional rigs were put on order in December 1962. Twelve of these units will be completed in May 1963, and the remaining 12 in June 1963.

906 PIN TEST FUEL ELEMENT PARTIALLY DISASSEMBLED



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Stress-rupture tests of 2-inch long specimens of fuel pin tubing have been run on six heat codes of material purchased for LCRE use in the as-received and/or swaged condition with a 2200F anneal before testing. Swaging from 0.312 inch outside diameter to 0.305 inch outside diameter had no appreciable effect on the rupture strength of the tubing. These tests showed however that some heats of Cb-1 Zr alloy material are consistently stronger than others. The rupture strength at 2200F and 10,000 hours for the various heats of material varied between 800 psi and 1050 psi.

Preliminary tests have been performed in the current program on several means of strengthening the as-received Cb-1 Zr alloy tubing. The three principal means are:

a. Solution heat treatment of the "as-received" tubing for 30 minutes at 3100F.

b. Nitriding by a gas phase surface treatment followed by

   (1) Driving the nitrogen into the material by a 2200F anneal, or

   (2) Driving the nitrogen into the material by a 3100F solution heat treatment.

c. Carburizing by a spraying technique followed by

   (1) Driving the carbon into the material by a 2600F anneal, or

   (2) Driving the carbon into the material by a 3100F solution heat treatment.

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A number of tests have been completed on as-received tubing which had been given a 3100F one-half hour solution heat treatment. This treatment apparently increases the extrapolated 2200F, 10,000 hour rupture strength by approximately ten percent. Several tests have also been completed on solution heat treated fuel pin tubing having increased carbon or nitrogen content. From the limited number of tests which have been run at this time, increased nitrogen (500 to 800 ppm) increases the 2200F, 10,000 hour rupture strength by approximately 30 percent. Increased carbon (1000 ppm) increases the 2200F, 10,000 hour rupture strength by approximately 100 percent.

The status and summary of the inpile capsule test program at the MTR are shown in Fig 7. A photograph of the MTR, with the top head removed during a recent shutdown, is shown in Fig 8. A total of 18 Pratt & Whitney Aircraft capsules are shown inserted in the reactor.

Engineering of a capsule to be irradiated in a fast flux region of the Engineering Test Reactor core is proceeding. The Phillips Petroleum Company is making thermal and fast flux measurements and determining gamma heating rates for the facility which appears most suitable. The purpose of these experiments is to mock up more accurately LCRE fast flux conditions and then to determine more precisely the helium generation and release rates in the $BeO-UO_2$ fuel.

Samples of beryllium and BeO have been inserted into the Battelle Memorial Institute reactor core so that a measurement of the n, 2n and n,α fast flux helium generation cross sections may be made without the complication of fission-produced helium. These specimens will be removed from test late in April. The main purpose of these experiments is to attempt to predict helium release in the LCRE. While helium release is measured from $BeO-UO_2$ inpile samples, the relative contributions from the n, 2n, and n,α reactions and from fission are markedly different in the LCRE flux spectrum and in the more thermal spectrum of the MTR.

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STATUS AND SUMMARY OF LCRE CAPSULE IRRADIATION PROGRAM

Capsule Number	Duration of Irradiation, hr	*** Average Operating Temp of Specimen Surface, F	Test Facility Thermal Flux, nv	*** Cesium-137 U^{235} Burnup, %	Fission Density Fissions/cc	Power Density kw/cc	% Xe Release	% He Release	Cladding OD Change, mils	Cladding Reaction	Irradiation Dates
A. Pin-type specimens (Cb-1 Zr alloy cladding), 0.312 OD x 0.035 wall cladding, 50 v/o BeO, 0.238 inch D x 0.5 inch long pellet height, 1 kw/cc in Li at 2000F design											
PW26-10	10,000*	1986**	1.7×10^{13}	9.7*							5/62 to 1/64*
PW26-11	1,120	1990	1.6×10^{13}	0.8	8.6×10^{19}	0.68	0.44	20. (2)	None	None	1/62 to 4/62
PW26-12	3,595	2000**	1.6×10^{13}	2.4	2.6×10^{20}	0.66	0.08	15. (2)	None	None	4/62 to 10/62
PW26-13	3,511	2000**	1.5×10^{13}	2.5	2.7×10^{20}	0.66	0.85(T)	18. (2)	None	None	2/62 to 8/62
PW26-14	324	900 (1)	1.3×10^{13}	0.2	2.0×10^{19}	0.56	0.04	6.3 (2)	None	None	2/62 to 3/62
PW26-15	6,000*	1989**	1.5×10^{13}	6.0*							5/62 to 5/63*
PW26-16	10,000*	2004**	1.6×10^{13}	9.7*							4/62 to 1/64*
PW26-17	734	2000**	1.7×10^{13}	0.7	7.3×10^{19}	0.9	0.71	17 (2)	None	None	3/62 to 4/62
PW26-18	6,000*	1994**	1.6×10^{13}	6.0*							4/62 to 5/63*
B. Pin-type specimens (Cb-1 Zr alloy cladding), 0.305 OD x 0.015 wall cladding, 50 v/o UO_2-50 v/o BeO, 0.272 inch D x 0.72 inch long pellet height, 1 kw/cc in Li at 2200F design											
PW26-40	1,000*	2200*		1.0*							4/63 to 5/63*
PW26-41	1,803	2161	2.5×10^{13}	1.3	1.4×10^{20}	0.7	2.1	16 (2)	2.4	Grain growth	10/62 to 12/62
PW26-42	1,494	2101	2.8×10^{13}	1.5*							12/62 to 3/63*
PW26-45	10,000*	2180	2.7×10^{13}	9.4*							10/62 to 6/64*
C. Pin-type specimens (Cb-1 Zr alloy cladding), 0.305 OD x 0.015 wall cladding, 60 v/o UO_2-40 v/o BeO, 0.262 inch D x 0.75 inch long pellet height, 1 kw/cc in Li of 2000F design											
PW26-50	1,228	1939	1.5×10^{13}	0.7	8.9×10^{19}	0.65	0.92	18.***	None	None	6/62 to 8/62
PW26-52	1,228	1879	1.4×10^{13}	0.6	7.7×10^{19}	0.56	0.82	17.***	None	None	6/62 to 8/62
D. Pin-type specimens (Cb-1 Zr alloy cladding), 0.305 OD x 0.015 wall cladding, 50 v/o UO_2-50 v/o BeO, 0.272 inch D x 0.72 inch long pellet height, 1 kw/cc in Li at 2200F design											
PW26-60	6,000*	2200*	2.8×10^{13}	5.8*							2/63 to 2/64*
PW26-61	10,000*	2177	2.5×10^{13}	9.4*							1/63 to 7/64*
PW26-62	10,000*	2192	2.5×10^{13}	9.4*							1/63 to 7/64*
E. Pin-type specimens (Cb-1 Zr alloy cladding E.B. welded), 0.305 OD x 0.015 wall cladding, 50 v/o UO_2-50 v/o BeO, 0.272 inch D x 0.72 inch long pellet height, 1 kw/cc in Li at 2200F design											
PW26-80	1,000*	2200*	2.3×10^{13}	1.0*							6/63 to 8/63*
PW26-81	3,000*	2200*	2.3×10^{13}	3.0*							6/63 to 12/63*

*Design or estimated values
**Approximate temperature - thermocouples are questionable
***Center test specimen
(1) Heater failed at start up because of an operator error
(2) Top test specimen

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FIG 8

MATERIALS TESTING REACTOR WITH THE TOP HEAD REMOVED SHOWING 18 PRATT & WHITNEY CAPSULE EXPERIMENTS

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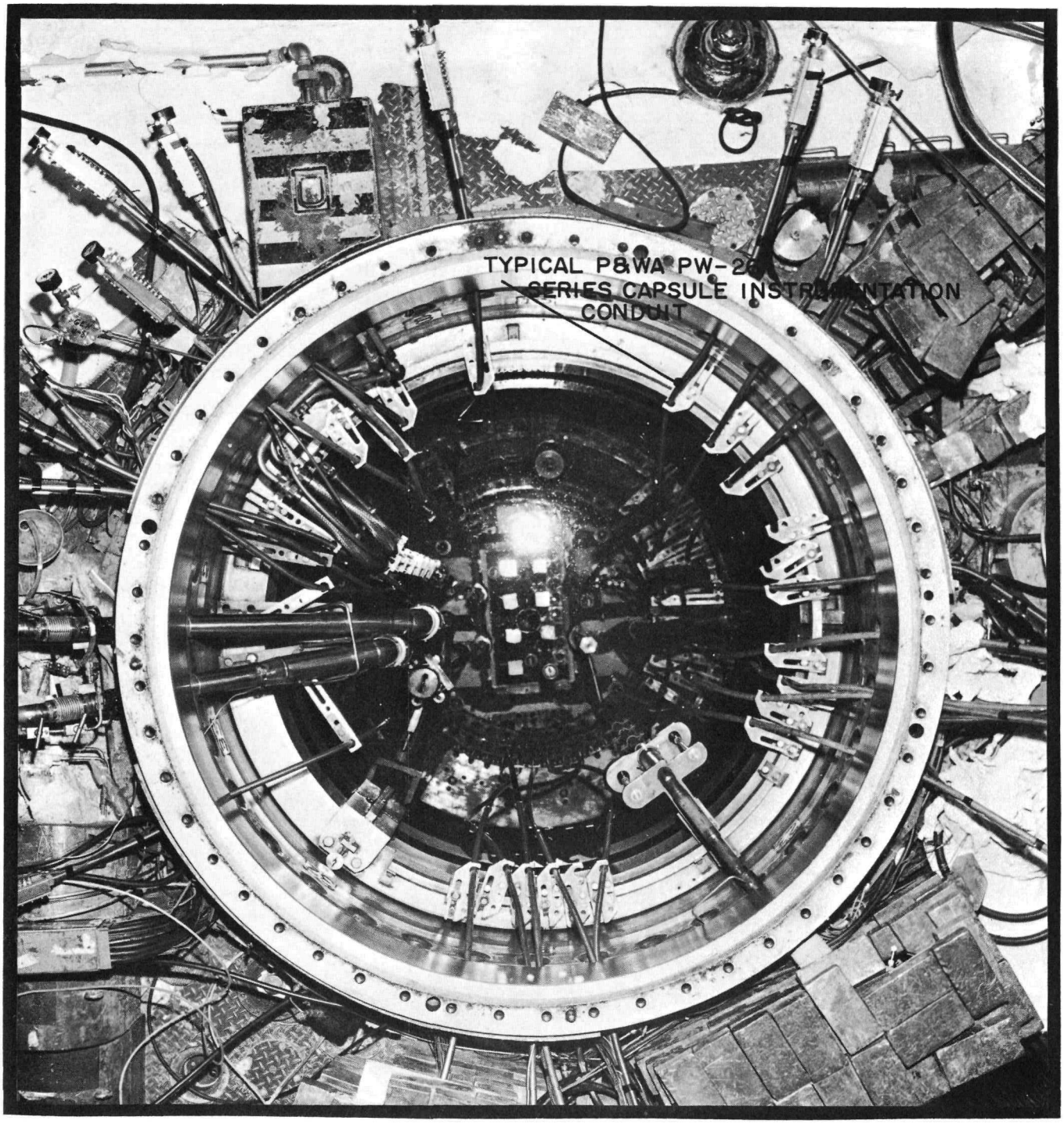

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2. Critical Experiments and Physics

Further experiments on the CCA-6 critical assembly with a 13.5 inch diameter, 12 inch long core were completed. The set of reactivity coefficients, for mock-up and reactor materials with control drums at the 60 degree orientation, was completed. Measurements were made for stainless steel in the reflector and on Li 7 in the core.

The as-built total worth of the LCRE control drums will be derived from single drum calibrations to be carried out during and following the Fuel Loading Experiment. Therefore, a set of single drum-worth measurements was taken in CCA-6 (12 inch) with the remaining drums set at 60 degrees, to determine the ratio of single to total drum-worth. The average value for eight of the ten drum mock-ups while turning a single drum from zero degrees to 180 degrees was found to be 157 cents plus or minus three. For the other two mock-ups, the single drum-worth was 167 cents.

All of the CCA-6 (12 inch) control drum mock-ups were then turned to the zero degrees. The U-235 critical mass for this configuration was 89.5 kilograms. A set of reactivity coefficients was measured for the mock-up and LCRE materials in the core and for selected materials in the end and side reflectors and filler regions. With few exceptions, no significant differences from the values measured at 60 degrees were observed.

Typical of material coefficients which did appear to change was that of BeO. It rose from a value of 79 cents per kilogram with drums at 60 degrees to 83 cents at zero degrees.

3. Reactor

A four-inch, Cb-1 Zr alloy pressure vessel, filled with lithium and pressurized to produce a hoop stress of 1200 psi at 2000F, completed a scheduled 10,000 hour test. This vessel was wrapped with a single layer of tantalum foil as a surface getter. Fig 9 shows the vessel in its foil wrapping after test. The tips of all the type 316 stainless steel thermocouples were swollen and discolored, apparently damaged by diffusion of oxygen through the electrical insulation during the long test period. The foil was badly discolored and dimpled under one thermocouple. The vessel is shown in Fig 10 with the foil removed. The dark spot on the vessel cylinder was under the dimple in the foil. The tantalum foil from the vessel exhibited nitrogen contents ranging from 32,000 to 39,000 ppm and oxygen from 2100 to 13,700 ppm. Nitrogen and oxygen contents from various representative areas of the test vessel itself were as follows:

Location	N_2 ppm	O_2 ppm
Flat Head inner surface	2800	175
outer surface	3600	530
Cylinder I. D.	4800	120
O. D.	6200	140
Dome I. D.	4400	105
O. D.	7200	175
Dark spot on cylinder	-	620

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FIG 9

Cb-1 Zr FOUR INCH PRESSURE VESSEL AFTER TEST

TEST CONDITIONS:
- TEMPERATURE - 2000F
- STRESS - 1200 PSI HOOP
- DURATION - 10,000 HOURS

ENVIRONMENT - INTERNAL, LITHIUM
EXTERNAL, HELIUM

FOIL WRAP - TANTALUM

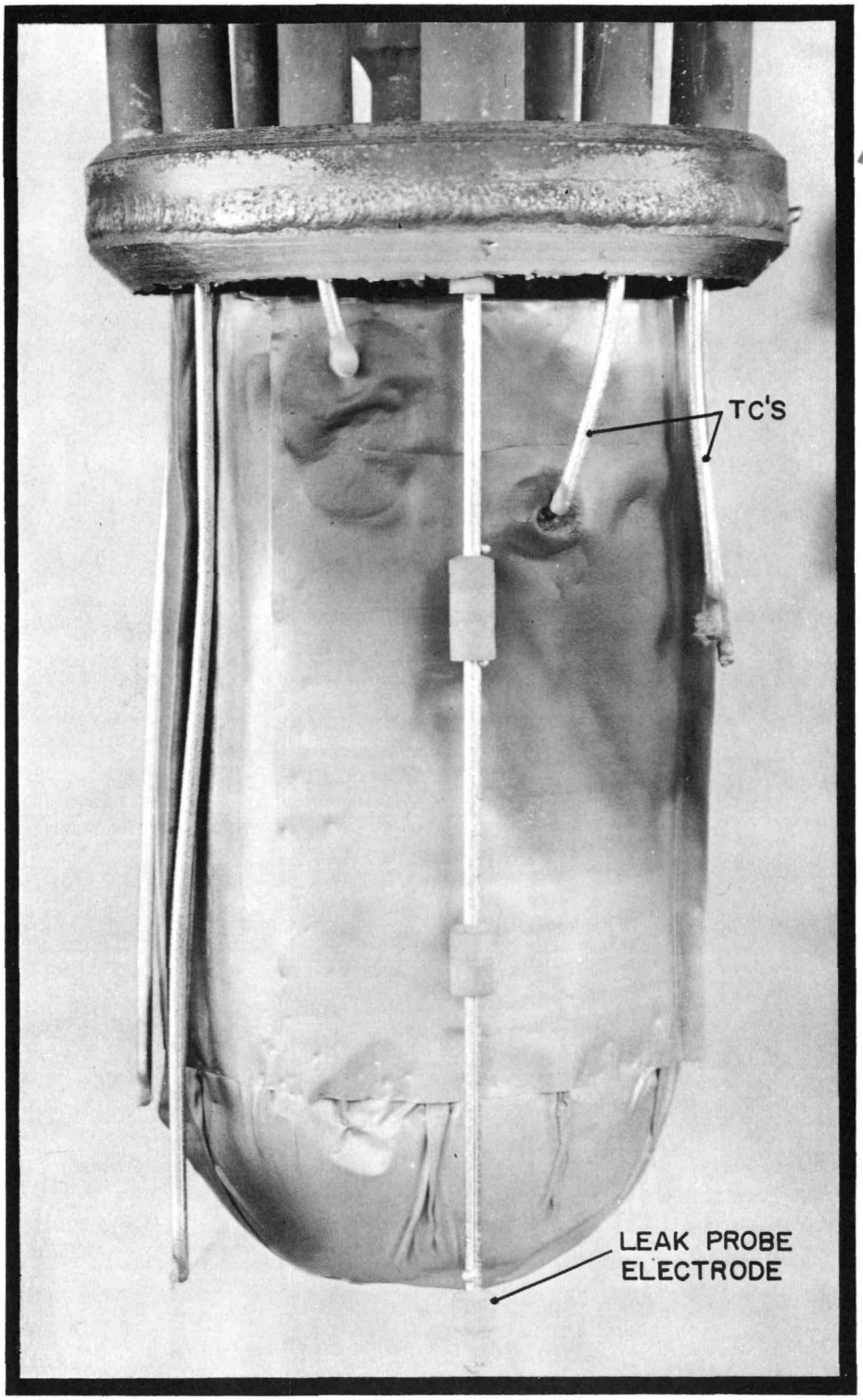

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FIG 10

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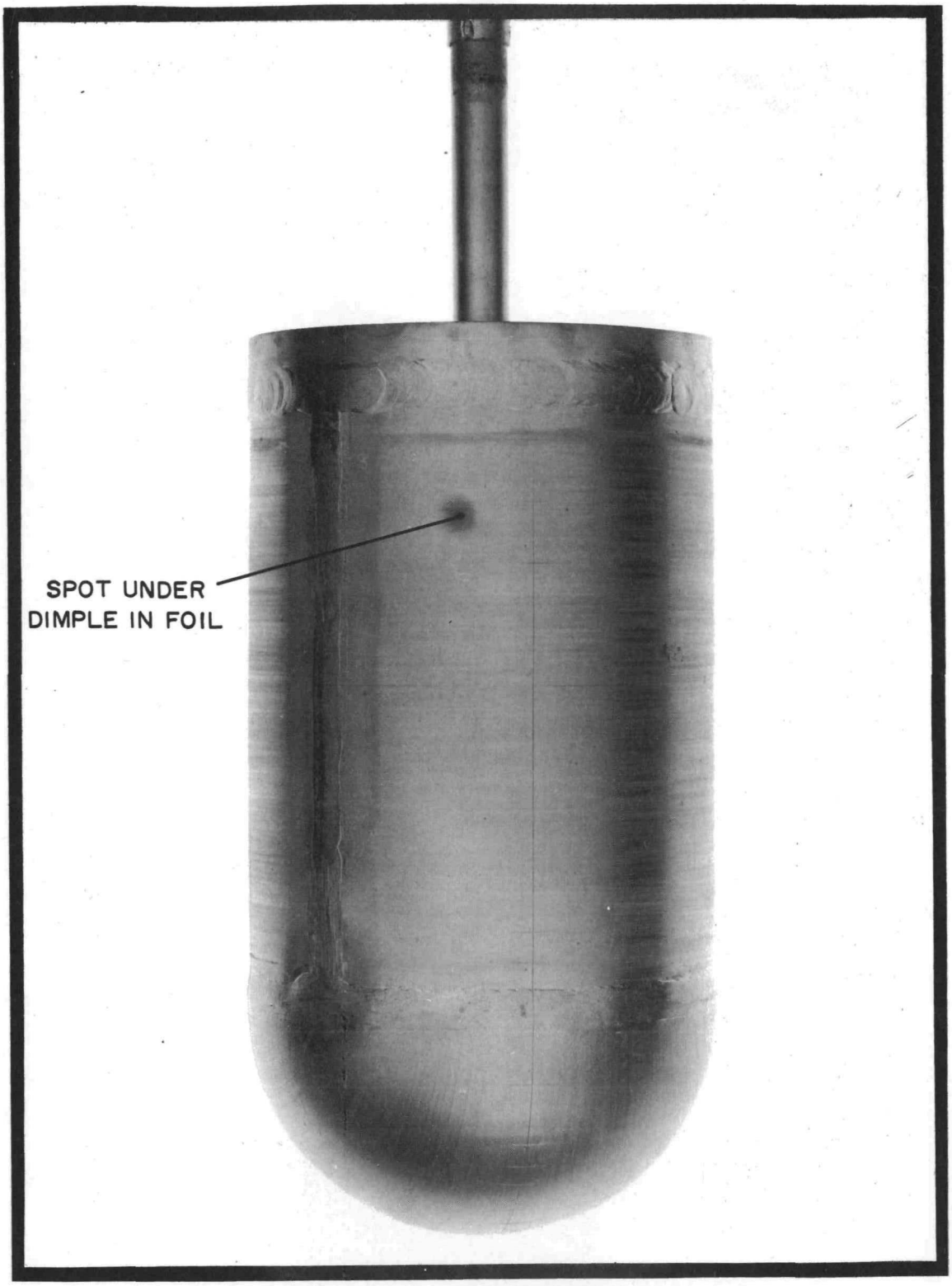
2000F Cb-1 Zr FOUR INCH PRESSURE VESSEL AFTER TEST WITH THE TANTALUM FOIL REMOVED



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The flat head of the vessel was bowed upward approximately 0.035 inch. No measurable creep was found in the cylinder.

Two 4-inch pressure vessels are being tested with single protective layers of columbium foil as a surface getter to prove its adequacy for wrapping the active core region of the LCRE core pressure vessel. One of these vessels, filled with lithium at 1600F and pressurized to produce a hoop stress of 6000 psi, has completed 2537 hours of a scheduled 10,000 hour test. The second vessel has completed 1296 hours of a scheduled 10,000 hour test at 200F with a 1200 psi hoop stress.

The Cb-1 Zr alloy 10-inch test vessel has now completed 7848 hours of a scheduled 10,000 hour test. This test operated for 4370 hours at 2000F with an effective stress of 2000 psi in the vessel cylinder. At that time the upper heating zone of the test rig burned out and the test temperature dropped to approximately 1900F. The effective stress in the vessel cylinder was then increased to 2300 psi to compensate for the loss in temperature.

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The fabrication of the full scale test vessel was completed. It was wrapped with a single layer of columbium foil along the cylinder and three layers of tantalum foil in all directions. The vessel was installed in its test stand, shown in Fig 11, and filled with lithium. The planned test conditions were 2000F, 45 psig internal pressure and 10,000 hours test duration. While bringing the temperature to test conditions the temperature had reached 1870F when a gas leak developed in the Inconel containment vessel where the tube from the expansion tank joined the top cover of the test specimen containment vessel. Helium was purged through the break and the rig was cooled as rapidly as possible by removing thermal insulation and heating elements. All portions of the columbium alloy system were cooled to 400F or below within 12 hours. Oxygen analyzers were connected to gas lines on the main containment vessel, the dump tank and the expansion tank containment vessel during the cooling process. The maximum reading was 3.5 ppm of oxygen in the expansion tank containment. This indicated negligible back diffusion of air into the cover gas. No apparent damage to the test vessel occurred. The break, shown in Fig 12, was apparently caused by a high stress in the tube due to differential thermal expansion in parts of the containment vessel. A section of Inconel tube was removed just above the break and a piece of the outer layer of tantalum foil was removed and examined. This foil contained 1.5 percent oxygen. The layer of foil beneath this was ductile and was not discolored. This region has now been redesigned to provide freedom for thermal expansion for the cover gas tank and the tank is being repaired.

The program for 20-millimeter non-lubricated bearings for use in the LCRE control drive gearbox was completed this period. Of the eight configurations tested, four are well qualified for the dry, 500F helium application in the drum drive gear boxes. The five tests completed this quarter were conducted at 500F under helium for 2000 hours. Material and configurations are tabulated below:

No. of Tests	Part No.	Races	Balls	Cage
2	1022900	Stellite 19	Stellite 3	17-4 PH
1	1022897	M-50	Carbaloy 44	Inconel X
2	1022894	M-2	M-2	17-4 PH

The posttest examinations showed no significant surface deterioration or increase in friction.

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FIG 11

FULL SCALE PRESSURE VESSEL TEST INSTALLATION

FIG 12

BREAK IN COVER GAS CONTAINMENT TUBE
FULL SCALE PRESSURE VESSEL TEST

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Six 600F dry non-lubricated harmonic bearing tests were completed this period:

Bearing		Materials			Test Condition	
Part No.	Type	Rings	Rolling Element	Cage	Cycles	Speed (RPM)
1029731	Ball	M-2	M-2	Inconel X	2×10^6	500
1029732	Ball	440-C Mod	M-2	Inconel X	5×10^6	10
1022917	Roller	Stellite 19	Stellite 3	17-4 PH	5×10^6	10
1031232	Ball	M-2	M-2	17-4 PH	5×10^6	10
1029732	Ball	M-2	M-2	Inconel X	2×10^6	500
1031232	Ball	440-C Mod	M-2	17-4 PH	5×10^6	10

Average peak posttest torques for these tests were less than ten inch-pounds in all cases. Wear was found to be well within acceptable limits. Five Harmonic Drive flex tubes have now accumulated 7 to 11 million flex cycles without damage.

A test of two Norden high temperature synchro transmitters was completed. One synchro failed at 1910 hours under 350F helium environment. The second operated 2120 hours at 350F and at 600F for an additional 340 hours before failure. The mode of failure was found to be open circuits in the rotors, probably caused by the extreme fragility of the magnet wire used. On the recommendation of the manufacturer, the mode of operation of the LCRE synchro transmitters was modified to permit a change from 60 cps to 400 cps. This made possible an increase in magnet wire diameter from 0.007 inch to 0.0126 inch.

Two LCRE prototype circuit switches each successfully completed 7000 electromechanical cycles at 350F followed by 31,000 cycles at 500F in dry helium gas.

A power train spur gear test operating at 600F in helium completed 2000 hours at LCRE shim speed with a scram condition imposed once per hour. Wear on the involute surfaces was found to be within acceptable limits. The gear material was silver-flashed AMS 5616.

A printed circuit, scram actuating motor test in 500F helium failed after 1020 scram cycles. Posttest examination showed that the aluminum oxide rotor had cracked, causing a high spot in the commutator region. This precipitated excessive brush wear. Better inspection techniques were developed to insure that defective (cracked) rotors will be screened out before installation.

Four control drum journal bearing tests operating under 37.5 pound radial loads were completed as follows:

No. of Tests	Material	Temp, F	Environment	Test Hours
1	CbC in Molybdenum	1000	Helium	500
2	WC in Molybdenum	900	NaK	2000
1	CbC in Molybdenum	Ambient	Air	100

The first two tests were successful. The third demonstrated excessive torque and subsequent examination showed that an assembly error caused an interference of parts.

One Harmonic Drive flex gear completed 1.36 million cycles on an AMS 5616 harmonic gear set. The test was operating under a 250 pound-inch reversing torque load at 600F in eutectic NaK. Posttest examination showed insignificant tooth form wear and a slight indication of metal pickup between the flex gear outside diameter and the flex tube inside diameter.

An inpile test of a shim actuating motor stator containing Phelps-Dodge Ceramic-Eze windings has operated 529 hours with an accumulated dose of 1.3×10^{19} nvt. No change in resistance has been observed.

All control console mock-up panels have now been installed except one which is awaiting the receipt of developmental models of the chamber and source position read-out packages. A relay control chassis and a power supply chassis for the control console circuits were constructed, installed in the mock-up and tested.

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All developmental models of the drum position read-out packages have been received, tested for performance and installed on the control console mock-up. A number of these indicators produced considerable noise when energized. This was found to originate with rotor vibration of the servomotors installed in the packages. The package structure and cover resonated as a result of the motor vibrations which in themselves were not excessive. The noise level was improved to within acceptable limits by the installation of one mil end-play shims in the servo motors of the noisy units.

The control system mock-up was operated for approximately 32 hours. Twenty hours of this time were required for initial checkout and trouble shooting of wiring errors in the interconnecting cabling. The remaining 12 hours were used for operation inthe closed loop flux control mode while connected to the analog computer representation of the LCRE. The system was found to perform satisfactorily.

A modification of the reactor control system to a flashing light indicator on the control console, whenever a large positive error signal occurs under servo mode of operation, was bench tested. (A positive error signal is one which calls for an increase in reactivity.) The nuclear operator will be alerted by this light whenever a significant positive change in power is requested or if the feedback indication of flux or temperature has been diminished appreciably for any reason.

The scram static contactor breadboard has now logged 2400 hours and 100 cycles without malfunction. The contactor controller breadboard has reached 1900 hours and 500 cycles of operation satisfactorily except for two lamp failures. The voltage has been reduced to increase bulb life.

The alarm light unit has logged 2100 hours and 250 operations with only bulb failures resulting. The power supply section of this unit was modified to increase its load carrying capability thereby increasing the number of lights which may be safely powered from each supply.

The low voltage, d-c power supply which will be used for manual tripping of certain safety actions has operated 3100 hours in breadboard version without failure. An OR logic unit breadboard has performed for 700 hours without failure. A breadboard TWO-AND unit was found to exhibit spurious tripping as a result of transient voltages introduced through the laboratory power lines. This behavior led to an extensive investigation of the possibility of spurious tripping of all breadboard logic circuits. As a result, small

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capacitors have been added to all logic unit, silicon controlled rectifiers from gate to cathode in order to bypass any transient voltage disturbances which might occur. The logic circuits now show stable operation for all input voltage values up to 0.5 percent of the tripping value.

Evaluation models of annunciators were received, tested and found to deviate from the characteristics specified for the LCRE application. These units were returned to the vendor for modification.

4. Primary and Reflector Coolant System

Three Cb-1 Zr alloy, primary system, pipe static tests have completed 5891, 5777 and 4169 hours of the scheduled 10,000 hour test program. The test specimens consist of short lengths of primary piping capped at each end and containing lithium at 2000F, pressurized to 56.5 psig. A fourth test was shut down after 3366 hours of test when lithium vapor was detected in the containment vessel. A complete metallurgical examination of the test specimen failed to reveal any indication of failure. It was concluded that the leak was too minute for detection by metallurgical examination.

The LCRE jacket heat transfer test rig began operation during the quarter. This test rig consists of a length of Cb-1 Zr alloy, 2.250 inch diameter pipe, surrounded by heat shields and jacketing as designed for the LCRE primary system. Calrod heaters are attached to the outside of the jacket as is planned for the LCRE. A Glo-bar type heater is inserted in the Cb-1 Zr alloy pipe to simulate hot lithium. Several experiments were conducted during this quarter. Test data were obtained for heating rates and temperature distribution in the pipe, baffles and jacket.

One of the three primary system pipe, mechanical strain cycling rigs has been completely fabricated and installed into the test stand. Fabrication of the other two test rigs was continued. Construction of the test stands was completed.

The Pratt & Whitney Aircraft isolation valve, designed for use in the primary fill and drain system, has accumulated over 2000 hours of a scheduled 10,000 hour test program in 1000F lithium. This value has been cycled, open-close-open, 850 times.

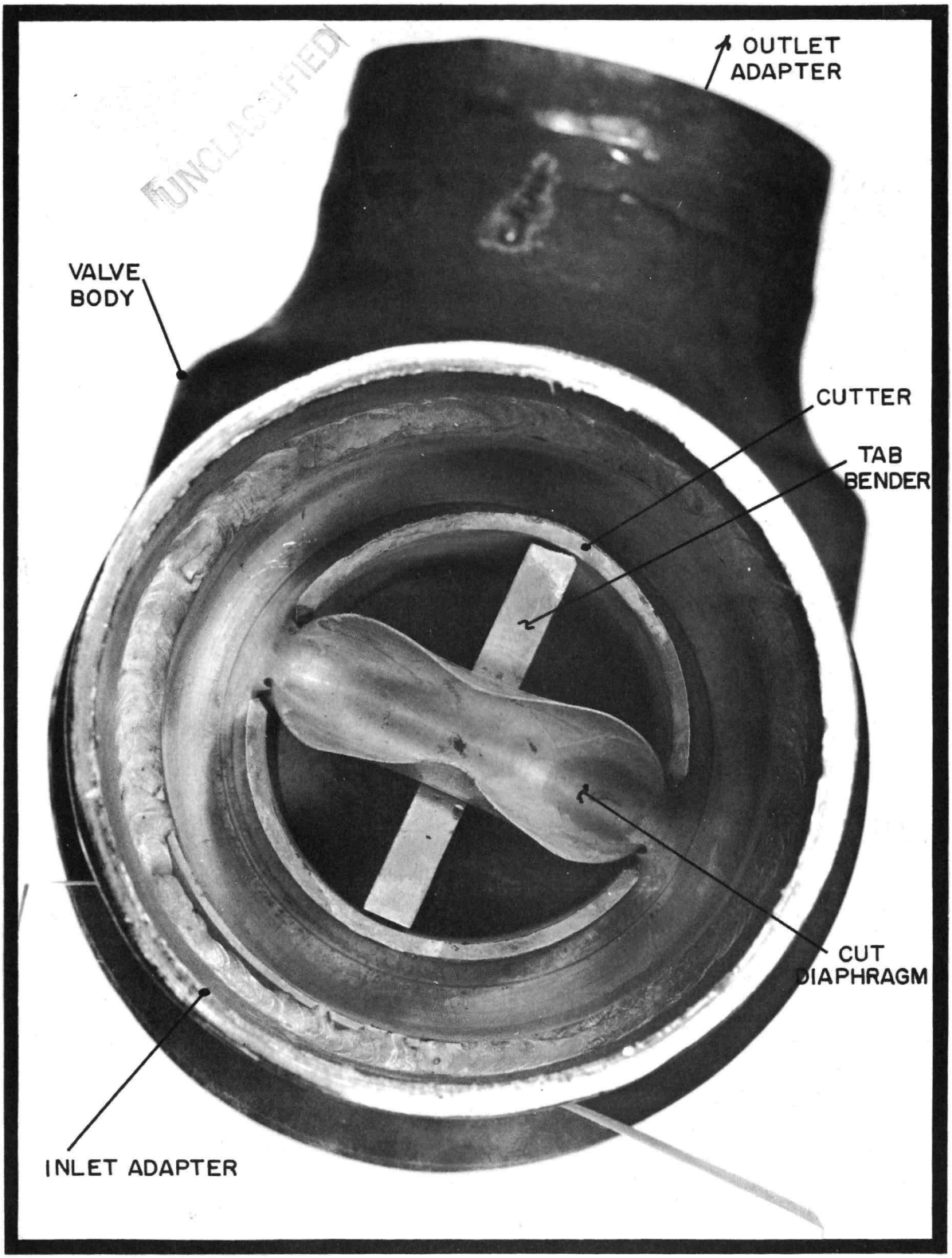
The diaphragm of the development punch valve was cut after having contained 1000F lithium for the scheduled 1000 hours. After cutting the diaphragm, the valve was tested as an isolation valve. The seat leakage rate was measured at approximately 40 cc of lithium per day. In cycling the valve to evaluate bellows integrity, the tab-bender-to-cutter fillet weld failed, allowing the shoulders on the tab-bender to ride out of the cutter slots. Interference between the cutter and the valve inlet adapter resulted. This prevented closing of the valve after seven open-close-open cycles and 1360 hours of testing at 1000F. Fig 13 shows the cut diaphragm after test.

Fabrication of the first LCRE type punch valve was completed during this period. A photograph of the completed valve prior to installation in the test rig is shown in Fig 14. Corrosion tests of 0.016, 0.018 and 0.025-inch thick type 316 stainless steel punch valve diaphragms have demonstrated satisfactory resistance to corrosion for periods ranging up to 5000 hours in 1000F lithium. Tests are continuing to 10,000 hours.

Testing of the Li 6 addition system, water-flow mock-up was completed during this quarter. Modifications to the gas system were made in order to maintain a constant supply tank pressure and to provide for better flow control. Additional instrumentation was provided to determine the flow characteristics of both the valves and the system.

FIG 13

PUNCH VALVE SHOWING DIAPHRAGM AFTER TEST



PROTOTYPE LCRE PUNCH VALVE

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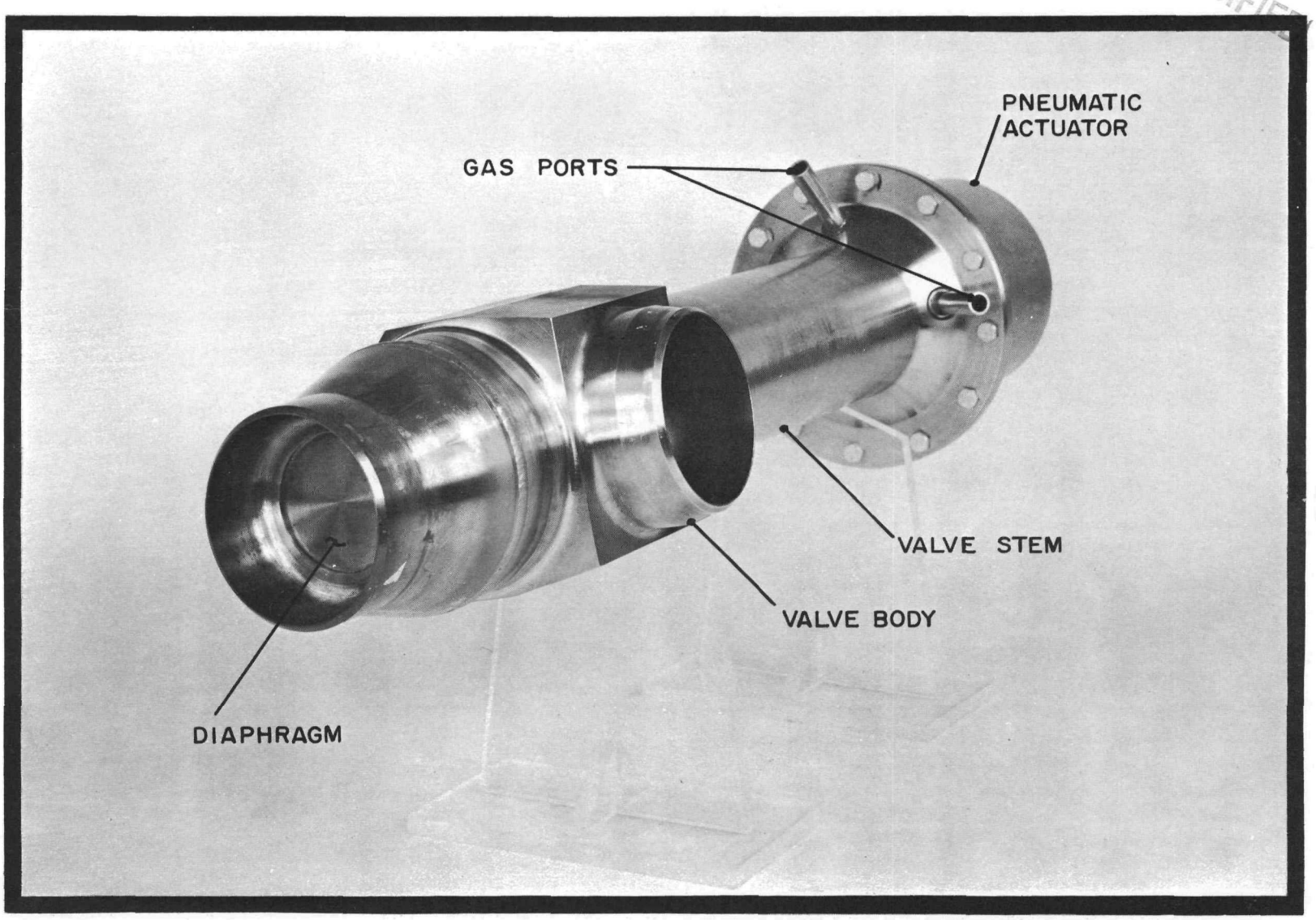

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The fabrication of vapor traps XI, XII and XIII was completed. These units are similar to traps IX and X which were described in report PWAC-624, and can be selectively fitted into a common vapor trap housing. All five cores incorporate an internally cooled, helical fin varying only in fin pitch and/or slope. Preliminary performance testing with lithium has been completed on all five units at temperatures ranging up to 1400F and helium flow rates to 200 SCFH. Two of five units will be selected, based on the preliminary testing, for more intensive performance and endurance test programs.

Vapor traps are also being tested in conjunction with the LCRE pump tests. Vapor trap X-LCRE-1 consisting of the X geometry fabricated to LCRE structural requirements, (i.e. all welded construction and heavy container wall thicknesses) was installed in the lithium pump test stand, PT-5-D. This trap has completed over 1000 hours of testing. A repeated plugging of a gas control valve downstream of the vapor trap by lithium particles, during the first 300 hours of operation was eliminated by adjustments to the air cooling flow and trap temperature.

The first of two lithium pump seal endurance tests was terminated after 15,045 hours of operation. A view of the test rig prior to the scheduled shutdown is shown in Fig 15. The three seal assemblies performed very satisfactorily throughout the test. The oil leakage rate past the upper shaft face seal was not measurable and the average leakage rate past the lower oil-cooled seal was approximately 0.2 cc/hr. The dry-gas seal operated at a temperature of approximately 430F and the average leakage rate was approximately 0.03 SCFH of helium for a one psi differential. The test rig was operated at a constant shaft speed of 5600 rpm.

Upon disassembly of the test unit only a slight film of hydrocarbons was found in the pot (Fig 16) surrounding the lower shaft region. The region about the dry-gas shaft seal (Fig 17) also revealed only slight deposits of carbonaceous material. All three shaft seal assemblies, as removed from the 15,045 hour test, are shown in Fig 18. All three seal faces exhibited polished wear track patterns. Extremely low wear was measured on both oil-cooled seals. The combined rotor and stator wear for these seals was less than 0.002 inch. The measured rotor wear for the dry-gas shaft face seal was 0.008 inch; the stator wear was about 0.009 inch.

Light wear marks were noted on the upper roller bearing assembly. Sufficient excessive wear of the lower half of duplex thrust bearing occurred so that, upon disassembly of the test rig, the outer race and the balls fell away from the bearing assembly. This wear resulted in a considerable amount of end play in the shaft assembly. The cause for this wear is not known. A view of the shaft assembly removed from the test rig is shown in Fig 19.

Operation of the second seal endurance rig test continued. The total test time is now 9012 hours. The pressure differential across the dry-gas shaft face seal has been reduced from 1 psi to 0.5 psi, to simulate the current pump tests.

The program of tests of dry-gas face seals on the Monobloc test rigs has continued. A program has been instituted to investigate the flatness of the seal rotors after installation on the shaft. The results of this program will determine permissible limits of flatness for satisfactory seal operation.

The hydrodynamic tests of the twin pump and sump in water have been completed. Motion pictures of the flow patterns in the sump trough with the "J" type level probes installed have been obtained.

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FIG 15

SEAL ENDURANCE RIG AFTER 15,045 HOUR TEST

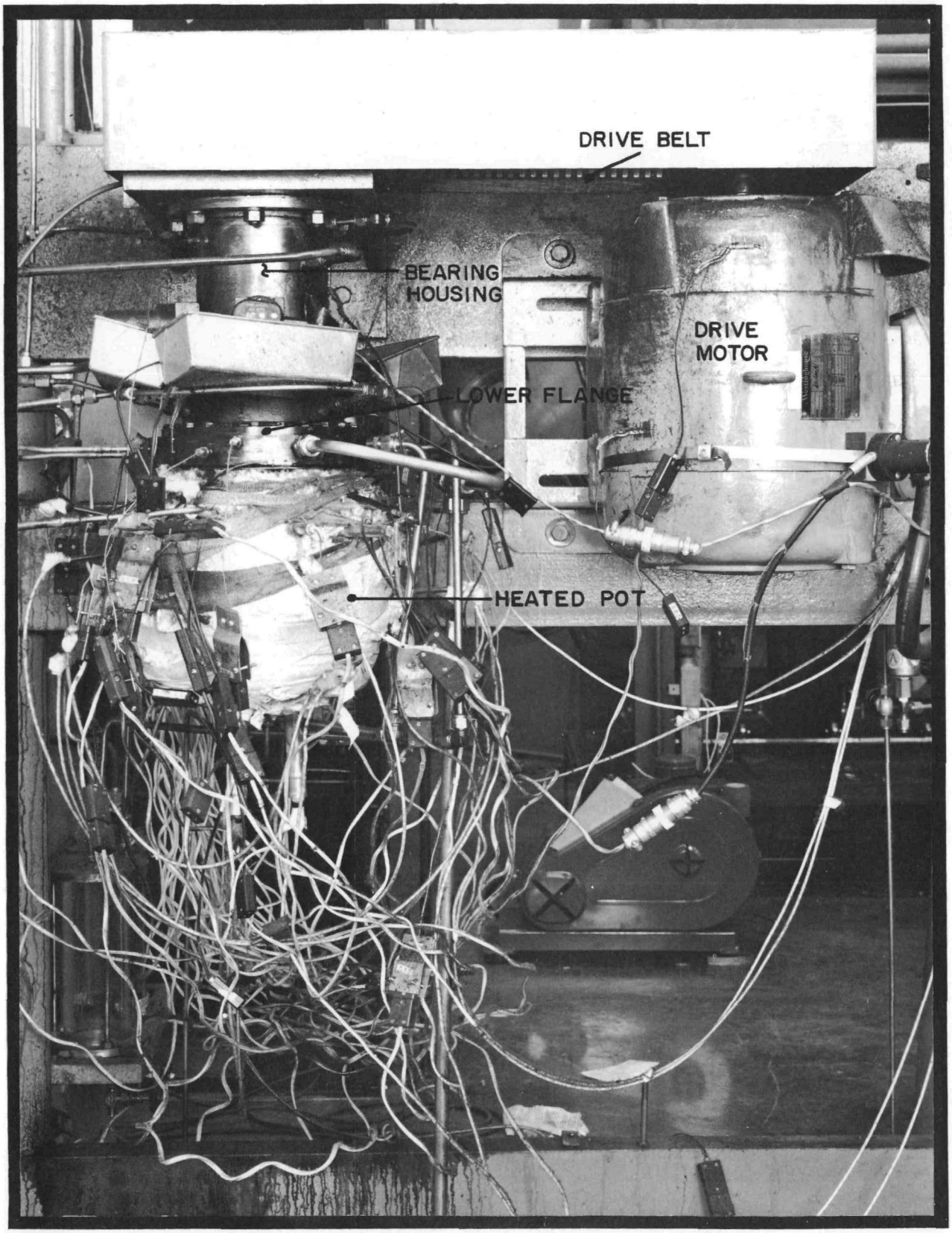

FIG 16

INTERIOR VIEW OF POT-SEAL ENDURANCE RIG

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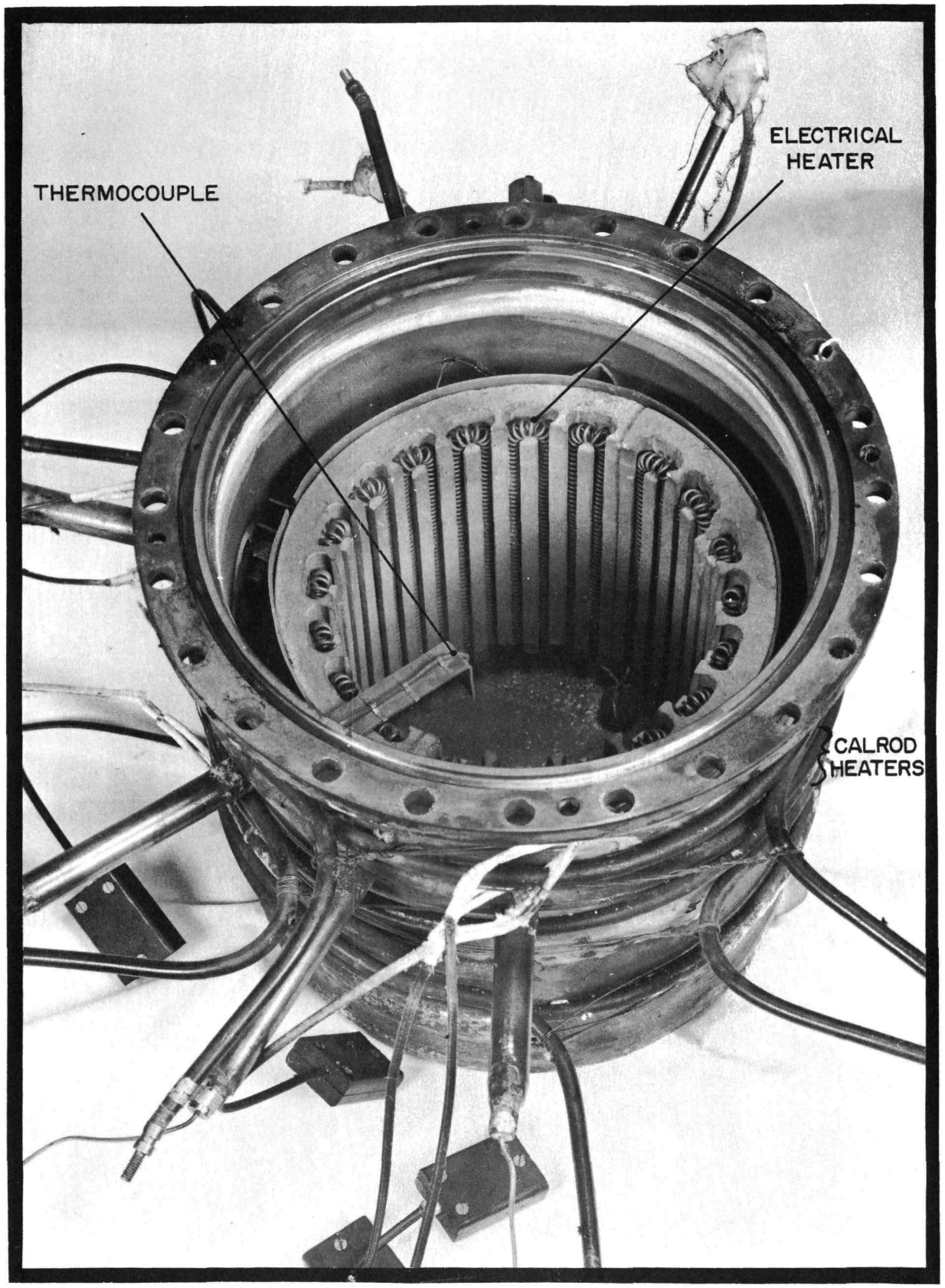

FIG 17

SEAL ENDURANCE RIG - DRY GAS SEAL REGION

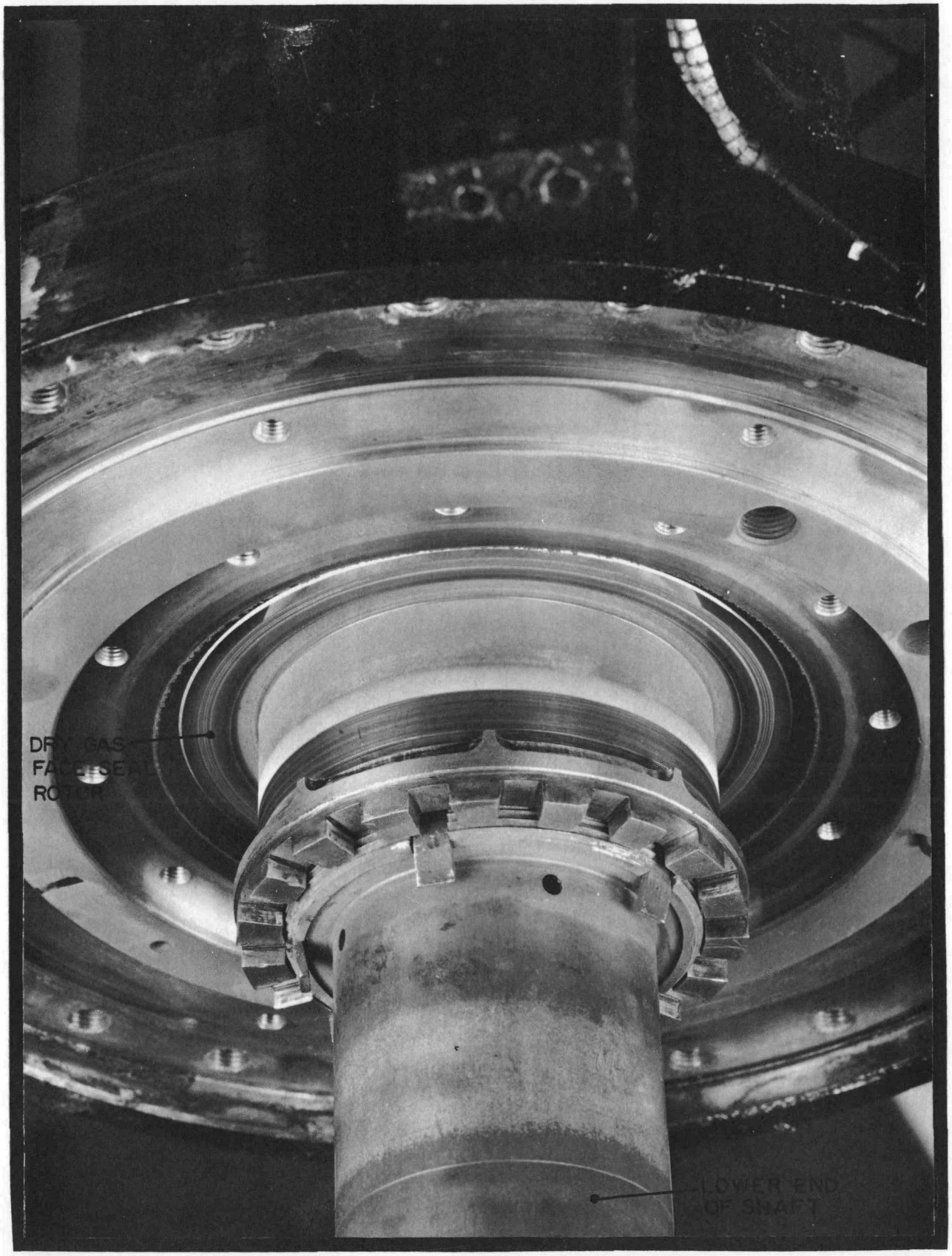

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FIG 18

SHAFT FACE SEALS AFTER 15,045 HOUR TEST

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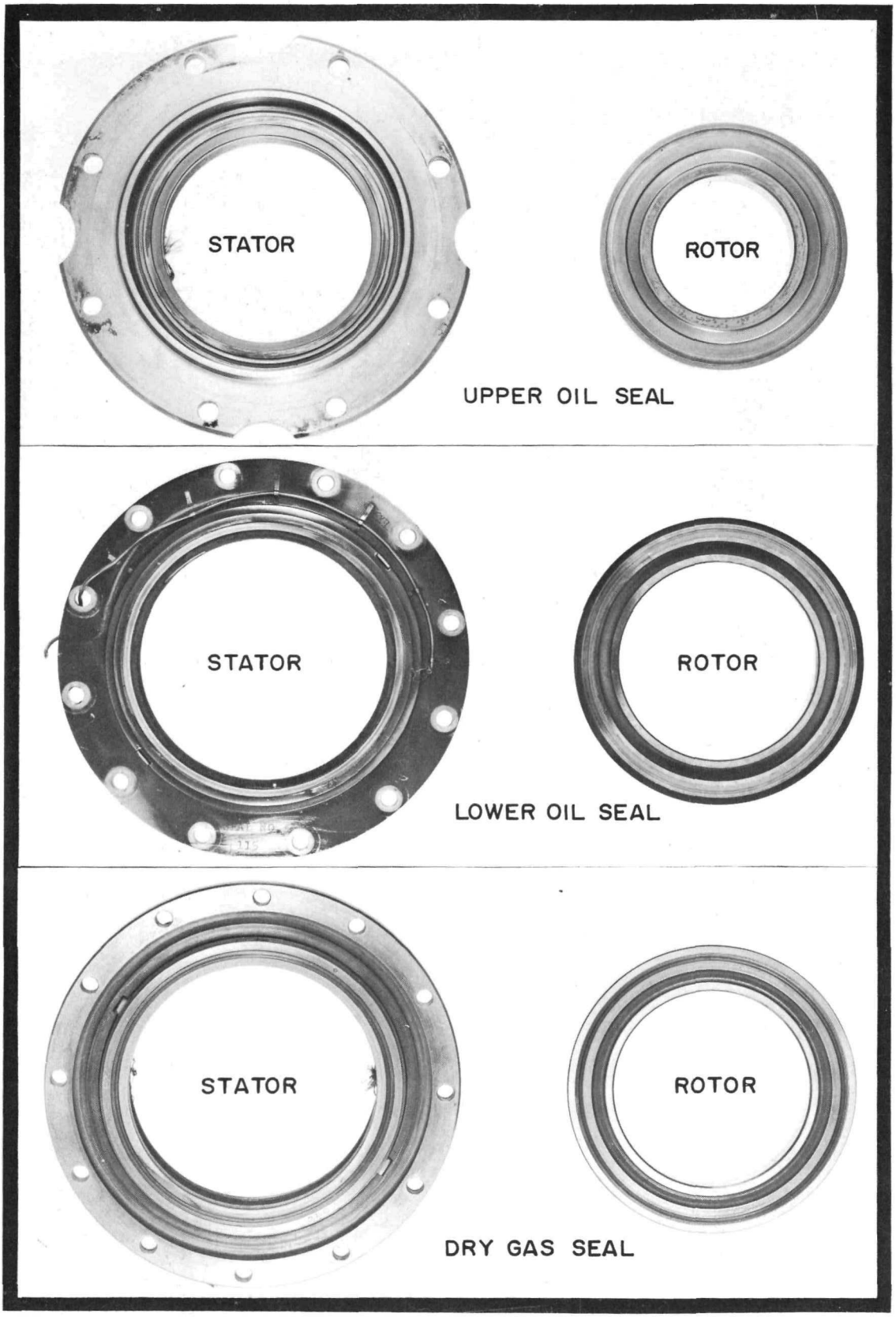

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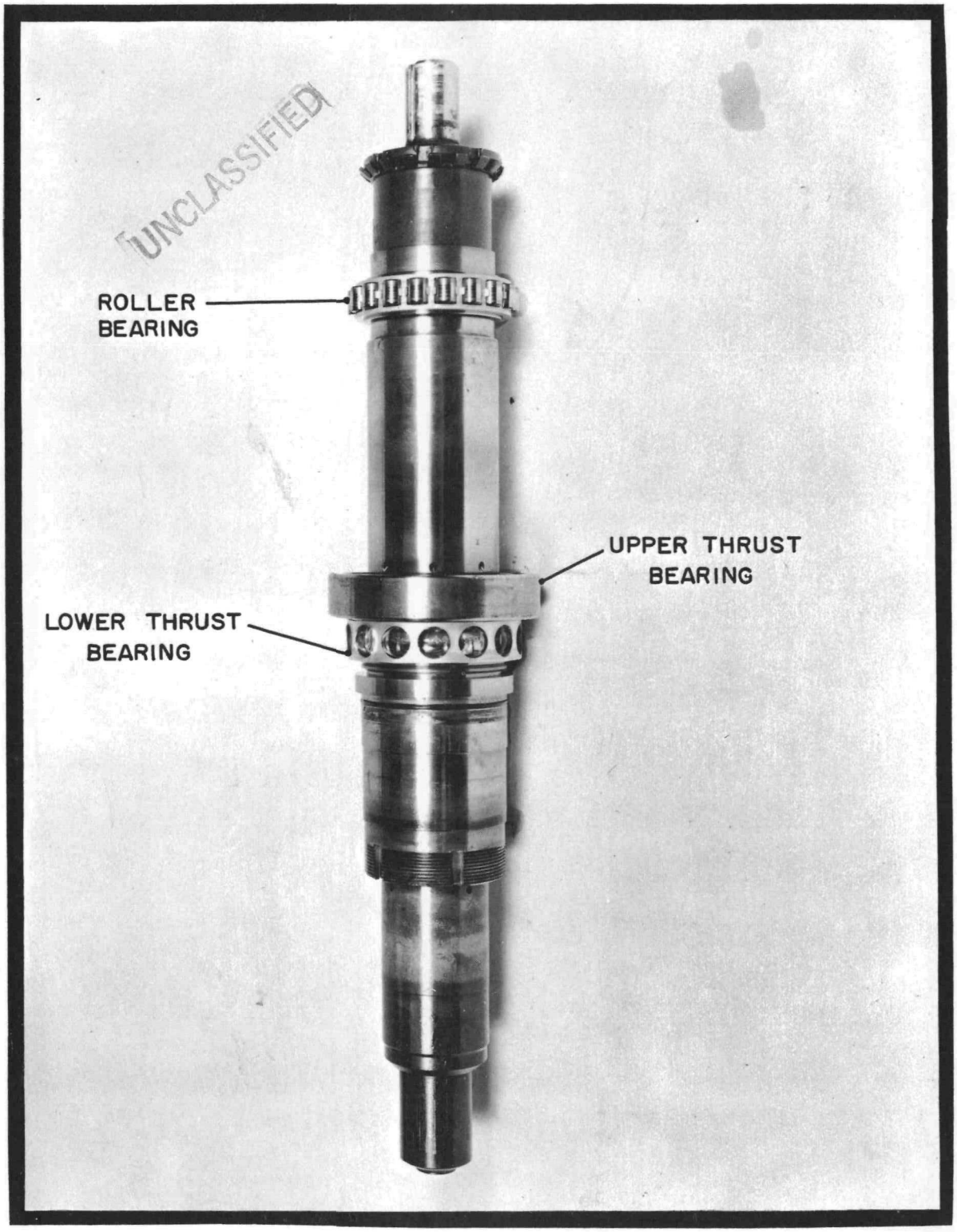
FIG 19

SHAFT AND BEARING ASSEMBLY AFTER 15,045 HOUR TEST



Two hot center section test units have continued to operate satisfactorily in 1000F lithium. These tests have accumulated 9725 and 6638 hours of total operation, respectively. The second center section test unit has been tested with a balanced pressure across the dry-gas shaft seal. Fluctuations in the oil leakage rate have been measured across the upper oil seal for this test unit. Although the leakage rate was not measurable for the first 3300 hours of test, it has now increased to an average value of about 4 cc/hr.

The first and second lithium pump and sump hot flow test units have now accumulated 6215 and 1064 hours of total operation, respectively. Both of these test units have continued to operate very satisfactorily.

Fabrication of the reflector coolant pump test unit and construction of its test stand were continued. This pump is of the same configuration as the lithium pump except that three equally spaced gas sweep-out lines are incorporated instead of one. This change resulted from the experience gained with eutectic NaK during the secondary pump development program. The test unit consists of a single pump and sump and is scheduled to begin testing with hot NaK during the next quarter.

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5. Secondary System Components

The first unit of the four development isolation valves has completed more than 7600 hours of a scheduled 10,000 hour test. This valve is being tested between 1200F and 1500F and has been cycled approximately 850 times.

The first of the prototype LCRE isolation valves, testing of which was terminated during the previous quarter because of mechanical wedging, has been rebuilt with a redesigned plug and seat with an included angle of 60 degrees. The rebuilt valve has been reinstalled in the valve test rig and has accumulated more than 150 hours of test operation at 1200F. During this test period the average seat leakage rate has been 40 cc per day.

Preliminary performance testing has been conducted on the five helical-fin vapor traps, described in the primary system development section, with eutectic NaK at temperatures ranging up to 750F and helium carrier gas flow rates up to 100 SCFH. From these evaulation tests, the two units exhibiting the best performance characteristics will be selected for further performance and endurance testing.

Endurance testing of the two 3 and 1/4-inch diameter Cb-1 Zr alloy-to-type 316 stainless steel coextruded joints, containing NaK at 1200F with an internal pressure of 64.5 psig, was continued during this period. These tests have completed 8121 and 7860 hours, respectively, of the scheduled 10,000 hour test period.

Cavitation tests of the secondary pump impeller, using a partial shroud constructed of Lucite, were terminated when the shroud fractured. A stainless steel partial shroud was fabricated and has been brazed to the impeller to permit experimental determination of the optimum shroud location.

The hot flow test of the secondary pump and sump has been successfully operated for 3412 hours in eutectic NaK. The current operating point is at 394 gpm with a head rise of about 143 feet in 695F NaK which corresponds to LCRE operating requirements. Minor plugging indications have been observed in the gas sweep circuit during this test but these have not caused any difficulties.

The secondary pump hot center section test unit is in the final stages of fabrication and assembly. Testing in hot NaK is scheduled to commence during the next quarter. A dry run of the center section test unit was terminated due to malfunction of the shaft face seals. The reassembly of this unit with relapped face seals has been started.

A cutaway illustration of the secondary pump, mounted in the sump for the LCRE experiment, is shown in Fig 20.

6. Non-Nuclear Systems Test

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Modifications to the gas system and to parts of the Non-Nuclear Systems Test (NNST) were completed. Improvements in the secondary pump seal and sweep-gas system were also completed and the pump was installed. A five hour shakedown run was completed on the secondary pump and a three hour run-in on the primary pump while adjustments were made on the gas controls and valves.

Helium and neon mass spectrometer leak checks were completed on the complete shroud gas system of the primary loop. Considerable time was expended in assuring a high quality leak tight system. The standard method of leak checking at CANEL is the detection of helium by a mass spectrometer. The NNST shroud had previously been filled with helium and the background caused by this residual helium on the mass spectrometer made it necessary to expend a great deal of time in heating and evacuation. It was not possible to remove fully the background but it was possible to detect 53 leaks. These were repaired. The shroud gas system was checked with neon to a sensitivity of 2×10^{-7} atmospheric cc/sec. The neon technique followed the helium and no further leaks were detected by it.

Approximately 25 pressure rate-of-rise tests were performed in order to detect leakage and to monitor off-gassing. During these rise tests the rates ranged from 30 down to six microns per hour for the shroud gas system and zero for the liquid metal side of the primary loop. The pressure rate-of-rise tests were performed by noting the increase in system pressure, as measured on a Pirani gage, after the system was purged and isolated.

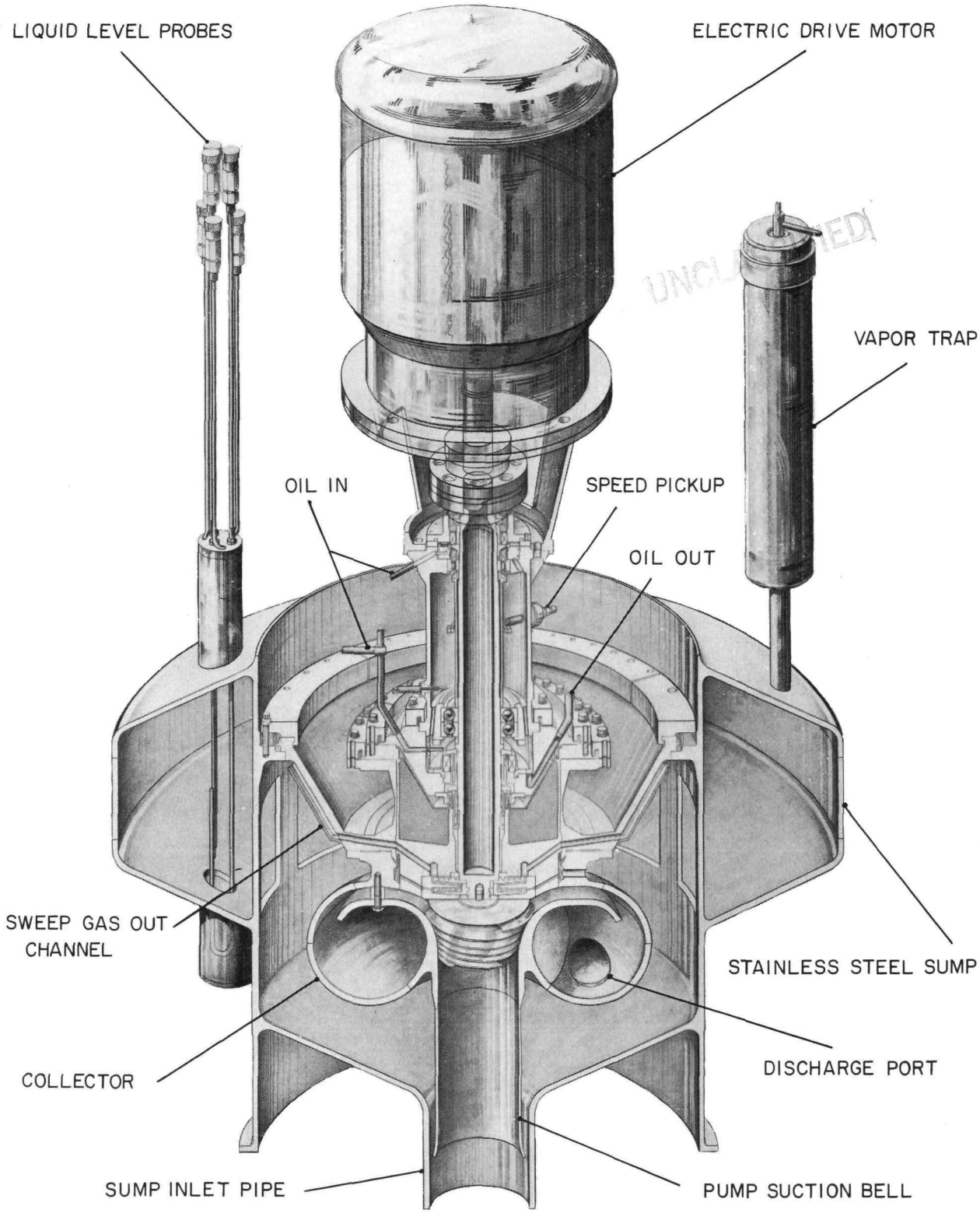
The cleanup of the shroud gas system of the primary loop was completed for temperatures ranging from 600F to 1000F. Three contamination rates-of-rise were taken and the improvement in moisture increase over a six hour period for the containment vessel was determined to be from approximately 31 to 3.8 ppm. This was accomplished by heating and recirculation purging at five volumes per hour. The final increases for a six hour period were 3.8 ppm moisture, 0.3 ppm oxygen and 0.3 ppm nitrogen.

The secondary system was filled with NaK on March 4, and cleanup of the liquid metal was accomplished with the sodium oxide trap and the calcium inhibitor system. A pure helium atmosphere in the primary loop shroud was established for temperatures up to 1000F. The primary loop was then preheated and filled with lithium and an 11 point performance test program was started.

Three points of this program were completed as follows:

FIG 20

CUTAWAY ILLUSTRATION OF THE LCRE PROTOTYPE SECONDARY COOLANT SYSTEM PUMP AND SUMP



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	Points 1	2	3
NaK			
Flow, gpm	394 (393.0)	394 (393.0)	394 (393.0)
Pressure at Pump Sump, psig	10 (9.5)	10 (9.5)	10 (9.5)
Temperature at Li-NaK Heat Exchanger Inlet, F	710 (710)	170 (708)	710 (706)
Pump Speed, rpm	4500* (4270)	4500* (4270)	&500* (4270)
Lithium			
Flow, gpm	177 (177.0)	180 (179.5)	183 (182.3)
Pressure at Pump Sump, psig	10	10	10
Temperature at Economizer Shell Inlet, F	940 (938)	1190 1175)	1460 (1450)
Pump Speed, rpm	4750* (4300)	4850* (4300)	4850* (4300)
Power, Mw	1* (0.93)	2* (1.99)	3* (3.11)

Values in the parentheses are the actual values, measured or computed for the conditions set.

* Approximate values only; other stated loop conditions were set.

Times accumulated were 1) primary system, 32 hours at 180 pgm and temperatures from 500F to 1460F, and 2) secondary system, 440 hours at 290 and 390 pgm and temperatures between 400F and 1000F.

An electrical short between the center terminal of the I^2R heater and the helium containment vessel caused a shutdown on March 23. The damaged area in the vicinity of the central terminal is shown in Fig 21. Both liquid metal loops were drained and the systems were cooled down for inspection. Examination revealed that the damage was confined to the seal area around the center terminal with no damage to the lithium system. A photograph of the center terminal taken after the heater tank was opened is shown in Fig 22. Samples of tantalum foil from the columbium heater pipe were analyzed for O_2 and no significant contamination was found. The inside of the heater containment vessel was examined for trace amounts of lithium and none was detected.

Methods of repair have been evaluated and design work is nearing completion on the selected scheme. No repair work will be required on the lithium system. Materials procurement and parts fabrication have been initiated. It is estimated that the rebuilt work will be completed by June 1. Re-establishment of the shroud gas system to the

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NON-NUCLEAR SYSTEMS TEST - CENTER LUG SHOWING DAMAGED AREA

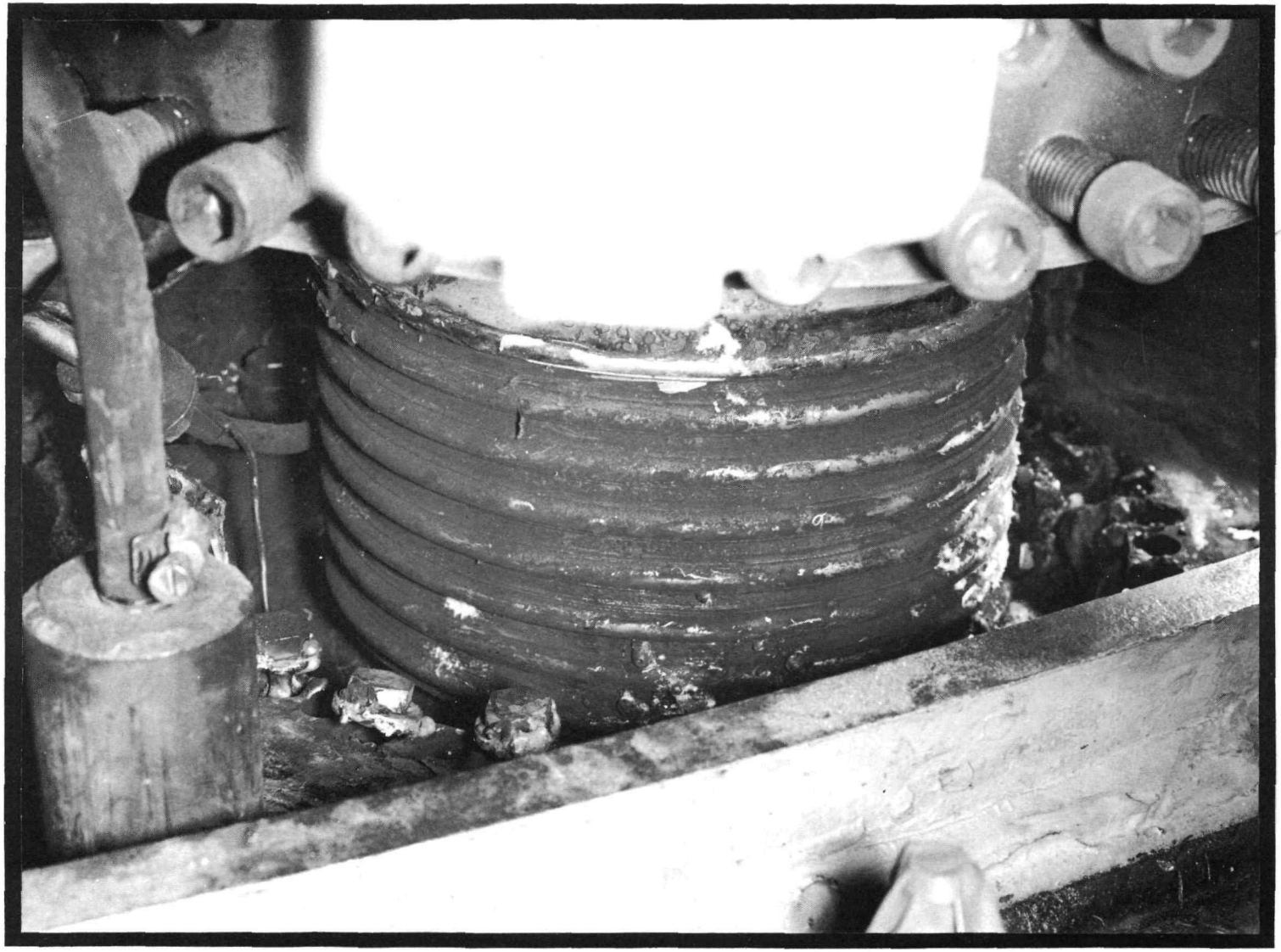

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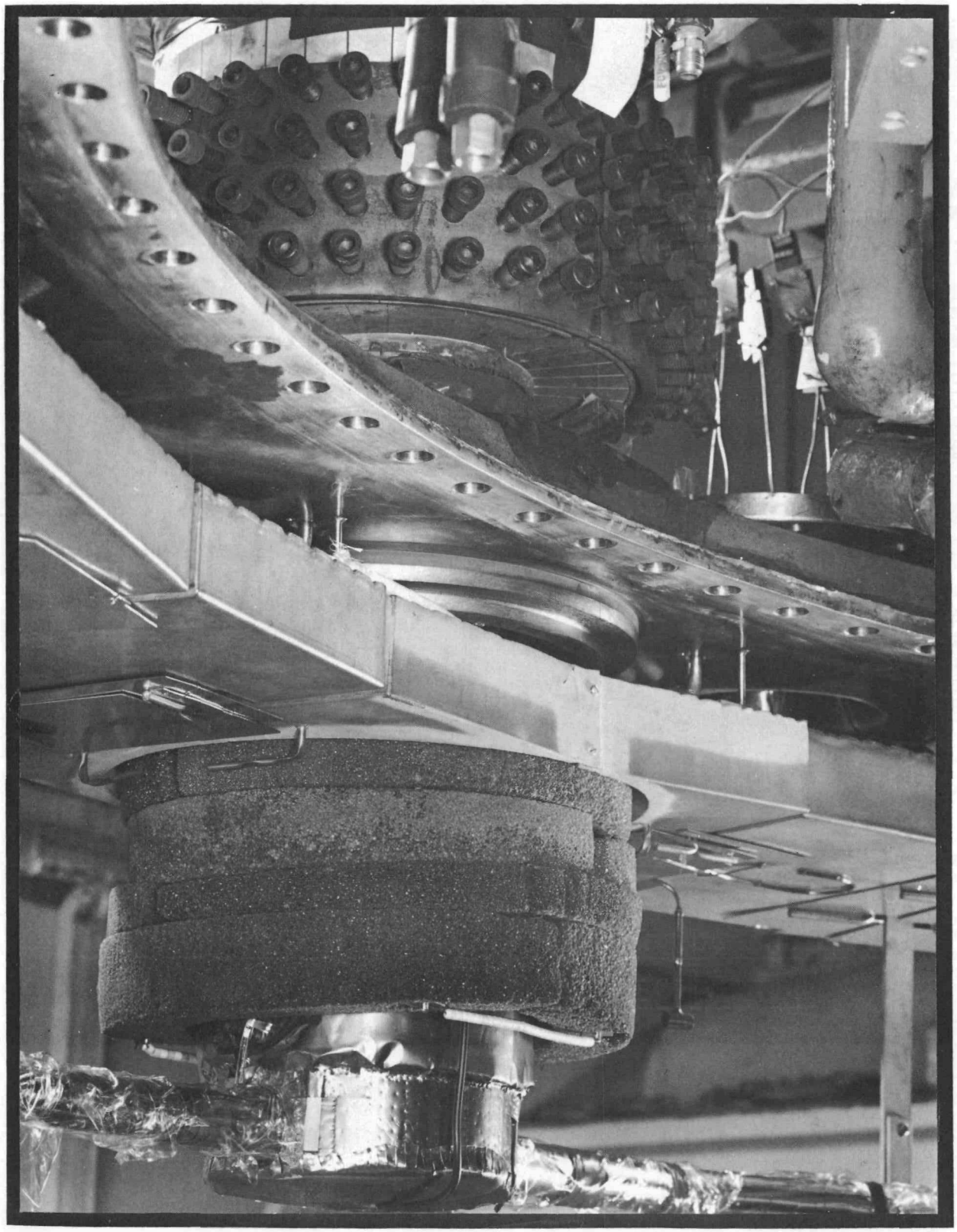
FIG 22

NON-NUCLEAR SYSTEM TEST - CENTER LUG SHOWING UNDAMAGED CONDITION OF PROTECTIVE FOIL



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startup configuration will be carried on concurrently with the terminal repair work. Leak check, shroud gas purification ans startup will follow completion of sonstruction.

To aid in future operation, a preliminary evaluation was completed on a newly developed method for the determination of calcium at concentrations of greater than 20 ppm in NaK. This method is intended for the monitoring and controlling of calcium present as a soluble getter in the NaK secondary circuit. Preliminary data, shown below, favor a procedure involving the titration to calcium with a 0.001 M solution of ethylenediaminetetraacetic acid (EDTA). An electrochemical end-point detector has successfully undergone initial investigation as a replacement for the Eriochrome Black T colorimetric indicator. Work will continue in an effort to improve the method.

Recovery of Calcium Added to NaK Samples

Parts per Million Added	Calcium, Total Found	Recovery of Total Calcium, Percent
350	300	86
480	520	108
760	610	80
1120	1080	96.5

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7. Instrumentation

A total of forty-seven 10,000 hour endurance tests of 1/8-inch diameter, Cb-1 Zr alloy clad thermocouples has now been started. Sixteen W-5 Re/W-26 Re thermocouples have operated with negligible drift after 1300 hours at 1950F in pure argon. The operating history of these is shown in Fig 23. Maximum and minimum readings, as well as mean error in degrees F, are plotted for each of two groups of eight thermocouples. The group of ten Chromel/Alumel thermocouples, with junctions insulated from the cladding, shows an average error of seven degrees after 1000 hours under the same conditions (Fig 24). Seven Mo/W-26 Re thermocouples showed no drift after 750 hours under these conditions but the initial error was 60 degrees when compared with the standard calibration (Fig 25). The open end of the cladding of all thermocouples in these tests is contained in an inert gas environment to prevent a calibration drift caused by oxidation of the wires.

As a result of experience obtained during endurance testing of this type of thermocouple, the specifications for all LCRE refractory metal clad thermocouples and level probes have been revised to require the maintenance of a high purity inert gas environment during fabrication. The effectiveness of this procedure in maintaining high quality insulation is shown by the fact that wire-to-sheath resistance in excess of a million megohms has been achieved.

Test results of the W-5 Re W-26 inpile thermocouple tests show drift which is greater than that to be expected from calculated transmutation. Capsules PW 25-30 and PW 25-31 have been returned to CANEL after exposure to a neutron flux of approximately 10^{20} nvt thermal in the Materials Testing Reactor and are currently being examined. In view of the fact that the out-of-pile control test showed a similar drift, it is felt that the anomaly may be caused by diffusion of air into the open end of the thermocouples. These units were fabricated before the need for sealing was realized.

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OPERATING HISTORY OF W-5 Re/W-26 Re THERMOCOUPLES

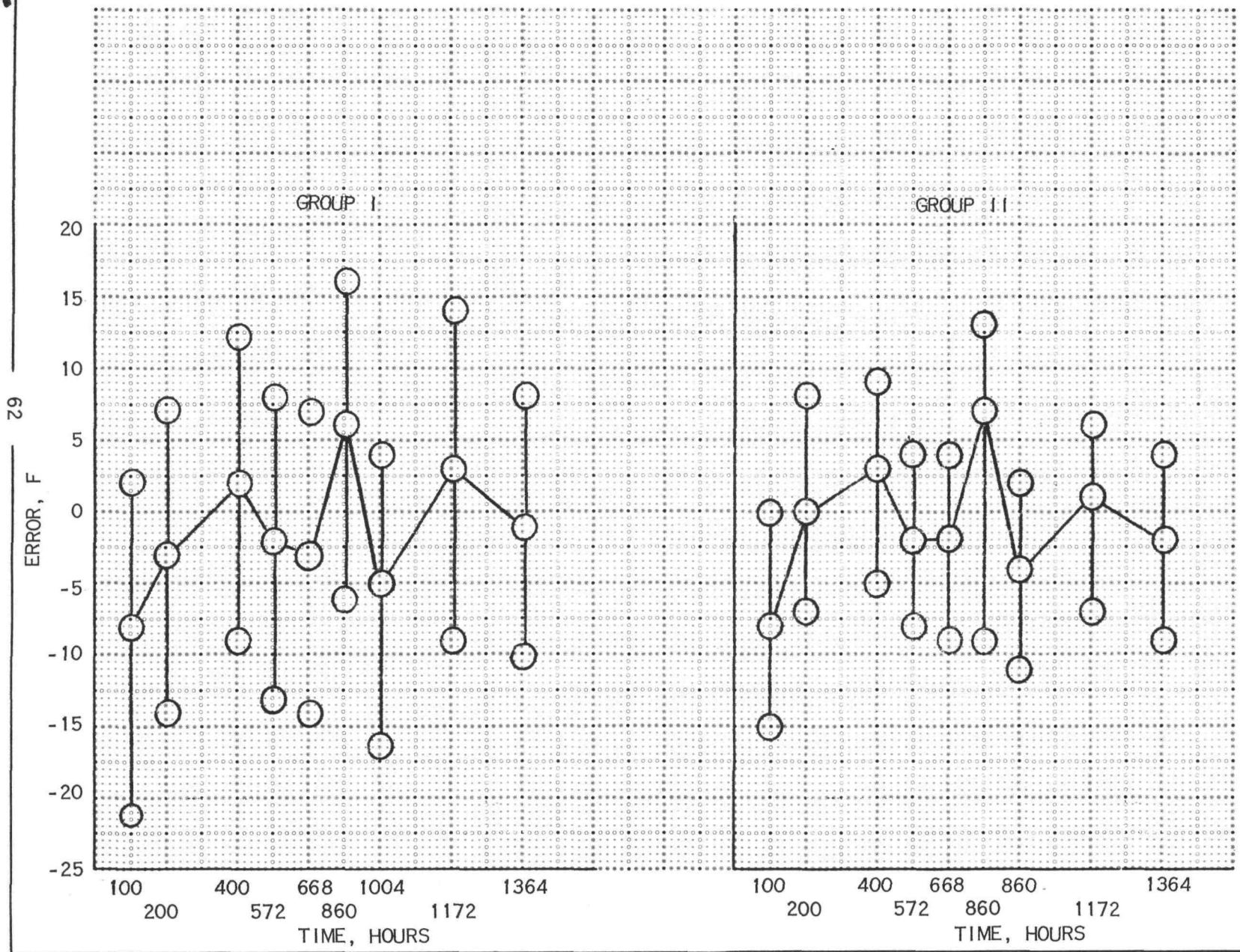

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FIG 23

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OPERATING HISTORY OF CHROMEL/ALUMEL THERMOCOUPLES

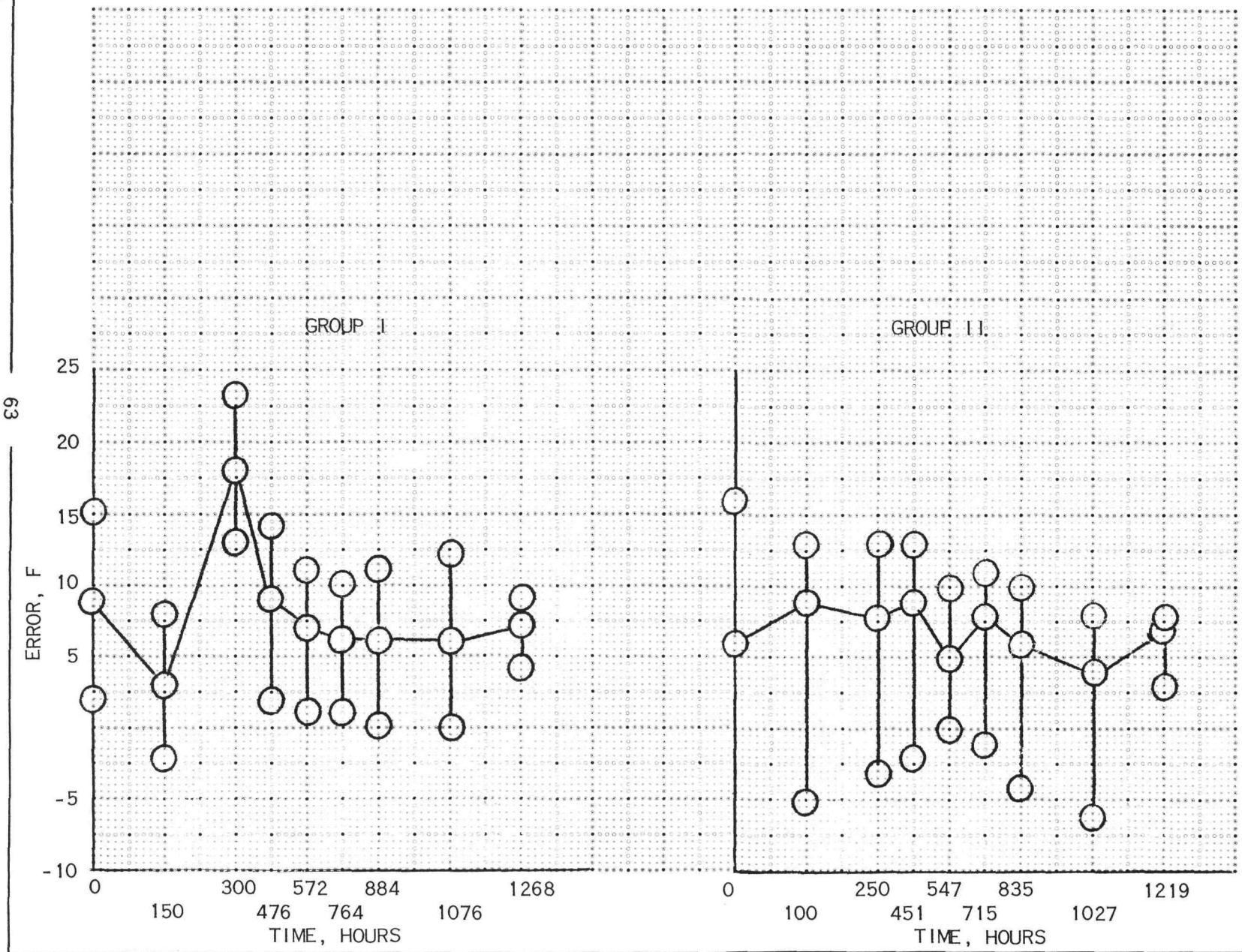

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FIG 25

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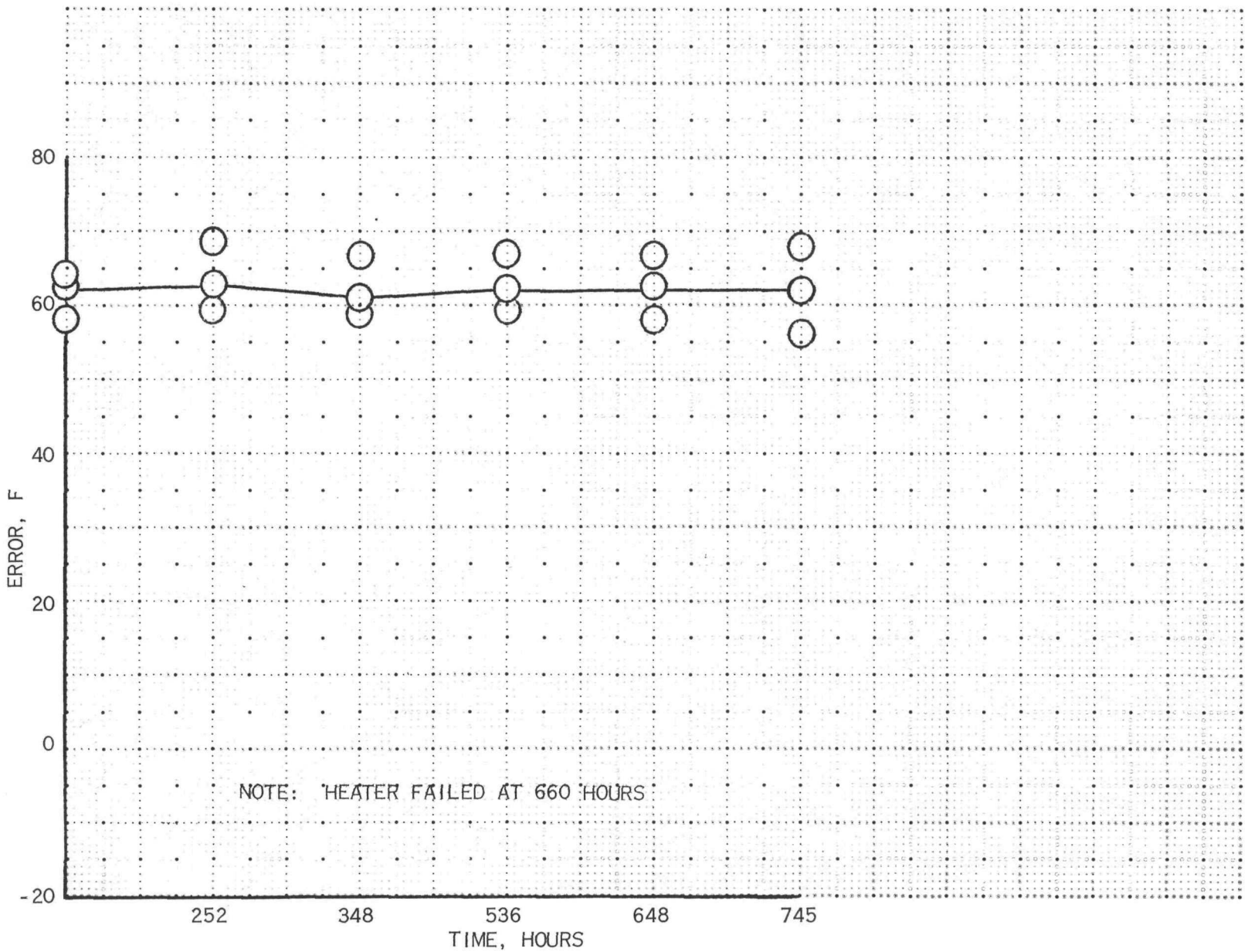
OPERATING HISTORY OF Mo/W-26 Re THERMOCOUPLES



Transmutation calculations based on the anticipated LCRE neutron flux environment showed a major contribution from neutrons in the epithermal range, due to a tungsten resonance at 18 ev. The total transmutation for the LCRE flux conditions for 10,000 hours is calculated to be as follows:

	Before Operation			After Operation		
	W	Re	Os	W	Re	Os
W-5 Re	95.00	5.00	0.00	94.40	5.56	0.03
W-26 Re	74.00	26.00	0.00	73.53	26.30	0.17

The percentage change in composition indicated is small but its effect on calibration is unknown. Since the wires are made by powder metallurgy techniques, it appears practical to fabricate and test wires of the anticipated post irradiation composition. The vendor will be consulted to confirm the feasibility of this approach.

The test stand for liquid metal testing of lithium system level probes is completed and five test assemblies, each containing two Cb-1 Zr alloy probes, are now operating satisfactorily after 800 hours at 1000F. Ten probes each of titanium and Cb-20 Ti alloy are currently being fabricated for testing under similar conditions.

Six of the 12 factory-calibrated, liquid metal pump speed transducers have been evaluated by the manufacturer and have been found to be undamaged after receiving a gamma exposure equivalent to that expected during LCRE operation. The remaining six are apparently undamaged after one-half of this dose and will remain in the MTR gamma grid for the completion of exposure.

Installation of four LCRE ion chambers and four fission counters in the Battelle Research Test Reactor was completed. The test configuration is shown in Fig 26. All units are operating satisfactorily at 400F with neutron flux levels ranging up to 10^{10} neutrons/cm^2/sec. and operating times of up to 1250 hours. Additional endurance testing of radiation sensors and high temperature pulse preamplifiers has been scheduled to improve statistical confidence. Rig designs are nearly complete and test units have been ordered.

8. Materials Development

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Mechanical properties data in the CS-1830, Cb-1 Zr alloy series have been procured to date on 17 heats of forgings and extrusions, 11 heats of bar, plate, rod and pipe, and 15 heats of sheet and tubing. Evaluation of the data shows a progressive decrease in long time rupture strength as the number of cold-reduction passes and anneals is increased. Minimum 10,000 hour rupture strength values determined from tests on various fabricated forms of Cb-1 Zr alloy are as follows:

Cb-1 Zr Alloy, Annealed at 2200F

	Minimum 10,000 Hour Rupture Strength		
Temp, F	Forgings & Extrusions (49 Tests), psi	Bar, Plate Rod, Pipe (47 Tests), psi	Sheet and Tubing (83 Tests), psi
1600	11,000	8,200	7,800
1800	5,800	4,400	3,400
2000	3,000	2,300	1,400
2200	1,500	1,300	700

FIG 26

TEST INSTALLATION OF GENERAL ELECTRIC AND REUTER STOKES ION CHAMBERS AND FISSION COUNTERS IN THE BATTELLE RESEARCH REACTOR

Forgings and extrusions represent material with hot work only. Bar, plate, rod and pipe are converted from forged stock with moderate amounts of cold work. Sheet and tubing are converted by cold reduction from forged stock. Evaluation of these strength data in conjunction with reactor stress analyses shows that the fuel pin cladding is unacceptable with respect to these latest strength data.

During the previous quarter (PWAC-624) a program was initiated to develop improved strength in Cb-1 Zr alloy fuel pin cladding. Several approaches were studied for treating finished tubing and modifications of melting and fabrication practices were initiated. Treatments to finished tube include solution heat treatment, nitriding, carburizing and a combination of nitriding and carburizing. These programs are directed toward determining:

1. The effect of solution heat treatment temperature and time on creep-rupture strength at 2200F and 2400F of several lots of finished tubing.

2. The effect of nitrogen on strength, with several conditions of nitriding time and temperature and solution treatment.

3. The effect of uniform carburizing on strength for several carburizing and solution heat treatment conditions.

4. The effect of several combinations of the above.

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Concurrently, the production of experimental quantities of tubing with a higher carbon content was initiated. This material will be melted to nominal carbon levels of 750 and 3000 ppm. Fabrication practices will be revised to minimize final cold reduction and to employ more effective annealing procedures. An indication of the extent of the improvement provided is given by comparison of the properties of the conventional low interstitial fuel element tubing with preliminary results on the specially treated tubing. On the conventional tubing, 24 tests at 2200F and 2400F have been completed. These indicate that the 10,000 hour, 2200F, minimum rupture strength is about 700 psi. This is about one-half of the strength of the same alloy in the forged condition. On the basis of the preliminary data so far obtained on the specially treated tubing, a carburizing treatment in combination with the solution heat treatment at 3100F will about double the 10,000 hour rupture strength at 2200F. This was accomplished with only a modest increase in the carbon (approximately 550 ppm). With the nitriding treatment, a strength increase of approximately 30 percent was obtained with approximately 800 ppm of nitrogen combined with a 3100F solution heat treatment. When the solution heat treat step was omitted, the indicated strength increase was about 20 percent,still for the 10,000 hour, 2200F rupture stress. The application of the solution heat treat to the as-received low interstitial tubing was considerably less effective, the increase in strength being from 10 to 15 percent. Further attention is being directed toward optimization of the solution heat treatment temperature and time.

Detailed data are presented in the following table:

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Fuel Pin Cladding Burst Test in Lithium
(All Tests Have Ruptured Unless Otherwise Noted)

Test Temp, F	Heat Code	Max. Stress, KIPS	Accumulated Time in Hours	Remarks
2400	PJHH	1.5	155	As rec'd, swaged 5%
		2.4	22	As rec'd, swaged 5%
		2.4	50	As rec'd, H. T. 3100F, swaged 5%
		2.4	18	As rec'd
		2.4	391	As rec'd, carburized H.T. 3100F, swaged 5%
		2.4	125	As rec'd, nitrided, H. T. 3100F, swaged 5%
2200	PIHH	4.62	6	As rec'd, swaged 5%
2400		1.0	650	As rec'd, swaged 5%
		1.0	754	As rec'd, H.T. 3050F, swaged 5%
		2.0	715	As rec'd, H.T., 3100F, swaged 5%
2200	PIXY	2.0	618	As rec'd, swaged 5%
		4.62	12	As rec'd, swaged 5%
2400		1.0	778	As rec'd, swaged 5%
2200	PJVI	2.4	312	As rec'd, H.T., 3100F, swaged 5%
		2.4	896*	As rec'd, carburized, H.T. 3100F swaged 5%
2400		1.5	336	As rec'd, swaged 5%
		2.4	44	As rec'd, swaged 5%
2200	PJVH	2.4	384	As rec'd, H.T., 3100F, swaged 5%
		2.4	896*	As rec'd, carburized, H.T. 3100F swaged 5%
2400		1.5	438	As rec'd, swaged 5%
		2.4	69	As rec'd, swaged 5%
2400	PILY	1.5	102	As rec'd, swaged 5%
		2.4	17	As rec'd, swaged 5%
		2.4	29	As rec'd, H.T. 3100F, swaged 5%
		2.4	270	As rec'd, carburized, H.T. 3100F swaged 5%
		2.4	42**	As rec'd, nitrided, H.T. 3100F swaged 5%

Note: All specimens have been annealed at 2200F for 2 hours just prior to testing.
* Still on test
**Discontinued Test

Testing of Cb-1 Zr alloy specimens prepared from plate was continued. Eight uniaxial creep-rupture specimens completed test in lithium during the quarter including four which exceeded 10,000 hours and one which was discontinued at 9905 hours because of a rig malfunction. Current status of uniaxial tests is as follows:

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Creep Tests in Vacuum

Test Temp, F	Heat Code	Test Stress, KIPS	Accumulated Time in Hours	Total % Extension	Remarks
1800	PJDC	12.5	25	22.6	Ruptured
2000	PJDC	7.5	51*	44.2	Discontinued
	PJDC	5.0	522*	17.7	Discontinued
2200	PJDC	2.5	793*	6.5	Discontinued
2400	PJDC	2.0	565	55.0	Ruptured
	PJDC	1.5	957*	4.1	Discontinued
	PJDC	1.0	312*	1.7	Discontinued

*Note: These tests were discontinued either because of a vacuum system leak or because the bellows extension limit was reached.

The following biaxial tube burst tests in lithium were continued.

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Biaxial Tube Burst Tests in Lithium

Test Temp, F	Heat Code	Max. Stress, KIPS	Total Time in Hours	Remarks
1600	PGSG(2)	6.0	2,345	On test
1800	PJYT(1)	3.5	1,506	On test
2000	PGSI(1)	5.0	12,920	On test
	PGSI(2)	4.0	2,409	On test
	PGSI(2)	3.0	3,330	On test
	PGSI(2)	2.0	3,379	On test

Note: (1) Tube to plate weld
(2) Girth weld

Long time rupture tests on specimens of type 316 stainless steel were continued at temperatures from 1200F to 1650F. Two specimens, one at 1500F and 3,000 psi and one at 1650F and 1,500 psi have exceeded 13,000 hours in continuing tests.

The system for measurement of alkali metal vapor pressures (Fig 27) has been installed. During preliminary tests, the equipment has been successfully operated at pressures lower than 1×10^{-5} Torr and at temperatures up to 2400F. Additional heat shields and water cooling are being installed to permit testing at higher temperatures. The oscillating cylinder viscometer has been reassembled and a Metrisite rotary variable transformer has been installed to measure angular position of the cylinder. Calibration tests on water have been started.

Compatibility testing of reflector, control and insulating materials is continuing as outlined below:

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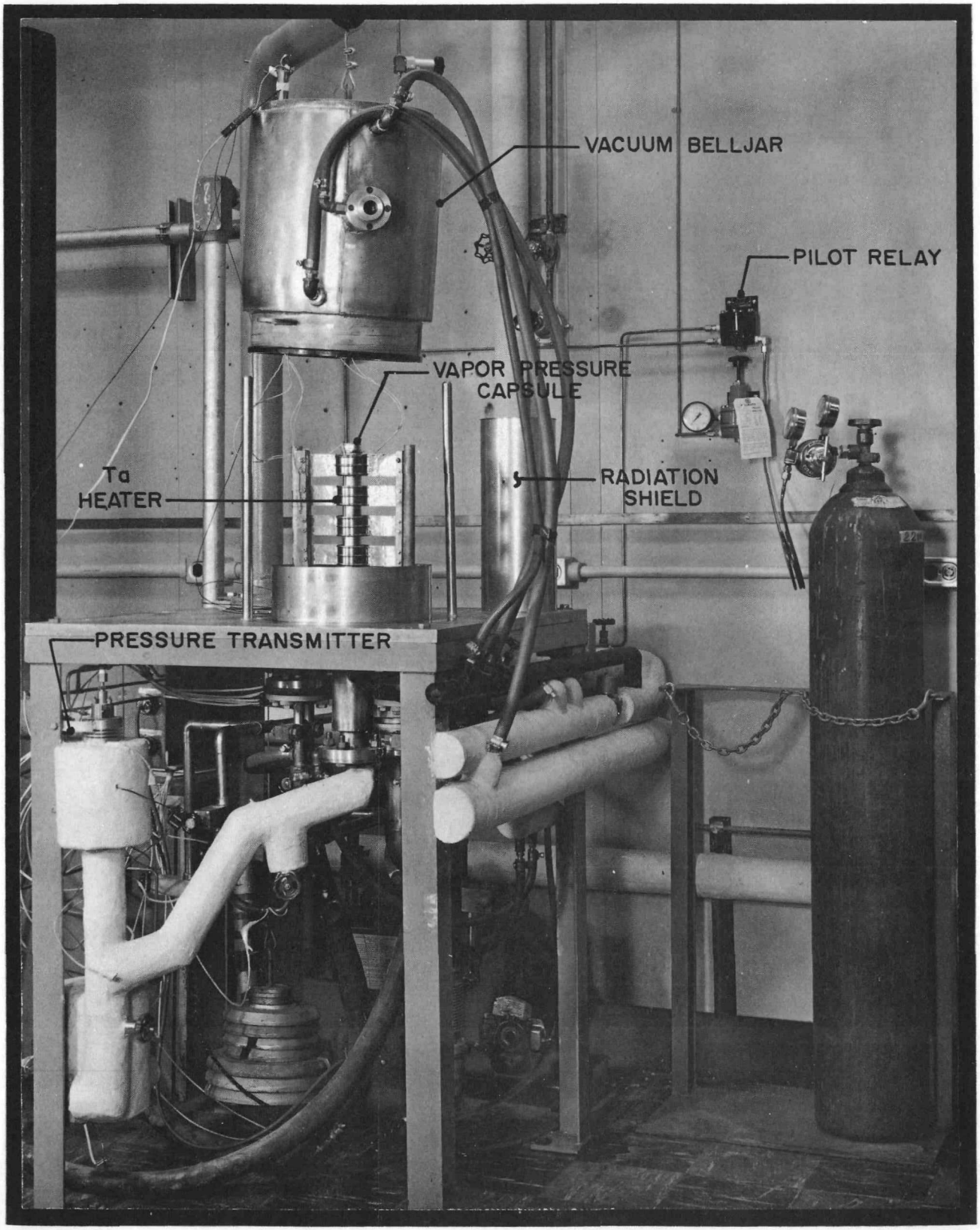
FIG 27

LIQUID METAL VAPOR PRESSURE RIG



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Materials	Purpose of Test	Type of Test	Temp, F	Time, hr.	Scheduled Duration, hr.
B_4C/NaK/Type 316 SS	Boron and carbon transfer study	Static	1000	9,036	10,000
	Performance of porous metal filler	Static	1000	2,976	5,000
Be/NaK/Type 316 SS	Diffusion study	Tilting	1000	7,788	10,000
		Tilting	800	1,268	3,000
Be(Nitrided)/NaK/ Type 316 SS	Diffusion study	Tilting	800	1,268	3,000 5,000 10,000

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Chemical analysis of specimens from the type 316 stainless steel tubing of the 3000 and 6000 hour NaK/B_4C tests previously reported (PWAC-624) has been completed. Boron was found in concentrations of up to 0.01 percent in the 0.0010 to 0.0016 inch band which was observed metallographically. In addition, the carbon content increased from a pretest value of 0.07 percent to a posttest value of 0.13 percent in the 0.010 inch increment of the tubing wall. Chemical analyses are in progress to determine the carbon gradient through the remainder of the wall. Neither the boron nor the carbon concentrations observed in these tests are considered detrimental to the performance of the stainless steel.

Tests completed during this period have demonstrated that B_4C fragments formed during 1000 and 5000 hour exposure to 1000F NaK were completely retained by 160 micron stainless steel filter. It seems likely, therefore, that the submicron size particles previously reported in the residue on a 20 micron stainless steel filter (PWAC-623) were produced during decontamination of the residue rather than by alkali metal attack.

A Cb-1 Zr alloy capsule, containing columbium and Cb-1 Zr alloy tabs contacting Al_2O_3 in a static, purified argon environment, has completed a scheduled 2000 hour test at 2200F. Posttest evaluation of the specimens is in progress to determine if any interactions occurred.

Testing of the beryllium/eutectic NaK/type 316 stainless steel erosion and corrosion loops continues. The first unit, NSBA-1, has completed 4979 hours of a 5000 hour test at 800F. A second loop, NSBA-2, with new specimens and holders, was put on test at 800F. A total of 948 test hours of a scheduled 2000 hours has been computed.

Metallographic examination of the four beryllium rod type specimens, which were exposed in the first NSBA-2 loop test, revealed that a dark region resembling a diffusion band was similar to that observed in capsule tested specimens previously reported (PWAC-624). Phase extraction studies are currently in progress to identify the constituents. There was no evidence of excessive spalling or other severe effects on the beryllium.

It was previously reported (PWAC-623) that wrought Cb-1 Zr alloy material, initially containing 300 to 400 ppm oxygen, could tolerate an additional 1500 ppm oxygen added at 2000F without being attacked by subsequent exposure to 2000F lithium. Additional tests of wrought specimens, containing 1 weight percent zirconium inoculated at 2000F

to contain up to 3100 ppm total oxygen have been completed. These tests show that at this concentration the alloy remains metallographically unaffected by lithium exposure. From the results of testing reported in PWAC-622 it would be expected that this material would exhibit a tolerance for approximately 3600 ppm oxygen without attack by lithium. In verification of this limit, two additional specimens, containing 4000 ppm oxygen, were attacked intergranularly by 2000F lithium. Thus, these tests effectively bracket the maximum tolerance value.

A tantalum foil-wrapped Cb-1 Zr alloy loop, LCCBK-8, successfully completed a scheduled 1000 hour test and posttest metallographic examinations and chemical analysis has revealed many areas of localized oxygen contamination associated either with tantalum foil attachment tack welds or with spiral discolorations on the surface of the tubing associated with laps in the foil wrapping. The contaminated areas were found to start at the center of the heater and extend into the cooler. No lithium corrosion was found in any of the sections examined, including areas where high oxygen concentrations were found which extended nearly across tubing wall.

LCCBK-9, the third in the series of tantalum foiled Cb-1 Zr alloy loops, was built up, installed and is currently on test at conditions identical to those in LCCBK-6, 7 and 8. A total of 81 hours of test operation has been accumulated to date. A 5000 hour test period is scheduled for this loop.

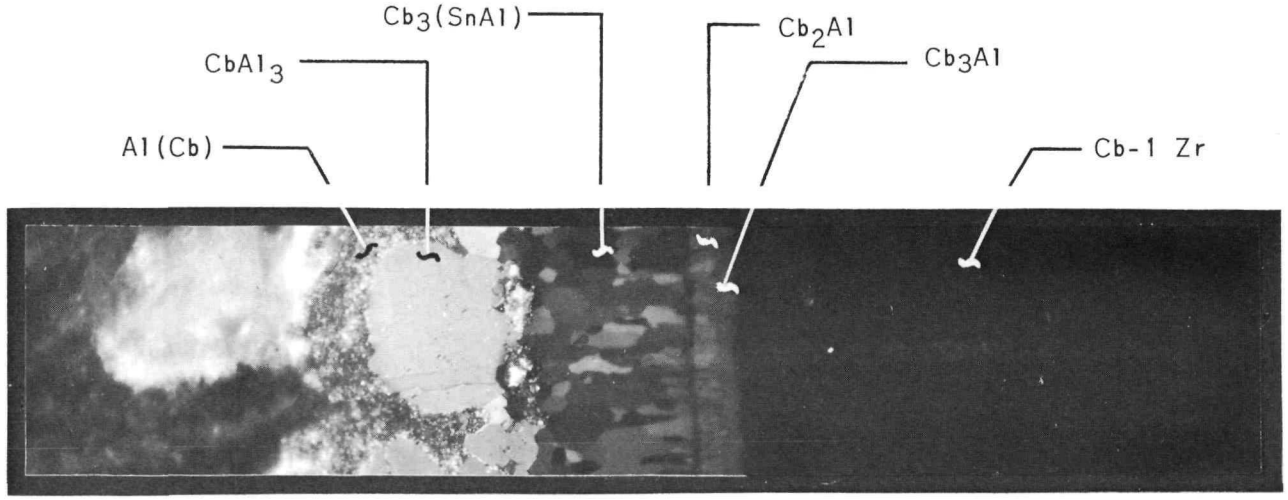
The first group in a series of tube-burst specimens of columbium alloy has been exposed in the LCCDA-1 slave system for 500 hours at a maximum lithium temperature of 2000F. The specimens have been removed, decontaminated and are currently being prepared for evaluation. A second group of specimens is being prepared for insertion into the slave system and test operation at 2000F will be started early in the next quarter. Some slight leakage of lithium vapor was experienced during the first test. This occurred at a mechanical joint connecting the columbium and stainless steel halves of the specimen tank. Since the test is being conducted in an inert atmosphere chamber, no adverse effects on the specimens are anticipated and the scheduled program will be continued.

Tests have been continued to evaluate the oxidation protection behavior and to measure the extent of interaction of SNAL-1 coatings on Cb-1 Zr alloy specimens in 2000F argon as described in PWAC-624. A series of specimens with two mil and eight mil coatings has completed 4700 hours of a scheduled 10,000 hour test with no sign of spalling. The test was terminated because of air contamination of the argon occurred which compromised the recession study. Additional metallographic and electron microbeam probe analyses were obtained on previously terminated specimens with exposures of up to 2500 hours in 2000F argon. Interdiffusion and composition gradients through a 4.5 mil wide band are illustrated in Fig 28.

In a study to evaluate methods of minimizing contamination of Cb-1 Zr alloy specimens during testing in inert atmosphere, two 4-inch Cb-1 Zr alloy pressure vessels containing lithium at 2000F continue to operate with no sign of lithium leakage. A SNAL-1 coated vessel, operating in static helium, has accumulated 6936 hours. A tantalum-wrapped vessel, operating with no coating in helium containing 100 ppm air and flow at the rate of 50 cc per minute, has been operated 6121 hours.

The posttest examination of the lithium-columbium alloy corrosion loop, LCCBK-7, which was coated with a tin-aluminum slurry (SNAL-1), has been completed. Preliminary evaluation of the data revealed good system compatibility with no evidence of lithium attack in the sections examined. There was also no evidence of separation of the coat-

ALUMINUM COATED Cb-1 Zr SPECIMEN



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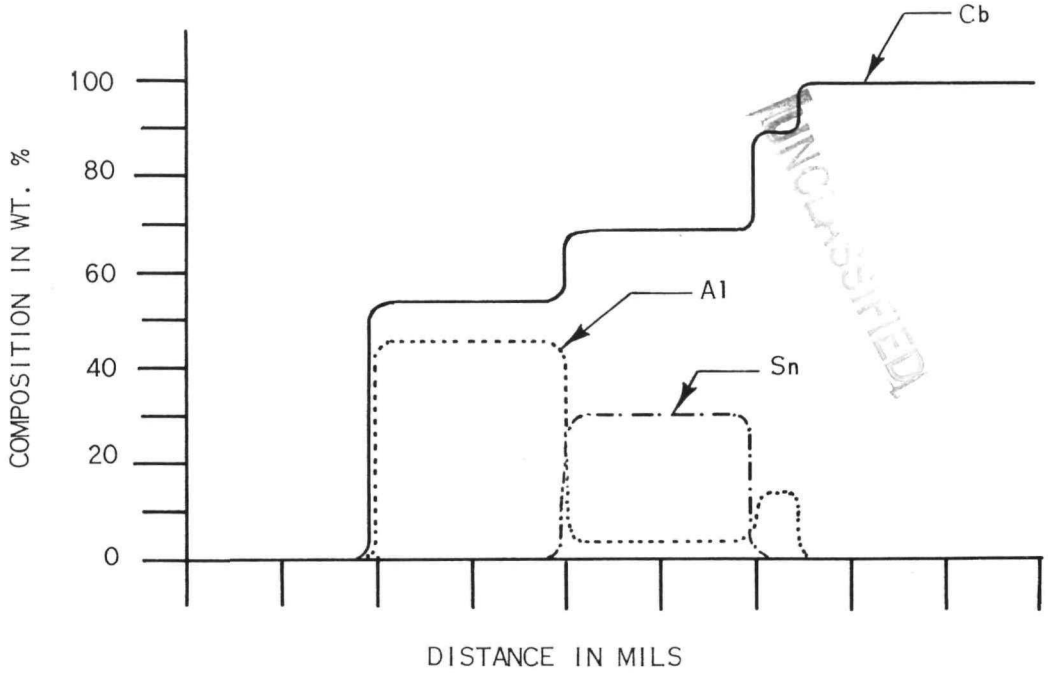

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MICROPROBE TRACE SHOWING Cb, Al AND Sn CONCENTRATION GRADIENTS IN SnAl-1 COATED SPECIMENS FIRED 2500 HOURS AT 2000F IN ARGON

ing from the surface of the Cb-1 Zr alloy. However, two localized areas of oxide contamination were found which were associated with cracks found in the coating. These cracks were perpendicular to the surface being protected. A detailed evaluation of the posttest loop examination data is continuing. The construction of the second SNAL-1 coated loop, LCCBK-6,was completed and the loop is currently being operated at test conditions. To date a total of 1112 hours has been accumulated at a maximum lithium temperature of 2000F with a ΔT of 1000F. This test is scheduled for 5000 hours.

An investigation has been initiated to develop a spectrographic method for the determination of trace quantities of rare earths in Cb-1 Zr alloy. This is required to determine that incoming Cb-1 Zr alloy meets the CS-1830 specification restriction of less than 100 parts per million total rare earths. Because of the complex spectrum of the Cb-1 Zr alloy and the low concentrations of the rare earths expected in the alloy, it has been found necessary to separate and concentrate the rare earths. This has been accomplished by taking advantage of the fact that the chlorides of columbium and zirconium are readily distilled in a chlorine gas atmosphere at about 500F while the chlorides of the rare earths exhibit negligible vapor pressure below 1200F. The spectrogram of the concentrated rare earth residue is examined to determine the presence of gadolinium and samarium as shown by the 3422.5 and 3568.3 angstrom lines. Gadolinium and samarium were chosen as representative of the rare earths since they are present in a natural abundance of about five percent each and their concentrations may be used to calculate the approximate total rare earth concentration. Preliminary recovery experiments have shown essentially complete recovery of two micrograms each of gadolinium and samarium and five micrograms of yttrium added to one gram of Cb-1 Zr alloy. Work will be continued to establish the spectral sensitivities of these elements and the statistical reliability of the method.

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D. LCRE FABRICATION

1. Reactor Components

Because of low strength found in Cb-1 Zr alloy tubing intended for use as LCRE fuel cladding, additional tubing with a higher carbon content is now being procured. The additional carbon coupled with appropriate heat treatment is expected to strengthen the tubing to acceptable levels, as reported in Section IIC of this report. Feasibility and Standard Practice reports for the LCRE core fabrication were drafted.

A core pressure vessel upper head and cylinder have been welded together in preparation for final machining. Another cylinder and upper head have been rough machined and are ready for welding. Welding of the latter is to be done in the welding box which is to be used in Idaho for the final closure weld on the LCRE core vessel. This operation will serve as the final qualification for this welding chamber.

Machining of the upper closure for the reflector pressure vessel has been completed. The development program for flow-turning the inner and outer reflector cases has been completed satisfactorily, but two attempts to flow-turn inner reflector cases were unsuccessful. The forgings for the outer reflector cases are being machined in preparation for flow-turning. An alternate design was completed for a welded and machined reflector pressure vessel assembly.

Forgings for control drum housings and all B_4C control drum poison segments have been received.

On the basis of consistently successful component tests, the following major control drive elements have been selected for incorporation in the LCRE:

Component	Material
Flex tube	AMS-5616
Harmonic gear pair	AMS-5616
Drum journal bearing	90 w/o WC, 8% molybdenum, 2% CbC
Drum thrust bearing	M-50 rollers and races, Rollube 3 cages
Harmonic bearing	M-2 balls and races, 17-4 PH cage

The reactor startup source drive and pile oscillator transmission were released for construction. Orders were placed for the LCRE shim actuating motors, scram actuating motors, limit switches and synchro transmitters. Manufacturing has been started on the neutron sensor assemblies, sensor positioning mechanisms and harmonic adapters. An extensive investigation was started to find vendors capable of assembling the LCRE control drive gear box assemblies under clean and dry conditions.

A construction contract was placed with a vendor for the assembly of LCRE control and safety systems chassis. The first two construction tasks have been summarized and work is scheduled to start early next quarter after receipt and inspection of the necessary parts by the contractor. Approximately 80 percent of the components and hardware required for the entire chassis construction effort have now been ordered.

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About 95 percent of the instruments for the prime viewing section of the "P" area control room are now either on hand or in various stages of procurement.

Preliminary specification, FPS-R-410, LCRE Shield, and preliminary construction drawings, were circulated to prospective bidders in preparation for a bidder's conference to be held at CANEL during the next quarter. It is the intent that a single shield subcontractor be selected who will fabricate, conduct a trial assembly of, and supervise final assembly of the shield in Idaho. At present, the design of the shield is sufficiently complete to define the scope of this fabrication and construction work. Complete detail design definition, principally in the areas of sealing gas coolant passages, the incorporation of control drive wiring ducts and the relocation of the nuclear sensor, will not be complete until August 1963. The following target dates have been established to be consistent with the LCRE installation schedule.

Final specifications by CANEL	August 1, 1963
Complete shield fabrication and preliminary assembly by subcontractor	November 1, 1964
Start shield installation in Idaho	January 1, 1965

2. Primary and Reflector System Components

Major construction work on the large, inert-gas welding chamber in the CANEL shop was completed during this period. This chamber will be utilized for assembly of all large Cb-1 Zr alloy components and assemblies for the LCRE.

Work continued on the fabrication of components for the primary system pumps and sump. Two Cb-1 Zr alloy double pump-sumps were formed at the East Hartford plant. The first of these sumps has been machined and assembly work started.

The remaining raw material required for the primary and regenerative heat exchangers was received. Fabrication of all of the components for these heat exchangers was completed and preparations for assembly work were started.

All work on the extrusion and drawing of Cb-1 Zr alloy primary system piping has proceeded satisfactorily. The large furnace at Harvey Aluminum Company for annealing the finished pipe was completed and put into operation.

The 3-1/4-inch diameter, co-extruded Cb-1 Zr-to-type 316 stainless steel joints have been received and inspected as were the primary system jacket expansion bellows. Orders were placed for the primary coolant system pump inlet duct forgings, pipe cross and tee forgings, and co-extruded, bimetallic forgings for the bimetallic thermocouples and liquid level transmitters. Fabrication of many of the detail parts for the primary gas jacket system was completed.

Fabrication of detail parts for the reflector cooling system pumps and sump continued during the quarter. Assembly work on the type 316 stainless steel double pump-sump was started. All of the raw material for the NaK-to-NaK reflector heat exchangers was received. Forming of the reflector system piping has been delayed pending the completion of a review of the as-bend pipe for the LCRE mock-up assembly. This review will include the inspection of coordinate points, the pipe wall thinning at the radaii, and the investigation of procedures for minimizing contamination of the pipe during bending.

A survey of the facilities of several vendors was made to establish interests and capabilities for subcontracting the fabrication and assembly of the primary and reflector coolant system fill and drain tank assemblies. Discussions have been centered around the capability of fabricating the assemblies to CANEL cleanliness specifications required. Several vendors expressed interest and appeared to be capable of complying with specifications.

The electric motors to drive the primary and reflector coolant system liquid metal pumps are special devices. They require special high temperature electric insulation, radiation resistant lubrication, and very careful detail design and fabrication of the rotating assembly since they must operate for 15,000 hours without maintenance. Because of specific experience with electric motors which were run inpile during the ANP program under conditions which are applicable to the present design problem, CANEL has specified the use of Dow Corning Sylkyd 1364 film insulated magnet wire and NRRG-159 grease. Proposals from three companies were solicited for the construction of these motors. All proposals were rejected because insufficient bearing, bearing housing and bearing lubrication data were presented. In addition it was learned that Dow Corning Sylkyd 1364 is no longer available and a substitute wire enamel must be found. Several studies will be subcontracted to develop the design of these motors. The selection of a final design will be made in the next quarter and an endurance-screening test program will be prepared to assure that the motors will receive a maximum amount of reliability testing prior to start of the LCRE.

3. Secondary System Components

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A quotation was received for the fabrication and assembly of three NaK-to-air radiators. This was reviewed and an addendum to the specification prepared covering all points discussed during the negotiations with the vendor. A meeting with the vendor's representatives has been scheduled to conduct the final review of the specification and the addendum.

Proposals were received from four vendors for the air system butterfly valves. These proposals were reviewed and a contract awarded. Quotations were solicited for the air system blower and drive packages. Two were received, reviewed and found unsatisfactory. Quotations on these packages will be solicited again.

Procurement of materials for all parts of the secondary pump-sumps was initiated except for those which may require modification as a result of further investigation of the NaK vapor-gas sweep plugging problems.

The secondary piping system materials have been ordered. To date the two-inch and three and 1/2-inch outside diameter, AMS-5573 (type 316 stainless steel) seamless tubing has been received. The secondary system isolation valves and fill and drain valves have been released for fabrication.

4. Auxiliary Systems

A review of work involved in fabrication, assembly and installation of the auxiliary and control systems in Idaho showed that design, fabrication, procurement and installation manpower resources at CANEL must be supplemented to complete installation of

these systems on schedule. As shown in Fig 29, portions of these systems must start arriving in Idaho January 1, 1964. Accordingly, planning was started to employ a single subcontractor for the bulk of this work.

Specification FPS-E-418, The Production of Installation Drawings, Fabrication, and Installation of the LCRE Control and Auxiliary System, outlines the work associated with auxiliary system construction to be performed to complete installation of the LCRE. This specification was completed and sent to five prospective bidders. Replies have been received from three and are being evaluated. The general plan of work to be performed by the contractor is shown in Fig 30. A review of these proposals will be completed and a firm plan of action will be prepared during the next quarter.

Proposals, which were solicited from three vendors for the design, fabrication, installation and test of a Cryogenic Trap System to remove inert fission products (mainly xenon and krypton) from the primary cover gas system, were evaluated during the past quarter. One of these was found to be technically unacceptable. The remaining two vendors were visited so that their facilities might be inspected and the proposals discussed in greater detail.

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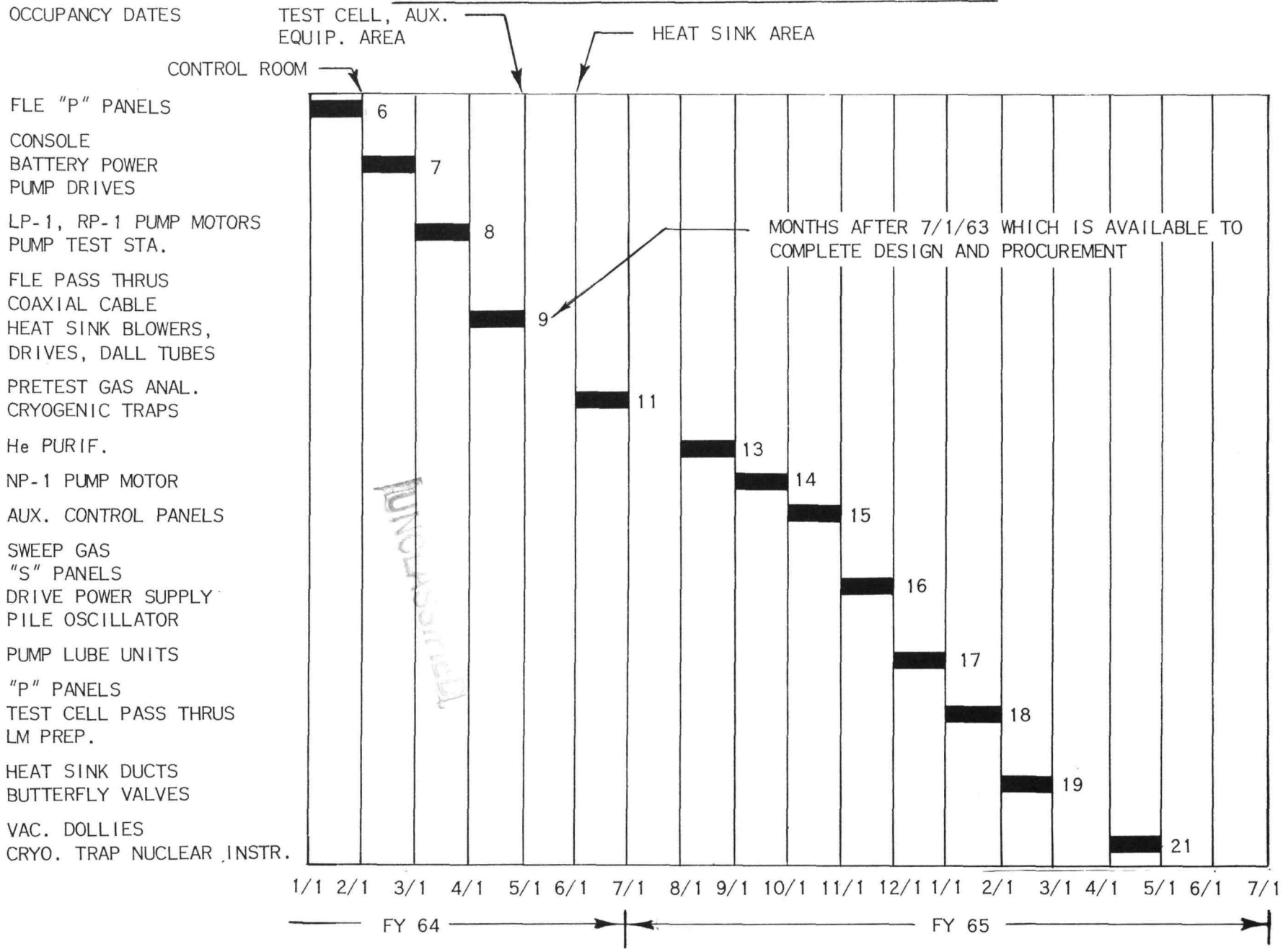
TARGET DATES FOR THE RECEIPT OF AUXILIARY AND CONTROL SYSTEMS EQUIPMENT AT THE IDAHO TEST FACILITY



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GENERAL PLAN FOR PROCUREMENT FABRICATION AND INSTALLATION OF LCRE AUXILIARY AND CONTROL SYSTEMS

	Auxiliary and Control System Work Package	Single Subcontractor		CANEL or Others	
		Installation (Ref. Fig 2)	Procure and Fabricate	Installation (Ref. Fig 2)	Procure and Fabricate
Instrumentation	1. Primary Circuit Instrumentation 2. Reflector Circuit Instrumentation 3. Secondary Circuit Instrumentation	Control and power wiring and plumbing from Test Cell and Heat Sink Rooms to P and S control areas. Gas control racks in S and Z areas. Nuclear sensors and gas control valves on Cryogenic Trap System including hookup to P and S areas. P area panels. Control Console. S area panels.	Notes 1, 2 Valve bellows backup gas control racks.	All Test Cell and Heat Sink Room LM components and assemblies procured and fabricated by CANEL and others.	Reactor, LM components and assemblies: pumps heat exchangers, radiators, vapor traps, piping, jacket, oxide control stations, valves, and pump motors, flow meters. P area instrumentation.
	4. Startup Source Drive -1 5. Chamber Check Source Drive - 1 6. Chamber Drives - 6			All sources and drives in Test Cell.	Reactor Startup source and drive. P area instrumentation. Chamber check source and drive. P area instrumentation. Chamber drives.
	7. Cryogenic Traps - Nuclear Instrumentation				
	8. Reactor Control 9. Reactor Safety		Notes 1, 2 and 21 bays P area panels		Control Console. 31 bays of P panels. All nuclear sensor assemblies. 4 bays in P area of area monitoring, test cell cooling, and shield cooling system controls by M. W. Kellogg. Reactor cooling shield controls in P and S area.
	10. Reactor Flux Monitoring 11. Reactor Gamma Monitoring			Nuclear sensor assemblies in Test Cell.	
	12. Shield Cooling - CANEL Instrumentation			Instr. in Shield in Test Cell.	
	13. S Area Controls		Note 2 and 50 bays of S area panels		See No. 29, Cryogenic Traps.
Electrical Heating	14. Primary Fill and Drain Instrumentation and Electrical Heating 15. Primary Coolant Addition Instrumentation and Electrical Heating 16. Reflector Fill and Drain Instrumentation and Electrical Heating 17. Secondary Fill and Drain Instrumentation and Electrical Heating 18. Secondary Oxide Control Instrumentation and Electrical Heating		Notes 1, 2 and valve bellows gas and cover gas control racks	All Test Cell and Heat Sink Room LM components and assemblies procured and fabricated by CANEL and others.	LM components and assemblies: fill and drain tanks, vapor traps, pipe, valves, and oxide control stations.
	19. Primary Electric Heating 20. Secondary Electric Heating	Wiring from Test Cell and Heat Sink Rooms to P and S area control and power distribution panels. Electric power distribution panels and powerstats in S area.	Notes 1, 2	Test Cell and Heat Sink Room LM mounted items.	Heaters, thermocouples, and insulation mounted on all LM components and assemblies
	21. Gas Line Electric Heating 22. S Area Electrical Equipment		Notes 1, 2 and heaters, thermocouples, and insulation, and power panels, powerstats, switch gears, recorders		

Notes: 1. All remaining instrumentation, electrical and pneumatic parts, wiring and plumbing.
2. All installation drawings for auxiliary and control systems such as wiring and plumbing diagrams, conduit and tray routing, area layouts, and mechanical and electrical equipment installation drawings except for the Cryogenic Trap System will be made by the single subcontractor.

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GENERAL PLAN FOR PROCUREMENT FABRICATION AND INSTALLATION OF LCRE AUXILIARY AND CONTROL SYSTEMS

CONTINUED

	Auxiliary and Control System Work Package	Single Subcontractor		CANEL or Others	
		Installation (Ref. Fig 2)	Procure and Fabricate	Installation (Ref. Fig 2)	Procure and Fabricate
Gas	23. Primary Pump Seal and Sweep Gas 24. Reflector Pump Seal and Sweep Gas 25. Secondary Pump Seal and Sweep Gas 26. Gear Box Gas 27. Jacket Gas 28. Helium Supply and Purification	Gas control racks in S and Z areas. He supply and purification system in G area. Bulk He supply. All interconnecting wiring and plumbing and hookup to P and S control areas	Notes 1, 2 and gas control racks. Complete helium supply and purification system		P area instrumentation.
Gas	29. Cryogenic Traps - Non-Nuclear Process	Hook up gas systems to Cryo Traps		Installation and acceptance test by Cryo Trap Subcontractor.	Design and fabrication including traps, liquid nitrogen supply, and 4 bays control panels in S area by Cryogenic Trap Subcontractor.
Mechanical-Electrical Equipment	30. Primary Pump Drive 31. Reflector Pump Drive 32. Secondary Pump Drive	Pump drives, switch gear and control panels in D area. Hook up to facility power and LM pumps. Control wiring to C, P, and S areas. One unit wired to U area.	Notes 1, 2		Pump drives (MG-MA sets) and electro-mechanical switch gear. P area instrumentation.
Mechanical-Electrical Equipment	33. Primary Pump Lube Oil 34. Reflector Pump Lube Oil 35. Secondary Pump Lube Oil	Pump lube units and control panels in L area. Hook up to facility power and lube lines to LM pumps. One unit in U area	Note 2 and complete lube oil system		
Mechanical-Electrical Equipment	36. Heat Sink	Blowers, blower drives, ducts, and Dall tube in Q area. Ducts and butterfly valves in Heat Sink Rooms. Hook up to facility power and control wiring to C, P, and S areas	Notes 1, 2 and Dall tubes and air ducts including supports		Large and small blowers and drives. Butterfly valves including motors. P area instrumentation.

Notes: 1. All remaining instrumentation, electrical and pneumatic parts, wiring, and plumbing.
2. All installation drawings for auxiliary and control systems such as wiring and plumbing diagrams, conduit and tray routing, area layouts, and mechanical and electrical equipment installation drawings except for the Cryogenic Trap System will be made by the single subcontractor.

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GENERAL PLAN FOR PROCUREMENT FABRICATION AND INSTALLATION OF LCRE AUXILIARY AND CONTROL SYSTEMS

CONTINUED

	Auxiliary and Control System Work Package	Single Subcontractor		CANEL or Others	
		Installation (Ref. Fig 2)	Procure and Fabricate	Installation (Ref. Fig 2)	Procure and Fabricate
Mechanical-Electrical Equipment	37. Pile Oscillator	Electrical power unit in F area. Control and power wiring to Test Cell and P area. Argon supply from facility to Test Cell	Notes 1, 2 and argon gas control rack	Pile oscillator, gear box, and drive motors in Test Cell	Synchro range electrical power unit, pile oscillator, gear box, drive motors.
Mechanical-Electrical Equipment	38. Battery Power Supply	Battery systems in O areas including power to Test Cell and control wiring to P area	Note 2 and complete system including batteries and chargers		
Mechanical-Electrical Equipment	39. Test Cell Pass Thrus	Pneumatic, electrical and instrumental power pass thrus including leak and pressure test	Note 2 and all Test Cell pass thrus (approximately 250) except LM piping		
Mechanical-Electrical Equipment / Portable	40. Fuel Loading Experiment Structure and Mechanical Equipment	Loading platform and F/E hoist. Core and F/E handling equipment. All installation is temporary and will be dismantled after FLE	All structure and equipment including fuel loading structure and hoist; handling equipment and assembly bench		Note 2
Mechanical-Electrical Equipment / Portable	41. Liquid Metal Handling	Electrical power and instrumental lines from S area to LM fill stations	Note 1 and LM transfer lines. Two NaK shipping containers and two gettering tanks delivered to CANEL. One furnace dolly	Transfer lines, transfer tanks, and transfer of LM to LCRE	Note 2
Instrumentation / Portable	42. Pretest Gas Analysis	Locate equipment as required	Note 1 and complete system including gas analysis and evacuation pump dollies		Note 2
Instrumentation / Portable	43. Core Installation		Note 1 and power supply, amplifier, and readout instrumental dolly	Used by CANEL as required for FLE	

Notes:
1. All remaining instrumentation, electrical and pneumatic parts, wiring and plumbing.
2. All installation drawings for auxiliary and control systems such as wiring and plumbing diagrams, conduit and tray routing, area layouts, and mechanical and electrical equipment installation drawings except for the Cryogenic Trap System will be made by the single subcontractor.

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E. FACILITY DESIGN AND CONSTRUCTION

Title II design of the LCRE test facility was approximately 70 percent complete at the end of the quarter. The revised date for completion of Title II drawings and specifications is April 19, 1963.

Construction of the facility was begun on January 21, 1963 with the start of demolition of the concrete floor in Building 629. At the end of the quarter, demolition was 98 percent complete, pile driving 45 percent complete and drilling for piers was 50 percent complete.

A preliminary critical path network based on Title I design, prepared by the construction contractor, M. W. Kellogg Co., indicates a construction completion date of August 1, 1964. A more complete network, based on Title II design, will be prepared.

Title I design of the Fabrication Support Facility was started by the Architect/Engineer, Burns & Roe, Inc., on January 8, 1963. Approximately 80 percent of Title I drawings were completed by the end of the quarter. Title II drawings and specifications are scheduled for completion during the next quarter.

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F. OPERATIONS

The operational requirements of the fuel loading phase of LCRE assembly were reviewed and a report was prepared describing the procedures and physics measurements required. The method of measuring reactivity changes as a function of core compaction was defined sufficiently to permit design of tooling to proceed.

An outline of the reactor operator training program was prepared. Initial training for supervisory personnel will commence on or about May 6, 1963. Planning for the course is approximately 90 percent complete.

A method for reactor disassembly has been evolved and is being studied in detail by the Disassembly and Hazards Group. Problems which are expected to be encountered during the control drum pinning procedure, and subsequent core removal and separation are being examined.

Thirty major disassembly tooling requirements have been determined, and design criteria for 11 of these tools have been forwarded to Tool Engineering.

A PERT network has been started to define the design and fabrication of the disassembly tools and mock-ups which will be required. Studies were begun to determine the cost and manpower requirements for the posttest disassembly, examination and waste disposal operations.

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III. SNAP-50-DR-1

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A. POWERPLANT ENGINEERING

1. Analysis and Design

Preliminary powerplant and component evaluation for a first flight 300 Kwe Rankine Cycle powerplant was completed. Preliminary design specifications presenting the initial flight system design ground rules, criteria and requirements including powerplant startup and control procedures were prepared. This powerplant, shown schematically in Fig 31 provides the point from which preliminary engineering and design studies of the first flight powerplant are to be initiated. Component selection will be based on simplicity, reliability, ease of manufacture and assembly, and minimization of development effort. The net power output was selected as being consistent with the SPUR requirement.* The power level is also compatible with a fixed radiator powerplant which can be incorporated within the Saturn C-1B upper stage envelope. Its weight is less than 20 lb/Kwe, exclusive of shield.

This powerplant utilizes three liquid metal coolants. Lithium transfers heat generated in the reactor to the potassium in the boiler. Potassium is vaporized and is the working fluid in the Rankine Cycle power conversion loop. Eutectic NaK is used in the main heat rejection system to reject waste heat to space. NaK is also used in the auxiliary heat rejection system which provides cooling for the pumps, turbo-alternator and transformer.

The basic characteristics of the powerplant are summarized below: UNCLASSIFIED

a. The reactor design power level is nominal eight Mwt.

b. The boiler is a once-through, total vaporization unit which incorporates an integral preheater section.

c. The turbine utilizes interstage liquid extraction.

d. The net electrical output of the powerplant, 300 Kwe, is supplied at generator output conditions of 1000 cps and a maximum of 1000 volts.

e. All liquid metal circulation is by canned motor, centrifugal pumps.

f. The main heat rejection system is segmented into four independent NaK loops.

g. The auxiliary heat rejection system is independent of the other fluid systems.

h. All radiators utilize type 316 stainless steel tubes and headers.

Analytical studies to determine system thermal losses from preheat on the ground to powerplant startup in orbit are in progress. Each liquid metal circuit of the powerplant is being treated separately to facilitate analysis. A two-dimensional, IBM-7090 heat transfer program has been modified and calculations are in progress to determine the temperature-time history of the lithium loops. An initial preheat temperature of 1000F and continuous circulation of lithium through the primary system is assumed. The analytical mockup incorporates a meteoroid barrier and concentric thermal shield around the components and piping. Similar studies to determine the temperature-time history of the ground preheated, thermally shrouded radiator are in progress.

* Report SY-5396-R2, Air Research Division, The Garrett Corporation; 10/15/62.

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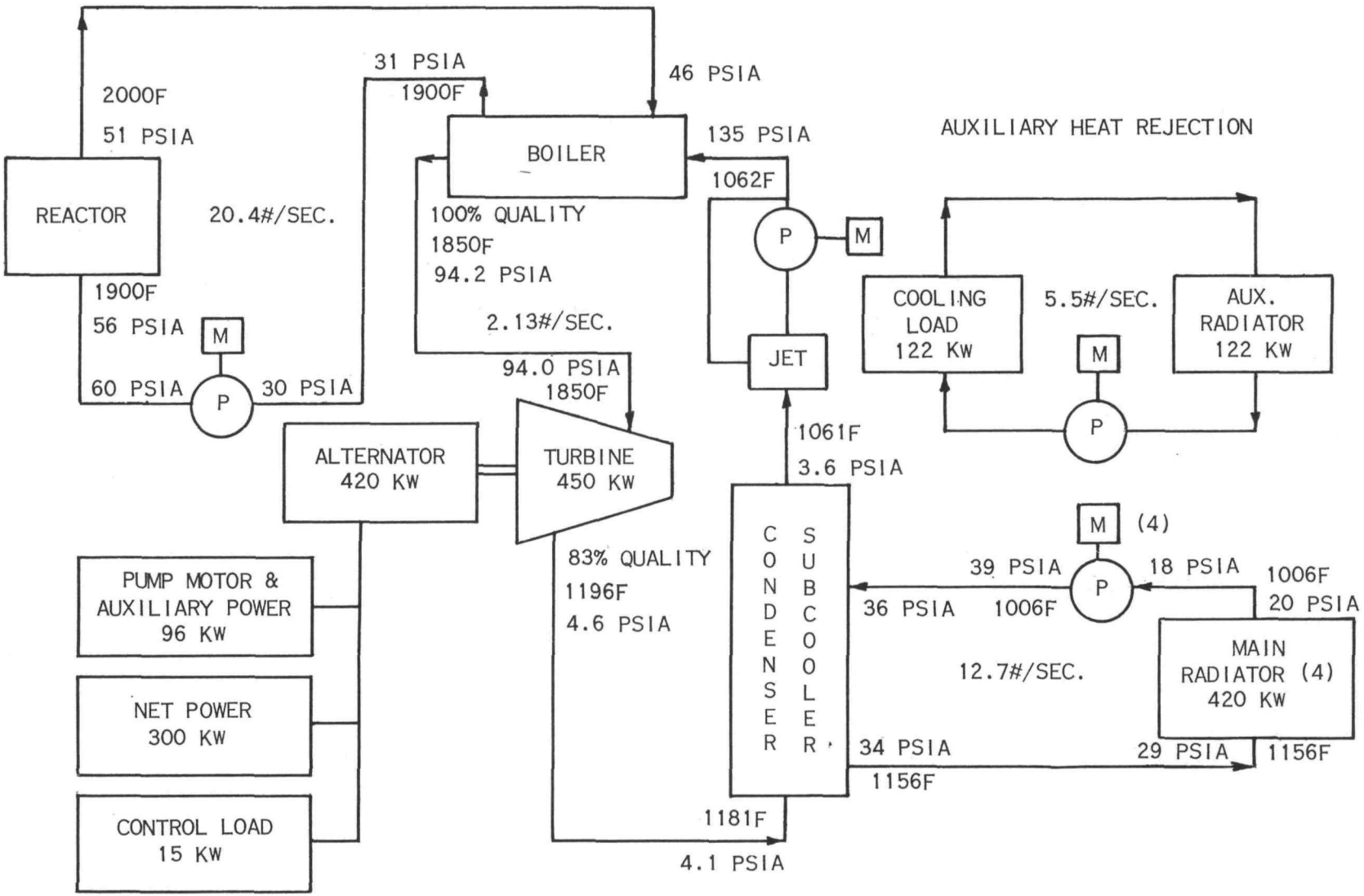
SNAP-50 FIRST FLIGHT FLOW SCHEMATIC NET POWER OUTPUT 300 KWE



PWAC- 631

FIG 31

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Analytical studies to investigate the radiator configuration and to provide parametric performance are in progress. Analyses of flat-plate radiators, fabricated of type 316 stainless steel tubes, copper finned, and utilizing eutectic NaK as the coolant, indicate that these are feasible for the initial powerplant design. Further analysis, to obtain parametric data for powerplant performance studies and to investigate radiators of cylindrical configuration, is in process.

Powerplant layout design studies to investigate structural support methods and powerplant arrangement in a Saturn C-1B booster were started. A preliminary powerplant schematic layout has been prepared for study purposes.

Preparation of detailed structural design criteria for the SNAP-50 first flight powerplant was completed. Included in the criteria are basic materials data for space applications design, including meteoroid protection parameters, sublimation rates, creep and structural data.

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2. Controls

Development of a mathematical model of the SNAP-50 powerplant was initiated. This model will be used in future analog computer, dynamic powerplant analyses and control system determinations. The basic model mockup has been completed and a check out case is in preparation.

A preliminary analysis of the proposed SNAP-50 reactor control system has proven the feasibility of core outlet temperature control. A suitable shim speed appears to be about one-quarter cent per second. An on-off control may be used. This will keep the outlet temperature within ten degrees of its set point and will allow as much as one cent of backlash in the reflector drive. More backlash and higher shim speeds can be accommodated by incorporating more compensation in the thermocouple circuit. It was found that this control mechanism could be used at 1.8 Mwt, (corresponding to 300 Kwe), and at 0.9 Mwt (150 Kwe). At higher powers, higher shim rates and increased backlash require additional compensation.

3. Systems Tests

The shock and vibration environment for the SNAP-50 system was specified in terms of the launch load spectra. These environmental requirements, coupled with estimated component and system masses, were utilized to determine initial facilities requirements for the SNAP-50 vibration and shock testing program. An investigation to determine the capabilities and availability of vibration ans shock test facilities within the United Aircraft Corporation was completed. Further investigation to determine other available facilities and their capabilities is in progress. A study to determine the feasibility and limitations of scale model testing has also been initiated.

Preliminary requirements for the non-nuclear powerplant test facility were completed. The test program, scheduled to start in mid-fiscal year 1966, will consist of a series of tests of the complete powerplant, exclusive of the reactor. It is planned to utilize the existing CANEL Radiator Laboratory and its basic services, modified to accept vacuum chambers, heat dumps and associated equipment, for this test program.

The preliminary criteria for the nuclear test facility have been prepared and submitted to the Idaho Operations Office for cost estimation. This facility will replace both the original SNAP-50-DR-1 facility and the ground prototype facility. It will be capable of testing the reactor separately or the reactor with any of the non-nuclear powerplant components. The powerplant will be tested in as near to flight configuration as is pos-

sible, within the limitations of gravity and ground test safety considerations. The test will be contained in a steel sphere approximately 100 feet in diameter. The facility will also contain a clean assembly area, a hot cell for remote disassembly and necessary engineering and equipment areas. Use will be made of LCRE service areas.

4. Reliability

A formal data collection system has been inaugurated at CANEL for the compilation and correlation of all development test data, and for the recording, analyzing, and reporting of all failures and malfunctions which occur during testing. This system should ensure coordination and dissemination of all vital test data. It provides a means based on accurate development history through which to initiate prompt action to prevent or correct failures.

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B. SNAP-50 REACTOR SPECIFICATIONS PRELIMINARY

Power Level, Mwt	8
Design Lifetime, hr	10,000
U^{235} mass loading, kg	86
Coolant	Lithium (99.9% Lithium-7)
Inlet temperature, F	1900
Outlet temperature, F	2000
Fuel material	UC (84% enrichment)
Structural material	Cb-1 Zr
Side reflector material	BeO
End reflector material	BeO
Method of cooling side reflector	Radiation to space
Method of control	Axially moving side reflector
Reactor weight, lb	860
Active core diameter, in.	10.43
Active fuel length, in.	10.43
Thickness of end reflectors, in.	1.5
Side reflector thickness, in.	1.6
Pressure vessel I.D., in.	11.78
Pressure vessel O;D;, in.	12.33
Average power density, kw/cc	0.96
Average heat flux, btu/hr-ft^2	6.5×10^5
Average ΔT in film, F	17.0
Average ΔT in clad, F	38.0
Maximum fuel temperature, no hot channel factors, F	3200
Maximum total temperature difference, bulk coolant to fuel centerline, F	1250 approx.
Coolant inlet pressure, psia	70
Coolant outlet pressure, psia	40
Coolant flow rate, lb/sec	76.6
Average coolant velocity in core, ft/sec	26.2
Average film coefficient, btu/ft^2-hr-F	41,500
Maximum to average power ratio	1.7
Average Reynolds number	53,000
Fuel pin O.D., in.	0.432
Cladding thickness, in.	0.025

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Pin spacing, in.	0.017
Fission gas void length, in.	4
Number of pins in core, nominal	434
Core volume fractions:	
Uranium carbide	0.573
Columbium	0.180
Lithium	0.247
Side reflector outer temperature, F	1825
Side reflector inner temperature, F	2000
U^{235} burn-up, atom percent	4.9
Moving reflector control worth, % ΔK/K	8.0
Median fission energy, kev	400
Prompt neutron lifetime, sec	3×10^{-8}
Temperature coefficient of reactivity, isothermal, excluding Doppler effect, ΔK/F	-1×10^{-5}
Meteoroid shield material	BeO
Meteoroid shield thickness	0.4
* Doppler coefficient at 1000F, Δk/F	$+0.05 \times 10^{-6}$
* Doppler coefficient at 70F, Δk/F	$+0.20 \times 10^{-6}$
* Radial expansion coefficient - core plus end reflector, Δk/F	-3.2×10^{-6}
* Radial expansion coefficient - side reflector, Δk/F	-2.2×10^{-6}
* Axial expansion coefficient - core plus end reflector, Δk, F	-2.7×10^{-6}
* Axial expansion coefficient - side reflector, Δk/F	-1.1×10^{-6}
* Coolant expansion coefficient, Δk/F	-2.4×10^{-6}
* Total coolant worth, Δk/k	0.024

*These items added

C. RESEARCH AND DEVELOPMENT

1. Reactor and Shield

Preliminary design specifications for a uranium carbide fueled, eight Mwt reactor selected as a reference design were modified in accordance with continued engineering analyses as shown in Section III B.

Preliminary structural analyses were begun on the design configurations. Methods of structural analysis of the pressure vessel were prepared and preliminary elastic analyses of the lower dome were completed. The highest stress in the lower dome occurs at the junction of the inlet pipe and the hemispheric shell and is estimated to be 1130 psi.

In present configurations the lithium flows up between the reactor pressure vessel and a core liner or skirt. It reverses in the upper plenum and flows down through the fuel elements, which are supported by a core plate at the top. With this arrangement, the pressure on the outside of any fuel subassembly can is equal to the core exit pressure and therefore is less than the pressure inside the can. The resulting outward pressure differential tends to cause bulging which may lead to undesirable changes in flow distribution in the cans. Various methods of reducing or reversing the pressure differential are being investigated.

Calculations of the heat transfer through the flow separator between inlet and outlet flow below the core were made. They showed a negligible temperature effect between the core inlet and outlet temperatures. The analytical optimization procedure to determine minimum weight reactors for given values of pressure drop, coolant temperature rise and power levels was coded for machine computers. Trial calculations showed relatively good agreement with previous optimizations done by hand. Existing codes for the calculation of flow distribution, pressure drop, power distribution and core temperature were modified to apply to the present design. Preliminary cases for the present reactor design were run and the resulting flow and temperature distribution was used in structural calculation and physics studies. Local pin temperatures and pressure levels calculated by these methods showed good agreement with the results obtained from previous parametric studies. A series of programs is being written for different core geometries to develop physical dimensions and specifications in evaluating various arrangements of core boundaries, core cans or internal flow dividers, core skirt, etc.

Further studies of comparable cylindrical and spherized reactor configuration have shown no appreciable weight differences between the two systems. Doppler and thermal expansion reactivity coefficients were calculated for the reference design and are included in the table in Section III A.

The Li^8 activity at the reactor outlet was calculated to be 0.49 millicuries per cc at 9 Mwt power level. The bremsstrahlung from the lithium piping in a typical powerplant arrangement at a distance of 60 feet from the reactor was estimated to be about 0.1 r/hr. This Li^8 activity is about 100 times less than in the LCRE because of the absence of large, low energy flux peaks in the end reflectors.

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A study of the change in neutron leakage with BeO reflector thickness was completed. Increasing the thickness from one to five inches reduces the fast neutron leakage by a factor of two and degrades the fast leakage spectrum significantly. To assess the possible effect of the fast neutron leakage spectrum on the shielding problem, a preliminary energy dependent damage function was derived for semiconductors, assuming damage to be proportional to displacement production. This showed that damage begins to occur at a minimum neutron energy of the order of 100 ev. Damage then increases approximately proportionately to neutron energy until an upper limit is reached in the Mev range after which damage is essentially independent of neutron energy.

Preliminary equations were derived to give approximate values for direct and scattered neutron flux at the payload and for shield weight for an 8 Mwt reactor with an extended-wing radiator. As an initial, simplified example for an all lithium hydride shield and an integral fast flux dose of 10^{13} nvt, a shield weight of 2240 pounds was calculated. Of this, 800 pounds was for the radiator shield. The minimum shield weight for this configuration was found to occur when the contributions of direct and scattered neutron fluxes were approximately equal. It was found that radiation scattering to the payload from the radiator shield exceeded radiation from the radiator itself.

A two-region shield is being evaluated which has a thermal and gamma shield region. This consists of beryllium, boronated steel and depleted uranium, followed by and insulated from a clad lithium hydride region. Conclusions from a review of the literature on the properties of lithium hydride and beryllium are favorable. Heat transfer calculations, which included the effects of cladding with ample gap allowance for expansion of lithium hydride, have indicated that tolerable temperatures can be maintained.

Neutron attenuation studies made with the DSN code are in poor agreement with empirical fast neutron relaxation lengths. The problem is being investigated by calculations for a point source geometry. Sample design specifications for a UO_2 fueled reactor were obtained from an initial parametric study, based on minimizing reactor weight over a wide range of fuel pin center temperatures. A typical set of reactor conditions was selected as a basis for designing fuel test specimens and to set irradiation test conditions for a UO_2 fuel development program. The selected reactor had a core length and diameter of 12.75 inches, fuel pin diameter of 0.22 inch, cladding thickness of 15 mils and maximum fuel pin center temperature of 3200F at 8 Mwt.

A brief study was made of the critical diameter of a UC-ZrC reactor. This was compared with UO_2 and BeO fueled reactors on the basis of one-dimensional, spherical DSN calculations. It was found that the diameter of a core with 25 volume percent ZrC in the fuel matrix was about the same as a UO_2 reactor. The diameter for 55 percent dilution was the same as a LCRE type UO_2-BeO fueled core. The study of an optimized SNAP-50 type reactor using LCRE type UO_2-BeO fuel pins was begun.

Preparations for critical experiments to obtain reactor physics data on reactors of the SNAP-50 type were completed and all non-fissionable materials were installed. After notice of approval of the Hazards Summary report was received from the Atomic Energy Commission, the first configuration of the CCA-7 series, a 12 inch diameter, 14.2 inch long core with 3 inch BeO side and end reflectors, was brought to criticality with a U^{235} loading of 97 kilograms.

2. Fuel

An area of the Fuel Element Laboratory was assigned to SNAP-50 fuel process development and installation of plumbing and electrical services has been completed. An automatic powder compacting press, incorporating a vacuum dry box chamber enclosing the working surfaces was received. Assembly and installation is in progress. Design of powder synthesis rigs was completed. Design of a cold-wall, vacuum sintering furnace utilizing resistance-heated graphite elements and graphite heat shields was completed and fabrication initiated.

Compatibility tests at 2200F and 2500F in vacuum of hyperstoichiometric uranium carbide against Cb-1 Zr alloy, utilizing polished faces in close contact, showed uranium diffusion into the Cb-1 Zr alloy regardless of fuel carbon content. Therefore excess carbon as U_2C_3 is not always effective in preventing uranium diffusion into Cb-1 Zr alloy cladding. This confirms Battelle inpile test results.

Tungsten foil barriers prevented reaction between stoichiometric uranium carbide and Cb-1 Zr alloy for 3000 hours at 2200F and for 100 hours at temperatures up to 3300F. No evidence of uranium diffusion into tungsten nor inter-diffusion between tungsten and columbium was observed. A series of 1000-hour, 2200F fuel-to-cladding compatibility tests were completed and posttest examination was started. Specimens consisted of $UC_{1.08}$ pellets in both bare and tantalum-lined Cb-1 Zr alloy claddings. The cladding on half of the specimens was intentionally defected and all specimens were immersed in lithium during test. Defected specimens, both with and without tantalum barrier liners, exhibited general bulging of the cladding in the region corresponding to fuel pellets, and local reactions and bulging at the ends of the fuel pellet stack. These local reaction spots contained uranium as indicated by radiographic inspection of specimens. A summary of data for the 1000 hour tests follows:

Results of Compatibility Test No. C-202
($UC_{1.08}$ versus Cb-1 Zr alloy in 2200F Lithium for 1019 hours)

Cladding	Cb-1 Zr	Cb-1 Zr	Cb-1 Zr	Cb-1 Zr
Barrier	None	None	2-mil Ta	2-mil Ta
Defective*	Yes	No	Yes	No
Fuel Composition, w/o C Pretest	5.08	5.08	5.21	5.21
Posttest	4.61	5.04	4.56	5.18
Cladding reaction, mils carburization	1.0	0.5	Nil**	Nil
U-Cb alloying	25	Nil	Nil	Nil
Barrier reaction, mils carburization	---	---	2.0	1.0
U-Ta alloying	---	---	0.5	Nil

*Drilled hole in one end of specimen
**Lithium attack to Cb-1 Zr alloy suspected

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Conclusions from this test are:

a. Tantalum liners provide effective protection against carburization of the Cb-1 Zr alloy cladding by hyperstoichiometric uranium carbide.

b. Defective specimens with no barrier show extensive uranium alloying with the cladding.

c. Defective specimens with tantalum barriers show no uranium in the cladding, but a small amount may alloy with the tantalum.

Compatibility tests of uranium nitride against Cb-1 Zr show slight interactions at temperatures above 2000F with the formation of Cb_2N precipitates and uranium-columbium alloy. Although this effect is markedly less than the uranium carbide-columbium interaction at 2200F, diffusion barriers are desirable. Tungsten foil barriers have effectively prevented uranium nitride to columbium interactions at 2200F. Neither nitrogen nor uranium was detected in Cb-1 Zr alloy protected by 1-mil-thick tungsten foil after 1700 hours at 2200F and no reactions were observed following 100 hours at temperatures up to 2800F.

Uranium carbide powder, received from a supplier for evaluation, although higher than specified in oxygen, appears to sinter to moderate density. Uranium nitride powder was received from another vendor. Chemical analysis indicated higher oxygen content, and preliminary compacting and bulk density tests indicated larger particle size than specified. Only moderate density pellets were achieved using the standard sintering cycle.

Three inpile capsules containing uranium carbide fuel, one containing uranium nitride fuel and the corresponding out-of-pile control capsules were examined following tests. Fig 32 summarizes data from the examination of uranium carbide capsules.

Examination of inpile capsule PW26-190, containing ten percent enriched, stoichiometric uranium monocarbide with tungsten-lined Cb-1 Zr alloy cladding, was completed. Metallographic examination, Fig 33, showed the tungsten foil to be an effective barrier to cladding-fuel interactions. Where cracking of the tungsten foil occurred, carburization of the Cb-1 Zr alloy was observed. Extensive reaction, as previously reported for the PW26-101 through -109 capsule series, was observed between fuel pellets and Cb-1 Zr alloy end spacers which were not protected by tungsten barrier foil. The fuel showed increased porosity and grain size compared with the micro-structure before irradiation. Porosity and grain size were greatest in the interior and generally corresponded to temperature gradients during iiadiation. Although no significant change in external specimen dimension or weight was detected, the top pellet of the bottom specimen was bulged approximately 50 mils into the gas void space. The specimen cover gas pressure and fission gas release from the fuel matrix were relatively high.

Examination of inpile capsule PW26-151, containing ten percent enriched, hyperstoichiometric (5.2C) uranium carbide in tantalum-lined Cb-1 Zr alloy cladding, was completed. Metallographic examination showed the tantalum foil to be an effective

RESULTS OF URANIUM CARBIDE CAPSULE IRRADIATIONS

Capsule Number	PW26-190C	PW26-190	PW26-150C	PW26-151	PW26-170C	PW26-171
Fuel Composition	$UC_{1.0}$	$UC_{1.0}$	$UC_{1.08}$	$UC_{1.08}$	$UC_{1.08}$	$UC_{1.08}$
Barrier	W (1 mil)	W (2 mils)	Ta (2 mils)	Ta (2 mils)	None	None
Cladding	Cb-1 Zr	Cb-1 Zr	Cb-1 Zr	Cb-1 Zr	Cb-1 Zr	Cb-1 Zr
Test Duration, hours	1000	1069	1000	1097	1000	892
Cladding Temperature, F						
Maximum	2200	2237	2200	2256	2200	2255
Average*	2200	2055	2200	2175	2200	2135
Fission Burnup %	Out-of-pile	6.8	Out-of-pile	6.6	Out-of-pile	In progress
Fission Density, f/cc x 10^{-20}	Out-of-pile	.1	Out-of-pile	2.0	Out-of-pile	In progress
Power Density, kw/cc	Out-of-pile	1.7	Out-of-pile	1.6	Out-of-pile	In progress
Cover Gas Pressure, psia	10.2	38.2	9.3	11.5	9.7	9.5
Fission Gas Release, % Xe	---	76.0	---	7.5	---	In progress
Fuel Density, % of Theor.						
Pretest	93.1	92.4	95.2	95.2	97.2	97.2
Posttest	ND	ND	95.3	ND	ND	95.0
Specimen Diameter Change, in.	Nil	+0.001	Nil	+0.005	Nil	+0.0025
Fuel Diameter Change, in.	ND	ND	ND	ND	Nil	+0.001
Fuel Length Change, in.	Nil	+0.050**	Nil	+0.020*	Nil	+0.001

Data are for center specimen unless otherwise indicated.
*Time weighted average cladding temperature
**Top pellet of bottom specimen
ND = Not determined

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HM26190186

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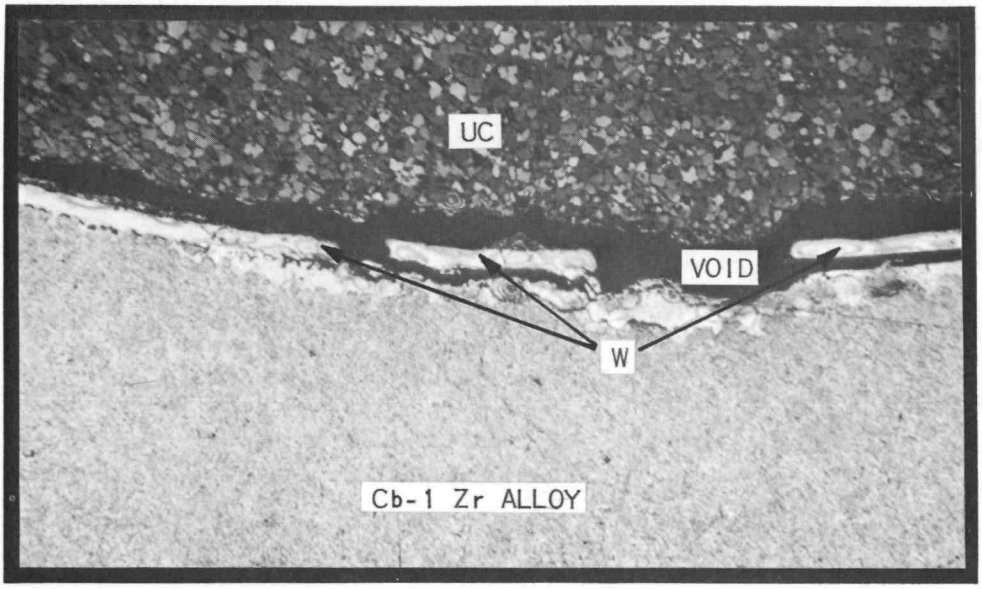
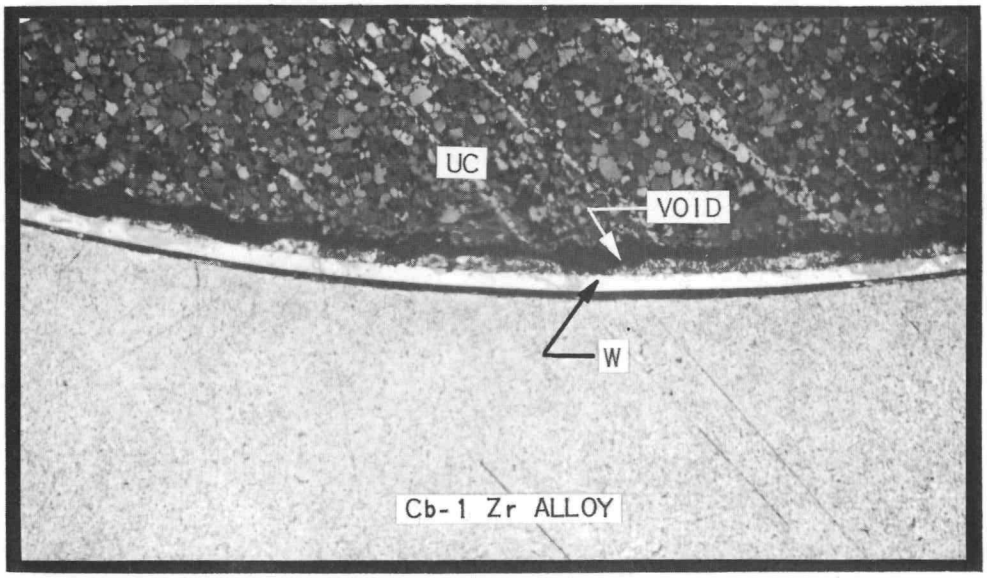
MIDDLE SPECIMEN, CAPSULE PW26-190



ETCHANT: 10 H_2O; 5 HNO_3; 2 HF MAGNIFIED. 100X

A TRANSVERSE SECTION SHOWING CARBURIZATION OF Cb-1 Zr ALLOY CLADDING BEHIND BREAKS IN TUNGSTEN BARRIER FOIL.



ETCHANT: 10 H_2O; 5 HNO_3; 2 HF MAGNIFIED: 100X

B TRANSVERSE SECTION SHOWING PROTECTION OF Cb-1 Zr ALLOY BY TUNGSTEN FOIL BARRIER.

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barrier for cladding-to-fuel interactions. Both irradiated and out-of-pile specimens showed generally uniform layers in the tantalum foil barrier about 0.2 mils thick and some intergranular carbide phase extending through the two-mil thick foil. No carbide phase was observed in the Cb-1 Zr alloy cladding (Fig 34). The fuel after irradiation showed a general depletion of carbon in contrast to the as-fabricated material which had U_2C_3 surrounding all grains (Figs 35 and 36). Some UC_2 was uniformly dispersed as a Widmanstatten precipitate in all grains. Slightly increased porosity was observed in the fuel following irradiation. In addition, a phase of hardness 1300 DPH extended from the surface to approximately eight mils into the fuel (Fig 37). Identification of this phase is being attempted. Approximately two percent increase in specimen diameter was measured on the irradiated specimens. The top fuel pellet of the bottom specimen bulged approximately 20 mils into the gas void space. Specimen cover gas pressures and fission gas release from the irradiated fuel were relatively low. Micro-examination of the fuel from the out-of-pile control test showed a zone at the surface approximately 20 mils thick which was depleted in carbon, some agglomeration of U_2C_3 and some UC_2 dispersed within grains. In general, the microstructure was unusually non-homogeneous, with areas of agglomerated U_2C_3 existing in close proximity to areas of UC grains surrounded by intergranular U_2C_3 as in the pretest structure.

Examination of inpile capsule PW26-171, with ten percent enriched, hyperstoichiometric (5.2C) uranium carbide against bare Cb-1 Zr alloy cladding, and the corresponding out-of-pile control capsule, PW26-170C, was begun. Results available to date are encouraging. Less than one percent increase in irradiated specimen diameter was measured. Fuel pellets were easily removed from specimens upon disassembly (Fig 38). The irradiated fuel pellets showed slight increases in dimensions and decreases in density. The inner surface of the cladding showed machining marks caused by sizing the cladding to fuel dimensions. Metallographic examination showed carburization of the cladding to a depth of approximately 2 mils, principally at the high spots of the machining marks (Fig 39). More extensive reaction, to a depth of approximately 13 mils, occurred on the flat end spacer. Identification of this reaction zone is in progress. Microexamination of the irradiated fuel showed some depletion of carbon from the interior of the pellets and agglomeration of a phase near the surface (Fig 40). In general, the effects in the irradiated fuel were similar to those for capsule PW26-151. Specimen cover gas pressures, which are a measure of fission gas release, were especially low. Specimens from the out-of-pile test showed effects in the cladding similar to those observed in the irradiated specimens. Microstructure of the fuel was similar to that in the out-of-pile, tantalum-lined test, PW26-150C. A comparison of results from the three uranium carbide fueled capsules examined to date may be summarized as follows:

a. Tungsten was an effective barrier between nearly stoichiometric uranium carbide and Cb-1 Zr alloy cladding. However, breaks in the tungsten barrier occurred and carburization of the Cb-1 Zr alloy was observed behind the cracks.

b. Tantalum was an effective barrier between hyperstoichiometric uranium carbide and Cb-1 Zr alloy. Carburization of the tantalum occurred as predicted, but no carbide phase was observed in the Cb-1 Zr alloy.

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MIDDLE SPECIMEN, CAPSULE PW26-151

UNETCHED MAGNIFIED: 100X

TRANSVERSE SECTION SHOWING PROTECTION OF Cb-1 Zr ALLOY BY TANTALUM.
NOTE CARBIDE PHASE IN TANTALUM BUT ABSENCE OF CARBIDE IN Cb-1 Zr ALLOY

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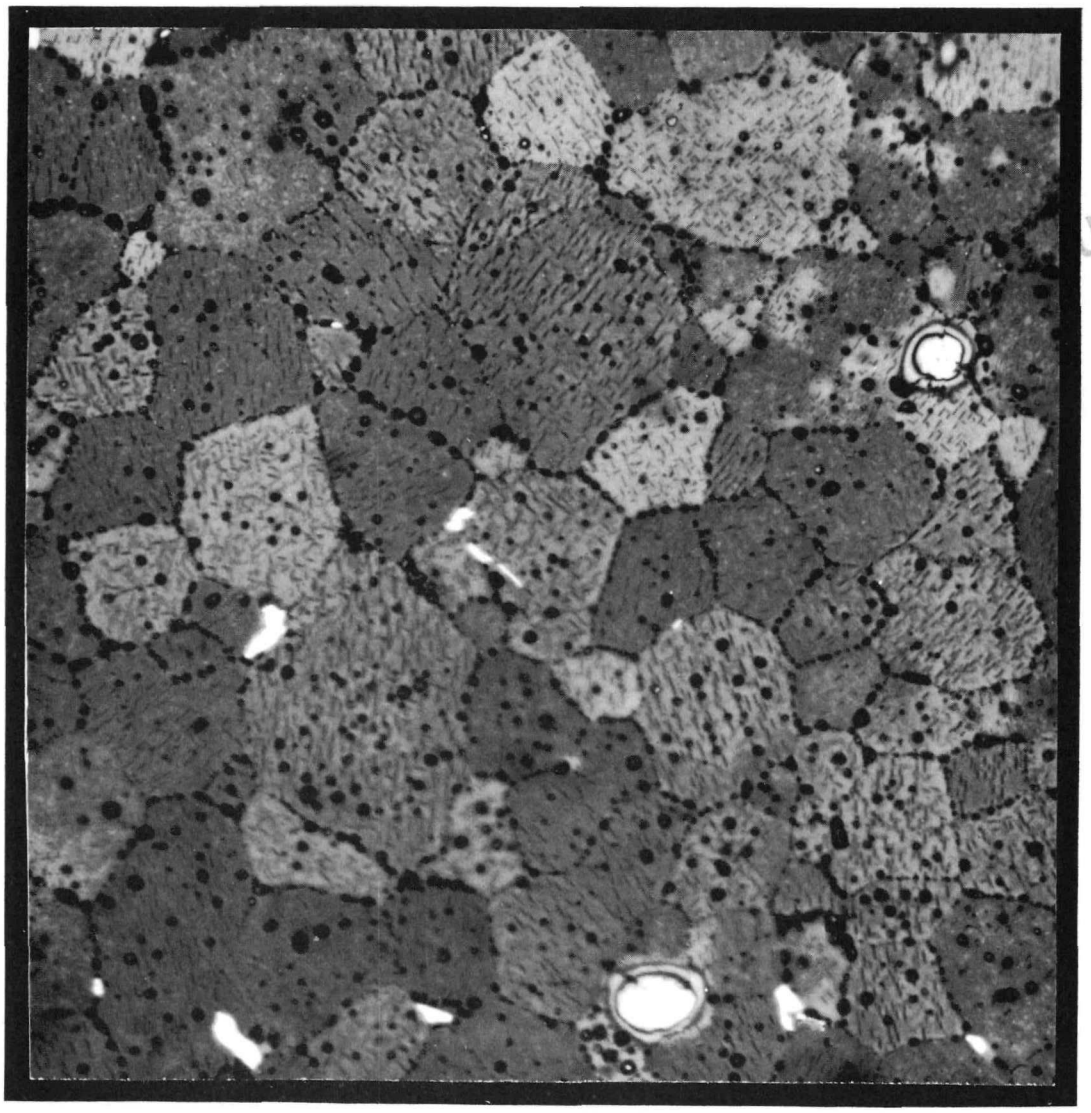
FIG 35

MIDDLE FUEL PELLET, MIDDLE SPECIMEN, CAPSULE PW26-151
HYPERSTOICHIOMETRIC URANIUM CARBIDE



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ETCHANT: 30 C_2H_5COOH, 25 HNO_3 MAGNIFIED: 500X

SHOWS MICROSTRUCTURE TYPICAL OF FUEL FOLLOWING IRRADIATION

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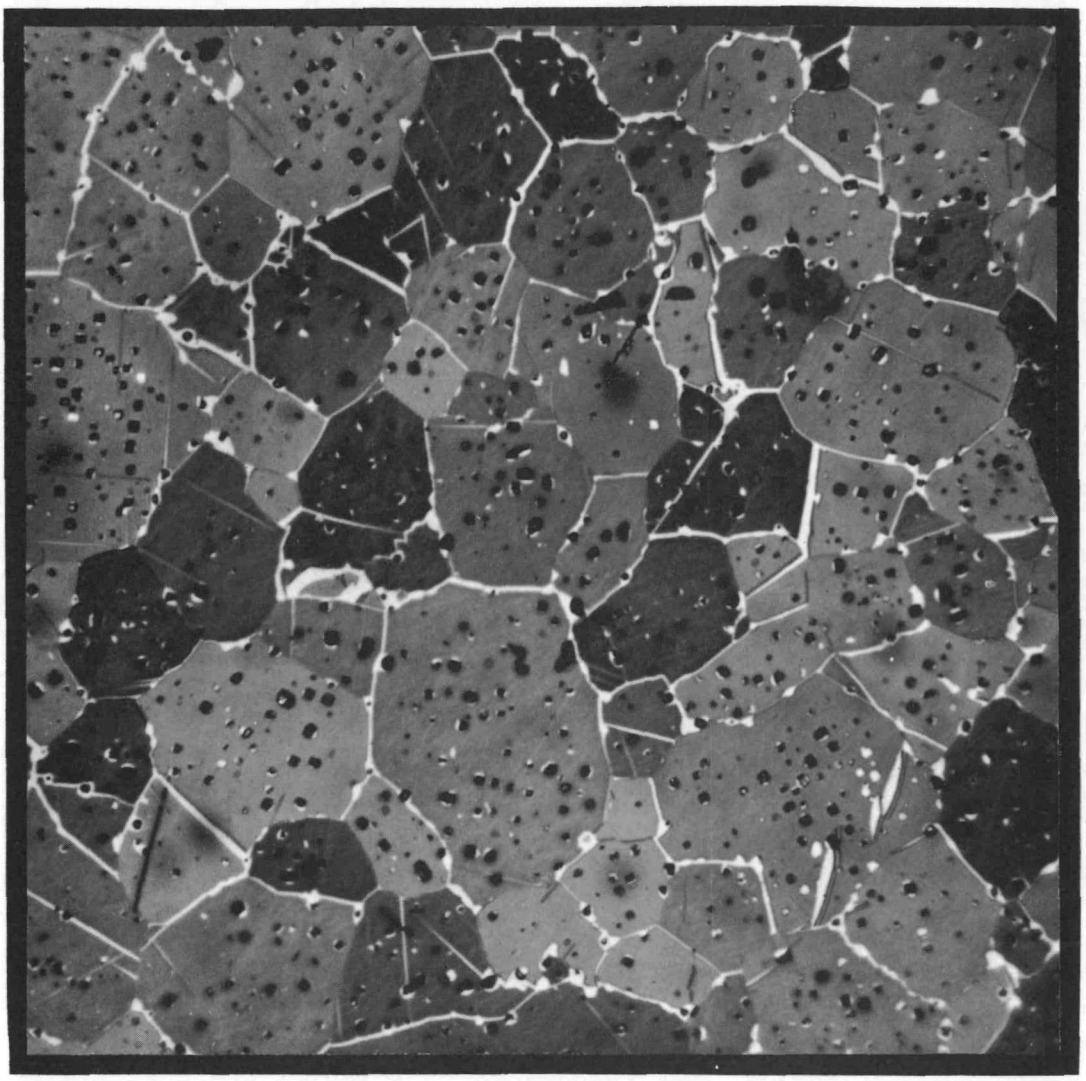
HYPERSTOICHIOMETRIC URANIUM CARBIDE FUEL PELLET



ETCHANT: 30 C_2H_5COOH, 25 HNO_3 MAGNIFIED: 500X

SHOWS MICROSTRUCTURE OF FUEL PELLET TYPICAL OF CAPSULE PW26-151 BEFORE IRRADIATION.

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FIG 37

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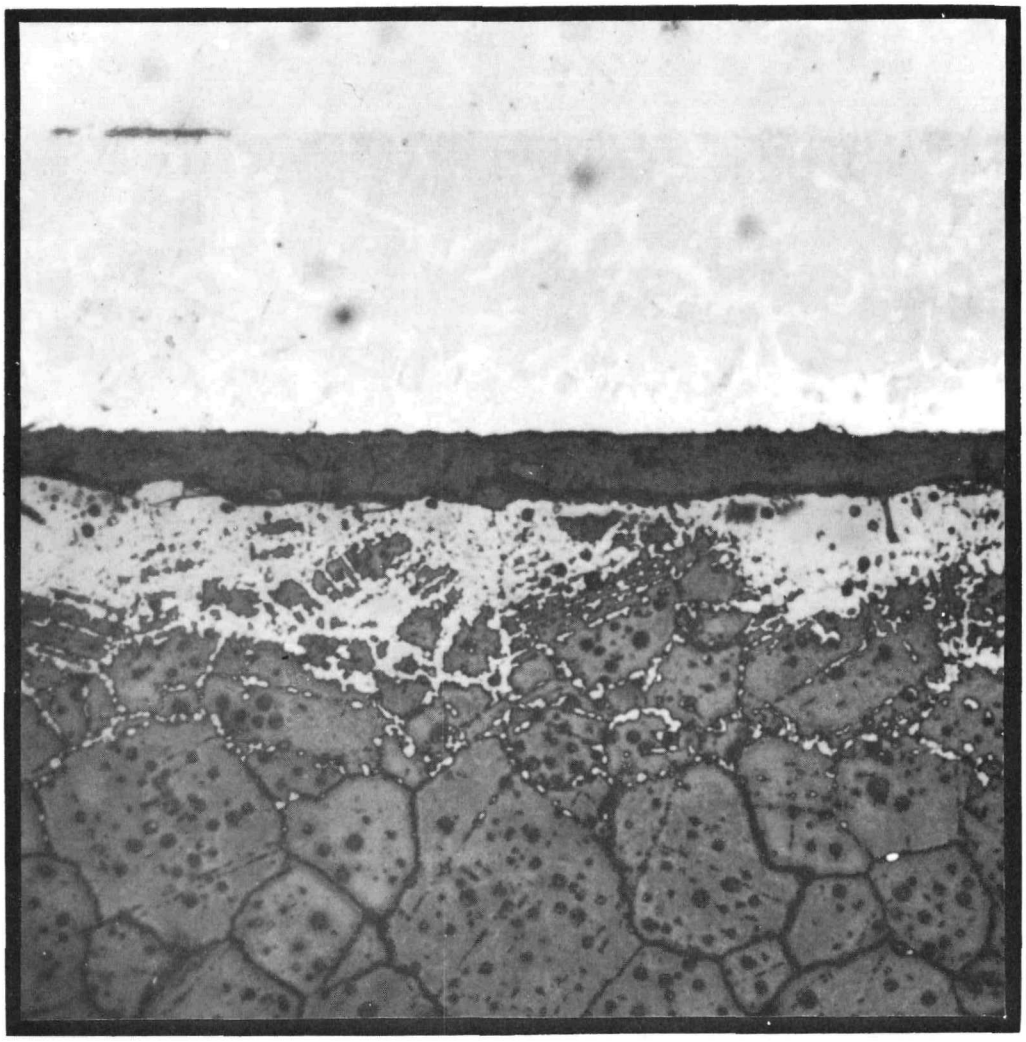
TOP SPECIMEN, CAPSULE PW26-151



ETCHANT: 30 C_2H_5COOH, 25 HNO_3 MAGNIFIED: 500X

LONGITUDINAL SECTION SHOWING PHASE AT PELLET SURFACE.
ALSO NOTE CARBIDE PHASE IN TANTALUM.

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FIG 38

BOTTOM SPECIMEN, CAPSULE PW26-171, DISASSEMBLED

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LEGEND

A END CAPS
B TUBULAR SPACER
C CLADDING
D UC PELLETS
E FLAT END SPACER

MAGNIFIED: APPROX. 4X

NOTE SHARP PELLET EDGES, MACHINING MARKS ON CLADDING ID AND DUCTILITY OF CLADDING.

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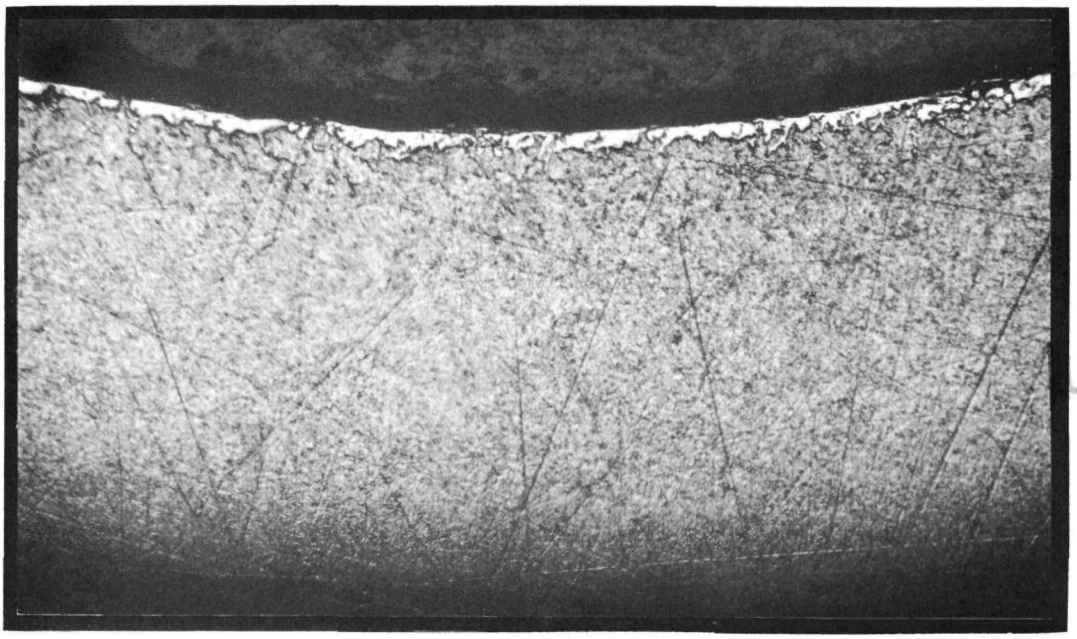
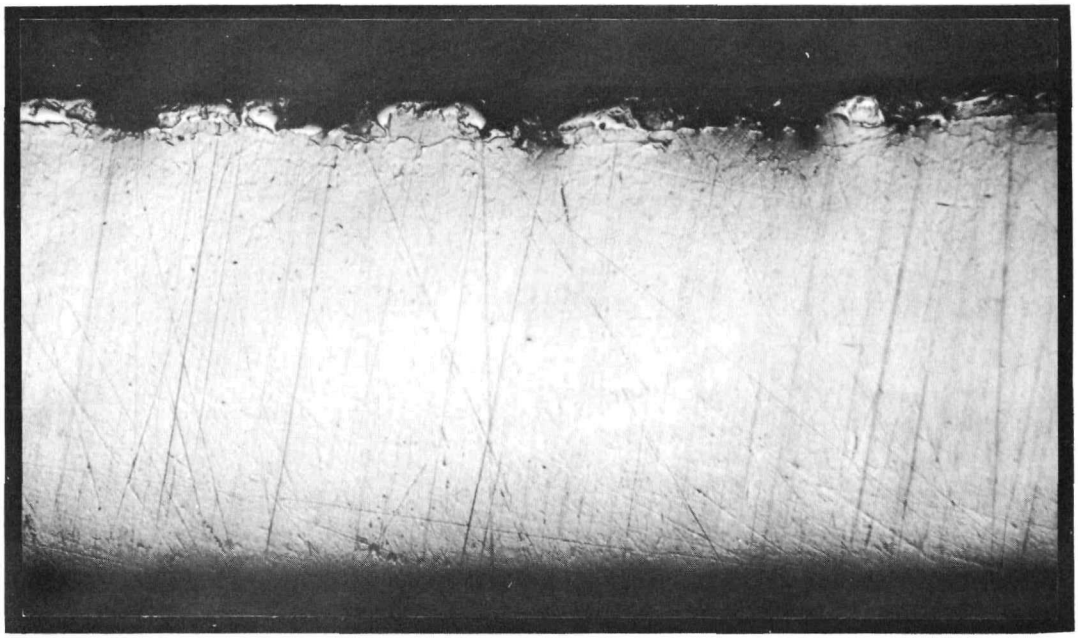
FIG 39

Cb-1 Zr ALLOY CLADDING FROM MIDDLE SPECIMEN, CAPSULE PW26-171



ETCHANT: 10 H_2O, 5 HNO_3, 2 HF MAGNIFIED: 100X

TRANSVERSE SECTION SHOWING CARBURIZATION OF CLADDING AT ID.



UNETCHED MAGNIFIED: 100X

LONGITUDINAL SECTION SHOWING CARBURIZATION AT HIGH POINTS OF MACHINING MARKS.

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MIDDLE FUEL PELLET, TOP SPECIMEN, CAPSULE PW26-171

HYPERSTOICHIOMETRIC URANIUM CARBIDE

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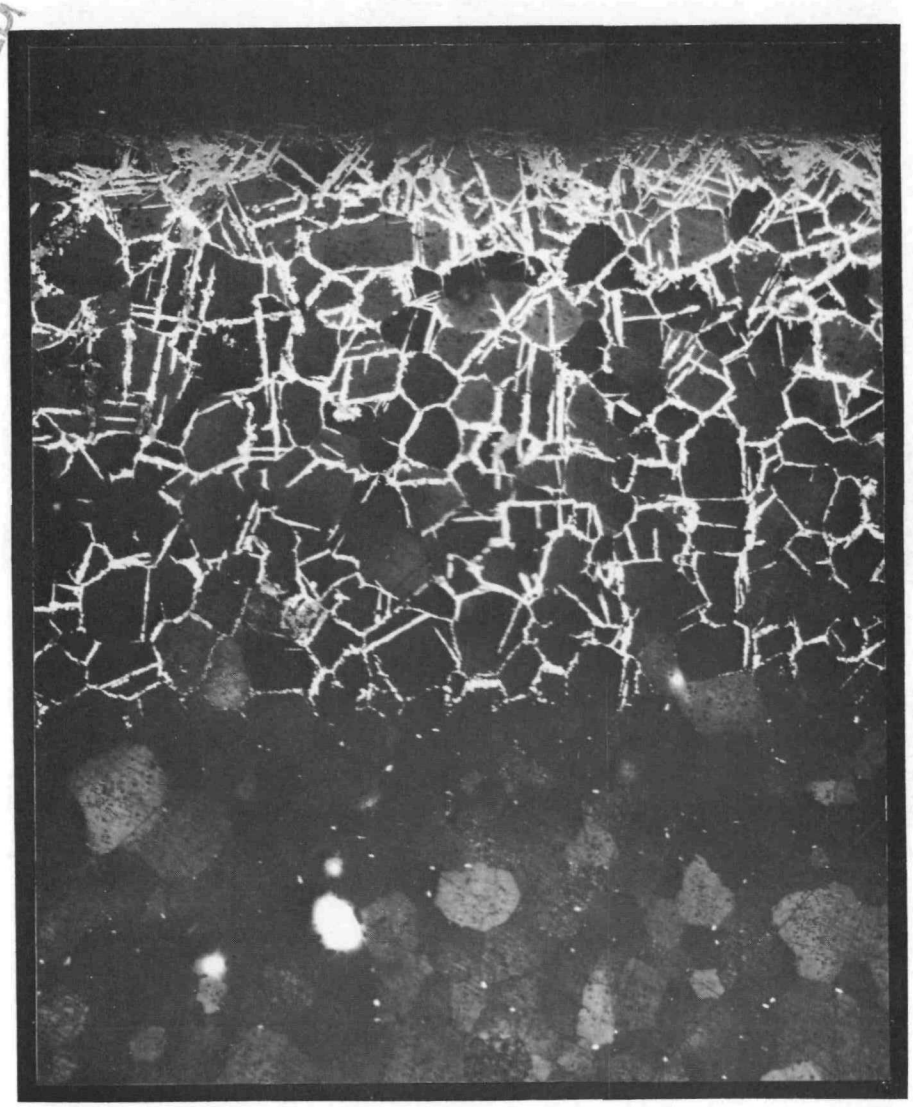

ETCHANT: 30 C_2H_5COOH; 25 HNO_3 MAGNIFIED: 100X

SHOWING DEPLETION OF SECOND PHASE FROM INTERIOR AND AGGLOMERATION NEAR THE SURFACE.

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c. Reactions between hyperstoichiometric uranium carbide and Cb-1 Zr alloy were influenced by surface finish of the cladding.

d. Good fission gas retention was observed for both capsules containing hyperstoichiometric fuel while poor retention was associated with the stoichiometric fuel. However, fission gas retention appears related to pre-irradiation density of the fuel.

e. Dimensional stability of the fuel appears related to pre-irradiation density but the compositional differences and the agglomeration and/or migration of a second phase may influence both dimensional and fission gas retention.

Examination of inpile capsule PW26-210, containing 31.5 percent enriched uranium mononitride with bare Cb-1 Zr alloy cladding, was completed. Results continue to indicate promise for this fuel as an alternate in SNAP-50 reactors. As previously reported (PWAC-624), specimens were disassembled with no difficulty (Fig 41). No change in specimen dimension was detected. Metallographic examination showed intergranular reaction (CbN) 8 mils deep into the cladding (Fig 42). The presence of uranium in a phase penetrating the cladding to 1.5 mils could not be positively identified but is suspected (Fig 43). Fuel showed only slight increases in porosity, generally near the surface, and exhibited good fission gas retention. Fig 44 shows microstructure typical of the fuel following irradiation. The following data were collected for the center specimen of this capsule.

Capsule	PW26-120
Test Duration, hours	1092
Cladding, Temperature, F	
Maximum	2190
Time Weighted Average	2106
Fission Burnup, % U^{235}	1.3
Fission Density, f/cc x 10^{-20}	1.4
Power Density, kw/cc	1.1
Cover Gas Pressure, psia	9.2
Fission Gas Release, % Xe	3.3
Fuel Density, % of Theor,	
Pretest	95.9
Posttest	94.9
Specimen Diameter Change, in.	Nil
Fuel Diameter Change, in.	+0.001
Fuel Length Change, in.	Nil

A summary and status of the UN and UC inpile test program is shown as Fig 45.

Enriched pellets for inpile capsule test of UO_2 fuel have been fabricated. The 1000 hour test of UO_2-Cb compatibility sandwiches is in metallography. The 3000, 6000, 10,000 hour samples are still on test at 2200F.

3. Primary Pump

A total of eight preliminary design layouts of the primary pump and drive have been completed to date to study the mechanical problems involved. The basic concept of a single

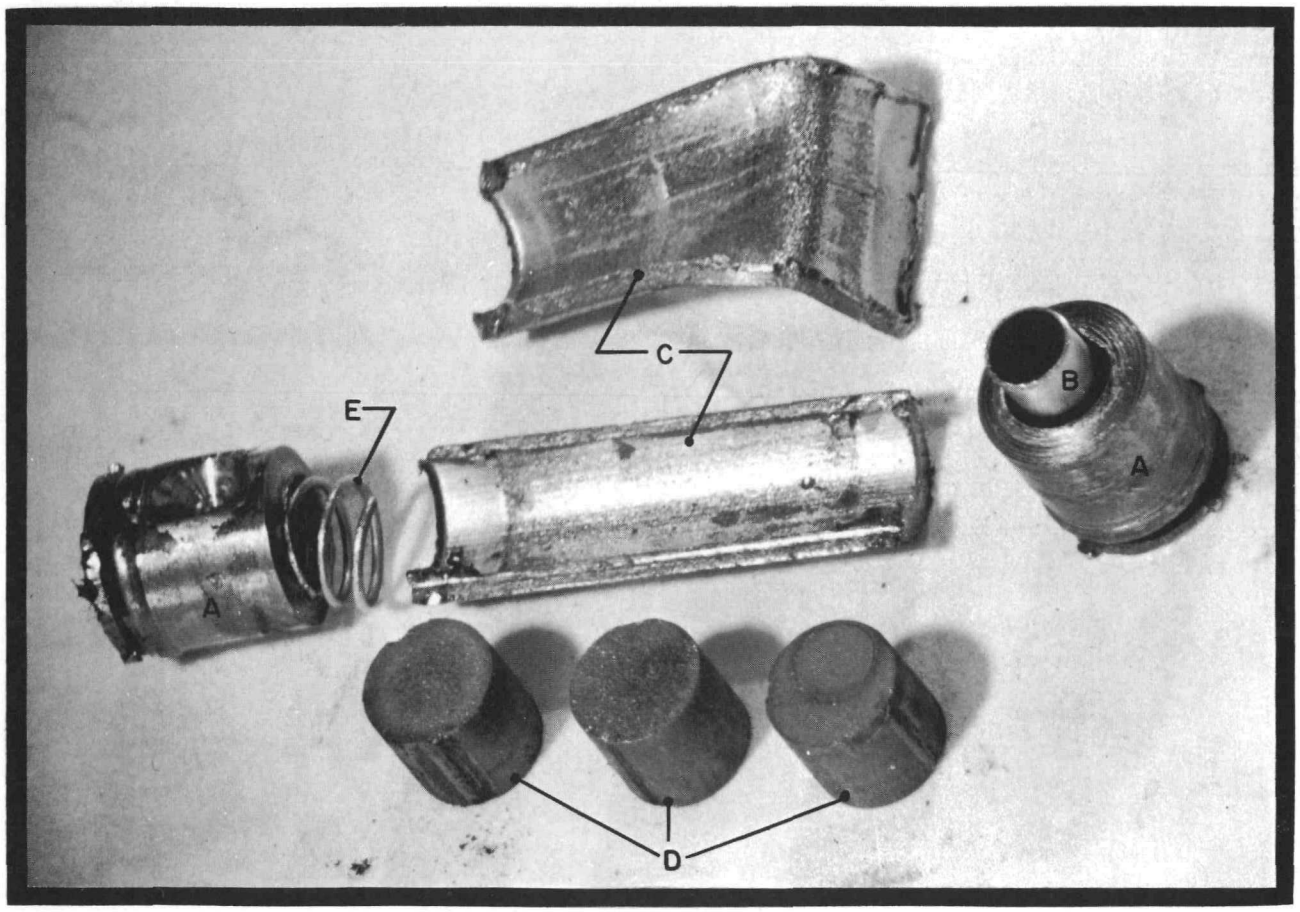
FIG 41

MIDDLE SPECIMEN, CAPSULE PW26-210, DISASSEMBLED

UNCLASSIFIED

LEGEND

A END CAPS
B TUBULAR SPACER
C CLADDING
D UN PELLETS
E SPRING SPACER



MAGNIFIED: APPROX. 3X

NOTE SHARP PELLET EDGES AND DUCTILITY OF CLADDING.

FIG 42

Cb-1 Zr ALLOY CLADDING, MIDDLE SPECIMEN, CAPSULE PW26-210-URANIUM NITRIDE

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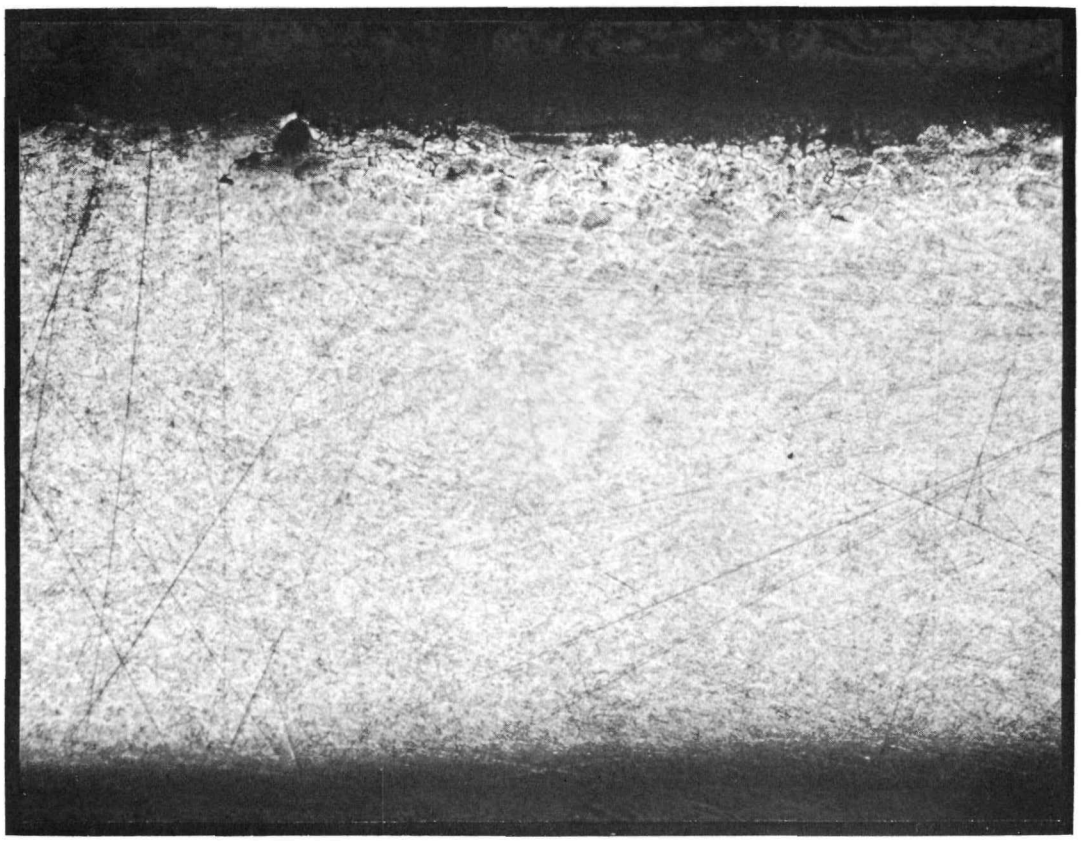

ETCHANT: 10 H_2O, 5 HNO_3, 2 HF

MAGNIFIED: 100X

SHOWING INTERGRANULAR REACTION AT CLADDING ID.

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SURFACE OF Cb-1 Zr ALLOY CLADDING, MIDDLE SPECIMEN
CAPSULE PW26-210-URANIUM NITRIDE

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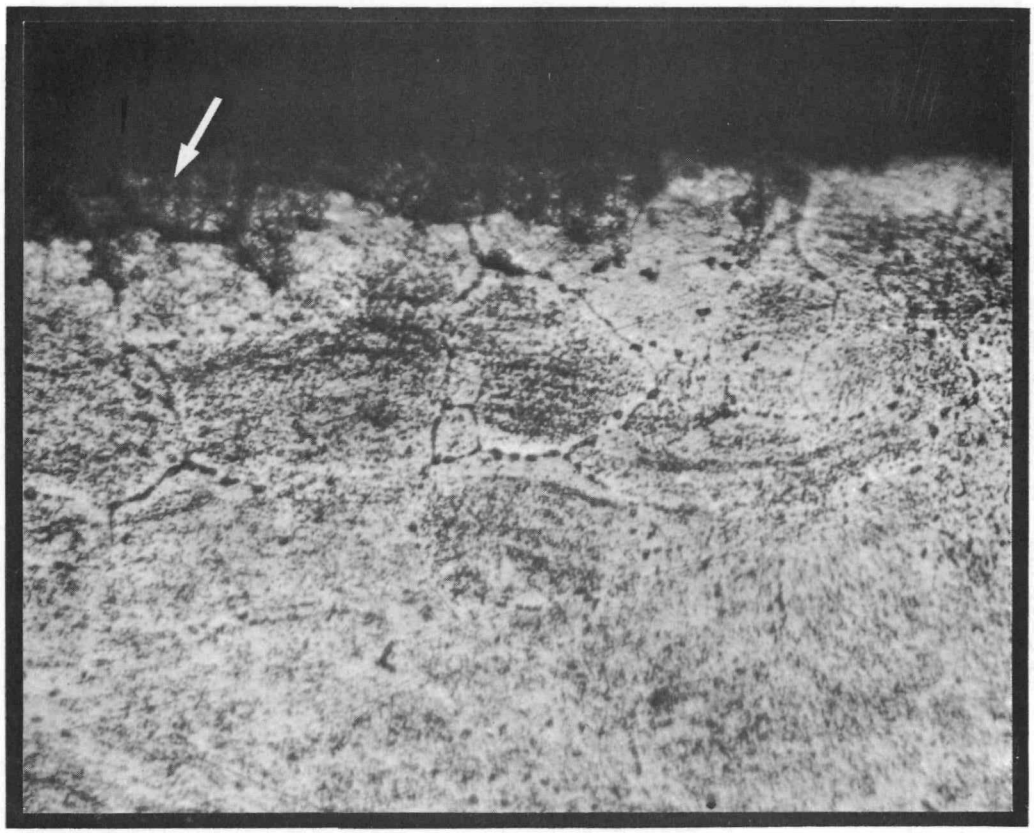

ETCHANT: 10 H_2O, 5 HNO_3, 2 HF MAGNIFIED: 500X

SHOWING PHASE BELIEVED TO CONTAIN URANIUM AT ID (ARROW) AND COLUMBIUM NITRIDE PHASE AT GRAIN BOUNDARIES.

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FIG 44

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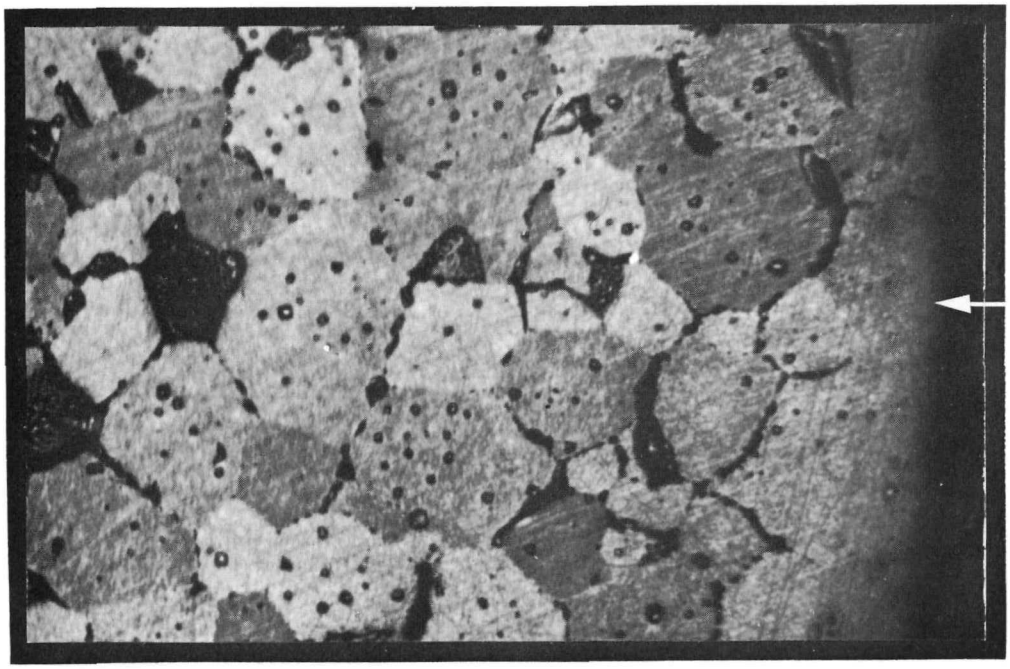
TOP FUEL PELLET, TOP SPECIMEN, CAPSULE PW26-210

URANIUM NITRIDE

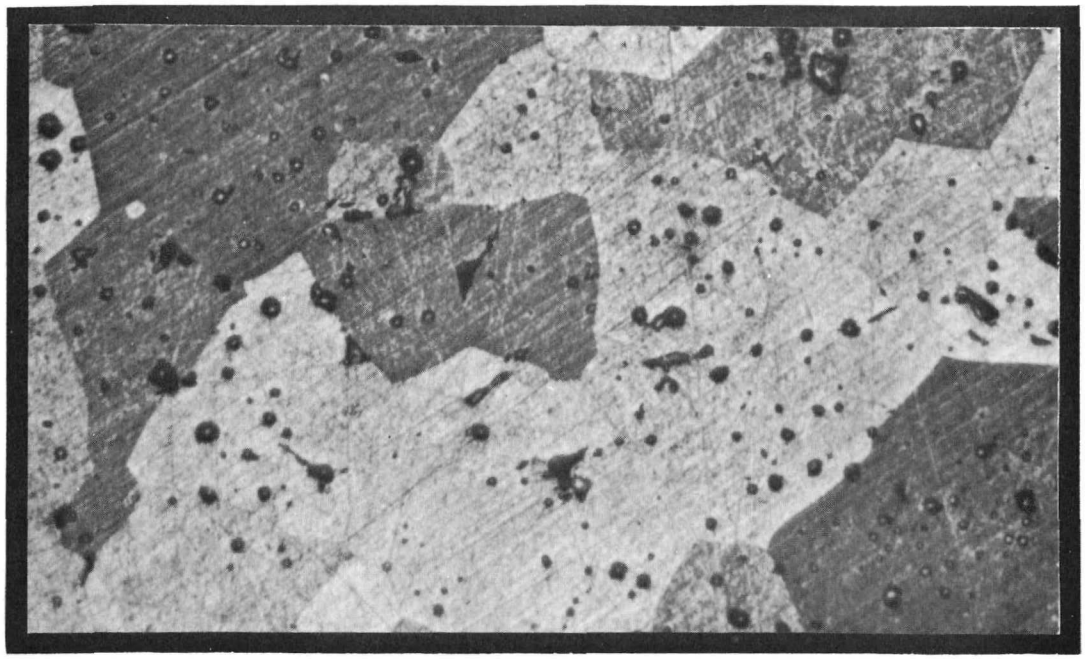


SURFACE

UNCLASSIFIED

ETCHANT: 30 $CH_3CHOHCOOH$, 10 HNO_3, 3 HF, 30 H_2O MAGNIFIED: 500X

A SHOWING FUEL MICROSTRUCTURE AT PELLET SURFACE FOLLOWING IRRADIATION.



ETCHANT: 30 $CH_3CHOHCOOH$, 10 HNO_3, 3 HF, 30 H_2O MAGNIFIED: 500X

B SHOWING FUEL MICROSTRUCTURE AT PELLET INTERIOR FOLLOWING IRRADIATION.

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SUMMARY OF CAPSULE IRRADIATION PROGRAM

Capsule Number	Duration of Irradiation, hr	Average Operating Temp of Specimen Surface, F	Test Facility Thermal Flux, nv	Cesium 137 / U^{235} Burnup, %	Fission Density Fissions/cc	Power Density kw/cc	% Xe Release	Cladding OD Change, mils	Cladding Reaction	Irradiation Dates
A. Pin type specimens, Cb-1 Zr alloy cladding (with Ta barrier) and end caps, 0.296 OD x 0.025 wall cladding, UC matrix 10% enriched, 0.244 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200F design										
PW26-150	3,000*	2108**	6.2×10^{13}	14.7*						11/62 to 5/63*
PW26-151	1,097	2175	7.0×10^{13}	6.6	2×10^{20}	1.6	7.5	+5.0	Nil	12/62 to 2/63
PW26-152	10,000*	2200*	7.0×10^{13}	40.0*						3/63 to 8/64*
PW26-153	10,000*	2200*	6.2×10^{13}	40.0*						4/63 to 10/64*
PW26-154	10,000*	2200*	6.2×10^{13}	40.0*						6/63 to 12/64*
B. Pin type specimens, Cb-1 Zr alloy cladding (with Ta barrier) and end caps, 0.296 OD x 0.025 wall cladding, UC matrix of alloy metal 10% enriched, 0.244 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200Fdesign										
PW26-160	3,000*	2200*	6.2×10^{13}*							4/63 to 10/63
C. Pin type specimens, Cb-1 Zr alloy cladding and end caps, 0.296 OD x 0.025 wall cladding, UC matrix 10% enriched, 0.244 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200F design										
PW26-170	3,000*	2191	6.5×10^{13}	14.7*						1/63 to 6/63*
PW26-171	892	2135	7.0×10^{13}	5.2*				+2.5		1/63 to 3/63
PW26-173	10,000*	2200*	6.3×10^{13}	40.0*						3/63 to 9/64*
PW26-174	10,000*	2200*	6.2×10^{13}	40.0*						4/63 to 10/64*
D. Pin type specimens, Cb-1 Zr alloy cladding (with W barrier) and end caps, 0.296 OD x 0.025 wall cladding, UC matrix of alloy metal 10% enriched, 0.242 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200F design										
PW26-181	3,000*	2200*	6.2×10^{13}*	14.7*						5/63 to 11/63
E. Pin type specimens, Cb-1 Zr alloy cladding (with W barrier) and end caps, 0.296 OD x 0.025 wall cladding, UC matrix 10% enriched, 0.242 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200F design										
PW26-190	1,069	2055	6.2×10^{13}	6.8	2.1×10^{20}	1.7	76.0	+1.0	Intermittent	
PW26-191	1,830	2103	6.5×10^{13}	9.4*						11/62 to 3/63
PW26-192	10,000*	2200*	6.5×10^{13}	40.0*						3/63 to 9/64*
PW26-195	10,000*	2200*	6.2×10^{13}	40.0*						6/63 to 12/64*
PW26-196	10,000*	2200*	6.2×10^{13}	40.0*						6/63 to 12/64*
F. Pin type specimens, Cb-1 Zr alloy cladding and end caps, 0.309 OD x 0.035 wall cladding, UN matrix 31.5% enriched, 0.237 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200F design										
PW26-210	1,092	2119	3.0×10^{13}	1.3	1.4×10^{20}	1.1	3.3	Nil	Spotty	10/62 to 12/62
G. Pin type specimens, Cb-1 Zr alloy cladding and end caps, 0.309 OD x 0.035 wall cladding, UN matrix 10% enriched, 0.237 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200F design										
PS26-220	3,000*	2213**	6.3×10^{13}	14.0*						1/63 to 6/63*
H. Pin type specimens, Cb-1 Zr alloy cladding (with W barrier) and end caps, 0.312 OD x 0.035 wall cladding, UN matrix 10% enriched, 0.237 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200F design										
PW26-230	853	2215	6.2×10^{13}	5.0*						1/63 to 3/63
PW26-231	3,000*	2200*	7.1×10^{13}	14.0*						2/63 to 8/63*
I. Pin type specimens, Cb-1 Zr alloy cladding (with W barrier) and end caps, 0.296 OD x 0.025 wall cladding, UN matrix 10% enriched, 0.241 inch D x 0.65 inch long pellet height, 1.5 kw/cc in Li at 2200F design										
PW26-240	10,000*	2200*	5.7×10^{13}*	38.2*						4/63* to 10/64*
PW26-241	10,000*	2200*	5.7×10^{13}*	38.2*						4/63* to 10/64*
PW26-242		2200*	5.7×10^{13}*							
PW26-243		2200*	5.7×10^{13}*							
PW26-244		2200*	5.7×10^{13}*							

*Design value
**Average to date

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stage centrifugal pump directly coupled to a canned electric motor drive has been adopted to insure absolute containment of the lithium and to provide a multiple start capability. A drawing of a pump and drive configuration, based upon detailed sizing of the various components, is shown in Fig 46. The drive motor for this configuration is straddled by two fluid film journal bearings. Internal cooling is accomplished by the lithium bearing lubricant flow, which transfers the heat to an auxiliary circuit through the heat exchanger attached to the motor casing. Other preliminary design layouts have investigated the feasibility of other shaft bearing arrangements and different types of cooling circuits.

The design of the canned electric motor is being obtained on subcontract with two companies. These subcontracts will include parametric studies and the design of motors at two different power levels.

Fabrication of the rotating disk test unit for cavitation studies has continued and procurement of the inert atmosphere test chamber has been initiated. A view of the partially machined Cb-1 Zr alloy bottom cover for the disk housing is shown in Fig 47. Bar and sheet stock of all refractory metal alloys except the Cb-18 W-8 Hf has been received and test specimens are in process of preparation for both water and hot lithium tests. Metallurgical tests will be conducted on specimens from identical heats of materials as for the disk specimens to provide data on short term tensile, impact, flexural fatigue and hardness at temperatures in the range of 1900F.

Tests of the rotating disks in the water test facility have continued. A plot of the damage data for aluminum and stainless steel specimens as a function of time is presented in Fig 48.

The drawing of the full scale primary pump impeller test unit is shown in Fig 49. All of the tapes have been produced to machine this impeller at CANEL on the tape-controlled milling machine from a contoured blank of stainless steel. The impeller will be tested in water to determine the complete hydraulic performance, including visual studies of cavitation formations at various values of net positive suction head. Testing is scheduled to begin during the next quarter.

The design of the water rotor dynamics test unit, shown in Fig 50, has been completed and is in process of fabrication. This unit will permit studies of bearing instability and other rotor dynamics behavior to be conducted on various mock-ups of primary pump configurations with the simulated mass of the drive motor overhung from the shaft support bearings. A similar type of water test unit with the mass of the motor straddled between the journal bearings is also in procurement. Both test units, will be driven by a small two stage air turbine through a quill shaft arrangement.

The assembly of the water hydrostatic journal bearing test unit is in progress. The finished shaft for this test unit is shown in Fig 51. The test journal is integral with the shaft and was produced with a concentricity not exceeding 0.0003-inch relative to the anti-friction support bearing shoulders. This will be driven by a single partial admission air turbine which has been calibrated on the test stand shown in Fig 52.

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FIG 46

SNAP-50 PRIMARY COOLANT PUMP

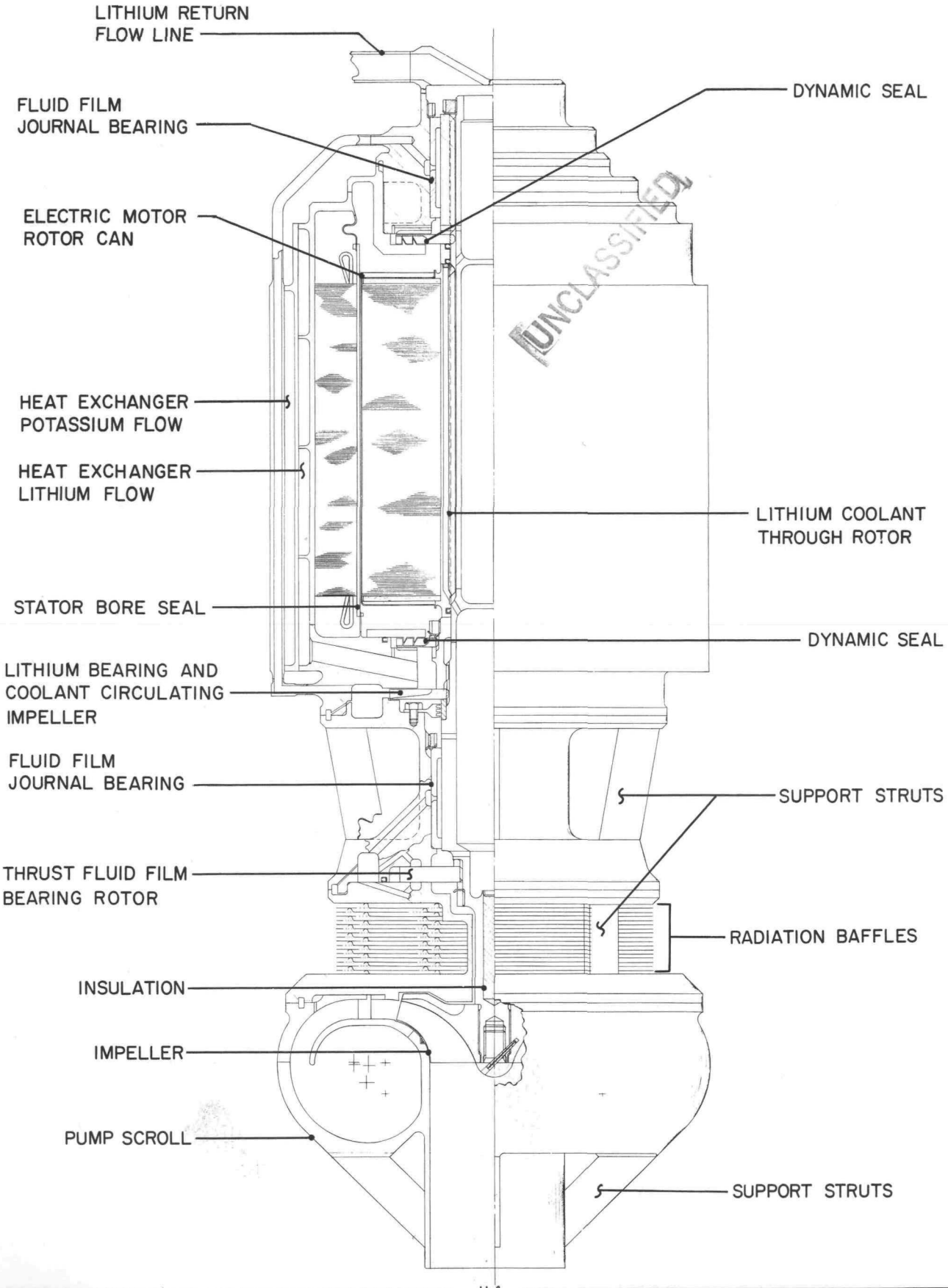

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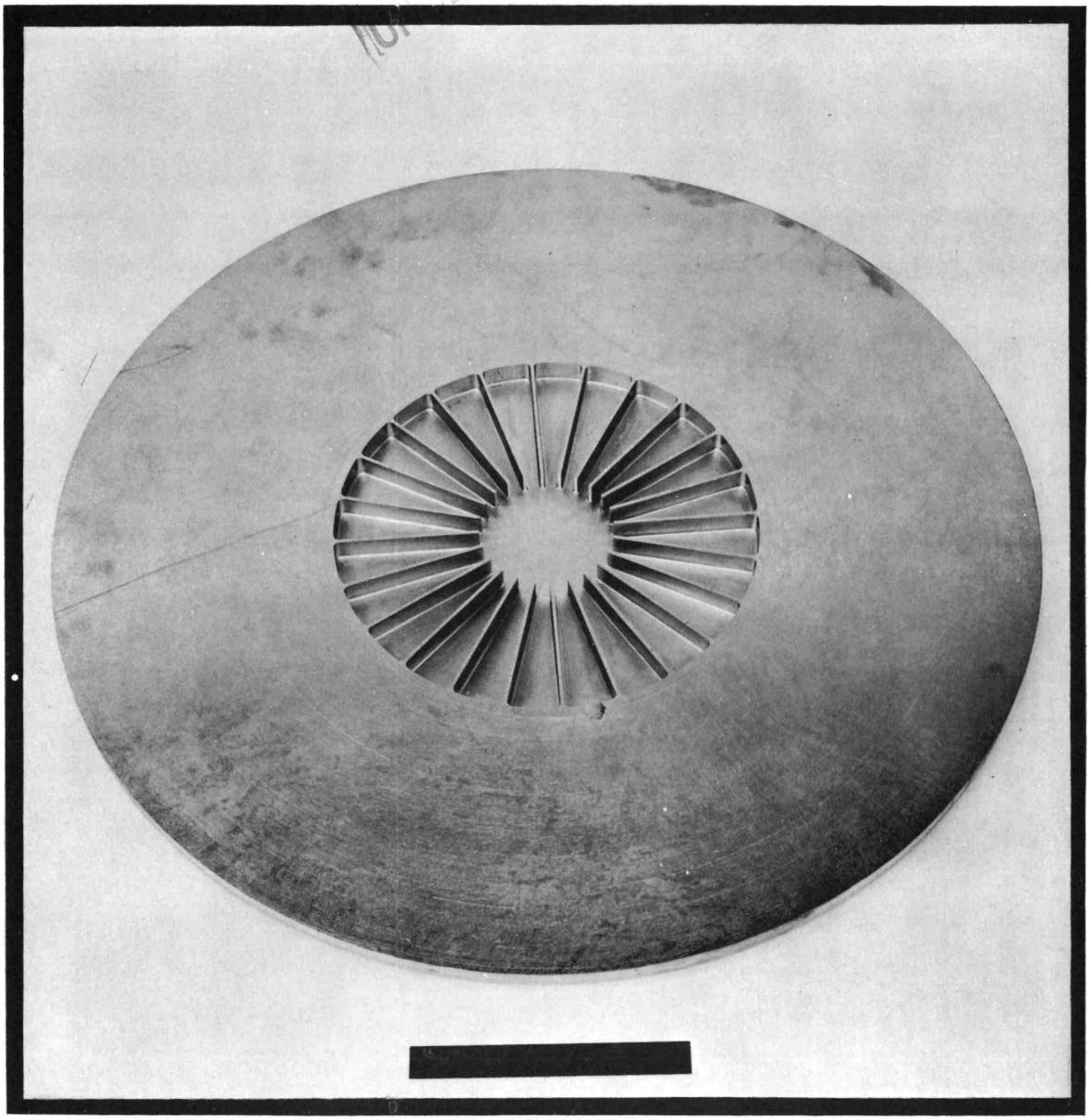
FIG 47

PARTIALLY MACHINED Cb-1 Zr BOTTOM COVER FOR ROTATING DISK HOUSING



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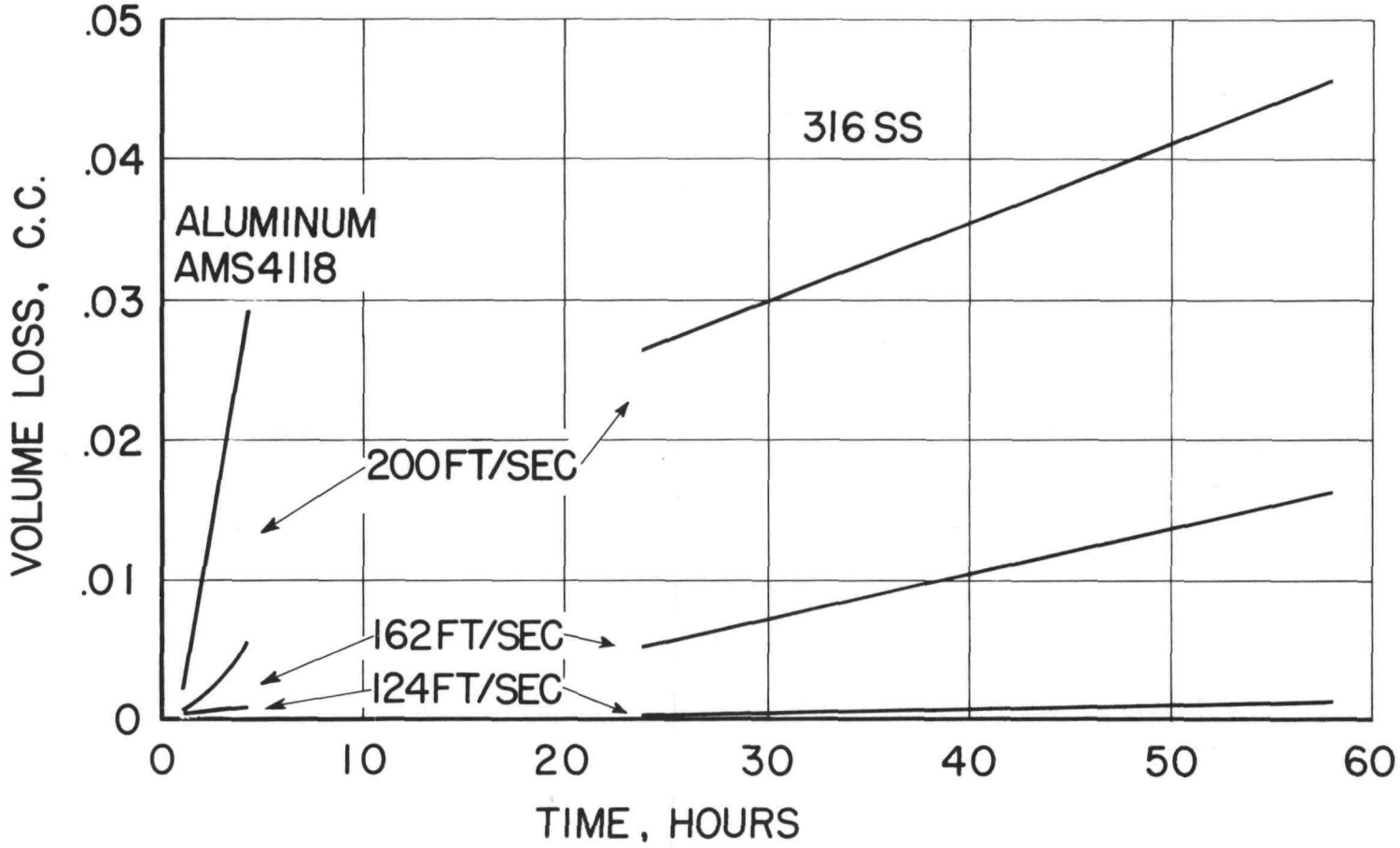
VOLUME DAMAGE RATE

SHAFT SPEED - 6000 RPM

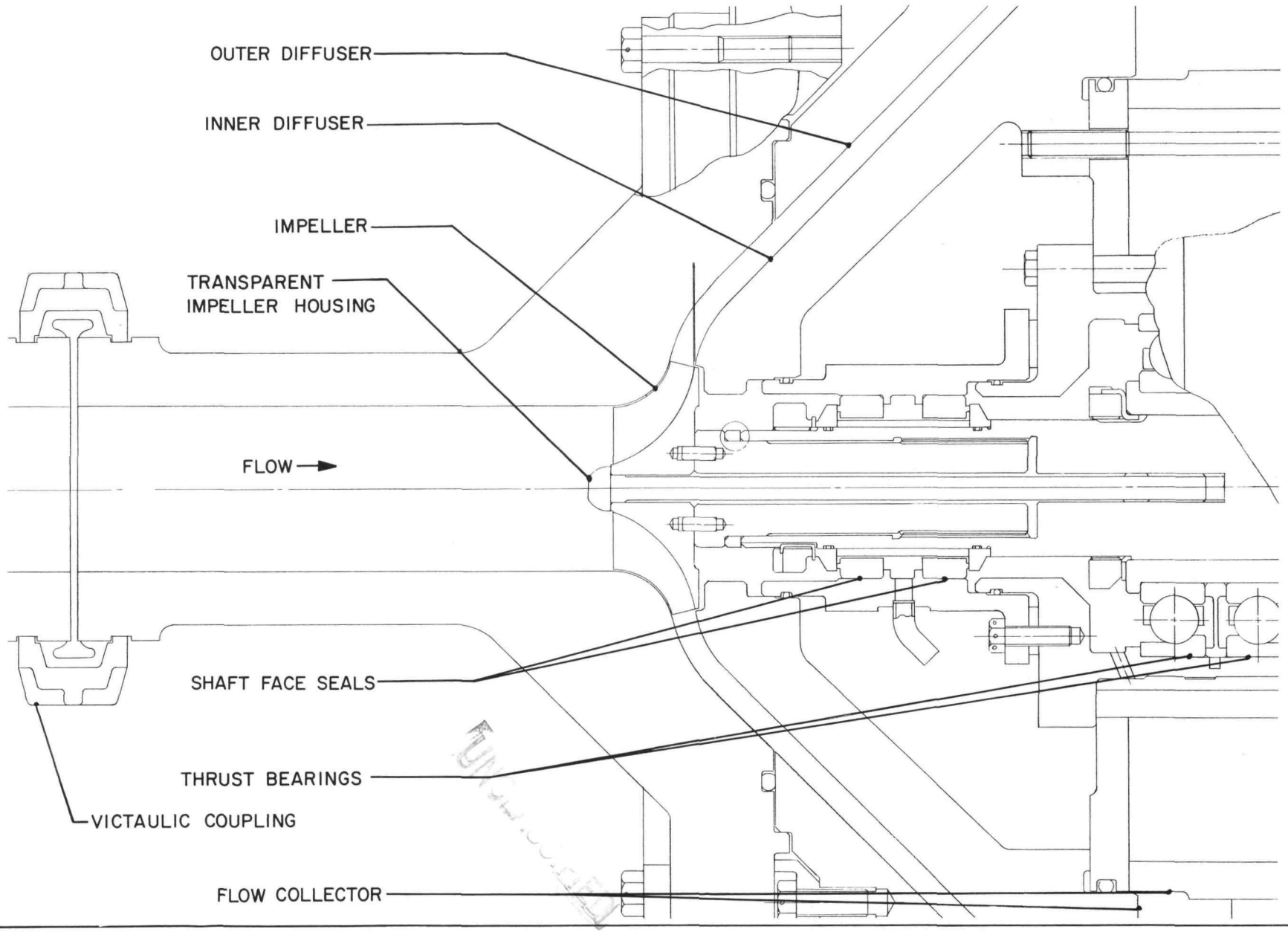
POT PRESSURE - 10 PSIG



PWAC - 631

FIG 48

RI-16 IMPELLER TEST UNIT



ROTOR DYNAMIC TEST UNIT FOR HYDROSTATIC BEARINGS WITH OVERHUNG LOAD

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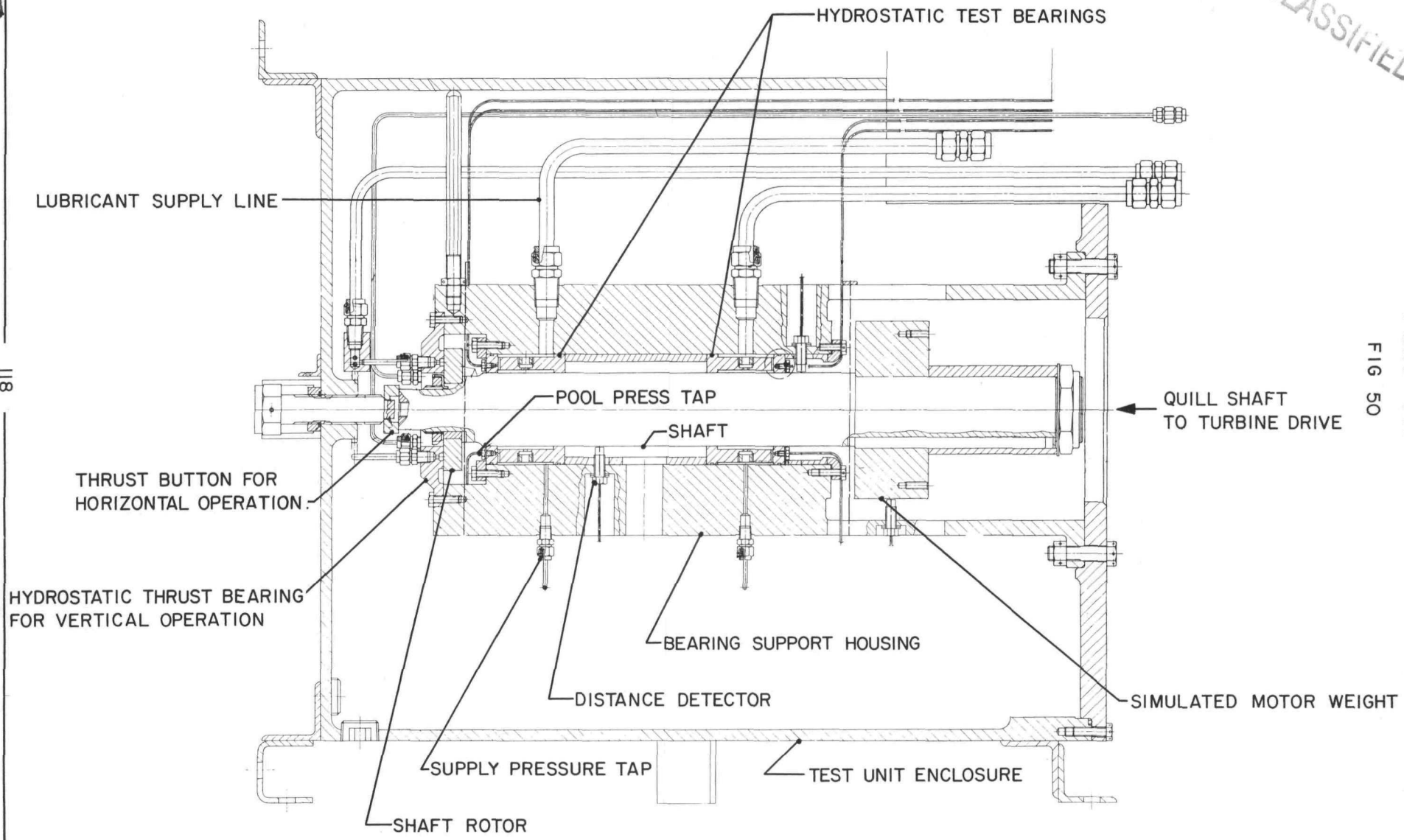
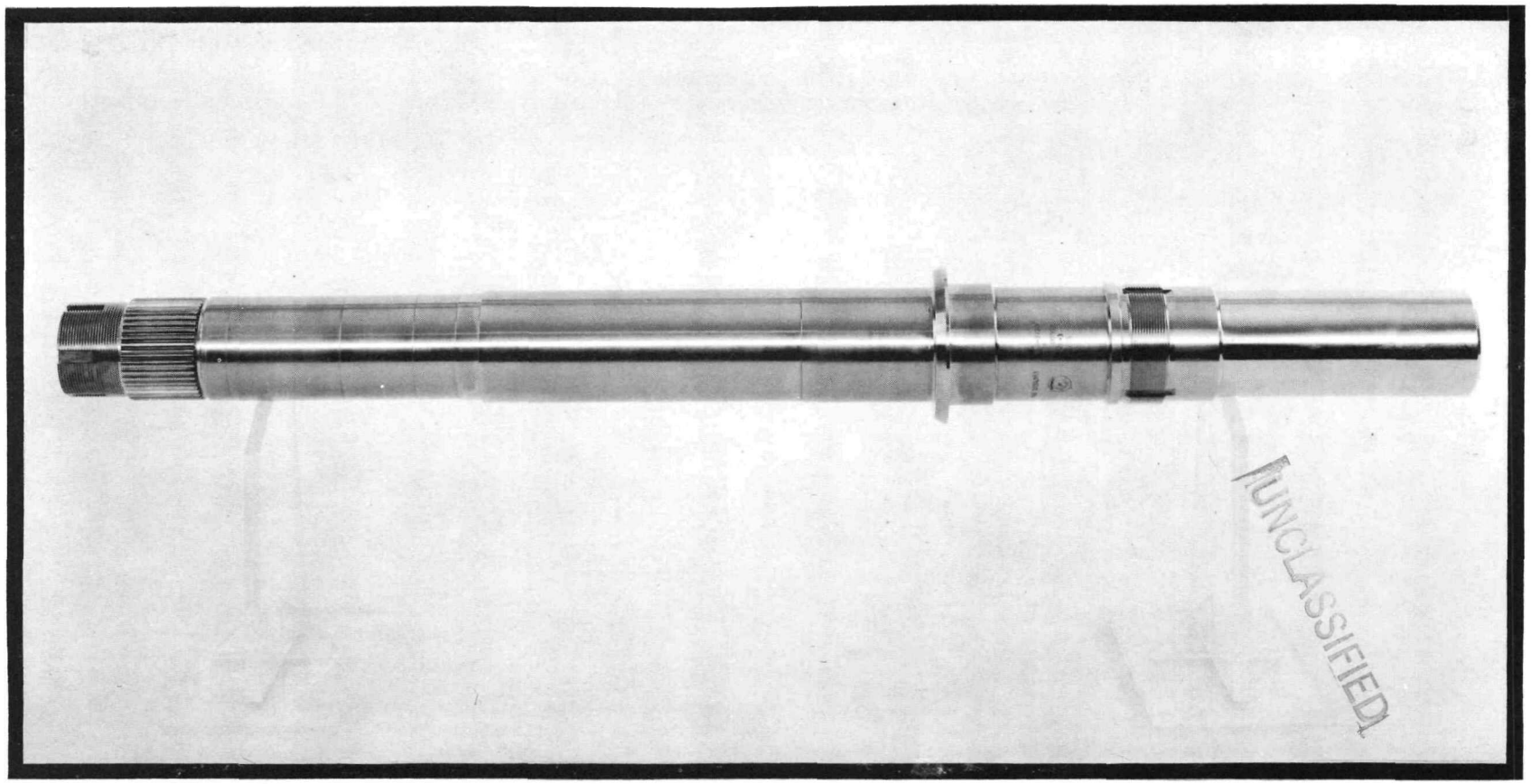


PWAC-631
FIG 50

FINISHED SHAFT FOR WATER HYDROSTATIC BEARING TEST UNIT



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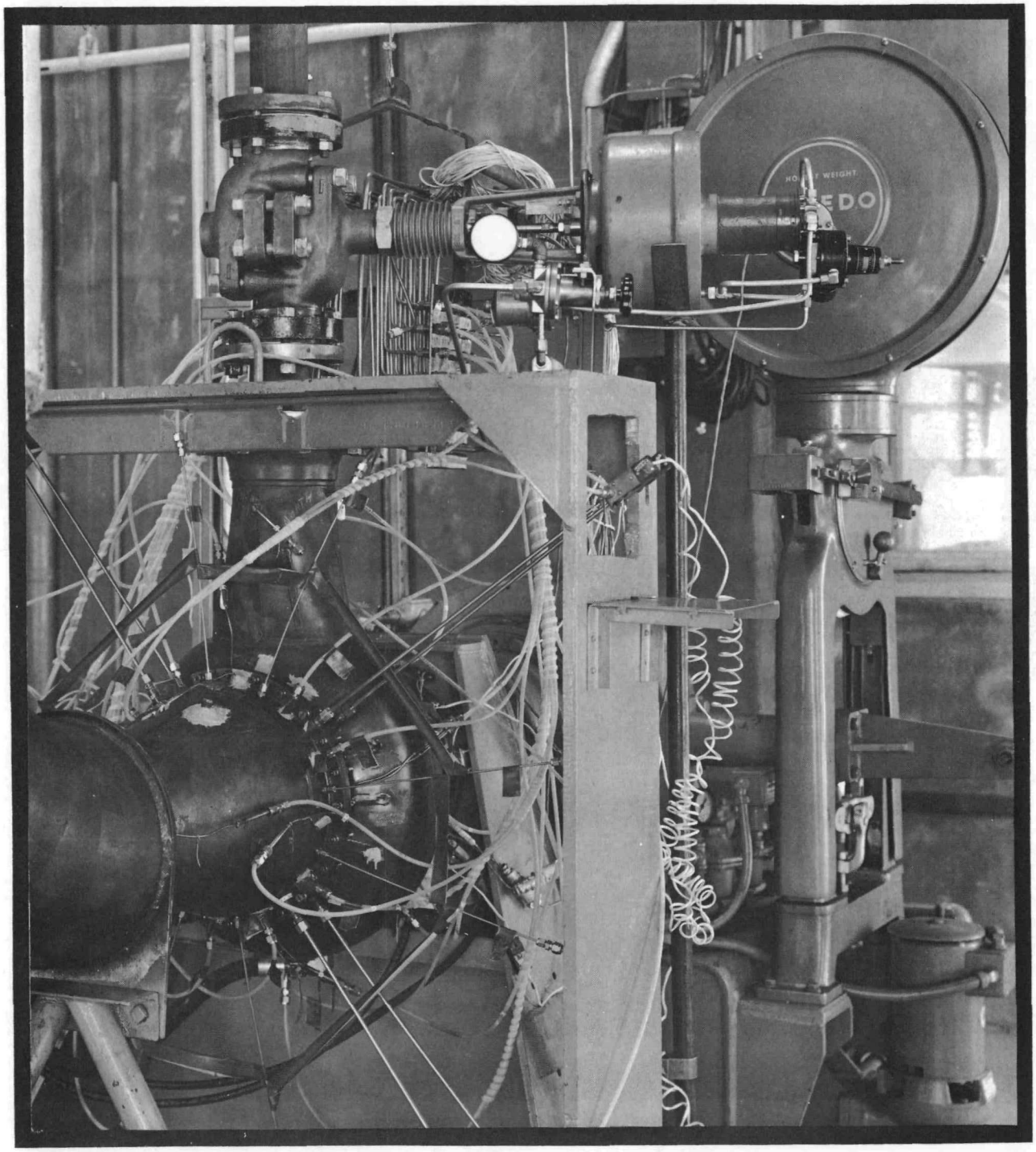
CALIBRATION TESTS OF PARTIAL ADMISSION AIR TURBINE DRIVE



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The first test of the lithium hydrodynamic radial bearing test unit (Fig 53) was started and a total of 225 hours of continuous operation at light loads has been completed. This test is being conducted at a shaft speed of 5000 rpm in 600F lithium with an imposed radial load of about 30 pounds.

A view of the lithium-to-vacuum seal test unit is shown in Fig 54, during assembly and in Fig 55 after installation in the dry box. The first attempt to run this unit was terminated at start-up by an accidental flooding of lithium. Upon disassembly, it was found that the end convolution of the shaft face-seal bellows had ruptured upon contact with the lithium. It is suspected that this was due to a faulty weld. The unit was reassembled and during dry running in the dry box a seizure occurred between the dynamic seal and the housing. The unit is being rebalanced and reassembled.

4. Boiler

Shop fabrication of the Haynes 25 alloy 1800F boiling potassium heat transfer test loop is 95 percent complete; the test stand is nearing completion. Installation of the loop into the test stand has begun. A photograph of the three-boiler assembly prior to instrumentation and lagging is shown in Fig 56. A photograph of the center boiler is shown in Fig 57.

A single test boiler section was fabricated and tested with boiling potassium at 1800F to determine creep of the containment vessel and burnout heat flux on the Haynes 25 alloy clad immersion heater. Creep at 80 psia was negligible in 200 hours; the heater failed at the cladding outside diameter at a flux of approximately 50,000 Btu/ft^2 hr. A photograph of this boiler after test is shown in Fig 58.

Two-phase operation was initiated on the stainless steel boiling potassium loop, NKSS-A during the quarter. Approximately 100 hours of boiling operation were accomplished at sodium loop temperatures to 1600F, potassium vapor temperatures to 1580F and potassium flow rates to 16 lb/hr. An average heat flux of 30,000 Btu/ft^2 hr was attained, with local heat fluxes of approximately 100,000 Btu/ft^2 hr. One hundred percent quality vapor was produced at stable conditions. Data analysis using the Bendix computer is underway at the present time.

The system was unstable at the higher heat fluxes and potassium flow rates. In an effort to improve stability, the potassium loop was drained, cut open, and an additional 0.027-inch-diameter orifice and pressure transmitter installed upstream of the boiler. After considerable trouble caused by plugging in the potassium system, two phase operation was re-established. System stability will be investigated along with further heat transfer experimentation.

A second stainless steel loop, designated as NKSS-B, is approximately 75 percent complete and will be installed when the useful life of NKSS-A is ended. This loop incorporates several features to improve stability, heat transfer capacity, and data accuracy. These include increased sodium flow capacity, decreased pressure drop in the potassium vapor region, a smaller orifice, an additional pressure transmitter, improved thermocouple installation and a more flexible condenser cooling method. The test boiler tube on NKSS-B will consist of a 3/8 inch outside diameter tube, five feet long, with the liner insert shown in Fig 59.

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FIG 53

LITHIUM HYDRODYNAMIC BEARING TEST STAND

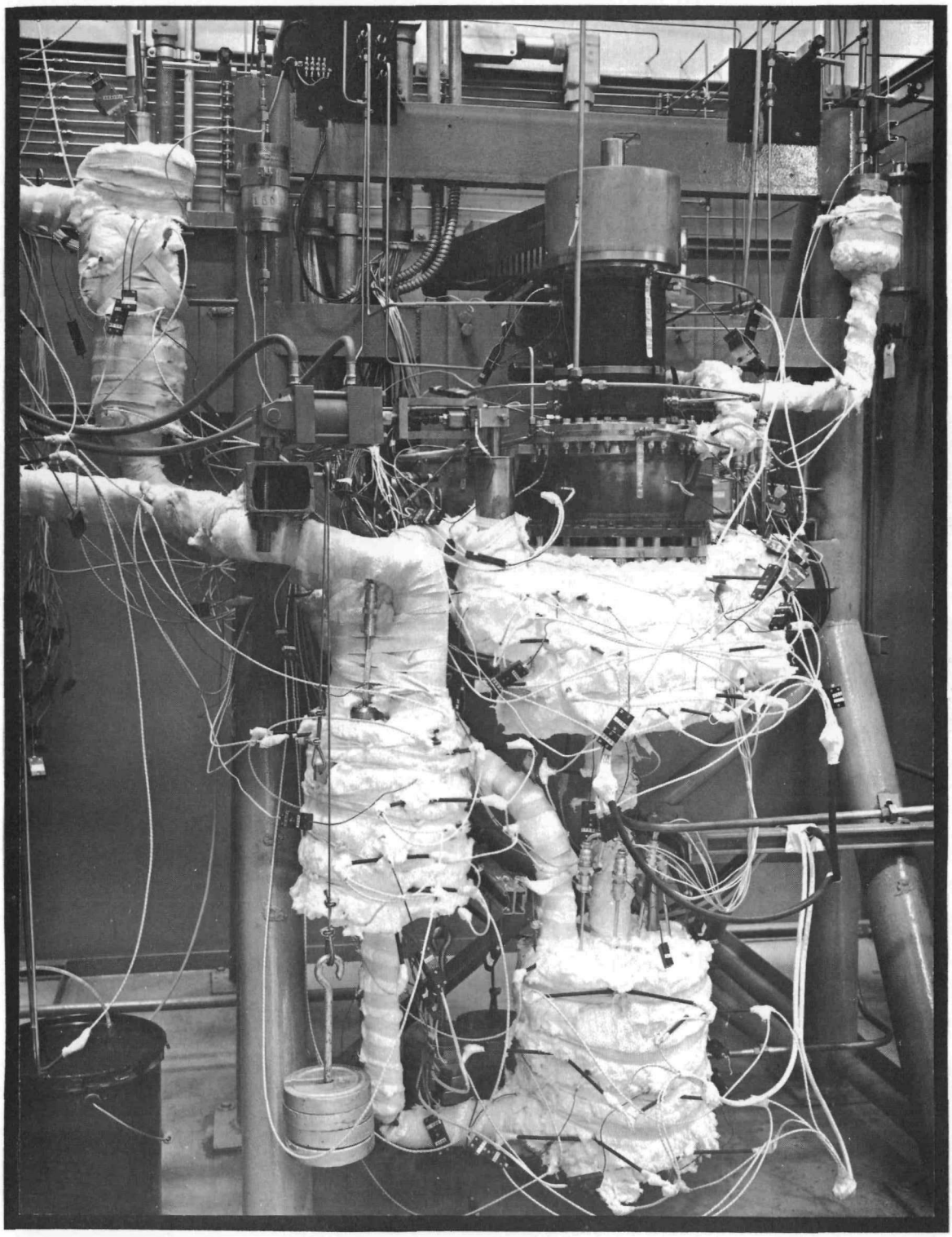

FIG 54

LITHIUM TO VACUUM SEAL TEST UNIT

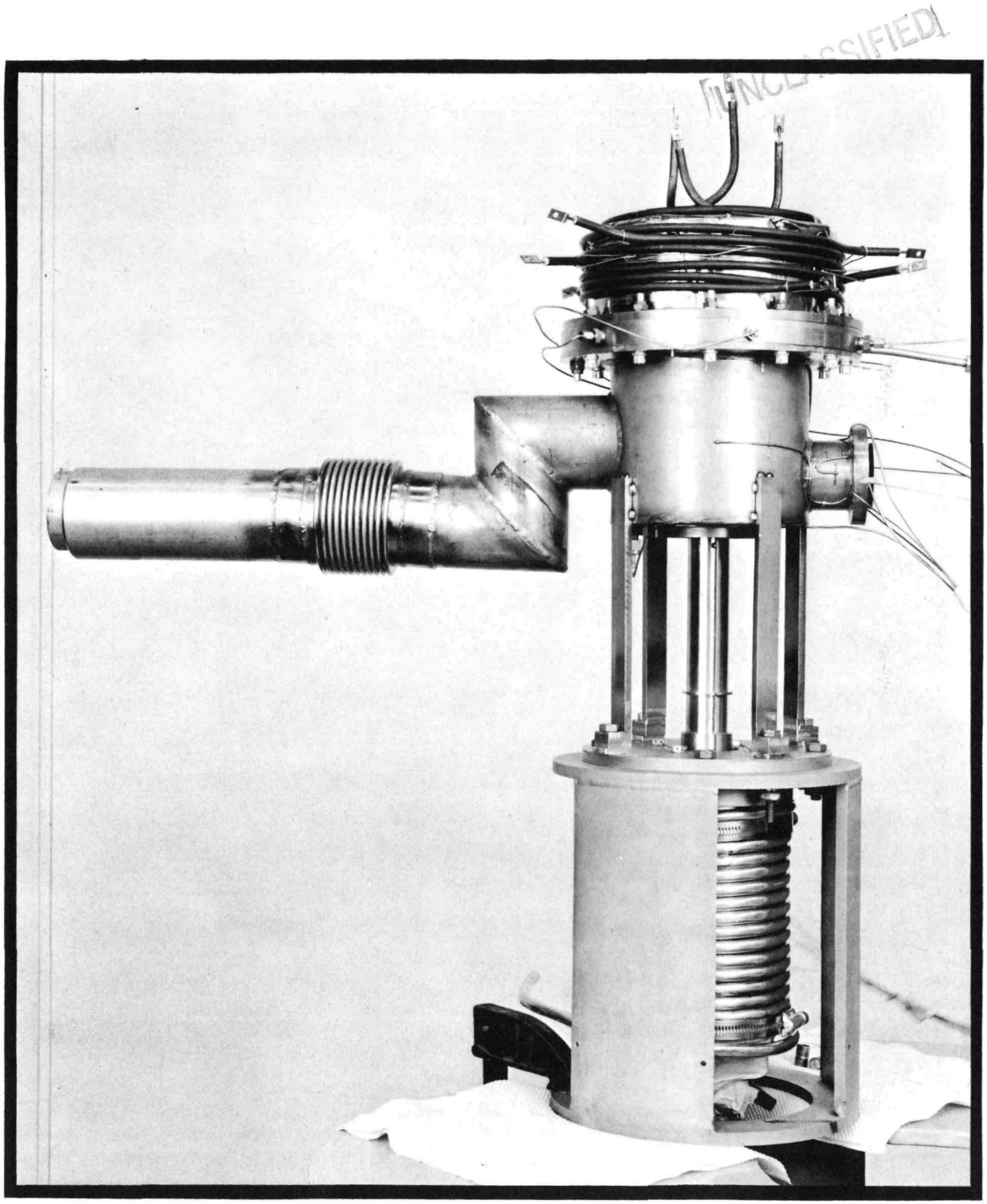

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FIG 55

INSTALLATION OF LITHIUM TO VACUUM SEAL TEST UNIT

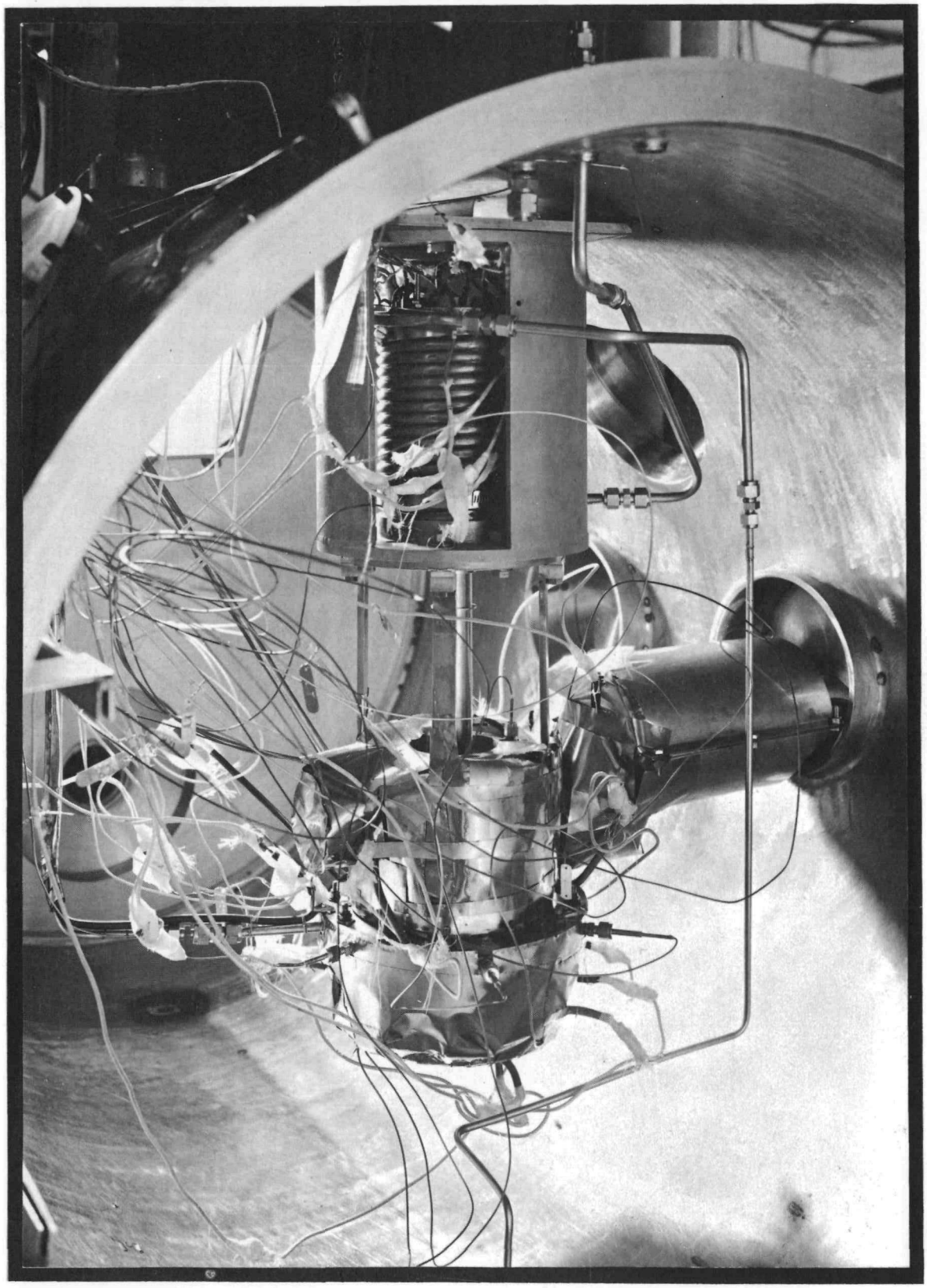

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FIG 56

THREE-BOILER ASSEMBLY

(TO BE OPERATED IN A HORIZONTAL POSITION)

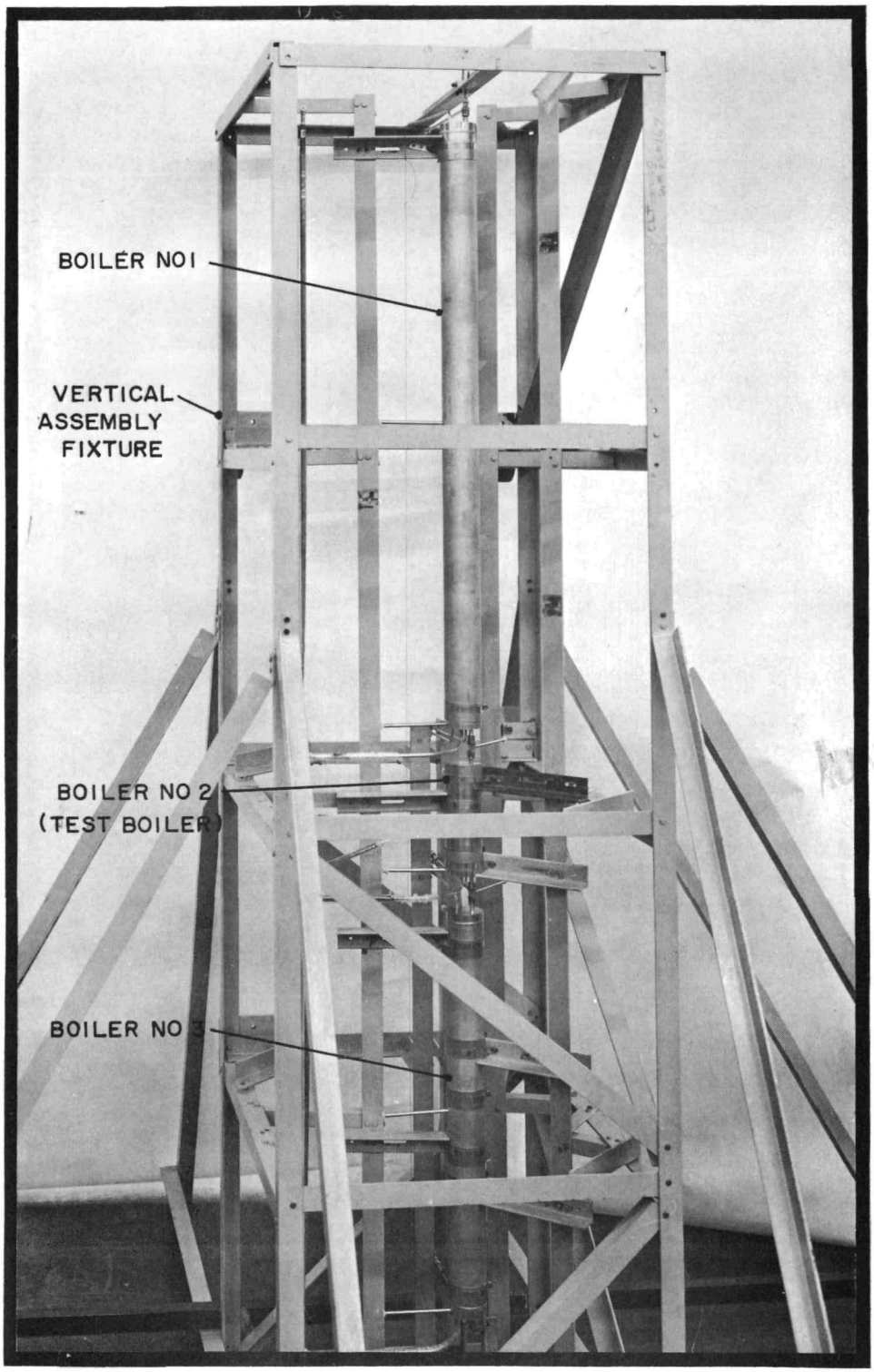

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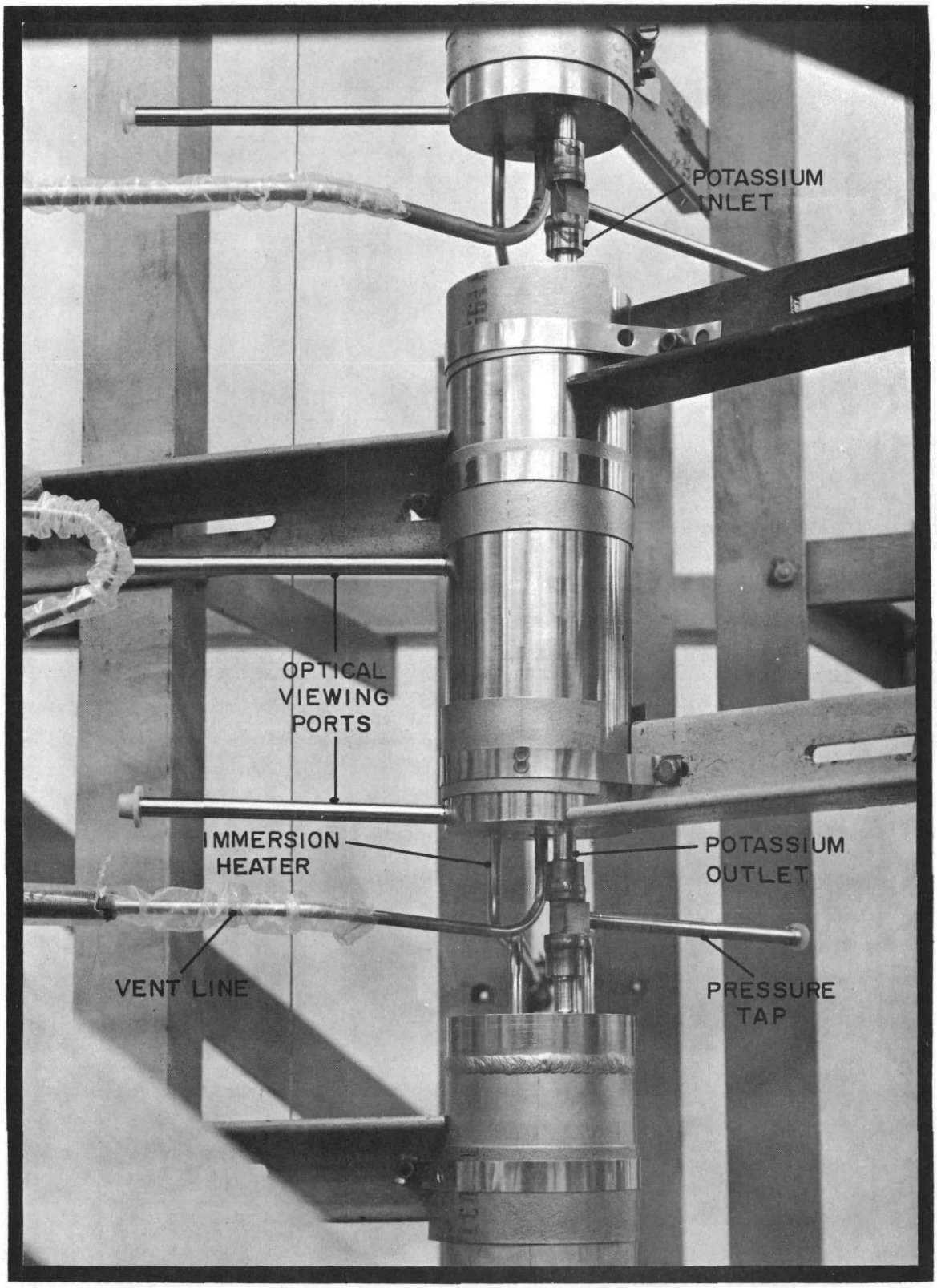
FIG 57

CENTER TEST BOILER



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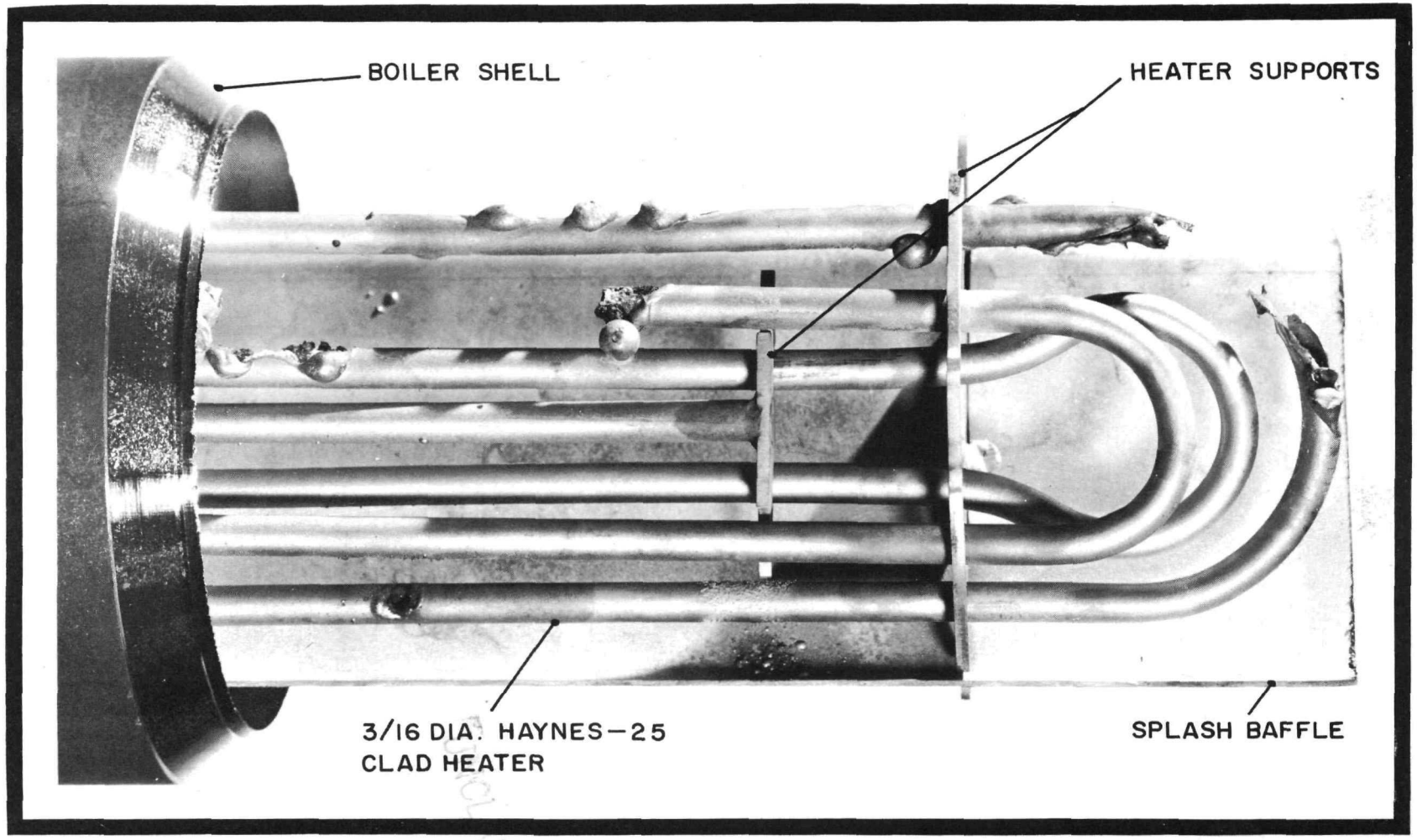
SINGLE BOILER AFTER TEST

(HEATER POWER DELIBERATELY INCREASED TO CAUSE BURNOUT)



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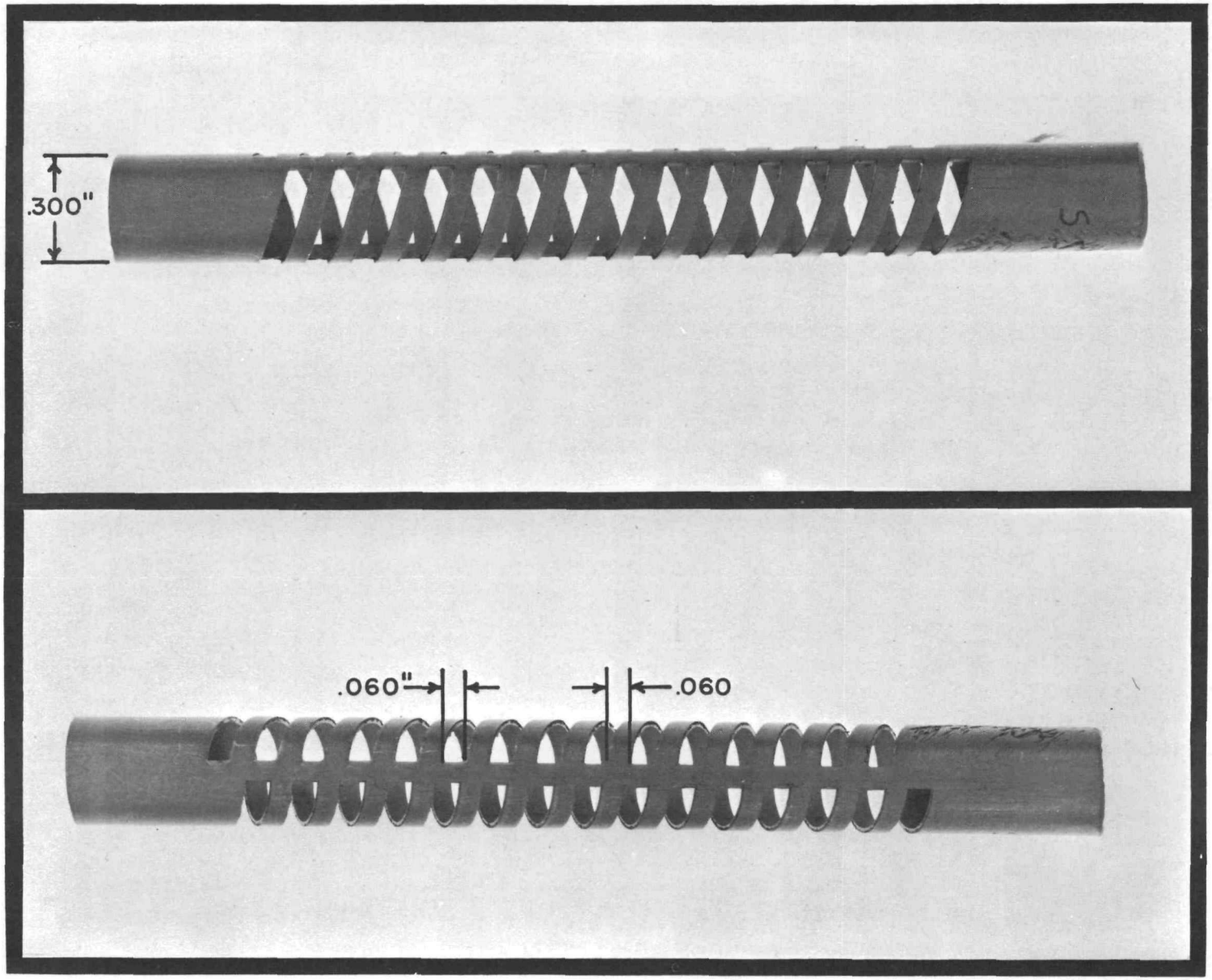
TEST BOILER TUBE INSERT - NKSS-B



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Two phase water heat transfer and pressure drop testing of various tube geometries was continued at heat fluxes to 60,000 Btu/ft^2 hr. Geometries investigated include straight and sine wave tubes, and tubes with internal helical fins and ribbon inserts. Plots of heat transfer coefficient versus quality for these tubes is shown in Fig 60.

A sample of columbium alloy, internally finned tubing was obtained and efforts are being made to obtain liquid metal quality tubing in this geometry. The general effort to develop sources of boiler tubing with internal parts of copper, stainless steel and columbium alloy was continued.

Designs of 80 and 100 percent vaporization boilers were completed. Preliminary studies were begun on a test system for a 19 tube module of the SNAP-50 boiler design.

5. Materials Development

Compatibility tests and evaluations of prospective bearing materials in lithium were continued, but have been limited to 1000F tests. Posttest evaluation was completed after 500 and 2000 hour tests of a hot-pressed cermet material whose composition is described in patent application 113565. Compared with pretest specimens, there was no evidence of change in argon tested or lithium tested specimens when examined by X-ray, metallographic inspection or posttest weight loss. Most pretest specimens had fine surface cracks that extended intergranularly to an estimated depth of 8 mils. These surface defects also did not show any posttest change. Hardness measurements listed below indicate that some slight hardening of the metal matrix is occurring as a function of time, probably through diffusion of a ceramic component into the metal. Long-time testing of this material is being continued to determine if the compatibility remains good under these conditions.

Time	Test Media	Hardness Range, DPH	
		Metal Matrix	Ceramic Areas
Pretest	None	475-528	1003-1620
500 hours	Lithium and Argon	475-528	1003-1620
2000 hours	Lithium and Argon	665-907	1003-1682

Methods for surface carburizing of columbium alloys were studied in an effort to produce carbide coatings up to one mil in thickness which will resist abrasion. Heating Cb-1 Zr alloy parts in contact with lithium-carbon solutions was known to yield columbium carbide surface layers. Data for a series of experiments performed to define the more important process variables are summarized below:

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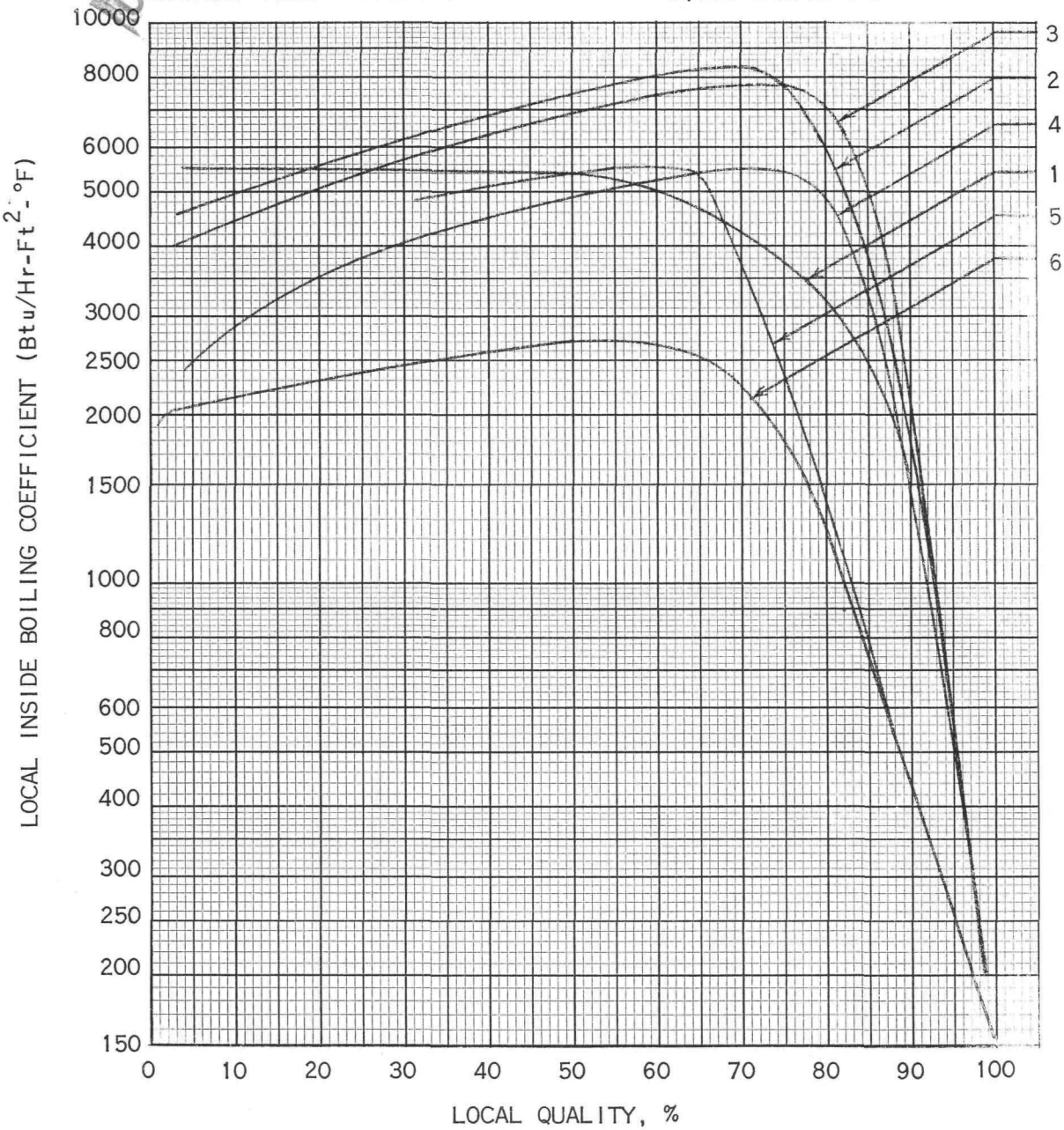
FIG 60

FORCED CONNECTION BOILING HEAT TRANSFER COEFFICIENTS FOR VARIOUS TUBE GEOMETRIES

BOILING PRESSURE 5 PSIG
WATER MASS FLOW RATE 26,850 LB/FT^2 HR

		HEAT FLUX AT TUBE ID
SINE WAVE TUBE - 0.190 ID	1	55,000 Btu/Hr-Ft^2
	2	25,000 Btu/Hr-Ft^2
	3	6,300 Btu/Hr-Ft^2
TWISTED RIBBON - 0.180 ID	4	6,300 Btu/Hr-Ft^2
STRAIGHT TUBE - 0.117 ID	5	6,300 Btu/Hr-Ft^2
STRAIGHT TUBE - 0.190 ID	6	6,300 Btu/Hr-Ft^2



Carbide Case Development by Reaction in Lithium Solutions

Lithium Melt Composition (parts by weight)	Temp, F	Time, hr	Case Depth, mils	Total Carbon full wall, ppm	Percent Total Carbon in 1st 2 mils	Remarks
50Li, 5C	1700	36	0.04	--	--	
50Li, 5C	1700	72	0.1	--	--	
50Li, 5WC, 0.03C	1700	100	0.12	1200	83	*
50Li, 5WC, 0.03C	1950	100	0.17	2600	84	*
50Li, 5WC, 0.03C	2200	100	0.7 outer 0.8 inner	9500	92	*Outer layer- 1100 DPH
50Li, 2.5MoC, 0.03C	2200	100	0.6 outer 0.4 inner	7000	90	*
50Li, 1.5VC, 0.03C	2200	100	0.5 outer 0.5 inner	6500	90	*
50Li, 5WC, 0.03C	1950	250	0.15 outer 0.1 inner	3400	78	*
50Li, 5C	2200	100	0.5 outer 0.8 inner	8600	92	*Outer layer- 800 DPH

*Fine, spheroidal carbide phase distributed throughout matrix

Since carbide layer microhardness is believed to be more important in this application than is thickness, the conditions shown for the 50 Li, 5 WC, 0.03 C; 2200F; 100 hours test were chosen for evaluation in bearing material screening tests. Specimens are being case hardened at this time.

An alternate carburizing method has been developed which involves thermal reaction of a film of graphite applied by spray coating one side of 30-mil thick, Cb-1 Zr alloy specimens with Dag Dispersion 154. A wide range of heat treatments was investigated in order to define the best condition for the case hardening of small parts for which extra-high temperature furnace facilities are available. Large sections must be restricted to a 2200F heat treatment because of furnace size limitations. The development of case thickness and extent of internal carbide precipitation for several graphite coating weights under varied heat treatment conditions is tabulated below:

Carbide Case Development by Reaction with Graphite Coatings

Coating Weight (mg/cm^2)	Heat Treatment Temp, F	Heat Treatment Time, hr	Nominal Carbide Case Depth (mils)		Remarks
0.28	2800	1	None	1, 2	
0.56	2800	1	0.2	1, 2	
1.13	2800	1	0.4	1, 2	
2.25	2800	1	0.5	2, 3	
0.28	2600	6	None	1, 2	
0.56	2600	6	0.2	1, 2	See Fig 61A
1.13	2600	6	0.4	1, 2	See Fig 61B
2.25	2600	6	0.5	1, 2	See Fig 61C
0.28	2600	4	0.05	1, 2	
0.56	2600	4	0.2	1, 2	
1.13	2600	4	0.3	2, 3	
2.25	2600	4	0.3	2, 3	
2.25	2600	0.5	0.1	2, 3	
2.25	2400	1	0.1	3	Carbide pptn-20 mils
0.28	2200	64	0.05	1, 2	
0.28	2200	16	0.1	1	Carbide pptn-10 mils
2.25	2200	7	0.2	3	Carbide pptn-8 mils
0.28	2200	4	0.1	3	Slight carbide pptn
0.56	2200	4	0.1	3	Slight carbide pptn
1.13	2200	4	0.1	3	Slight carbide pptn
2.25	2200	4	0.1	3	Slight carbide pptn

[1] coating complete absorbed

[2] spheroidal carbide precipitation throughout specimen

[3] graphite remaining on specimen surface

The conditions listed in line eight of the table (2.25 mg carbon/cm^2, 6 hours, 2600F) were selected for evaluation in bearing material screening tests and specimens are presently being prepared. Surface layers resulting from this coating process are shown along with hardness data in Fig 61.

Bearing materials screening tests were continued in a laboratory test apparatus described in report PWAC-624. A total of five materials combinations consisting of Mo-0.5 Ti alloy, two sets of carburized Cb-1 Zr alloy specimens, nitrided Cb-1 Zr alloy and carburized Mo-0.5 Ti alloy were tested in 700F lithium during this period. In general, the carburized specimens performed well but further testing of these and other materials is required before a relative ranking of candidate materials can be prepared.

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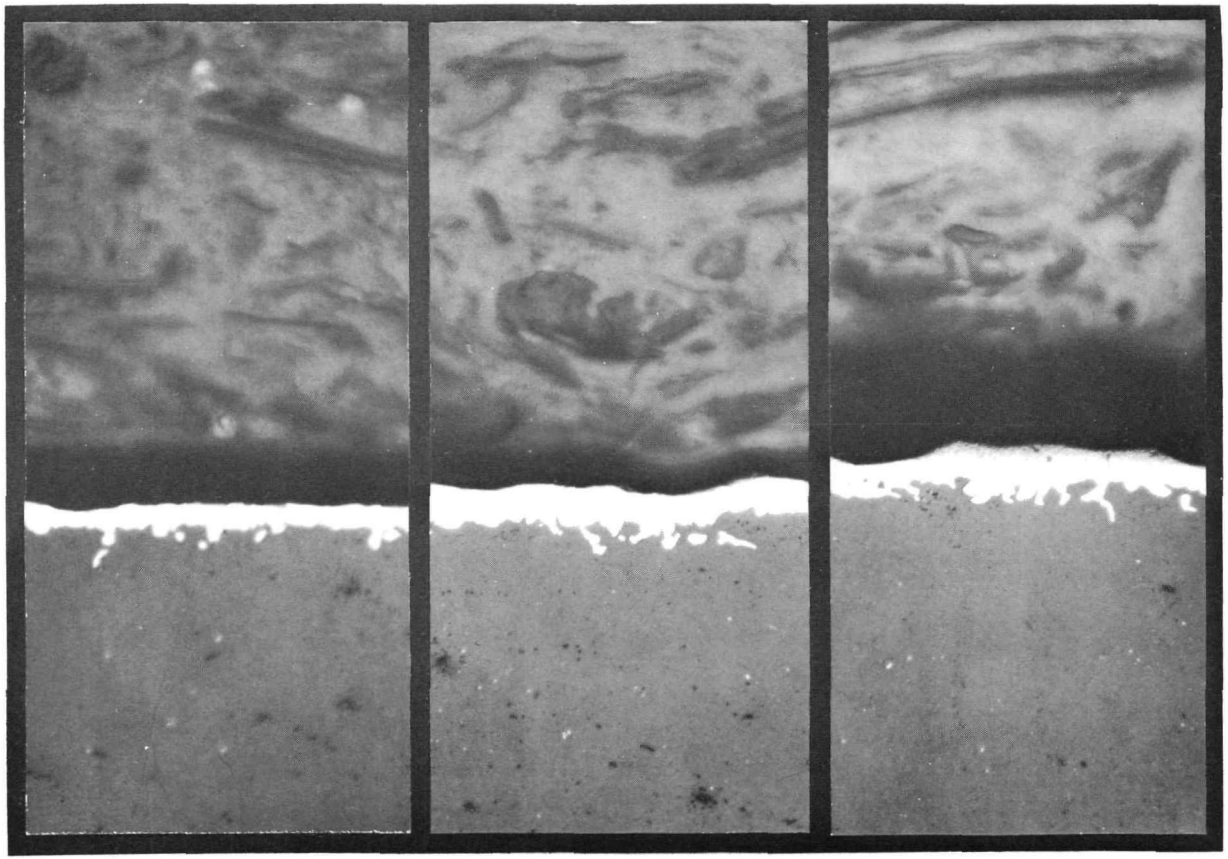
COLUMBIUM CARBIDE SURFACE LAYERS ON Cb-1 ZR ALLOY



1 MIL

MAG: 500X

	A	B	C
WEIGHT OF GRAPHITE (MG/CM^2)	0.56	1.13	2.25
CASE THICKNESS (MILS)	0.2	0.4	0.5
CASE HARDNESS (DPH)	-	1008	1017

PHOTOMICROGRAPH SHOWING CASE DEPTH AS A FUNCTION OF CARBON COATING WEIGHT AFTER FIRING FOR 6 HOURS AT 2600F.

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IV. ADVANCED SYSTEMS ENGINEERING

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IV. ADVANCED SYSTEMS ENGINEERING

A report, SNAP-50 System Optimization Studies, PWAC-391 was published during the last quarter. This presents the results of the latest system optimization studies of the SNAP-50 powerplant. The cycle conditions shown in Fig 62 and 63 were derived from this study. The latest SNAP-50 component data for the primary system, power conversion loop, main radiator system, radiation shielding, auxiliary heat rejection system and power conditioning equipment were incorporated into this analysis. Alternate types of turbines and radiator systems, using direct condensing and NaK radiators, were studied. Several alternate methods of providing auxiliary cooling were studied. A method utilizing a separate potassium circuit was selected.

A report entitled CNLM-4448, SNAP-50 Preliminary Data for Application Studies, was published during the quarter. Its purpose was to present preliminary performance and installation characteristics for use in space vehicle and mission studies. The report is being called to the attention of government agencies and industrial groups who might be interested in using SNAP-50 in their missions.

The report presents the flow schematics and weight breakdowns of a series of powerplants which cover the power output range from 150 to 1200 Kwe. The series is based on the modular concept which makes it possible to cover a wide range of power outputs using a single set of component designs. The 300 Kwe powerplant is shown schematically in Fig 62. Two segments are removed to provide 150 Kwe output. Two, three, or four of the power conversion loops shown in this figure are used with the same reactor design to produce 600, 900, or 1200 Kwe respectively. The four loop, 1200 Kwe powerplant is shown schematically in Fig 63. The weights of the 300, 600, 900 and 1200 Kwe powerplants are summarized in the following table:

SNAP-50 Preliminary Weight Summary

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Power Output, Kwe	300	600	900	1200
No. of Power Conversion Units	1	2	3	4
Weight, lb:				
Reactor & Primary System	1030	1400	1400	1400
Power Conversion Equipment	940	1880	2820	3760
Heat Rejection & Miscellaneous Equipment	1160	2320	3480	4640
Radiator Meteoroid Shielding	520	1040	1560	2080
Totals	3650	6640	9260	11,880
Specific Weight, lb/Kwe	12.2	11.0	10.2	9.0

Approximately 20 SNAP-50 powerplant arrangement studies have been conducted since the beginning of the program. These studies, which are summarized in PWAC-387, SNAP-50 Arrangment Studies, cover powerplants ranging from 150 to 1200 Kwe and deal with booster payload envelope limitations and spacecraft integration problems. The boosters used in these studies were Atlas-Centaur, Titan III, Saturn C-1 and Saturn C-1B.

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300 KW SNAP-50 FLOW SCHEMATIC

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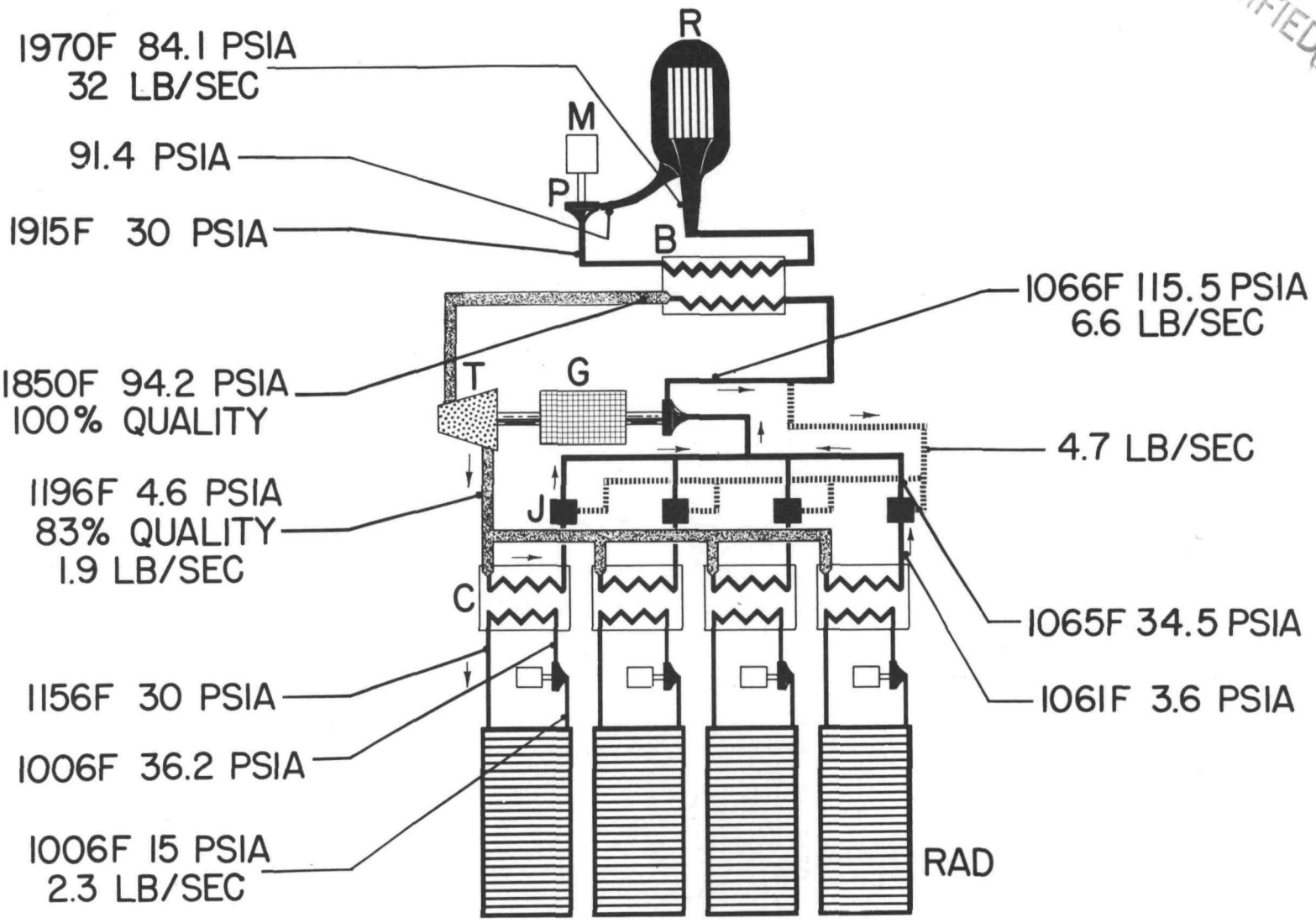

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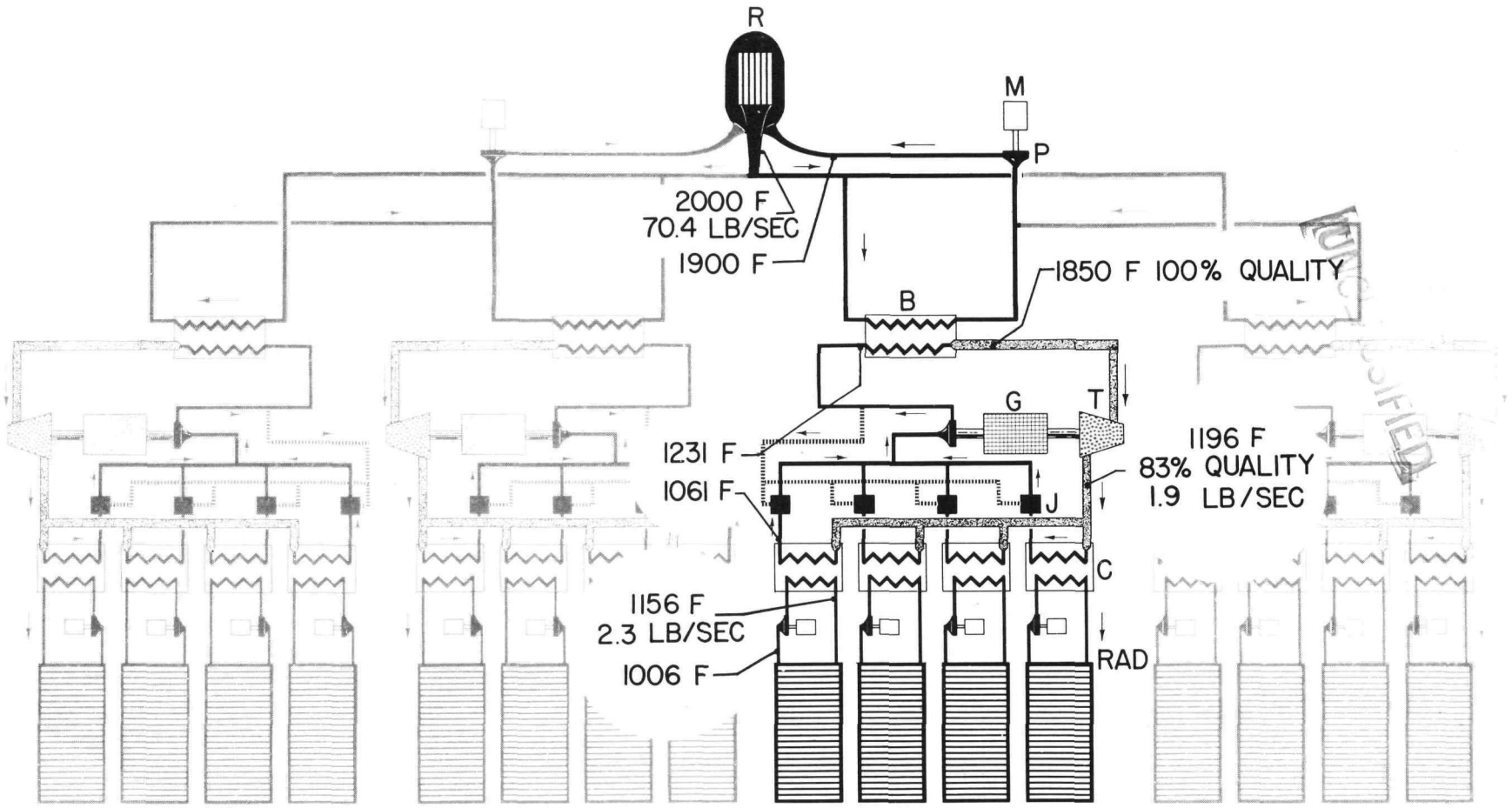
1200 KW SNAP-50 FLOW SCHEMATIC



The powerplants studied were various combinations of a single reactor, one to four power conversion units, and five different folding radiator schemes. Complete spacecraft design problems were studied including a simulated payload in several of the studies. The following general conclusions resulted from the studies:

a. The radiator is a dominant factor in determining the power level of the powerplant which can be installed in any given booster payload envelope. It was shown that a folding radiator increases the maximum power level which can be installed in given envelope by a factor of about four over that obtainable using a rigid non-deployable planar radiator. A consequence of this increase is the necessity for developing the liquid metal pipe hinges required by these systems.

b. Except for very small envolopes, such as that of the Atlas-Centaur, there is usually sufficient volume remaining in the launch envelope after the powerplant has been installed for the convenient stowage payload, communications antennae, and propulsion devices.

c. It appears to be desirable, although not mandatory, to deploy the spacecraft's payload from its launch position. This introduces another degree of freedom in the spacecraft design which permits more efficient use of the launch envelope volume. It also reduces payload shielding requirements, due to the fairly large distances obtainable between the radiation source and sensitive items, and allows unhibited erection of communications antennae, payload radiators, and scientific experiment sensors.

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