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SNAP-50/SPUR

PROGRAM SUMMARY

PRESENTED AT

CONNECTICUT ADVANCED NUCLEAR ENGINEERING LABORATORY

MIDDLETOWN, CONNECTICUT

SEPTEMBER 24, 1964

[redacted]

Pratt & Whitney Aircraft DIVISION OF UNITED AIRCRAFT CORPORATION

GARRETT

AIRESEARCH MANUFACTURING COMPANY

A DIVISION OF THE GARRETT CORPORATION

PHOENIX, ARIZONA

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GROUP 1
EXCLUDED FROM AUTOMATIC
DOWNGRADING AND DECLASSIFICATION

SPECIAL REVIEW FINAL DETERMINATION
Class: U
Reviewer: HFC
Class. U
Date 12-03-82

CLASSIFICATION CANCELLED
DATE SEP 10 1973
For The Atomic Energy Commission
Division of Classification

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PREFACE

This Briefing Book presents a review of the SNAP-50/SPUR Program, including the charts used in the presentation on September 24, 1964, at the Connecticut Advanced Nuclear Engineering Laboratory, Middletown, Connecticut.

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Section VI

CONNECTICUT ADVANCED NUCLEAR ENGINEERING LABORATORY
BRIEFING AGENDA
SEPTEMBER 24, 1964

8:00	Depart America Hotel, Hartford, Connecticut	
8:45	Introductions at CANEL	W. H. Pennington, AEC SNAP-50 Project Manager
9:00	MPRE, 710 Reactor and Brayton Cycle Presentations	Dr. E. E. Sinclair, AEC Ass't. Director, DRD
10:30	SNAP-50/SPUR Presentation:	
	Program Background	Dr. R. I. Strough, P&WA Program Manager
	Technological Background	
	Current Program at CANEL: System Engineering and Testing Reactor and Shield Component Development Instruments and Controls	
12:15	Current Program at AiResearch: Boiler and Condenser Turbo-alternator Radiator	J. H. Dannan, AiResearch Project Manager
12:45	Program Summary	Walter Doll, P&WA General Manager-CANEL
1:00	Lunch and Exhibit Tour	
2:00	Discussion of SNAP-50 Presentations and Written Questions	
3:30	Tour of CANEL Facilities: General Laboratory Shop Laboratory Heat Exchanger Laboratory Nuclear Physics Laboratory Hot Laboratory Fabrication Facility	
5:15	Adjourn	
6:00	Arrive Hartford Airport	

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I PROGRAM OBJECTIVES

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I PROGRAM OBJECTIVES

I. PROGRAM OBJECTIVES AND RESPONSIBILITY ASSIGNMENTS

The primary task (Fig. 1) of the SNAP-50/SPUR Program is to develop a nuclear-electric powerplant for space use in the 300-Kwe to 1000-Kwe power level range, having a lifetime of 10,000 hours and a minimum weight. The present reference powerplant design, used for program direction and guidance, would produce 300 Kwe of useful power. However, the power level of the prototype powerplant will be set by consideration of probable Air Force and NASA space power needs. A reactor design suitable for a 1-Mwe powerplant is in progress at CANEL to cover the upper level of the SNAP-50/SPUR power range.

The long-life objectives of the SNAP-50/SPUR powerplant can be met by conservative design at a relatively high specific weight. The salient development problem is then to reduce the specific weight to a minimum and still meet the performance and long-life requirements.

The SNAP-50/SPUR contractor responsibilities as assigned by the Atomic Energy Commission and the Air Force are presented in Fig. 2. Pratt & Whitney Aircraft-CANEL is responsible for development of the reactor, shield, pumps auxiliaries, and controls system. It is also responsible for the over-all integration of the powerplant, the design approval of all powerplant components, fabrication, and testing of all powerplant columbium alloy components and the testing of the SNAP-50/SPUR ground prototype powerplant. The AiResearch Division of the Garrett Corporation has been assigned the design and initial development of the boiler, condenser, and turboalternator. It has also been assigned a radiator study contract. The responsibility for the vehicle and the radiator design and development has not been assigned to date.

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SNAP-50/SPUR POWERPLANT PROGRAM OBJECTIVES

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POWER	300 TO 1000 K_{we}
CYCLE	RANKINE
USEFUL LIFE	10,000 HRS.
SPECIFIC WEIGHT	20 LBS/K_{we}

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FIG 1

SNAP-50/SPUR
CONTRACTOR RESPONSIBILITIES

A PRATT & WHITNEY AIRCRAFT

- DEVELOPMENT OF REACTOR AND SHIELD, INCLUDING PERFORMANCE OF ALL REACTOR TESTS
- DEVELOPMENT OF PUMPS, PIPING AND AUXILIARIES
- DESIGN APPROVAL OF ALL POWERPLANT COMPONENTS
- FABRICATION AND TESTING OF ALL COLUMBIUM ALLOY COMPONENTS AND SYSTEMS
- DEVELOPMENT OF CONTROL SYSTEM
- OVER-ALL INTEGRATION OF POWERPLANT DESIGN AND DEVELOPMENT
- TEST OF GROUND PROTOTYPE POWERPLANT

B AiRESEARCH

- DESIGN AND DEVELOPMENT OF BOILER, CONDENSER AND TURBO-ALTERNATOR
- RADIATOR STUDY CONTRACT

C RADIATOR DESIGN AND DEVELOPMENT UNASSIGNED

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FIG 2

II POWERPLANT DESCRIPTION

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II. POWERPLANT APPLICATIONS AND DESCRIPTION OF THE REFERENCE DESIGN

A. The Need for SNAP-50/SPUR Power Levels in Space

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SNAP-50/SPUR power levels will be needed in space for electric propulsion and power supply, as indicated in Figure 3. Based on studies of unmanned space probes by the California Institute of Technology, Jet Propulsion Laboratory, and the General Electric Missiles and Space Division, electric propulsion will be required for the more difficult missions, such as investigations of the sun at one-tenth astronomical unit, out-of-the ecliptic probes at angles of 30 degrees or more, and orbiter type missions to the outer planets (See Refs. 1, 2, and 3). Unmanned probe missions to the near planets can be accomplished by chemical rockets, nuclear rockets, and electric propulsion, and will probably use the most appropriate available propulsion system.

Studies of the manned Mars mission, sponsored by the NASA Marshall Space Center at the United Aircraft Research Laboratories and the General Electric Missiles and Space Division, are considering the possibility of using a combination of nuclear rocket and nuclear-electric propulsion as the most economical and expeditious way of performing this mission in reasonable trip times. These studies are considering the possibility of using several Saturn-V rockets to put the spacecraft into Earth orbit. A nuclear rocket will be used for Earth escape and propulsion early in the trip. From that point on, nuclear-electric propulsion will be used until the spacecraft is put into low orbit around Mars. The Mars excursion module will then be landed on the planet surface by chemical propulsion and, after a suitable period of exploration, return to the space vehicle by chemical rocket in much the same manner as the Lunar Excursion Module in the Apollo Program. The return of the space vehicle to Earth orbit is then accomplished by nuclear-electric propulsion. Electric propulsion is attractive for this case, since alternate propulsion schemes increase severalfold the required number of Saturn-V boosters.

Electric power supplies in the 50 to 1000-kilowatt or more power range will eventually be required for military weapons systems, large orbiting space stations, and for lunar base power requirements. The Aerospace Corporation, under contract to the Air Force Space Systems Division, completed a survey type study of the possible military uses of SNAP-50/SPUR power levels in space (Ref. 4). This report concluded that such power systems may be eventually required to supply electrical power for reconnaissance and surveillance, command and control, space-based weapons systems, and electronic counter measures. The Air Force Space Systems Division intends to issue a request for proposal to review these military mission possibilities for the purpose of further defining the power level for which the SNAP-50/SPUR prototype powerplant should be designed and developed.

Several lunar base studies have been made to date, most notably Refs. 5, 6 and 7. These studies show that the more advanced lunar bases will require power levels of the order of 50 to 3000 Kwe for periods longer than six months. The Westinghouse Lunar Power Supply study shows that SNAP-50/SPUR type power systems can supply this requirement. Our own studies (Refs. 8 and 9) show how the SNAP-50/SPUR powerplant can be modified for this application and delivered with integral shielding by one Saturn-V launch vehicle.

B. Choice of Powerplant

Once it was established that SNAP-50/SPUR power levels would be needed in space, the next question which had to be answered was which powerplant would be best to develop considering both the early assurance of success as well as future growth potential. Thermionic and Brayton power conversion systems, as well as the Rankine cycle were originally considered as possibilities to meet the SNAP-50/SPUR objectives. The Rankine cycle was chosen on the basis that the objectives can be met at lower temperatures and therefore much sooner than the other two powerplant possibilities. This is illustrated in Fig. 4, where the results of studies are compared to determine at approximately what fuel clad temperature level each of the three cycles would have to operate. To meet the SNAP-50/SPUR specific weight objective of 20 pounds per kilowatt, the Brayton cycle, based on a General Electric study (Ref.10), would have to operate at fuel cladding temperatures greater than 3000F. The thermionic cycle powerplant, based on a Pratt & Whitney Aircraft East Hartford study (Ref.11), would have to operate at a fuel cladding temperature on the order of 3200F. The problems of gas retention and fuel burnup of the ceramic and cermet fuels required by these three

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SNAP-50/SPUR APPLICATIONS

	POWER
ELECTRIC PROPULSION:	
UNMANNED PROBES	300 TO 1000 Kwe
LOGISTICS VEHICLES	1 TO 5 Mwe
MANNED VEHICLES	2 TO 5 Mwe
ELECTRIC POWER SUPPLY:	
LUNAR BASE	50 TO 300 Kwe
MANNED SPACE STATIONS	40 Kwe AND OVER
MILITARY WEAPON SYSTEMS	SEVERAL Kwe TO OVER 1 Mwe
DEEP SPACE COMMUNICATIONS	1 Kwe TO SEVERAL Mwe

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FIG 3

POWERPLANT COMPARISON

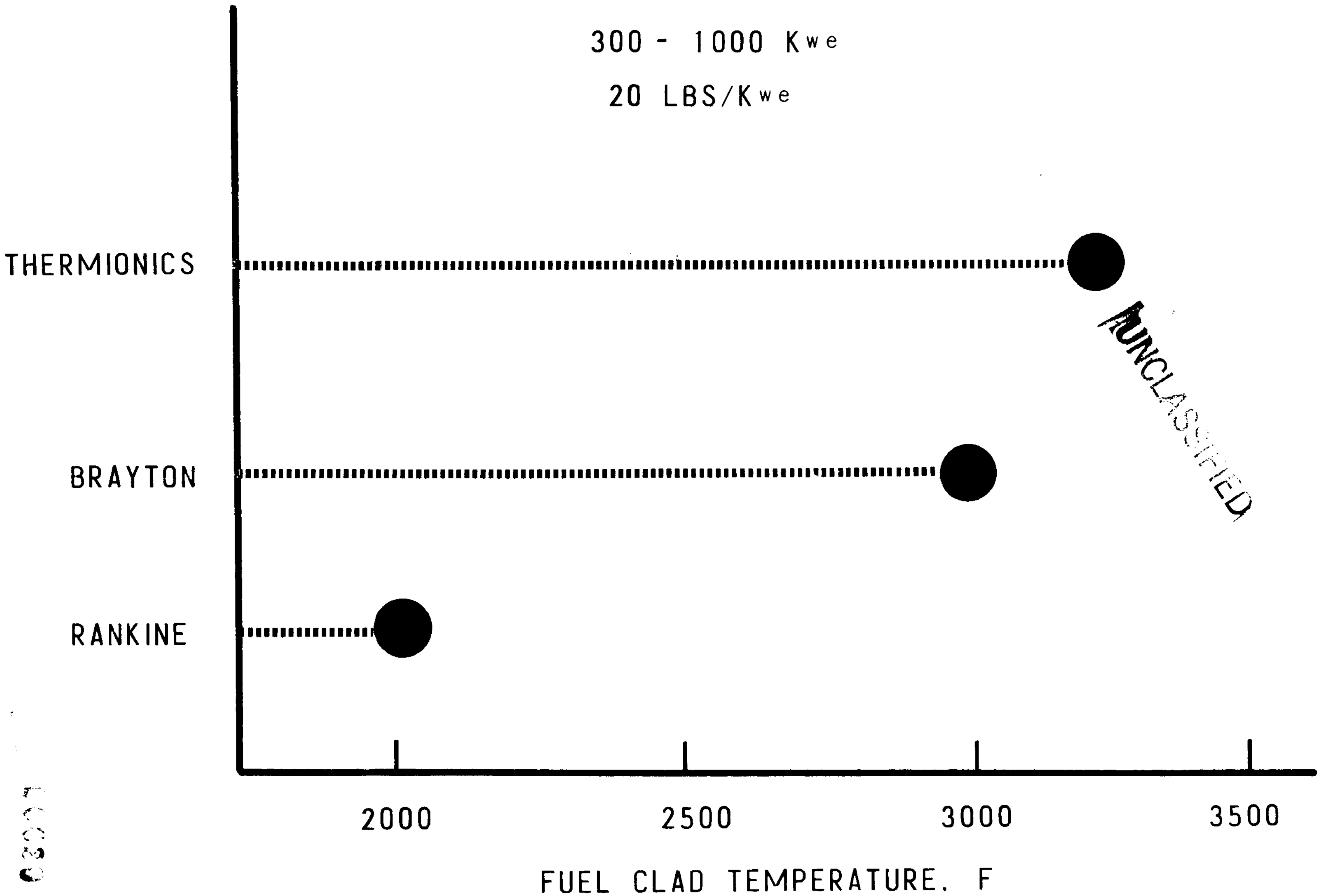

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FIG 4

II-3

powerplants are similar except that they can be solved more readily at 2000F. Another aspect of the development of these three powerplants is the compatibility and structural strength of refractory alloys at these extreme temperatures. These problems increase exponentially with temperature.

C. Reference Design Powerplant Description

The reference design power level of 300 Kwe was chosen to guide the technology and initial development phase of the SNAP-50/SPUR Program. The cycle in its simplest form is shown in Fig. 5.

Lithium is circulated through the reactor and transfers the reactor heat to the boiler (Fig. 6). Potassium condensate enters the boiler, vaporizes to 100 percent quality or to a slightly superheated condition, and is then delivered to the turbine where the energy of the potassium vapor is converted to mechanical power to drive the alternator. After leaving the turbine, the potassium flow is condensed and subcooled in the condenser. The heat of condensation is transferred to the main radiator where the condenser heat is rejected to space. The auxiliary coolant (NaK) is pumped to the individual component heat exchangers where it absorbs the waste heat of the electrical components, maintaining them at their required operating temperatures. This heat is then rejected to space by radiation from the auxiliary radiators.

The reactor outlet temperature of 2000F was selected on the basis of the technology built up at CANEL. The turbine inlet quality was set at 100 percent to maintain a high cycle efficiency. The subcooling of the potassium flow in the condenser was set at 100F to provide adequate margin for avoiding pump cavitation effects. The auxiliary coolant temperature was set at 600F so that the maximum hot spot temperature in the electrical machinery would not exceed the capabilities of the magnetic alloys used in the motors and alternators. Utilizing estimates of the efficiencies and pressure losses of the components, a weight optimization program was conducted to determine the reactor coolant temperature rise and the turbine inlet and outlet conditions and the radiator conditions.

The 300-Kwe reference design powerplant consists of four major systems: a primary system, a power conversion system, a main heat rejection system, and an auxiliary heat rejection system. The primary system consists of a reactor, a canned electrical motor-driven centrifugal pump, an accumulator to maintain the desired system pressure, and necessary piping. The reactor, utilizing uranium carbide fuel, is to be designed to produce approximately two megawatts thermal power for the 300 Kwe system.

The components of the power conversion system include a boiler, a turboalternator unit, a segmented condenser, a canned electrical motor-driven centrifugal condensate pump in series with a jet pump, piping and valves, and a potassium accumulator system to compensate for system inventory changes during off-design operation and start-up (Figs. 7 and 8).

The main heat rejection system consists of four independent loops, each containing a non-deployable radiator, motor-driven pump, accumulator, and associated piping.

Use of a direct condensing radiator system is also being considered. This would usually result in lower powerplant weight, but poses certain materials, testing and integration problems which have not yet been resolved.

The auxiliary heat rejection system (Fig. 7) is composed of four independent loops, each containing a radiator and the required piping to provide NaK coolant at temperatures of 520F. Each auxiliary radiator loop is paired with a main radiator loop, with the auxiliary NaK coolant being circulated by the lubricant impeller of the main radiator NaK pumps.

The estimated weight summaries for both the current reference design and a powerplant design based on the development potential using improved reactor fuel and radiator systems are shown in Fig. 9.

SNAP-50 / SPUR SCHEMATIC

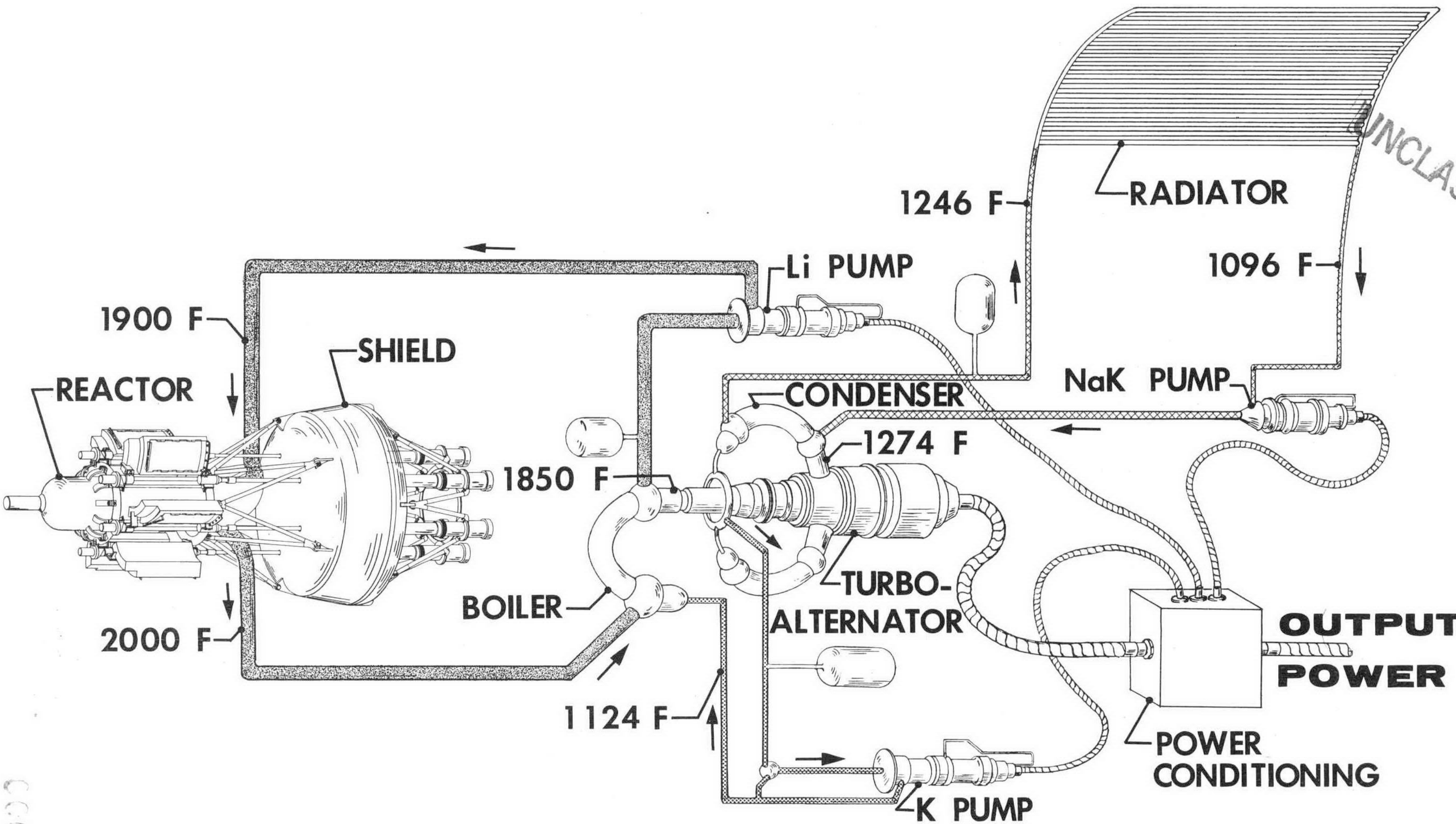

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FIG 5

300 KWE SNAP - 50 FLOW SCHEMATIC

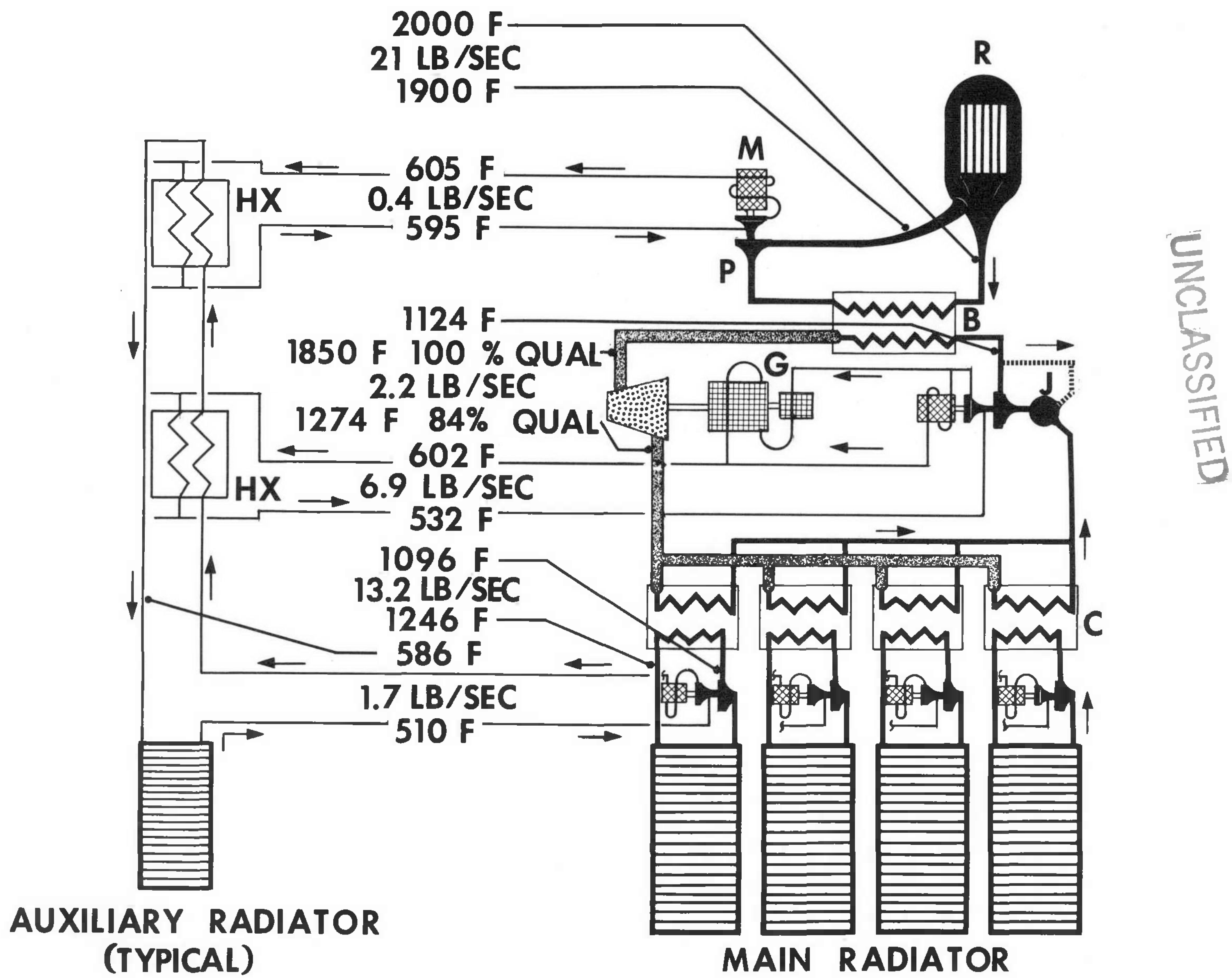

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FIG 6

SNAP - 50 POWERPLANT COMPONENTS & ASSEMBLY

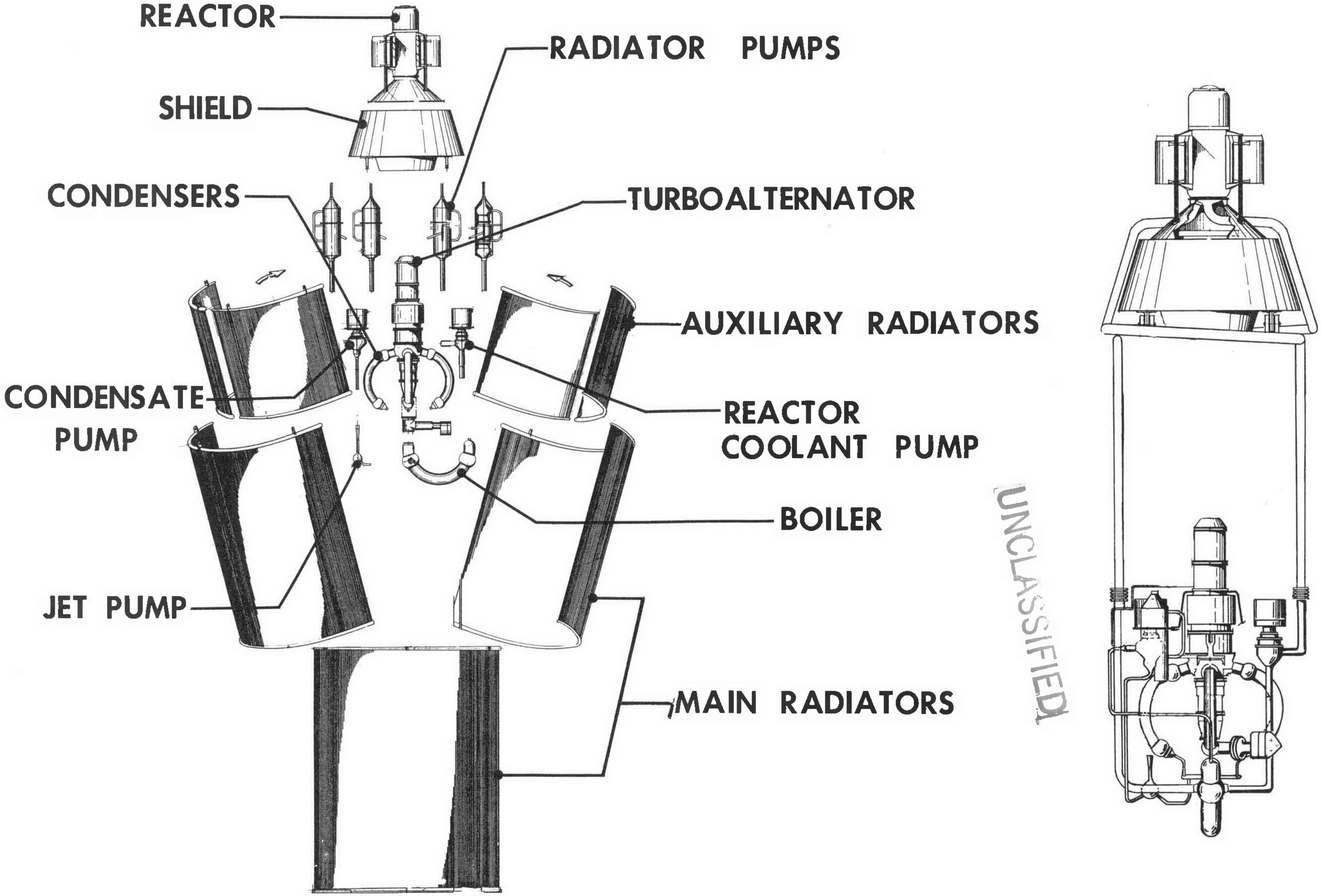

FIG 7

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FIG 8

MODEL OF SNAP-50/SPUR POWERPLANT

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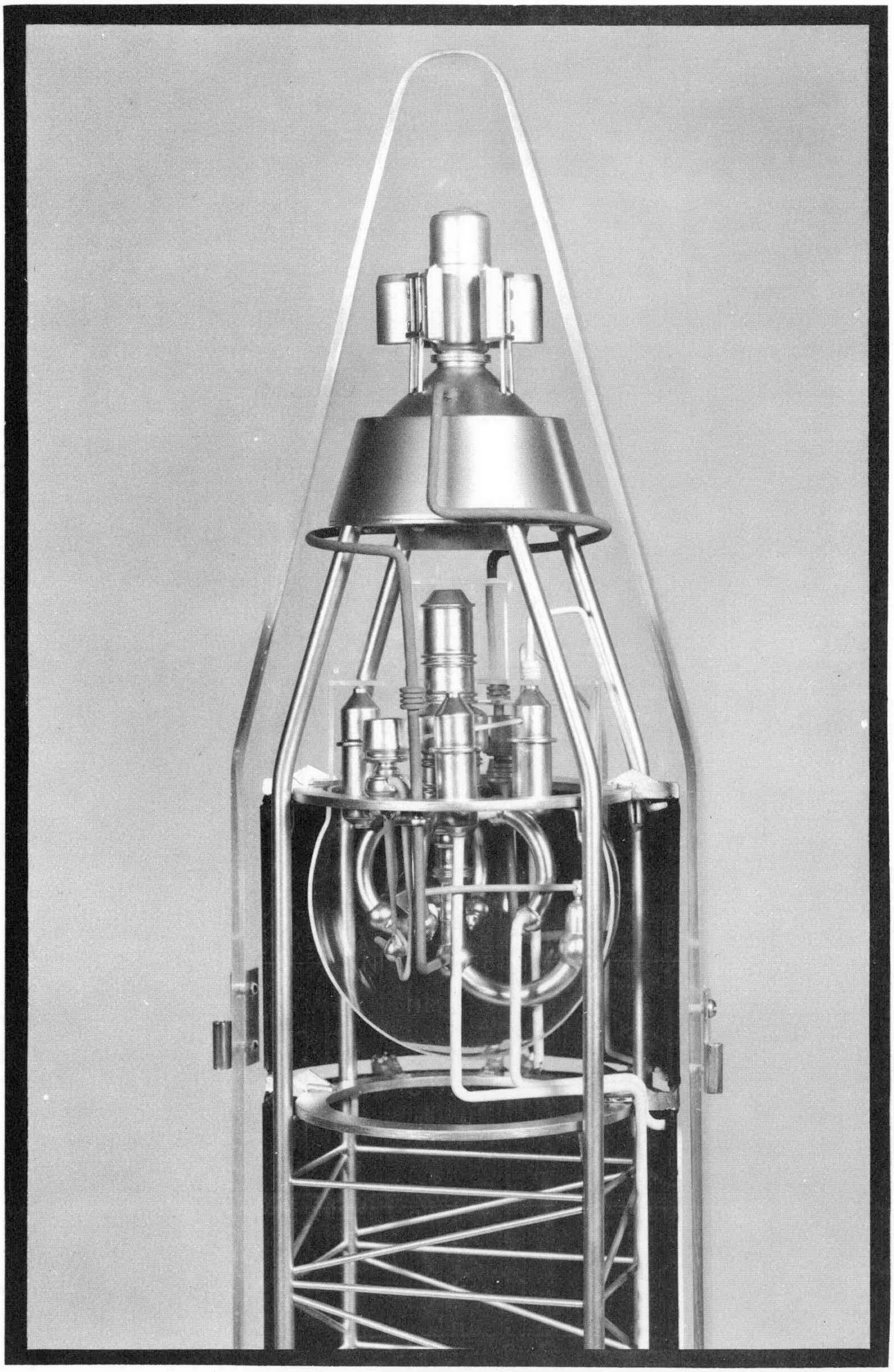

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300 Kwe SNAP-50 / SPUR WEIGHT SUMMARY

	CURRENT DESIGN	DEVELOPMENT POTENTIAL
REACTOR	1800	1300
PRIMARY SYSTEM	440	440
POWER CONVERSION	1710	1415
HEAT REJECTION SYSTEM	2615	1395
STRUCTURES. CONTROLS SUBSYSTEMS	925	915
TOTAL	7490 LBS	5465
	25.0 LBS/Kwe	18.2 LBS/Kwe
SHIELD		
$5 \times 10^{11} - 10^{13}$ nvt	1800-3000	1500 - 2400

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FIG 9

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III DEVELOPMENT PLAN

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III. DEVELOPMENT PLAN

A. Approach

The powerplant development approach requires a balanced engineering program, consistent with both the complexity of the technical problems to be solved and the anticipated time period when the resulting powerplant will be required.

The major program steps are:

1. Extend the basic technology to 10,000+ hours.
2. Test subcomponents separately.
3. Develop full-scale components separately or in subsystem tests.
4. Test the reactor individually, utilizing a heat dump system.
5. Conduct tests of the complete power conversion system using an electrical heat source to simulate the reactor.
6. Run a ground test of the powerplant, including the reactor.

B. Schedule

The currently proposed development schedule (Fig. 10) consists of concurrent nuclear and non-nuclear programs, beginning with initiation of prototype component liquid metal tests in 1966 to 1967, and culminating in both a non-nuclear systems test and a reactor test in 1969. These latter tests are followed by a ground test of the integrated powerplant in about 1971. It is presumed that a flight test vehicle development program would be conducted in parallel with the powerplant development. This would permit a flight test in the 1972 to 1975 time period.

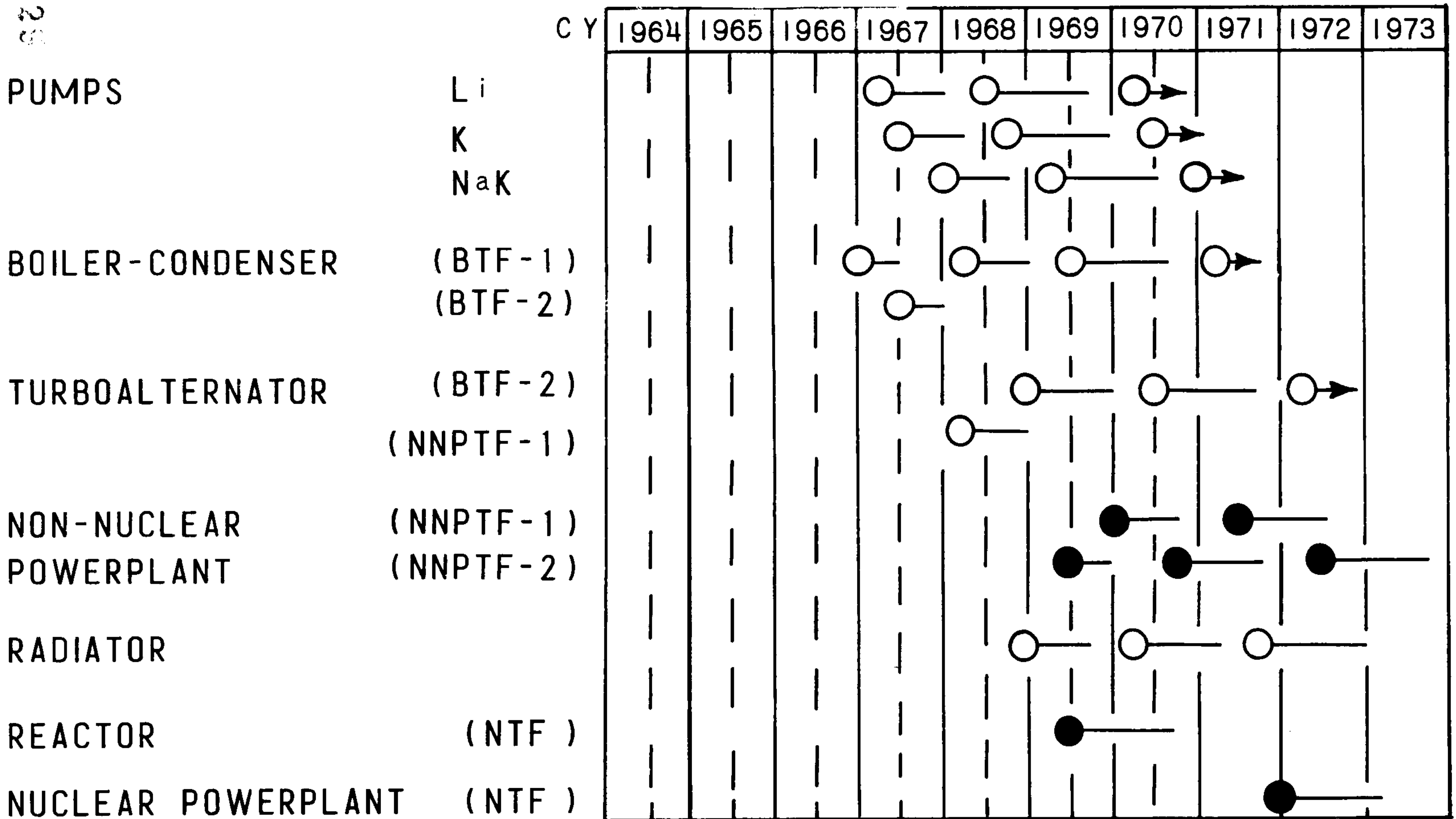
Estimated program costs are also shown in Fig. 10. All values include an annual five percent escalation adjustment.

Costs have not been established beyond Fiscal Year 1970, although a funding level of about $50 million per year would be anticipated for the ensuing years. Thus, the estimated program costs through the reactor and first non-nuclear powerplant tests are about $250 million. Continuing, the development program, including component, systems and nuclear ground powerplant testing through 1974, would increase the program cost to the $450 million level.

C. Facilities

The major test facilities required to support this program are indicated in Fig. 11. It is proposed to use the existing CANEL facilities complex as a basis for these test cells. The boiler test facility (BTF) has been authorized in the Fiscal Year 1964 budget. It will be built in the present CANEL Heat Exchanger Laboratory containing two test cells (BTF-1 and BTF-2). Two non-nuclear powerplant test facilities (NNPTF) have been proposed for construction in the existing Radiator Laboratory. The first would require FY 1966 funds and the second would require FY 1967 funds. The initial facility would first be utilized as a turboalternator development stand. A nuclear test facility (NTF) is planned at CANEL which would permit both initial reactor testing and nuclear powerplant operation. This facility would require funding from the FY 1966 budget. A separate radiator development building is needed. Detailed engineering studies have not been made on this facility. However, the CANEL Pilot Radiator Laboratory appears to be usable for this purpose. The planned development schedule will require FY 1968 construction money for this program. The total estimated construction costs for the above facilities is $27.5 million. Twenty six million dollars plus the Radiator facility remain to be funded.

PROPOSED SNAP-50/SPUR DEVELOPMENT SCHEDULE



	FY 65	66	67	68	69	70	TOTAL
OPERATING FUNDS:							
CANEL	15.8	23.9	33.5	42.0	43.5	45.9	204.6
AiRESEARCH	4.0	7.7	6.8	7.0	7.0	7.0	39.5
RADIATOR CONTR.	0.1	0.4	0.8	1.3	2.0	2.7	7.3
TOTAL	19.9	32.0	41.1	50.3	52.5	55.6	251.4
CAPITAL EQUIPMENT	2.8	4.3	4.5	4.5	3.6	3.5	22.2

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FIG 10

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SNAP-50/SPUR DEVELOPMENT PROGRAM

NEW FACILITIES

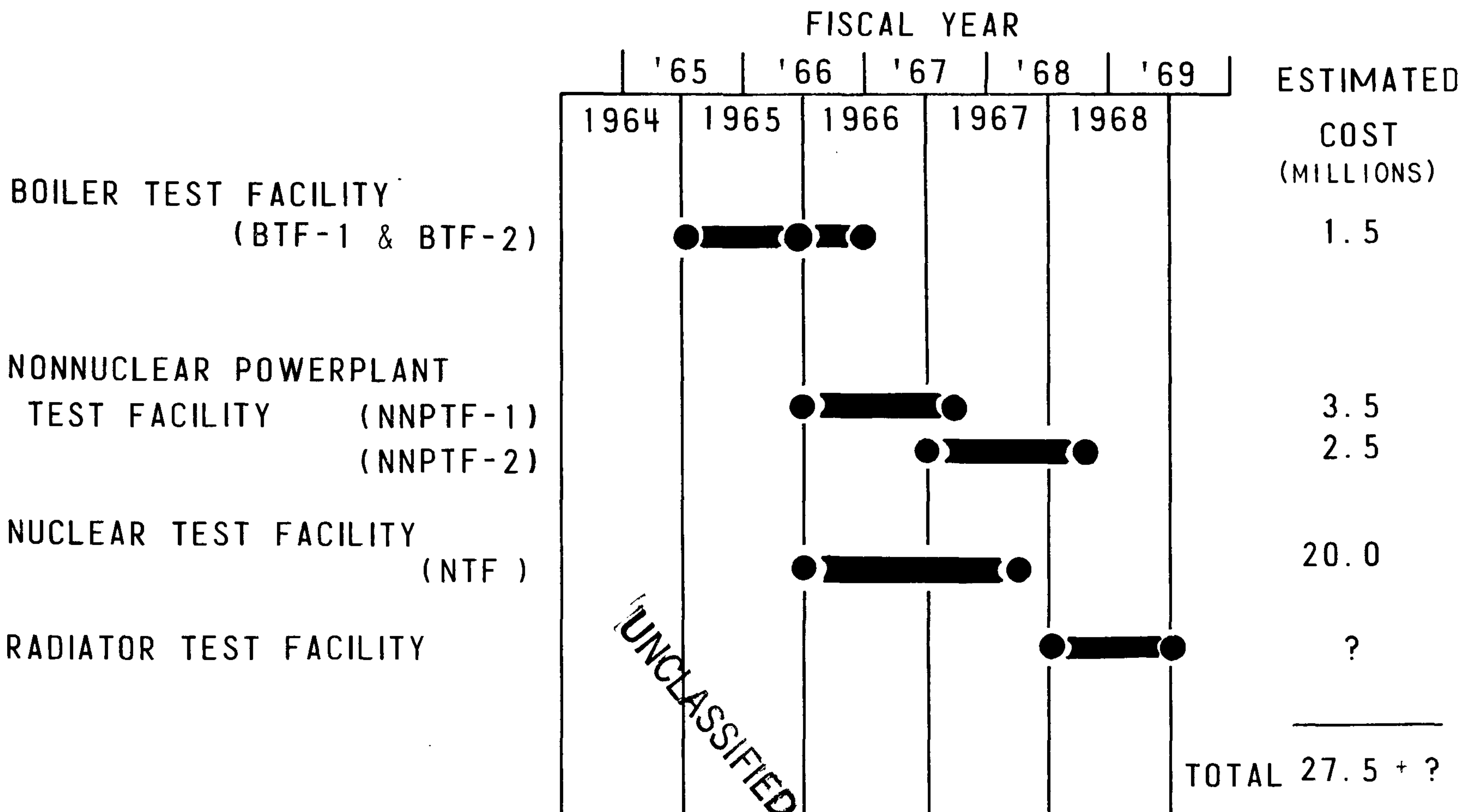

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FIG 11

D. CANEL Cost Structure

The structure of costs at CANEL is shown in Fig. 12 for Fiscal Years 1964, 1965, and 1966. The AEC programs listed comprise virtually all of the CANEL funding support during these years. The man years supported by these funds is indicated at the bottom of the chart. The direct labor portion averages about 70 percent of the total work force. The scientific staff comprises about 25 to 30 percent of the total manpower.

CANEL PROGRAM COSTS
FISCAL YEARS 1964 - 1966

(IN THOUSANDS)

	FY1964	FY1965	FY1966
SNAP-50/SPUR	22,303.0	15,800.0	23,900.0
ADVANCED MATERIALS	1,025.1	1,100.0	1,400.0
MCR	94.3	70.0	---
CAFEE	7.4	10.0	10.0
TOTAL	23,429.8	16,980.0	25,310.0
SUMMARY			
DIRECT LABOR	7,426.2	6,394.0	8,830.6
OVERHEAD	8,456.7	7,181.0	9,176.2
DIRECT MATERIAL	6,191.3	2,120.0	5,122.2
SUBCONTRACTS	370.6	485.0	1,006.0
FEE	985.0	800.0	1,175.0
TOTAL	23,429.8	16,980.0	25,310.0
MAN YEARS	1,541	1,241	1,620

FIG 12

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IV TECHNOLOGICAL BASIS

IV TECHNOLOGICAL BASIS

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IV. TECHNOLOGICAL BASIS FOR THE SNAP-50/SPUR DEVELOPMENT PROGRAM

The technological basis on which the SNAP-50/SPUR development rests has its roots deep in the old Aircraft Nuclear Propulsion Program. Pratt & Whitney Aircraft's nuclear experience dates back to 1951, when we became involved in feasibility and development studies of the super-critical water cycle. However, it was 1953 when our attention was first attracted to liquid metal as a heat transfer medium. At that time, we set up and operated our first liquid metal test loop which simulated the temperature and flow cycle of the heat transfer loop of an indirect cycle nuclear aircraft powerplant. From 1953 to 1958, we accumulated an extensive background in the development of fuels, corrosion loops, pumps, heat exchangers and radiators, using sodium and NaK in stainless steel in the temperature range of 1000F to 1600F (Fig. 13). Extensive reactor physics data were obtained from critical experiments on a mockup of a sodium-cooled, stainless steel, beryllium-moderated reactor, (see Fig. 46).

In 1957, Pratt & Whitney Aircraft's nuclear powerplant development efforts were redirected to an advanced technology program based on lithium as a reactor coolant with the objective of developing a high temperature powerplant. The reasons for choosing lithium over other liquid metals as a heat transport fluid is its relatively high specific heat as compared with sodium and NaK, its very low vapor pressure, and its relatively low irradiation activation as a result of exposure to a fast neutron spectrum (Fig. 14). This combination of properties makes lithium the best available heat transfer fluid in the temperature range of 500F to 3000F and above.

Having chosen lithium as the heat transfer fluid, the problem became one of finding a containment material. Mass transport with lithium in stainless steels and other materials known in 1957 was intolerable for temperatures above 1000F. Our Materials Laboratory here at CANEL studied the general properties of a number of alloy systems and short-time tests were run. It became evident from this work that the columbium alloys offered the possibility of a good solution to the lithium containment problems in the temperature range of 100F to 2200F (Fig. 15). The initial columbium alloy chosen for extensive development was Cb-1 Zr . We have since added other columbium alloys to our materials program for engineering development (Fig. 16).

Before any extensive experimental work could be done, it was necessary to learn how to fabricate columbium alloys. Since the beginning of this learning process, we have used more than 150 tons of columbium at CANEL in the fabrication of test loops, pumps, heat exchangers, parts for the aircraft nuclear propulsion test reactor and the Lithium-Cooled Reactor Experiment (Fig. 17). As a result of this pioneering effort at CANEL, it is now possible to purchase, commercially, columbium alloys in the following shapes and sizes: vacuum arc melted ingots up to 16 inches in diameter, forgings up to 36 inches in diameter, plate up to 40 inches wide and 1/4-inch thick, tubing drawn down to 1/10-inch diameter and 0.008-inch wall. At the present time, columbium alloys are being used in considerable quantities in the Apollo and Minuteman rocket engine programs.

An important aspect of learning to fabricate columbium alloys was the development of the required degree of cleanliness in the Shop processes and, particularly, in welding. Figs. 18 and 19 show the Clean Room and the welding facilities in the CANEL Shop. Dust particles are controlled to less than 50 ppm in the Clean Room and the moisture and oxygen in the argon atmosphere of the welding facilities are kept below 8 ppm.

The lithium-columbium technology is summarized in Figs. 20 and 21 in the categories of fuel development, structural tests, corrosion tests, pumps, and heat exchangers. The number of tests and the total number of test hours shows the amount of test work that was done in each of these areas since 1957. While this is very extensive, it is important to note the best performance achieved in each case, since this is the measure of the status of the lithium-columbium technology. You will note that in the case of uranium carbide and uranium nitride, the best performance in irradiation tests to date at temperatures of interest has been 2500 to 2700 hours. The best in-pile performance of UO_2-BeO fuel capsules has been successfully tested 10,000 hours in the Materials Testing Reactor at the design power density. The Materials Test Reactor flux is lower than what a SNAP-50/SPUR reactor will be. Considerable fuels development work is in progress and remains to be done. This is discussed under the current program. You will note that in all of the remaining test areas, a number of successful 10,000-hour tests at the temperatures of interest have been completed. Six years ago, the columbium-lithium technology as demonstrated in Fig. 20 was non-existent and, since a vigorous research and development program is being continued at CANEL, we can expect to see considerable improvement before starting final design of the first SNAP-50/SPUR reactor and powerplant.

CANEL has been actively engaged in liquid metal pump development for the past ten years. Work was initiated under the Aircraft Nuclear Propulsion Program for the design and development of large circulating pumps powered by air turbines. Development testing of a 3000-gpm NaK turbopump (Fig. 22) was accomplished at temperatures up to 1300F. Development of a 1500F, 50-gpm motor-driven NaK pump was accomplished as part of the fuel element testing program. New designs of sump-mounted pumps were evolved to circulate lithium and NaK for the LCRE Program. These pumps were driven by electric motors and also utilized conventional lubrication systems with antifriction bearings and shaft face seals. The columbium-lithium pump, shown on the right side of Fig. 22, was developed for 10,000-hour service at 1000F. Over 82,000 hours of liquid metal pump testing were accomplished during the ANP and LCRE programs. Details of these pump development programs are described in Refs. 12, 13, and 14.

The type of liquid metal-to-liquid metal heat exchangers developed at CANEL in both columbium alloys and other materials is illustrated in Fig. 23. These heat exchangers have been developed in sizes up to 8-1/2 Mw heat transfer capacity and have been proof-tested at design temperature for over 9500 hours.

During 1961 and 1962, we performed a number of boiling potassium tests to investigate and to obtain basic heat transfer data for use in boiler design work (Ref. 15). Fig. 24 is a summary of this two-phase potassium experimental work. Figs. 25 and 26 show the test apparatus made from Haynes-25 alloy, in which local boiling potassium heat transfer coefficients were obtained. A total of 3625 hours of test time were logged on this apparatus. More than 1200 hours of test time were logged in a stainless steel boiling potassium heat transfer test loop in determining the heat transfer performance of various boiler tube configurations. Fig. 27 is a summary of the boiling potassium heat transfer data produced on a serpentine tube configuration in the CANEL tests. These data provide the boiler designer with sufficient information to size a boiler for a wide range of conditions. Excellent performance is indicated over the entire range. The serpentine tube geometry of these tests exhibits pressure drops which are only slightly higher than what would be experienced in a straight tube. Fig. 28 is a full-scale boiler model of a design of a refractory metal-potassium boiler suitable for the SNAP-50/SPUR reference design powerplant. The model was made to obtain shell side flow and pressure drop data with water, and also to demonstrate the fabrication feasibility. The tubes were made of copper and the shell was made of plexiglass in order to view the shell side flow. The potassium heat transfer capacity of this design operating with the potassium vapor outlet temperature of 1850F covers the range of 7 to 12 times 10^6 Btu/hr.

The best proofs of the status of columbium technology demonstrated to date at CANEL are the successful completion of 10,000-hour tests of both a 10-inch reactor pressure vessel (Fig. 29) and a full-scale LCRE 10-Mw reactor pressure vessel (Fig. 30), and the 5-Mw heat transfer test now in progress at the CANEL Heat Exchanger Laboratory. This latter test simulates the complete reactor and liquid metal heat rejection system of the Lithium-Cooled Reactor Experiment. Fig. 31 is a schematic of this test showing the operating temperatures of the various components. The test was started the week of June 17, 1963, and on August 20, 1964, it completed its 12th month of successful operation at design flow and 2000F lithium outlet temperature from the reactor simulator. The lithium and NaK loops of this test were thermal-shocked several times when the system was shut down to repair the electrical system. No maintenance or repair has been required for the liquid metal system at any time since the start of the test. A total of 10,000 operating hours will be completed on October 11, 1964. Fig. 32 is a photograph of the heating loop which simulates the reactor by electrical I^2R heating. The electrical input to this loop is 74,500 amps at 67 volts. The lithium circuit is fabricated from columbium alloy and the lower temperature NaK system is fabricated from type 316 stainless steel. The practicality of designing, fabricating and operating full size lithium-columbium systems is conclusively demonstrated in this test. A comprehensive report, Ref. 16, has been issued covering this test from design through 4200 hours of operation.

STAINLESS STEEL TESTS IN Na &NaK
P&WA 1953-1958

TEMPERATURE RANGE 1000F TO 1600F

	TEST HOURS
UO_2-SS IRRADIATION CAPSULES	40,000
RADIATOR CORE	7,500
RADIATOR - OTHER	14,600
HEAT EXCHANGERS	500
PIPES, PIPE JOINTS	5,750
VALVES	4,000
PUMPS, CENTRIFUGAL	320
CORROSION TESTS (FORCED CONVECTION)	28,700
CORROSION TESTS (THERMAL CONVECTION)	9,000
MISC. TESTING (THERMAL SHOCK, CREEP, PLASTIC STRAIN, BEARINGS AND SEALS)	16,000

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FIG 13

COMPARISON OF REACTOR COOLANT PROPERTIES

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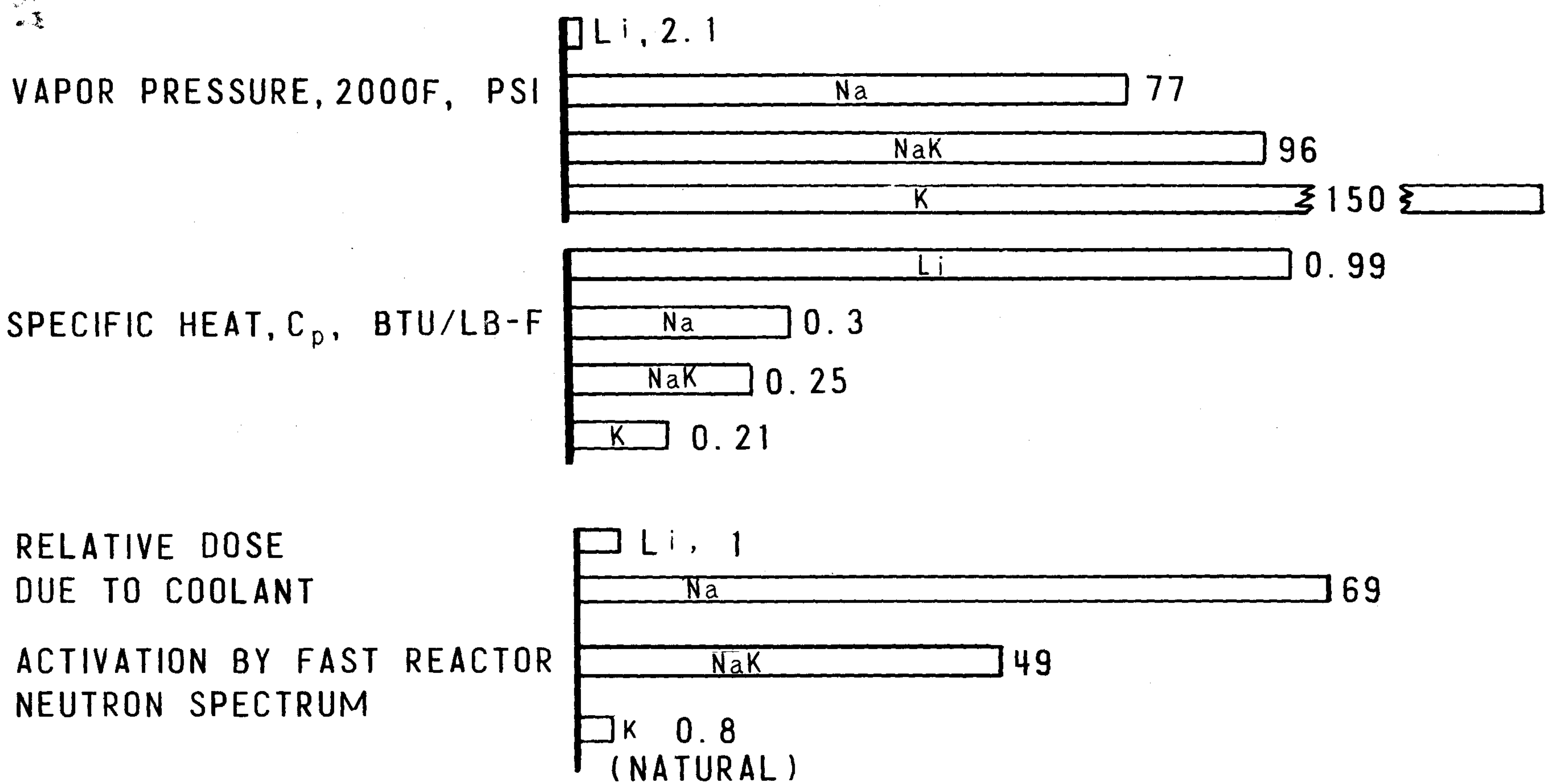

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FIG 14

STRUCTURAL ALLOYS EVALUATION

ALLOY	MAXIMUM USEFUL TEMP. F	LITHIUM CONTAINMENT 2000F	DUCTILITY AND FABRICABILITY
STAINLESS STEEL	1500 - 1600	VERY POOR	EXCELLENT
NICKEL AND COBALT	1600 - 1850	VERY POOR	FAIR
TITANIUM AND ZIRCONIUM	800 - 1200	EXCELLENT	EXCELLENT
COLUMBIUM	2000 - 2200	EXCELLENT	EXCELLENT
TANTALUM	2200 - 2600	EXCELLENT	GOOD
MOLYBDENUM	2200 - 2400	EXCELLENT	POOR
TUNGSTEN	2800 - 3200	EXCELLENT	POOR

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FIG 15

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FABRICATION PROGRESS
SNAP-50/SPUR STRUCTURE COMPONENTS

FY 1963-1964

	ACHIEVED		
	LAB	PILOT	PRODUCTION
WELDING			
Cb-1 Zr	●	●	●
PWC-11 (Cb-1 Zr-0.1C)	●	●	
D-43 (Cb-10 W-1 Zr-0.1C)	●	●	
DIE FORGINGS			
Cb-1 Zr			●
MILL PRODUCTS			
Cb-1 Zr			●
Cb-1 Zr-0.6 C			●
D-43 (Cb-10 W-1 Zr-0.1 C)			●
T-111 (Ta-8 W-2.5 Hf)			●

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FIG 16

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COLUMBIUM ALLOY PARTS FABRICATION

150 TONS PROCESSED AT CANEL
1958 TO SEPT. 1964

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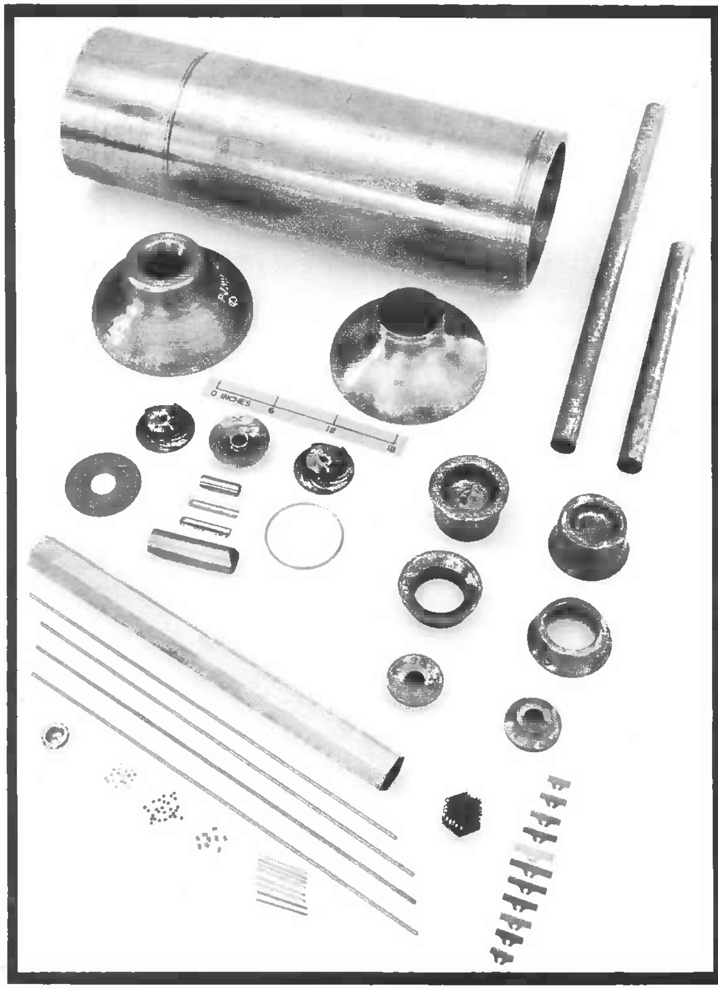
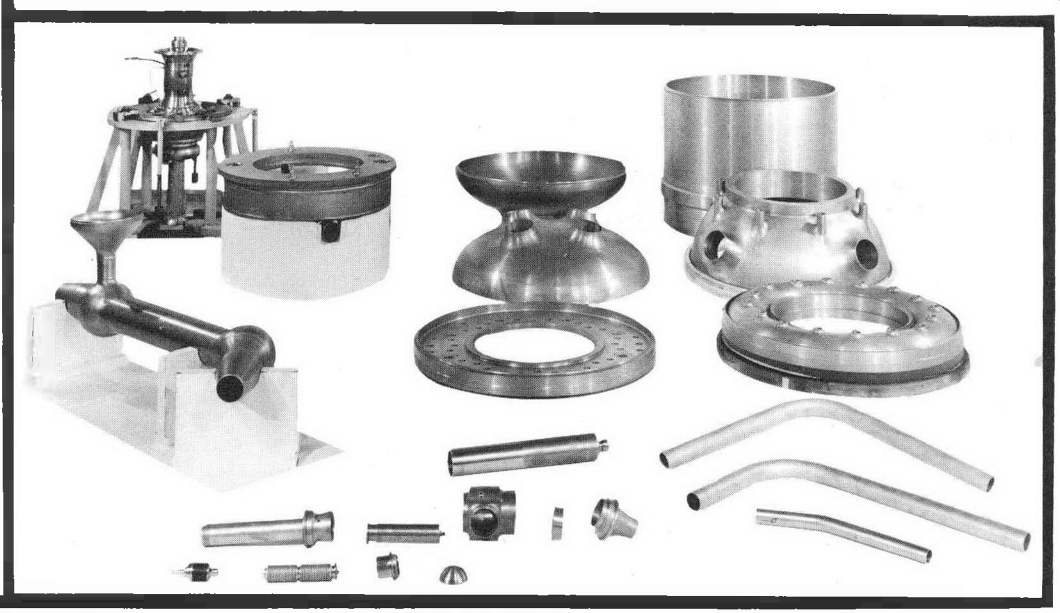
FIG 17





CANEL CLEAN ROOM FACILITY

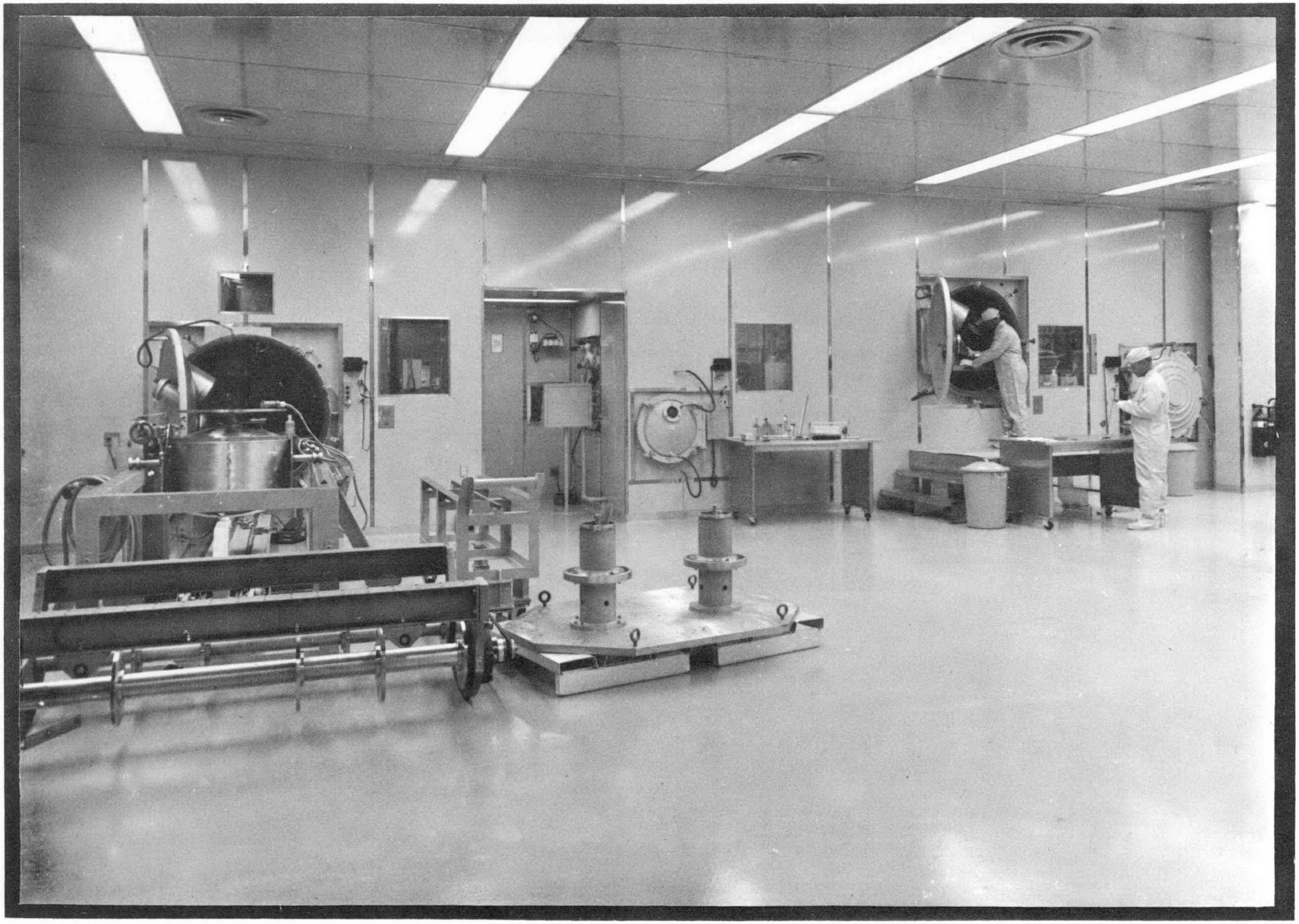


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FIG 18

[redacted]

CANEL REFRACTORY METAL WELDING FACILITY

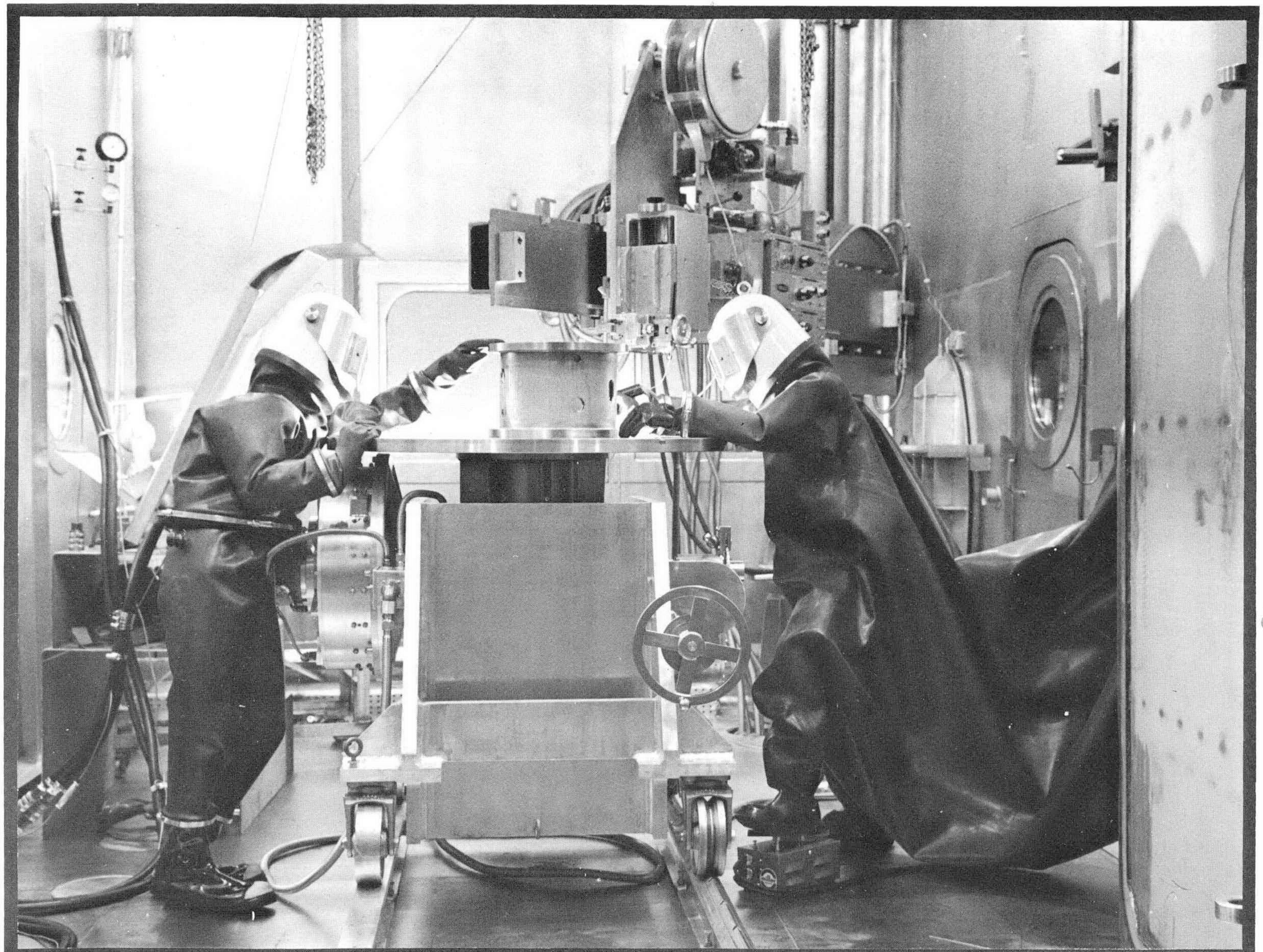

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FIG 19

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[redacted]

Cb-Li TECHNOLOGY STATUS
SEPT. 24, 1964

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	NO. OF TESTS	TOTAL TEST HRS	BEST PERFORMANCE TEMP. F	BEST PERFORMANCE TIME. HR
FUEL IRRADIATION TESTS				
UC	51	34,200	2100	2,560
UN	27	13,200	2225	2,750
UO_2-BeO	66	68,800	1700-2000	10,000
Cb-UO_2 AND UN CERMETS	48	11,059	2000	1,000
STRUCTURAL TESTS				
TUBE BURST	87	137,300	2000	18,000
4-INCH PRESSURE VESSEL	17	69,200	2000	10,000
10-INCH PRESSURE VESSEL	3	11,874	1900 - 2000	10,000
FULL SCALE VESSEL	1	10,000	2000	10,000
CORROSION TESTS				
DYNAMIC LOOP	38	42,115	2000	10,000
			2200	2,000
			2400	500
CENTRIFUGAL PUMPS 200 GPM	5	43,378	1000 - 1300	10,000
			1000 - 1600	6,600
HEAT EXCHANGERS				
Li TO Li, NaK	2	19,200	2000	9,600

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FIG 20

Cb-Li TECHNOLOGY STATUS (GRAPHICAL) SEPT. 1964

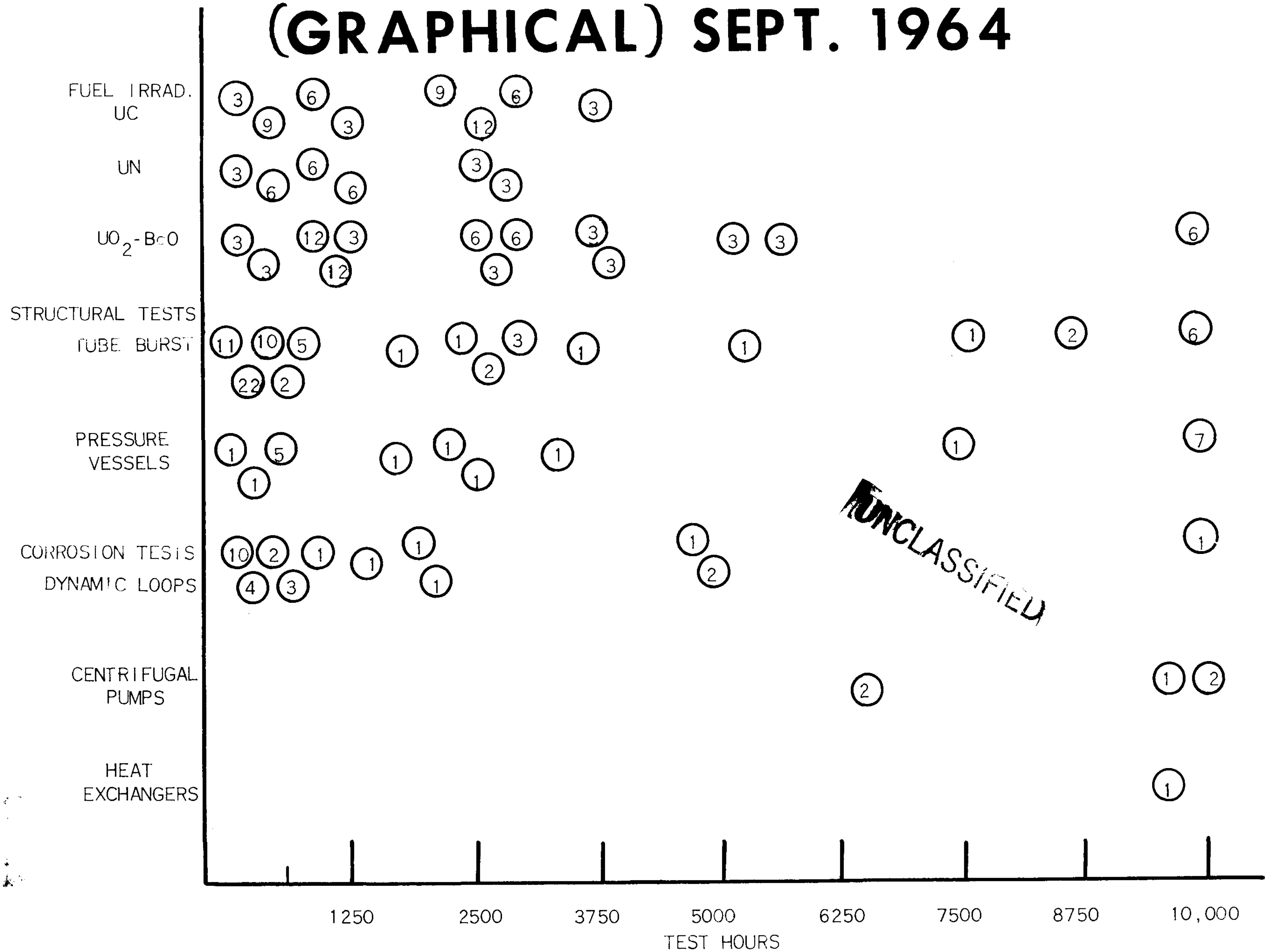

FIG 21

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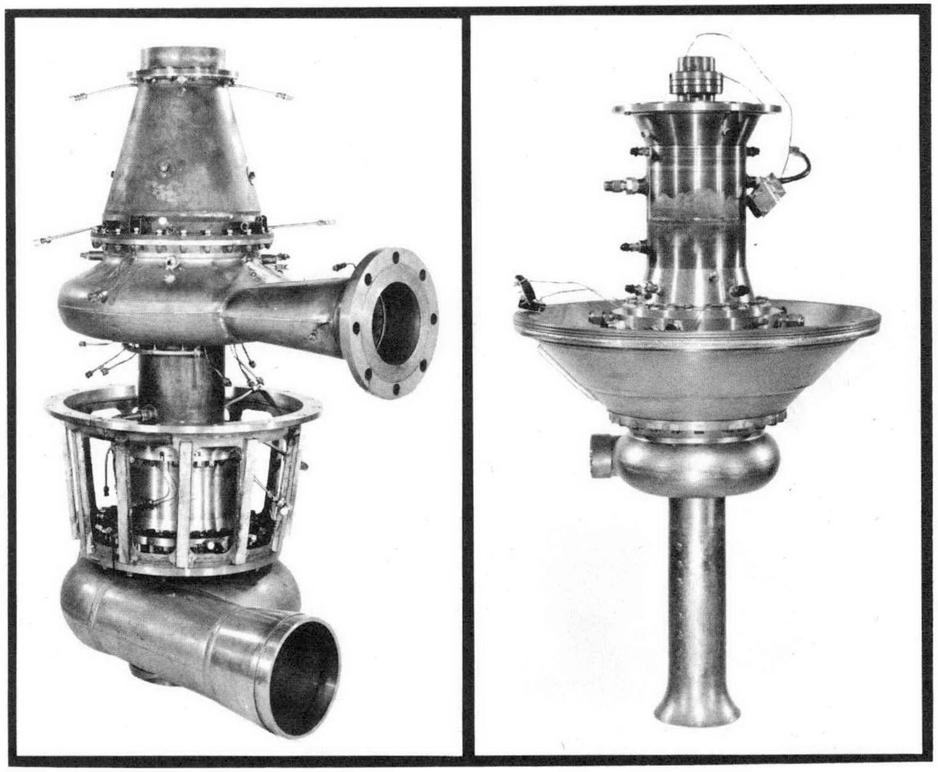
NaK, K, & Li PUMP DEVELOPMENT



STAINLESS STEEL & Cb ALLOY

AIR TURBINE & ELECTRIC MOTOR DRIVES

50 - 3000 GPM 100 - 450 FT

700 - 1600F 5-350 SHP

17 TESTS 47,781 HRS

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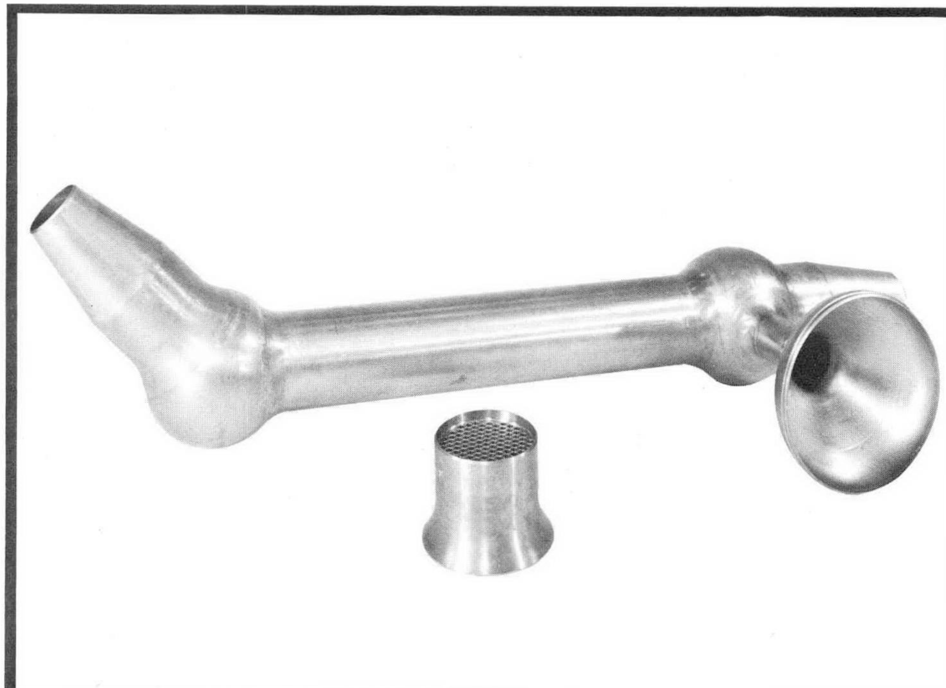
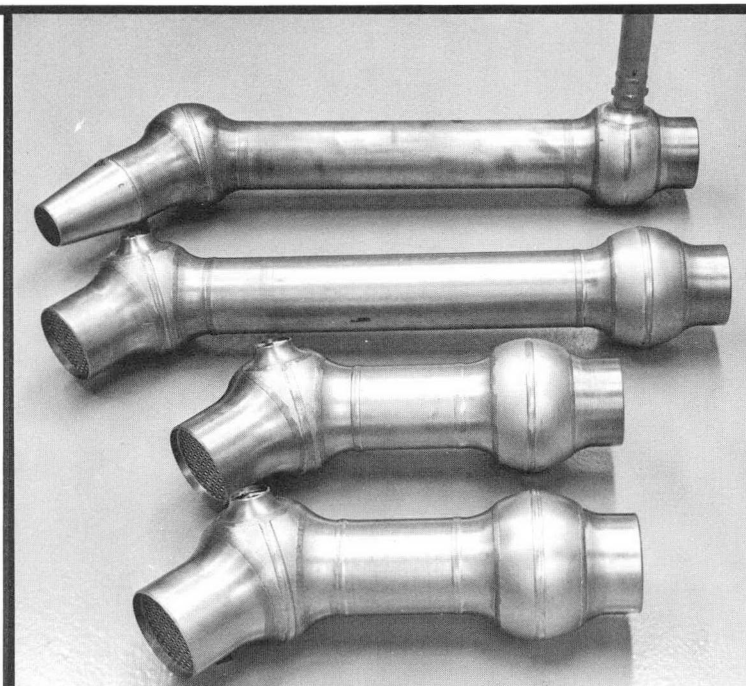
FIG 22

HEAT EXCHANGER DEVELOPMENT





LITHIUM. SODIUM. NaK

Cb-1 Zr, 316 SS, HS-25, INCONEL

PERFORMANCE AND ENDURANCE TESTS

1500F - 2000F UP TO 8.5 M W

6 TESTS 21,886 HRS.

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FIG 23

SUMMARY OF TWO PHASE POTASSIUM TESTS

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TYPE	TUBE	PURPOSE	TEMP. LEVEL. F		TEST TIME. HRS.
THERMAL CONVECTION	PLAIN	MATERIAL INVESTIGATION	2000	Cb ALLOYS	9,200
FORCED CONVECTION	SERPENTINE	BOILING HEAT TRANSFER DATA	1800	HS-25	3,625
FORCED CONVECTION	TUBE INSERTS	TWO-PHASE SYSTEM PERFORMANCE	1600	316 SS	1,292
				TOTAL HOURS	14,117

FIG 24

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CONDENSER SECTION OF BOILING POTASSIUM HEAT TRANSFER RIG

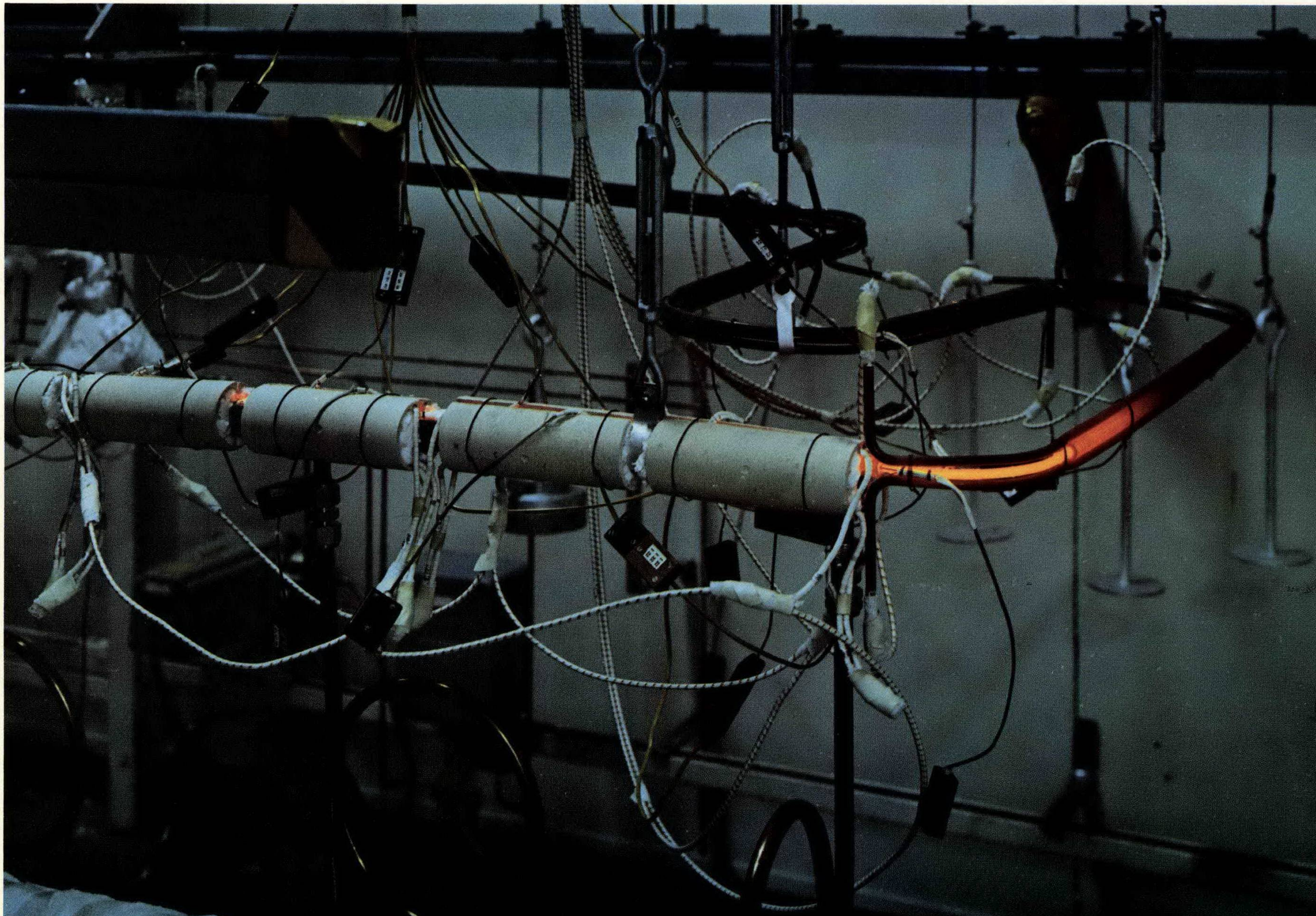

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FIG 25

POTASSIUM BOILER AFTER 3625 TEST HOURS

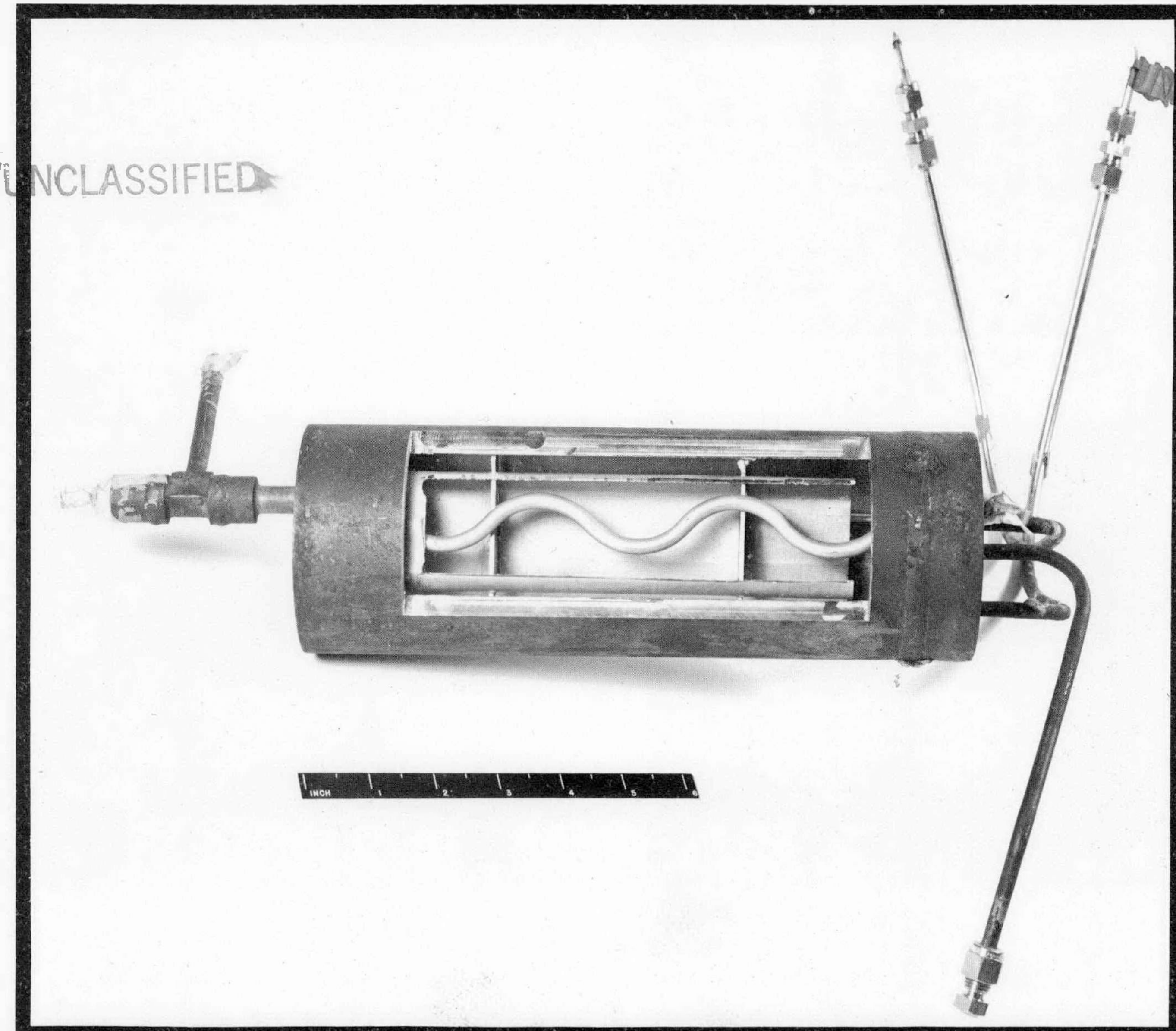

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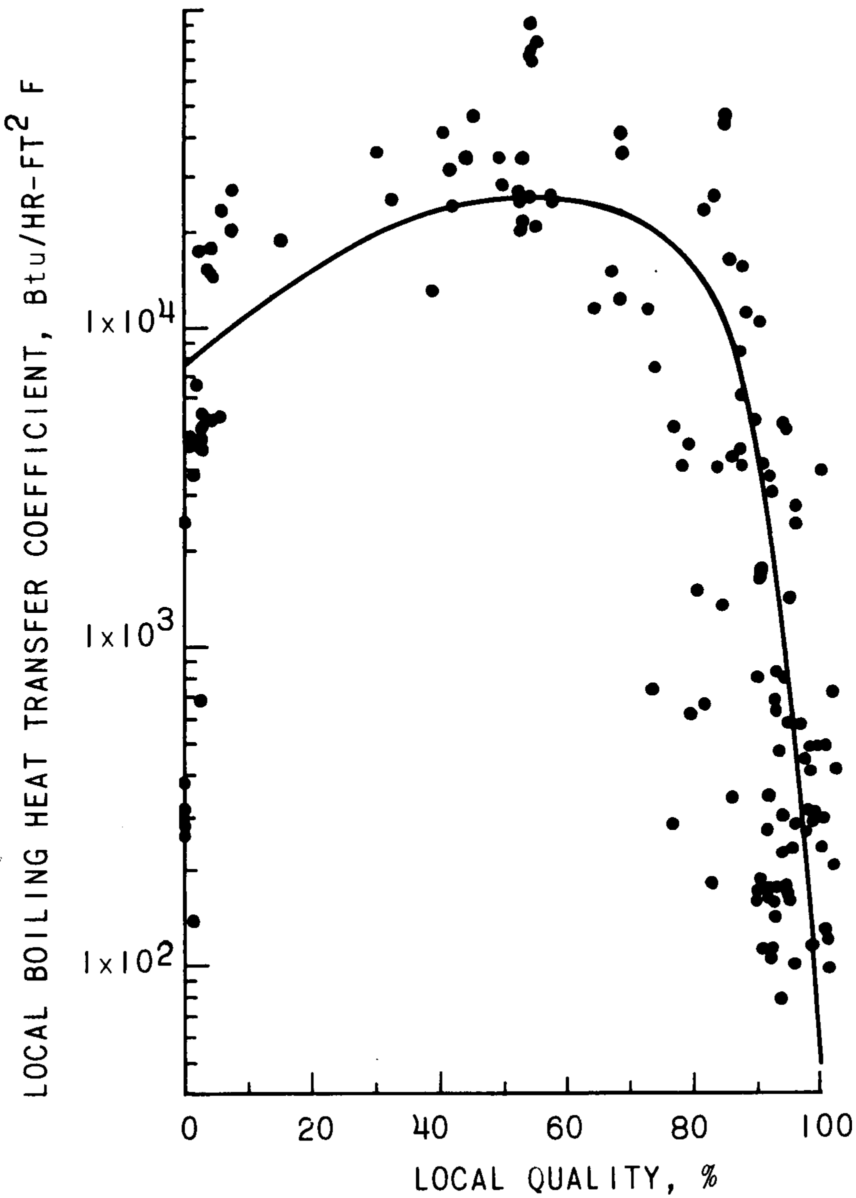
FIG 26

TWO PHASE LOCAL HEAT TRANSFER FOR POTASSIUM



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TUBE GEOMETRY-SERPENTINE
TUBE ID-0.186"
TUBE OD-0.250"
WAVE LENGTH-3"
AMPLITUDE-3/8"
OVERALL LENGTH-10.35"
MATERIAL-HAYNES 25

HEAT FLUX-17,000-52,000Btu/HR-FT^2
VAPOR TEMPERATURE-1625-1725F
VAPOR PRESSURE-42-62 PSIA
POTASSIUM FLOW RATE-17-25 LB/HR

FIG 27

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FULL-SIZE SERPENTINE TUBE BOILER MODEL

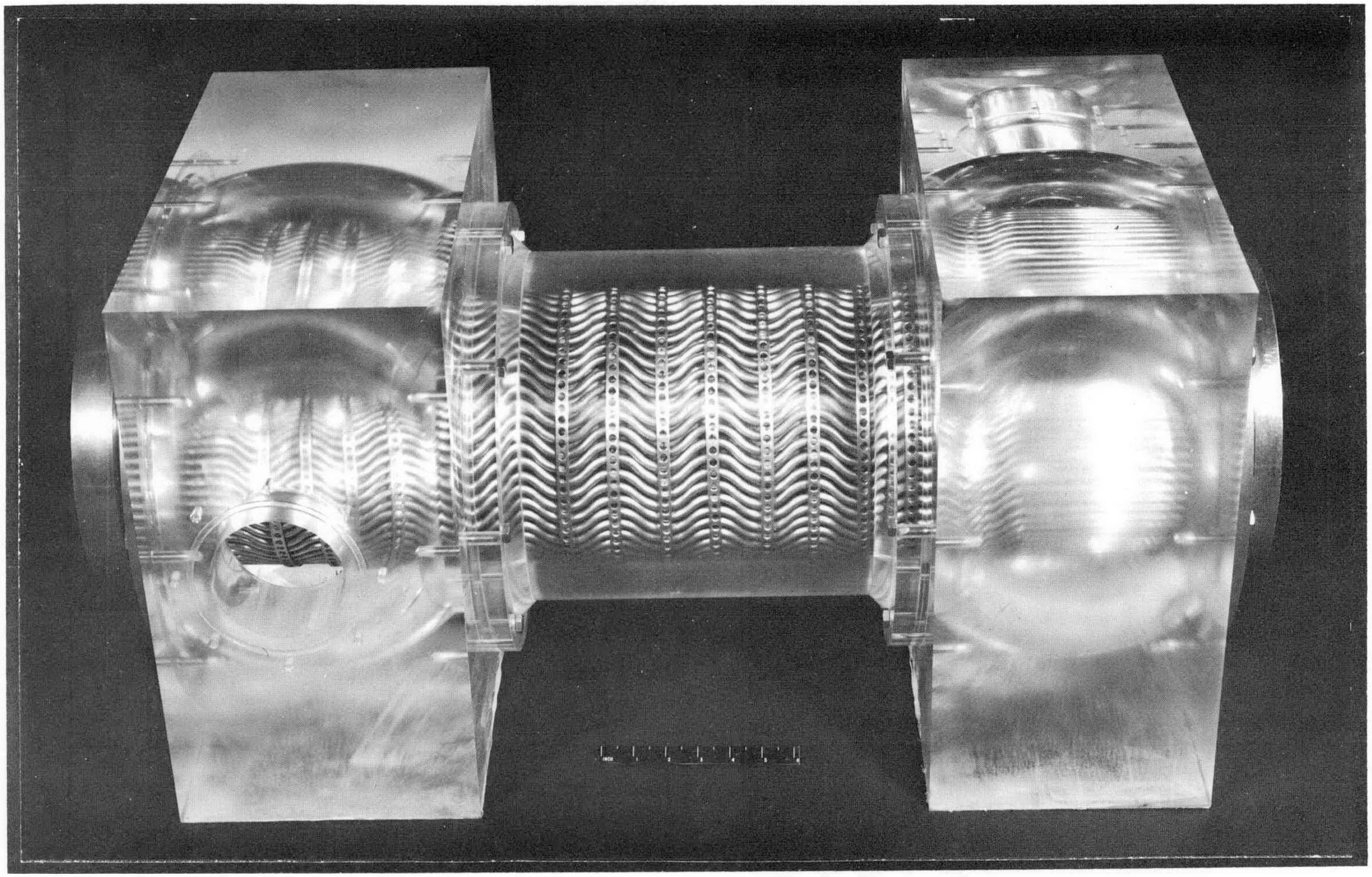

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FIG 2.8

TEN INCH PRESSURE VESSEL AFTER 10000 HOUR TEST

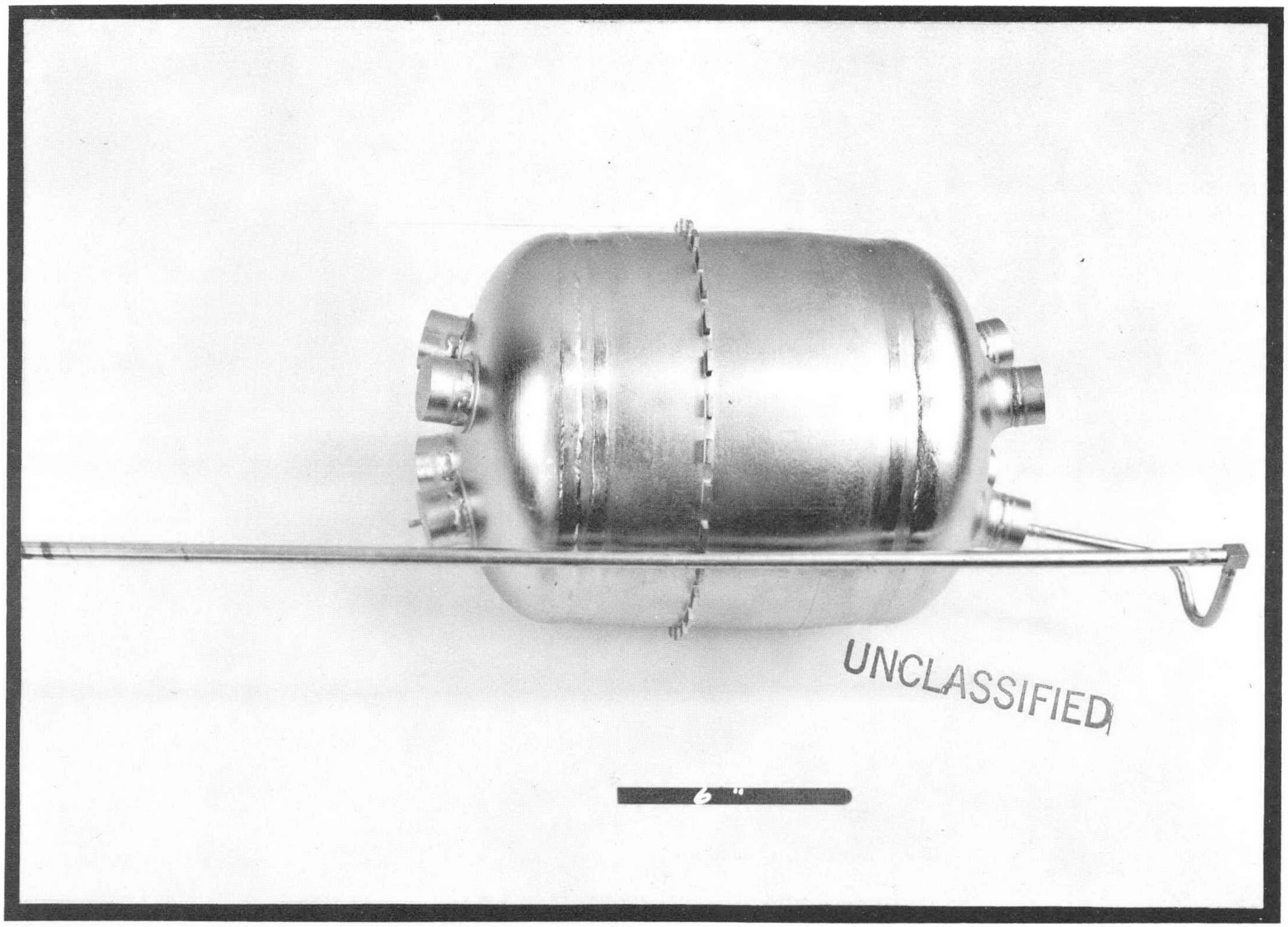

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FIG 29

[redacted]

LCRE FULL SCALE PRESSURE VESSEL AFTER 10,000 HOUR TEST

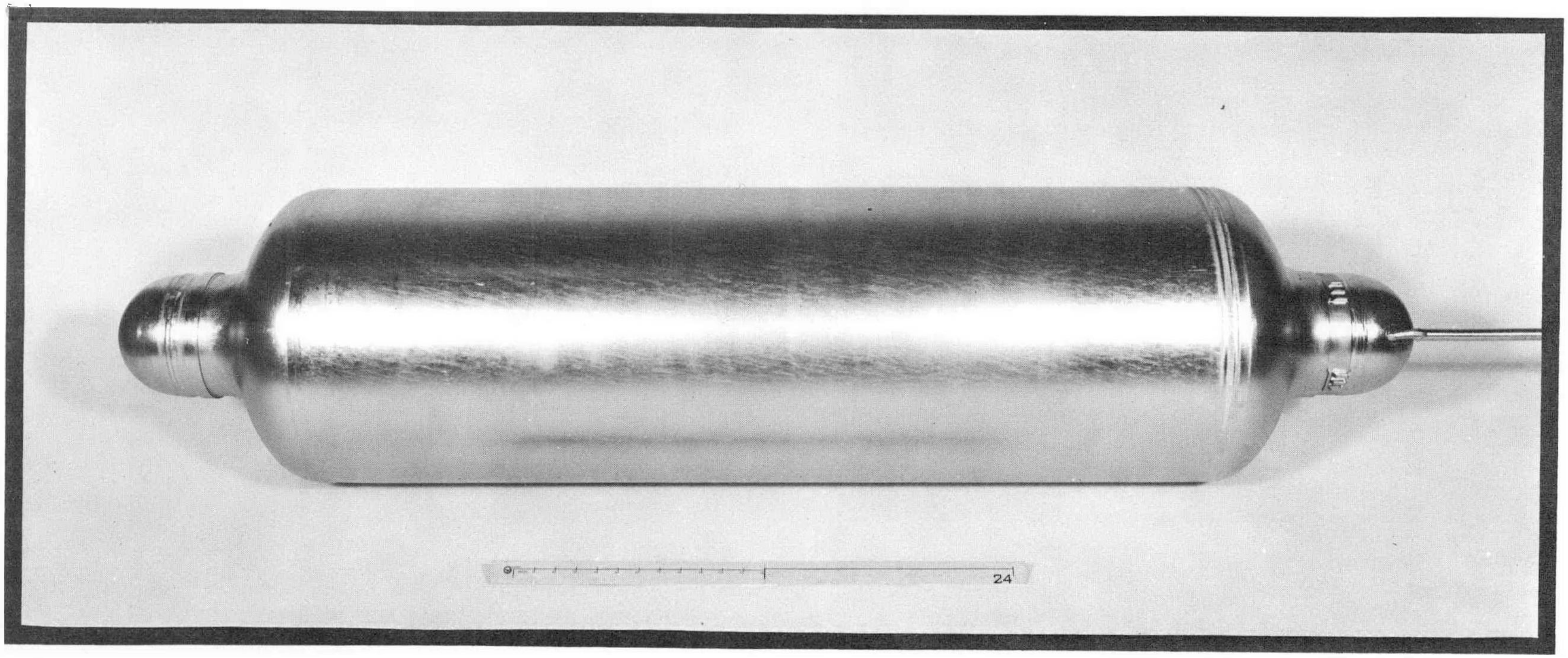

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FIG 30

LCRE 5 Mw SYSTEM TEST

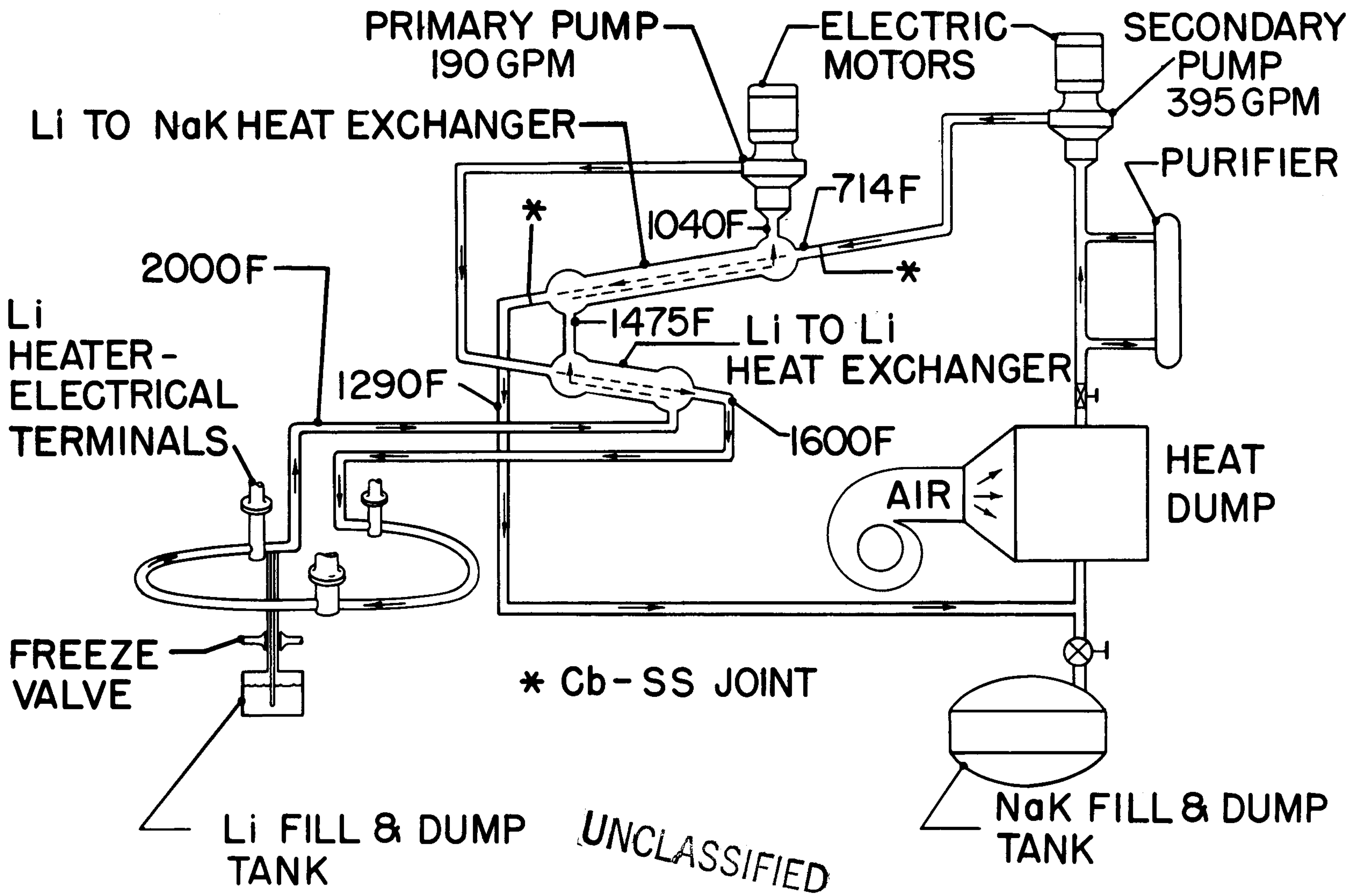

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FIG 31

LCRE 5 Mw SYSTEM TEST Li HEATER

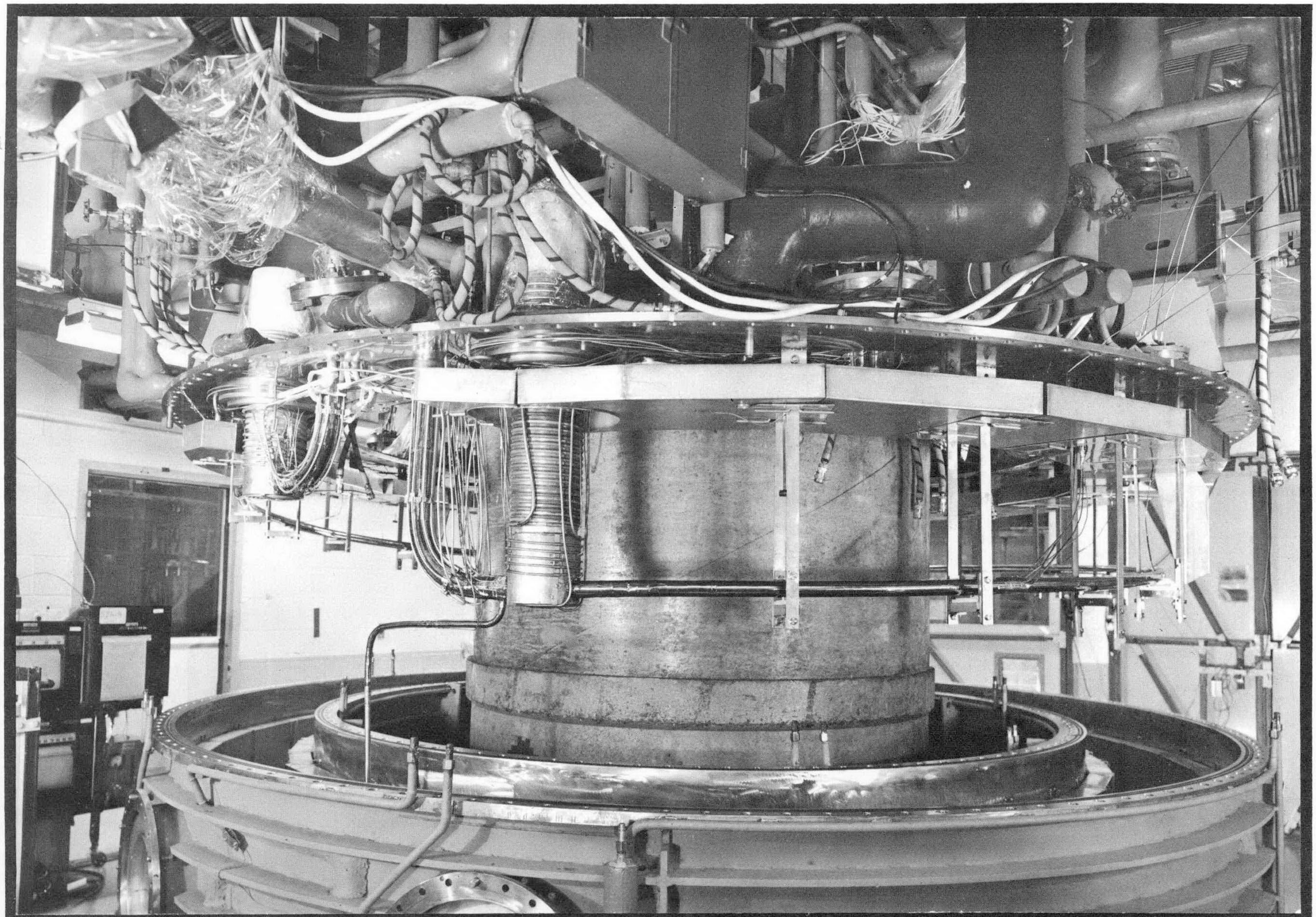

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FIG 32

00355

V CURRENT PROGRAM

V CURRENT PROGRAM

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PWA CANEL

 [redacted]

V. CURRENT PROGRAM

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Pratt and Whitney Aircraft

A. System Engineering and Major System Tests

1. System Engineering Management of the SNAP-50/SPUR development program consists of technical direction, cost and schedule control, and reliability assurance. Technical direction of research programs and systems integration of design and development are guided by powerplant optimization and performance studies.

   Specifically, powerplant optimization and performance studies are performed to establish the design conditions of the powerplant. This work includes studies of powerplant control, off-design performance and startup, The requirements for the 300-Kwe reference design described in Section II of this report have been defined and preliminary drawings have been completed. A 1-Mwe design study is now in progress to define the SNAP-50/SPUR design requirements at the upper end of the 300 to 1000-Kwe range. Powerplant configuration studies are being carried out to define the vehicle powerplant interfaces and to evaluate operational requirements.

   The attainment of reliability goals is based on:

   a) Thorough knowledge of the properties of the materials to be used.

   b) A meticulous design effort, supported by subcomponent development during the design stage to provide accurate information on which to base the design choices.

   c) Meticulous control of all material procurement from ingot melting to availability in shop forms.

   d) Rigid controlled manufacturing procedures with detail inspection at prescribed intervals during manufacturing.

   e) Extreme attention to detail during the assembly, check-out, test planning and operational phases of devel opment.

   f) Extensive component testing to determine failure modes early in the program in order to delineate the problems.

   The Program Evaluation and Review Technique (PERT) is utilized for detailed planning and schedule control of the powerplant program. All development work is outlined and progress monitored using this system. This function is administered by the Schedule Control Group, responsible to the Program Manager. Program cost control data is collected, evaluated and prepared for use by Program Managers as part of the Budget Group responsibilities. Program cost is compared to PERT reported progress and are monitored to insure program schedules are being kept within funding limitations.

2. Major System Tests

   Development of the powerplant calls for the operation of a number of full-scale tests (Fig.10), including pumps, boiler, condenser, turboalternator, reactor and power conversion machinery system. All of these tests will be performed in a vacuum environment in order to simulate space operations and to provide oxidation protection for the refractory metal components. A vacuum level of 10^{-3} to 10^{-5} torr is adequate for space simulation, however, the metal protection requirement necessitates a 10^{-7} to 10^{-9} torr vacuum test level.

[redacted]

a) Boiler and Condenser Test

Boiler and condenser tests are included in the development program to verify component performance and structural design. The Heat Exchanger Laboratory (Fig. 33) will be modified to contain two test stands as shown in Figs. 34 and 35. A line schematic of the test circuit including approximate operating condition is shown in Fig. 36. An electrical I^2R heater and a facility heat dump are used in place of the powerplant reactor and space radiator respectively.

b) Turboalternator Test

Full-scale turboalternator tests at powerplant conditions are planned in the non-nuclear powerplant test facility in the Radiator Laboratory (Fig. 37) and in the Boiler Test Facility. By necessity, the turboalternator test will also include a powerplant boiler and condenser. Heat would be supplied to the system by an electrical I^2R heater and rejected from the system by a liquid metal-to-air heat exchanger. The test system will be very similar to the boiler-condenser test. The turboalternator would replace the turbine simulator. In addition, loading devices to absorb the electrical output of the alternator would be included.

c) Non-Nuclear Powerplant Test

Tests of the non-nuclear powerplant are also planned for the radiator laboratory in the second cell. The non-nuclear powerplant will contain all powerplant components except the reactor and possibly the radiator (Fig. 38). The components will be arranged and integrated in the flight configuration. Later tests will contain flight instrumentation and controls and will be operated to simulate space missions.

d) Reactor and Nuclear Powerplant Tests

In order to demonstrate the feasibility of SNAP-50/SPUR technology, a reactor test (Fig. 39) followed by a nuclear powerplant test will be performed in the Nuclear Test Facility which will be located at the CANEL site, Middletown, Connecticut. Due to the nature of the tests, the facility design criteria stress personnel safety, system reliability and containment of any potential credible accident. The facility will be capable of accommodating reactor tests in the power range of up to 10 Mw thermal and nuclear powerplant tests of up to 1 Mw electric at temperatures of up to 2000F. The reactor and nuclear powerplant tests, as well as all of the facility support systems, are based on 10,000 hours of endurance test time plus additional time required for pretest and post-test operations. The facility will utilize existing CANEL support facilities, services, and utilities.

An Architect-Engineering firm, the Catalytic Construction Company, supported by Nuclear Utilities Services, Incorporated, was selected by the Atomic Energy Commission to perform pre-Title I and Title I engineering and design services for the Nuclear Test Facility. Pratt & Whitney Aircraft-CANEL is now in the process of evaluating various facility concepts proposed by the Architect-Engineer.

HEAT EXCHANGER LABORATORY

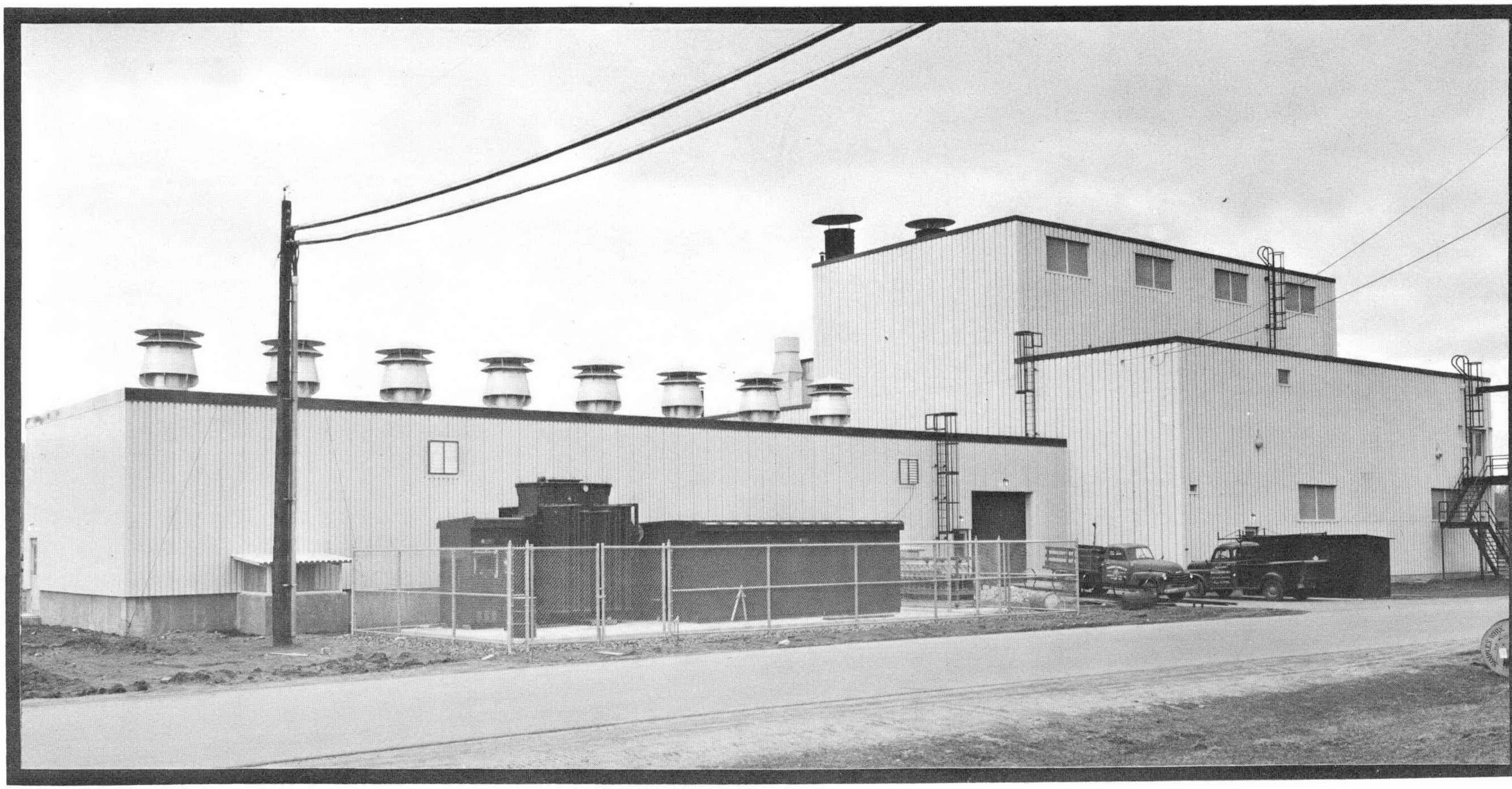

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FIG 33

BOILER-CONDENSER TEST FACILITY - GROUND FLOOR PLAN

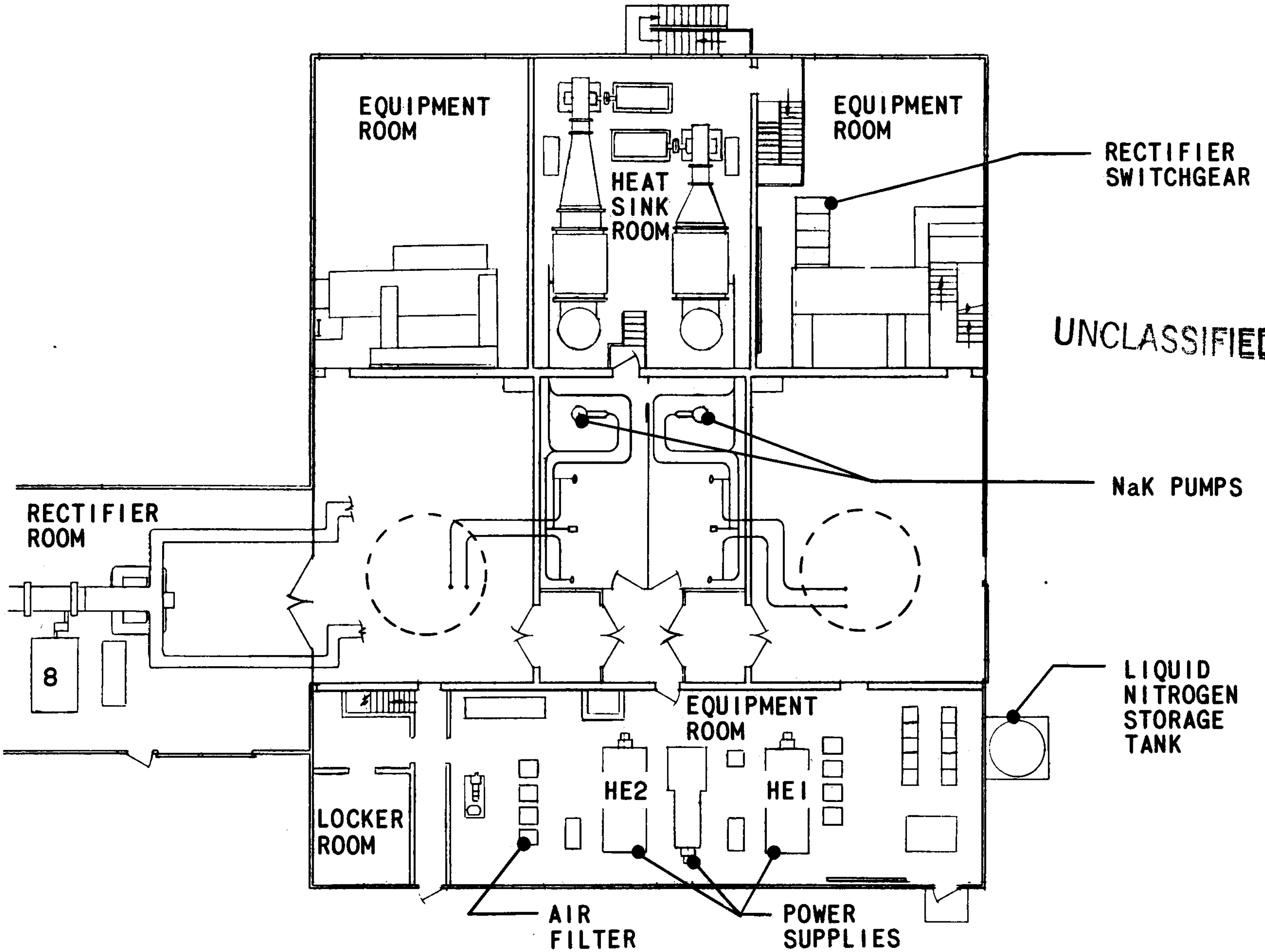

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FIG 34

BOILER TEST FACILITY VACUUM CHAMBER

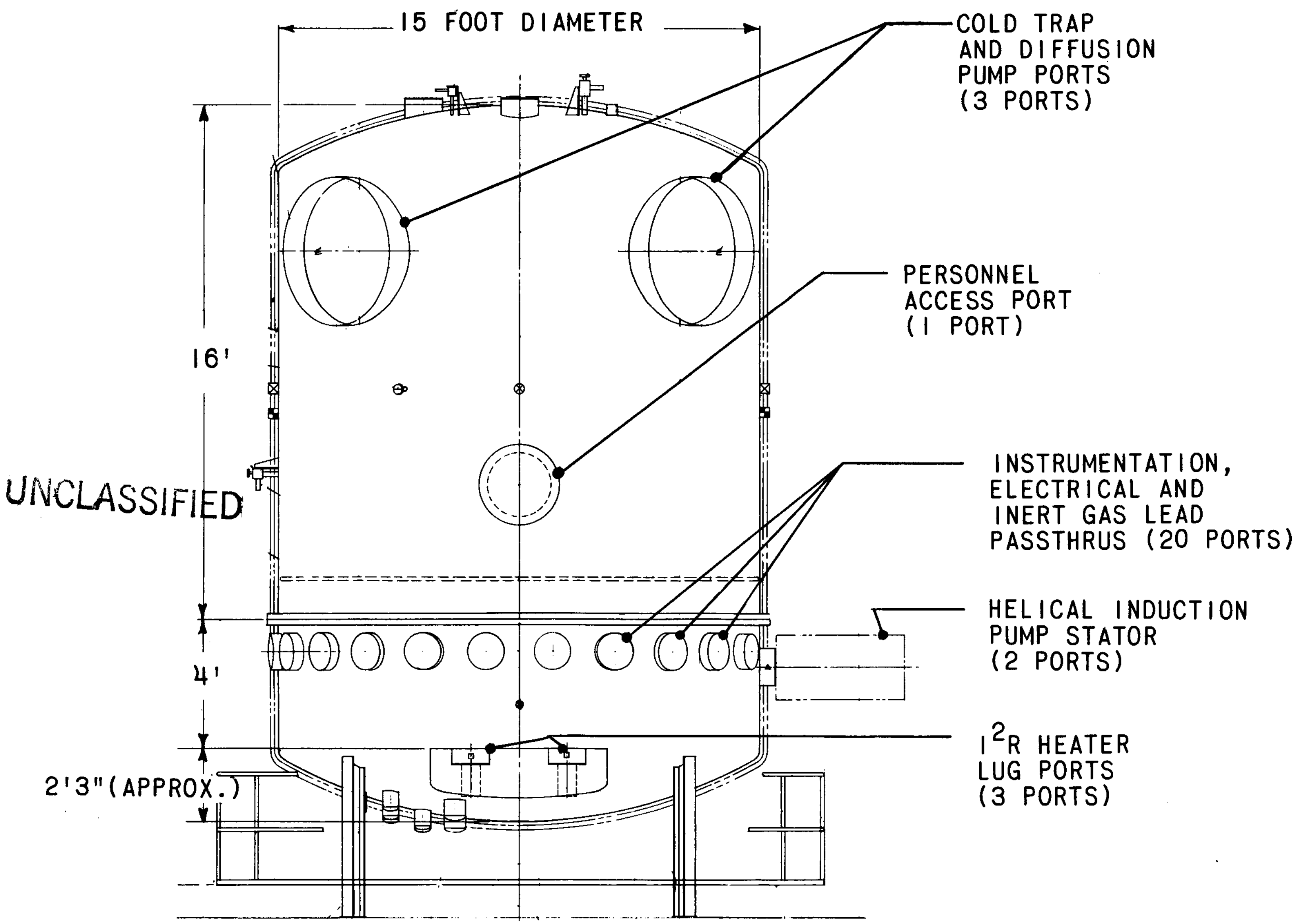

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FIG 35

BOILER-CONDENSER TEST SCHEMATIC

100% POWER CONDITION

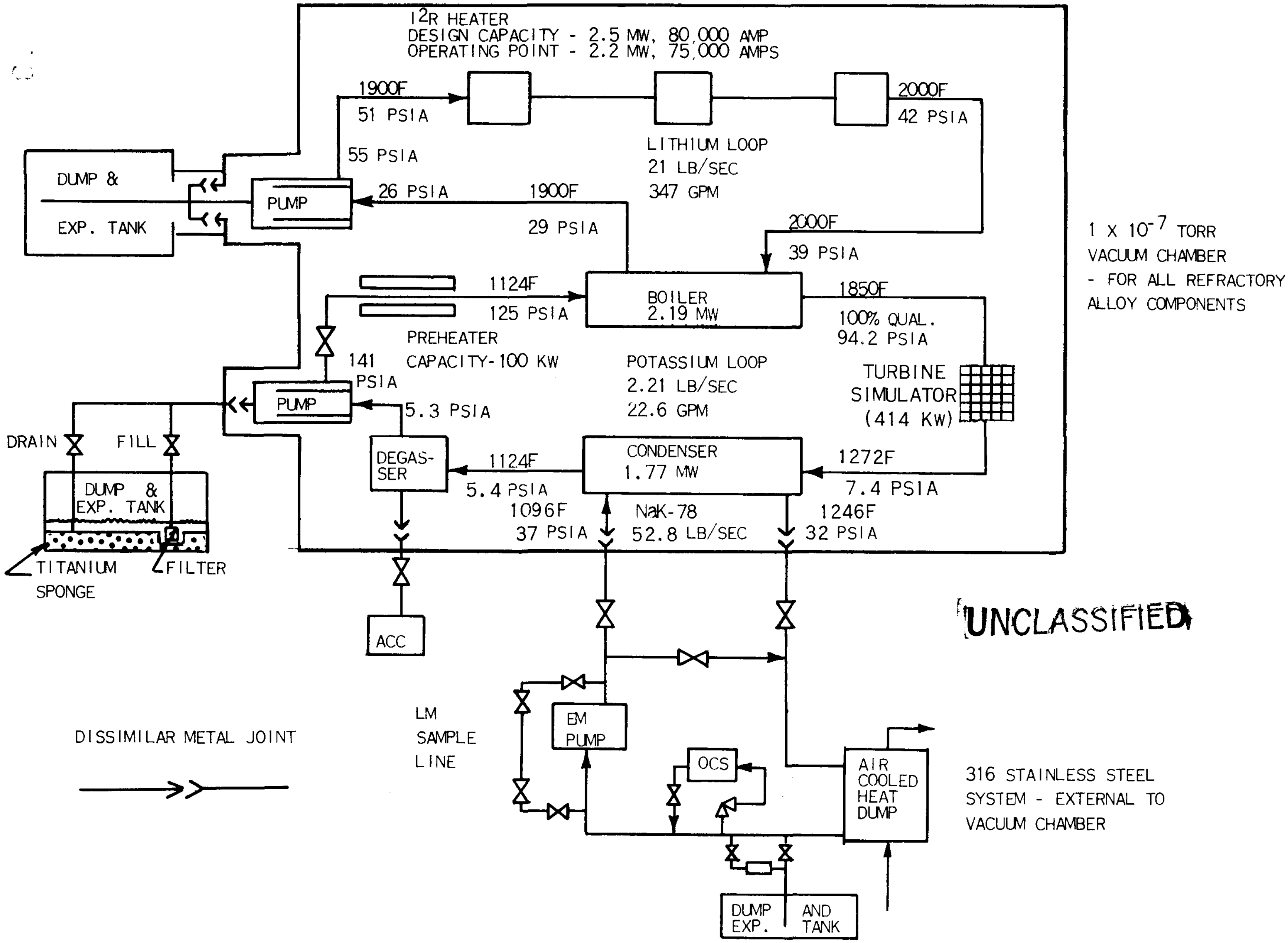

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FIG 36

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RADIATOR LABORATORY

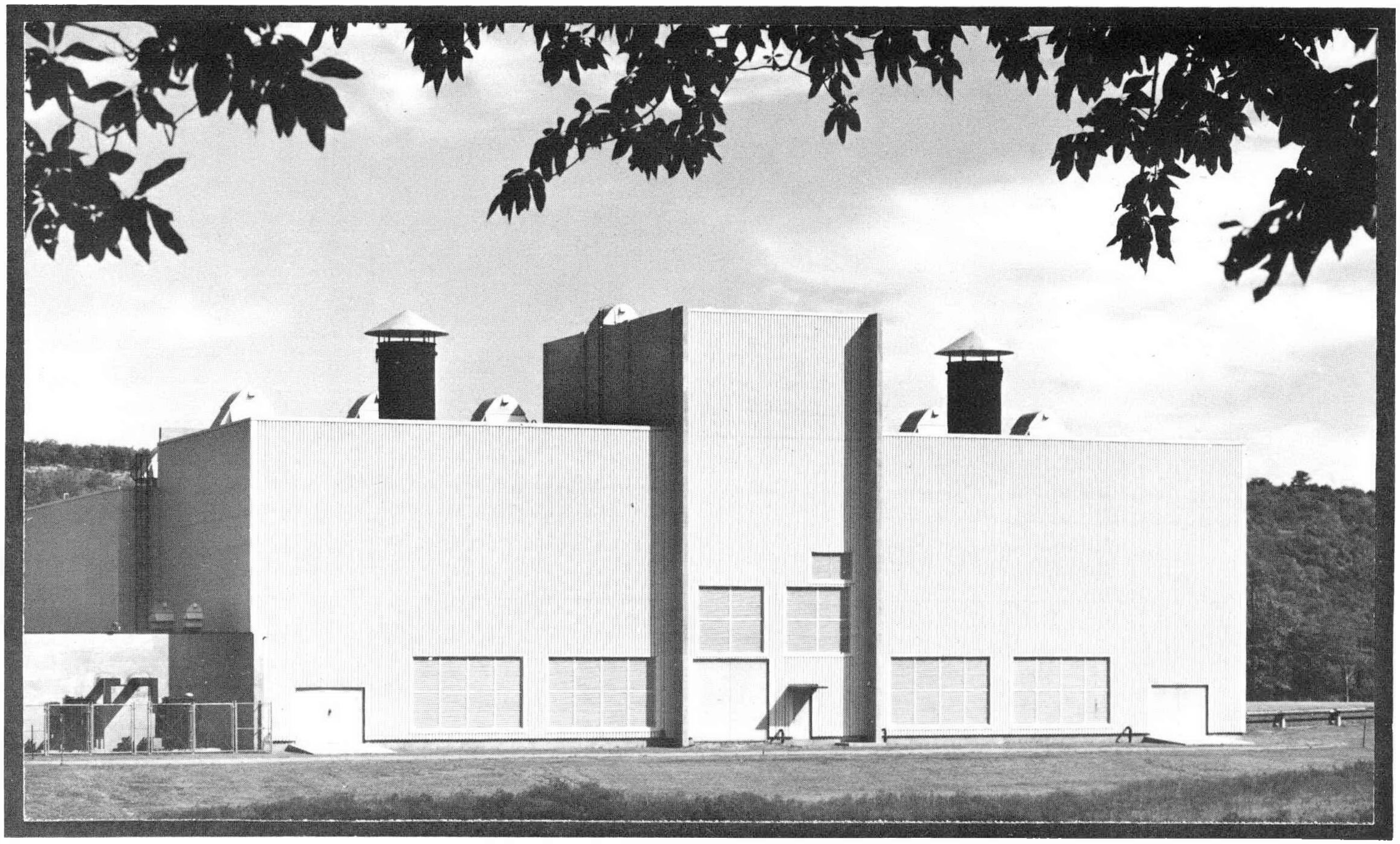

FIG 37

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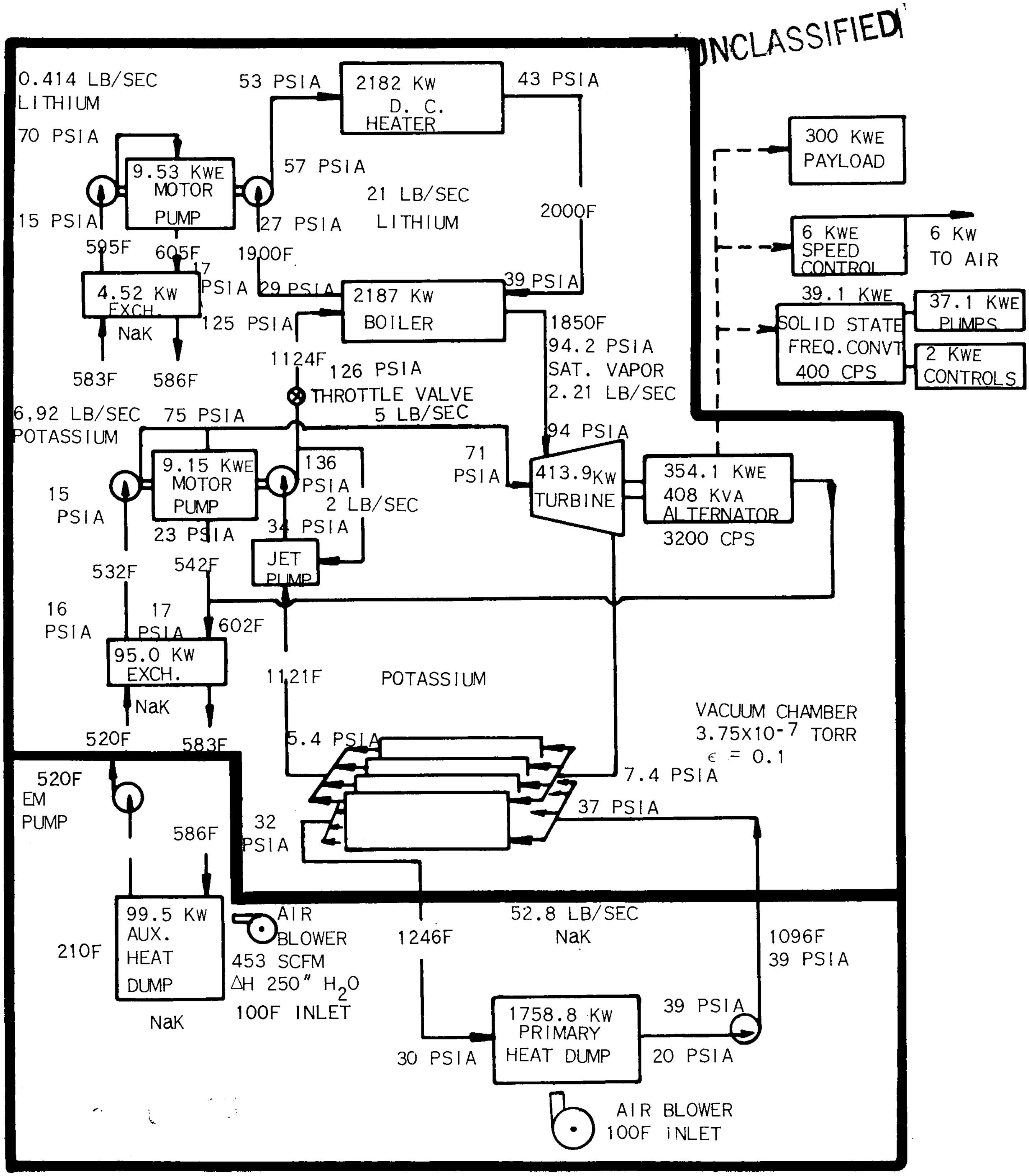
V-7

FIG 38

FLIGHT CONFIGURATION NON-NUCLEAR POWERPLANT TEST
HEAT BALANCE



SYSTEM SCHEMATIC

2 MW REACTOR TEST

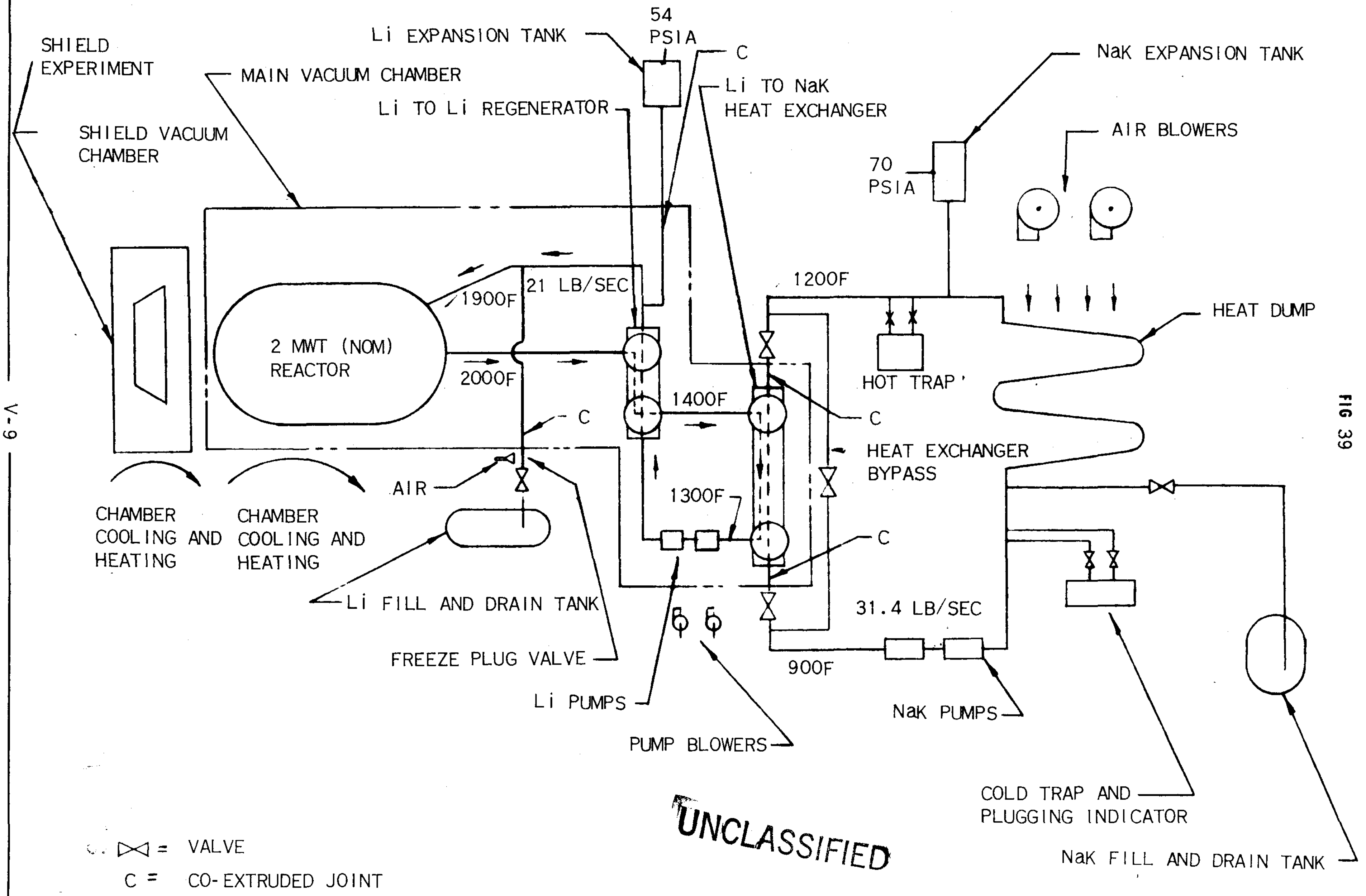


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FIG 39

B. Reactor and Shield

1. Reactor and Shield Design

a. Present Design Status

1) 2-Mw Power Level

a) Reference Design, PWAR-20

(1) Design Criteria

A preliminary design of the PWAR-20 has been completed. The major design criteria were 2-Mw thermal output, 10,000-hour life, separated lithium coolant, 2000F liquid metal outlet temperature and a dose at the payload of 10^{13} nvt >.1 Mev and 10^7 rad gammas. A more complete list of design criteria is given in Fig. 40. The PWAR-20 design is based on conservative assumptions such as 1 a/o maximum permissible fuel burnup and 2300F maximum centerline temperature.

(2) Description of Reactor and Shield

The PWAR-20 (Figs. 41 and 42) is a lithium-cooled reactor which will operate in the fast neutron energy spectrum. It will utilize columbium alloys for the pressure vessel, core support structure, and fuel element cladding, and will use titanium alloys for that portion of the reactor reflector support structure which operates below 1000F. The fuel will be enriched uranium in the form of UC or UN. A summary of the physics and engineering data is given in Fig. 43.

Six pivotted reflector segments (made of blocks of beryllium oxide clad with columbium alloy) are arranged so that in the fully closed position they completely surround the reactor core. These segments are connected to the reactor reflector support structure by means of flexure bearings at each end. They control the reactor by opening or closing to change the neutron leakage.

Each segment is individually driven by a co-linear drive consisting of a stepping motor and a harmonic gear reduction unit. The motors and Harmonic Drives are placed behind the neutron shield and are connected to their respective segments by shafts operating through penetrations in the shield. The shafts and penetrations are stepped to minimize neutron streaming.

The present shield concept is constructed of two regions of LiH, radiatively-cooled and having the shape of a frustum of a circular cone with dished heads on top and bottom. It is designed to allow no more than 10^{13} nvt fast neutrons and 10^7 rad gamma dose to a payload 42 feet from the reactor. The shield and the reactor reflector shadow criteria are shown in Fig. 44.

The first 10-inch thickness of the shield adjacent to the reactor consists of 99.99 percent Li^7H to reduce heating from the Li^6 (n, α) reaction. The remainder is natural LiH. The hydride is cast into the stainless steel shell containing a perforated stainless steel honeycomb and sealed in a helium or hydrogen atmosphere. Patch shielding is required to reduce the radiation scatter from the primary piping to the payload to acceptable levels.

A block diagram of the reactor, primary loop, and reactor control system is shown in Fig. 45. The controlled variable is core outlet lithium temperature. This temperature is sensed by thermocouples located on the lithium piping at

SNAP-50 / SPUR PWAR-20 REACTOR & SHIELD DESIGN SPECIFICATIONS

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REACTOR	
DESIGN POWER, MW (TH)	2.2
FULL POWER LIFETIME, HRS.	10,000
CORE COOLANT OUTLET TEMPERATURE, F	2000
CORE COOLANT TEMPERATURE RISE, F	100
REACTOR COOLANT ΔP, PSI	10
TYPE OF CONTROL	MOVING REFLECTOR-RADIATIVELY COOLED
REACTOR MATERIALS	
STRUCTURE	Cb-1 Zr-0.1C
COOLANT	Li (99.9% Li^7)
REFLECTOR	BeO
FUEL-CLAD	UC/UN; Cb-1 Zr-.06C (BARRIER-0.005-INCH Ta or W)
SHIELD	
TOTAL DOSE AT PAYLOAD	
NEUTRON	$5 \times 10^{11}-10^{13}$ nvt >0.1 Mev
GAMMA	10^7 RAD
MATERIALS	
NEUTRON SHIELD	Li^7H, NATURAL LiH
CLADDING	STAINLESS STEEL

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FIG 40

PWAR-20 REACTOR & SHIELD

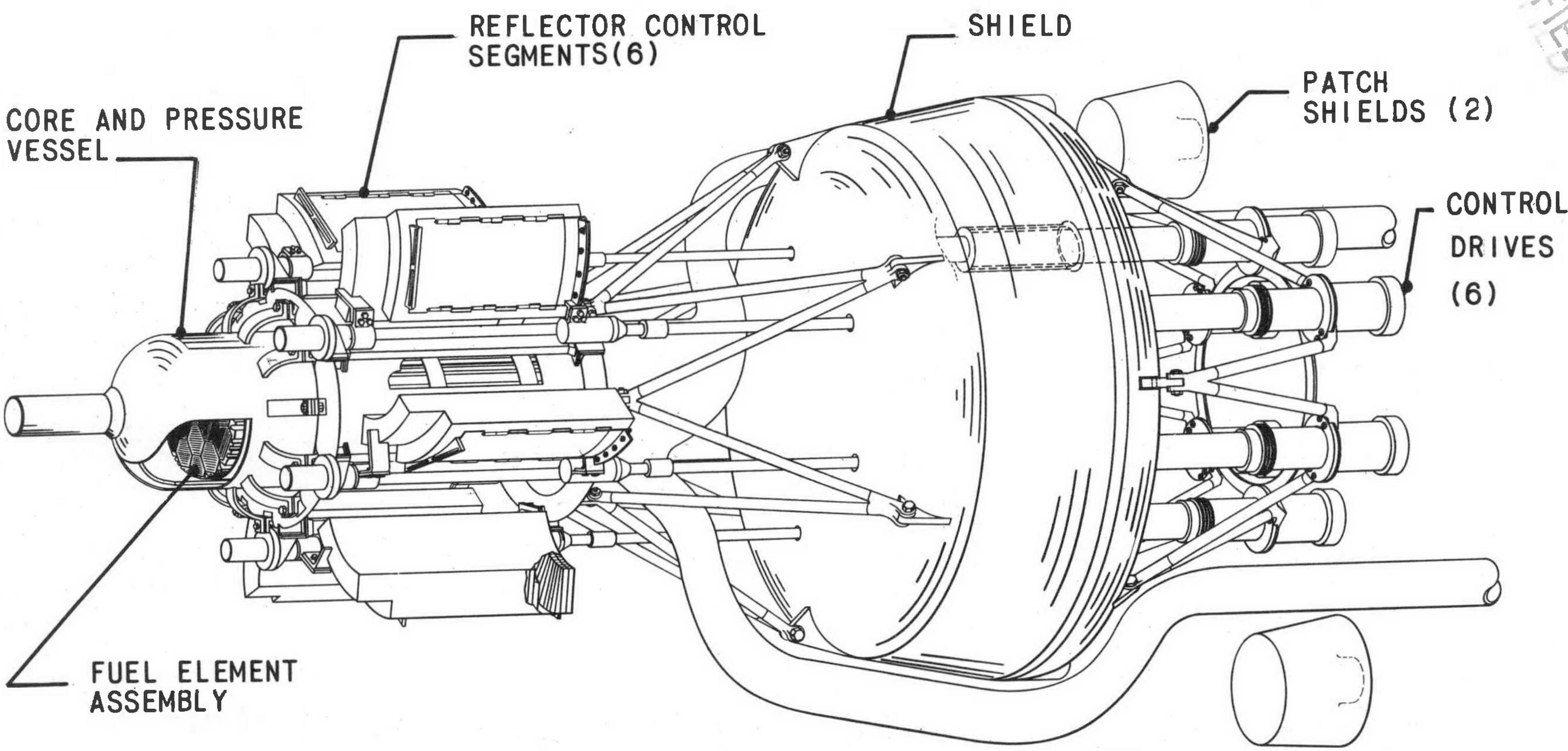

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FIG 41

PWAR-20 REACTOR CONCEPT

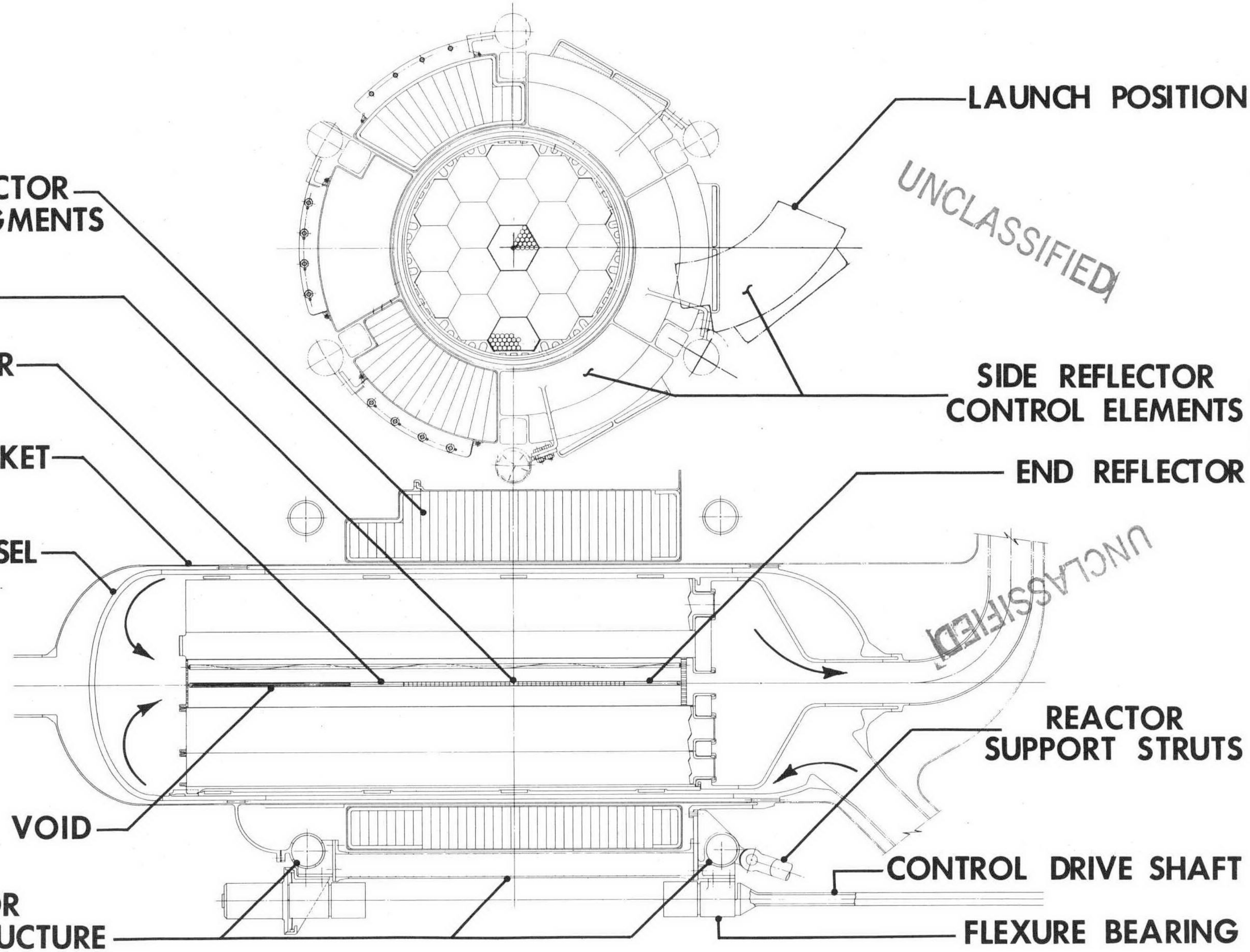

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FIG 42

PWAR-20 DESIGN DATA SUMMARY

00071

PIN CLAD STRENGTH, PSI	1500
REACTOR PRESSURE DROP, PSI	10
FUEL PIN MAXIMUM TEMPERATURE, F (WHCF)	2400
SIDE REFLECTOR TOTAL WORTH, % ΔK	20
REACTIVITY REQUIREMENT, START-UP TO END-OF-LIFE, % ΔK	4.0
EXCESS REACTIVITY, WET/700F, % ΔK	6.0
U^{-235} MASS LOADING, KG	100
FUEL ENRICHMENT, %	63
MAXIMUM FISSION BURNUP, % U	1.0
DESIGN FISSION GAS RELEASE, %	20
DESIGN HELIUM RELEASE, %	25
MAXIMUM FUEL POWER DENSITY AT STARTUP, Kw/cc	0.283
REACTOR DIMENSIONS, INCHES	
PIN CLADDING THICKNESS	0.015/0.025
PIN DIAMETER	0.25/0.35
EFFECTIVE CORE DIAMETER	11.2
CORE LENGTH	12.3
SIDE REFLECTOR THICKNESS	4.0
END REFLECTOR THICKNESS, EACH END	3.0
FISSION GAS VOID LENGTH	8.8/6.8
REACTOR ESTIMATED DESIGN WEIGHT, LBS.	1800

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FIG 43

PWAR-20 SHIELD CONCEPT

(PAYLOAD DOSE: 10^{13} NVT)

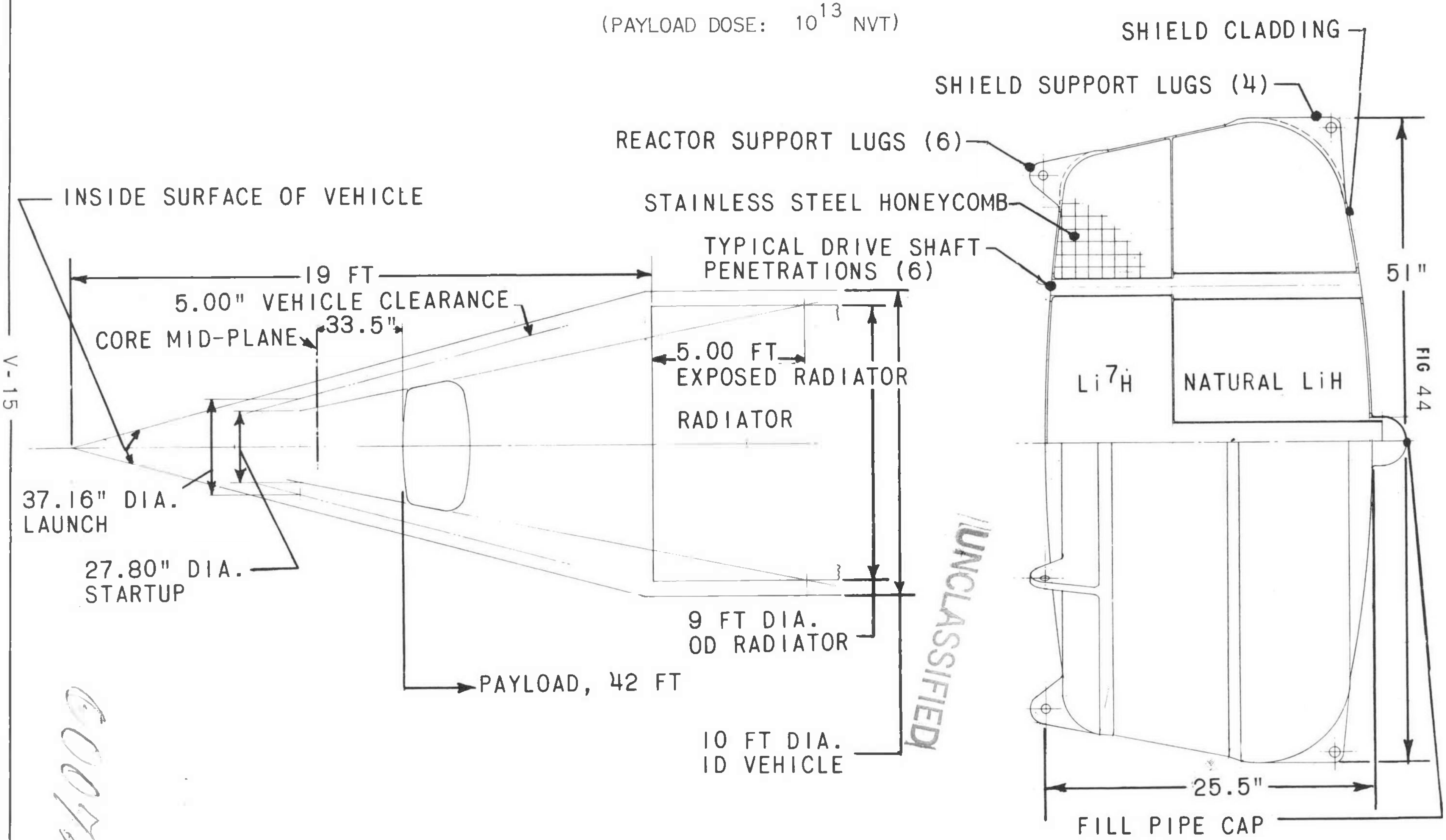

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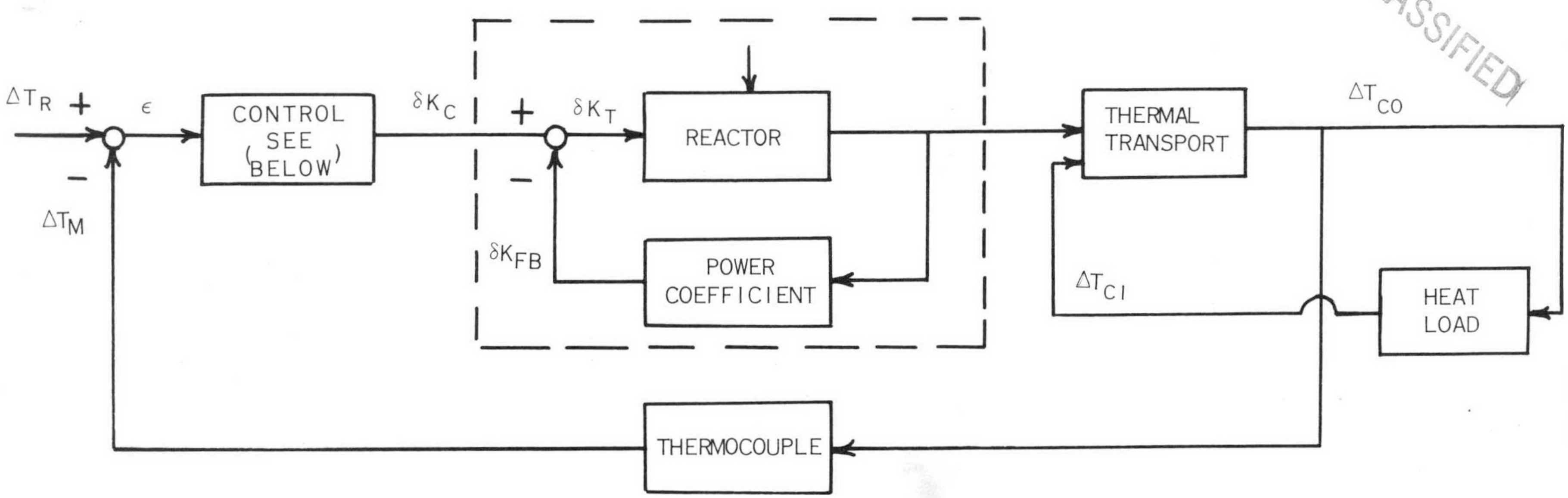
FIG 44

V-15

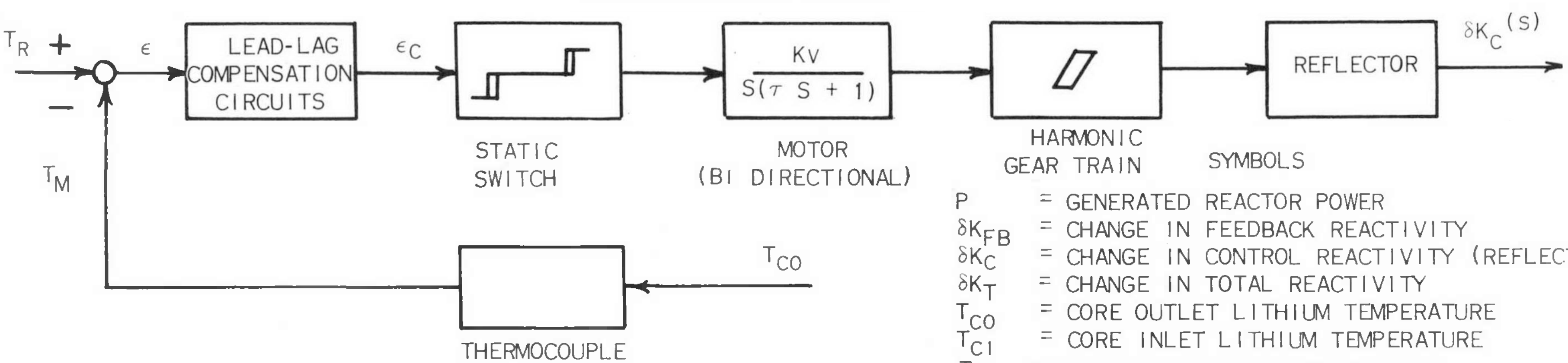
60072

BLOCK DIAGRAM OF REACTOR, PRIMARY COOLANT LOOP, AND CONTROL

REACTOR LOOP



REACTOR CONTROL SYSTEM



SYMBOLS

P = GENERATED REACTOR POWER
δK_{FB} = CHANGE IN FEEDBACK REACTIVITY
δK_C = CHANGE IN CONTROL REACTIVITY (REFLECTOR)
δK_T = CHANGE IN TOTAL REACTIVITY
T_{CO} = CORE OUTLET LITHIUM TEMPERATURE
T_{CI} = CORE INLET LITHIUM TEMPERATURE
T_M = MEASURED LITHIUM TEMPERATURE
T_R = REQUEST VALUE OF TEMPERATURE
ϵ = TEMPERATURE ERROR
ϵ_C = COMPENSATED TEMPERATURE ERROR

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FIG 45

core exit. The measured temperature is compared with a requested value to generate an error signal. This temperature error signal is acted upon by the lead-lag compensation circuits. When of sufficient magnitude, the compensated temperature error signal actuates a static switch which, in turn, actuates the control drive motor. The motor drives the reflector through a harmonic gear train. The displacement of the reflector results in an alteration of reactor power to reduce the temperature error. Control action ceases when the compensated temperature error is corrected to within the deadband of the static switch. The criteria for control gains and characteristics will be the optimization of performance and stability margin.

(3) Confirmation of Nuclear Design by Critical Experiments

Critical experiments have been conducted using the materials of PWAR-20 and covering a range of possible core and reflector dimensions (Fig. 46). These tests have provided the following results:

(a) Improved cross section data for the spectrum and materials of PWAR-20.

(b) Verified the accuracy of CANEL's two dimensional S-4 multigroup transport computing code employing the 16-group cross sections of Hansen and Roach (Ref. 17).

(c) Direct experimental verification of the worth of the pivoting reflector control method for various reflector thicknesses.

b) Development Potential of PWAR-20 Reactor with Improved Performance of Fuel System (Ref. 18).

The development potential of PWAR-20 is largely dependent on the performance of the fuel. Parametric studies of the effects of various fuel systems and assumed fuel performance levels on the weight of 2-Mw reactors are summarized in Fig. 47. The core sizes have been calculated using the TDC two-dimensional transport theory code.

The U^{235}-based fuels which show the most promise and which are the subject of present CANEL development are:

1) UC/UN
2) UC/UN-1 W solid solution
3) 90 UC/UN-ZrC solid solution
4) UC/UN with a central hole

These UC/UN fuels result in lower weight reactors at 2-Mw than UO_2, UO_2-BeO, and the UC/UN and UO_2 cermets because their higher uranium densities permit smaller critical reactor diameters.

The present design of the 2-Mw reactor is predicated on being able to attain 1.0 a/o U burnup from the fuel. If the fuel performance could be extended to 1.5 a/o U burnup (criticality limited for this core), a weight savings of approximately 400 pounds would result. It is felt that the burnup limit of UC/UN type fuel can be extended to 1.5 percent or higher by the use of additives such as tungsten and ZrC to improve the creep strength of the fuel. The use of fuel pellets with a central hole also appears promising as a means of controlling swelling. If the development of plutonium of U^{233} fuels were pursued, it is clear that the reactor weight could be reduced appreciably provided swelling and gas release could be controlled up to approximately 4.0 a/o U burnup.

CANEL CRITICAL EXPERIMENTS

ASSEMBLY	PROGRAM	REACTOR	DIAMETER, IN.	LENGTH, IN	CORE TYPE	NO. OF EXPERIMENTS
CCA-1	ANP	PWAR-5	30	30	UO_2-Be;NA; SS	181
CCA-2	ANP	PWAR-11	18	18	UO_2-BeO;Li; Cb	785
CCA-3	ANP	PWAR-12	18	18	UO_2-Cb; Li; Cb	539
CCA-5	SNAP	PARAMETRIC STUDY	10-14	12-15	UO_2-BeO; Li; Cb	228
CCA-6	SNAP	LCRE	13.5	12-15	UO_2-BeO; Li; Cb	714
CCA-7	SNAP	PARAMETRIC	10-12	10.6-14.2	UC; Li; Cb	649
CCA-8	MCR	MCR	13.7	12.4	UO_2; K; Cb	279
						3375

FIG 46

FUEL SYSTEM EFFECT ON REACTOR DESIGN

2 Mw - 10,000 HOUR REACTORS

FUEL	60 UO_2- 40 W	50 UO_2- BeO	60 UC/ UN 40 W	LOW DENSITY UC/UN	UO_2	REF. 4 DESIGN UC/UN	UC/UN W/ CENTRAL HOLE	LOW DENSITY UC/UN	UC/UN UC/UN-1 W UC/UN-10 ZrC	$U^{233}C$/ $U^{233}N$ 40 W	PuC/ PuN- 1 W
PERFORMANCE ASSUMPTIONS											
FUEL DENSITY, % THEORETICAL	95	95	95	8	95	95	95	8	95	95	95
MAX. FUEL TEMP., W/HCF, F	2300	2300	2300	2500	2300	2300	2300	2700	2300	2300	2300
FISSION GAS RELEASE, %	10	25	10	100	10	20	10	100	10	10	10
FUEL CLAD ALLOY	Cb	Cb	Cb	Cb	Cb	Cb	Cb	W Re	Cb	Cb	Cb
FUEL CLAD STRENGTH, psi	1500	1500	1500	1500	1500	1500	1500	8000	1500	1500	1500
RESULTS											
MAXIMUM BURNUP, a/o U	0.8	1.4	1.3	1.3	1.1	1.0 (1)	1.4	1.3	1.5	3.8	3.9
MAXIMUM POWER DENSITY, Kw/CC	0.1	0.15	0.22	0.30	0.24	0.28	0.35	0.31	0.42-0.38	0.65	1.07
ESTIMATED ENGINEERED WEIGHT, LBS.	3100	2200	2050	2050	1900	1800	1600	1500	1400-1500	[illegible]00	1000

(1) BURNUP LIMITED REACTOR

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FIG 47

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2) Design Study of an 8 to 10-Mw Power Level Reactor

A preliminary design of a reactor of the 8 to 10-Mwt power level is in progress. The burnup capability of the fuel appears to be the major consideration in this size range. The effect of permissible burnup on reactor weight is shown in Fig. 48. If UC/UN is considered for the 8-Mw application, it is evident that an improvement in performance of from 1.0 to 1.5 a/o U burnup would result in a reduction in weight of approximately 2000 pounds. If the performance level could be raised to accommodate a burnup of two percent, a weight saving of approximately 2700 pounds would result.

Certain selected fuel systems which offer possible alternate solutions to the problem of minimizing reactor weight for the 8-Mw power levels are also shown in Fig. 48 and summarized in more detail in Fig. 49. The 60 V % UO_2-40 V % W cermet at a burnup of 3 a/o U would result in a weight about comparable to that for a UC/UN fuel system at a burnup of 1.85 percent. Additional weight savings are possible for a 60 V % UN-40 V % W fuel if a burnup of 4.4 a/o U can be attained. Essentially equivalent to this is a fuel system of low density UC/UN designed for 100 percent gas release and clad with a high strength (W-Re) alloy.

Fig. 50 shows estimates of neutron shield weights versus reactor power for two dose limits for the payload, 10^{11} nvt and 10^{13} nvt. These estimates are based on a simple plug shield concept. It is expected that shield shaping will result in a moderate reduction in weight.

3) High Power Level Reactor and Shield Systems

Studies of high power level reactors (10 to 40-Mwt) are in progress to determine what basic changes would have to be made in the SNAP-50/SPUR reactor concept to reach high power levels. Preliminary results indicate that temperature limitations in the reflector structure will restrict the use of radiantly-cooled reflectors to a definite maximum power level. This power level may be extended somewhat by finding the optimum core length-diameter ratio (assumed to be 1.1 in present designs), by using reflector materials having higher conductivities than BeO, such as graphites or beryllides, and by replacing the columbium alloy reflector support by a molybdenum, tantalum or tungsten alloy structure. Above the maximum power level for radiantly-cooled reflectors, other reactor control concepts such as liquid metal-cooled reflector-poison elements or movable fuel must be used.

Control by external reflector movement also is not usable above certain combinations of power and core diameter because of insufficient reactivity change. Typical control requirements are summarized below:

Temperature Effects	3%ΔK
Fuel Swelling	1
Allowance for Failed Control Element	2
Shim for Fuel Burnup	1/2 x Percentage Burnup (approximately)
	6% + 1/2 x Burnup %

It is apparent that the reactivity requirement increases with burnup at a given core size, which means that it increases with power level at a given core size. At the same time, the reactivity controlled by a reflector of a given thickness decreases as core size increases. Fig. 51 indicates the limiting core diameters and powers for external reflector control assuming a 4-inch thick BeO reflector and a core L/D of 1.1. Reflector control can be extended to higher power levels by using thicker reflectors and larger core L/D's.

EFFECT OF FUEL PERFORMANCE ON REACTOR WEIGHT

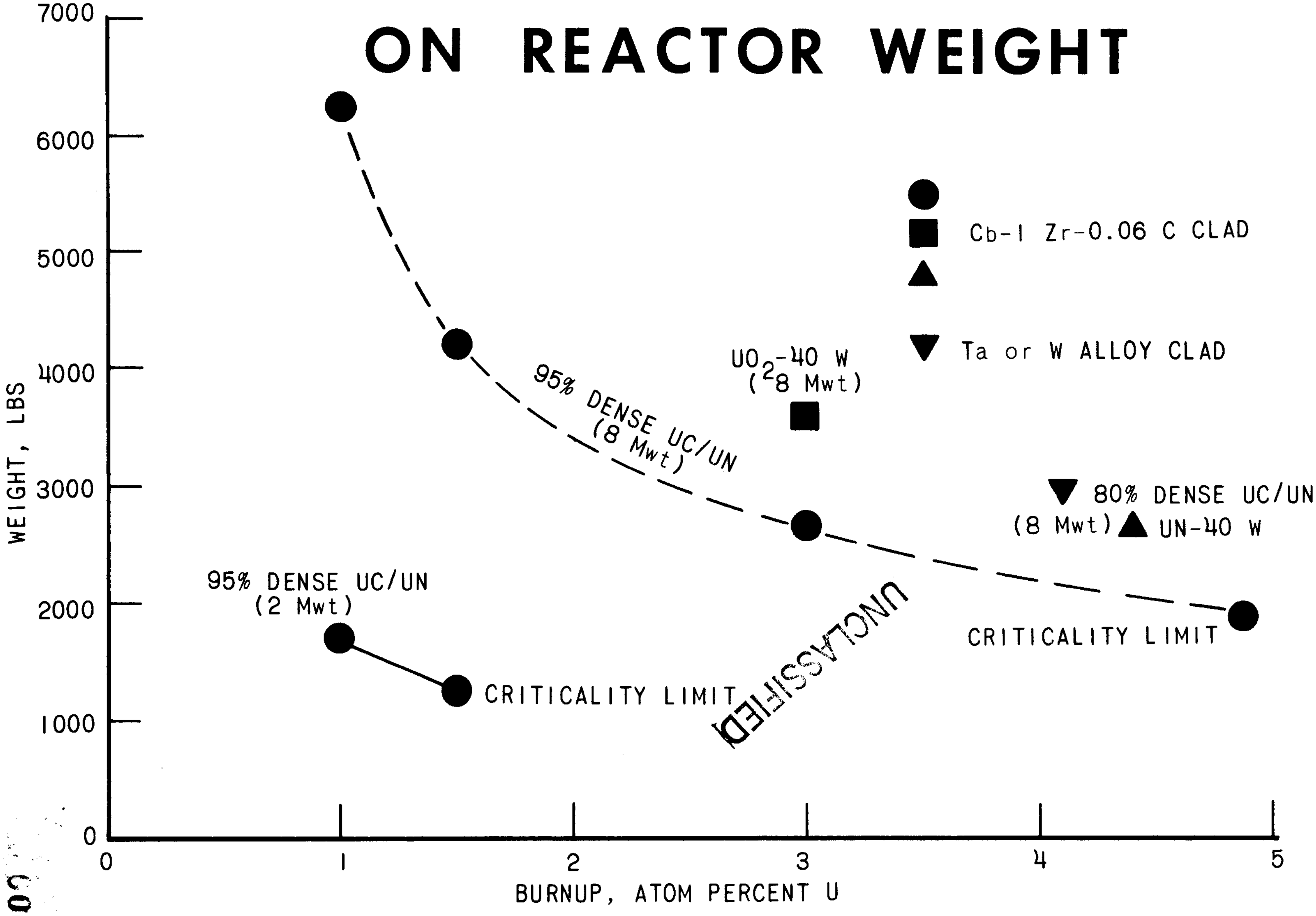

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FIG 48

FUEL SYSTEM EFFECT ON REACTOR DESIGN

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8 Mw - 10,000 HOUR REACTOR DESIGNS

FUEL	UC/UN	UC/UN	LOW DENSITY UC/UN	UO_2-40 W	50 UO_2-50 BeO	LOW DENSITY UC/UN	UN-40 W
PERFORMANCE ASSUMPTIONS							
FUEL DENSITY, % THEORETICAL	95	95	80	95	95	80	95
MAXIMUM TEMP, W/HCF, F	2300	2300	2700	2700	2700	2700	2300
FISSION GAS RELEASE, %	20	10	100	10	25	100	10
FUEL CLAD ALLOY	Cb	Cb	Cb	Cb	W,Ta	W	Cb
FUEL CLAD STRENGTH, psi	1500	1500	1500	1500	4000	4000	1500
RESULTS							
MAXIMUM BURNUP, a/o U	1.0 (1)	1.5 (1)	3.9	3.0	3.0 (1)	4.1	4.4
MAXIMUM POWER DENSITY, Kw/cc	0.28	0.42	0.93	0.37	0.32	0.98	0.73
ESTIMATED ENGINEERED WEIGHT, LBS	6250	4200	4000	3600	3550	2950	2650

(1) BURNUP LIMITED REACTOR

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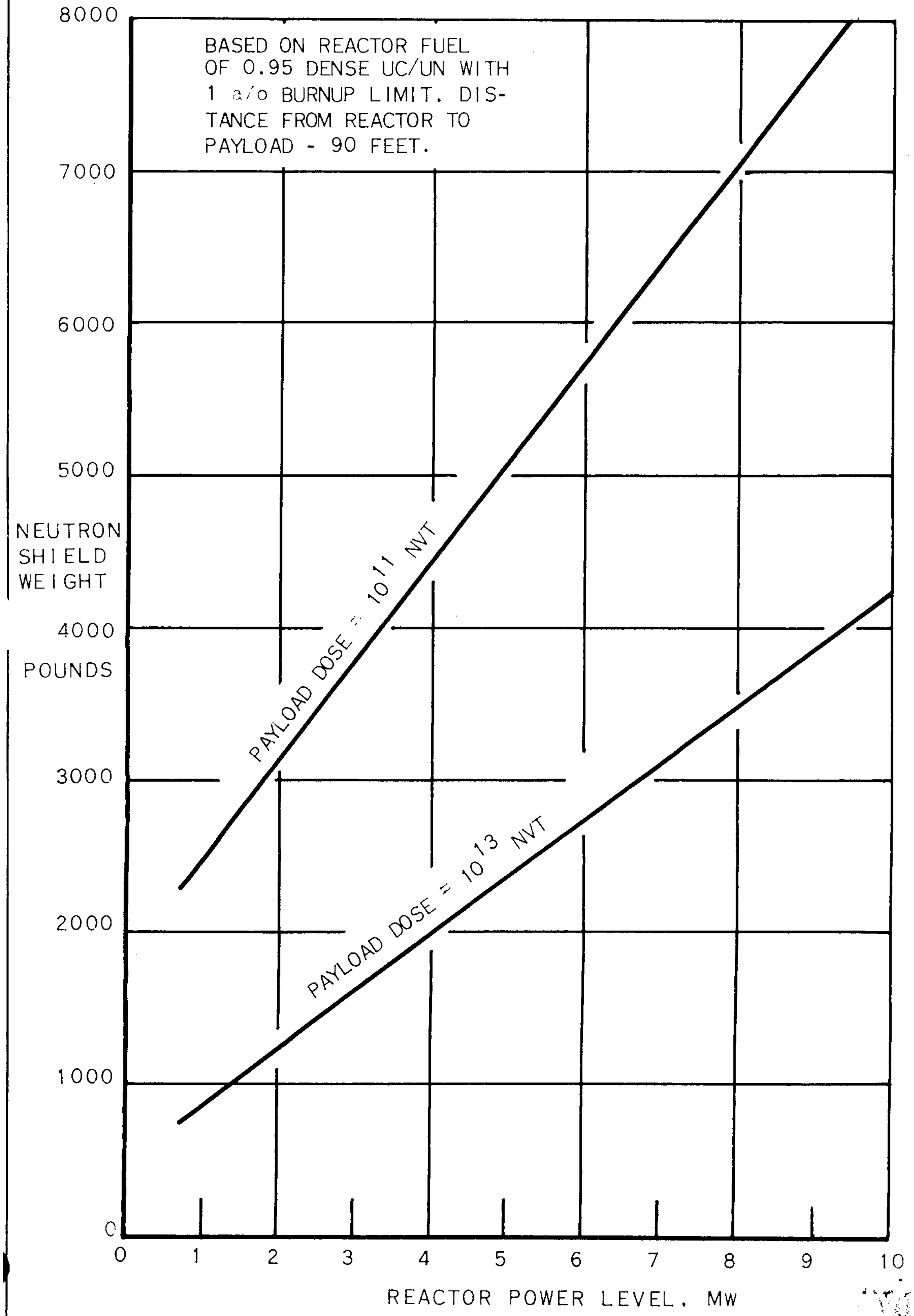
FIG 49

[redacted]

FIG 50

LiH NEUTRON SHIELD WEIGHT (INCLUDING STRUCTURE) VERSUS REACTOR POWER



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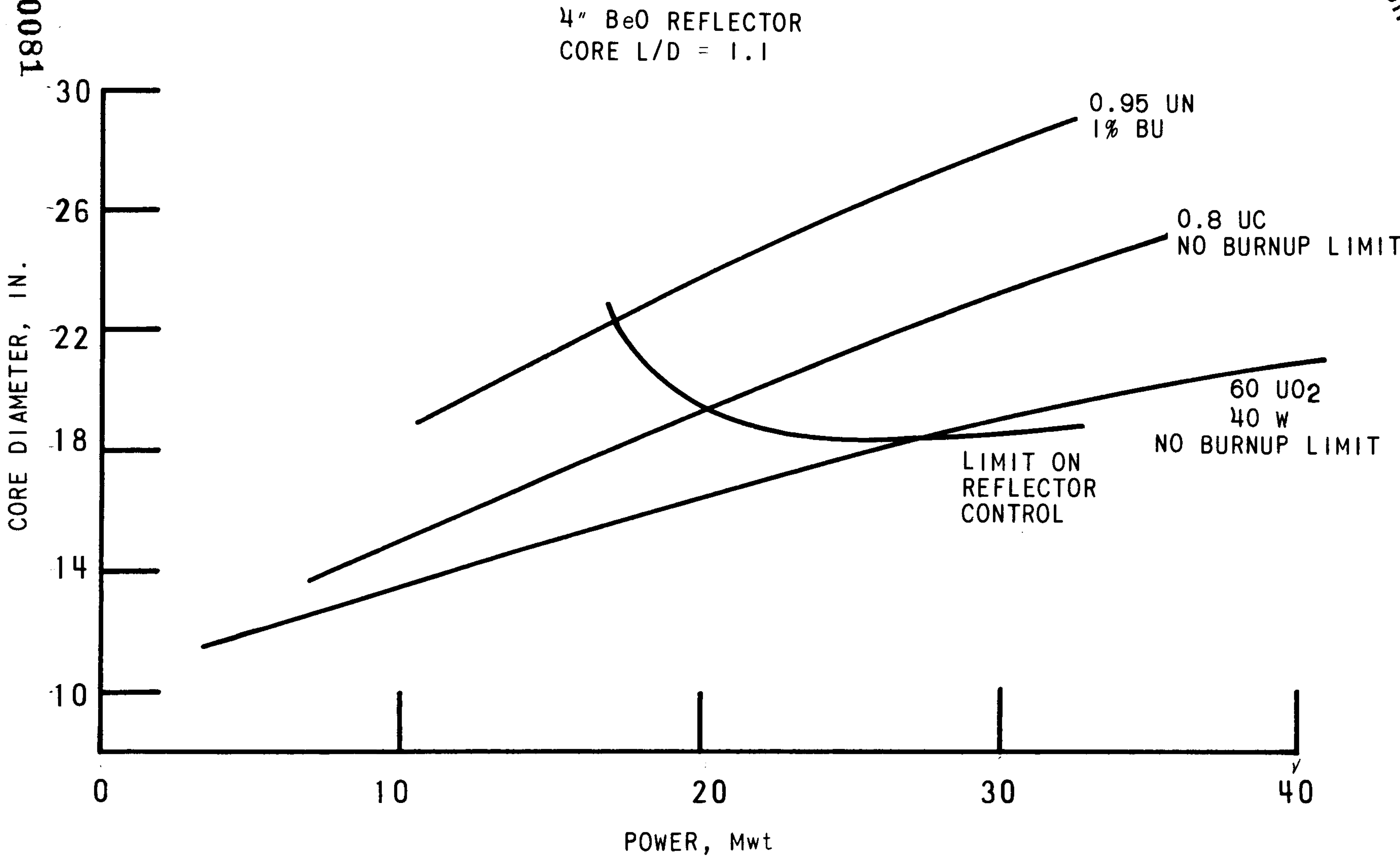
LIMITATIONS IN REACTOR CONCEPT



V-2.4

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FIG 51

b. Immediate Program Plans (FY 1965)

1) Physics, Engineering and Control Analysis of the PWAR-20 Reactor

The design of the PWAR-20 will be weight optimized on the basis of parametric studies currently in progress. It is expected that the core length to diameter ratio (currently 1.1) will be raised.

A detailed program designed to define and explore the stability and controlability of the PWAR-20 reactor will be completed.

Two design studies of an 8 to 10-Mw(th) reactor will be completed, each based on a different fuels criterion:

1) an extension of UC/UN fuels capability to permit attainment of 1.5 a/o burnup.
2) a non-burnup limited UN-W cermet fuel.

In conjunction with these design studies, engineering design studies of auxiliary control mechanisms such as poison rods, etc., will be completed.

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2) Critical Experiment Program

The current program of SNAP-50 critical experiments will be completed. This program consists of a parametric investigation of UC-fueled reactors with core diameters of 10 to 12 inches and length-to-diameter ratios of 0.9 to 1.4. Fig. 52 shows the critical assembly split table with a view of a 10-inch diameter core. The effectiveness of reactor control by displacement of the side reflector is being evaluated and the effects on power distributions are measured. Reactivity coefficients are measured to provide data for analytical evaluation of cross section data used in reactor calculations. Special experiments are also performed to provide reactivity data for evaluation of nuclear safety for assembly and handling of ground test and flight reactors.

3) Reactor Design Studies to Cover the Potential Range of Application

a) 2 to 20-Mw Reactor Parametric Studies

The first phase of the high power reactor studies will consider 2 to 20-Mw reactors with radiatively-cooled reflectors. The plan for conducting this study is summarized in Fig. 53.

The fuels chosen for study were those shown to be the most promising from the results of the reactor fuels parametric studies (Ref. 18 and 19). In addition, the effect of changing some of the design parameters such as reflector material, core pressure drop, coolant outlet temperature, and clad strength will be studied.

The results of this evaluation will be a detailed understanding of reactor characteristics in the 2 to 20-Mw range.

b) 10 to 40-Mw Reactor Studies

It is expected that the high power reactor studies will be extended to cover the 10 to 40-Mw power range. The first phase of these studies, which is in progress, consists of a materials evaluation to establish the compatibility, temperature, radiation, and structural limitations of the candidate reflector and poison materials.

SNAP-50 CRITICAL ASSEMBLY

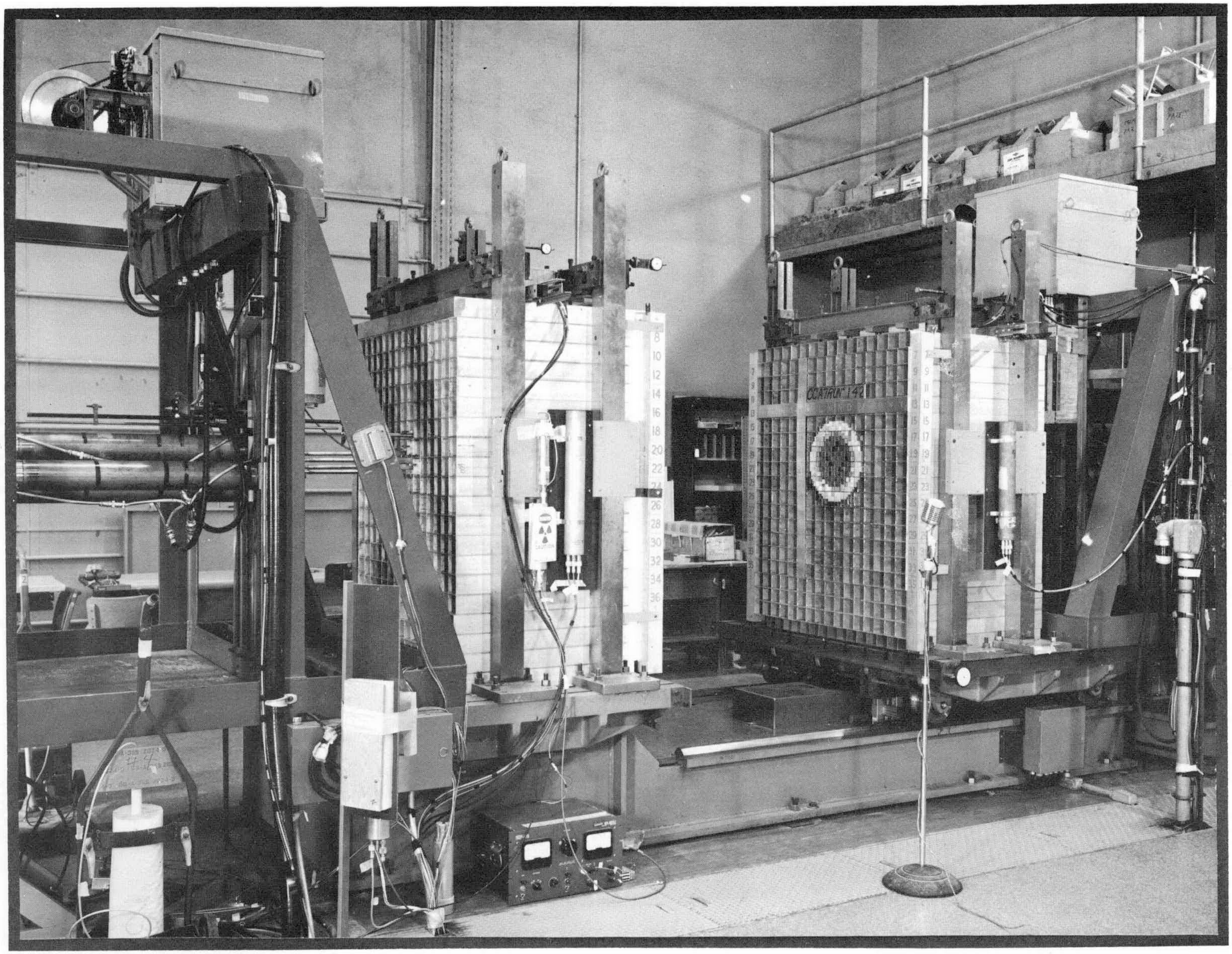

FIG 52

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[redacted]

FIG 53

2-40 Mw REACTOR PARAMETRIC STUDIES

PHASE I - 2-20 Mw RADIATIVELY-COOLED REFLECTOR CASES

Initial Evaluation

Fuel	UC/UN, 0.95 Dense	UC/UN, 0.80 Dense	60 UO_2-40 W	60 UC/UN-40 W
Power, Mw	10	10	10	10
Poison Safety Rod	Yes	Yes	Yes	Yes
Maximum Fuel Burnup, a/o U	1	Unlimited	Unlimited	Unlimited
Maximum Fuel Temperature, W/HCF, F	2300	2600	2600	2600
Fission Gas Release, %	20	100	0	0
Clad Strength, 10,000 Hrs.	1500	4000	750	750
Maximum Reflector Clad temperature, F	2400	2400	2400	2400
Control Requirements, % ΔK	7+BU	7+BU	7+BU	7+BU
Reflector Material	BeO	BeO	BeO	BeO
Coolant Pressure Drop, psi	30	30	30	30
Coolant Outlet Temperature, F	2000	2000	2000	2000
Coolant Inlet Temperature, F	1900	1900	1900	1900

Parametric Cases	Parameter	Range or Point Values
Fuel Burnup Limit, a/o:	UC, 0.95 Dense	2, 3%
	60 UO_2-40 W and 60 UC-40 W	3%
Reactor Power		2, 20 Mw
Fission Gas Release	60 UO_2-40 W and 60 UC-40 W	5%
Fuel Clad Strength, psi		750 to 10,000
Maximum Reflector Clad Temperature, F		2600, 2800
Control Requirements, ΔK		5% + Burnup
Reflector Materials *		Zr Be_{13}, Pyrolytic Graphite
Coolant Pressure Drop, psi		60, 90
Coolant Outlet Temperature, F		1900
Coolant Inlet Temperatures, F		1850, 1800, 1750
Power Flattened Core Fuel		UC/UN 0.95 Dense, 1% Burnup
Clad Alloy		Cb or Mo vs. W Alloy
Cermet Fuel		Cb or Mo Cermet vs. W Cermet

00084

* Each of the reflector cases will be weight optimized considering core L/D and side reflector thickness as variables.

[redacted]

4) Shield Design Studies

Parametric shield studies for 2 to 20-Mw reactors are in progress to study the effects of reactor and powerplant variables on shield weight and to clarify the power and application range in which the radiatively-cooled shield concept is feasible.

During this fiscal year, the range will be extended to cover shields for reactors capable or producing up to 40 Mwt. Concurrent with these studies, an effort will be made to develop improved methods of parametric analysis for plug and shaped shields and to develop the Monte Carlo FMC-N code to analyze in more detail the specific shield designs of interest.

c. Future Program Plans (FY 1966)

1) Design Optimization of Reactor and Shield

It is assumed that a reactor power level for initial development will have been chosen. During FY 1966, design optimization of the reactor and shield for this power level will take place using the 6000-hour inpile test data on fuel, BeO and lithium hydride irradiation data, the 3000-hour creep test data on 10-inch pressure vessels, and the environmental test data on control drive components.

2) Ground Test Reactor and Shield Design Studies

Design studies of the flight reactor and shield will be conducted in order to modify the structures for operation in an earth-gravity field and to incorporate safety features (scram devices, etc.) which are unique requirements with the ground test. These design studies will culminate with the detailed design of the ground test modifications to the reactor and shield.

2. Reactor Development

a. Fuel Element Development

1) Fuel Development Program Plan

Candidate fuel materials will be selected on the basis of their merit as indicated by parametric and design studies of the reactor and on available information on physical properties, compatibility, and burnup capability. Physical properties such as thermal conductivity and thermal expansion coefficient will then be determined precisely for the candidate materials so that this information may be used in reactor design. Concurrently, out-of-pile tests will be performed to determine the chemical compatibility of the matrix and cladding materials with and without protective coatings. Clad strength will also be measured under simulated service conditions except for radiation. Concurrent inpile tests employing capsules will determine the stability of the fuels at temperature under irradiation and their ability to retain fission products.

A final selection of one fuel would be made on the basis of the properties, compatibility, and burnup capability determined by the above program, as they affect the over-all reactor design. Work will then begin on fabrication and process control methods for the actual reactor fuel elements.

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[redacted]

2) Present Status of Fuel Development

a) Physical Properties of Fuel Matrix

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A knowledge of accurate fuel matrix properties is necessary to perform a reactor design. The acquisition of required accurate properties data for UC is nearly completed at CANEL. Data acquisition is in progress for UN. The status of UC and UN property data at CANEL is summarized below:

	UC	UN
Coefficient of Expansion	100F to 2800F	In Progress
Heat Capacity	600F to 2200F	600F to 2400F
Thermal Conductivity	1650F to 2900F	In Progress
Vapor Pressure and Thermodynamics	3600F to 4000F	3050F to 3600F
Hot Hardness	1800F to 2600F	1800F to 2600F

Some data of UO_2 and UN-refractory metal cermets are available from work at Battelle Memorial Institute. In addition, it is possible to make estimates of the properties from those of the constituents, since these materials are dispersions.

b) Compatibility of Fuel Matrix with Fuel Cladding and Coolant

A compatibility problem exists between Cb-1 Zr alloy and uranium carbide and uranium nitride, because the alloy reduces these fuel matrix materials to form metallic uranium. Three methods have been considered to prevent these interactions:

(1) Hyperstoichiometric UC may be used with Cb-1 Zr alloy cladding. In this case, a columbium carbide layer is formed on the cladding ID. Since diffusion of carbon through this carbide layer is slow compared with the diffusion of carbon through columbium, the loss of carbon from the fuel is decreased and uranium formation is prevented.

(2) A tantalum barrier may be used with hyperstoichiometric UC. In this case, a tantalum carbide layer is formed instead of the columbium carbide layer described above, but the rate is much slower and, therefore, much slower carbon losses are experienced. In addition, the tantalum carbide formed is more stable thermodynamically than columbium carbide, and, therefore the transfer of carbon to the cladding is minimized.

(3) A tungsten barrier may be used with stoichiometric UC and UN. In this case, the interposition of tungsten between the fuel and cladding results in an interface which is very stable thermodynamically and little or no reaction occurs.

Testing experience with these three methods of solving the matrix-cladding compatibility problem is summarized below. Specimens were placed in lithium and heated for long periods of time at 2200F. Subsequently, they were examined metallographically. The method described in (1) above for UC was found to be undependable, due to selective carburization of the zirconium in the Cb-1 Zr alloy and concomitant lack of the protective columbium carbide layer.

00086

Fuel Cladding Compatibility at 2200F[1] (Sound Pins)

Fuel	Number of Tests	Barrier	Max. Test Time, Hr.	Effect on Fuel	Effect on Barrier	Effect on Cb-1 Zr Clad
$UC_{1.1}$	10	None	11,000	C Loss	--	Carburized
$UC_{1.1}$	10	Ta	11,000	C Loss	Carburized	Select Carburized
$UC_{1.0}$	11	W	9000	None	None	None
UN	2	None	3000	None	--	None[2]
UN	6	W	6000	None	None	None

[1] Li-Encapsulated

[2] Free uranium formed in diffusion couple tests

The possibility of contact between the lithium coolant and the fuel matrix in defected pins introduces an additional compatibility problem due to possible reactions between the lithium and the fuel matrix. To investigate this problem, tests have been made using purposely defected pins to simulate a cladding failure. The results of these tests are indicated below.

Fuel-Cladding Compatibility at 2200F
(Defected Pins Immersed in Lithium)

Fuel	Barrier	Effect on Fuel	Effect on Barrier	Effect on Clad	Time for Uranium To Penetrate Clad, Hrs.
$UC_{1.1}$	None	R	--	D	1000
$UC_{1.1}$	Ta	R	D	D	3000
$UC_{1.0}$	W	R	D	D	5000
UN	None	None	--	D 4 mils in 3000 hours	----
UN	W	None	D	D	5000[1]

R - Partial Reduction to Elemental Uranium

D - Diffusion of Elemental Uranium

[1] Tested at 2400F

Summarizing the results of testing with sound and defected pins, the long time results indicate that tantalum and tungsten are satisfactory barriers for containing UC and that tungsten is satisfactory for containing UN in undefected Cb-1 Zr alloy pins at 2200F. In addition, tungsten barriers suppress lithium reaction of UC and UN and are particularly effective in the case of UN. The performance which can be forecast with certainty on the basis of these tests is shown below.

Status of 2200F Fuel-Cladding Compatibility

Fuel	Problem	Solution	Satisfactory Performance To, (Hours)
UC	rx with Cb-1 Zr to liquid U	$UC_{1.2}$ with Ta barrier	11,000
		$UC_{1.0}$ with W barrier	11,000
UN	rx with Cb-1 Zr to liquid U	$UN_{1.0}$ with W barrier	6000

c) Strength and Fabricability of Fuel Cladding

(1) Structural Design Criteria

The creep strength of the fuel element cladding largely determines the ability of the fuel element to remain dimensionally stable under the effect of fuel matrix swelling and fission gas evolution. The present strength and ductility criteria for the fuel cladding are 1) adequate strength to contain a release of 20 percent of the fission gases released in the fuel, and 2) sufficient rupture ductility to permit a 2 percent diametral swelling of the fuel over 10,000 hours without premature rupture. Usable strengths of Cb-1 Zr-.06 C, the present nominal cladding material, which met these criteria, are shown in Fig. 54. The stress is based on one-half the rupture strength for 10,000 hours.

(2) Measurement of Creep and Rupture Strength

Since the strength is markedly affected by fabrication processes, it is essential to determine the strength of the cladding materials in the form of tubing. The strength of Cb-1 Zr tubing (low carbon, 100 ppm) has been measured in the past and found to be marginal for the SNAP-50 application.

Strength studies are in progress on fuel cladding of improved alloys. The alloys being tested include: Cb-1 Zr-0.06 C, Cb-10 W-1 Zr-0.1 C, Ta-8 W-2 Hf, and tungsten-lined Cb-1 Zr-0.06 C alloy. The test specimens with nominal dimensions of 0.304 OD x 0.015-inch wall by three inches long and 0.250 OD x 0.025-inch wall by three inches long are internally pressurized by helium to subject the specimens to a biaxial stress and are contained in a lithium bath at test temperature (Figs. 55 and 56).

Over 200 tests in the temperature range of 2000 to 2400F and for times to 4600 hours were completed or are in progress. The test data (Ref. 20, 21, 22, and 23) which are summarized below show the following:

(a) A 10,000-hour, 2200F rupture strength of 700 psi for Cb-1 Zr alloy.

(b) Increase in carbon level to 600 ppm for the Cb-1 Zr system, coupled with a solution heat treatment, doubles the strength.

(c) Addition of cold-working after solution heat treatment increases rupture strength, but decreases rupture ductility.

(d) An estimated 10,000-hour, 2200F rupture strength of 1500-2000 psi for solution heated Cb-1 Zr-0.6 C alloy.

(e) An estimated 10,000-hour, 2200F rupture strength of 1700-2400 psi for Cb-10 W-1 Zr-0.1 C alloy.

(f) An estimated 10,000-hour, 2200F rupture strength of 6400 psi for Ta-8 W-2 Hf alloy.

(g) The 10,000-hour, 2200F rupture strength fuel clad requirement of 1500 psi with 8 percent minimum rupture ductility is expected to be attained with the Cb-1 Zr-0.06 C alloy, the Cb-10 W-1 Zr-0.1 C alloy, and the Ta-8 W-2 Hf alloy, based on results of tests extending to 2000 hours.

STRUCTURAL DESIGN CRITERIA AND MATERIAL SPECIFICATIONS

68000

	Forgings 2000F	Forgings 2200F	Cb-1 Zr Structure Plates 2000F	Cb-1 Zr Structure Plates 2200F	Tube and Sheet 2200F	Tube and Sheet 2400F	Cb-1 Zr 0.06% C Fuel Element Cladding 2200F	PWA 1202 Ti-8 Al-1 Mo-1 V 1000F
10,000 Hour Stress to Rupture - psi	3000	1500	2300	1300	700	500	1500	
10,000 Hour Stress for 1% Creep - psi	1400	600	1300	600	600	250	750*	
Allowable Stress - 1/2 Stress-Rupture or 1% Creep Stress - Whichever is lower, psi	1400	600	1150	600	350	250	750**	
1/2 Stress Rate to Rupture Psi/Hr	-	-	-	-	-	-	0.11	
Minimum Rupture Ductility, %	-	-	-	-	-	-	8	
Minimim Initiation of 3rd Stage Creep %	-	-	-	-	-	-	2	
0.2% Yield Strength	-	-	-	-	-	-	-	60,000
Ti Alloy Allowable Stress - .87 Times Yield Strength, psi	-	-	-	-	-		-	52,000

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*2% Creep used for Fuel Element Cladding

**Note: All other references to cladding strength are stress to rupture rather than allowable stress

FIG 54

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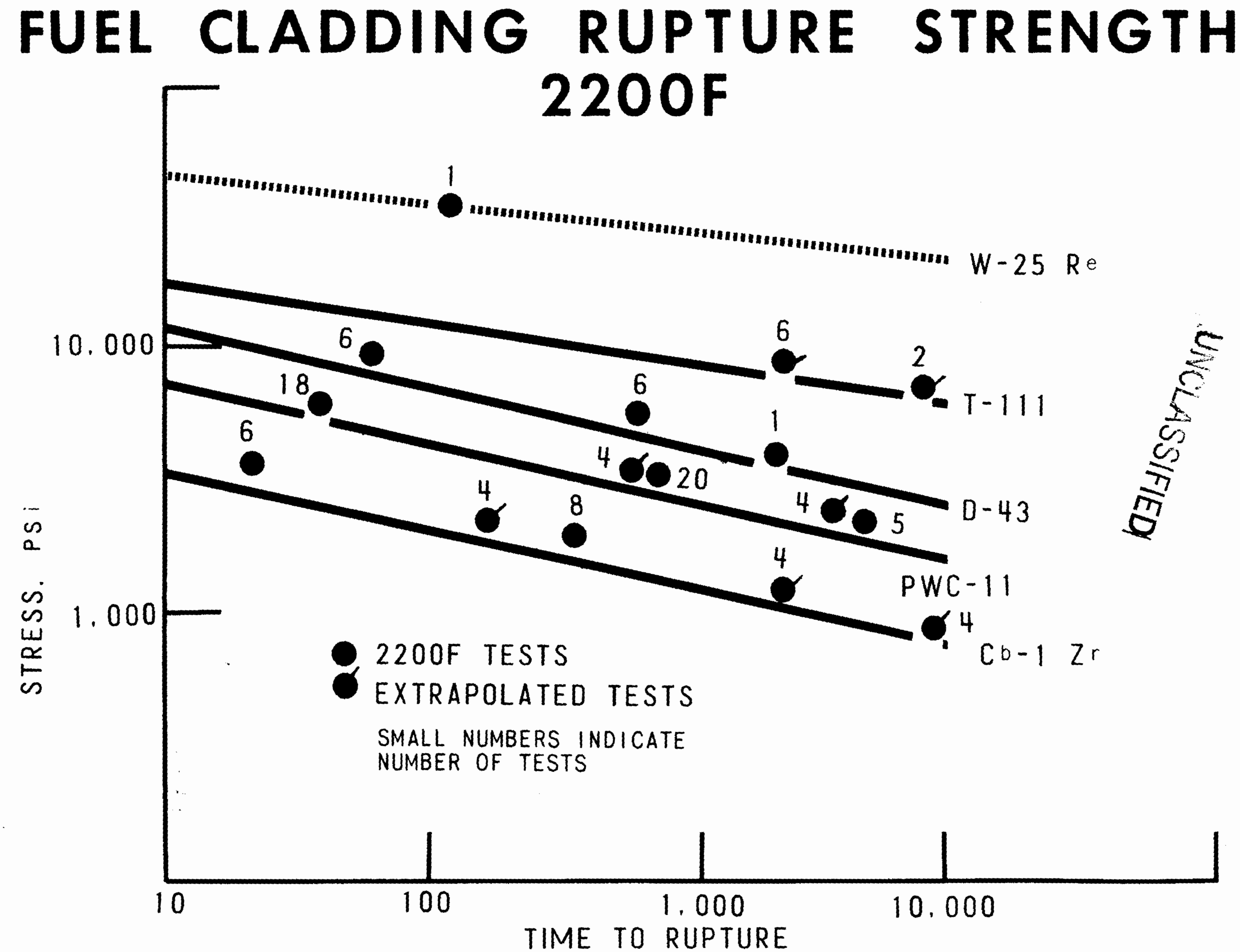

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FIG 55

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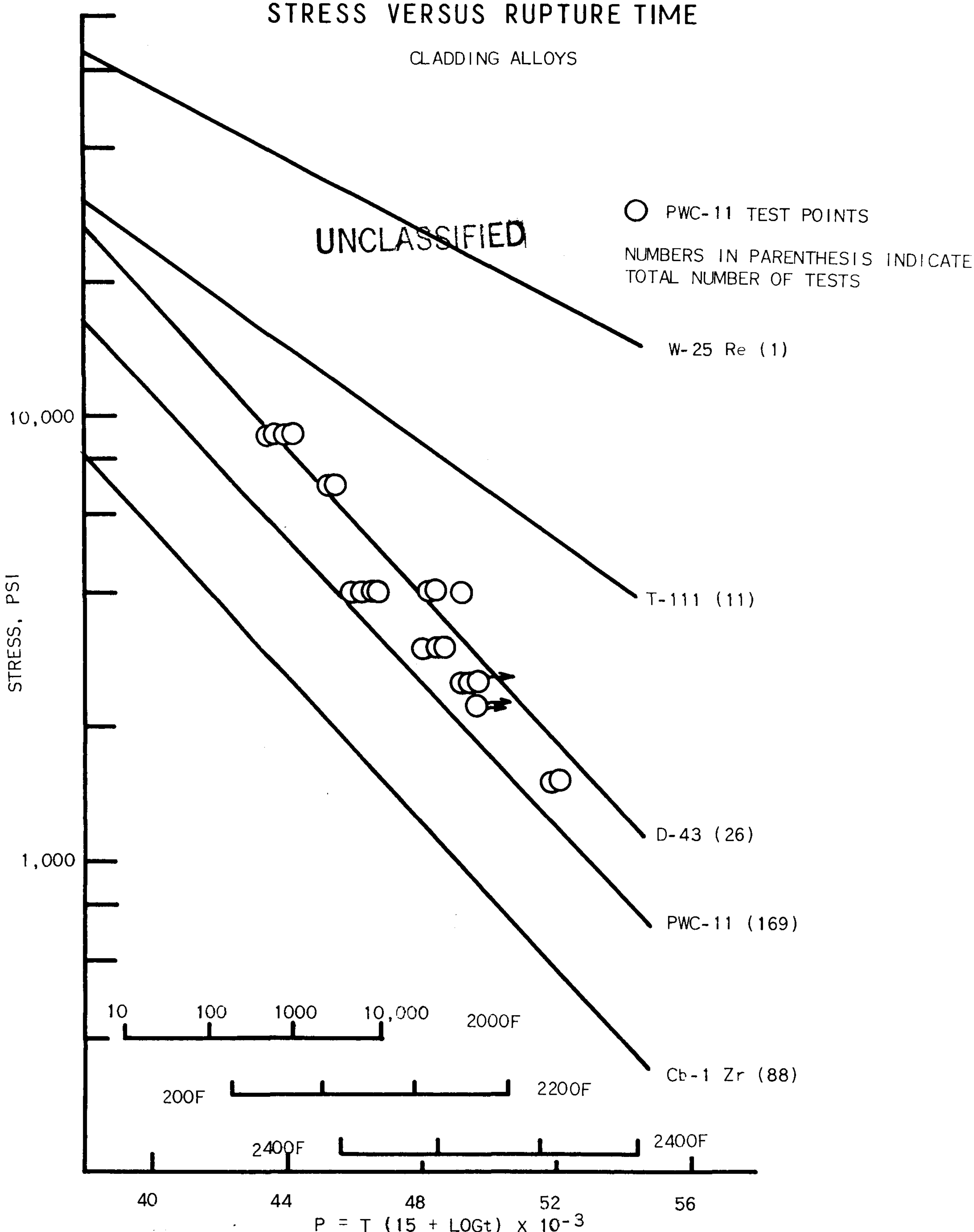
FIG 56

STRESS VERSUS RUPTURE TIME

CLADDING ALLOYS



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SNAP-50 Fuel Clad Strength Tests UNCLASSIFIED

Tests Completed or in Progress

Alloy	Temp. Range, F	Stress Range, psi	Max. Time, Hrs.	Total Tests	Est. 10^4 Hr 2200F Rupture Strength
Cb-1 Zr	2000 2400	866 4000	780	88	750
Cb-1 Zr-.06 C	2000 2400	10,000 10,000	4630	169	1500
D-43	2000 2400	2000 12,000	3300	26	2400
T-111 (T-222)	2400 2600	5000 9100	2300	11	6000
W-25 Re	2400 2600	15,000 25,000	-	-	20,000

Cb-1 Zr-.06 C alloy is presently the nominal fuel cladding material for the PWAR-20 2-Mwt reactor design.

(3) Fabrication Development

The fabrication of tubing from these refractory alloys is difficult because they must be protected from the atmosphere and other sources of contamination throughout the fabrication process. Some of the alloys are also difficult to cold work.

Through close metallurgical engineering liaison with CANEL, commercial tube mills have demonstrated the ability to produce high quality tubing of Cb-1 Zr and carbon-modified alloys with wall thicknesses of 0.008-inch

.

Non-destructive test methods have been established to assure close control of tubing quality.

An additional fabrication problem is introduced by the requirement of a tungsten barrier layer between the fuel matrix and the cladding to solve the compatability problem. Tungsten vapor deposition methods are being developed for applying a barrier coating on the inner diameter of fuel cladding tubes. Although a method which assures reproducible quality as regards to thickness, adhesion and grain structure has yet to be defined, the general process has been demonstrated to be capable of producing full length, tungsten-clad fuel element tubing of required quality. Studies are in progress at San Fernando Laboratories and at CANEL to evaluate operating parameters with the objective of improving the reliability of the process.

d) Performance of Fuel Under Irradiation

As was brought out in the discussion of the reactor design program, the reactor weight is strongly affected by the burnup capability of the fuel, especially in the case of 8-Mwt and larger reactors. Tests to determine the burnup capability of candidate SNAP-50 fuels have been in progress for several years employing encapsulated fuel element specimens which are irradiated in test reactors. The fission rates expected in service are equalled or exceeded in these tests.

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Irradiation tests of UC and UN were performed in support of the current reactor design at relatively high power density conditions, nominally 1.5 kw/cc. Duration of test and average cladding surface temperature are shown in Fig. 57. The shaded portion of the data mark represents that fraction of the maximum burnup achieved, i.e., 2 a/o, which the particular specimen attained.

The important interpretations of the irradiation results may be deduced from Fig. 58. These data indicate a linear relationship between density change, i.e., fuel swelling, and burnup for UN. An apparent threshold value exists for burnup of approximately 1.5 a/o beyond which excessive swelling occurs for UC at these high power density conditions. Data from UC irradiations by United Nuclear Corporation (Ref. 24) are also included in Fig. 58 and illustrate the improved fuel swelling expected from lowered centerline temperatures achieved by decreased power densities. Fuel swelling is caused principally by fission gas diffusion and its agglomeration at grain boundaries. Deformation of the fuel results when very high pressures are attained in these gas-filled pores. Internal gas pressures are relieved only if the porosity becomes connected to the fuel surface. The extent of gas release is related to the amount of surface-connected porosity.

The data show that swelling and gas release in UC and UN are well below design limits at a burnup of 1.0 a/o U and the high power density conditions tested.

3) Plans for the Balance of FY 1965 and FY 1966

a) Physical Properties of Fuel Matrix

The main effort in property studies for FY 1965 will be the determination of the thermal expansion and thermal conductivity of UN.

b) Compatibility of Fuel Matrix with Fuel Cladding and Coolant

The compatibility program for FY 1965 is designed to demonstrate the satisfactory containment of UC with tantalum and tungsten barriers, and UN with tungsten barriers in Cb-1 Zr-.06 C alloy cladding. A summary of the scheduled proof tests is listed below.

Li-Encapsulated Fuel - Cb-1 Zr-.06C
Compatibility Proof Tests

Fuel	Barrier	No. of Tests: Sound	Defected	Couple*	Maximum Test Time Planned, Hr.
$UC_{1.0}$	W	6	6	2	11,000
$UC_{1.2}$	Ta	6	6	2	11,000
UN	W	6	6	2	11,000

* Li-encapsulated wafer specimens simulating fuel/clad mass.

The compatibility of UC and UN with W-25 Re alloy and Ta-8 W-2 Hf alloy will also be determined. Test experience to date indicates that a barrier will not be required between fuel and W-Re alloys. The need for a barrier in the case of Ta-8 W-2 Hf alloy will be determined. A summary of the scheduled tests of these advanced fuel systems is shown below.

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RANGE OF UC/UN FUEL IRRADIATION TESTS

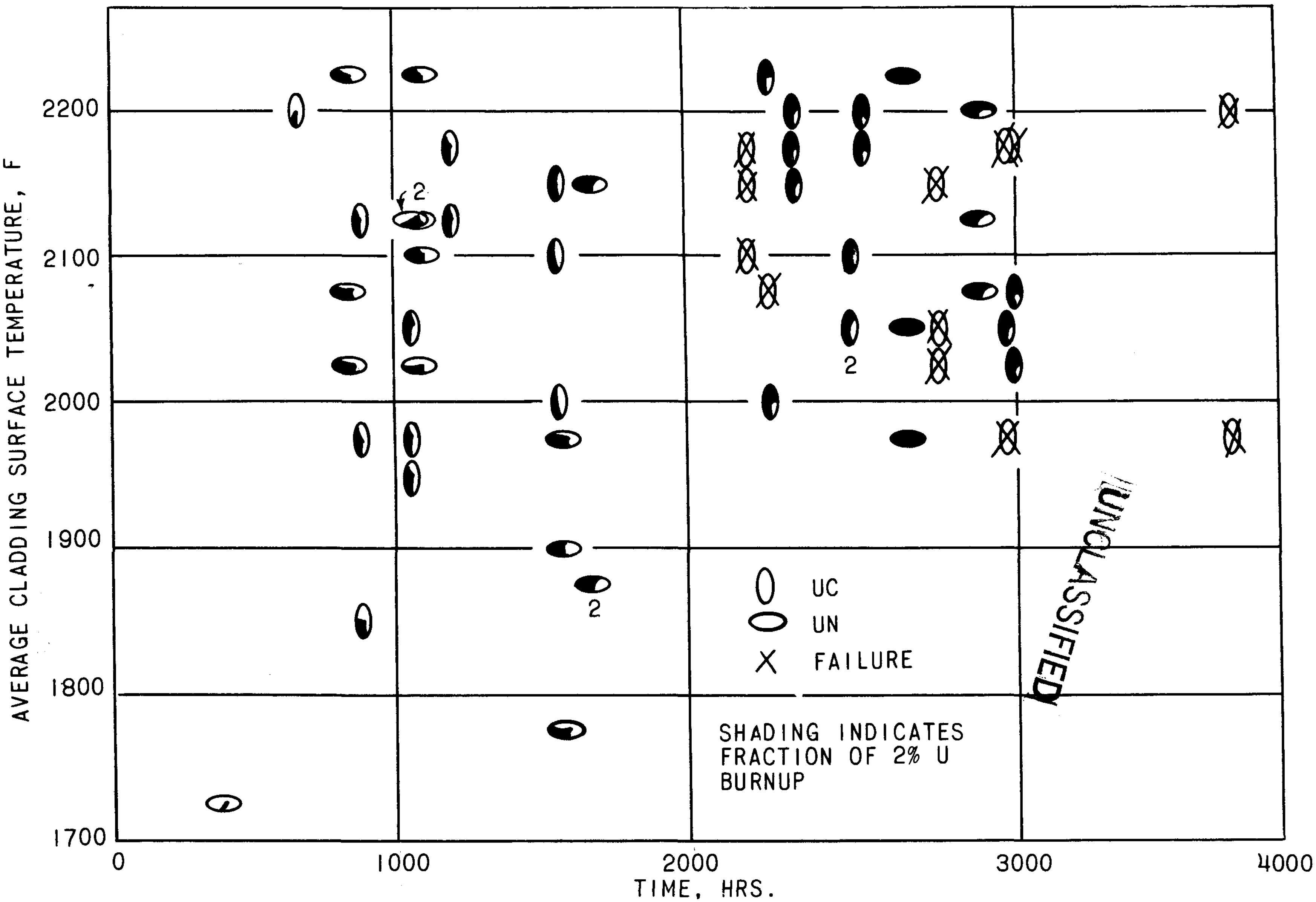

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FIG 57

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FUEL SWELLING VS BURNUP FOR UC & UN

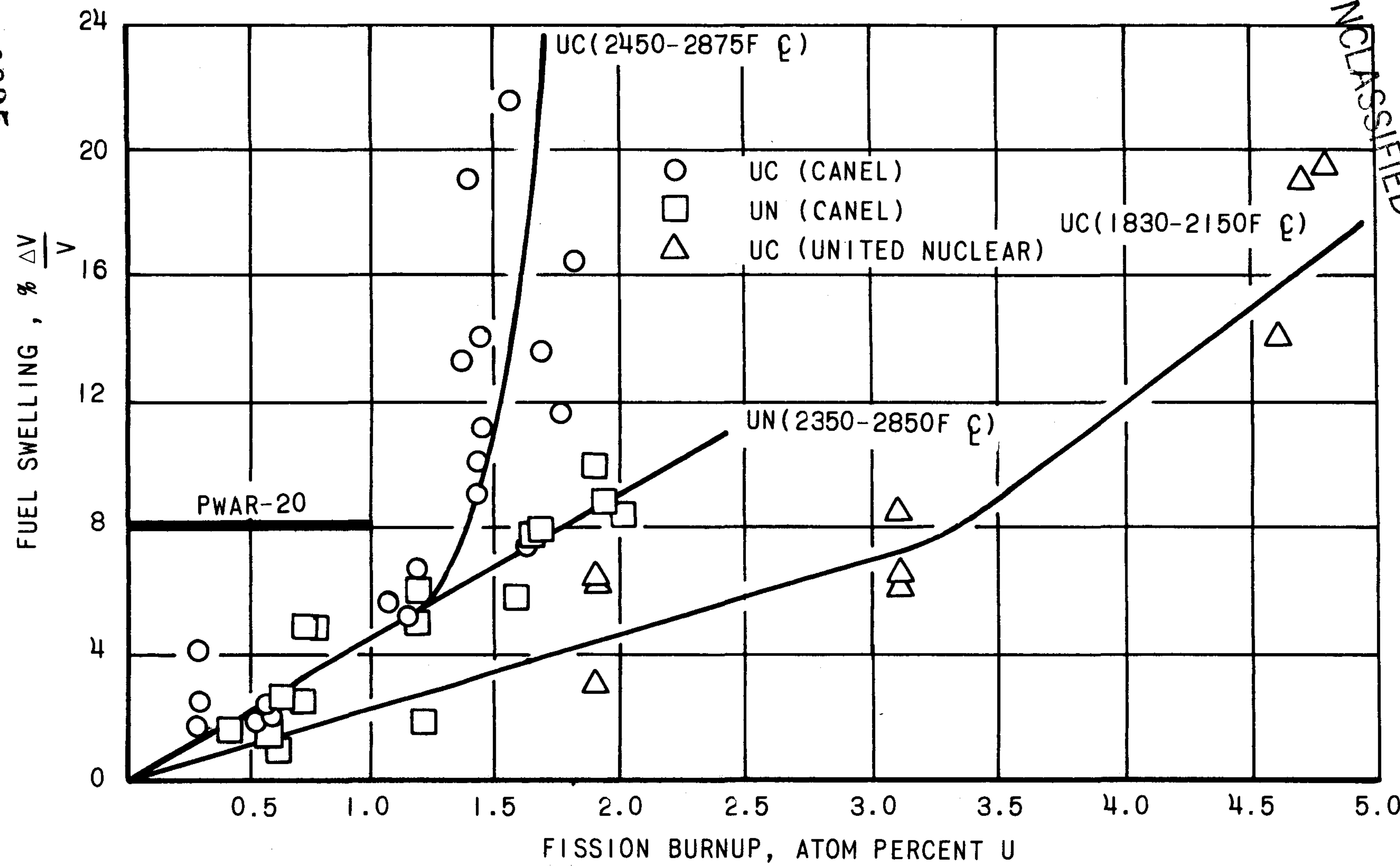

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FIG 58

Advanced Fuel Systems Compatibility Evaluation Program

Fuel	Clad	Barrier	No. of Tests Pins	No. of Tests Couples	Max. Test Time Planned
$UC_{1.0}$	W-26 Re	--	4	6	10,000
$UC_{1.2}$	W-26 Re	Ta	--	6	10,000
UN	W-26 Re	--	4	6	10,000
$UC_{1.0}$	T-111	--	6	6	10,000
$UC_{1.2}$	T-111	W	6	6	10,000
UN	T-111	W	6	6	10,000

c) Strength and Fabricability of Fuel Cladding

In FY 1965 and FY 1966, strength tests will be continued for times to 10,000 hours for Cb-1 Zr-0.06 C, tungsten-lined Cb-1 Zr-0.06 C, and Ta-8 W-2 Hf. In addition, tests will be initiated on Cb-1 Zr-0.06 C (tantalum-lined) Cb-1 Zr-0.06 C (tungsten-lined), Ta-8 W-2 Hf, Ta-0.5 W-2.5 Hf-0.01 C and W-25 Re.

Fabrication activities will be directed toward procurement of high quality tantalum and tungsten alloy tubing. Tungsten vapor deposition development studies will be continued as required to assure a reliable product.

d) Performance of Fuel Under Irradiation

The current UC and UN irradiation program in support of the 2-Mwt SNAP-50/SPUR reactor design (proof tests) is shown in the upper portion of Fig. 59. This program will provide information on the relative stability of UC and UN at reduced centerline temperatures, and at 2000F and 2200F cladding surface temperatures with claddings of different strengths, and will provide inpile confirmation for selected compatibility barriers.

The program aimed at extending the burnup of UC and UN without exceeding swelling and/or fission gas release design criteria is shown as "Fuel Geometry", "Density Variation", and "Alloy Strengthened" in Fig. 59. This program will test the effects of fuel density, fuel alloying, extended fuel length, and internal swelling accommodation.

In addition to the proof tests and the UC and UN burnup extension programs, a preliminary program aimed at high powered reactor designs is also shown in Fig. 59. The latter program will evaluate 1) W-UO_2 and W-UN cermets, and 2) low density UN clad in W-25 Re alloy which should release all fission gas from the fuel matrix and, thereby, limit swelling.

b. Reactor Structural Materials

1) Present Status

a) Structural Design Criteria

For the design of reactor structural members, the working stress is based on one-half the stress to rupture in 10,000 hours or the stress required to produce 1 percent creep in 10,000 hours, whichever is lower. Nominal temperatures at which these strengths are to be developed are determined by the estimated service temperature of the component in question.

b) Columbium Alloy Components

An alloy consisting of Cb-1 Zr-0.1 C is used for all portions of the reactor in contact with lithium other than the cladding. However, conservative strength

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FUELS IRRADIATION PROGRAM

TYPE	NUMBER OF SPECIMENS UC/UN	APPROX. MAX. BURNUP (a/o U)	TOTAL HOURS	DATE, MAXIMUM TEST TIME RESULTS
PROOF TESTS (UC, UN)				
Cb-1 Zr	24-24	1.0	6,000	MID '65
Cb-1 Zr-.06 C	36-36	1.5	10,000	LATE '66
MEAT LENGTH (UC, UN)				
L/D = 18	2-2	1.5	3,300	EARLY '65
	6-6	1.5	10,000	LATE '66
FUEL GEOMETRY (UC, UN) (CENTRAL HOLE)	2-2 6-6	1.5 1.5	3,300 10,000	EARLY '65 LATE '66
DENSITY VARIATION (UC)	4*	1.5	2,500	EARLY '65
ALLOY STRENGTHENED (UC-ZrC & UC-W)	8-12	1.5	2,500	EARLY '65
80 PERCENT DENSE UN WITH W-25 Re CLAD	12*	1.5	2,500	LATE '65
60 v/o UO_2-W CERMET	12*	5	2,500	LATE '65
60 v/o UN-W CERMET	12*	5	2,500	LATE '65

* 1 PIN PER CAPSULE

FIG 59

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criteria based on Cb-1 Zr alloy (low carbon, 100 ppm) are used for design purposes. This has been done to allow maximum latitude in the choice of carbon content, should a modification be necessary to insure carbon stability in the fuel cladding as discussed in the materials section of this report. Usable strengths based on Cb-1 Zr-.01 C are shown in Fig. 54 for:

a. Forgings and plate, pressure vessels, core support plate

b. Tube and sheet fuel element, hex can, reflector cladding

It is expected that when the stability of the Cb-1 Zr carbon system has been adequately defined, the design criteria will be modified to take maximum advantage of the strength of this alloy system.

c) Structural Material for High Temperature Portion of Reflector

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Tantalum base alloys such as T-111 and T-222 are being explored as substitutes for Cb alloys for the high temperature portions of the reflector support structure. Present design studies are based on the assumption that these alloys will provide strength at 2600F equivalent to the Cb-1 Zr-0.1 C alloy at 2400F. This will permit the use of a radiatively-cooled reflector for higher power reactors.

The columbium or tantalum alloy reflector support structure and other sections in the proximity of the reactor will require an oxidation protective coating to permit pre-launch heating of the SNAP-50/SPUR powerplant at 1000F in impure noble gas or air. The coating is to afford 1000-hour protection from oxidation and is not to adversely affect strength of structural members when subsequently exposed to design temperatures in space environment. Oxidation protection of Cb-1 Zr alloy for 1000 hours in 1200F air has been obtained on laboratory specimens with 95 percent reliability, using a tin-aluminum slurry coating PWK-35 (Ref. 71). Tests are in progress to evaluate oxidation protection of engineering-size specimens and to evaluate the effect of coating on creep strength of Cb-1 Zr alloy.

d) Structural Material for Low Temperature Portions of the Reactor

Strength-weight considerations for structural members associated with reactor system support during pre-launch and launch and operation at temperatures below 1000F have led to the choice of a titanium alloy (Ti-8 Al-1 Mo-1 V). Long time creep-rupture strength extrapolated from present limited data for temperatures to 800F show promise for this alloy for this application. A creep-rupture investigation at current design temperatures and allowable creep deformations (1% in 1000 hours, max.) is required to confirm present extrapolated data.

2) Plans for FY 1965 and FY 1966

Plans for future work on reactor structural materials and further details on the present knowledge of structural materials properties may be found in the materials section of this report.

c. Reflector

1) Present Status

Sintered BeO was chosen as the reflector material in the PWAR-20 reactor design because of its favorable nuclear performance and its advanced state of development for high temperature reactor applications. The available information on BeO is summarized by the following statements:

a) Physical and mechanical properties required for reflector design have been measured at applicable temperatures on unirradiated material (Refs. 25 and 26). Selected physical and mechanical properties of irradiated material have been determined (Refs. 26, 27, 28, and 29) and the measurement of others such as thermal conductivity and emissivity is planned.

b) High temperature compatibility of BeO with Cb-Zr alloy containers has been demonstrated through out-of-pile couple tests at 2000F, 2400F, and 2800F extending to 1000 hours (Ref. 21). Irradiation tests of LCRE-type BeO end reflectors clad in Cb-1 Zr alloy for 1000 hours at 2200F showed no adverse effect on compatibility (Refs. 30 and 31).

c) The fabrication technology for sintered BeO in various sizes and configurations has been established and parts are manufactured in production quantities (Ref. 32).

d) Irradiation stability tests of BeO indicate that the material is capable of retaining integrity for the SNAP-50/SPUR design conditions of 1×10^{21} nvt (> 1 Mev) at temperatures above 2000F (Refs. 26 and 28). Irradiation effects which result in anisotropic growth and cracking of BeO appear to be dependent upon temperature, time, and neutron flux. Volume expansion of less than one percent has been observed for irradiation conditions of 1850F and 0.4×10^{21} nvt (> 1 Mev) at 1 to 3×10^{14} nv (> 1 Mev), (Ref. 26). These data suggest that swelling of BeO at SNAP-50/SPUR reactions would not be a problem although verification is needed at specific design conditions of temperature, time, and flux.

As noted in statement (a) above, emissivity is one of the properties which has not yet been measured on irradiated BeO. Since the PWAR-20 reactor design is radiatively-cooled, stable emissivity characteristics are required over the reactor lifetime; thus, the effect of irradiation on emissivity must be established. In addition, improved BeO cooling through the development of a stable high temperature coating is a desirable goal. A minimal effort on such high temperature coatings is now in progress.

2) Plans for the Balance of FY 1965 and FY 1966

During FY 1965 and FY 1966, an irradiation program will be initiated, if required, to verify design extrapolation concerning BeO stability for 10,000-hour, 2300F to 2500F service. The stability of candidate high-emissivity coatings will be studied at design environmental conditions.

3. Shield Development

a. Present Status

Honeycomb-reinforced, cast lithium hydride was selected for the neutron shield material as the best compromise of shielding and physical properties. Extensive test data are available to support the choice, although additional testing is considered necessary to provide the required confidence in extrapolations to design conditions. The following statements summarize the present information on LiH:

1) Physical and mechanical properties of unirradiated lithium hydride have been measured through the temperature range of design interest (Refs. 33, 34, 35, and 36).

2) Honeycomb-reinforced construction is capable of limiting thermal stress cracking in lithium hydride (Refs. 35, 36, and 37).

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3) Hydrogen migration and permeation studies indicate acceptable stability up to 1000F in stainless steel canned systems although data are limited. Additional hydrogen stability tests are planned using improved experimental techniques (Refs. 38 and 39).

4) Irradiation stability as measured by gas release, swelling and cracking is not believed to be a problem based on limited tests at conditions approximating SNAP-50/SPUR temperature and integrated neutron dosage, but in one-fiftieth of the time (Refs. 36, 40, and 41). The absence of time-dependent effects must be verified through irradiation experiments at more realistic conditions.

5) Compatibility tests of lithium hydride with various austenitic and martensitic stainless steels have demonstrated no appreciable corrosion effects in the stressed or unstressed condition (Refs. 35 and 36).

b. Plans for the Balance of FY 1965 and FY 1966

The FY 1965 and FY 1966 program will include irradiation testing to determine time-dependent effects at SNAP-50/SPUR conditions where not available from other government sponsored research.

4. Control Drive Development

a. Present Status

The development of non-lubricated control drive components capable of operating at elevated temperatures under radiation has been in progress for several years. A brief summary of recent control drive component tests is shown below:

Number of Tests	Component	Temperature and Environment	Total Hours Tested
61	Bearing, Anti-friction and Journal	500-1000F helium, NaK	76,000
12	Gears, Spur, Harmonic	600F NaK, helium	23,000
10	Electric Motors, Switches	500-600F helium	20,000
6	Electric Motors, Switches	600F-10^{19} nvt helium	17,000

The test experience has been in inert atmosphere. It will probably be possible to encapsulate much of the control drive in such an atmosphere, but some components will necessarily be exposed to the vacuum of space. For this reason, and because encapsulation entails additional weight and complexity, it is desirable to develop components which can operate in vacuum.

The problems of self-welding in bearings, sublimation of solid lubricants, and electrical arcing under vacuum conditions will require a comprehensive evaluation of suitable materials and methods to be used in the final control devices.

The component development work currently in process is oriented toward extending the reliable operational range of electro-mechanical devices to the 800F level under space environmental conditions. Components adjacent to the reactor, such as flexure pivots, are to be tested at 1000F to determine their applicability as high temperature, vacuum environment bearings. The present status of this work is as follows:

1) Screening tests on bearings and motors at 800F have been initiated.

2) Harmonic gears in vacuum are being tested.

3) The vacuum laboratory is being equipped and ion pump shakedown tests have been started.

4) Orders have been placed for experimental 1000F flexure pivots and related test stand parts.

In addition to the component testing effort, a more basic program is in progress to provide the information for improvements in the control drive. This work includes:

1) Evaluation of basic control concepts, including design studies.

2) Evaluation of materials.

3) Procurement of experimental hardware for feasibility determinations.

4) Extension of operational environments on key components to 800F.

5) Initiation of high temperature vacuum environmental screening tests.

b. Plans for the Balance of FY 1965 and FY 1966

Design studies on several reflector and control arrangements will be completed to permit selection of a reflector suspension system and the location of the power package. Candidate anti-friction bearing materials will be screened in environmental bearing test stands and electrical potting materials at 800F will be evaluated.

Existing bearing and motor test stands are to be modified to permit development work for the new application including vacuum environmental testing.

Reactor control stability determinations will be made by use of the analog computer and control mockups.

Work in FY 1966 will essentially consist of an increased activity in procurement of test hardware for evaluation of materials and control method feasibility. Endurance testing will be accelerated.

A primary control drive concept incorporating materials found acceptable in the screening tests will be selected.

C. Liquid Metal Pumps and Other Components

1. Liquid Metal Pumps

a. Introduction

For the SNAP-50/SPUR powerplant, CANEL has the responsibility for designing and developing all pumps in both the reactor coolant and power conversion systems. As presently conceived for the 300-Kwe powerplant, this work entails the development of a reactor coolant pump circulating lithium at 1900F, a condensate pump circulating potassium at 1121F, and a condenser coolant pump circulating NaK at 1096F. These pumps must operate with a high degree of reliability and with no external leakage for periods in excess of 10,000 hours without maintenance. High performance is also mandatory to meet the system size and weight requirements. After detailed studies of all the variables involved, hermetically-sealed centrifugal pumps powered by canned electric motor drives have been selected for all three pumps.

b. Design Description

1) Reactor Coolant Pump

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A cross section of the reactor coolant pump is shown in Fig. 60. This pump is required to circulate 345 gpm of 1900F lithium with a head rise of 158 feet. The design shaft speed of the pump is 11, 300 rpm with an inlet pressure of 27 psia.

As shown in Fig. 60, the rotor assembly is mounted on a single shaft which is supported by lithium lubricated journal and thrust film bearings. The rotor of the canned motor is straddled by the two journal bearings. A double-acting thrust bearing is located between the motor and the secondary impeller. The main stage pump impeller is positioned at one end of the shaft to provide a satisfactory thermal gradient in the structure connecting the reactor coolant passages and the lower temperature lithium used to cool and lubricate the motor and bearings. Multiple centrifugal dynamic shaft seals are located at both ends of the motor rotor to scavenge the liquid lithium out of the rotor gap at start-up. These seals are designed to maintain a liquid-to-vapor interface on the rotating vanes of the dynamic seal impeller during operation, thereby maintaining a saturated lithium vapor in the motor rotor gap. A ceramic sleeve is fitted into the rotor gap to serve as a bore seal and prevent lithium from contacting the electrical materials in the stator. Similarly, a thin refractory metal liner is utilized to encapsulate the rotor and prevent lithium attack of the rotor materials.

As shown in Fig. 60, a rather complex flow circuit is required to circulate the lower temperature lithium to cool the rotor and stator of the motor and lubricate the journal and thrust film bearings. All of the return flow is collected at the inlet to the secondary impeller. At the discharge of the impeller, the lithium is ducted to the thrust bearing and the two journal bearings. The flow from the journal bearing at the far end of the shaft is circulated through the motor rotor before mixing with the flow from the other two bearings. All of the flow, less the direct leakage from the thrust bearing back to the secondary impeller, is then circulated through the annular heat exchanger surrounding the motor stator. After cooling the stator, the lithium is circulated through an external lithium-to-NaK heat exchanger to reject the heat picked up in the pump. The flow from the external heat exchanger is then ducted back to the inlet of the secondary impeller.

Pressure control is maintained by an accumulator in the 1900F lithium circuit. A narrow annular passage statically connects the high and intermediate temperature lithium inside the pump. The liquid evacuated from the motor rotor gap on start-up is forced through this annular passage and into the accumulator by the pumping action provided by the dynamic seals. As shown in Fig. 60, instrumentation of the liquid metal pump is limited to a speed indicator and thermocouples at strategic locations.

SNAP-50/SPUR REACTOR COOLANT PUMP

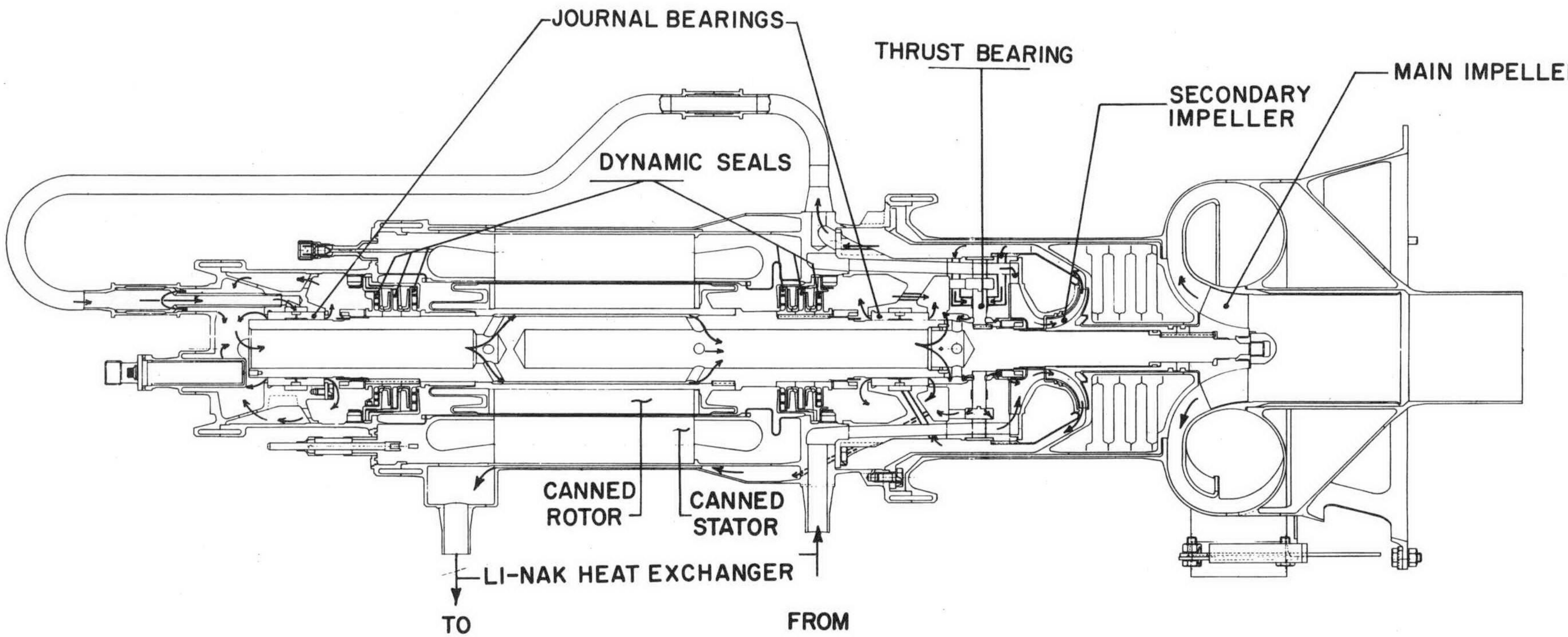

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FIG 60

2) Condensate and Condenser Coolant Pumps

The general design features of the two power conversion system pump are similar to the reactor coolant pump just described, except for different materials of construction. The condensate pump is required to circulate 43.5 gpm of 1121F potassium with a head rise of 339 feet and an inlet pressure of 34 psia. The condenser coolant pump is required to circulate 130 gpm of 1096F NaK with a head rise of 60 feet and an inlet pressure of 20 psia. A portion of the flow through the secondary impeller of the condensate pump is used to lubricate and cool the turboalternator. In a similar fashion, the flow from the secondary impeller of the NaK pump is used to provide auxiliary cooling as required throughout the powerplant system.

The design shaft speed for both of these pumps is 11,300 rpm. UNCLASSIFIED

c. Development Program

Primary emphasis has been placed on the development of the reactor coolant pump. The development of this pump represents the greatest advancement in the state-of-the-art as compared to the condensate and condenser coolant pumps.

The development program can be divided into four major phases. The first phase involves the necessary detailed testing of such vital components as the journal and thrust film bearings, centrifugal dynamic shaft seals, impellers and scrolls, motor stator bore seal, and development of the electric drive motor. In addition, component testing of jet pumps is required for the power conversion system. Several supplementary tests will be performed to obtain the necessary material data, particularly with regard to the film bearings and cavitation erosion damage of the pump impellers.

The second phase of the program is concerned with the development tests of full-scale water pump models. Here, for the first time, all parts of the pump are assembled into a test unit and tested at simulated liquid metal operating conditions. Due to the relative ease of instrumentation in water, detailed calibrations of all important parts of the pump will be accomplished prior to the liquid metal tests.

The third phase in the pump development program is the liquid metal testing of full-scale pumps in vacuum test chambers, such as shown in Fig. 61 . The principle objective of this phase is to develop the necessary degree of reliability of these pumps for the extended service life requirements by conducting multiple tests at simulated operating conditions.

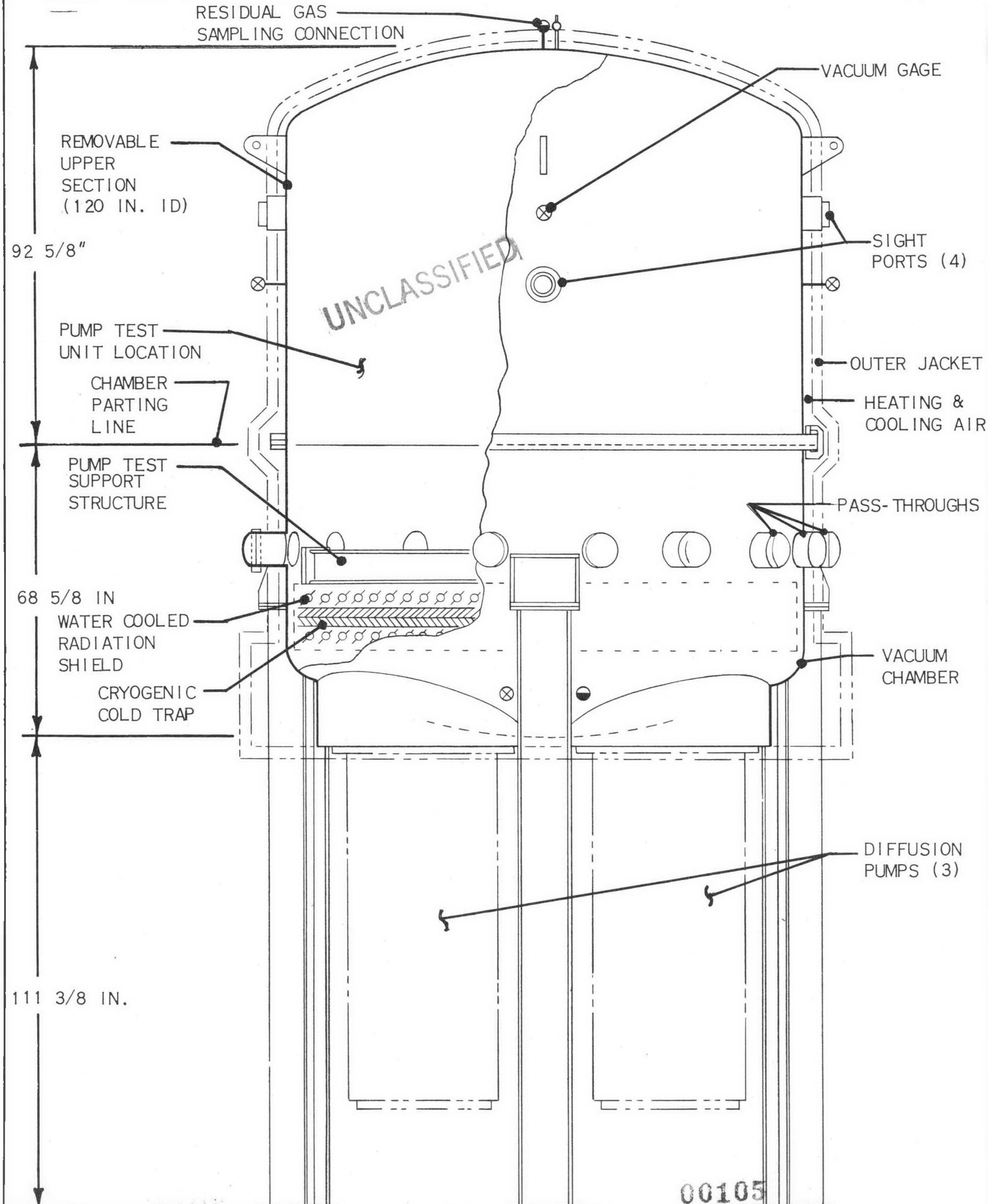
The last phase in the ground test program for the liquid metal pumps will be the incorporation of these pumps into the major systems tests, such as the reactor test, boiler test, and the turboalternator test.

At present, the design and analysis of reactor coolant, condensate, and condenser coolant pumps is in progress with major emphasis on the reactor coolant pump. All phases of the component development work have been initiated. One of the vital component tests is the successful scavenging of the liquid metal from the motor rotor gap on start-up. The water test units shown in Fig. 62 are being used to visually observe with high speed photography the manner in which the rotor scavenging occurs. Successful scavenging has been accomplished with room temperature water in about four seconds during repeated tests. Another important development at CANEL has been the machining of pump impellers on the tape-controlled milling machine shown in Fig. 63 . This figure also displays an initial full-scale model of the reactor coolant pump impeller fabricated from Cb-1 Zr alloy and a full-scale condenser coolant pump impeller machined from type 316 stainless steel. Full-scale pump impeller and scroll tests are being conducted in the water test facilities shown in Fig. 64 . These

[redacted]

FIG 61

SNAP-50/SPUR REACTOR COOLANT PUMP VACUUM TEST CHAMBER



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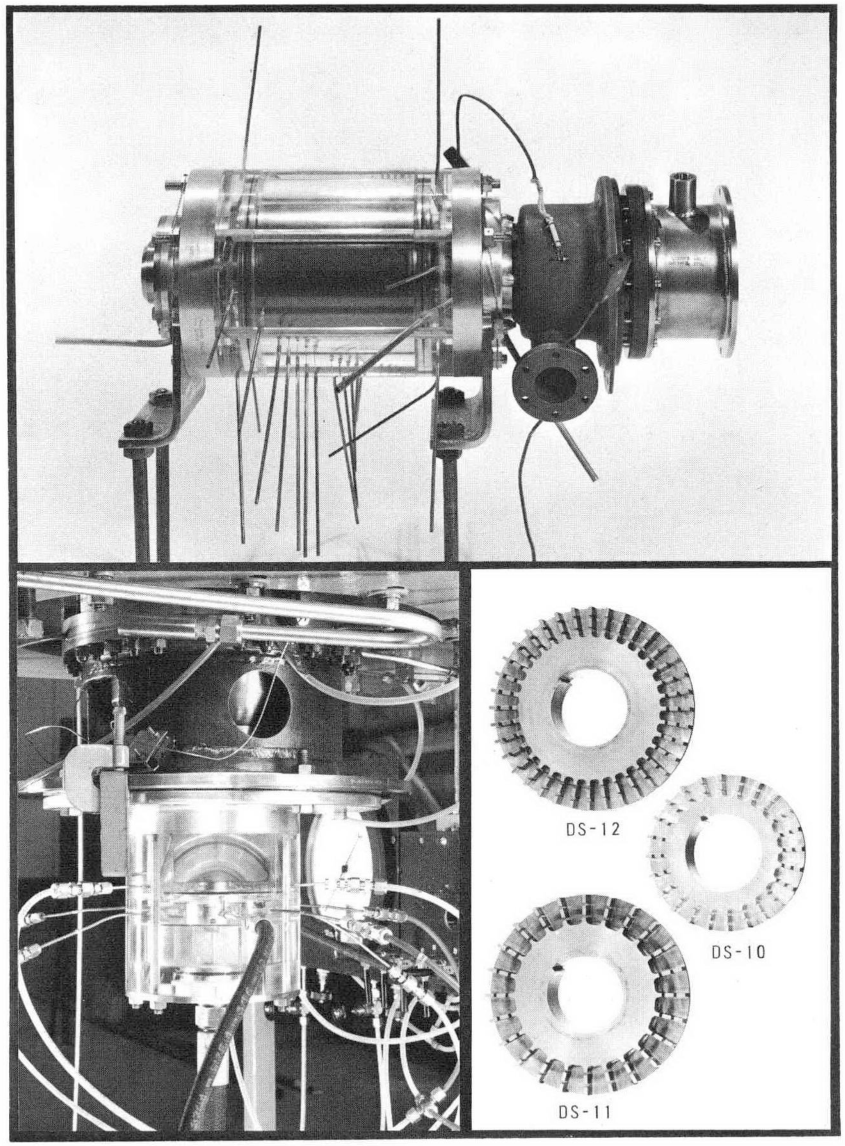
MOTOR ROTOR GAP SCAVENGING TESTS

A.
MOTOR ROTOR CAVITY
TEST UNIT



B.
DYNAMIC SEAL
TEST STAND

C.
DYNAMIC SEAL
IMPELLERS

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FIG 62

FABRICATION OF PUMP IMPELLERS

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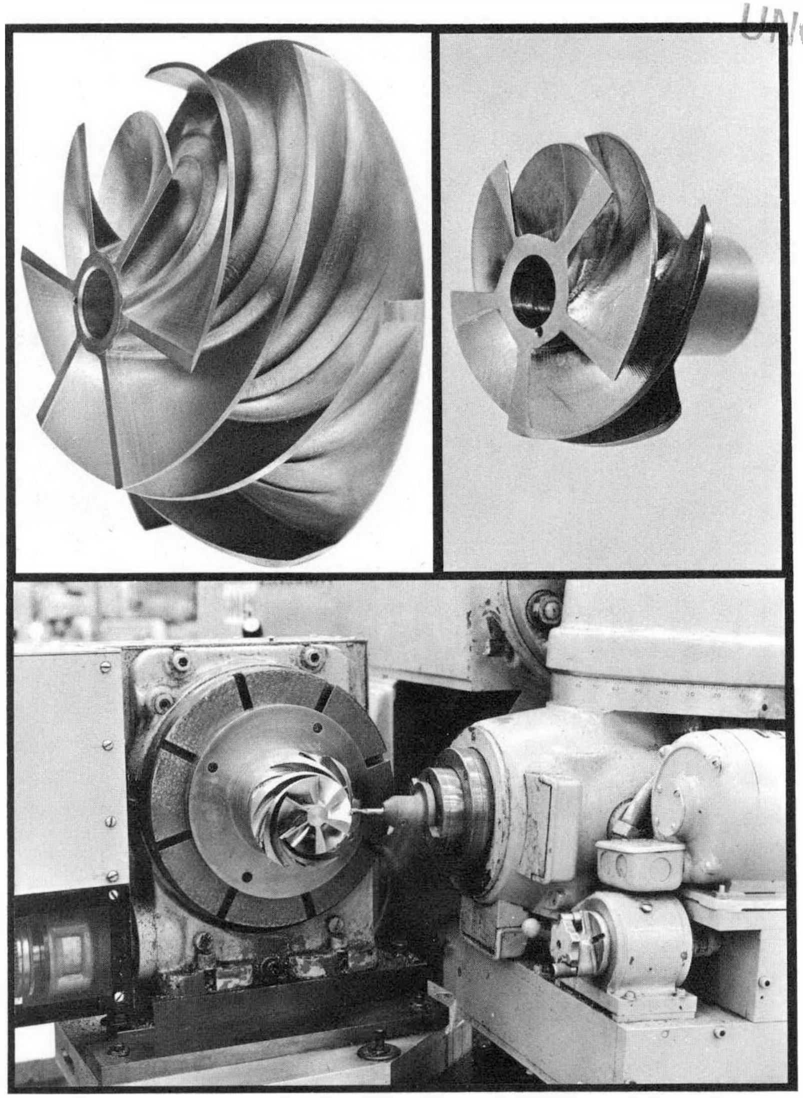
A.

REACTOR COOLANT
COLUMBIUM
PUMP IMPELLER



B.

CONDENSER COOLANT
STAINLESS STEEL
PUMP IMPELLER

C.

NUMERICALLY
CONTROLLED
IMPELLER MACHINING

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FIG 63

FIG 64

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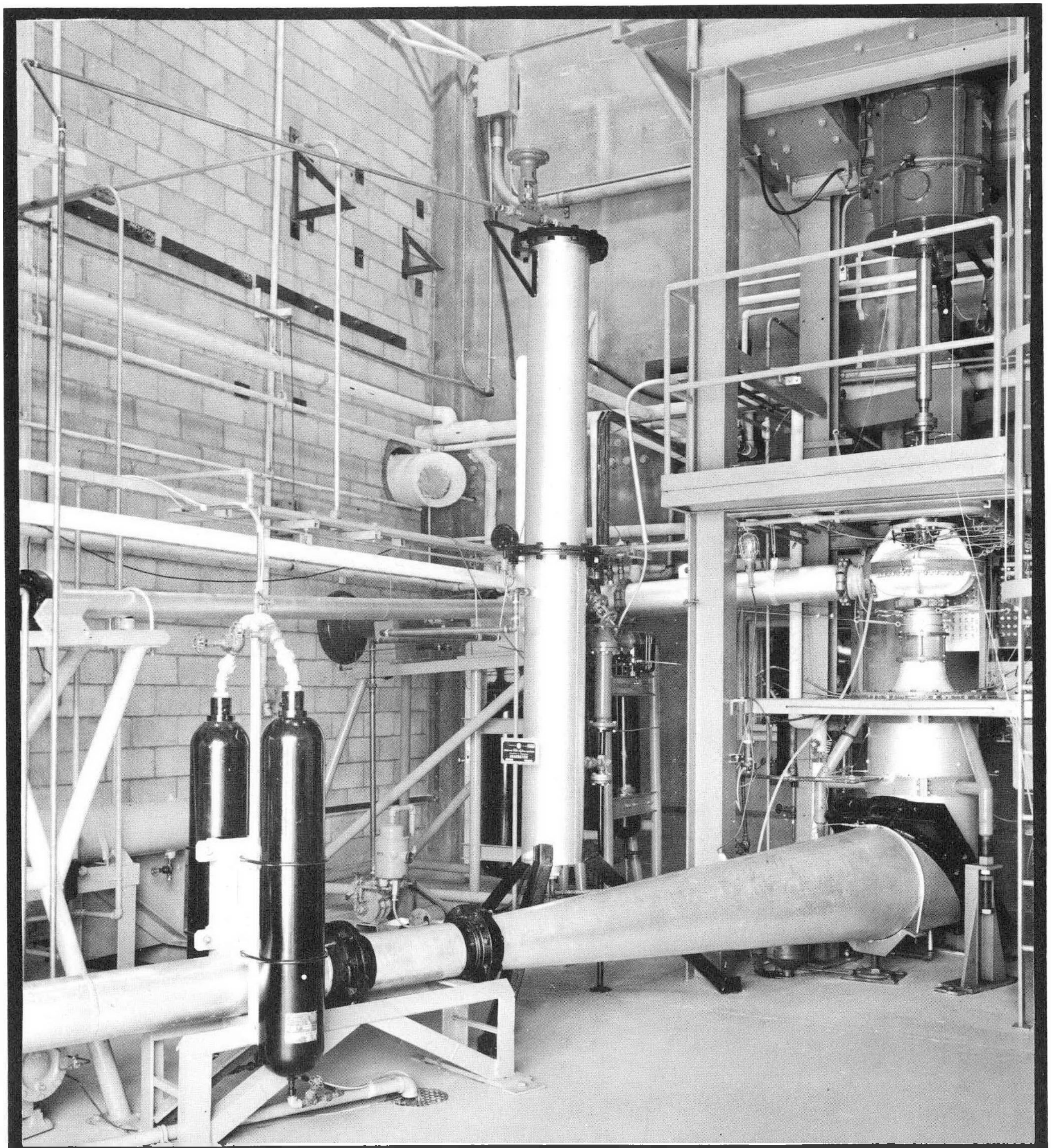
IMPELLER AND SCROLL PERFORMANCE TEST STAND



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component hydraulic tests are needed to establish the specific head rise and flow rate requirements for the fixed shaft speed of these pumps. A summary of impeller and dynamic seal development tests is presented in Fig. 65. One of the prime areas of component development has been in the area of the fluid film journal and thrust bearings. A summary of bearing development tests is shown in Fig. 66. To date, three configurations of hydrodynamic journal bearings have been tested in lithium at 600 to 800F in the test facility shown in Fig. 67. The third test currently in progress consists of a Mo-CbC bearing with a journal diameter of 2.500 inches, which has been successfully tested at shaft speeds up to 8400 rpm in 625F lithium for over 3000 hours. Typical journal bearing configurations being evaluated in water and liquid metal are shown in Fig. 68. Component tests of full-scale thrust bearings in water are being conducted in the test unit shown in Fig. 67. Other test units for testing component journal and thrust bearings in water and liquid metal are in process of fabrication and assembly. In addition, rotor dynamics characteristics of water test units simulating the rotating mass of the pump are being conducted, as shown in Fig. 69. Rotor dynamics test units with the simulated motor rotor mass overhung from the journal bearings have also been fabricated and tested. The rotor dynamics development program is summarized in Fig. 70. Initial emphasis was placed on hydrostatic bearings in the bearing program. Presently, hybrid bearings incorporating both hydrodynamic and hydrostatic features have been selected for the pumps. The friction and wear apparatus shown in Fig. 71 has been used to conduct multiple tests in NaK and lithium to screen bearing materials for the fluid film bearings. Each set of specimens has been subjected to several starts and stops to simulate the start-up of the pumps.
Another important area of component development is the ceramic-to-refractory metal bore seal joint depicted in Fig. 72. The specimen shown has successfully survived 25 thermal shocks in lithium from 800F to 1200F without evidence of leakage. Tests of full-scale bore seal joints are scheduled in the near future. The ceramic bore seal protects the stator of the motor from attack by liquid metal.

The test facility for screening candidate materials for cavitation damage resistance in liquid metal is shown in Fig.73 together with a fully assembled test disk. The disk is spun in a pot of hot liquid metal and the cavitation produced by the inducer holes collapses on the target specimens. Extensive work on rotating disks in water was performed prior to conducting a 30 hour test in lithium at 1000F.

Design and development of the 400-cycle induction electric motor will be accomplished under subcontract. A summary of the motor development program is presented in Fig. 74.

2. Other Components

a. Valves

High temperature liquid metal valves are required to start and control the powerplant. Similar valves were developed for the LCRE, as shown in Fig. 75 and reported in Ref. 42. Two of the LCRE valves will soon complete 10,000 hours of operation in the non-nuclear systems test. The basic development approach to the valves required for the SNAP-50/SPUR powerplant is to use features of the LCRE valves and modify as necessary to meet the space environment conditions. Tests will be conducted on the thin wall bellows stem seal to insure that long term operation can be attained. Complete valve assemblies will be incorporated and tested in conjunction with major systems tests such as the non-nuclear powerplant test.

b. Accumulators

Each liquid metal system requires an accumulator to establish system pressure and to provide for volume variations due to temperature and/or phase changes. The bellows type of accumulator has been selected and is being developed. Refractory alloy bellows of the size required, approximately 12 inches in diameter, are being formed out of thin two-ply butt-welded material. A typical in-process bellows is shown in Fig. 76 . The development program calls for the evaluation of these bellows under simulated operating conditions.

IMPELLER & SEAL DEVELOPMENT

WATER TESTS OF 25 IMPELLER CONFIGURATIONS ESTABLISHED CORRELATIONS FOR:

* SLIP FACTORS
* HEAD COEFFICIENTS AT BEST EFFICIENCY
* CAVITATION DESIGN MARGIN
* TIP CLEARANCE EFFECTS

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WATER TESTS OF 12 DYNAMIC SEAL CONFIGURATIONS ESTABLISHED CORRELATIONS FOR:

* TORQUE COEFFICIENTS
* HEAD COEFFICIENTS
* INGASSING LIMITS

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FIG 65

HYDROSTATIC & HYDRODYNAMIC BEARING DEVELOPMENT

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WATER TESTS OF 5 HYDROSTATIC AND 3 HYDRODYNAMIC CONFIGURATIONS ESTABLISHED CORRELATIONS FOR:

* LOAD CAPACITY
* FLOW REQUIREMENTS
* ORIFICE COMPENSATION

LITHIUM TESTS OF 3 HYDRODYNAMIC CONFIGURATIONS

* 2100 HOURS, 600-800F, 200 LB. LOAD, 5000 rpm
* 110 HOURS, 600-800F, 55 LB. LOAD, 5500 rpm
* OVER 3400 HOURS, 625F, 11.3 LB. LOAD, 8400 rpm (TEST IN OPERATION)

CNLM - 5889

FIG 66

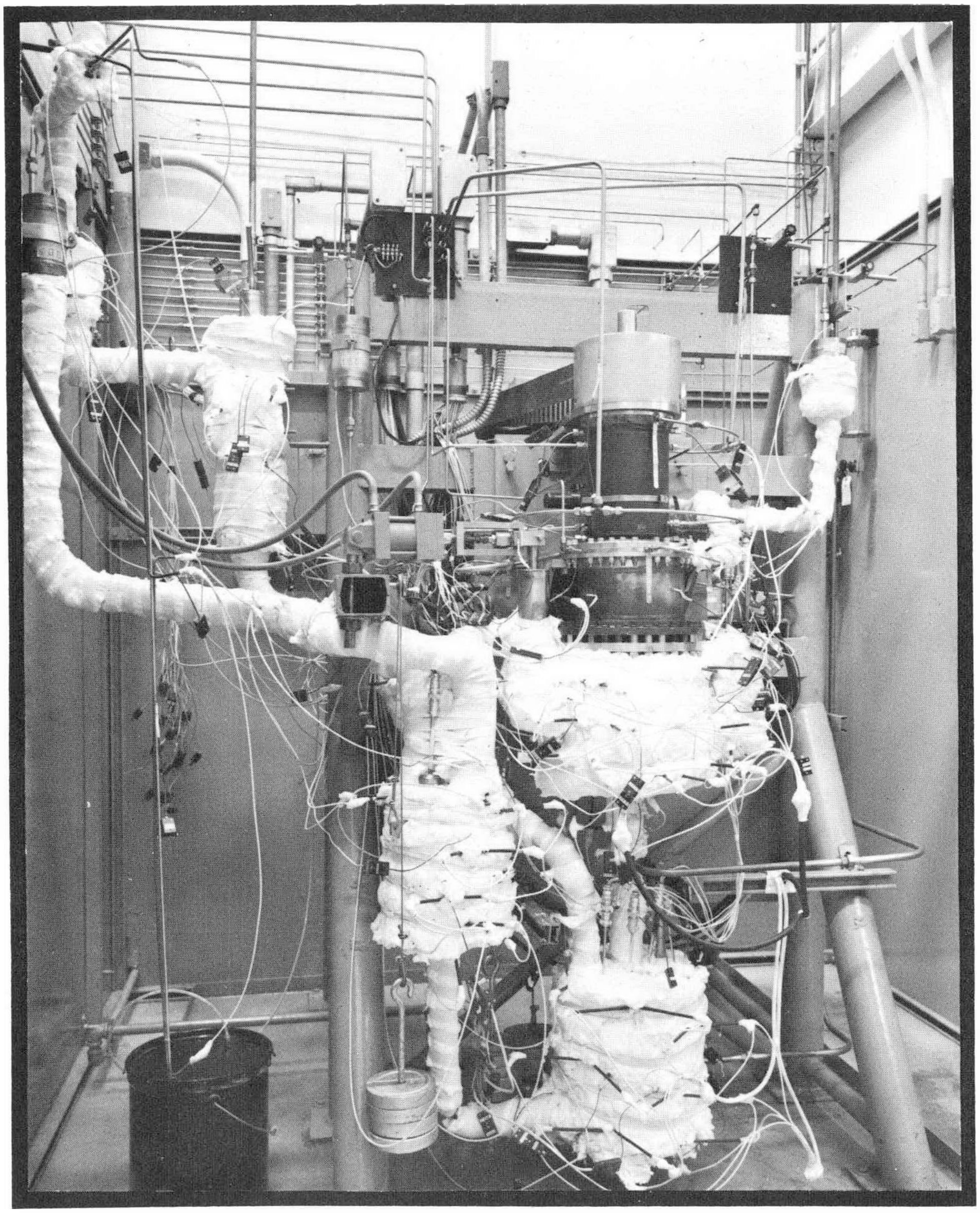
V-54

000111

FIG 67

JOURNAL AND THRUST BEARING TESTS

UNCLASSIFIED



A. LITHIUM JOURNAL BEARING RIG

000112

[redacted]

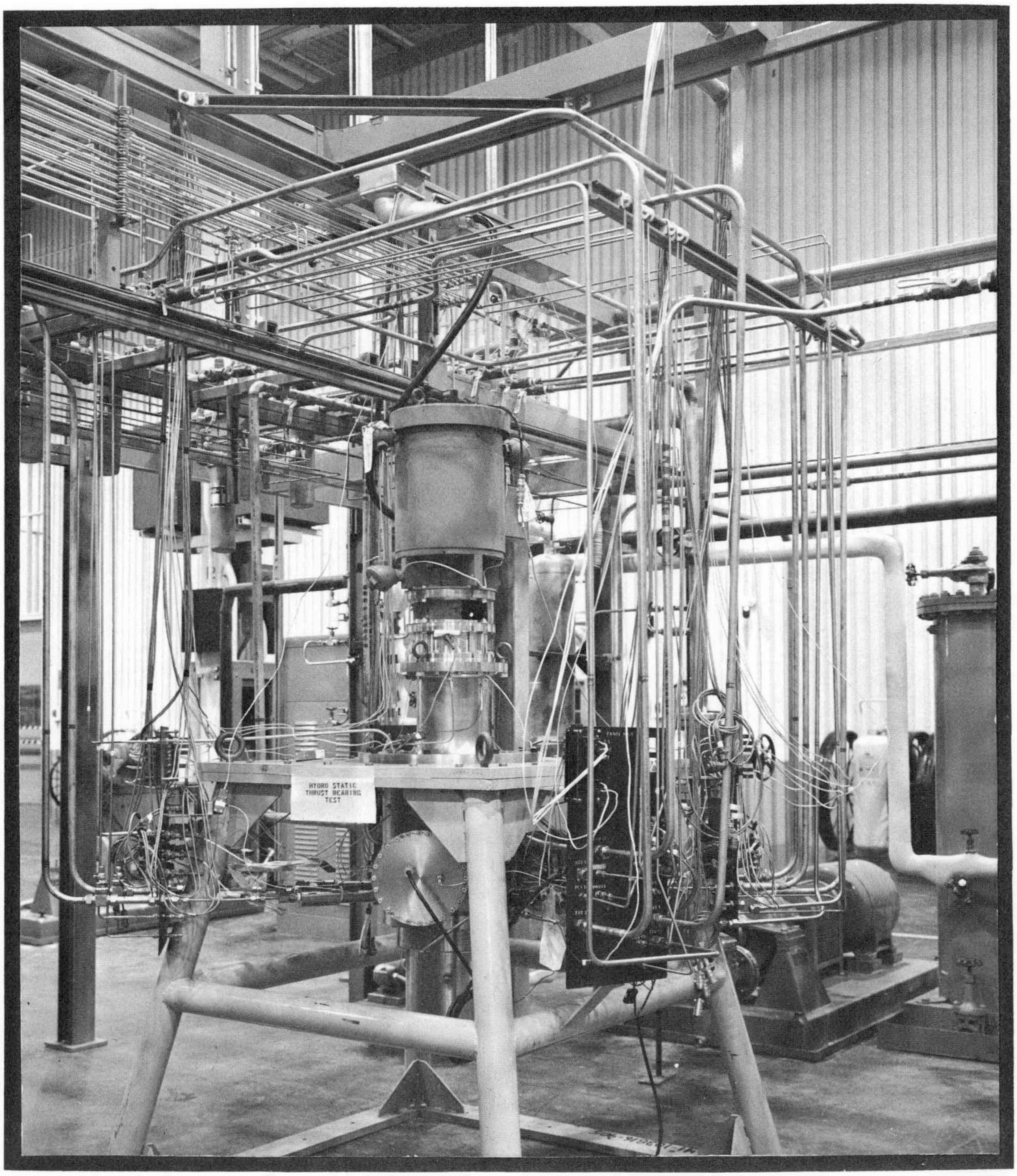
FIG 67

C-13368

UNCLASSIFIED

JOURNAL AND THRUST BEARING TESTS

(CONTINUED)



B. WATER THRUST BEARING RIG

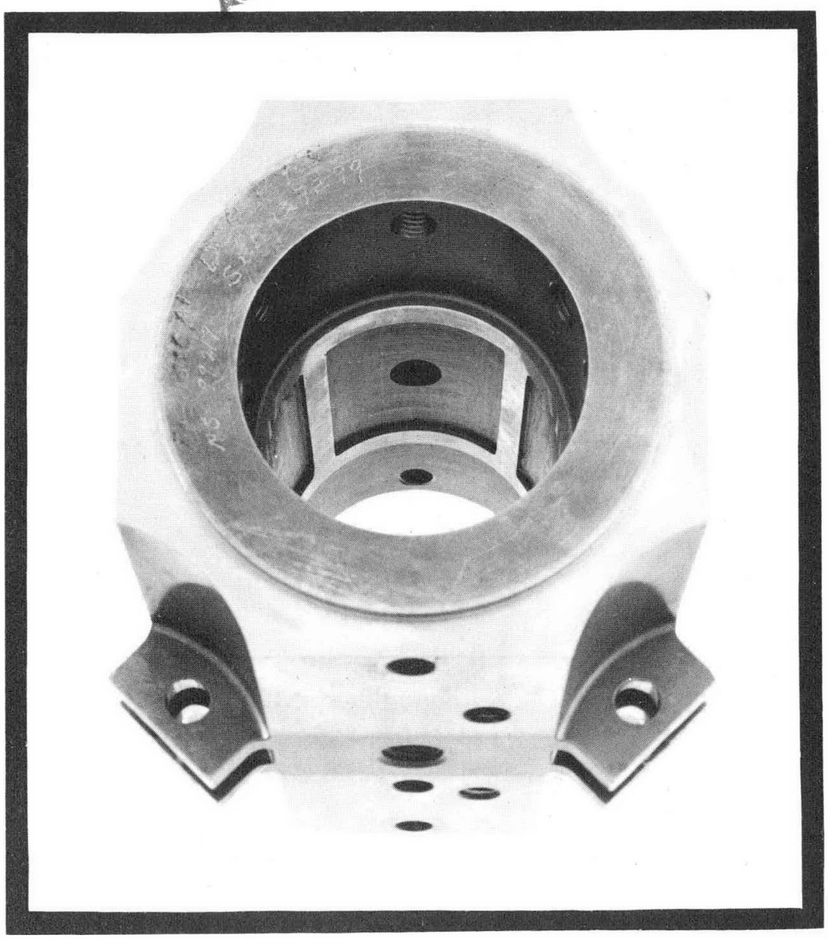
00113

[redacted]

FIG 68

TYPICAL JOURNAL BEARING CONFIGURATIONS

UNCLASSIFIED



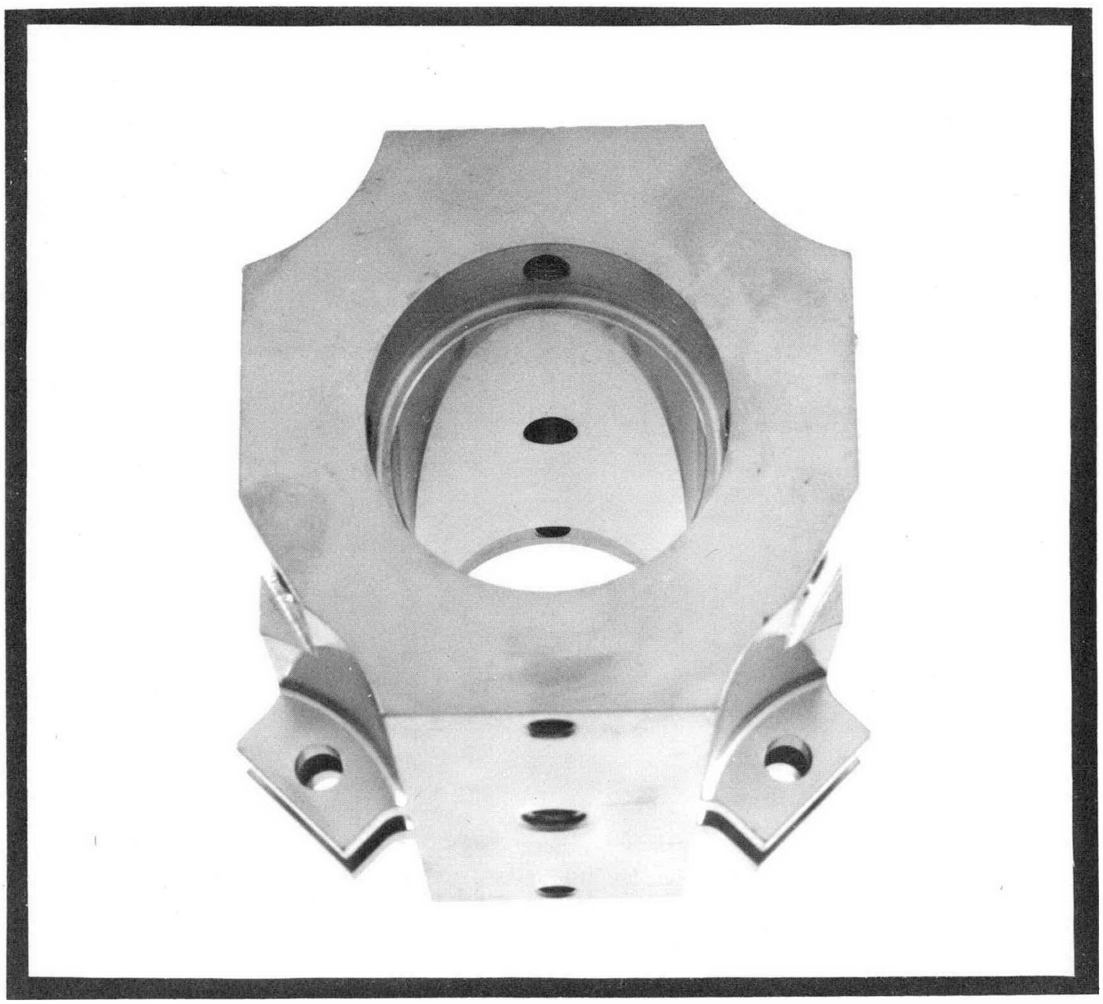
L/D = 1

A. HYDROSTATIC BEARING

00114

TYPICAL JOURNAL BEARING CONFIGURATIONS

(CONTINUED)

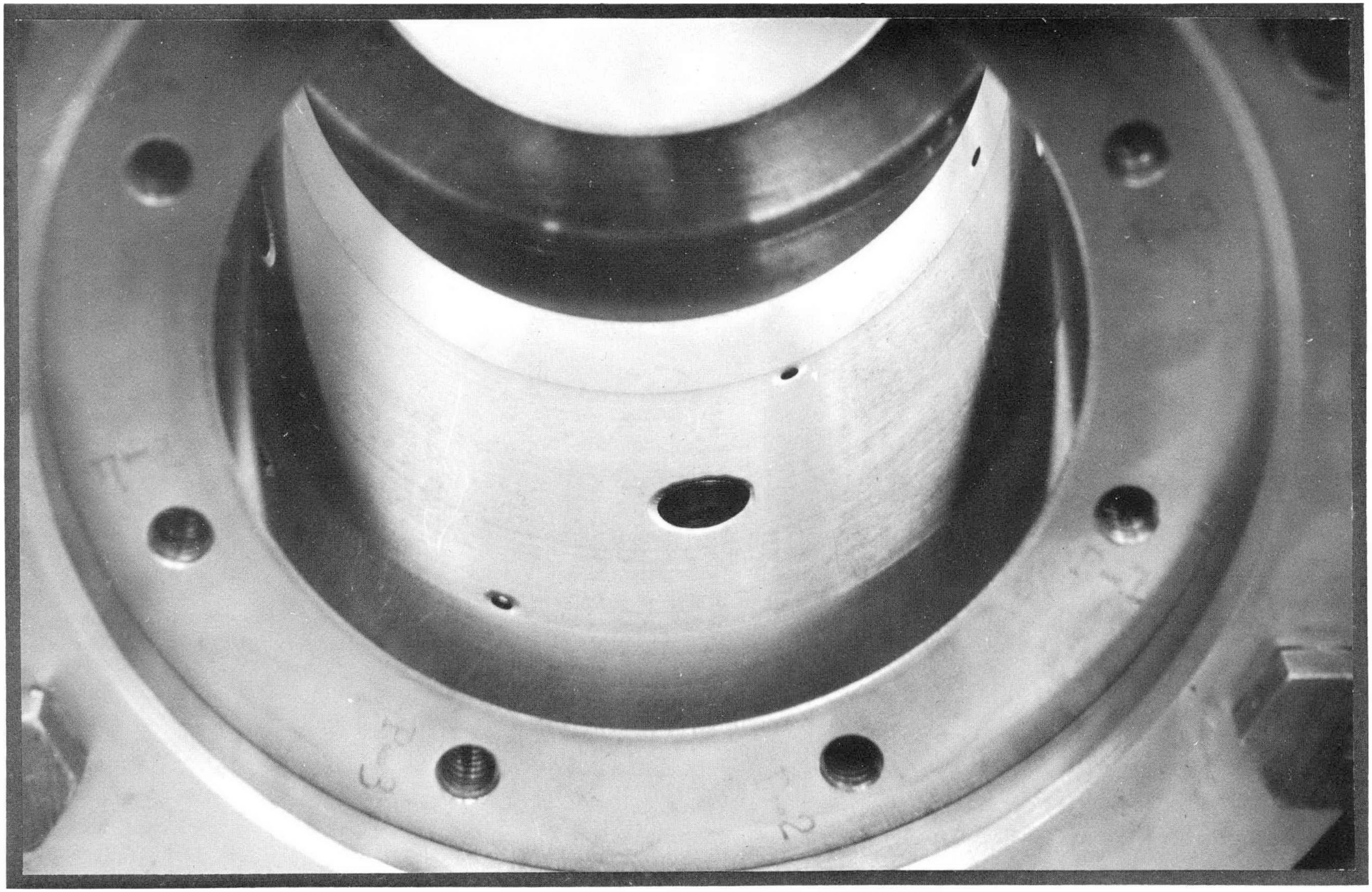


DIA. = 2.375"   L/D = 1.5

B. HYDRODYNAMIC BEARING

TYPICAL JOURNAL BEARING CONFIGURATIONS

(CONTINUED)



C. STEP BEARING

CNLM - 5889

FIG 68

UNCLASSIFIED

0C116

[redacted]

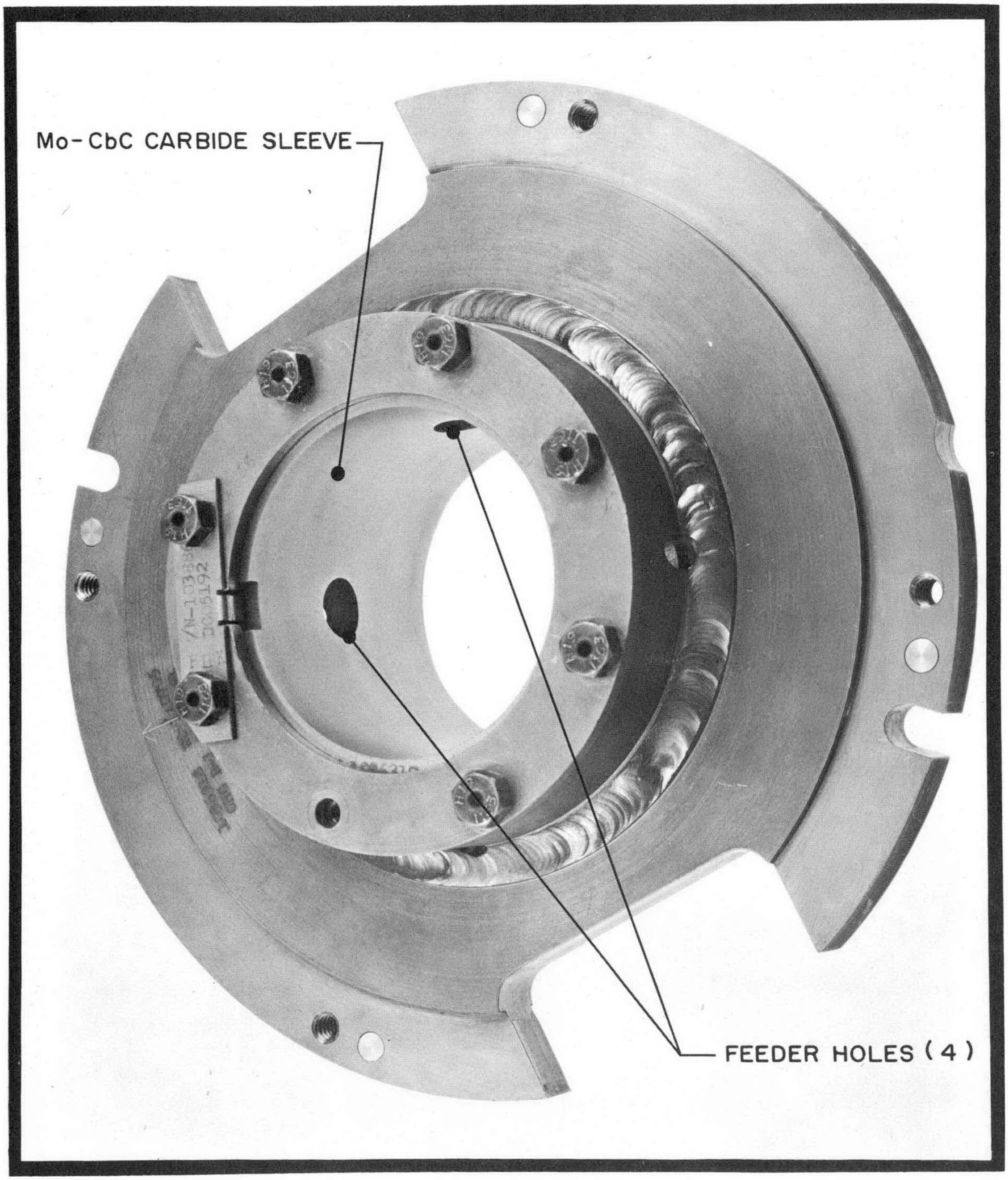
FIG 68

C-13198

UNCLASSIFIED

TYPICAL JOURNAL BEARING CONFIGURATIONS

(CONTINUED)



00117

D. LIQUID METAL HYDRODYNAMIC BEARING

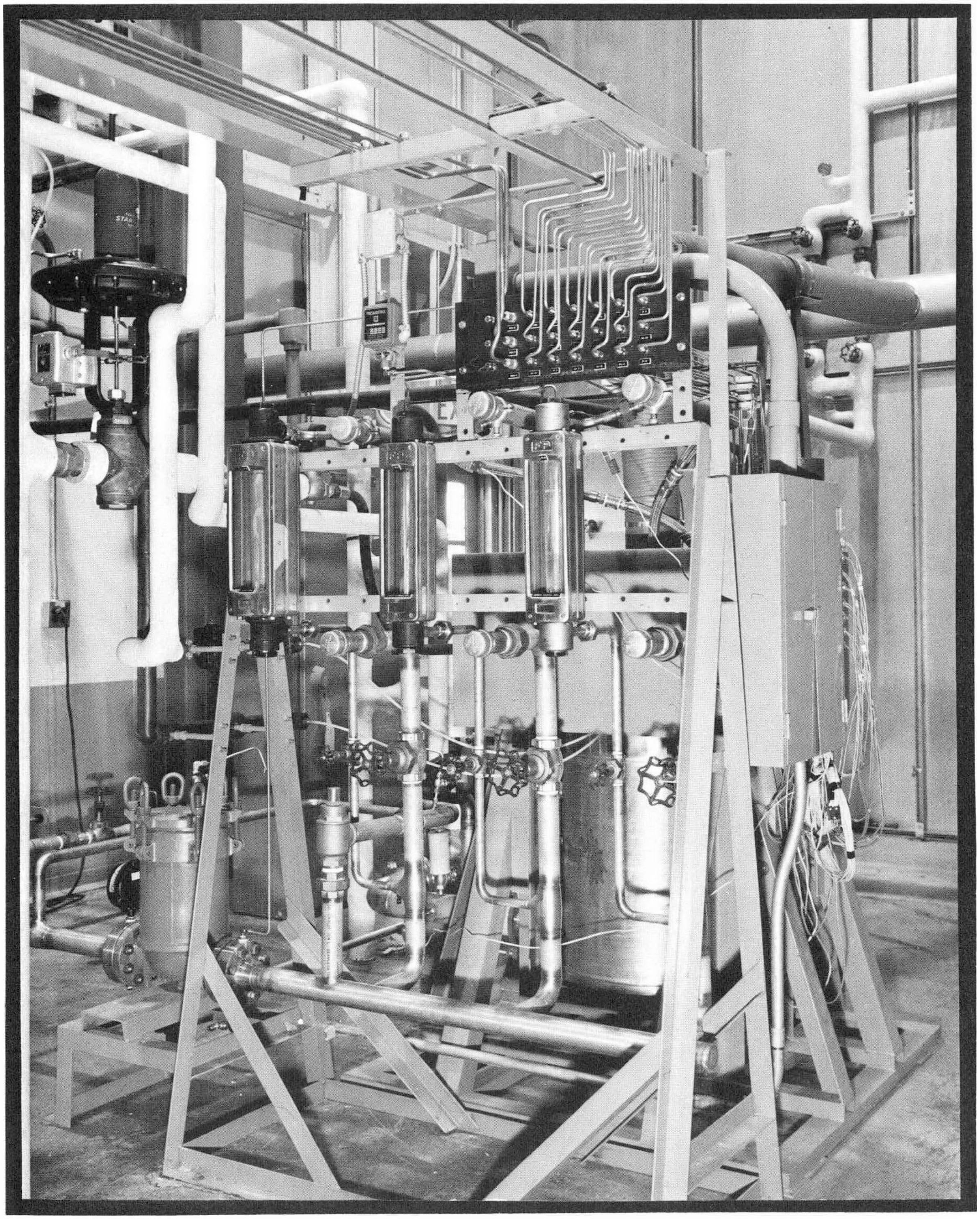
[redacted]

[redacted]

FIG 69

ROTOR DYNAMICS TESTS

UNCLASSIFIED



A. ROTOR DYNAMICS TEST UNIT

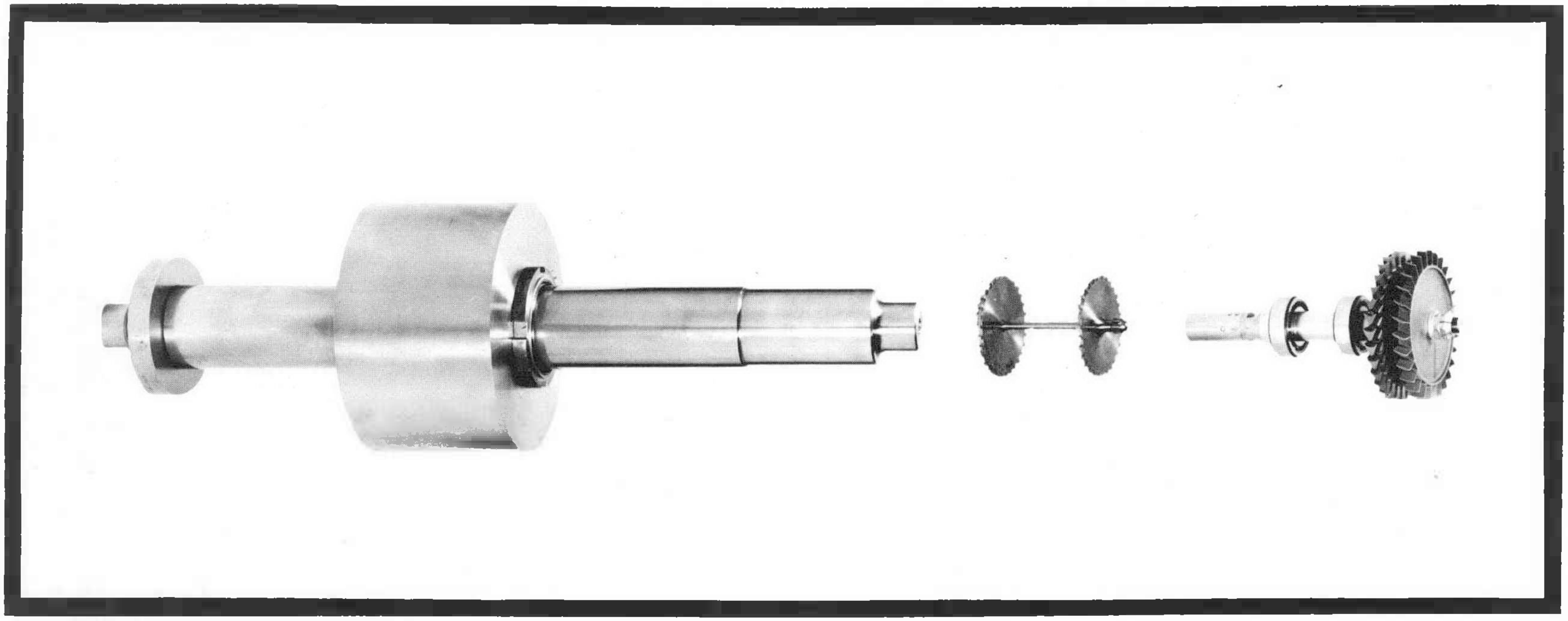
00118

[redacted]

UNCLASSIFIED

ROTOR DYNAMICS TESTS

(CONTINUED)



B. ROTATING ASSEMBLY

CNLM - 5889

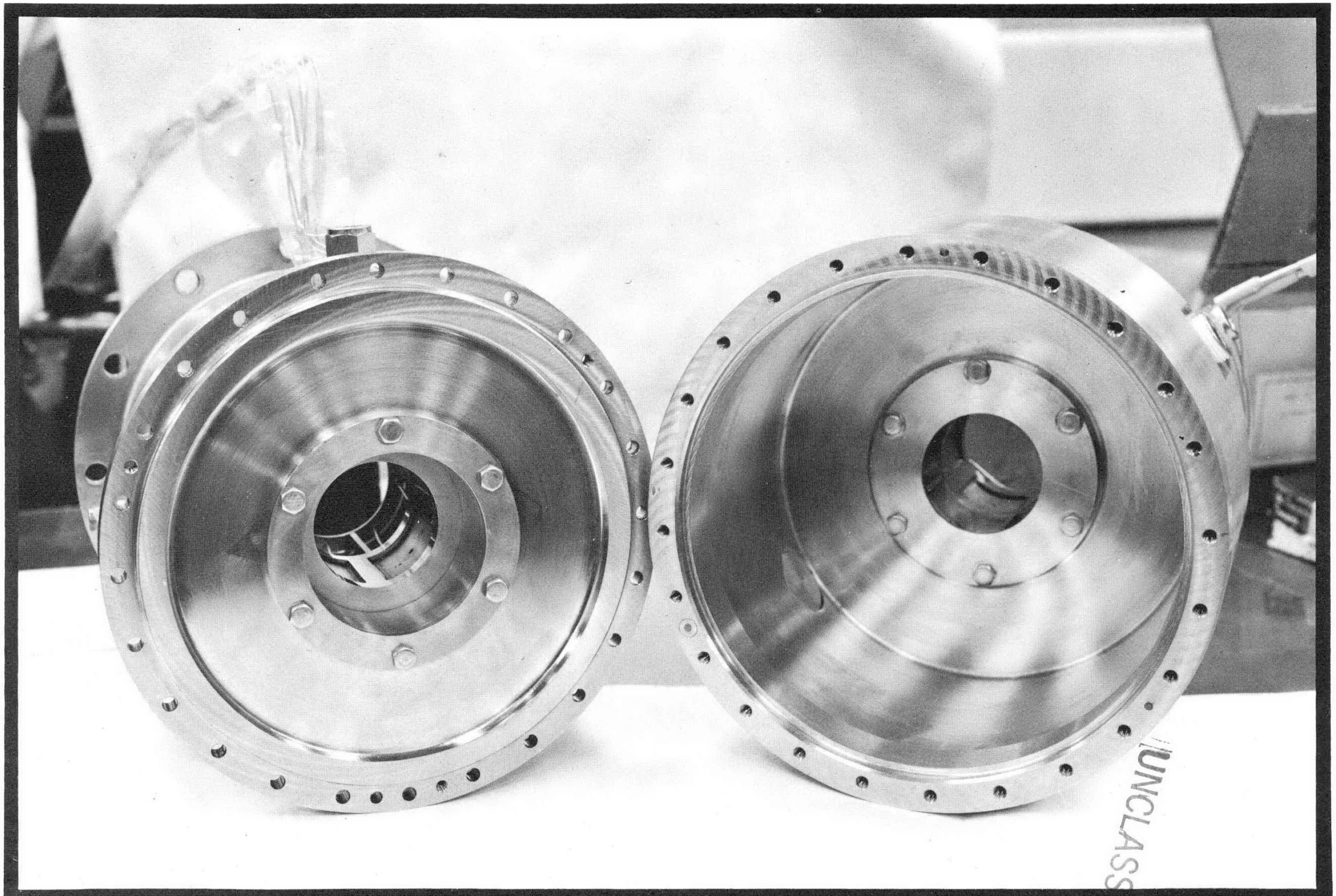
FIG 69

V-62

00119

ROTOR DYNAMICS TESTS

(CONTINUED)



C. BEARING CASE

FIG 69

UNCLASSIFIED

CONFIDENTIAL

ROTOR DYNAMICS TESTS

UNCLASSIFIED

FOUR CONFIGURATIONS TESTED IN WATER ESTABLISHED

* STABILITY LIMITS
* FRICTION COEFFICIENTS
* STARTUP CHARACTERISTICS

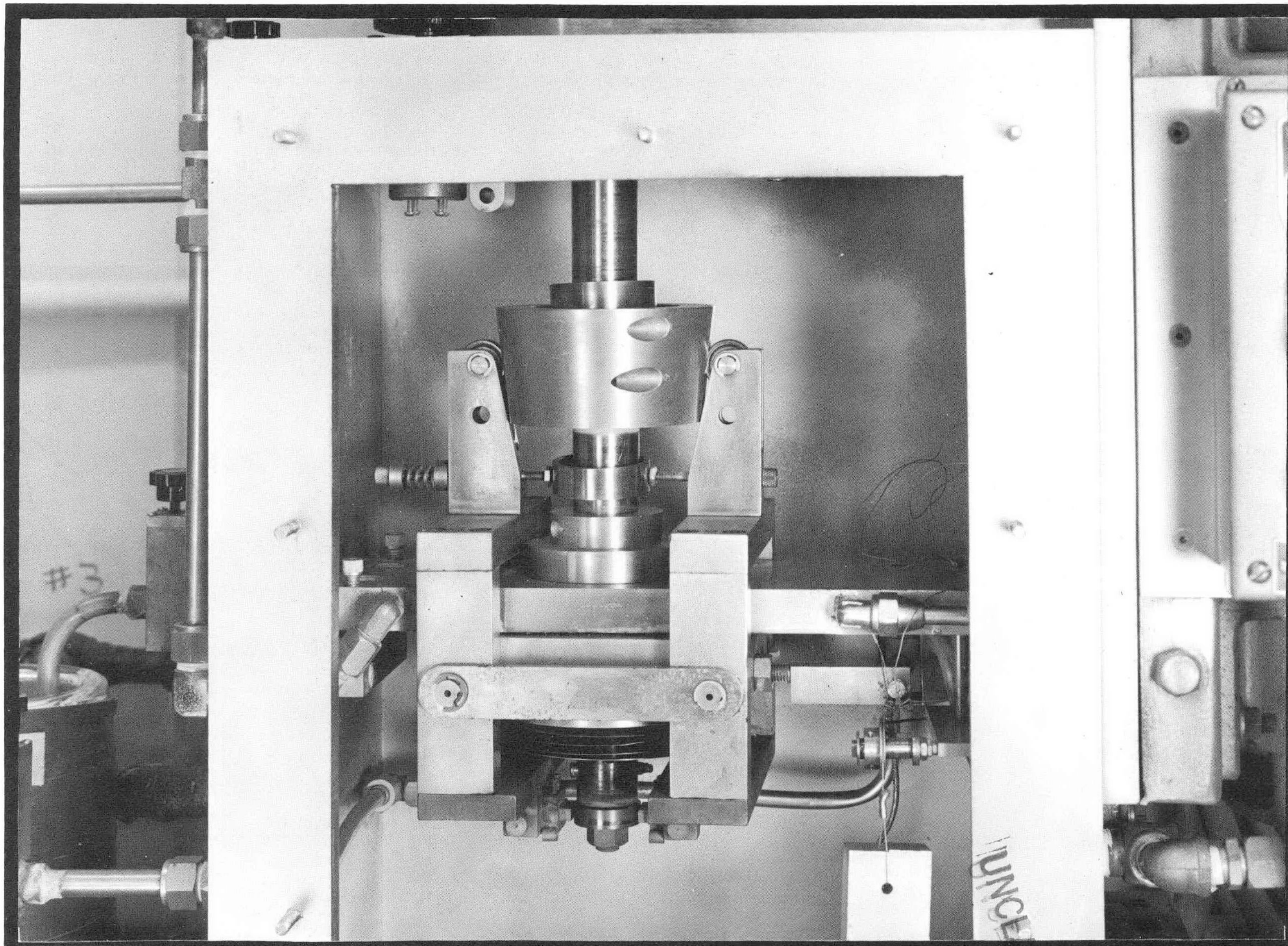
ACHIEVED

* STABLE OPERATION UP TO 14.000 rpm AT DESIGN PRESSURE LEVELS
* SELECTION OF STRADDLED MOTOR CONCEPT FOR PUMP DESIGNS

CNLM - 5889

FIG 70

BEARING MATERIALS SCREENING TESTS



A. SPECIMEN ASSEMBLED IN RIG

FIG 71

CNLM - 5889

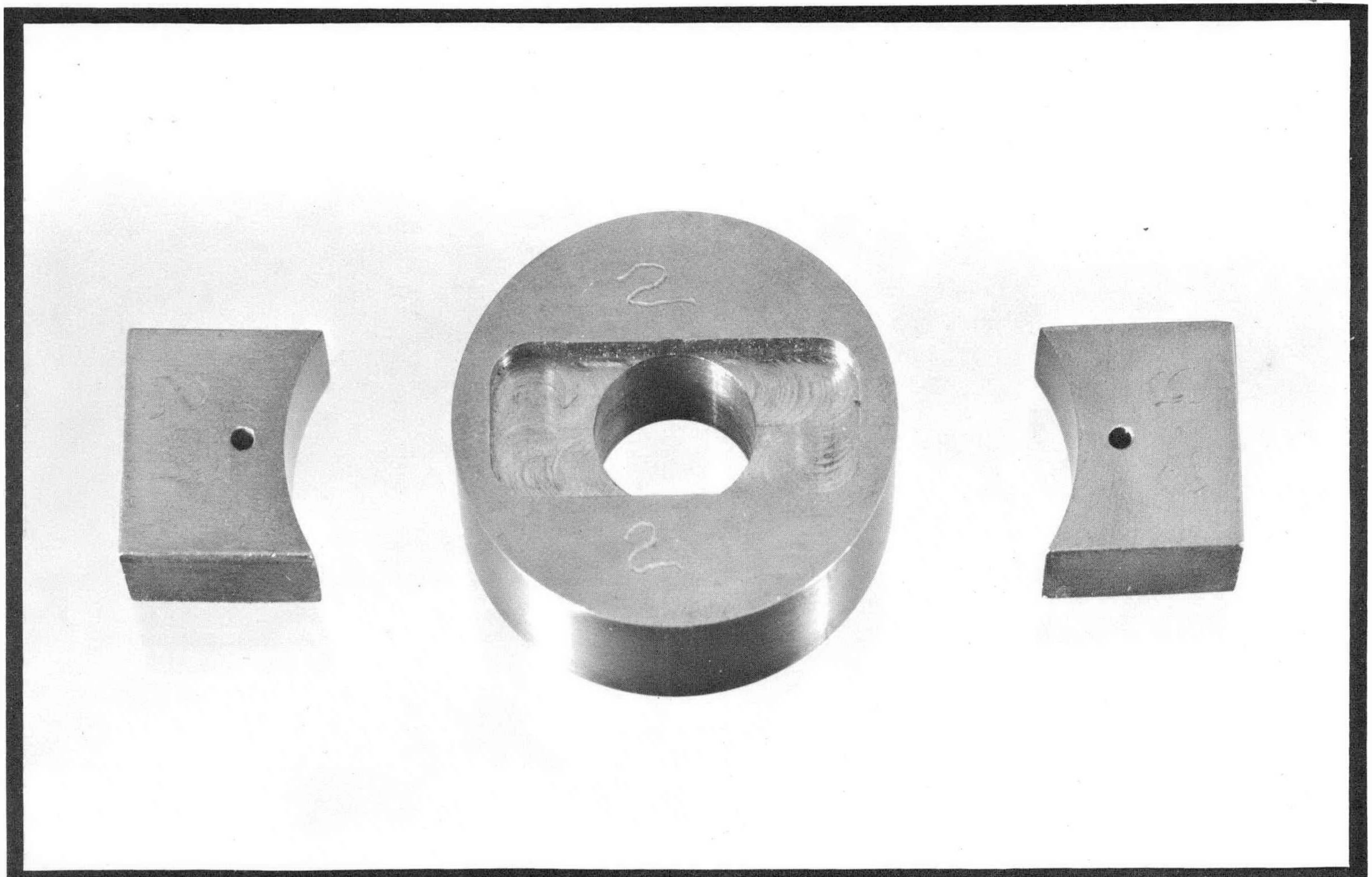
V-65

00122

BEARING MATERIALS SCREENING TESTS

UNCLASSIFIED

(CONTINUED)



B. TYPICAL TEST SPECIMENS

CNLM - 5889

FIG 71

CERAMIC TO REFRACTORY METAL BASE SEAL JOINT

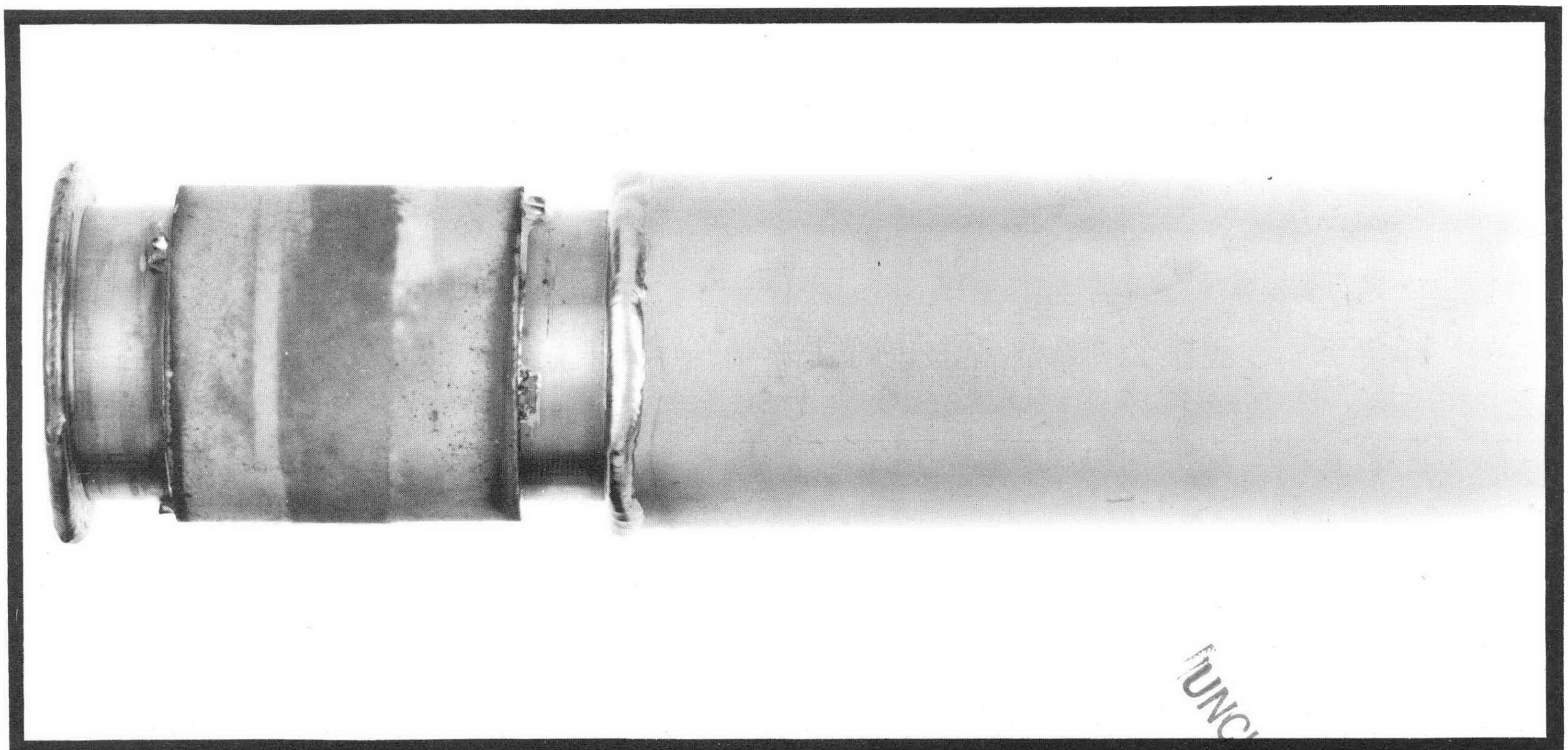


FIG 72

CNLM - 5889

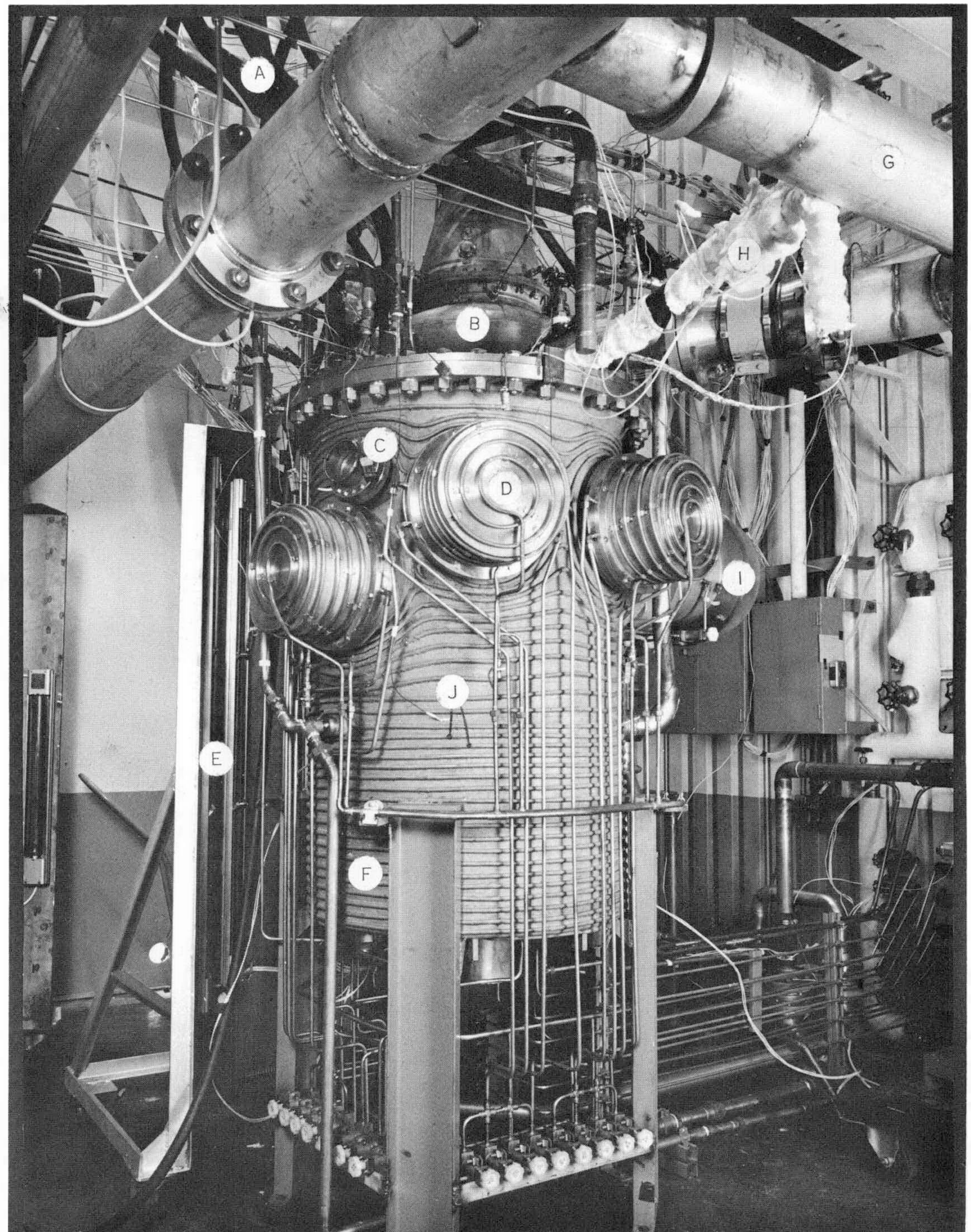
[redacted]

FIG 73

MATERIALS SCREENING TEST FOR CAVITATION PUMP RESISTANCE

LEGEND

A - TEST UNIT AIR COOLING LINES
B - TURBINE DRIVE
C - VIEWING POINT
D - GLOVE PORTS
E - CHAMBER BAKE-OUT HEATERS
F - TEST CHAMBER
G - TURBINE INLET AIR INLET PIPING
H - SWEEP OUT GAS
I - SPECIMEN PASS-THROUGH PORT



A. ROTATING DISK LIQUID METAL TEST RIG

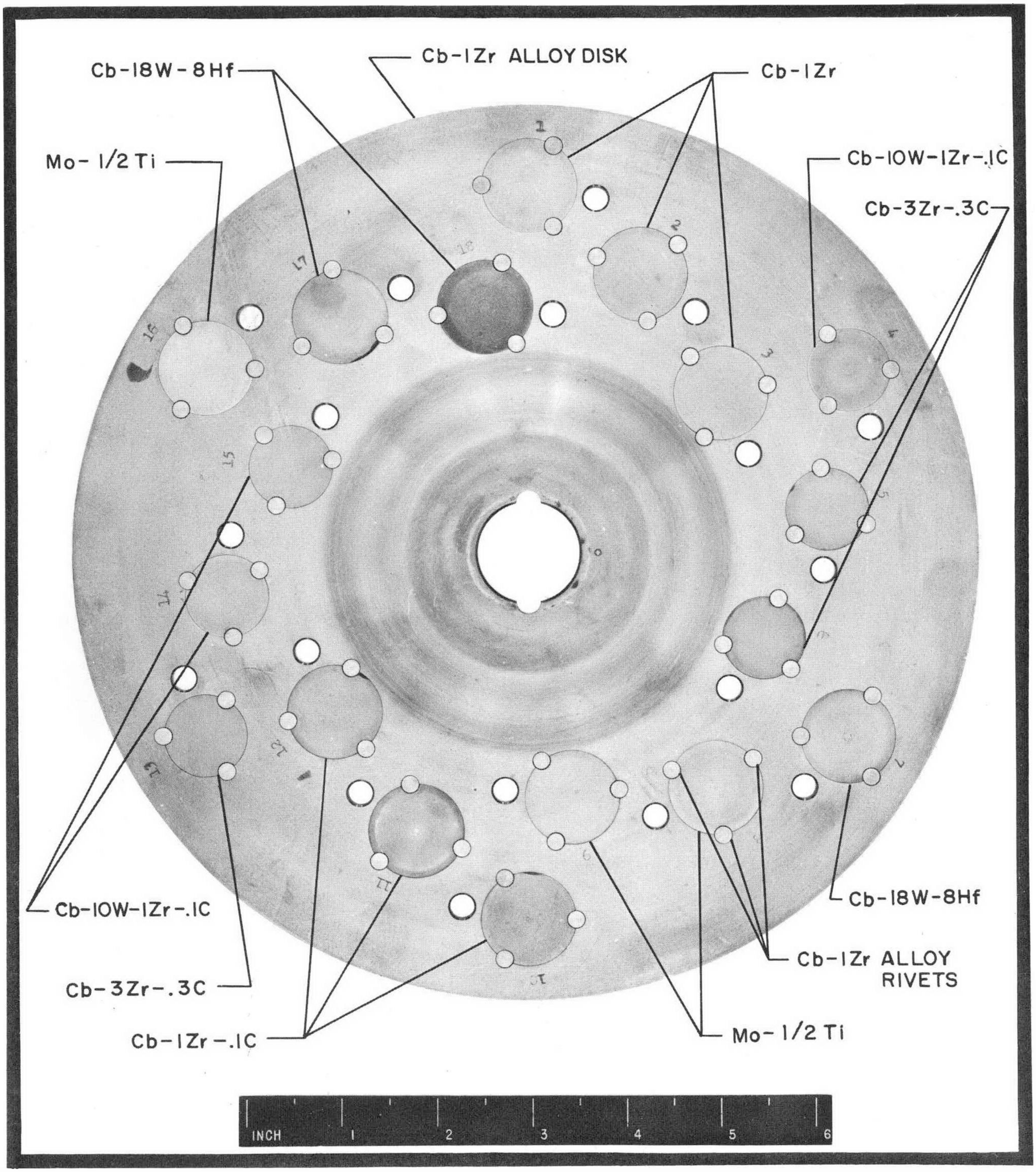
[redacted]

FIG 73

MATERIALS SCREENING TEST FOR CAVITATION PUMP RESISTANCE

(CONTINUED)

UNCLASSIFIED



B. ROTATING DISK ASSEMBLY

00126

MOTOR DEVELOPMENT

PRELIMINARY DESIGN STUDIES COMPLETED UNDER SUBCONTRACT

SELECTED

* POWER LEVEL
* MAXIMUM TORQUE
* INDUCTION TYPE MOTOR
* OPERATING FREQUENCY AND VOLTAGE
* SIZE AND WEIGHT
* POWER FACTOR AND EFFICIENCY

UNCLASSIFIED

ACHIEVED

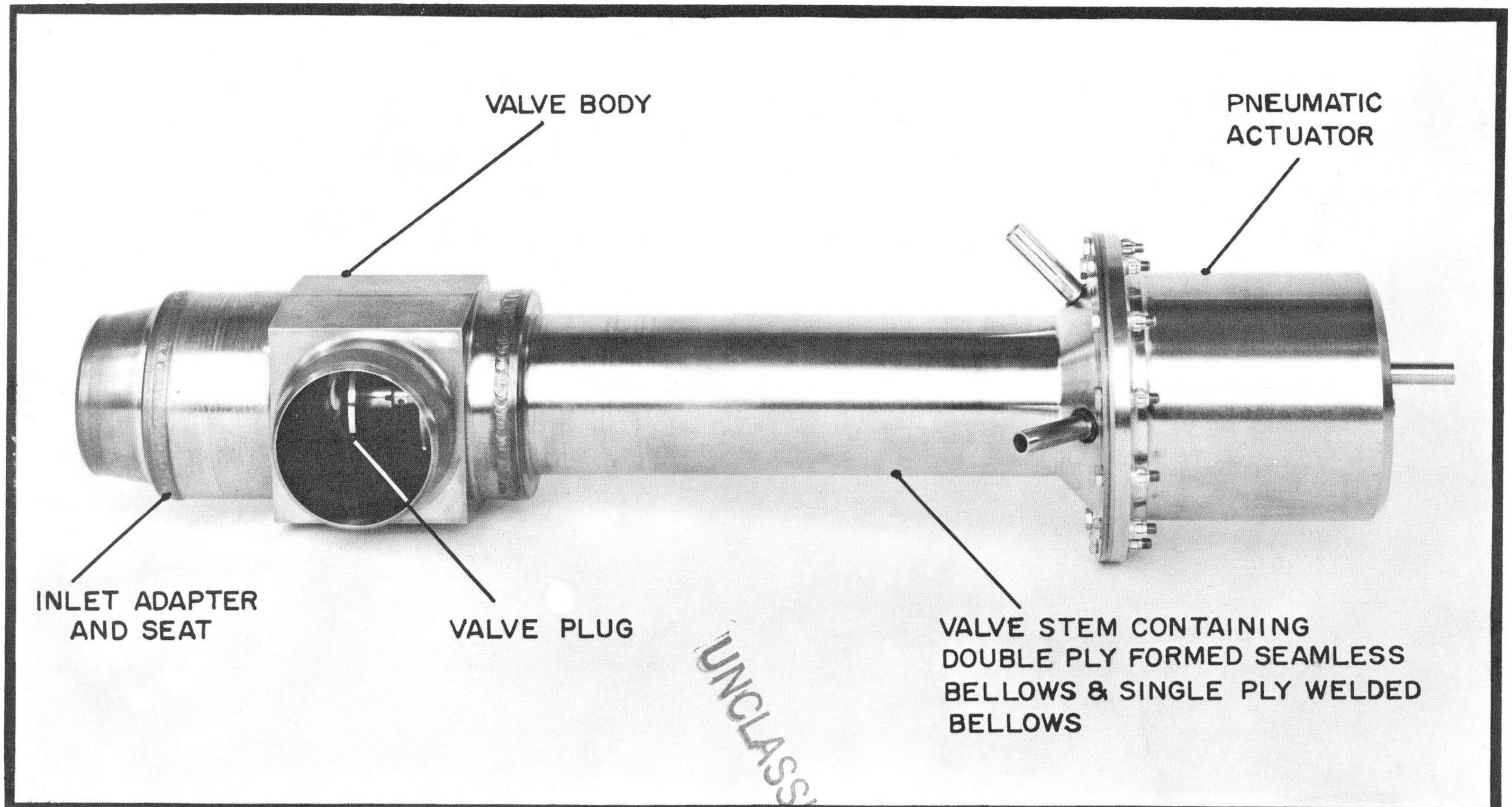
* 25 THERMAL CYCLES (800-1200F) OF BeO TO Cb ALLOY JOINT WITHOUT LEAKAGE
* FABRICATION OF FULL SCALE BeO STATOR LINER

00127

CNLM - 5889

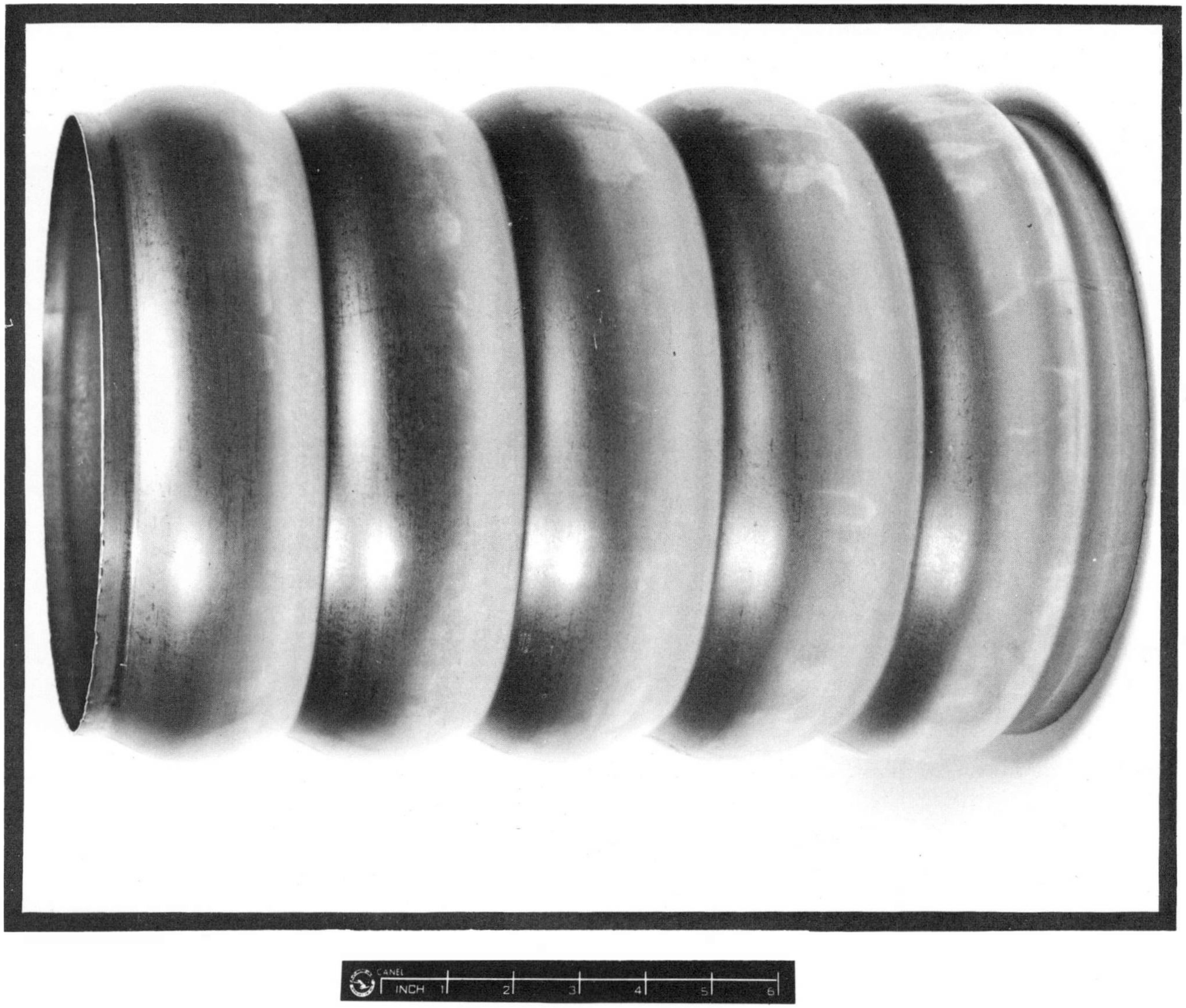
FIG 74

LCRE ISOLATION VALVE



CNLM - 5889
FIG 75

ACCUMULATOR SHELL



CNLM - 5889
FIG 76

D. Instrumentation and Controls

1. Description

The control and instrumentation system for SNAP-50/SPUR is designed to provide control of the powerplant during start-up, design power, and off-design operation, insuring accurate, stable operation at all operating levels (Fig. 77). The basic control functions which must be achieved are:

a. Initiate and control powerplant start-up.

b. Maintain proper temperature levels during start-up, off-design, and design operation.

c. Regulate turboalternator frequency.

d. Regulate turboalternator output voltage.

e. Prevent powerplant failure during transients caused by load changes or system abnormalities.

Safety functions will be incorporated into the design to prevent powerplant operation under unsafe conditions.

Direct reactor control is on neutron flux during start-up and on reactor outlet temperature during power operation, each mode being a closed loop control. After powerplant idle is reached, the control system transfers the pump and control system loads from the auxiliary power supply to the powerplant after which loads are applied to the generator.

The electrical output of the powerplant is controlled by correcting voltage and frequency error in closed loop systems. Voltage error is brought to zero by varying the generator field excitation and frequency error by varying the generator speed. Small speed errors are corrected by adding or removing portions of a parasitic load in parallel with the main load. Large speed errors are corrected by throttling potassium flow and by adjusting the reactor outlet temperature demand.

2. Development Status

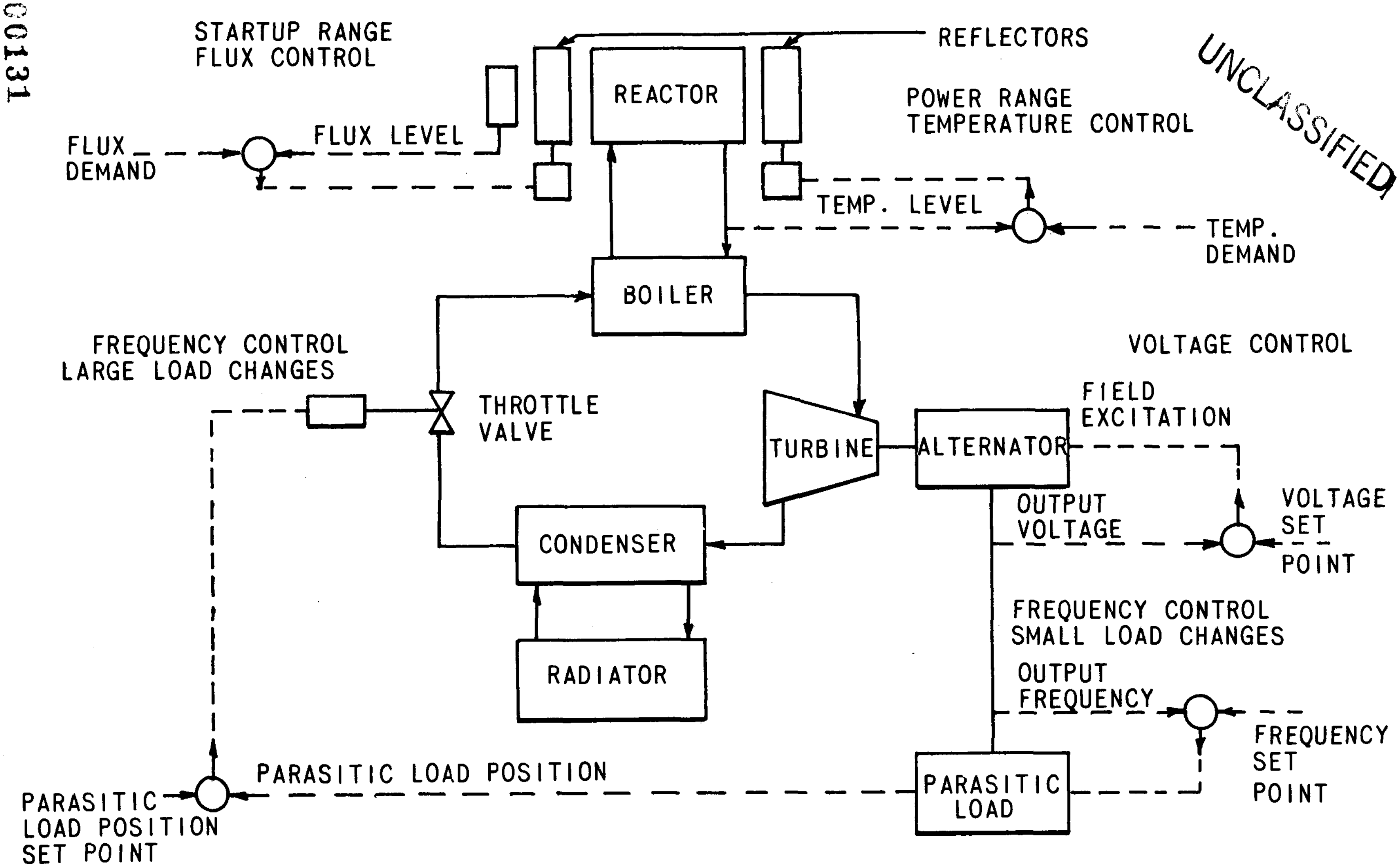
a. Sensors

The environmental capabilities of the various process sensors which have been or are now being developed are shown in Fig. 78 .. Proven long-term capabilities is denoted by an asterisk.

A summary of the test program on thermocouples is shown in Fig. 79 Also included in this table is a summary of tests now in the process of buildup which are required to evaluate performance at higher temperatures, as dictated by the fuel development program and in anticipation of SNAP-50/SPUR requirements. Testing is scheduled in both inert gas and liquid metal environments and is intended to establish confidence in new junction designs, material combinations, and fabrication techniques.

Fig. 80 summarizes the test experience with liquid metal level probes, pressure transmitters and flowmeters installed in several long-term liquid metal system tests. The pressure transmitters used are of all-welded, stainless steel construction. A diaphragm is used to separate the process fluid from the measuring fluid (NaK). The liquid metal flowmeters are of the permanent magnet type.

SNAP-50/SPUR CONTROL SYSTEM



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C0131

V-74

CNLM - 5889

FIG 77

PROCESS SENSOR CAPABILITY

MEASUREMENT	SENSOR	MAXIMUM ENVIRONMENT TEMPERATURE, F: AIR	INERT GAS & VACUUM	L.M.	LITHIUM
TEMPERATURE	CLAD THERMOCOUPLES				
	a) 316 SS CLAD C/A	1500*	2000*	2000 1800*	1000 800*
	b) Cb-I Zr CLAD C/A	--	2350 2000*	2200	2200
	c) Cb-I Zr CLAD W-Re	--	2600 2000*	2600 2000*	2400 1000*
LIQUID LEVEL	J & I PROBES				
	a) 316 SS CLAD	--	--	2000 1800*	1000 800*
	b) Cb-I Zr CLAD	--	--	2600 2000*	2400 1000*

MEASUREMENT	SENSOR	MAXIMUM DIAPHRAGM TEMPERATURE, F: AIR, INERT GAS, VACUUM, L.M.	LITHIUM
PRESSURE	TRANSMITTERS, DIAPHRAGM TYPE, 316 SS	1000 600*	800 600*

MEASUREMENT	SENSOR	MAXIMUM MAGNET TEMPERATURE, F: AIR, INERT GAS
FLOW	PERMANENT MAGNETS	900

MEASUREMENT	SENSOR	MAXIMUM ENVIRONMENT TEMPERATURE, F: AIR, INERT GAS
RADIATION	FISSION AND IONIZATION CHAMBERS	700 400*

*PROVEN LONG TERM PERFORMANCE

THERMOCOUPLE DEVELOPMENT - Cb-1 Zr CLAD
THERMOCOUPLE TESTS

C0133

Completed

Type	Qty.	Temperature - F	Time - Hr.	Environment	Post Test Accuracy	Percent Surviving
Chromel/Alumel	20	1950	10,000	Inert Gas	±1%	100
W-5 Re/W-26 Re	16	1950	10,000	Inert Gas	±1%	100
Mo/W-26 Re	7	1950	10,000	Inert Gas	+2%	100

In Progress

Type	Qty.	Temperature - F	Scheduled Time - Hr.	Time to Date - Hr.	Environment	Accuracy	Percent Surviving
W-3 Re/W-25 Re	9	1950	10,000	7075	Inert Gas	±1%	100
Mo/Cb	10	1950	10,000	4600	Inert Gas	<1% drift	100
W-5 Re/W-26 Re	20	2300	10,000	1400	Inert Gas	±1%	100

In Process of Buildup

Type	Qty.	Temperature - F	Scheduled Time - Hr.	Environment
Chromel/Alumel	6	2200	10,000	NaK
W-5 Re/W-26 Re	6	2200	10,000	NaK
W-5 Re/W-26 Re	30	2200-2400	1000-10,000	Lithium
W-5 Re/W-26 Re	70	2200-2500	10,000	Inert Gas
W-3 Re/W-26 Re	40	2300-2500	10,000	Inert Gas
Chromel/Alumel	50	2300-2400	10,000	Inert Gas
Mo/Cb	40	2300-2500	10,000	Inert Gas

UNCLASSIFIED

FIG 79

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3

FIG 80

PROCESS SENSOR TESTS

Liquid Metal Level Prob

UNCLASSIFIED

Completed

Type Clad	Quantity	Temperature - F	Time - Hr.	Environment
"J" Cb	2	1000	6600	Lithium
"J" Cb	6	1000	10, 000	Lithium

In Progress

Type Clad	Quantity	Temperature - F	Scheduled Time - Hr.	Time to Date - Hr.	Environment
"J" Cb	2	900	10, 000	9100	Lithium
"J" 316 SS	1	1000	10, 000	9100	Lithium
"I" 316 SS	2	1000	10, 000	9100	Lithium
"J" 316 SS	2	800	10, 000	9100	NaK

Pressure Transmitters

In Progress

Quantity	Temperature - F	Scheduled Time - Hr.	Time to Date - Hr.	Environment
2	600	10, 000	9100	Lithium
7	600	10, 000	9100	NaK

Flowmeters

Completed

Quantity	Pole Face Temperature - F	Time - Hr.
1	900	6600
1	900	10, 000
2	200	10, 000

In Progress

Quantity	Pole Face Temperature - F	Scheduled Time - Hr.	Time to Date - Hr.
2	200-300	10, 000	9100

00134

Fig. 81 summarizes the test experience with nuclear sensors and preamplifiers. Both Reuter-Stokes and General Electric ion chambers and fission counters have been evaluated, the latter being intended for backup. Both manufacturers rated their units for 700F. The Reuter-Stokes units are of coaxial design with integrally-attached, metal-clad, ceramic-insulated coaxial cables. Construction of the General Electric units is similar, although they require separate, metal clad cables extending from the sensor body into a relatively mild environment where connection to conventional cables could be made. Initial development work on fission counter preamplifiers included the use of General Electric's ceramic tubes and RCA's Nuvistor tubes. Recent work has incorporated only the ceramic type tube because of its higher temperature capabilities. In general, operation at 250F, both in and out-of-pile, has been completely satisfactory. Although short-term tests have been run up to 475F, capacitor failures begin to occur in the range of 300F to 350F, limiting the long-term, unattended use to several hundred hours in this temperature range.

b. Control System

SNAP-50/SPUR control system work to date has dealt with preliminary studies of the powerplant system. Powerplant control modes have been evaluated and a casualty analysis has been performed. Shim rates have been established as a function of the reactor control system stability margin. Preliminary studies have been made on the effect of position feedback of the control elements, as opposed to series compensation techniques on the reactor outlet temperature loop. Position feedback poses a problem of scheduling the amount of fuel burnup so that the control system can compensate for the new positions of the control elements. Preliminary control gains have been determined for the power conversion loop. Initial start-up studies have been made. A scram rate of $1.00 per second was established for a sensor loop having a response time of 50 m sec. A scram trip point on a start-up accident was also established. The scram is based on a power level trip supplemented by a period trip. With no core coolant or with zero flow rate, a period trip or a low level power trip is required for a start-up accident.

Further studies during this and the next fiscal year will include:

1) Further studies of the powerplant control system and techniques of adding and deleting loads.

2) Parametric studies of various lags and deadtimes in the sensor instrumentation relating to powerplant control.

3) A detailed study of the start-up procedure.

4) Control studies on the first nuclear powerplant test to determine additional requirements peculiar to this test.

The results of these studies will provide sufficient information for the preliminary design of the powerplant control. Components will be procured and their environmental capabilities determined through a series of short-term tests. This will be followed by design and fabrication of a control system mock-up to undergo testing with an analog reactor system.

c. Electronics

Post-test analysis of irradiated fission counter preamplifiers will be completed.

Electronic circuit elements will continue to be tested at high temperatures as new types become available. Since short-term temperature effects and damage caused by nuclear radiation are somewhat similar, obviously unsuitable component types will be determined without expensive inpile testing.

00135

[redacted]

NUCLEAR SENSOR AND PREAMPLIFIER DEVELOPMENT

Completed

Type	Manufacturer	Qty.	Temperature - F	Time - Hr.	Environment
Ion Chamber	Reuter-Stokes	1	400	2200	Inpile $1x10^{11}$ nv(th)
Ion Chamber	Reuter-Stokes	1	400	10,000	Inpile $3x10^{9}$ nv(th)
Ion Chamber	General Electric	2	400	10,000	Inpile $1x10^{9}$ nv(th)
Fission Counter	Reuter-Stokes	1	400	2300	Inpile $2x10^{9}$ nv(th)
Fission Counter	Reuter-Stokes	1	400	10,000	Inpile $3x10^{9}$ nv(th)
Fission Counter	General Electric	1	400	2300	Inpile $2x10^{9}$ nv(th)
Fission Counter	General Electric	1	400	3000	Inpile $2x10^{9}$ nv(th)

UNCLASSIFIED

In Progress

Type	Manufacturer	Qty.	Temperature - F	Scheduled Time - Hr.	Time to Date - Hr.	Environment
Ion Chamber	Reuter-Stokes	3	400	10,000	9650	out of pile
Ion Chamber	General Electric	2	400	10,000	7500	out of pile
Fission Counter	Reuter-Stokes	3	400	10,000	1900	out of pile

Completed

Type	Manufacturer	Qty.	Temperature - F	Time - Hr.	Environment	Comments
Preamp	CANEL	2	250	1400	Inpile $3x10^{7}$ nv(f)	--
Preamp	CANEL	2	350	3500	Out of Pile	Failed

In Progress

Type	Manufacturer	Qty.	Temperature - F	Scheduled Time - Hr.	Time to Date - Hr.	Environment	
Preamp	CANEL	5	250	10,000	6000	Out of Pile	No Failures
Preamp	CANEL	3	350	10,000	600	Out of Pile)	Capacitor Failures
Preamp	CANEL	1	350	10,000	2400	Out of Pile)	During Test, Replace
Preamp	CANEL	1	350	10,000	1800	Out of Pile)	and Continue Test
Preamp	CANEL	5	to 475	--	--	Out of Pile	Short Term Tests

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FIG 81

00136

Commercially available control circuits for shim motors are being modified to meet our requirements. These modified devices are being used in analog studies and will be used as parts of various shim motor component tests. This concept will be expanded to include as much of the electronic system as possible as component tests are extended. Thus, actual running experience will be gained with these circuits prior to system ground tests.

Survey studies of space electronic systems and devices will be continued. It is our intent to take full advantage of prior work throughout this entire program. The major effort will lie in system and component testing.

E. Materials

1. Structural Alloys

a. Basic Strength Evaluation of Cb-1 Zr

Laboratory strength studies in lithium on the Cb-1 Zr alloy for times to 10,000 hours are essentially completed. Over 290 uniaxial and biaxial creep-rupture tests were made on approximately 30 heats of material. The test data is tabulated in Refs. 20, 21, 22, 43, 44, and 45. Minimum 10,000-hour rupture strength values derived from tests on various fabricated forms of material are listed below. The strength of welded material was determined to be equal to that of material in the unwelded condition.

Cb-1 Zr Alloy, Annealed 2200F

Lithium Temp, F	Minimum 10,000-Hour Rupture Strength: Forgings & Extrusions (49 tests)	Bar, Plate, Rod, Pipe (47 tests)	Sheet and Tubing (83 tests)
1600	11,000 psi	8200 psi	7800 psi
1800	5800	4400	3400
2000	3000	2300	1400
2200	1500	1300	700

b. Engineering Type Tests of Cb-1 Zr

In addition to the laboratory tests, engineering tests were completed on Cb-1 Zr alloy 10-inch and 15-inch diameter pressure vessels. The vessels which contained lithium were pressurized with helium and tested in static inert gas. The 10-inch diameter vessel, which completed 4370 hours at 2000F under an effective stress of 2135 psi and completed 5630 hours at 1900F under an effective stress of 2475 psi, exhibited a creep growth of three percent. Fig. 29 shows the 10-inch vessel after test. The 15-inch diameter vessel which recently completed 10,000 hours at 2000F under an effective stress of 1400 psi is currently undergoing disassembly for post-test evaluation. Fig. 30 shows the 15-inch diameter vessel after test. Test details are presented in Ref. 46 and 47.

c. Further Strength Evaluation of Cb-1 Zr and Cb-1 Zr-0.1 C

Fabrication of two Cb-1 Zr and three Cb-1 Zr-0.1 C ten-inch pressure vessels has begun. Uniaxial creep-rupture tests in lithium were started on improved columbium base alloys. The principal effort is on the Cb-1 Zr-0.1 C alloy. These tests are intended to provide preliminary design data for use in the pressure vessel program. Test data are tabulated in Refs. 22 and 23 and summarized in Figs. 82 and 83.

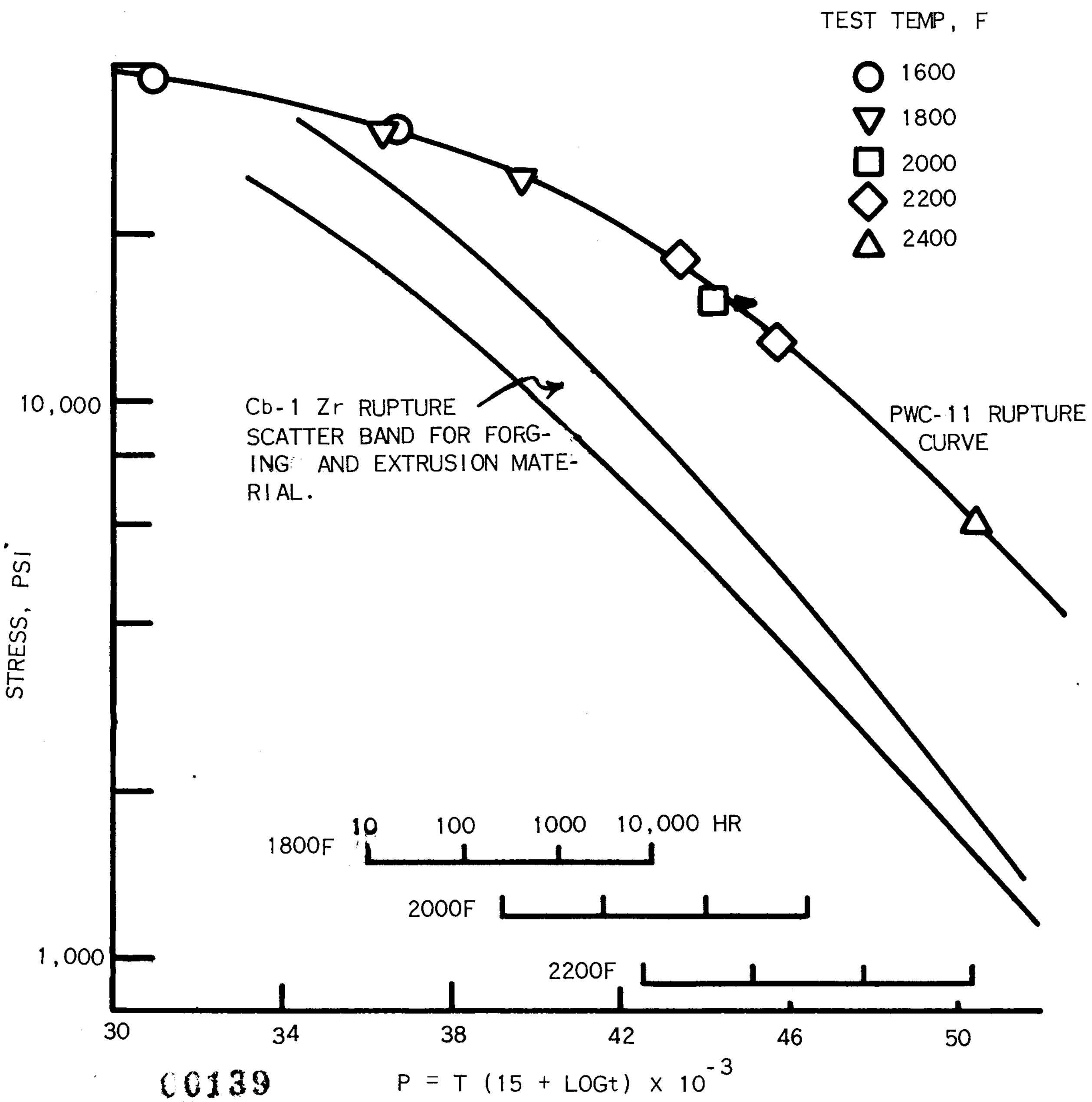
SNAP-50/SPUR Structural Alloy Strength Tests

Alloy	Temp. Range F	Stress Range psi	Max Time Hrs	Total Tests	Est 10^4 Hr 2000F Rupture Stress	Est 10^4 Hr 2000F 1% Creep Stress
Cb-1 Zr	1600F 2400F	27,500 750	10,000+	293	3000	2000
Cb-1 Zr-.09 C	1600F 2400F	37,500 3000	2,702	32	11,500	6000
D-43 (Cb-10 W-1 Zr-0.1 C)	2000F 2400F	20,000 1800	2,168	8		5800

FIG 82

STRESS VERSUS RUPTURE TIME

STRUCTURAL ALLOYS



C0139

STRUCTURAL ALLOY 1% CREEP STRENGTH
2000F

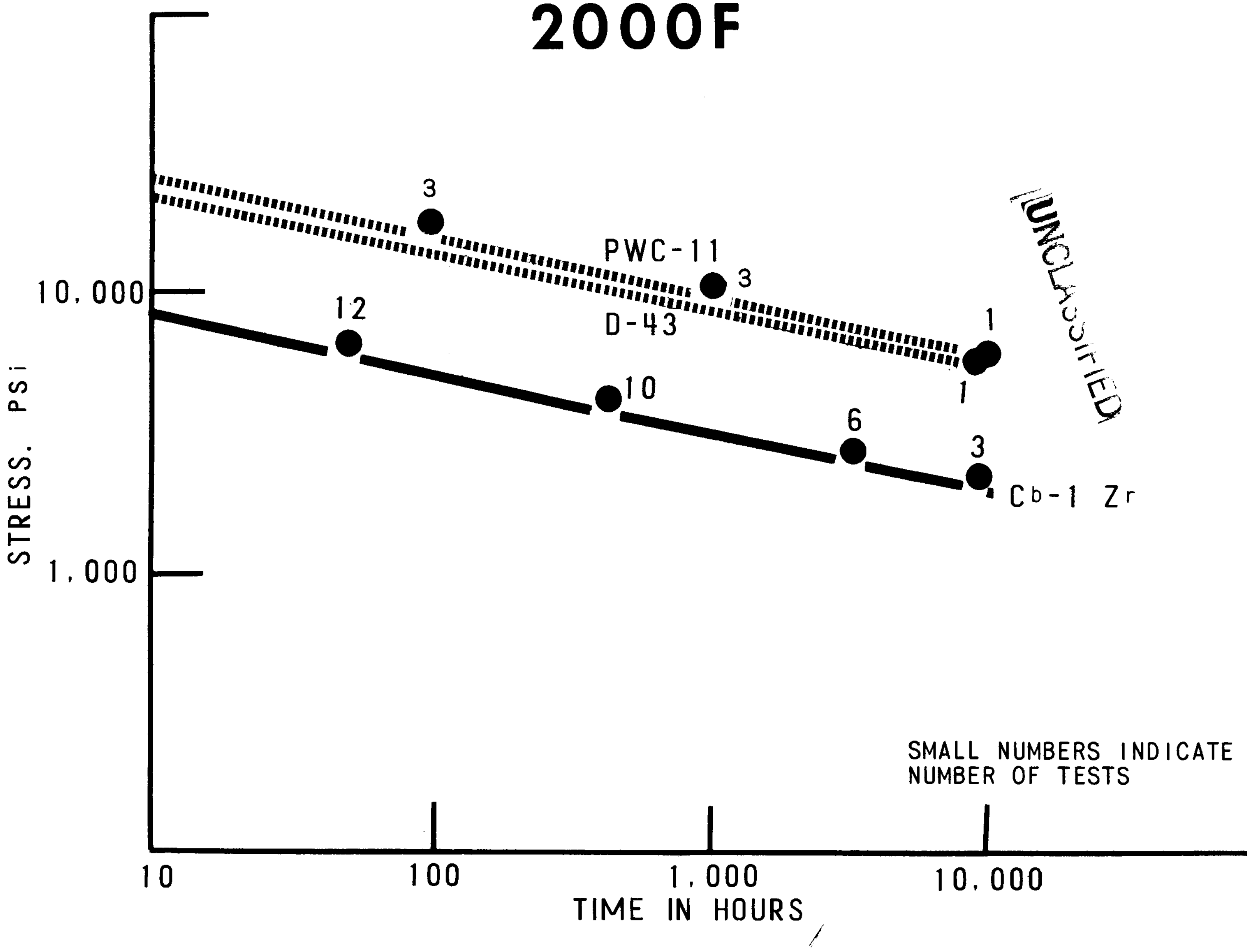


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FIG 83

00140

d. Fabrication of Cb-1 Zr Alloys

Cb-1 Zr alloy was fabricated in all forms of mill product (Fig. 16) through development efforts under the ANP and LCRE programs. Laboratory, pilot scale and limited production experience indicate that no serious problem should be expected in the production of carbon-modified columbium zirconium alloy in required forms of mill products. Cb-1 Zr alloy and carbon-modified columbium zirconium alloys are capable of producing ductile welds in thin or thick wall sections (Refs. 22 and 23).

e. Irradiation Testing of Cb Alloys

Irradiation effects studies of Cb-1 Zr alloy have consisted of limited inpile biaxial rupture tests performed by Oak Ridge National Laboratory (Ref. 48) and post-irradiation bend ductility tests by Oak Ridge National Laboratory and CANEL (Refs. 22, 44, 49 and 50). Results of these tests have not indicated an irradiation stability problem. The effect or irradiation on carbon-modified columbium zirconium alloys requires investigation to evaluate the stability of the dispersion-strengthening carbide system.

Post-irradiation bend ductility studies of D-43 alloy (Cb-10 W-1 Zr-0.1 C) (Ref. 49) indicate that irradiation effects measured by room temperature yield strength and by ductility at -110F are minor and are removed by an 1800F anneal. Post-irradiation strength tests and electron microstructure studies are planned for Cb-1 Zr-0.1 C alloy specimens now in preparation (Ref. 51). These investigations will provide a further understanding of irradiation effects on dispersion-strengthened alloys and also determine whether or not there is a serious material instability which would require immediate project attention at specific design conditions.

f. Future Strength Testing of Columbium and Tantalum Alloys

In FY 1965 and FY 1966, uniaxial creep-rupture tests will be continued on the Cb-1 Zr-0.1 C alloy in lithium and in a vacuum of 10^{-8} torr. Also, preliminary strength curves will be established for Cb-10 W-1 Zr-0.1 C, Cb-18 W-8 Hf, Cb-27 Ta-10 W-1 Zr, and Ta-9.5 W-2.5 Hf-0.01 C alloys. In addition, engineering tests of two Cb-1 Zr and three Cb-1 Zr-0.1 C ten-inch diameter pressure vessels will be completed or will be in progress. All five tests will be conducted at 2000F for 3000 hours with the outer surfaces of the specimen subjected to a vacuum of 10^{-8} torr. The vessels will be filled with lithium and pressurized to produce a stress expected to cause one percent or five percent diametral creep growth in the cylinder section. Details of this program are presented in Ref. 52. Production quantities of Cb-1 Zr-0.1 C alloy in various mill product forms will be fabricated. Post-irradiation evaluation tests of carbon-modified Cb-1 Zr alloy will be completed.

2. Coolants and Working Fluids

a. Compatibility Studies

1) Lithium with Cb-1 Zr Alloys

Compatibility of Cb-1 Zr alloy with lithium has been evaluated over a wide range of operating conditions through tests of forced circulation corrosion loops, pumps, pressure vessels and biaxial-stressed tube specimens. Over 300 tests were conducted to evaluate the effect of the following operating parameters: a) maximum temperature 2000F to 2400F, b) thermal gradient 400F and 1000F, c) fluid velocity up to 50 fps, d) heat flux 4.0 to 50 x 10^4 Btu hr^{-1} ft^{-2} and e) applied stress up to that required for short time rupture. More than 30 heats of Cb-1 Zr alloy have been tested in obtaining the experience summarized in Fig. 21. Detailed results are given in Ref. 22, 44, 50, 53, 54, 55, 56, 57 and 58. The following conclusions can be made:

a) Cb-1 Zr alloy is capable of containing 2000F lithium for 10,000 hours with no significant corrosion or mass transfer when oxygen contamination in alloy and environment is minimized.

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b) Cb-1 Zr alloy compatibility with lithium is not significantly altered by wide variations in thermal gradient, fluid velocity, or applied stress.

c) Oxygen contamination of Cb-1 Zr alloy in excess of two oxygen atoms per zirconium atom results in lithium penetration corrosion.

d) Inert gas atmosphere must be of ultra-high purity for oxidation protection of long duration tests.

e) Vacuum systems at a nominal 10^{-8} torr appear to afford reliable oxidation protection necessary to prevent corrosion.

Compatibility studies (Table 1) are in progress to evaluate the effect of added carbon (0.06 to 0.1 percent) to Cb-1 Zr alloy. The objectives of this study are to determine if the oxygen tolerance of the alloy is seriously reduced by the presence of carbon and if the carbon content of the alloy is essentially stable when subjected to temperature and concentration gradients in a lithium system. Tests are being conducted using 1) laboratory capsules, 2) a forced circulation loop constructed of Cb-1 Zr alloy with facilities for exposure of specimens in 2000F to 2200F flowing lithium and 3) forced circulation corrosion loops with heater regions of carbon-modified Cb-1 Zr alloy.

Table 1

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Status of Compatibility of Carbon-Modified Cb-1 Zr Alloys in Lithium

Test Type	Total Tested	Temp. Range, F	Time hrs.	Results
Oxygenated specimens (1150 ppm O_2) in capsules	2	2000	100	No attack when in annealed condition.
Tube specimens exposed in Cb-1 Zr forced circulation loop	3	2000	500	No detectable change in chemistry or strength
	3	2200	1000	Inconclusive, uncertain thermal history
	3	2200	1000	Slight carbon and zirconium depletion, no significant change of rupture strength
Forced circulation loop of Cb-1 Zr-0.06 C (Hot Leg) and Cb-1 Zr (Remainder)	1	2200/1900	2175 (leaked)	Inconclusive, contaminated by external gas environment

Results to date:

The oxygen threshold for lithium corrosion of annealed carbon-modified Cb-1 Zr alloy is above 1150 ppm. Thus carbon addition does not cause a catastrophic reduction in the oxygen tolerence determined for Cb-1 Zr alloy.

Surface depletion of carbon and zirconium does not result in a significant change in rupture strength.

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2) Lithium with Tantalum and Tungsten Alloys

Testing of tantalum-base and tungsten-base alloys in lithium has been initiated. Capsule corrosion tests have established that hafnium in T-111 alloy (Ta-8 W-2.0 Hf) provides a tolerance for oxygen, thereby preventing lithium attack (Ref. 59). This is the same effect achieved by zirconium in columbium-zirconium alloys. The ductile behavior of T-111 alloy in biaxial rupture tests of up to 2200 hours in 2400F lithium indicate that compatibility of the material in the 2400F annealed condition is not stress-sensitive. Tube specimens of T-111 alloy and W-25 Re alloy have shown no gross effects when exposed 1000 hours in the 2200F lithium region of a forced circulation Cb-1 Zr alloy loop, but detailed examinations are not complete.

3) Potassium with Cb-1 Zr Alloys

Compatibility of Cb-base alloys with boiling potassium is being evaluated in natural circulation loops operated in a vacuum environment for 3000 hours. Tests have been completed on two Cb-1 Zr alloy loops and one Cb-5 Mo-3 Zr-3 Ti alloy loop. Results to date showed negligible corrosion effects for both alloys based on weight change data, metallography and chemical analyses, Fig. 84 (Ref. 23 and 59). Initial information concerning stability of carbon-modified Cb-1 Zr alloy will be obtained from examination of the GFL-3 test. In these tests, a vacuum environment of a nominal 10^{-8} torr proved effective in preventing oxygen contamination during test.

Additional results of natural circulation and forced circulation loop tests at Oak Ridge National Laboratory verify compatibility of Cb-1 Zr alloy, Fig. 85 (Refs. 49, 60 and 61). Other improved strength alloys such as D-43 and T-111 are being evaluated.

4) NaK and 316 SS and Cb-1 Zr Alloy

The SNAP-50/SPUR condenser cooling circuit will use NaK in a Cb-1 Zr/type 316 stainless steel system. We are now running representative forced circulation loop tests to determine the extent of mass transfer of stainless steel and the migration of carbon, oxygen, and nitrogen in such a system. Results of one 3000-hour test to evaluate reaction rates in 1250F NaK showed that the carbide nitride reaction zone penetration of Cb-1 Zr alloy was negligible and was not significantly time-dependent. Ductility of the Cb-1 Zr alloy tubing was not reduced by carbon and nitrogen contamination.

An engineering corrosion loop which matches materials, area ratios, and fluid velocities of the condenser coolant system, has been constructed. It is now in start-up at ORNL and will be used to verify materials compatibility. Improved radiator materials such as high-strength alloys will be evaluated when they become available.

b. Physical and Chemical Properties

Measurements of the thermophysical properties of lithium, eutectic NaK (78K), and potassium liquid required for powerplant design have been completed (Refs. 62, 63, 64, 65, 66, 67 and 68). Measurements of potassium vapor properties are incomplete for viscosity, thermal conductivity, and construction of an improved Mollier diagram, but existing government-sponsored research is expected to provide the required data by FY 1966 to validate estimates used in design.

Purification methods have been established for reducing objectionable contaminants in lithium, NaK and potassium to tolerable levels (Ref. 69). However, additional information is required for design of hot trap purifiers.

Cb-ALLOY COMPATIBILITY WITH BOILING POTASSIUM

NOMINAL TEST CONDITIONS

TEMP. F	2000 MAX-1550 MIN
FLOW. LB/HR	10-12 LB/HR
TIME. HRS	3000
CHAMBER PRES.. TORR	$0.7-2.5 \times 10^{-8}$

RESULTS

LOOP	ALLOY	OD OXYGEN, PPM	WEIGHT CHANGE, Mg/Cm^2	1950 F TENSILE STRENGTH, PSI
GFL 1	Cb-1 Zr	160	0.8	
	Cb-5 Mo 3Zr-3 Ti INSERTS	(65-300)*	<0.04	32,500 (33,300)*
GFL 2	Cb-5 Mo 3Zr-3 Ti	415 (310-460)*	1.7	21,300 (23,300)*
GFL 3	Cb-1 Zr Cb-1 Zr-0.1 C INSERTS	BEING EXAMINED		

*CONTROL VALUES

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FIG 84

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TEST SUMMARY
Cb-ALLOY-POTASSIUM COMPATIBILITY

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ALLOY	TYPE	TEST SITE	MAX. TEMP. F	TIME. HRS	COMPATIBILITY EVALUATION
	TEST EXPERIENCE				
Cb-1 Zr	CONV. LOOP	CANEL	2000	6000	GOOD
	PUMP LOOP	ORNL	2000	3000	GOOD
	CONV. LOOP	ORNL	2000	2800	GOOD
	PUMP LOOP	GE	1700	5500 (ABOVE 800F)	ON TEST
PWC-533 (Cb-5 Mo-3 Zr-3 Ti)	CONV. LOOP	CANEL	2050	3000	GOOD
D-43. (Cb-10 W-1 Zr-0.1 C)	CONV. LOOP	GE	2000	2000	GOOD
	CONV. LOOP	ORNL	2200	3000	GOOD (PRELIM. EXAM)
	REFLUX CAPS	ORNL	2200	5000	GOOD

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FIG 85

c. Future Plans for FY 1965 and FY 1966

1) Lithium Compatibility:

The oxygen tolerance limit for carbon modified Cb-1 Zr alloys will be determined over a range of carbon levels and conditions for annealing will be optimized. The effect of temperature and concentration gradients on carbon migration will be investigated by thermal convection loop tests and by forced circulation corrosion loop tests as summarized in Table 2. In addition, it is expected to continue the evaluation of tantalum, tungsten and solution-strengthened columbium alloys through capsule and loop tests.

Table 2

Lithium Compatibility Loop Program

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Type	Loop Material: Hot Leg	Loop Material: Remainder	Number Planned	Temp. F Max/Min	Scheduled Time, hrs
Thermal Convection	Cb-1 Zr-0.06 C	Cb-1 Zr-0.1 C	1	2000/1500	1000
			1	2000/1500	3000
Forced	Cb-1 Zr 0.06 C	Cb-1 Zr	1	2200/1900	3000
	Cb-1 Zr-0.06 C	Cb-1 Zr-0.1 C	2	2200/1900	3000
			1	2200/1900	10,000
	Cb-1 Zr-0.1 C	Cb-1 Zr-0.1 C	2	2200/1900	3000
			1	2200/1900	10,000
	Ta-8 W-2 Hf	Cb-1 Zr	1	2200/1900	3000

2) Potassium Compatibility:

Loop tests will be directed toward evaluating structural and turbo-machinery materials in boiling potassium. These materials will include PWC-11, TZM (Mo-0.5 Ti-0.1 Zr), type 316 stainless steel and prospective bearing and seal materials.

3) NaK Compatibility:

Limited forced circulation loop testing is planned to evaluate the temperature dependence of mass transfer and migration in the Cb-1 Zr alloy--type 316 stainless steel--NaK system. Improved radiator materials such as high-strength titanium alloys will be evaluated when available.

4) Physical and Chemical Properties:

Estimates of thermophysical properties required for powerplant design will be verified from data of existing government-sponsored programs or by CANEL if necessary. In addition, the effect of operating parameters on purification of potassium will be determined in order to design improved "hot-trap" purifiers.

3. Special Component Materials

a. Bearing Materials

Laboratory tests are in progress to screen prospective bearing materials for compatibility with coolants and structural alloy, and for rubbing and wear behavior under boundary lubrication conditions representative of pump starts and stops. Compatibility tests are performed in tilting capsules at 1100F for up to 7000 hours.

Rubbing and wear tests are performed using a rotating disk and two static rub shoes immersed in 1000F lithium or in 700F NaK. Results, summarized in Table 3 and described in detail in Ref. 21, 22 and 45, indicate the following.

1) Most promising bearing materials for 600F to 1000F lithium service are as follows: (a) WC - 8 Mo-2 CbC and Kennametal K-601 cermets which exhibit excellent rubbing behavior, but complicate the design to accommodate thermal expansion mismatch, and (b) surface-carburized Cb-1 Zr alloy which essentially eliminates thermal expansion mismatch, but is less effective under rubbing conditions.

2) Most promising materials for 700F NaK service are cermets WC - 8 Mo-2 CbC, Kennametal K96, and plasma-sprayed WC - 6 Co and WC.

3) Improvements in the plasma-spraying procedure are required to assure reliable coating adhesion for lithium service, but this is probably not an insurmountable problem.

4) Engineering tests have been initiated to evaluate promising materials as full size bearings.

Table 3

PROMISING MATERIALS FOR BEARING AND SEAL APPLICATIONS

Based on Compatibility and Rubbing Test Results

1000F LITHIUM	600F NaK (78K)
WC-8 Mo-2 CbC	WC-8 Mo-2 CbC
Kennametal K-601 (WC-TaC-No Binder)	Kennametal K-96 (92 WC-6 Co-2 (TaCb) C)
Kennametal K-96 (92 WC-6 Co-2 (TaCb) C)	Kennametal K-94 (86 WC-12 Co-2 (TaCb) C)
Carboloy-78 (76 WC-8 Co-4 (TaCb) C-12 TiC)	Kennametal K-162B (64 TiC-25 Ni-5 Mo-6 (TaCb) C)
Carburized Cb-1 Zr	Carboloy-999 (97 WC-3 Co)
	Kennametal K-151A (70 TiC-20 Ni-10 (TaCb) C)
	Plasma-Sprayed WC
	Plasma-Sprayed WC-6 Co

b. Motor Materials

We have begun development of a ceramic-to-metal joint for lithium service to provide a stator bore seal in support of a subcontracted motor development program. Results of compatibility and containment tests on one-quarter size specimens (Ref. 22) are summarized as follows:

1) Containment of 1000F lithium by a BeO ceramic-to-Cb-1 Zr alloy assembly joined with Ti-8.5 Si braze alloy is probably feasible, based on successful 1000-hour, 1100F lithium containment by a test assembly previously subjected to 25 thermal cycles between 800F and 1200F.

2) Improvements in fabrication methods and nondestructive inspection procedures are required to increase reliability.

c. Future Plans for FY 1965 and FY 1966

1) Bearing Materials:

Screening tests in lithium, potassium and NaK will be completed. Material specifications and fabrication procedures will be established for control of quality of bearing materials selected from engineering test program.

2) Motor Materials:

Full-size BeO-Cb alloy joint assemblies will be produced and evaluated for lithium containment and thermal shock resistance. Materials and process specifications for fabrication of motor bore seals will be established.

4. Advanced Materials Studies

A program has been conducted since FY 1962 under funds from the Fuels and Reactor Materials Branch with the objective of extending materials technology for advanced reactor applications. The general areas of study, Fig. 86, provide general support but are not limited to the SNAP 50/SPUR powerplant concept.

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ADVANCED MATERIALS

DEVELOPMENT PROGRAMS

MATERIALS	TO IMPROVE OR EXTEND
FUELS:	
UC, UN MIXED CARBIDES	PHYSICAL PROPERTIES CLAD COMPATIBILITY IRRADIATION BEHAVIOR
ALLOYS:	
Cb ALLOYS Ti ALLOYS	AGING BEHAVIOR Cb-Zr-C PHASE DIAGRAM STRENGTH OF LOW DENSITY ALLOYS
COATINGS:	
Sn-Al, Ti-Si-Cr	OXIDATION PROTECTION
COOLANTS:	
Li, K, Cs	CORROSION MECHANISMS BOILING POTASSIUM COMPATIBILITY
SPACE RADIATOR:	
Be, Ti, V, STAINLESS STEEL	FABRICATION FEASIBILITY

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FIG 86

F. Critical Experiments

1. Introduction

At the beginning of CANEL's program of development of lithium-cooled reactors in 1957, only a limited amount of nuclear cross section data existed for lithium and for the refractory structural materials such as columbium required for high-operating temperatures. Reactor physics data consisted of reactivity coefficients for small quantities of these materials measured for the fast reactor programs at Los Alamos and Argonne and in the naval reactor work at KAPL. Since that time, critical experiments for lithium-cooled reactor concepts have been performed at CANEL for a wide range of reactor configurations. Core diameters have varied from 10 to 18 inches, with critical masses between 40 and 175 kilograms of U^{235}. Both moderated and unmoderated cores have been studied and reflector materials have been high-Z materials as well as good moderators. For reliable calculations of fuel loading, reactivity control, power distribution, fuel burnup and other significant reactor characteristics, it was essential that a large volume of experimental data on a number of such various reactor concepts be provided, so that methods of calculation and nuclear cross sections could be developed and improved. The summary of CANEL critical experiments presented in Fig. 46 (Reactor and Shield Section) also includes the current program conducted by CANEL for United Nuclear Corporation.

2. Current Program

a. Present Status

Two series of critical experiments on CANEL critical assemblies CCA-5 and CCA-7 were undertaken to obtain reactor physics data to substantiate the nuclear analysis of compact, high-temperature reactors for space applications based on the lithium-columbium technology developed at CANEL. The CCA-5 experiments consisted of a parametric study of UO_2-BeO fueled reactors. The CCA-7 series is a parametric study of UC-fueled reactors, but including a study of the PWAR-20 Reactor design. A third series of experiments on CCA-6 was a detailed investigation for the Lithium Cooled Reactor Experiment.

The purpose of the parametric studies CCA-5 and CCA-7 was to provide data to indicate basic trends in variations of critical masses, reactivity coefficients, fission rate distributions, and effectiveness of possible methods of reactivity control with variations in core size, reflector thickness, and material composition. Emphasis has been on simplicity of critical assembly configurations in order that the results be most easily interpreted by analyses with multigroup transport theory digital computer codes. The results have been useful in discovering deficiencies in analytical methods and cross section data.

In the CCA-5 series, critical masses were determined for 18 basic assembly configurations including cylindrical cores with diameters of 10 to 14 inches and lengths of 12 to 15 inches, side reflector thicknesses of 4 to 8 inches, and end reflectors from zero to 6 inches thick. Core compositions corresponded to three different uranium to beryllium ratios but emphasis of the experiments was on compositions representing approximately 50 volume percent of 93 percent enriched UO_2-BeO mixture. Side reflectors of beryllium oxide were investigated as well as the basic beryllium composition. Side reflectors of nickel and polyethylene were also studied in connection with investigations of the hazards of reflection of bare cores by water or dense metals. The results of these investigations as well as these of the CCA-6 program are reported in Ref. 70.

In the current program, the CCA-7 series, for UC-fueled reactors, two core diameters, 10 and 12 inches, are being investigated each for core lengths of 10.8, 12.4, and 14.2 inches. At approximately the midpoint of this program, critical masses have been determined for 30 basic configurations. Emphasis in this program has been on determinations of the dependence of critical mass on side reflector thickness, and on the thickness of pressure shell and void annuli between the core and side reflector, and of the effectiveness of reactivity control by displacement of the side reflector.

The basic side reflector material is BeO, but the effects of hydrogeneous reflectors has also been investigated in connection with hazards studies. The following tabulated critical masses determined for the 10-inch diameter 14.2-inch long core are typical of the results obtained:

1" Cb. Annulus Between Core and Side Refl.		2" Cb. Annulus Between Core and Side Refl.	
Side Refl.	C. M.	Side Refl.	C. M.
0" BeO	123 Kg U^{235}	0" BeO	112 Kg U^{235}
2	97	1	101
4	83	3	88
6	76	5	83
8	73	7	81
9" CH_2	90	8" CH_2	93

These results indicated that, for a BeO reflector, thickness of about 4 inches, the increase in columbium thickness from 1 to 2 inches required an increase in uranium loading of only 2 to 3 kilograms. But in terms of uranium mass, the reflector worth was reduced by about 10 kilograms, a fact which is important to the feasibility of reactivity control by side reflector displacement. They also indicated that to guarantee safety under conditions of reflection by water, the uranium loading must be less than 90 kilograms or that the reflector thickness must not be less than about 4 inches.

The results of calculations of a few of these assemblies, listed below, were typical in that they showed that significant improvement in two-dimensional Sn calculations could be obtained by increasing the order of n. The 16-group cross sections of Hansen and Roach were used (Ref. 17).

Configuration	Sn	k_{eff}
1" Cb 2" BeO	S2	1.025
	S4	1.007
1" Cb 4" BeO	S2	1.029
	S4	1.001
	S8	1.000
1" Cb 6 CH_2	S2	1.110
	S4	1.032
	S8	0.999

The evaluation of the quality of the cross sections currently used for SNAP-50 reactor analysis is a continuing process, and significant improvements have been made where discrepancies have been found in comparisons of calculated with experimental reactivy coefficients. Reactivity coefficients are calculated by means of a two-dimensions perturbation theory code, developed at CANEL, which makes use of forward and adjoint flux distributions calculated with the TDC code. Some results of measurements and S8 calculations of reactivity coefficients for the 12-inch diameter 12.4-inch length core show that further work is required in adjusting the cross sections of some of the refractory metals and of Li^7.

Material	Experiment	Calculated	Data Source
U^{235}	88.9 ¢/Kg	87.4 ¢/Kg	LAMS-2941
U^{238}	6.38	6.57	LAMS-2941
Al	25.2	25.4	LAMS-2941
Be	117	133	LAMS-2941
		126	Modified
C	60.5	62.	LAMS-2941
Li^7	84	64.	LAMS-2941
0	40.9	49.5	LAMS-2941
Cb	5.47	1.93	LAMS-2941
		5.17	Modified
W	-1.97	0.64	GAM-II
		- .01	ANL-6656
Ta	-18.1	-13.6	LAMS-2941
		-19.1	Modified
Mo	6.12	4.23	LAMS-2941
		8.28	GAM-II
Re	-32.9	-23.3	GAM-II

Experiments with a detailed mockup of the PWAR-20 Reactor reflector control segments with a 12-inch diameter 12.4 long core, demonstrated that the total range of control was approximately 0.22 Δk, or significantly more than required. Three-dimensional flux synthesis Sn methods are currently under development for calculating such configurations.

b. Plans for FY 1965 and FY 1966

The current series of critical experiments is expected to be completed near the end of FY 1965. Upon completion of the current series of experiments, the critical experiment program will be extended to include studies of a wider range of core dimensions, investigations of methods of poison control of reactivity and fuels other than UC, such as UN and refractory cermets.

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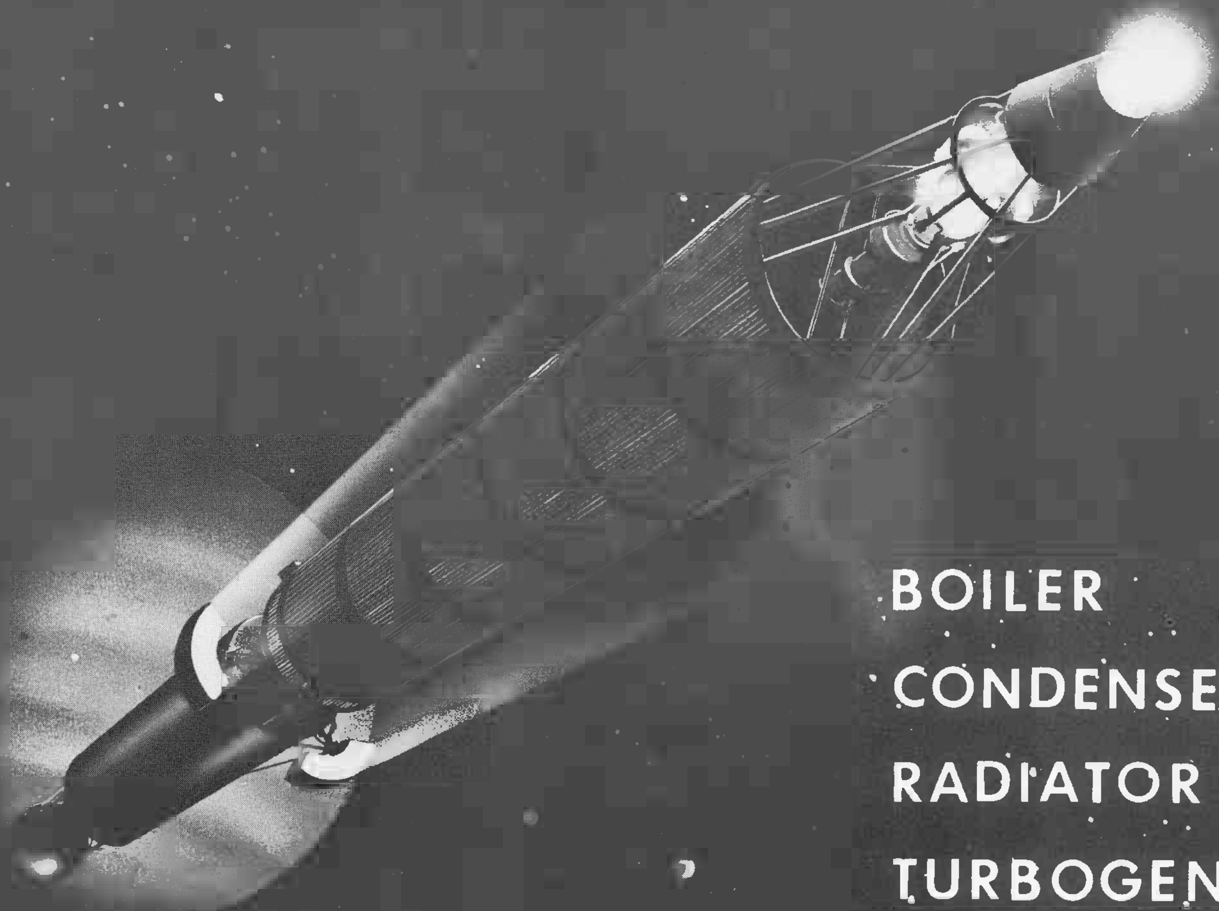
SNAP 50/SPUR



BOILER
CONDENSER
RADIATOR
TURBOGENERATOR

GARRETT

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[redacted]

[redacted]

SNAP 50/SPUR POWER CONVERSION COMPONENT DEVELOPMENT STATUS

This document briefly describes the SNAP 50/SPUR boiler, condenser, radiator, and turbogenerator component development work being done by AiResearch.

The program approach is to establish an overall system design, and derive from it the individual component performance objectives. This aids in assessing design risk and apportioning development effort appropriately among the various components, and assures that the end result of each component program will be compatible with system requirements. The system design and resulting component design objectives have been established by detailed analyses on the part of CANEL and AiResearch. We believe that adequate data exists to substantiate the predicted system performance and the material selections. Therefore, a high degree of confidence exists that the resulting component design objectives can be met.

The development effort described herein is being conducted with test systems constructed principally of nonrefractory materials in order to simplify test procedure and to avoid the time and cost that would be associated with early testing of refractory-alloy components. The development program provides test experience on important items (bearings, seals, turbine rotors, generator rotor and stator, etc.) under conditions duplicating those of the final refractory alloy components, and also provides the bulk of the additional data needed for the design of the refractory alloy components.

The following pages describe the design of each of the components listed, the major development problems of each, and the methods (analysis, test, etc.), being used in the SNAP 50/SPUR program to resolve these problems.

SNAP 50/SPUR BOILER

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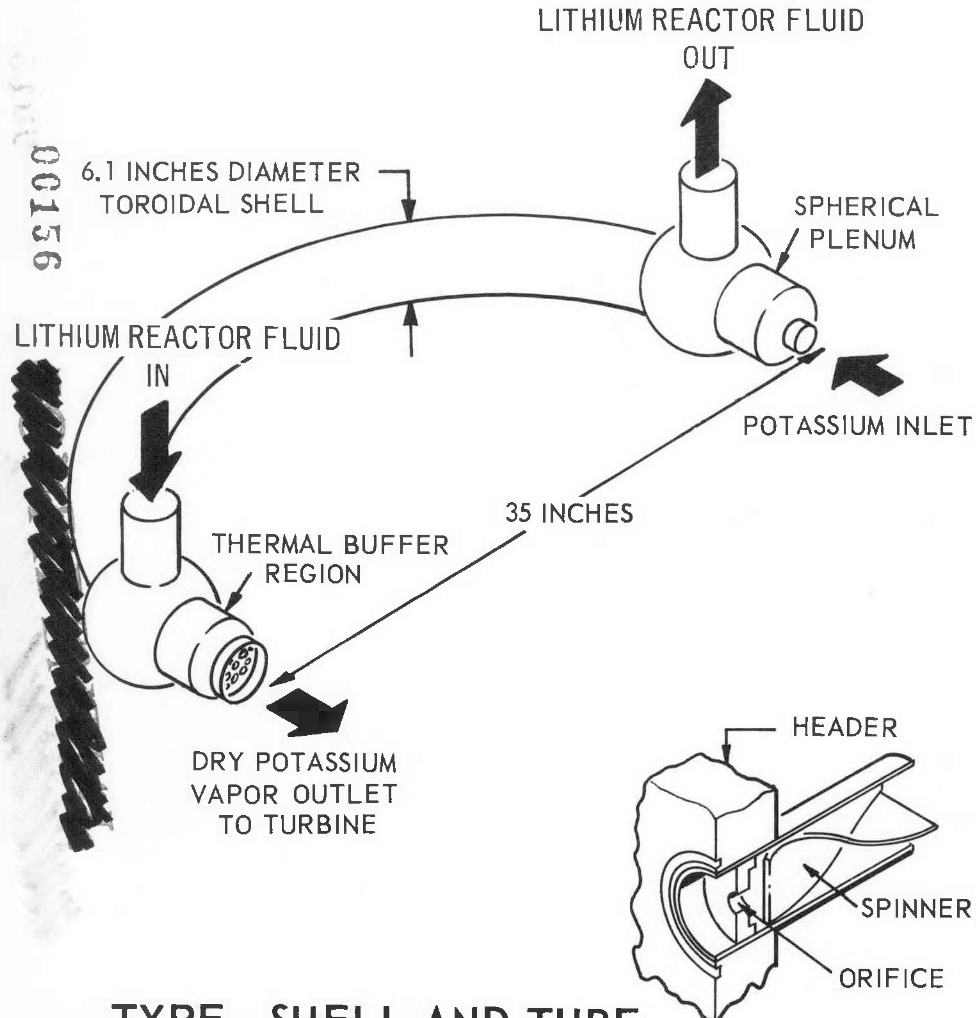

TYPE: SHELL-AND-TUBE
RATING: FOR 300 KW (e) SYSTEM
WEIGHT: 220 LBS
MATERIAL: CB - 1% ZR

DESIGN OBJECTIVES:

- VAPORIZE ENTIRE POTASSIUM FLOW TO DRY VAPOR (100% QUALITY)
- OPERATE INDEPENDENT OF GRAVITY
- MINIMUM WEIGHT
- 10, 000 HOUR LIFE

DEVELOPMENT AREAS

- HEAT TRANSFER { BOILING / LIQUID
- STABILITY
- STRUCTURAL DESIGN AND FABRICATION

[redacted]

MS-752-1

The task of the boiler is to supply dry vapor to the potassium turbine. The boiler is a tube-and-shell unit fabricated of Cb - 1 Zr. The size and weight shown are for a system of 300 kw electrical output. Potassium flowing through the 3/8 inch diameter tubes is vaporized by heat transferred from the lithium reactor coolant, which flows outside the tubes in a counterflow arrangement with respect to the potassium flow. The circular-arc shall shape is selected to minimize differential thermal stresses between shell and tubes. Twisted-tape swirlers are inserted in the tubes to provide a force field to centrifuge entrained droplets to the wall where they can be vaporized. The twisted-tape approach is also used in the SNAP 8 boiler. Orifices at the tube inlets are used to provide intertube flow stability.

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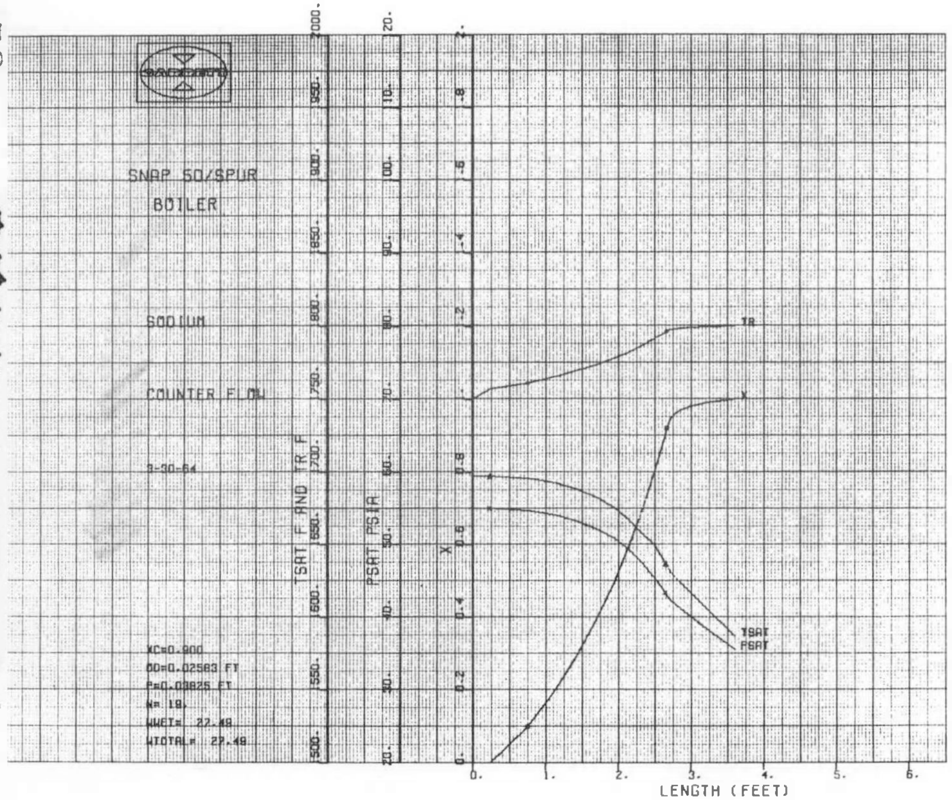

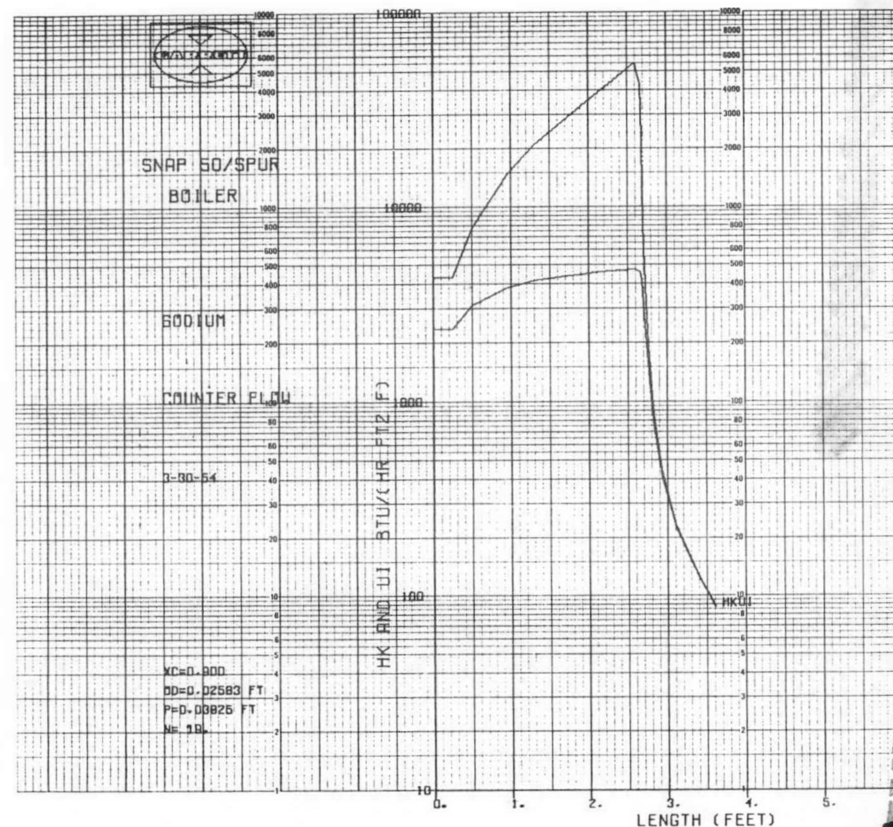

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TYPICAL PRINTOUTS OF COMPUTER ANALYSIS DATA

The boiler heat-transfer design is accomplished using a digital computer program whose heat transfer and pressure drop equations are based on correlation with test data. Typical printouts (in this case for a lower-temperature test boiler) are shown above; the left-hand plot shows lithium temperature (TR), quality (X) and potassium temperature and pressure (TSAT, PSAT); the right-hand plot shows potassium-side heat transfer coefficient (upper curve) and overall coefficient (lower curve). These curves illustrate that over most of the quality range it is not the boiling potassium heat transfer which sizes the boiler; until the boiling coefficient falls off at high quality, the lithium-side heat transfer is determining. Therefore, there are different levels of accuracy required for knowledge of various coefficients; the lithium-side coefficient, and the high-quality boiling coefficients, must be known accurately; of the low-quality boiling coefficients, it need only be confirmed that they are much higher than the lithium coefficient.

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SOURCES OF BOILER DEVELOPMENT DATA

TYPE OF DATA	SOURCE	
	OTHER PROGRAMS	SNAP 50/SPUR PROGRAM
HEAT TRANSFER		
BOILING		
LOW-MEDIUM QUALITY	ADEQUATE	SINGLE-TUBE BOILING HEAT TRANSFER TEST
HIGH QUALITY (>80%)	INSUFFICIENT	
LIQUID	ADEQUATE	
BOILING STABILITY	INSUFFICIENT	FREON FLOW VISUALIZATION TESTS
STRUCTURAL DESIGN	PARTIAL	EXPERIMENTAL STRESS ANALYSIS (AIRESEARCH) FABRICATION DEVELOPMENT (CANEL)

MS-744-2

This being the case, the existing data on liquid metal coefficients and on low-to-medium quality boiling coefficients, available as the result of previous and current liquid metal heat transfer programs such as those at ORNL, CANEL, GE, NASA, Geoscience Corp., University of Michigan etc., is adequate for design of the SNAP 50/SPUR boiler. Thus the major heat transfer efforts in this program are directed toward development of high-quality data and of means to increase high-quality coefficients, and of means to assure stable boiling operation. The final boiler design is selected on the basis of weight optimization studies such as that shown below. Structural design likewise relies on previous experience, with supplemental experiments directed specifically at SNAP 50/SPUR programs.

These efforts are briefly described in the following pages.

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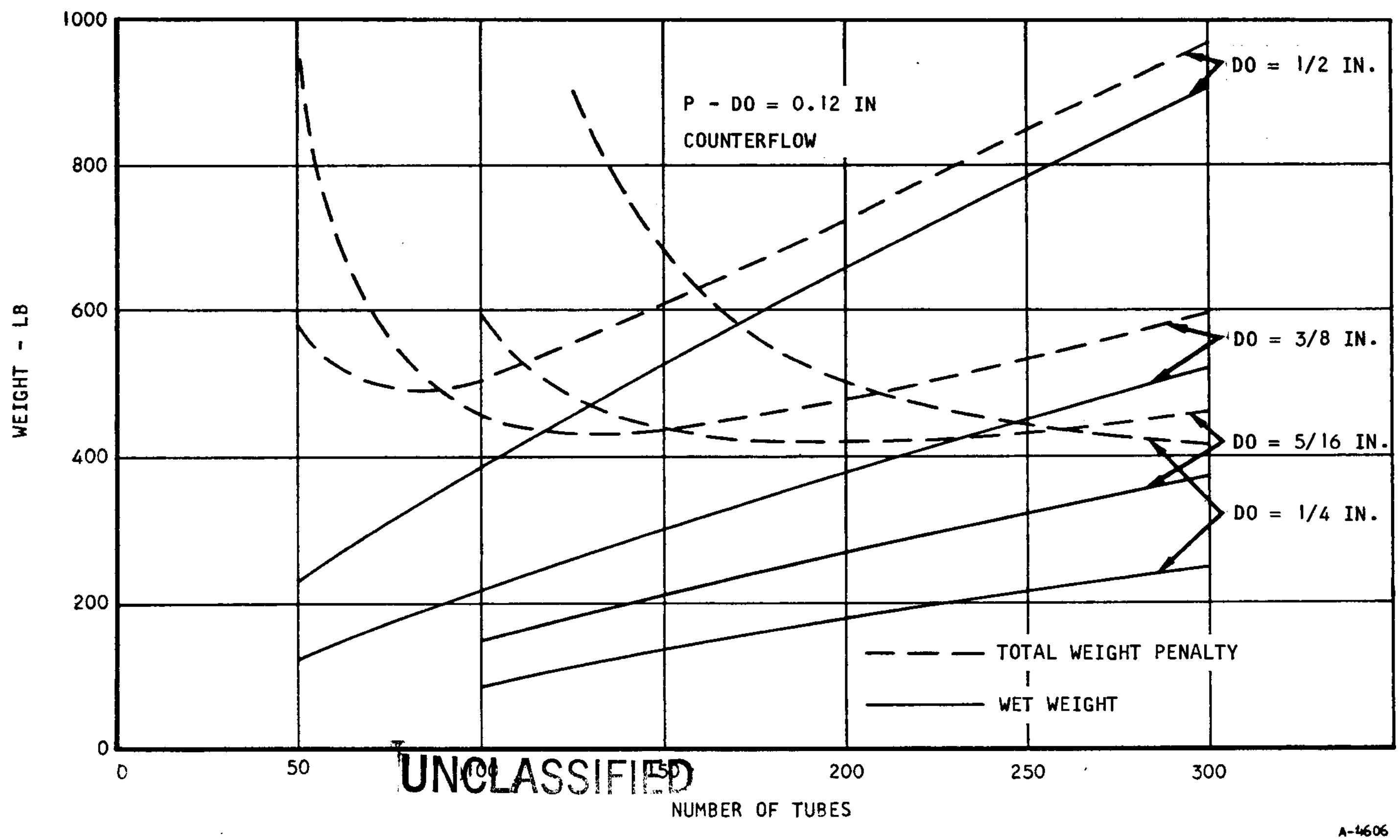
SNAP 50/SPUR BOILER PARAMETRIC STUDY



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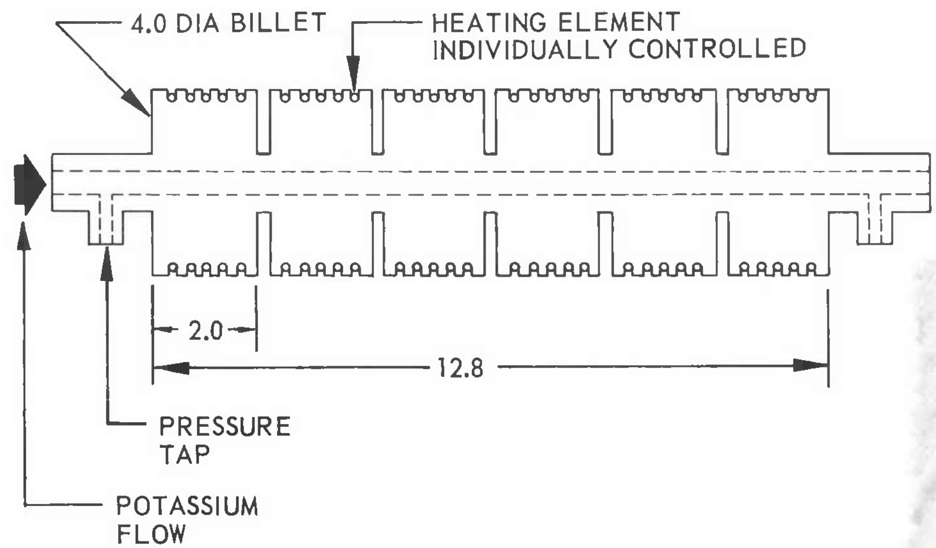
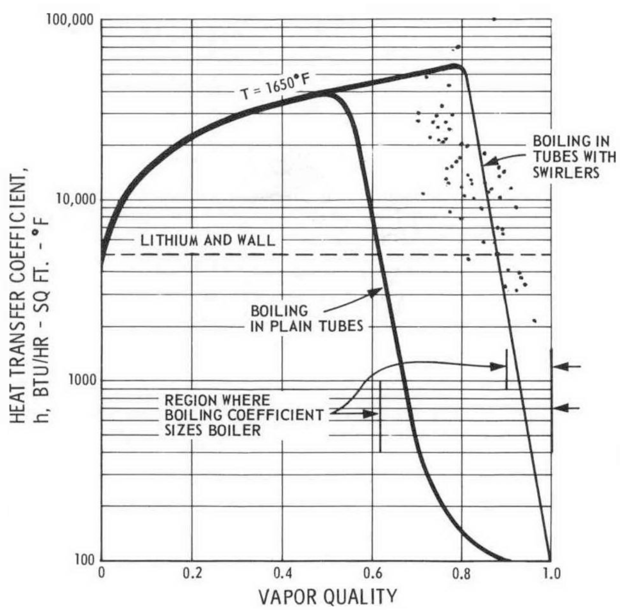
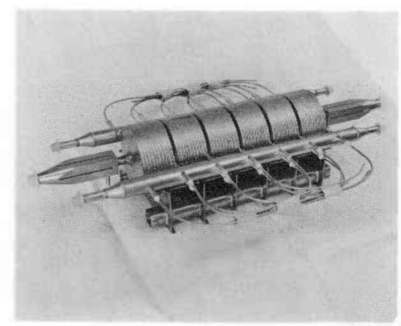
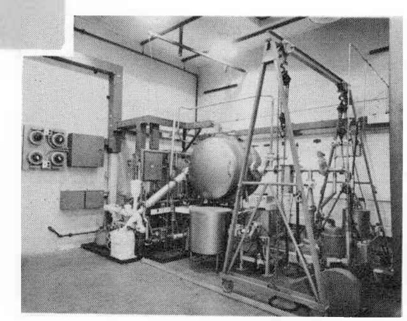
Single-Tube Potassium Boiling Heat Transfer Tests

- ***PROVIDE BOILER DESIGN DATA***
- ***CONFIRM EFFECTIVENESS OF SWIRLERS IN INCREASING PERFORMANCE AT HIGH QUALITY***









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The achievement of high heat transfer coefficients at high vapor quality is essential to minimize boiler weight. The potassium coefficient is not the determining factor throughout most of the low-to-medium quality region, as the lithium-side coefficient determines the required heat transfer area in this region. When, with increasing quality, the tube wall becomes dry while there are still liquid droplets in the stream, the potassium coefficient falls to the gas value of about 100 Btu/hr ft^2-^{o}F, and this coefficient sizes the remainder of the boiler. To minimize boiler size, it is desired to postpone the transition to dry wall to as high a quality as possible, in order to minimize the amount of heat to be transferred under this condition. This is done in the SNAP 50/SPUR boiler by inserting a twisted tape in the tube, to produce a force field tending to centrifuge liquid droplets to the outer wall, where they can be directly vaporized. This has the effect, shown in the curve, of raising the dry-wall-transition quality from about 50 percent to 80-90 percent. The improvement in potassium coefficient shown for swirlers over plain tubes represents about an order of magnitude reduction in the required boiler size, with saving of hundreds of pounds in system weight.

The figure illustrates the test apparatus being used to measure high-quality boiling heat transfer coefficients. Potassium vapor of known quality is passed through the six-section, electrically-heated test boiler, where quality increases of 2 to 5 percent per section are accomplished. Heat transfer coefficients are calculated in each section from measurement of the heat flow, wall temperature, and fluid temperature. Heat flow is calculated independently by measurement of net electrical power input and from the test section radial temperature gradient. Wall temperature is calculated from test section radial temperature gradient. Fluid temperature is estimated from measurements at boiler inlet and outlet. The resulting data, shown above and in the following curves, provides verification of the effectiveness of the swirlers in increasing high-quality heat transfer. This work is reported in AiResearch Report L-9452, "Forced-Convection Vaporization of Potassium in a Single Tube."

The data shown here is supported by a large amount of similar data obtained at CANEL in previous years covering a wide range of tube geometries in addition to that selected for the SNAP 50/SPUR boiler.

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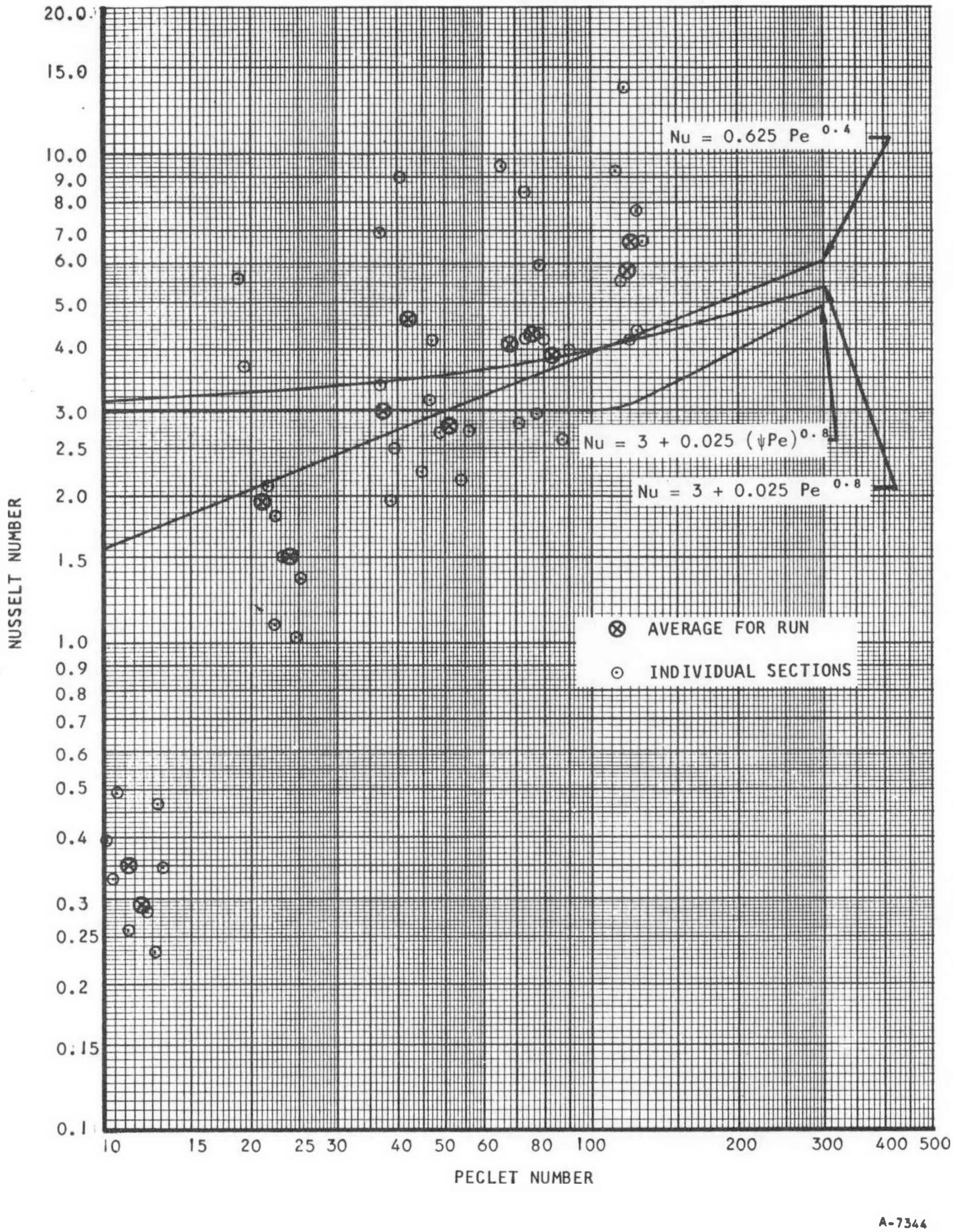

POTASSIUM LIQUID HEATING RESULTS

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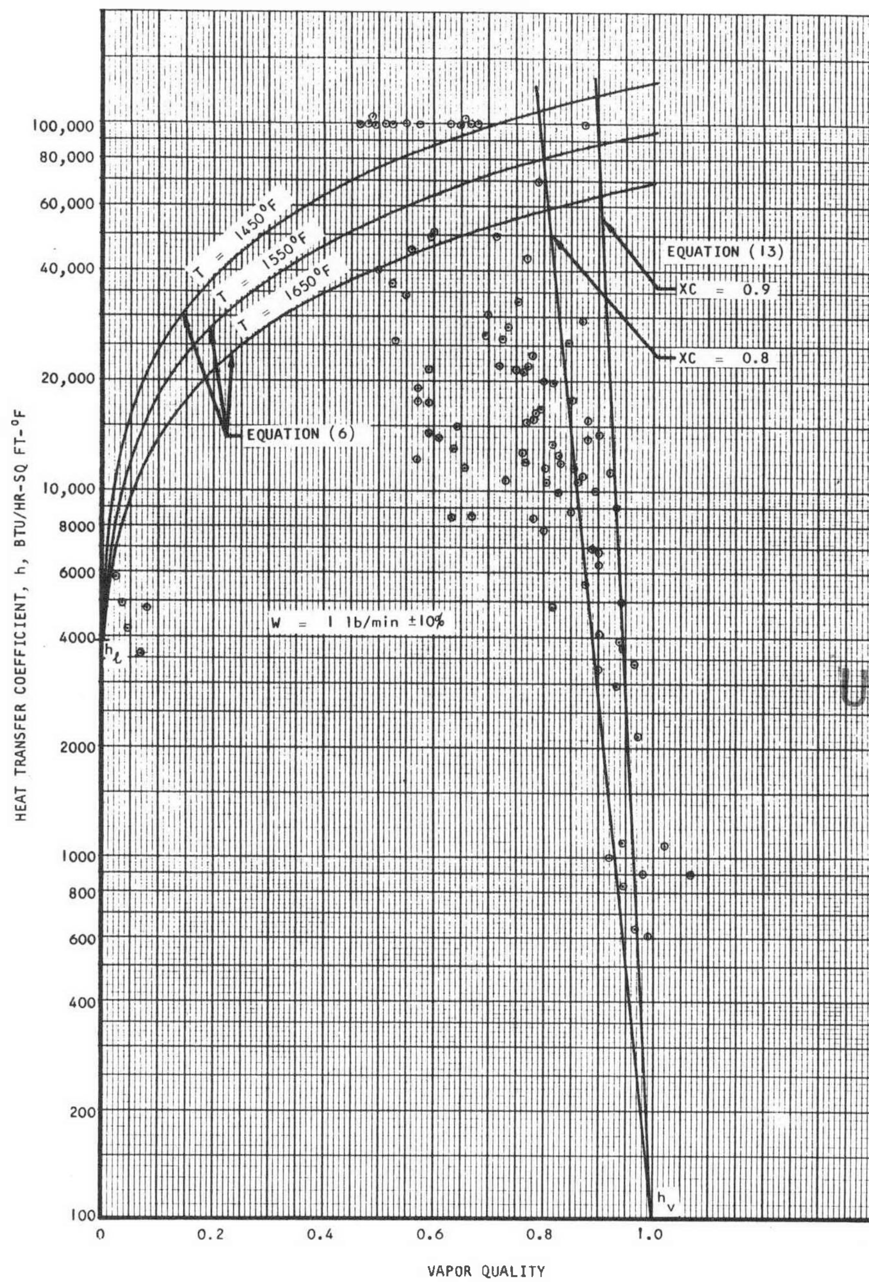
[redacted]



POTASSIUM VAPORIZATION RESULTS

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FREON FLOW VISUALIZATION TESTS

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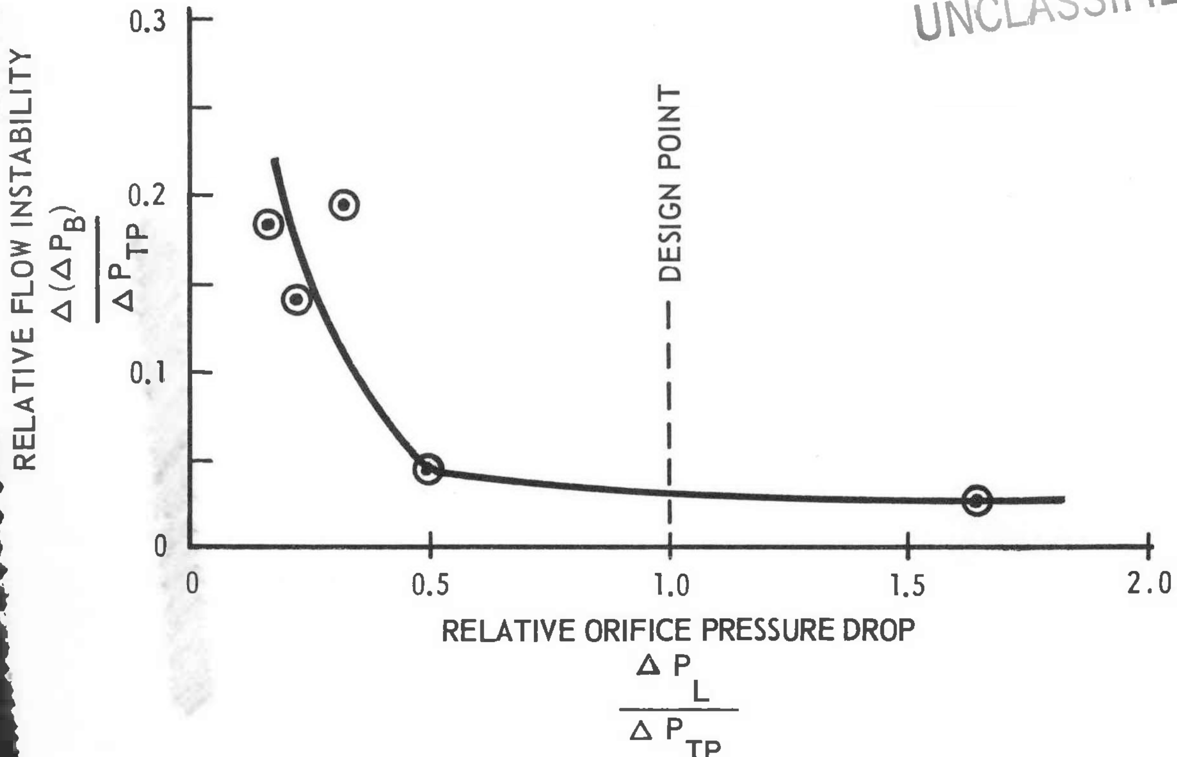
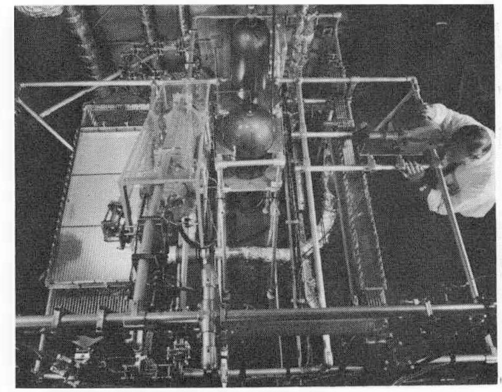
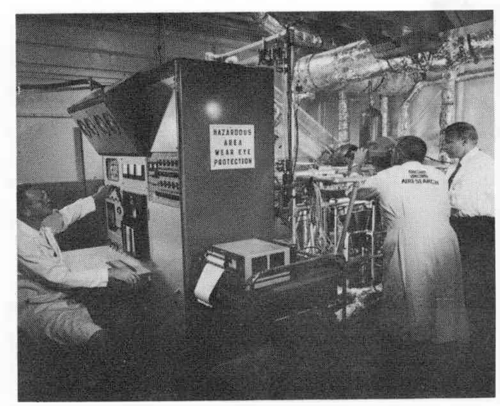
- ***PROVIDES STABILITY CRITERIA***







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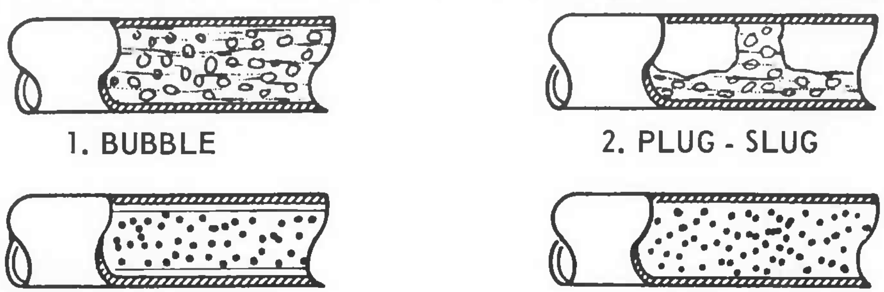
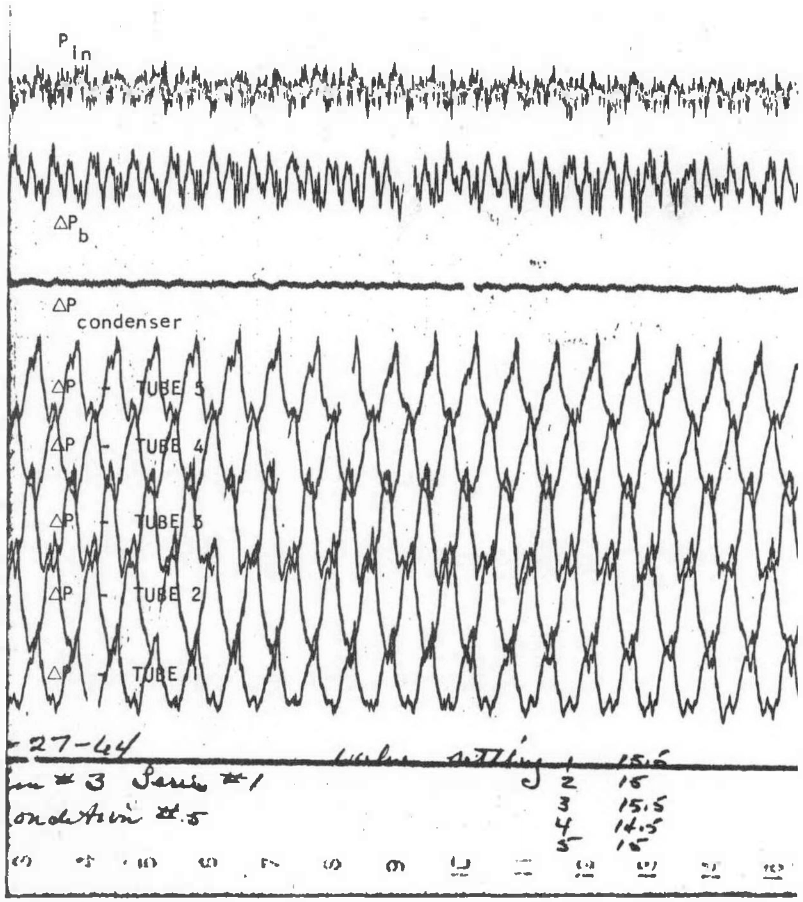
- ***PROVIDES DATA ON FLOW MECHANISMS***



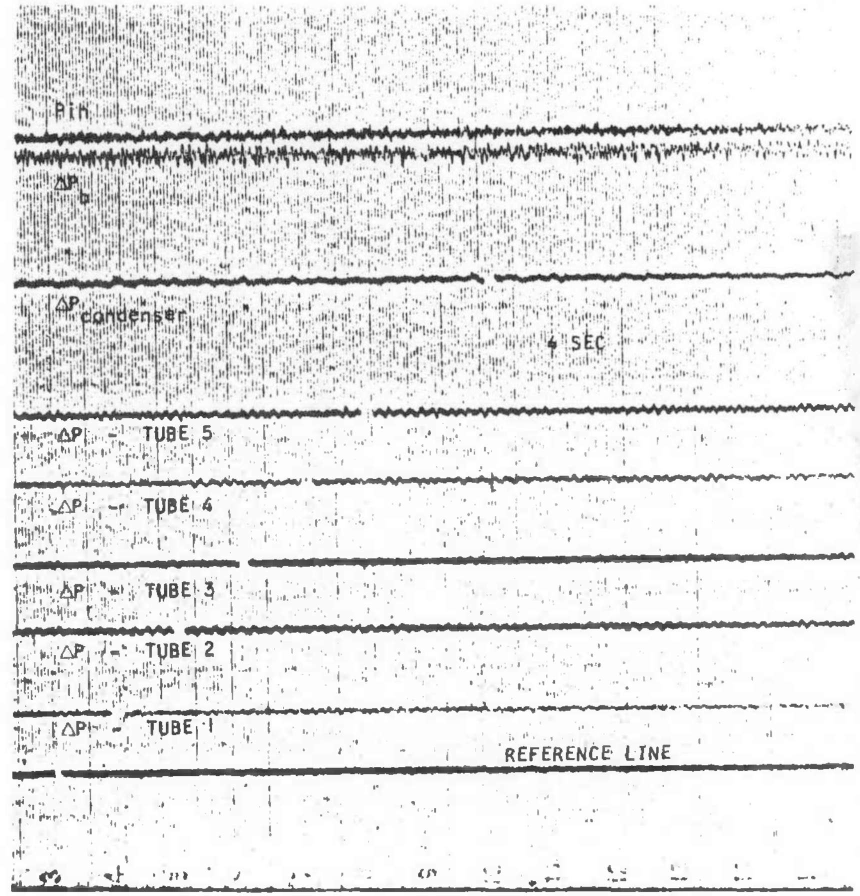
MS-752-3

Provision of adequate boiling stability is essential to meeting the objective of dry vapor output with no liquid carry-over. As is typical of SNAP 50/SPUR programs, the flow visualization tests are aimed primarily at confirming the effectiveness of the method chosen in providing stability, and only secondarily at producing general data on stability.

The test loop shown is a closed cycle in which Freon 113 is vaporized in an air-heated 5-tube test boiler, and condensed in a 30-tube air-cooled test condenser and a water-cooled workhorse condenser. The system is used to investigate flow stability, multichannel flow distribution, and high-quality dry-wall transition. Stability is investigated by applying varying values of dimensionless parameters representing the relative degree of inlet liquid subcooling, the vapor-liquid density ratio, the quality, the tube L/D, the ratio of liquid ΔP to two-phase ΔP, and the ratio of two-phase pressure drop to inlet pressure, and noting the effect on boiling stability, represented by the magnitude of fluctuations in two-phase ΔP. The value of inlet orifice pressure drop required to provide stability is determined for each set of parameters, providing data as shown in the curve.



TRANSDUCER TRACES: TYPICAL UNSTABLE FLOW

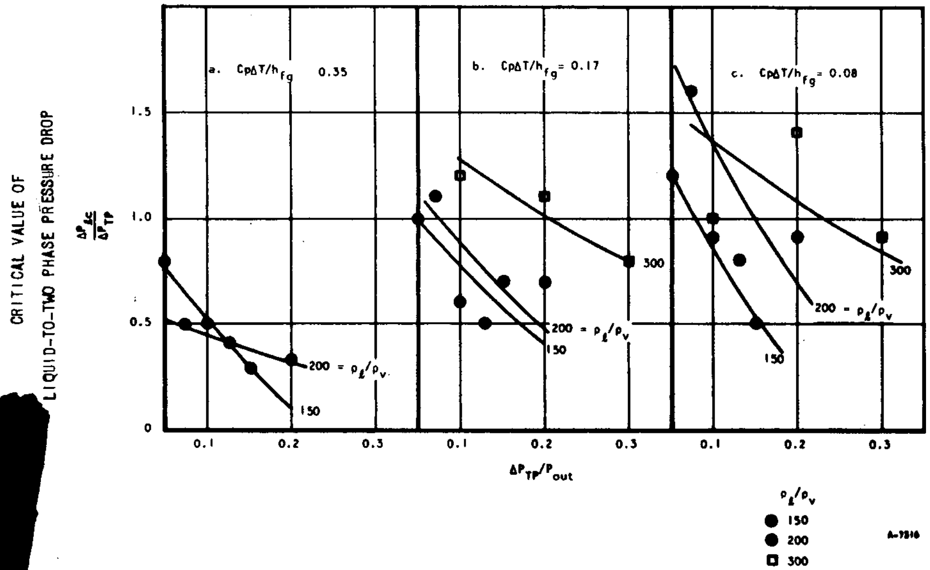


TRANSDUCER TRACES: TYPICAL STABLE FLOW

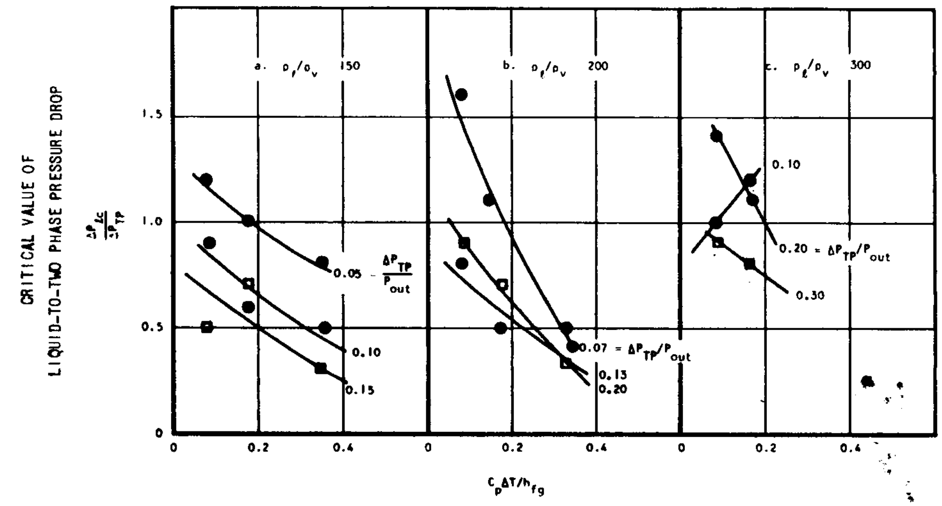
Stability is evaluated from oscillograph traces of boiler flow and ΔP, such as the two shown above. The effect on the critical value of orifice pressure drop of those parameters is illustrated in the curves below. On the basis of this data, the inlet orifices for the first multitube potassium test boilers have been sized.

In addition to quantitative data, this program has provided understanding of the flow mechanisms of forced-convection boiling, leading to identification of the nature of the high-quality dry-wall problem and selection of the swirler insert as a means of solving it.

This program is reported in AiResearch Report L-9448, "Flow Stability in Multitube Forced-Convection Vaporizers."

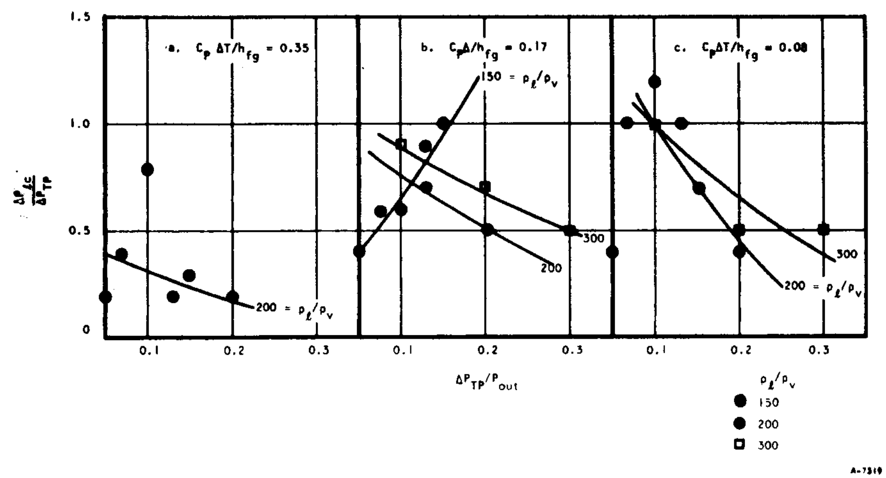


Effect of Fractional Pressure Drop with 3/4 Tape

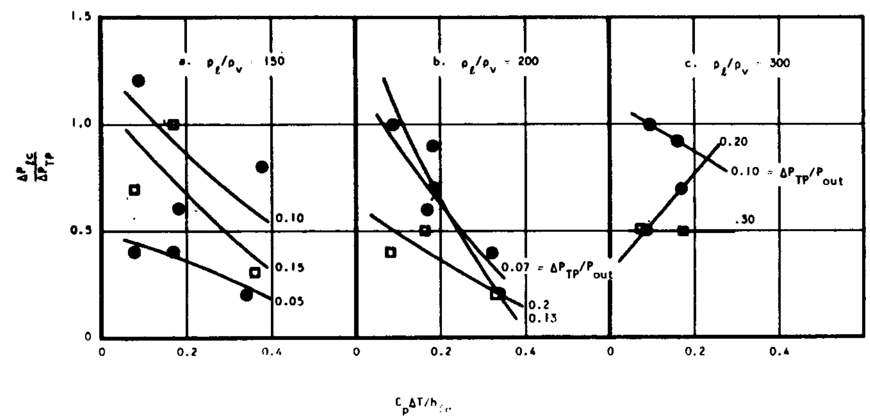


Effect of Subcooling with 3/4 Length Tape

00169



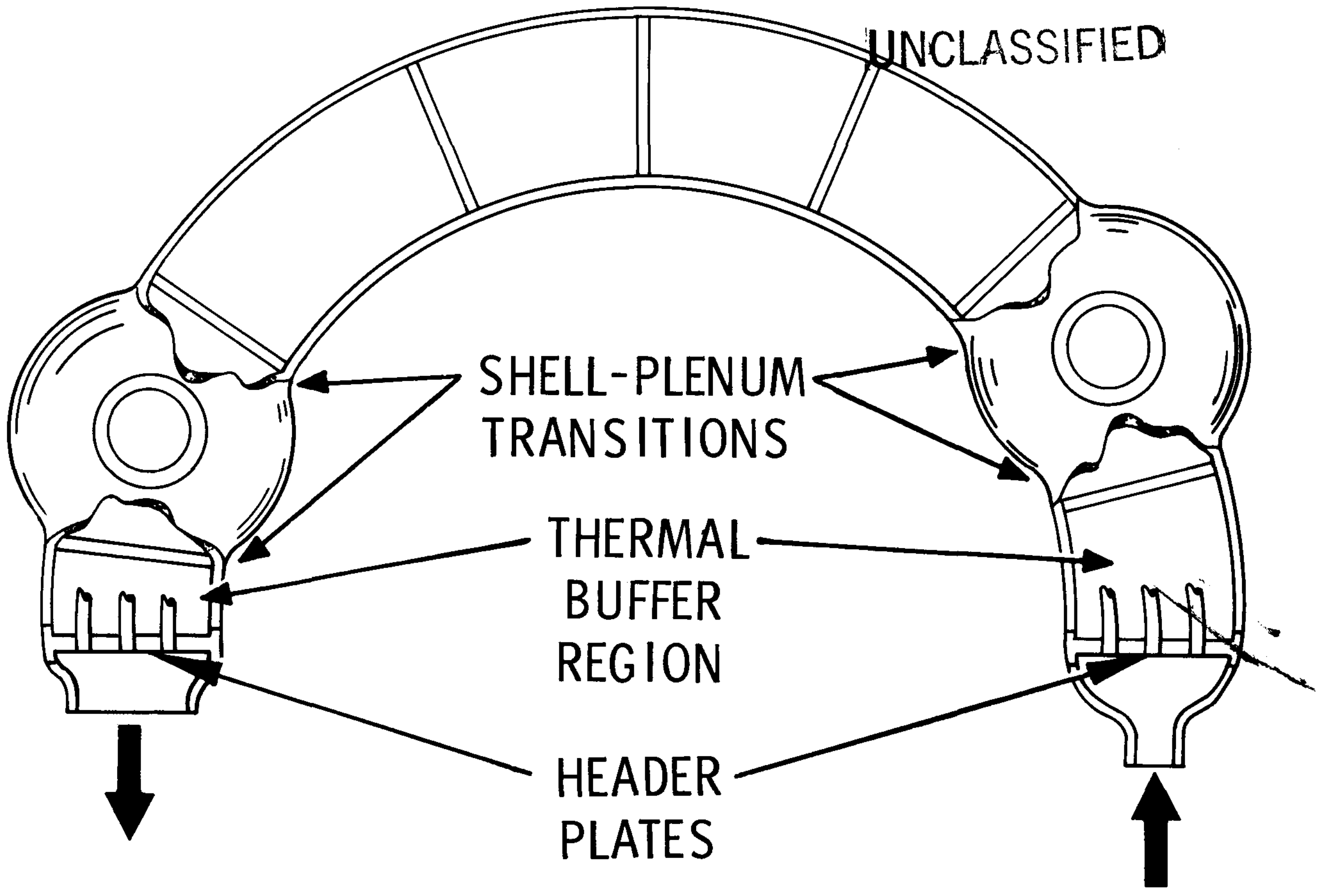
Effect of Fractional Pressure Drop with Full Tape



Effect of Subcooling with Full Length Tape

EXPERIMENTAL STRESS ANALYSIS PROGRAM

CRITICAL STRUCTURAL DESIGN AREAS:



PROGRAM:

1. SMALL-SCALE HS-25 SECTIONS → HS-25 BOILER DESIGNS; → GENERALIZED DESIGN DATA
2. FULL-SCALE CB-1% ZR SECTIONS → FULL-SCALE BOILER DESIGN

Critical structural areas are the shell-plenum transition, due to high discontinuity stresses at the sphere-cylinder transition; the thermal buffer regions, designed to moderate sharp temperature gradients between shell-and tube-side fluids; and the header plates, whose complex geometry requires experimental design evaluation.

The shell-plenum transition will be investigated by photoelastic analyses of plastic models, strain gauge tests of air-heated Haynes-25 models, and stress-rupture tests of Haynes-25 models. Header plate stresses will be evaluated by tests of four-times-size aluminum models which compare deflection and stress patterns on plain discs, drilled headers, and headers with tubes inserted. Other future tests will evaluate temperature and stress gradients in thermal buffer regions by use of NaK-filled electrically-heated Haynes-25 model sections.

Another important area is developing a positive means of bonding the tape insert in the tube; if the tape is not firmly bonded, it may vibrate loose and break up under the boiling flow oscillations. The following photographs illustrate progress in diffusion-bonding of tapes in test boiler tubes.

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SECTION OF DIFFUSION BONDED TUBE-SWIRLER ASSEMBLY

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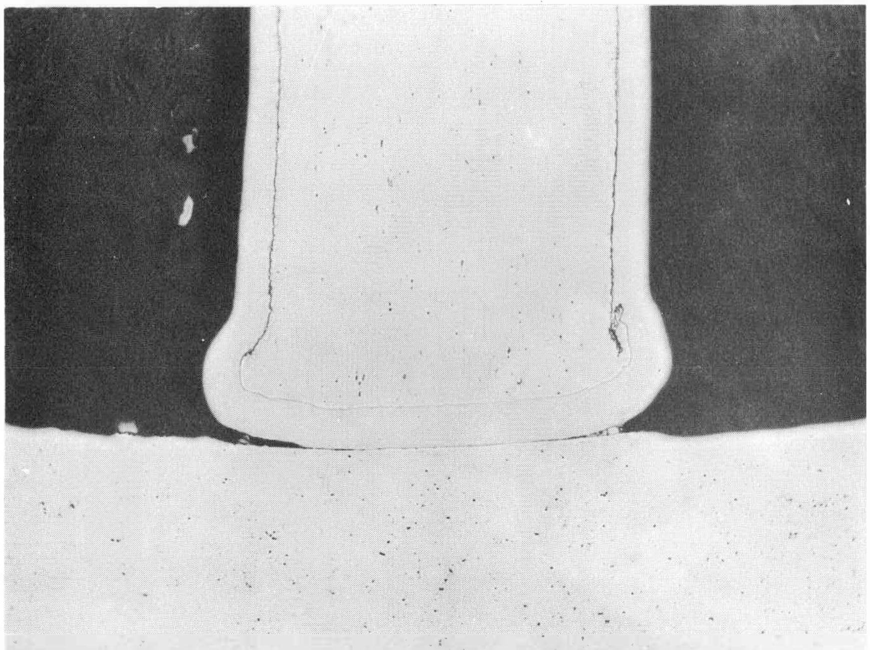
DIFFUSION BONDING OF SWIRLER-TO-TUBE JOINT



SWIRLER-TUBE ASSEMBLY AS DRAWN, BEFORE BONDING
SULFAMATE NICKEL PLATE ON TAPE IS .0015 INCHES THICK.

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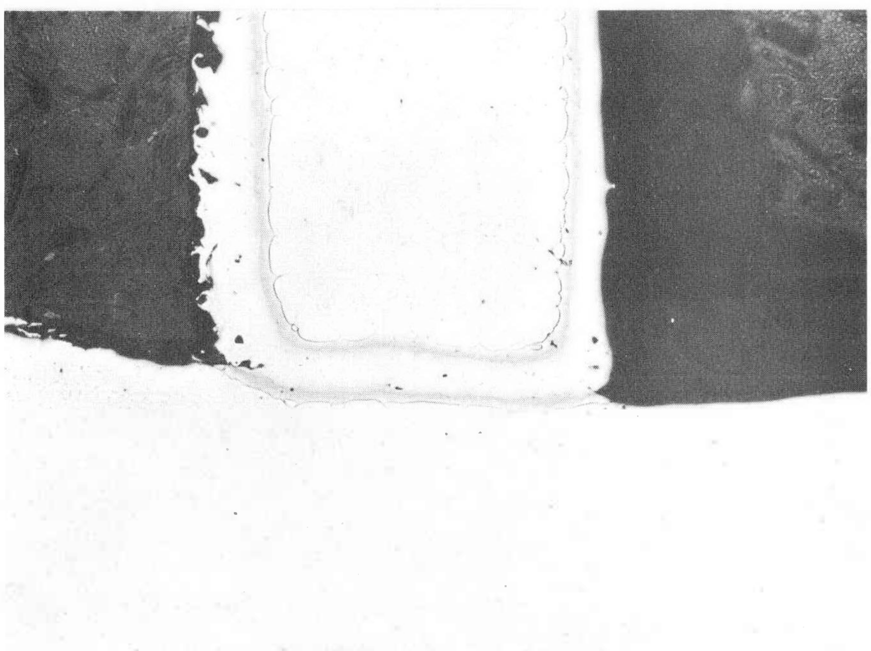

SWIRLER-TUBE ASSEMBLY AFTER DIFFUSION BONDING
AT 2200°F IN HYDROGEN FOR ONE HOUR.

A-4609

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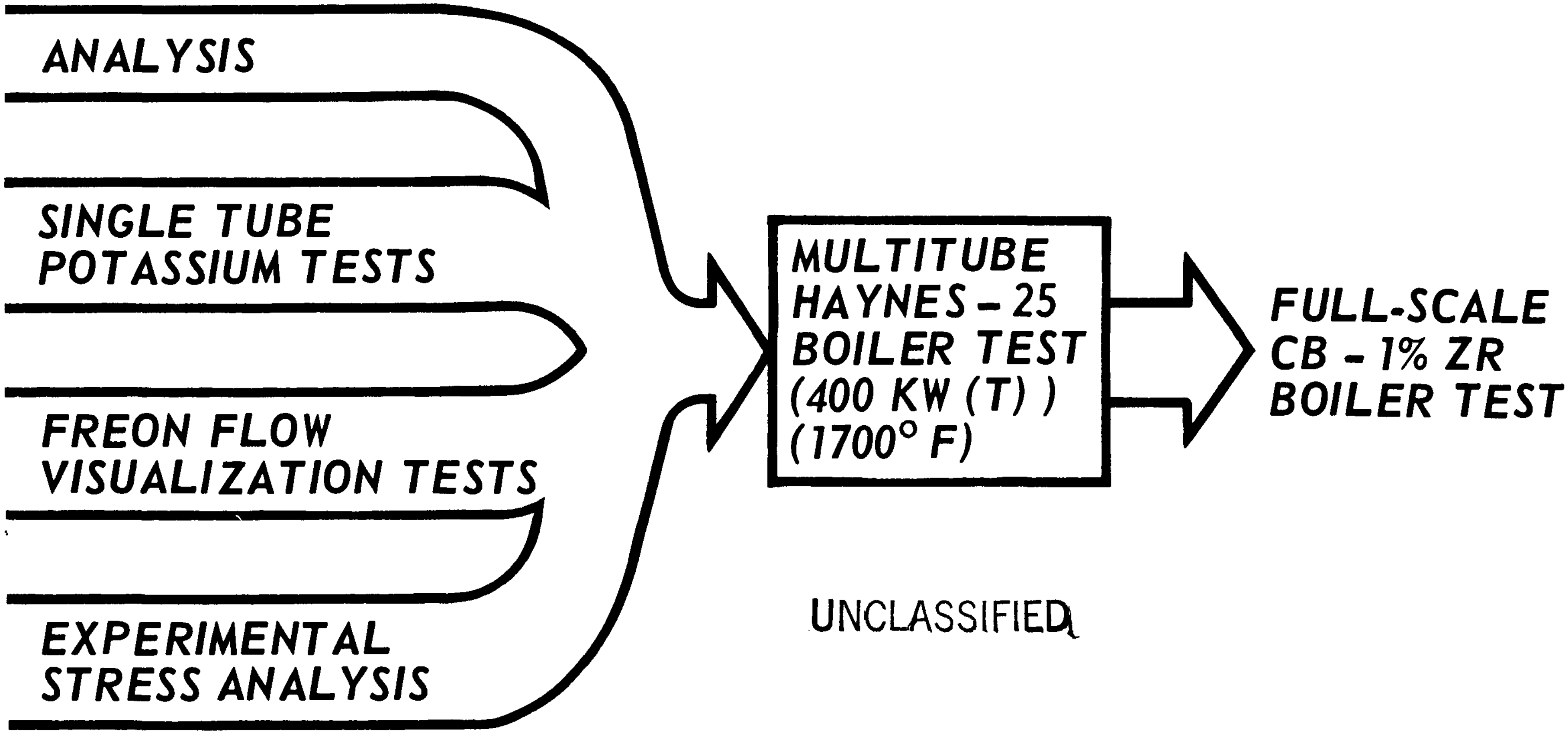
00173

BOILER DEVELOPMENT PROGRAM



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The programs described on the previous pages all provide data needed for high-temperature multitube potassium boiler tests, which will begin this fall. These tests will use boilers containing 15 to 20 tubes identical to those of 100- to 150-tube SNAP 50/SPUR boilers. This test program will in turn lead to future tests of full-scale Cb - 1 percent Zr boilers.

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00175

400-KW (T) HAYNES - 25 MULTITUBE BOILER TESTS

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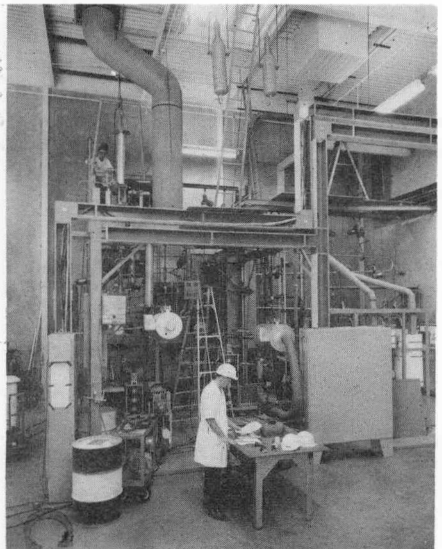

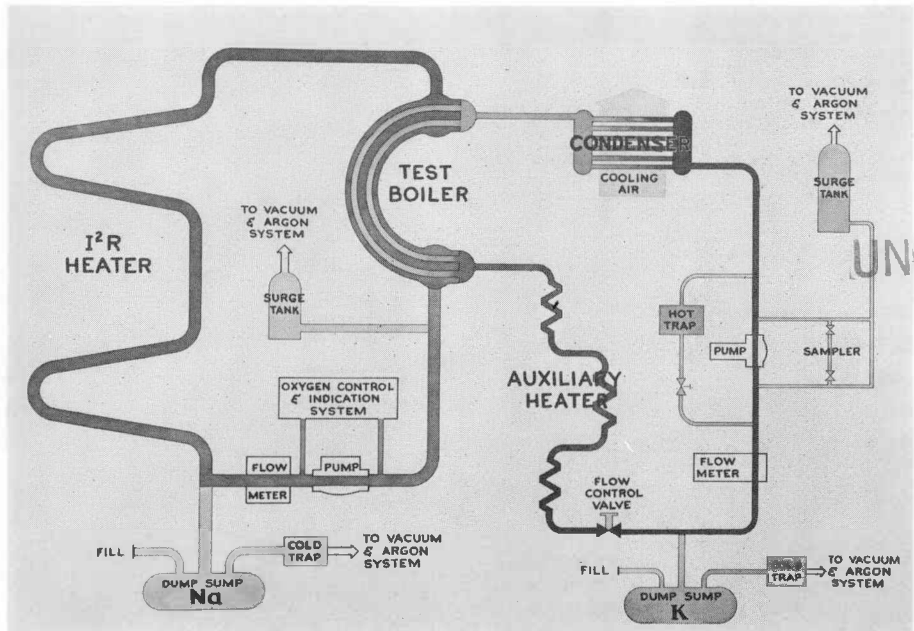

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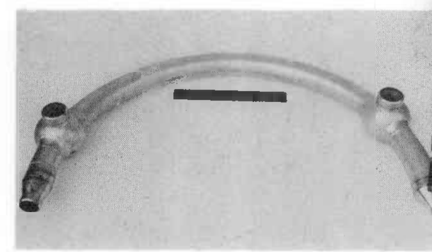

PROVIDES:

- OVERALL MULTITUBE PERFORMANCE
- STABILITY DATA
- STRUCTURAL DESIGN DATA

In this test loop, an electrical-resistance-heated sodium loop simulates the lithium reactor coolant. The potassium vaporized in the boiler is condensed in an air-cooled condenser and recirculated by a pump. An auxiliary potassium heater is used to vary the degree of subcooling of the potassium entering the boiler. Test data includes overall heat transfer performance, plus shell temperature gradient data which aids in both heat transfer and structural design. Initial performance tests will be followed by longer-term endurance tests.

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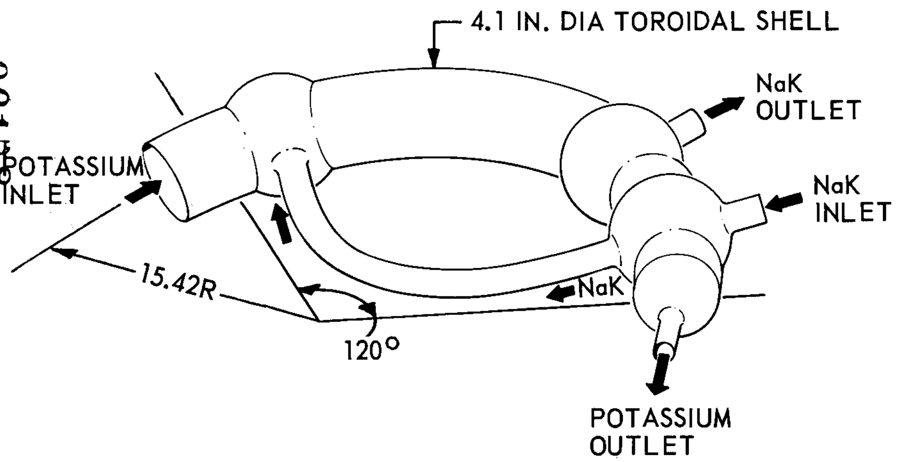
SNAP 50/SPUR - CONDENSER



MATERIAL-COLUMBIUM 1% ZIRCONIUM

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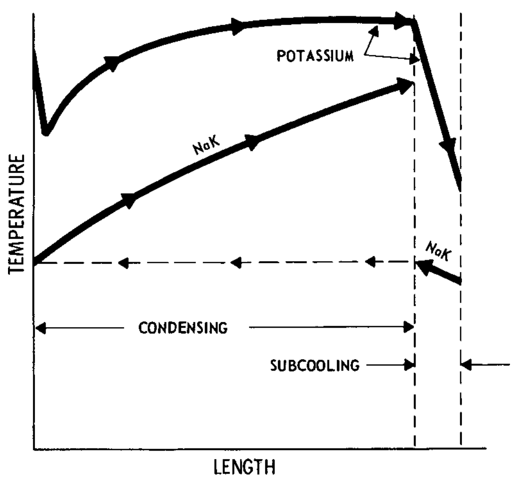
WET WEIGHT - 40 LBS



- TUBE-AND-SHELL
- CROSS-FLOW SUBCOOLER + PARALLEL - FLOW CONDENSING
- FOUR UNITS PARALLELED IN POTASSIUM CIRCUIT, SEPARATE NaK LOOPS

SIMILAR TO BOILER BUT:

- LOWER TEMPERATURES
- NO MAJOR HEAT TRANSFER OR STABILITY PROBLEMS

THUS DEVELOPMENT WILL PARALLEL BOILER DEVELOPMENT

The condenser is similar in many respects to the boiler, but represents a lesser development task. Condenser design studies are pursued using a digital computer program, typical results of which are presented on the next page. The design shown is selected on the basis of weight optimization studies as illustrated on the following page. The configuration shown was selected to provide parallel-flow condensing, selected because of the unique increasing-temperature characteristic of the condensing process. This results because the pressure rise due to momentum recovery exceeds the friction loss in a compact condenser.

The condenser program includes testing of scaled-down components in the 400-kw loop starting in about one year. The condenser program is planned for development generally concurrent with the boiler, to maximize utilization of test facilities.

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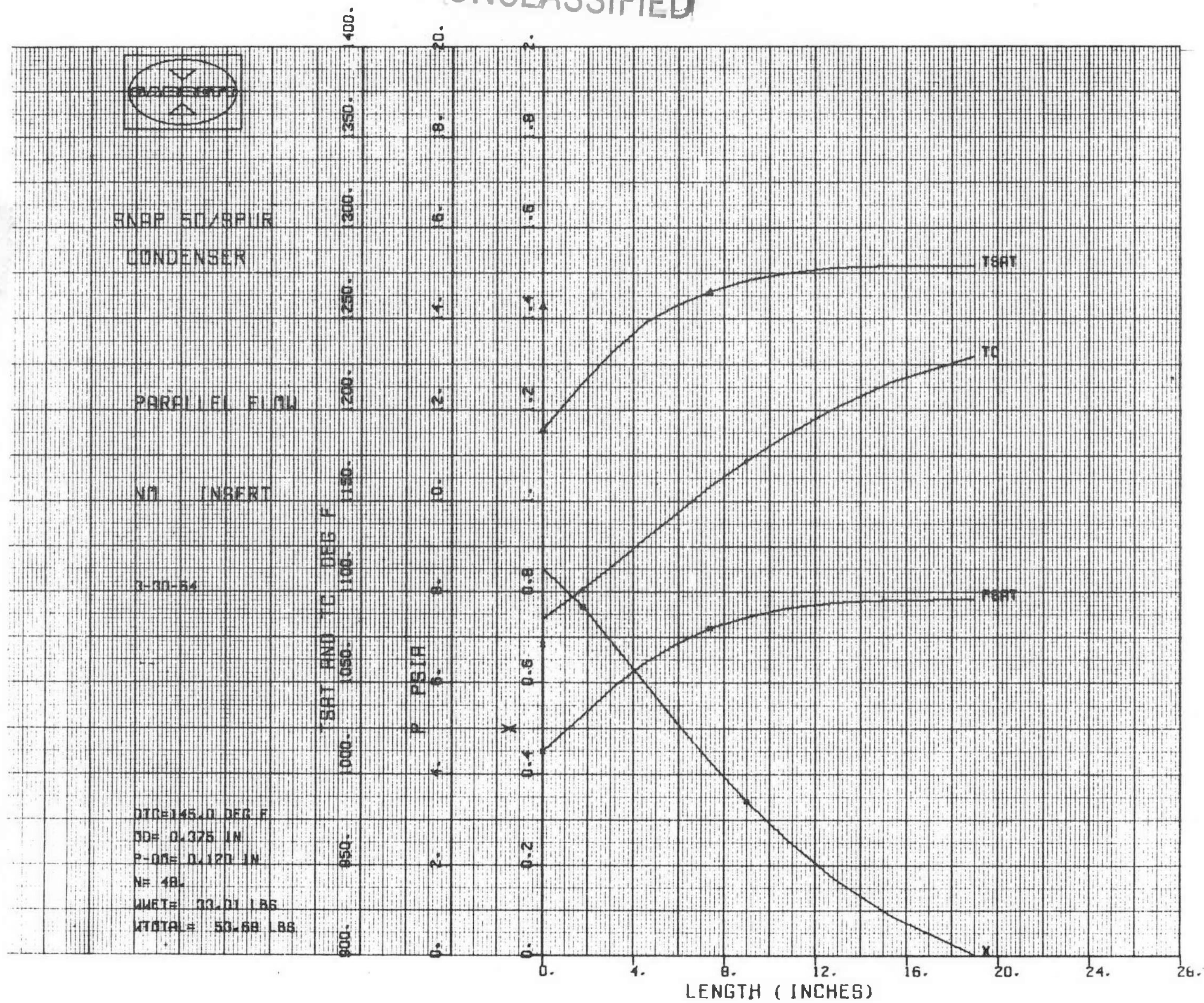

CONDENSER COMPUTER ANALYSIS PRINT-OUT
SHOWING REASON FOR PARALLEL FLOW ARRANGEMENT

SNAP 50/SPUR CONDENSER PARAMETRIC STUDY

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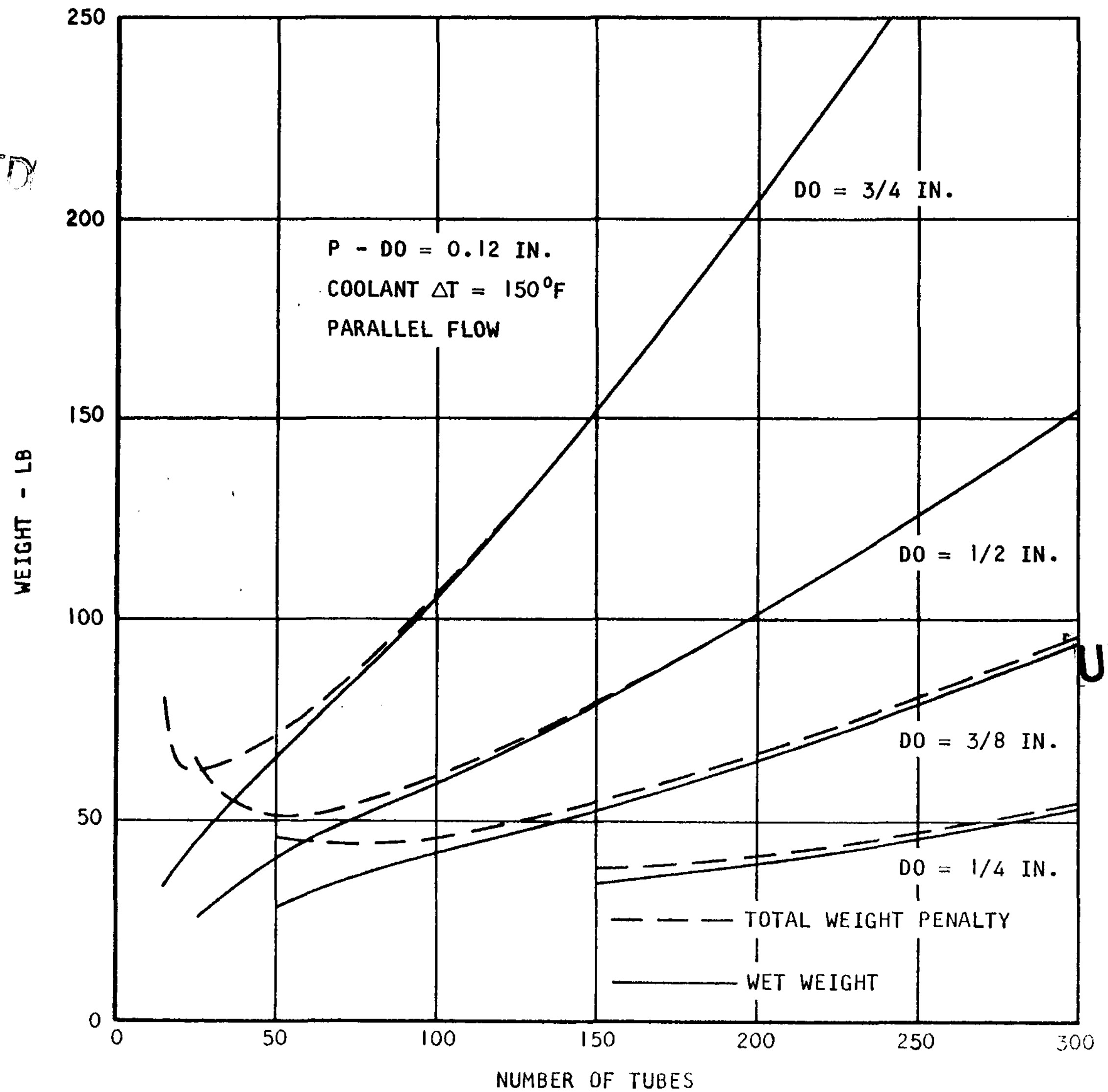

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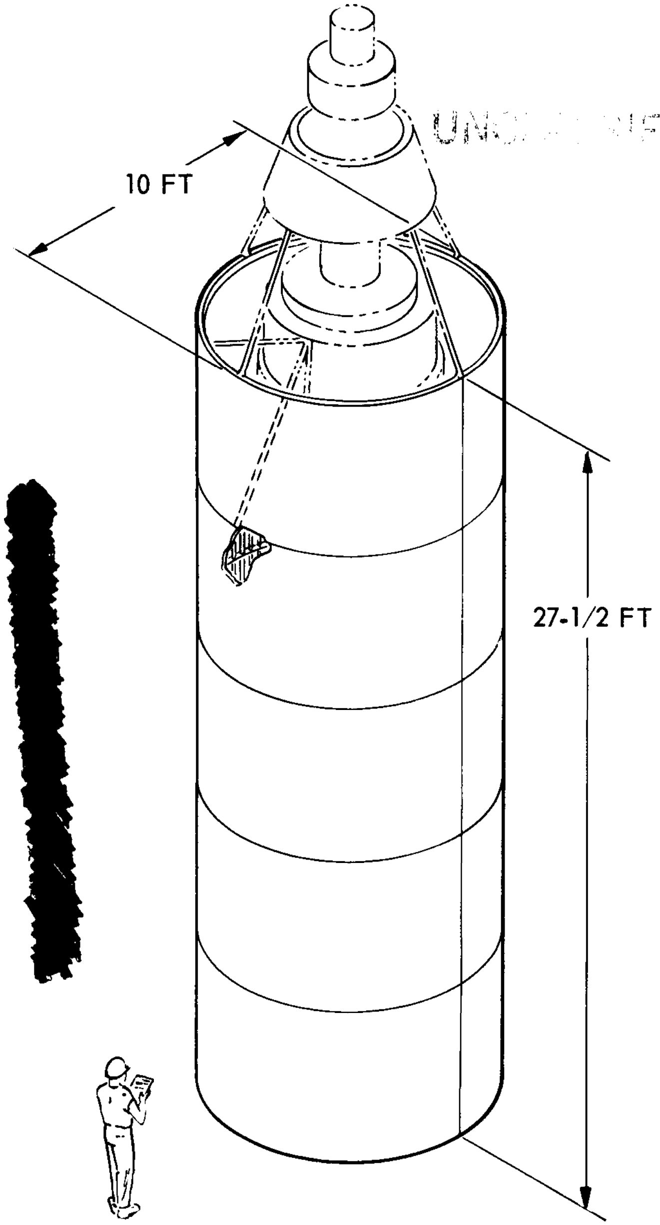
SNAP 50/SPUR RADIATOR PROGRAM



MS-752-5

00182

PURPOSE:

- PROVIDE WEIGHT, PERFORMANCE, AND COMPARISON DATA FOR POWER SYSTEM OPTIMIZATION

PROBLEM AREAS:

- SELECTION OF TUBE, FIN, ARMOR MATERIALS
- STRUCTURAL DESIGN
- RELIABILITY

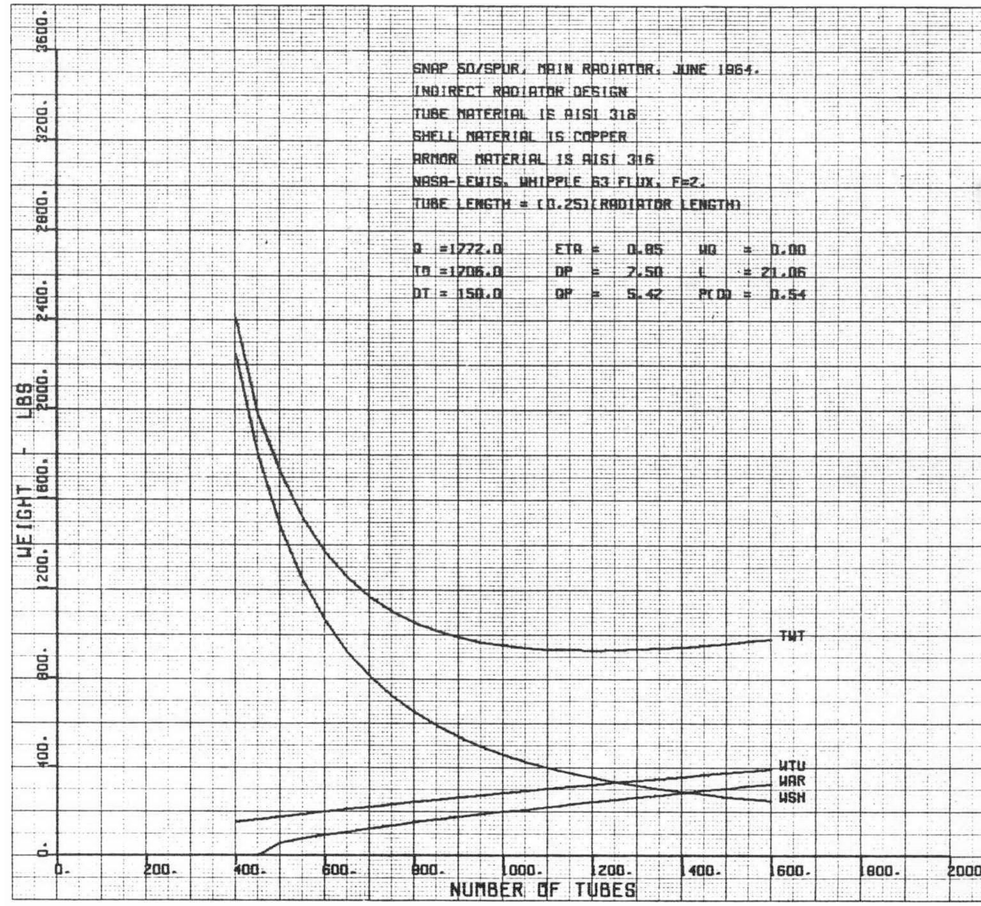
METHODS:

- ANALYSIS
- DESIGN STUDIES
- SMALL-SCALE MATERIAL TESTS
- ORNL BIMETAL LOOP TESTS

The indirect-cycle radiator design presently envisioned is a cylindrical tube-and-fin radiator with tubes parallel to the cylinder axis. It is divided longitudinally into sections, to permit optimization of tube length and segmentation of NaK loops. Radiator design studies are aided by use of digital computer programs which evaluates weight as a function of configuration, pressure drop, temperatures, fin effectiveness, and meteoroid protection requirements. Programs are in use for both liquid and direct-condensing radiators. Results, permitting selection of optimum-weight designs, are illustrated in the curve, which plots tube weight (WTU), armor weight (WAR), and fin/shell weight (WSH), plus total weight (TWT versus number of tubes.

SNAP 50/SPUR radiator studies are primarily concerned with weight optimization, material evaluation, and structural design. The Pratt and Whitney (East Hartford)/NASA program is relied upon for emissive coating development, and the NASA/Lewis and other programs are used for meteoroid penetration design data.

Radiator design is discussed in detail in AiResearch Report L-9443, "Radiators for SNAP 50/SPUR," from which the data following is excerpted.



B-721

Indirect Radiator Analysis, Reference Conditions, Tube Length Equal to One-Quarter Radiator Length

RADIATOR MATERIAL EVALUATION

00184

TABLE

POSSIBLE TUBE-FIN-ARMOR COMBINATIONS

Tube-Armor Combinations	Fin: Be	Fin: Cb	Fin: 316L- or 318-clad Copper	Fin: L-605-clad Copper	Fin: A-286-clad Copper	Fin: Inconel 718-clad Copper
L-605 (Haynes 25)	C (1,2,3)	C (1,2,3)	B (3)	B (3)	B (3,4,6)	B (3,6)
Cb-1Zr	C (1,2,3)	C (1,2)	C (1,2,3)	C (1,2,3,6)	C (1,2,3,4,6)	C (1,2,3,6)
316L or 318SS / 316 or 318SS	C (1,3)	C (1,2,3)	A	C (3,6)	C (4,6)	C (3,6)
316L or 318SS / A-286	C (1,3,4)	C (1,2,3,4)	A (4)	C (3,4,6)	[illegible]	C [illegible]
Cb- or V-lined 316L or 318 / 316 or 318SS	C (1,3,5)	C (1,2,3,5)	A (5)	C (3,5,6)	C (4,5,6)	[illegible]
Cb- or V-lined 316L or 318 / A-286	C (1,3,4,5)	C (1,2,3,4,5)	A (4,5)	C (3,4,5)	B (4,5)	C (3,4,5,6)
Cb- or V-lined A-286 / A-286	C (1,3,4,5)	C (1,2,3,4,5)	A (4,5)	C (3,4,5)	B (4,5)	C (3,4,5,6)
Cb- or V-lined Inconel 718 / Inconel 718	C (1,3,5)	C (1,2,3,5)	B (3,5)	C (3,5)	B (3,4,5)	B (3,5)
Cb- or V-lined Be / Be	C (1,5)	C (1,2,3,5)	C (1,3,5)	C (1,3,5,6)	C (1,3,4,5)	C (1,3,5,6)

LEGEND

A. Principal candidate materials for SNAP 50/SPUR Radiator.

B. Alternative choices.

C. Not considered candidates at present time for reasons noted.

(1) Fabrication Problem -- Adequate brazing alloys not available.

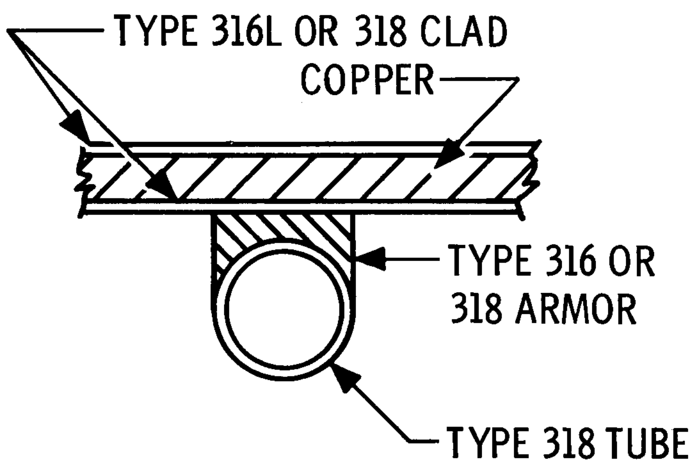
(2) Fabrication and Testing Problem -- Must be protected against contamination during fabrication and testing by use of high vacuum or very pure inert atmosphere.

(3) Fabrication and Design Problem -- Differences in thermal expansion coefficients of components make feasibility of combination doubtful.

(4) Fabrication Problem -- Rapid cooling from 1650°F/1800°F is required for A-286 to solution-anneal the material. Adequate fixturing is necessary to avoid warpage.

(5) Care must be taken to avoid contamination of liner during brazing, heat-treating (if required), and testing.

(6) Advantage of high-strength cladding combined with lower strength tube-armor materials is questionable.



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● BASIS OF SELECTION

- MINIMUM DEVELOPMENT COST & RISK
- SIMPLEST FABRICATION
- SIMPLEST THERMAL STRESS PROBLEMS
- WEIGHT ONLY ABOUT 1 lb/KW (e) HIGHER THAN Be SYSTEM

MS-752-6

A large number of combinations of tube, fin, and armor materials have been considered. The present design selection has been made as a best evaluation of incomplete data, and is therefore tentative. Much additional data on material properties, fabrication techniques, and meteoroid damage mechanisms is needed for final selection.

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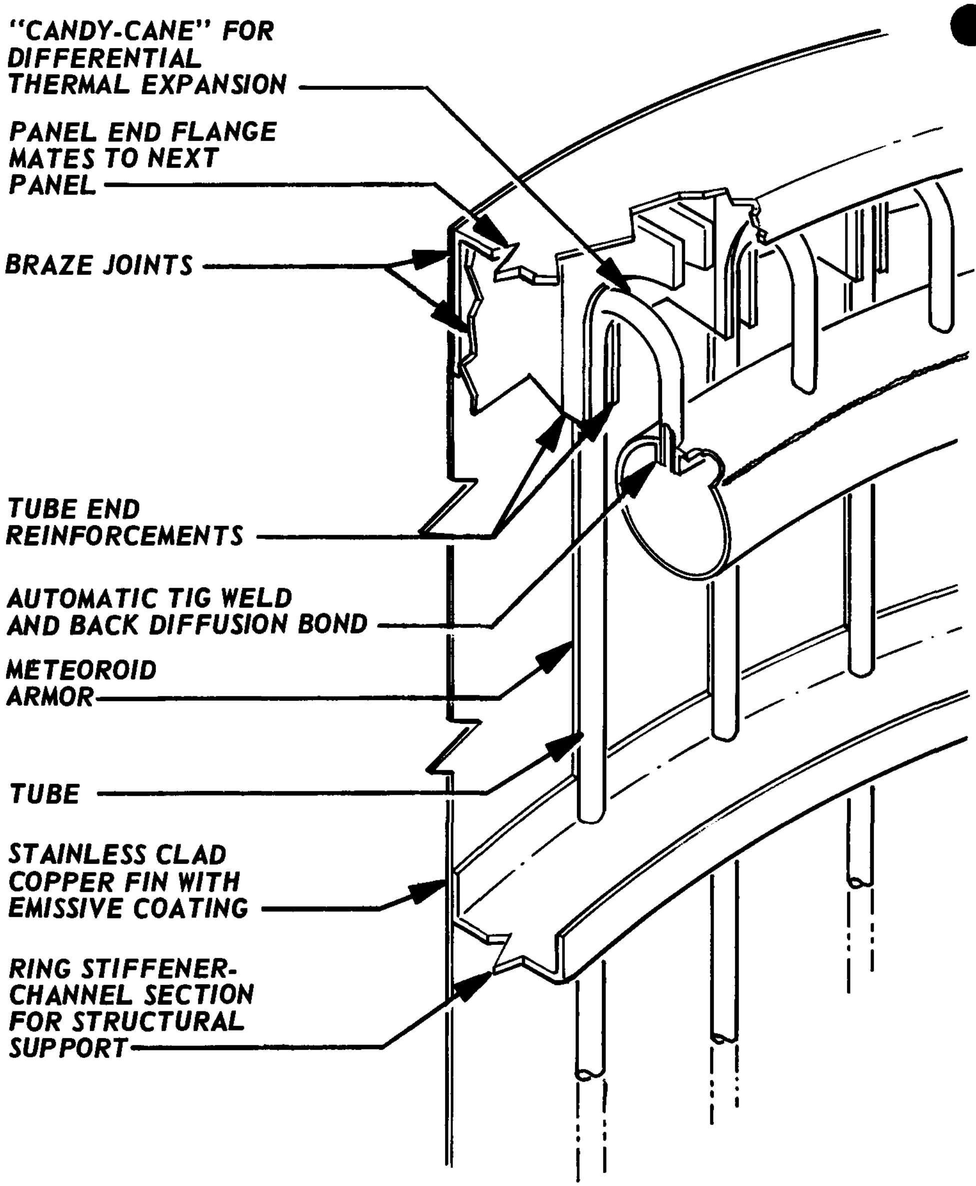
GARRETT

STRUCTURAL DESIGN



- ***RADIATOR AS MAIN SYSTEM STRUCTURE FOR LAUNCH LOADS:***

	NON-STRUCTURAL RADIATOR	STRUCTURAL RADIATOR
RADIATOR	1600	1800
SEPARATE TRUSS STRUCTURE	700-1400	—
TOTAL WEIGHT, LB	2300-3000 LB	1800 LB

PROGRAM:

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- ANALYSIS
- SMALL-SCALE PANEL TEST
- COPPER/STAINLESS SANDWICH STRENGTH TEST

The radiator is a large and complex structure. The great apparent weight advantage of using it as the basic system structure provides a strong incentive to investigate this possibility. In addition, a number of detailed structural design problems must be solved even to assure structural integrity of the radiator when subjected to its own weight and thermal gradients.

Small-scale tests are now under way to evaluate the effects of thermal cycling or of plugged-tube conditions on the structural integrity of the radiator. Other tests are aimed at evaluating the load capability and resistance to buckling of the tube-armor-shell matrix. Mechanical properties of the stainless-clad copper shall material are being experimentally determined, at launch temperatures and at operating temperatures.

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RELIABILITY EVALUATION

WEIGHTS OF DIRECT VS. INDIRECT SYSTEMS FOR GIVEN RELIABILITY

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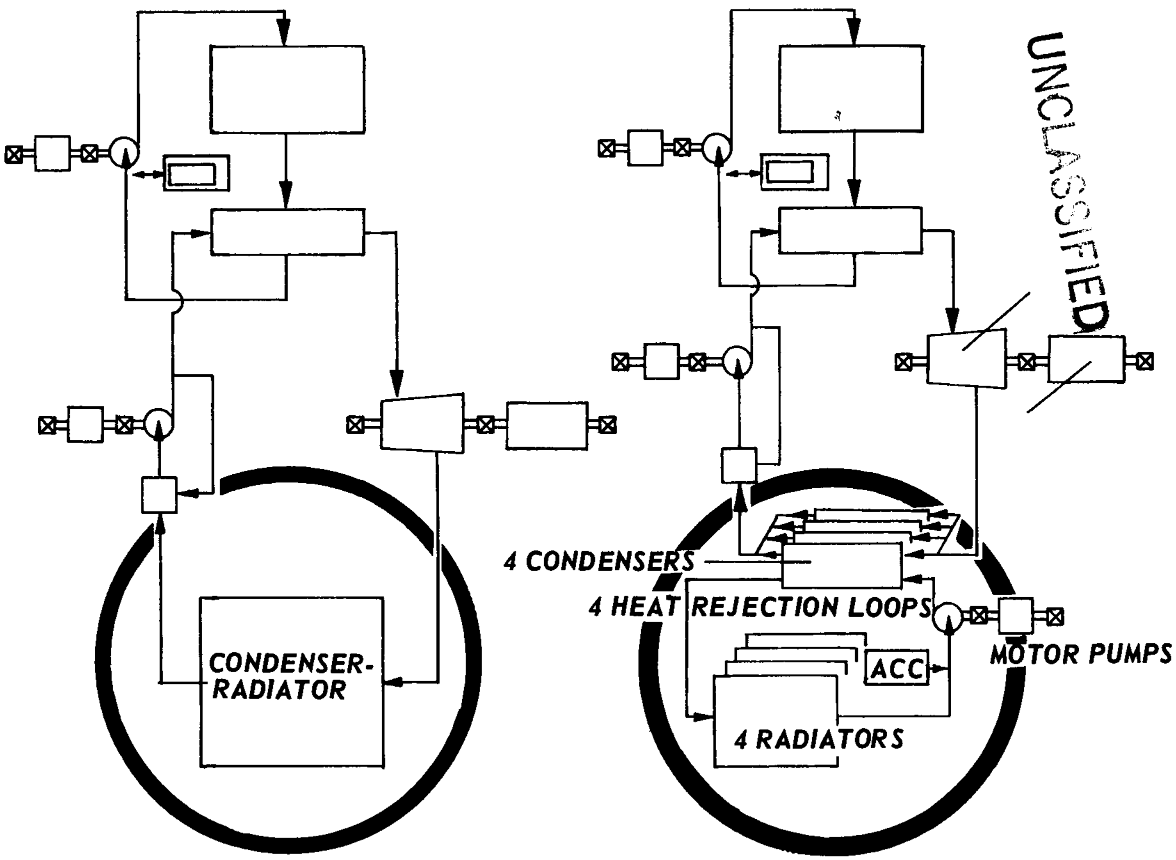
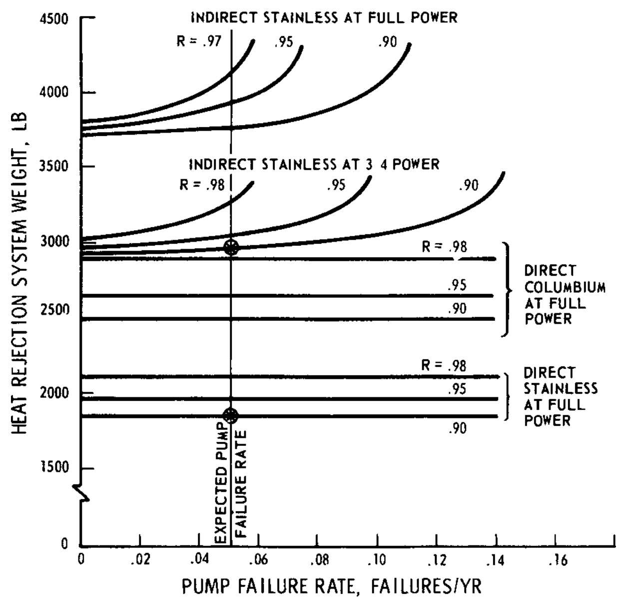


DIRECT SYSTEM

INDIRECT SYSTEM



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Current evaluations show weight savings of 1,100 to 1,800 lbs. for a 300-kw(E) system by using direct-condensing radiators instead of indirect cycles, and indicate that, if material compatibility problems prevent use of a stainless radiator with a Cb - 1% Zr system, a Cb - 1% Zr direct-condensing radiator would still offer substantial weight advantage over an indirect system. This illustrates another major area which must be resolved before the radiator weight, and thus the system weight, can be predicted accurately.

The comparison of direct and indirect systems is discussed more fully in the answer to "Topics of Interest - Question 4c" elsewhere in this report.

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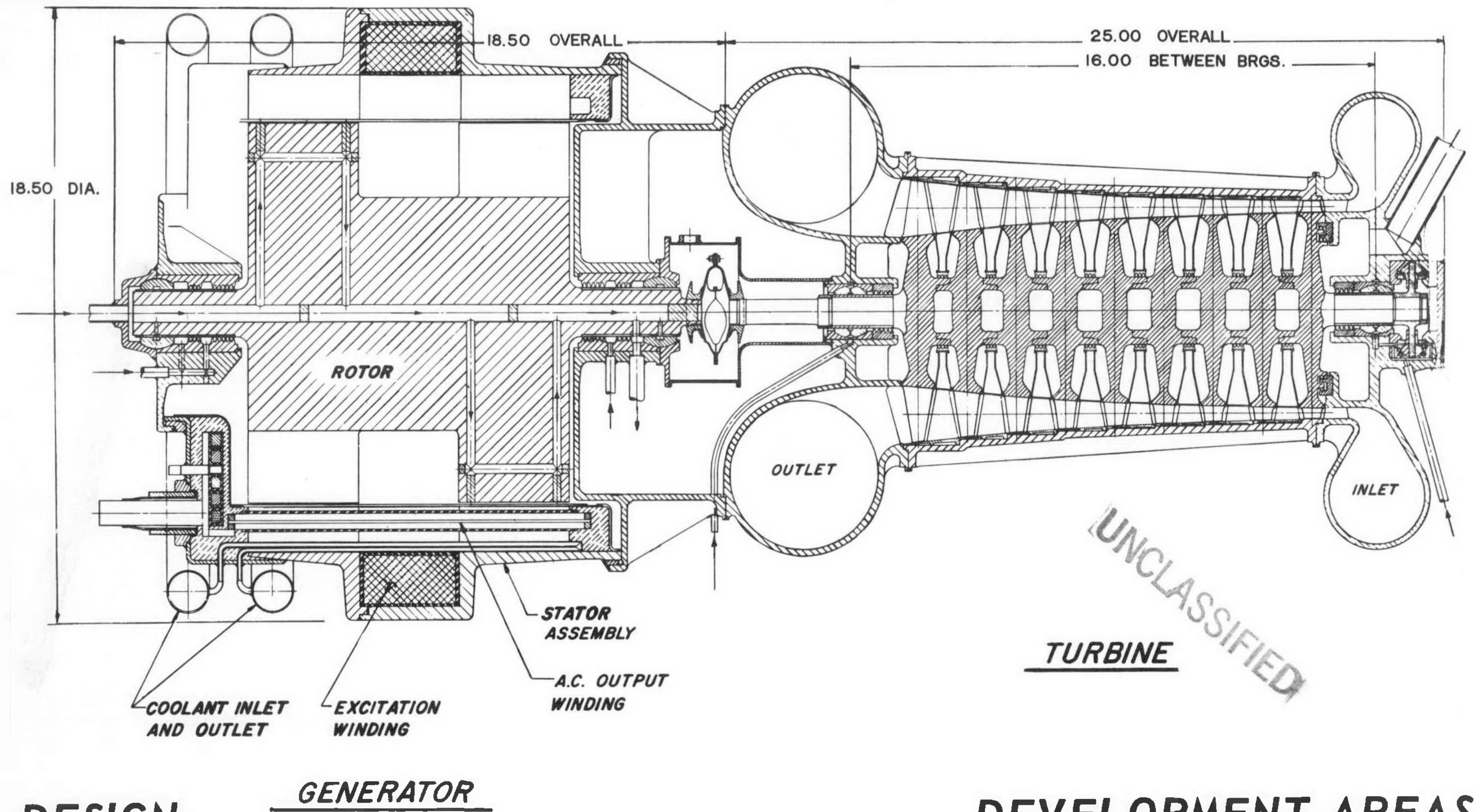
00189

SNAP 50/SPUR - TURBO GENERATOR



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DESIGN

- 300KW-3200CPS-115/208V-.75pf
- REFRACTORY METAL CONSTRUCTION
- 10,000 HR LIFE
- POTASSIUM GENERATOR COOLANT
- POTASSIUM LUBRICATED BEARINGS
- SATURATED TURBINE INLET

DEVELOPMENT AREAS

- BEARINGS & SEALS
- TURBINE
- GENERATOR

An axial turbine directly drives a radial gap inductor generator at 24,000 rpm. Both rotors are "straddle supported" by potassium lubricated bearings.

Saturated potassium vapor enters the smaller turbine volute on the right, and discharges from the larger volute at approximately 88 percent quality. The number of turbine stages is dictated by the allowable blade speed which is limited by potential liquid droplet impingement erosion of the last stages. (The design shown is for a blade speed of 800 ft/sec.) The candidate rotor material is the molybdenum base alloy, TZM, based on considerations of strength, fabricability and availability. The columbium alloy (Cb-1Zr) was chosen for the turbine housing on the basis of joining requirements to the columbium-alloy inlet and outlet vapor headers, formability, and moderate strength requirements. The use of interstage liquid extraction techniques, or reheat, can improve turbine efficiency and may slightly reduce turbine erosion. The extent to which condensed potassium can be removed from the turbine will be determined in development tests. Some improvements in turbine performance may be forthcoming if these techniques can be developed for incorporation into the system.

The radial gap homopolar inductor generator was chosen since it requires no brushes; the rotor is one piece and symmetrical with low stresses; its efficiency is inherently good; and its electromagnetic unbalance is least affected by rotor-stator misalignment. Liquid potassium cooling channels are provided in the stator and rotor. Seals are provided at each end of the generator rotor in order to prevent liquid potassium from entering the rotor cavity. A ceramic "bore seal" is incorporated in the stator assembly to protect the stator insulation against attack from potassium vapor within the rotor cavity.

The development work on bearings, turbine, and generator is described on the following pages.

SNAP 50/SPUR - TURBO GENERATOR BEARINGS

DESIGN REQUIREMENTS

- POTASSIUM LUBRICANT
  (TEMP. PRESS. PER SYSTEM)
- UNBALANCED LOADS, GROUND & ZERO "G"
- MULTIPLE STARTUP & SHUTDOWN
- POWERPLANT STRUCTURE
  (RESILIENT MOUNT)
- OTHER

MAJOR PROBLEMS

- STABILITY (BEARING/ROTOR SYSTEM)
  CAUSE = HIGH SPEED, LOW VISCOSITY, LIGHT LOAD
- STARTUP & SHUTDOWN (MATERIAL COMPATIBILITY)
  CAUSE = BOUNDARY LUBRICATION
- WEAR (EXTENDED OPERATION)
  CAUSE = EROSION, CORROSION, & FATIGUE

DEVELOPMENT PROGRAM

- ANALYTICAL PROGRAM WITH CONVENTIONAL FLUID TESTING
  (BEARING/ROTOR SYSTEM STABILITY)
- MATERIAL COMPATIBILITY (POTASSIUM)
- STABILITY VERIFICATION IN POTASSIUM
- ENDURANCE TEST IN POTASSIUM (WEAR)

MS-744-13

The key design requirements and major problems are indicated.

The problem requiring solution initially is that of journal stability, a problem common to all high speed rotating machinery. Stability problems are aggravated by the low viscosity of the potassium (i.e., low film spring rate), and the low journal bearing loads that may occur during ground, and particularly zero "g" operation, with balanced rotors.

Multiple restarts will occur during the development program, which introduces a requirement for this capability. Since system pumps can be operated to provide bearing lubrication during starting, this is primarily a problem of material compatibility under conditions of boundary lubrication.

Erosion/corrosion and fatigue causing general bearing wear arise from the scrubbing action of the fluid velocity, chemical attack (mass transfer) of the potassium, and the oscillatory load arising from the rotating "out of balance" load vector.

The four major parts of the bearing development program as listed on the figure are described on subsequent pages.

BEARINGS - ANALYTICAL PROGRAM

- ***BEARING/ROTOR SYSTEM ORBITAL COMPUTER PROGRAM***

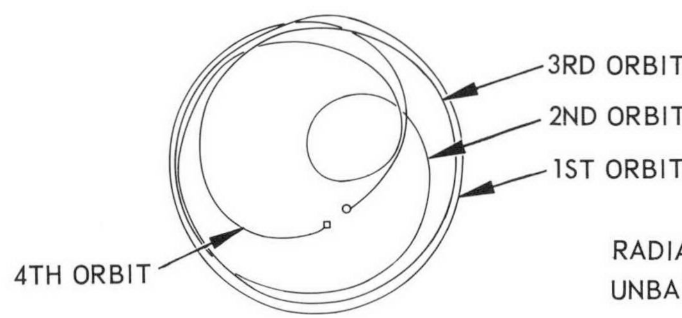
  -JOURNAL BEARING SUBROUTINE (LAMINAR/TURBULENT FILM; SHAFT MOTION; MISALIGNMENT)

  CYLINDRICAL
NONCIRCULAR & SHOE
FLOATING SLEEVE
HYBRID
RESILIENT MOUNT

  L/D; CLEARANCE; LUB SUPPLY GEOMETRY; ETC.

  -ROTOR/BEARING SYSTEM DYNAMICS PROGRAM (ROTOR MASS, UNBALANCE, INERTIA, BRG. FORCES, ETC.

- ***RESULTS***



RADIAL CLEARANCE 0.0015 INCH
UNBALANCE 0.0006 C.G. ECCENTRICITY

- ***CONVENTIONAL FLUID TESTING*** (WATER/OIL OF BEARING & BEARING/ROTOR SYSTEMS)

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In the bearing analytical program an extensive computer program is being utilized to evaluate the stability of candidate journal bearings and of bearing/rotor systems. With this program, the interaction between the motion of the rotor and the fluid pressure forces generated by the film due to the shaft motion can be studied for a number of journal bearing types such as those shown on this figure. A typical result from this program, presented above, shows the path followed by the centerline of an unbalanced rotor over four orbits when supported by a cylindrical bearing. The case shown has marginal stability. A rerun of these conditions, with increasing radial clearance, indicates instability of the system, with the journal contacting the stationary housing (failure).

Testing of bearings and of bearing/rotor systems with conventional fluids as lubricants is utilized in conjunction with the analytical computer program to evaluate bearing types and configurations as a screening procedure prior to actual potassium testing.

To date some limited bearing/rotor system stability tests have been run under NASA support (General Electric) with water which indicate that the two axial groove cylindrical, noncircular and shoe type bearings merit continual detailed analysis in view of their ability to operate at high speeds and moderate loads without destructive half frequency wheel. Detailed analysis, with the AiResearch dynamics computer program, is underway on these and other types to establish stability margins. Verification screening tests on selected types and configurations will be run in water.

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BEARING DEVELOPMENT-POTASSIUM-MAJOR COMPONENT TESTS

Materials		Test Hrs	Load Lbs	Temp °F	Dia inch	Length inch	Speed rpm	Radial Clearance inch	Brg Type	Facility
Journal	Bearing									
• K-94 WC	Mo-0.5 Ti	250	25	600	1.0	1.0	24,000	.0012	HYDRODYNAMIC	AiResearch
• K-94 WC	Chrome Plate-316S/S	4085	25	800	3.5	3.0	4,500	.0025		ORNL
• LW-1 WC Flame Sprayed Coating	Carboloy 883	710	15-25	600-950	1.0	1.0	24,000	.0016		Rocketdyne

START/STOP BEARING TESTS CONDUCTED BY ORNL

K-94 WC Journal and Chrome Plated 316S/S Bearing

Bearing Load - lbs	Number of Start/Stops
2.2	150
11.0	100
22.0	100
33.0	100

00106

The above major successful bearing component development tests, conducted in the turbulent operating regime, have proved the general overall feasibility of potassium lubricated bearing for use in the SNAP 50/SPUR turbogenerator package. The start/stop bearing test results indicate the feasibility of multiple ground test starts and stops in a one "g" environment. They also indicate that successful operation of bearings during boundary lubrication conditions, which might occur during multiple transient changes of temperature (structure and lubricant) and over the speed range, could be obtained with the proper selection of materials.

[redacted]

[redacted]

BEARINGS - POTASSIUM TESTING (DYNAMIC JOURNAL & THRUST)

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- **VERIFICATION OF STABILITY OF SELECTED TYPES (SIMULATED ROTORS)**
- **WEAR WITH EXTENDED OPERATION (EROSION/CORROSION & FATIGUE)**

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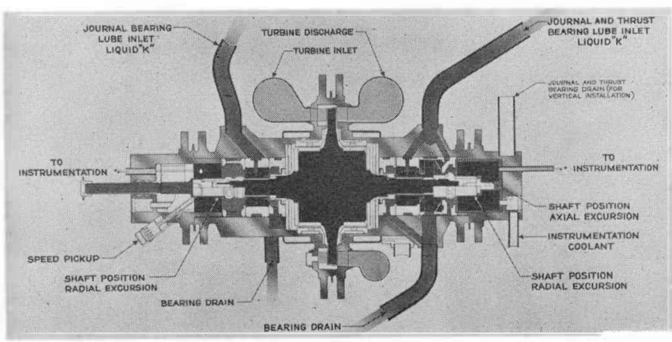
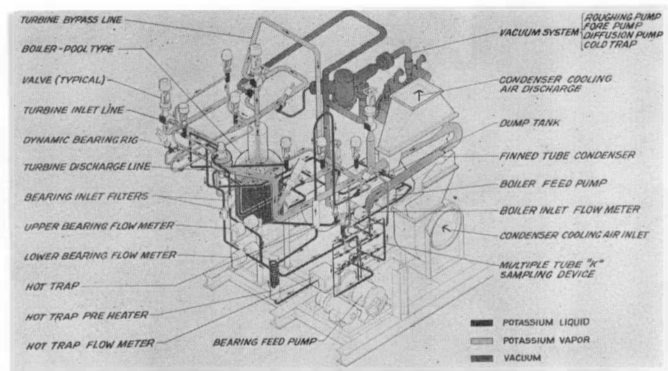


TURBODYNAMIC RIG



TURBODYNAMIC LOOP

Stability tests of journal bearing designs, selected from the analytical and conventional fluid test program described previously, will be tested with potassium utilizing the test system shown above.

The test rig consists of a potassium vapor turbine supported by two journal and one thrust bearing. The rotor simulates the dynamics of the larger turbine rotor system. The special AiResearch designed bearing instrumentation noted on the figure, has been developed for measuring dynamic shaft motions in a liquid potassium environment. The development of this instrumentation represents an important feature of the test system in the investigation of bearing stability characteristics in a dynamic system utilizing potassium bearing. Both radial and horizontal shaft excursions can be measured within an accuracy of 50 millionths of an inch.

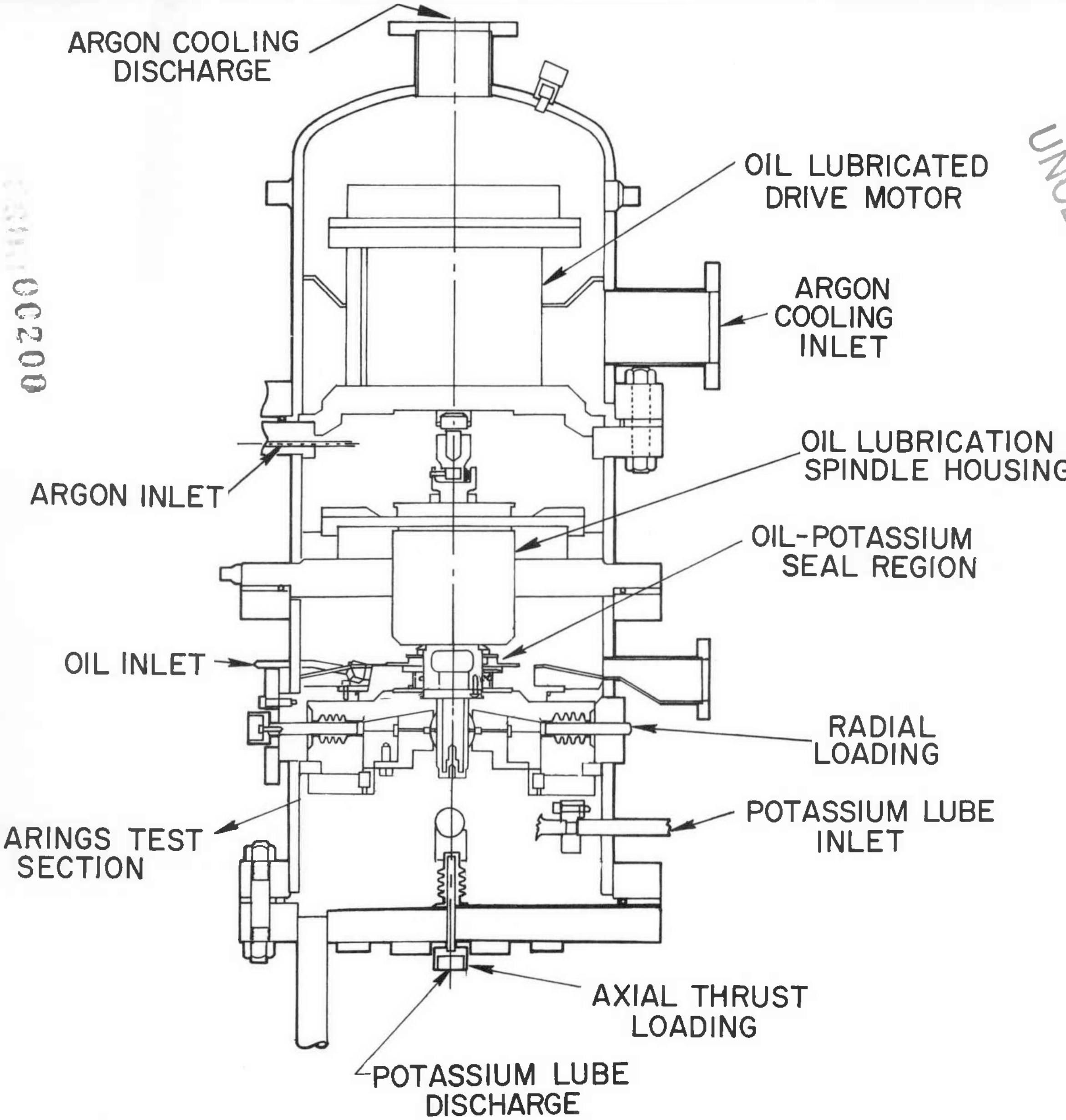
The loop supplying potassium vapor to the turbine, liquid to the bearings, etc. is shown on the adjacent figure.

This test system will also be utilized to evaluate bearing designs and selected materials for general wear properties under oscillatory loads (fatigue), and erosion/corrosion with extended endurance testing.

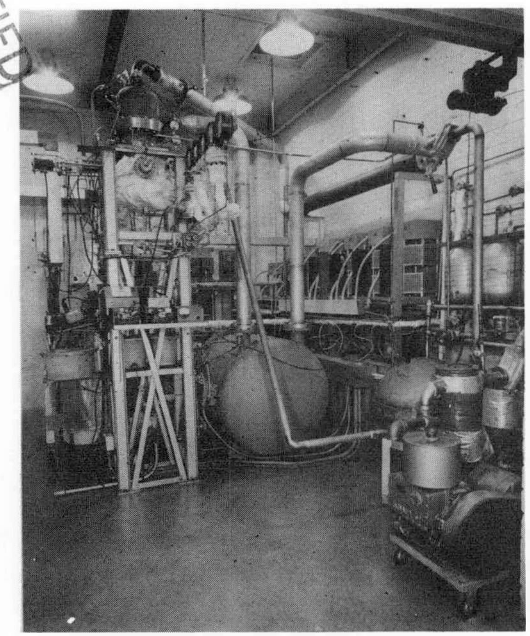
A dynamic bearing/rotor system stability test of 5 hours duration has just been completed utilizing the above facilities. A plain spherically seated circumferential grooved journal bearing and a fixed tapered pad spherically seated thrust bearing were run (5 hours) to 20,000 rpm with 600°F potassium when a slight rub between the turbine wheel and nozzle ring caused termination of the test. Design modification have been completed and the test is being resumed.

BEARINGS- UNIDIRECTIONAL JOURNAL & THRUST (POTASSIUM)

[redacted]



TEST RIG



TEST LOOP

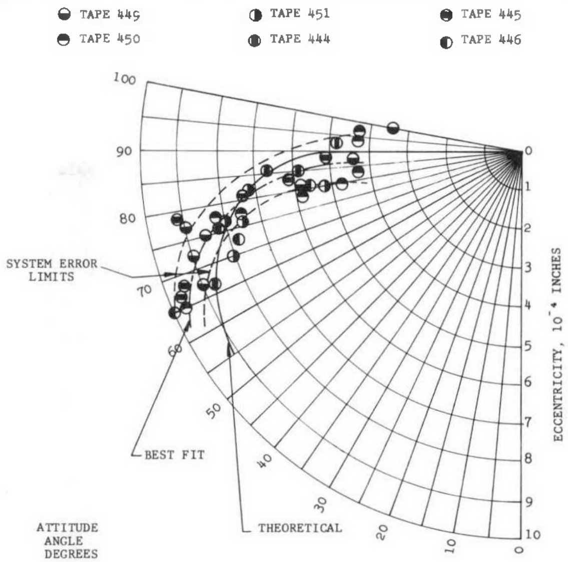
[redacted]

MS-744-16

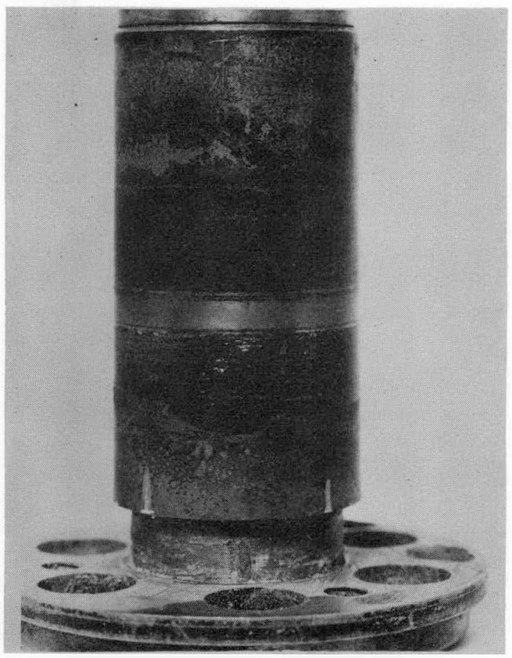
The test system shown above is utilized for testing unidirectionally loaded potassium journal and thrust bearings. Tests can be conducted in this system at speeds above 30,000 rpm, potassium lubricant temperatures in excess of 1200°F, and with loads exceeding 200 lbs, thus covering a range of variables anticipated in the final turbo-generator. Instrumentation is provided for measuring all variables including lubricant film thickness, attitude angle, etc.

Results of tests from this sytem are represented on the following figure.

BEARING - UNIDIRECTIONAL JOURNAL TESTS (POTASSIUM)



ATTITUDE ANGLE VERSUS ECCENTRICITY



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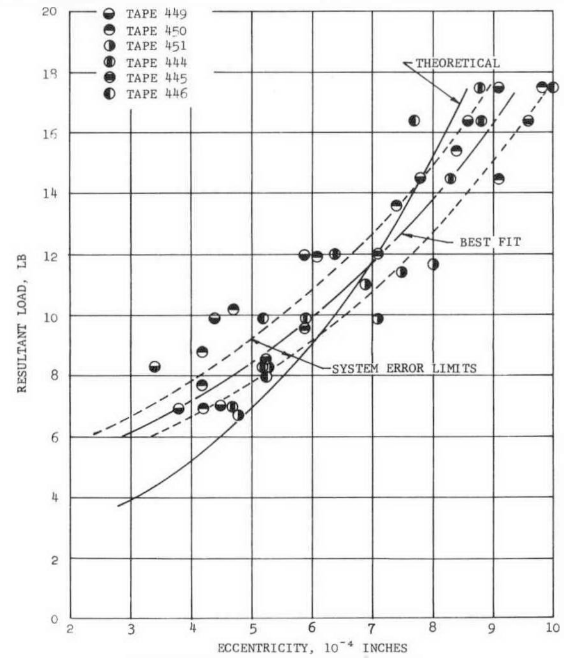
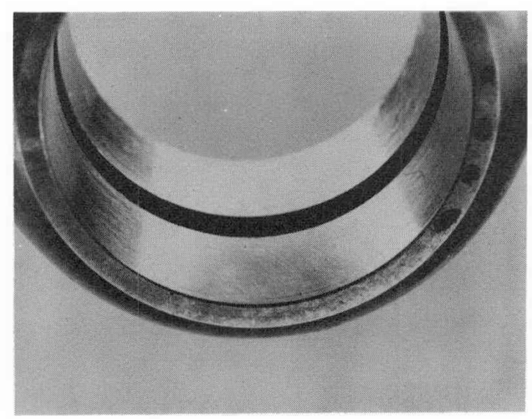


JOURNAL ECCENTRICITY VERSUS LOAD



This figure shows the results of tests of a unidirectionally-loaded cylindrical journal bearing, conducted in the test system described on the previous figure. The bearing attitude angle and eccentricity are shown for a speed of 24,000 rpm and variable load, with a potassium lubricant temperature of 600°F. Bearing (Mo-1/2%Ti) and journal (WC) surface conditions are shown after a 300 hour test run.

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[redacted]

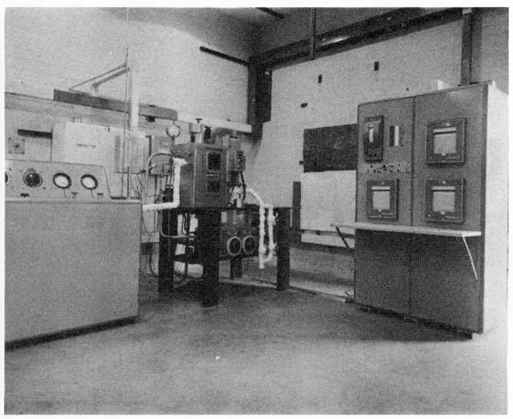
[redacted]

BEARING - MATERIAL COMPATIBILITY

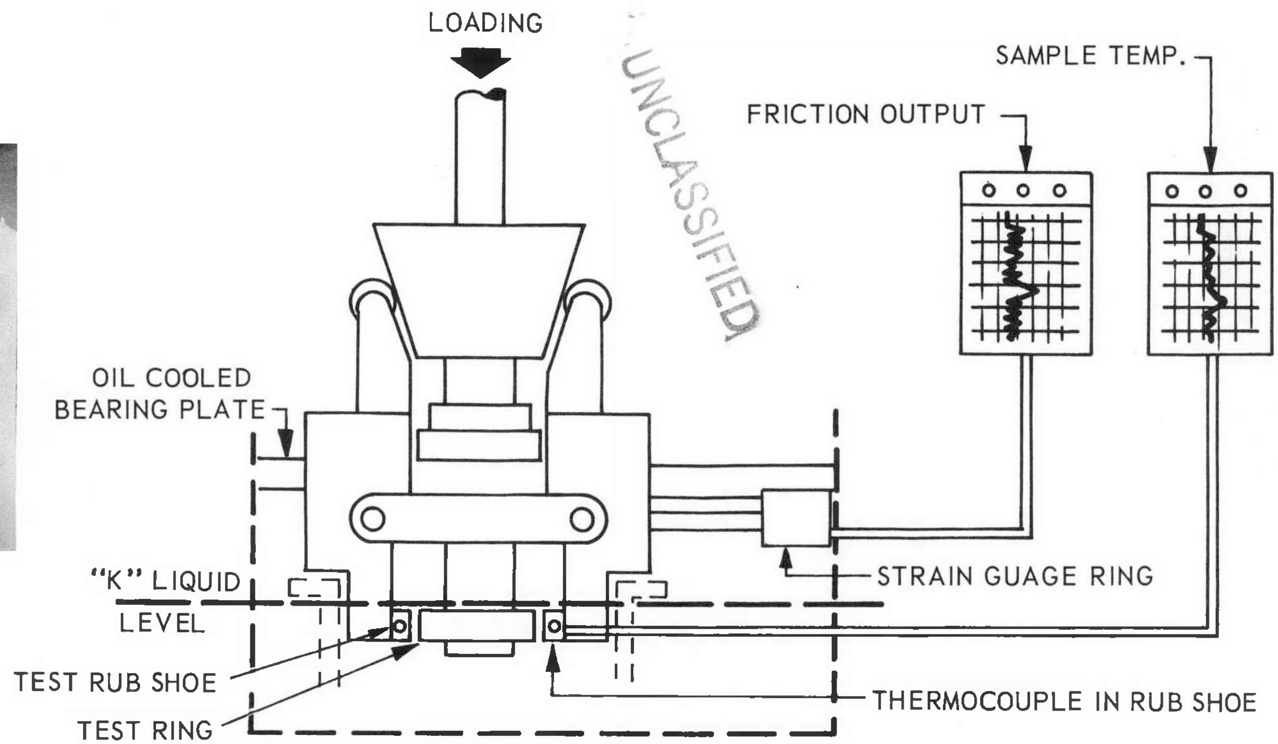
- **_STARTUP & SHUTDOWN (MATERIAL COMPATIBILITY)_**
  - CAPSULE TESTS (PRELIMINARY SELECTION, COMP. WITH "K")
  - HOHMAN A-6 FRICTION & WEAR TESTER
  - OTHER SOURCES - BMI, CANEL, NASA (GE), ORNL, ROCKETDYNE

00204

[redacted]



TEST FACILITY



TEST RIG

[redacted]

The compatibility of bearing materials under conditions of boundary lubrication, simulating startup and shutdown or transient conditions, will be conducted in the test system shown above with potassium.

Selected combination of materials for the bearing are tested for engine, galling, friction, etc., in potassium over a range of speeds, loads, and temperatures approximating conditions in the turbogenerator.

A considerable number of friction and wear studies have been conducted by KAPL, B.M.I., and Canel in sodium-potassium alloy (NaK) and lithium but no direct data on what are considered to be applicable materials for potassium has been generated. The data available on the other liquid metals from the above sources has however been used to great advantage for initial bearing material selections and is establishing the base-line for future testing in the above facility with potassium.

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BEARING MATERIAL COMPATIBILITY - FRICTION AND WEAR BOUNDARY LUBRICATION TESTS

[redacted]

Materials		Stator Specimen	Load lbs	Rotor Speed fps	Temp °F	Test Fluid	Rating	Material Composition	Facility
Stator	Rotor								
Carboloy 44A	Carboloy 44A	5/16 dia	34	7.4	850	NaK	Excellent	WC + Co Binder	KAPL
Carboloy X3040D	Carboloy X3040D	5/16 dia	34	7.4	850	NaK	Excellent	WC + Co Binder	KAPL
Carboloy 55A	Kentanium 138	5/16 dia	34	7.4	850	NaK	Good	WC + Co Binder TiC + Co Binder	KAPL
Kentanium 138A	Carboloy 779	5/16 dia	34	7.4	850	NaK	Excellent	TiC + Co Binder WC + Co Binder	KAPL
Kentanium	Carboloy 779	5/16 dia	34	7.4	850	NaK	Excellent	TiC + Fe Binder	KAPL
Kentanium	Kentanium 138A	5/16 dia	34	7.4	850	NaK	Fair	See above	KAPL
Aluminum Bronze	Stellite 98M2	5/16 dia	34	7.4	850	NaK	Fair	Non-carbides	KAPL
Stellite 98M2	Stellite 98M2	5/16 dia	34	7.4	850	NaK	Poor	Non-carbides	KAPL
Kentanium 138A	Kentanium 138	5/16 dia	34	7.4	850	NaK	Very poor	See above	KAPL
Kennametal	Kennametal	-	20	-	1000	Lithium	Good	WC-Mo-CbC	CANEL
Carboloy 78	Carboloy 78	-	20	-	1000	Lithium	Good	WC + Co +(TaCb)C + TiC	CANEL
Kennametal K-96	Kennametal K-96	-	20	-	1000	Lithium	Good	WC + Co +(TaCb)C	CANEL

[redacted]

Excellent: Neglible wear, no material transfer, polish both specimens.

Good: Smooth wear fixed specimen, no material transfer polish rotating specimen.

Fair: Smooth wear both specimens, minor material transfer.

Poor: Excessive wear, surface roughening, material transfer high friction.

Very Poor: Test not completed due to excessive wear, friction, material roughening and transfer.

The above referenced data from a number of material compatibility, friction and wear tests run on metal, cermets, ceramics, etc., has narrowed the range of applicable materials for possible use in the SNAP 50/SPUR potassium lubricated bearings. Only limited friction and wear tests in potassium, on the existing potential materials need to be conducted to determine which combination is the best. Also some future testing on new materials will be required to take advantage of advanced developments.

The Carboloy 44A material shown has the same composition as the Kennemetal K-94 which is currently being used in the journal bearing designs. Also Carboloy 44A is being used in a second vintage pivoted pad equa-film thrust bearing currently being fabricated. The Carboloy 44A is being used for the pivots and pads.

It is evident from the above tests that the type and percentage of bender used will undoubtedly be an important factor in the design of 10,000-hour bearings. This factor will be investigated in future AiResearch friction and wear studies.

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00207

BEARING MATERIAL COMPATIBILITY - SELF WELDING STATIC TESTS

Materials Combination		Contact Area-Load psi	Test Duration hrs	Separating Cycles No.	Separating Force psi	
Top	Bottom					
Tungsten	Tungsten	5000	100	5	All 0	No visible evidence of bonding
Molybdenum	Molybdenum	5000	100	5	0-2300	No visible evidence of bonding
Molybdenum	Tungsten	5000	100	1	100-400	No visible evidence of bonding
Kennametal K-94*	Molybdenum	5000	100	1	100-200	No visible evidence of bonding
Molybdenum	Mo_2C-4 W/o Ni	5000	24	1	0	No visible evidence of bonding
Kennametal K-94	Kennametal 138A	Specimen combination eliminated by 24-hr tests				
Kennametal K-96	Kennametal 162B	Specimen combination eliminated by 24-hr tests				
Kennametal K-162B	Kennametal 162B	Specimen combination eliminated by 24-hr tests				
Molybdenum	Chrome Plated S/S	Specimen combination eliminated by 100-hr tests				
Kennametal K-96	Kennametal K-96	Specimen combination eliminated by 100-hr tests				

NOTE: Tests conducted by Battelle Memorial Institute in Nak at 1500°F.

Dynamic Material Compatibility Tests - 1500°F Nak

Best material combination of 50 metal versus metal, metal versus ceramic, metal versus cermet, cermet versus cermet, and coated materials run at 7-1/2 tps was found to be Mo-0.5 W/o Titanium running against Kennametal K-94 WC-12 W/o Cobalt. Tests were conducted at Battelle Memorial Institute with 170 psi loading.

Another way of helping to establish which materials might be suitable for bearings is to screen by conducting static self-welding tests. Above are some results, of a large number of material combinations (metal, cermets, carbides, ceramics, coatings, etc.) that show which material combinations could be applicable for SNAP 50/SPUR even though they were run in Nak.

The Kennemetal K-94 versus Molybdenum combination was used for the initial selection of SNAP 50/SPUR bearing materials (K-94 and Mo-0.5 Ti) which has been run successfully as discussed earlier. The compatibility with molybdenum was also used when selecting TZM (Mo-0.5 Ti 0.08 Zr) as another candidate bearing material being evaluated in the turbodynamic bearing test rig. TZM has a much lower brittle transition temperature than Mo-0.5 Ti and therefore is of interest.

Also, the friction and wear dynamic studies discussed above, however conducted with flat plates which more closely simulate face seals and thrust bearings, shows that K-94 versus Mo-0.5 Ti was the best combination tested.

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BEARING MATERIAL COMPATIBILITY - CORROSION TEST

Materials	Temp °F	Time hr	Weight Change mg (1)(2)	Facility
Mo-0.5 Ti	1650	500	-0.1	AiResearch
Cb - 1 Zr	1800	500	+0.1	AiResearch
Tungsten	1000	1000	-13.6	AiResearch
Mo-0.5 Ti	1000	1000	+0.4	AiResearch
Kennametal K-94	1000	1000	-5.0	AiResearch
K-94 + Mo-0.5 Ti	1000	1000	-5.5	AiResearch
Al_2O_3	1000	500	+44.6	AiResearch
Flame Sprayed Al_2O_3	1000	500	Coating Failed	AiResearch

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(1) Plus sign denotes weight gain, negative weight loss.

(2) All tests run with liquid potassium

Bearing material compatibility from a corrosion standpoint is another important criteria in the design and development of 10,000 hour bearings. This work conducted on numerous combinations (not all shown) does verify the initial bearing materials selected on the basis of corrosion resistance.

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GARRETT

SNAP 50/SPUR TURBINE

DEVELOPMENT AREAS

00212

- **AERODYNAMIC PERFORMANCE WITH POTASSIUM**
  - -GOOD EFFICIENCY WITH WET STAGES (INTERSTAGE MOISTURE REMOVAL)

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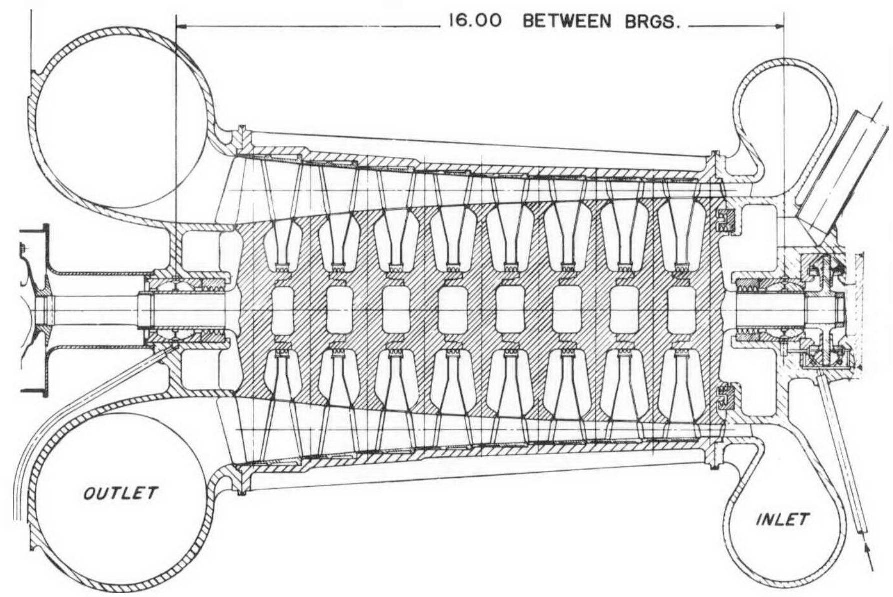
- **BLADE EROSION (DROPLET IMPINGEMENT)**
  - -BLADE MATERIAL

[redacted]

- **ROTOR MATERIAL & CONSTRUCTION**
  - -CREEP, FATIGUE, FABRICATION, COMPATIBILITY WITH "K"

- **CONSTRUCTION OF HOUSING (Cb-1% Zr)**



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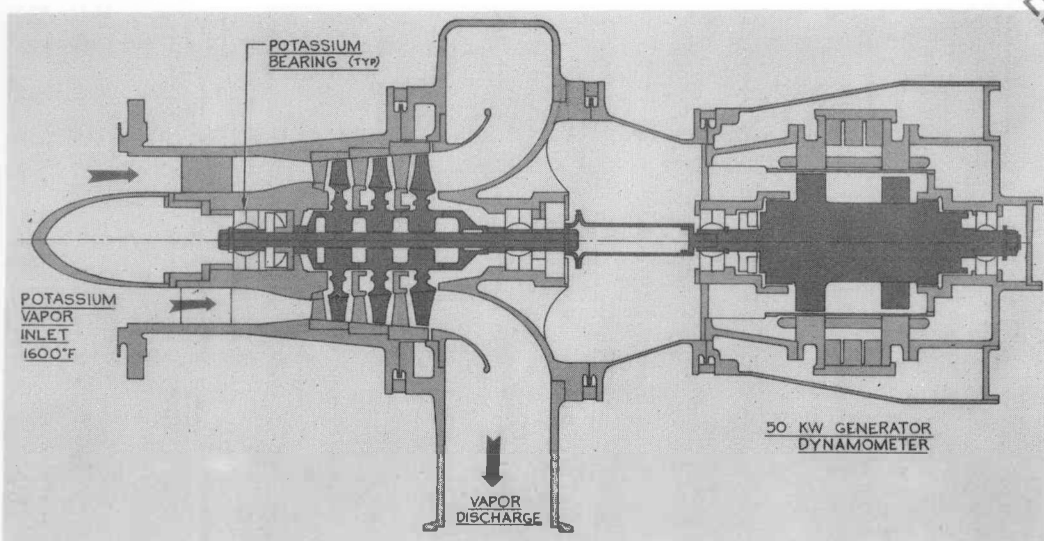
The major turbine development areas are shown on the figure. Aerodynamic performance has been calculated using data from gas and steam turbine operation, and the design procedures used to obtain optimum turbine performance with saturated potassium vapor must be verified.

Erosion of blades due to liquid droplet impingement may be a problem in the last stages due to condensation of the potassium vapor. The degree of erosion is a function primarily of blade speed and blade material. Allowable blade speeds will be of the order of 1,000 feet per second in order to preclude onset of erosion.

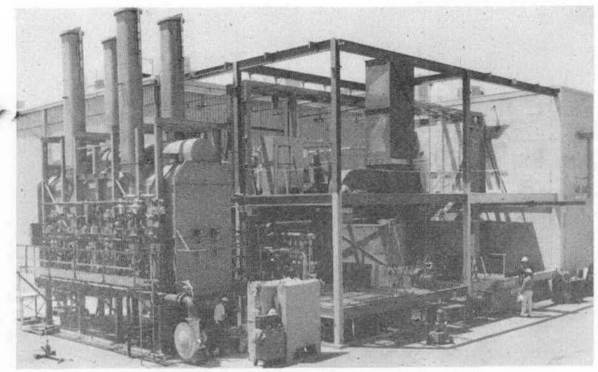
The selection of rotor materials with low creep properties for over 10,000 hours operation is required in order to minimize the progressive out-of-balance of the rotor and to prevent contact with the rotor housing. Good fatigue and creep properties in a potassium environment are required.

TURBINE - AERODYNAMIC PERFORMANCE

- ***INITIAL AERODYNAMIC CALIBRATIONS - AIR (SINGLE STAGES)***
- ***POTASSIUM TESTING (SINGLE & MULTIPLE STAGES)***
  - -AERODYNAMIC PERFORMANCE
  - -BLADE EROSION
  - -MOISTURE REMOVAL



TEST TURBINE



TEST LOOP

AIRESEARCH MANUFACTURING COMPANY
A DIVISION OF THE GARRETT CORPORATION
PHOENIX, ARIZONA

The objective of the turbine design program is to obtain high efficiency within the limits of existing knowledge of the fluid characteristics and properties and the properties of the candidate turbine construction materials.

Of basic concern to the aerodynamic design of a turbine is knowledge of the thermodynamic properties of the fluid to be used. The thermodynamic properties for potassium were initially derived by machine computation using conventional relationships and limited experimental data. The results of this effort are shown plotted in the adjoining Mollier Diagram. Since, this time the Battelle Memorial Institute and the Naval Research Lab have been conducting extensive experimental programs to determine the actual thermodynamic properties. Results of the NRL work are shown in the following tabulation and the results show significant differences from the calculated properties. Current turbine performance analyses are based on the calculated data and will be revised based on this new experimental properties in the near future.

Other fluid characteristics also influence the design of the turbine flow passages to obtain good efficiency. For potassium two characteristics are of interest--supersaturation and the kinetics of potassium dimerization associated with the rapid expansion of the fluid in the turbine flow passage. Based on studies of the supersaturation characteristics of other fluids such as steam, mercury and

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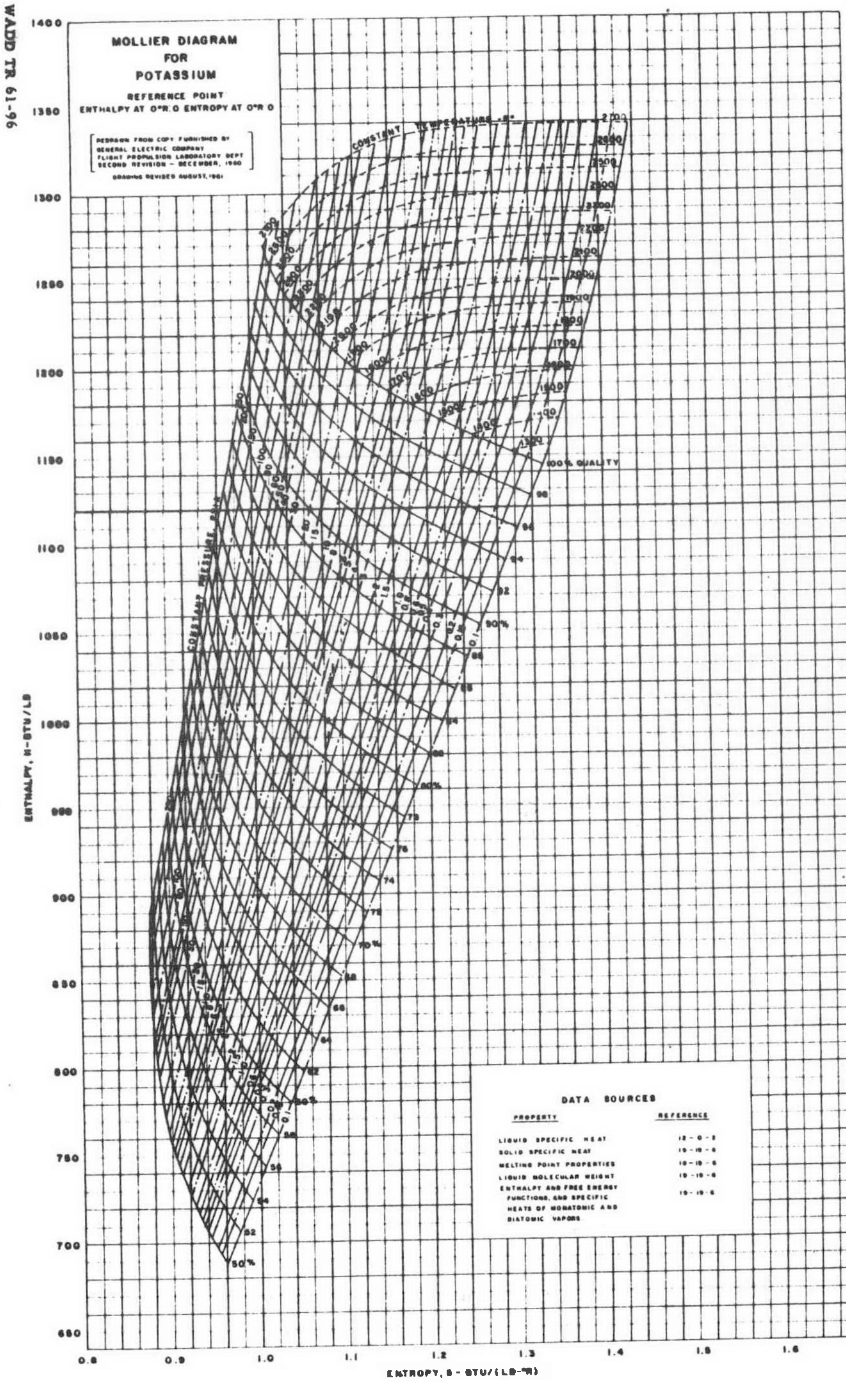

K-HS-a

00216

air, the characteristics of potassium vapor were predicted. The following figure shows the supersaturation index of several fluids. This figure shows that saturated potassium vapor in the temperature range of interest should behave similarly to steam at low temperature which is known to supersaturate until an undercooling of 50° to 100°F is achieved at which point minute droplets form and reversion to equilibrium conditions is quickly achieved. Further analysis based on the work of Tolman, Volmer, and others concerning fractional condensation rates of pure vapors and critical droplet sizes, shows that isentropic expansion of potassium vapor should result in condensation and reversion to equilibrium conditions at a Mach number of approximately 0.5 which is well before the throats of the present turbine flow passage designs.

Tests have been conducted at the General Electric Company under a NASA contract on a converging-diverging nozzle with saturated potassium vapor. Typical results of this program are shown in the next figure. This figure shows the calculated total to static pressure ratio and the experimental data along the nozzle length. This data indicates supersaturation exists downstream of the nozzle throat with reversion beginning at pressure tap No. 15. The next figure shows the experimental polytropic exponent as a function of the inlet quality of the vapor entering the nozzle for a nominal inlet temperature of 1580°F. These test results must also be factored into the continuing turbine design program to insure proper stage matching, flow areas and velocity distributions for optimum efficiency.

[redacted]

SATURATION PROPERTIES OF POTASSIUM
(Monomer Gas Base)

$t\,^{\circ}F$	p_s (Atm)	v^{ℓ} ft³/lb	v_s^g ft³/lb	h^{ℓ} Btu/lb	Δh_v Btu/lb	h_s^g Btu/lb	s^{ℓ} Btu/lb-°F	Δs_v Btu/lb-°F	s_s^g Btu/lb-°F
2500.	33.9590	.03191	1.2383	603.67	645.37	1250.02	.7692	.2181	.98[redacted]
2400.	27.7249	.03097	1.5025	577.11	667.24	1244.35	.7603	.2333	.99[redacted]
2300.	22.2899	.03009	1.8410	552.83	686.20	1239.02	.7518	.2486	1.00[redacted]
2200.	17.6159	.02926	2.2867	529.98	703.63	1233.61	.7435	.2646	1.00[redacted]
2100.	13.6577	.02848	2.8888	508.01	720.42	1228.43	.7352	.2814	1.01[redacted]
2000.	10.3632	0.2775	3.7226	486.51	737.11	1223.62	.7267	.2997	1.0[redacted]
1900.	7.6745	.02706	4.9081	465.25	753.96	1219.21	.7180	.3195	1.03[redacted]
1800.	5.5287	.02641	6.6414	444.07	771.04	1215.11	.7089	.3412	1.05[redacted]
1700.	3.8596	.02580	9.2563	422.92	788.24	1211.17	.6993	.3650	1.06[redacted]
1600.	2.5988	.02522	13.3420	401.84	805.34	1207.18	.6894	.3910	1.08[redacted]
1500.	1.6785	.02467	19.9890	380.89	822.04	1202.94	.6790	.4195	1.09[redacted]
1400.	1.0327	.02414	31.3207	360.19	838.05	1198.24	.6682	.4506	1.11[redacted]

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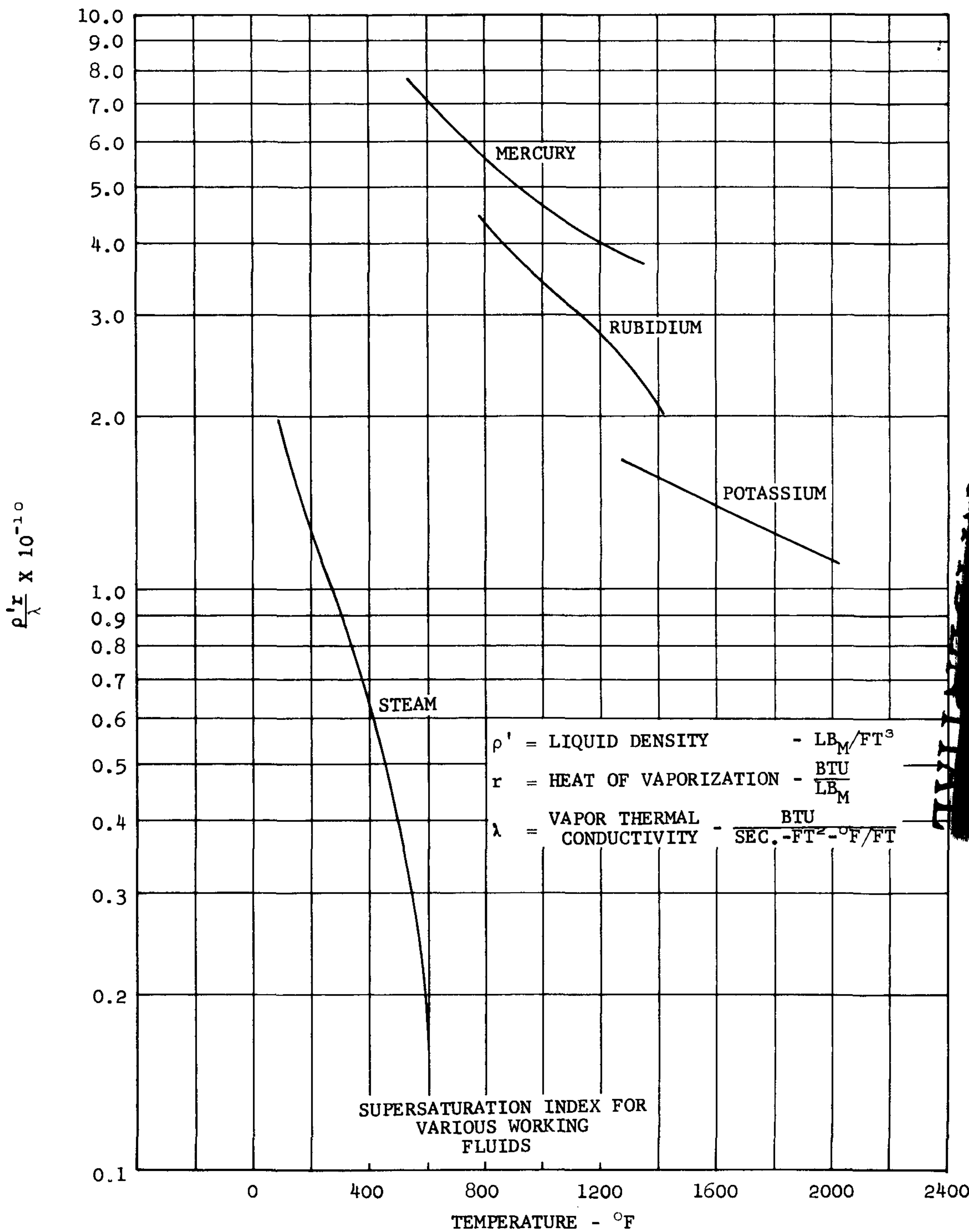
[redacted]

The kinetics of potassium dimerizations have been studied by Dr. Robert D. Waldron of the AiResearch Preliminary Design Section based on a monomer-dimer reaction, reaction rate theory and kinetic theory. For an expansion with a pressure ratio of 2.7 and a temperature drop of 180°F, he concludes that the departure from equilibrium conditions is probably too small to detect experimentally. The validity of this conclusion will be evaluated by actual performance testing of turbine stages with potassium vapor. The overall aerodynamic design of the turbine is based on calculated stage efficiencies which are corrected to account for coordination and the losses occurred in accelerating the condensed vapor to rotor blade speed. An empirical constant is involved in this correction and this constant is based on published steam turbine data. Evaluation of this empirical constant is planned in future potassium turbine tests.

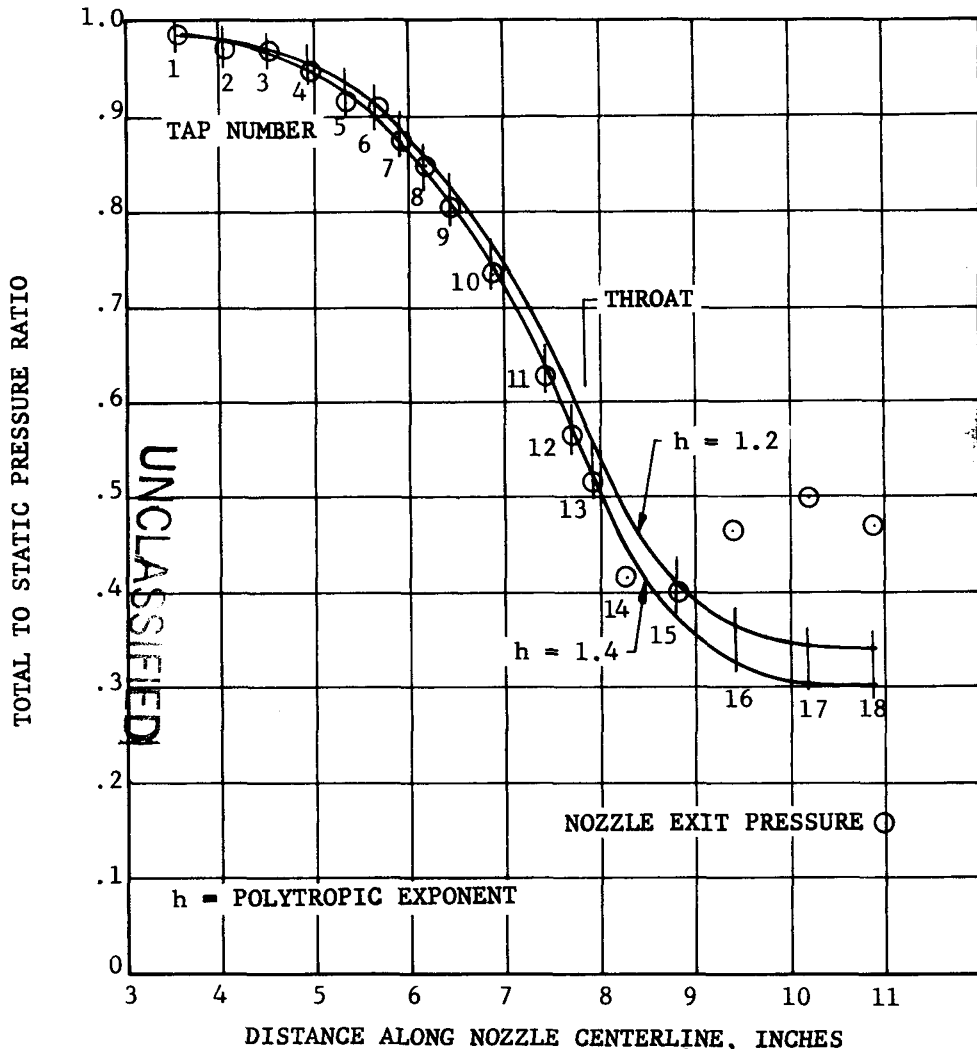
The potassium test turbine and potassium test loop, to be utilized for turbine aerodynamic performance and endurance testing, are shown in a previous figure. The test loop, constructed of stainless steel, permits testing up to approximately 1600°F. The operating characteristics of this loop permit testing of full-size turbines at partial load and testing of the latter stages of the turbine at design and off-design conditions. The turbine is primarily constructed from conventional materials, with the principal housing materials stainless steel and HS-25 and the rotor material Waspaloy. The refractory alloy TZM and a cermet WC with cobalt binder are used for the bearings which operate in liquid potassium.



Initial aerodynamic calibration of the various turbine stages are being made utilizing air prior to actual potassium tests.

00218

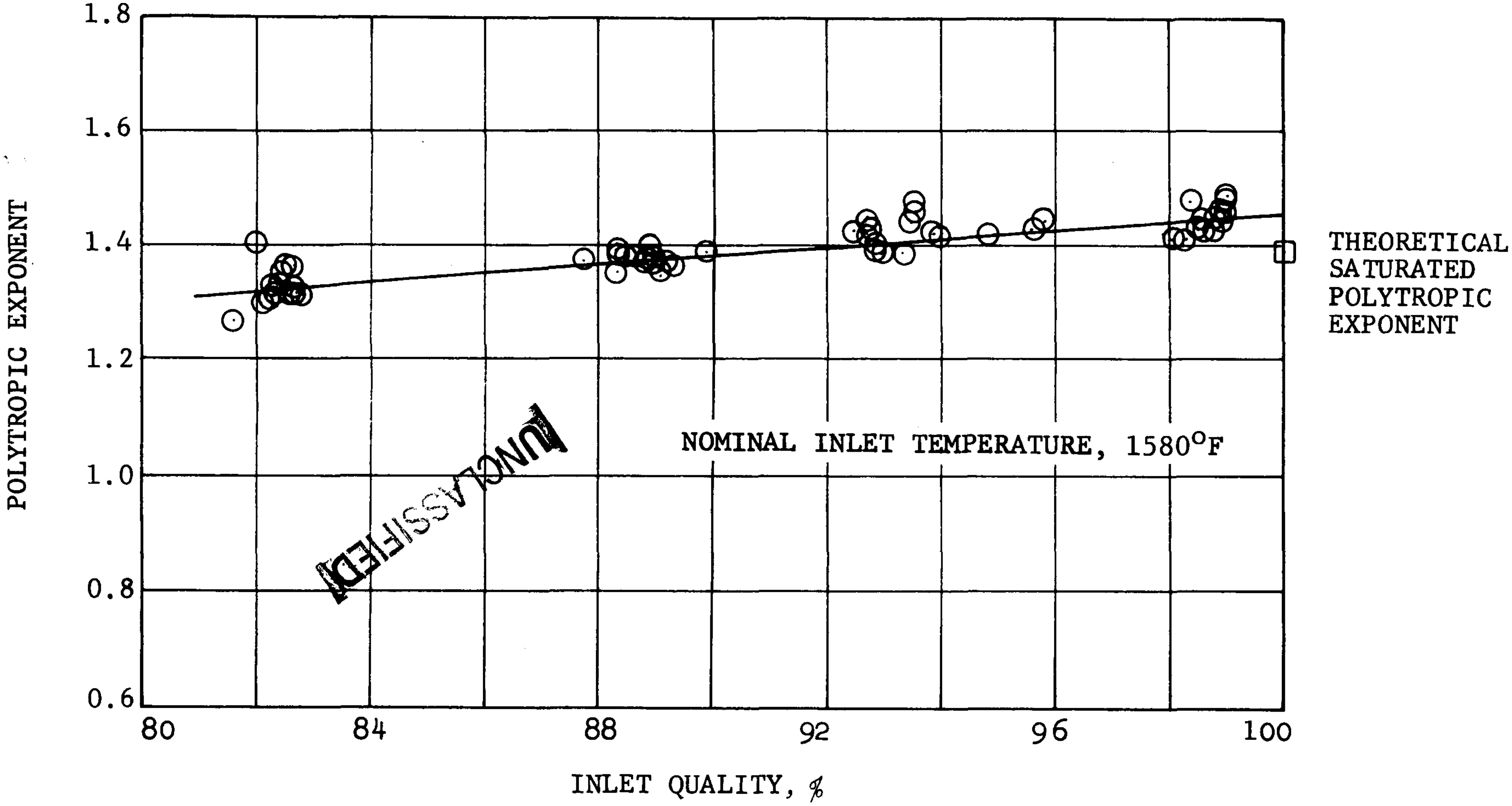
[redacted]



(c) INLET TEMPERATURE = 1580°F

[redacted]

EXPERIMENTAL CONVERGING-DIVERGING NOZZLE PRESSURE DISTRIBUTIONS IN POTASSIUM. INLET TEMPERATURE = 1580°F



VARIATION OF AVERAGE POLYTROPIC EXPONENT WITH INLET QUALITY.

TURBINE - BLADE EROSION

- *RELATIVE EROSION RESISTANCE OF ROTOR MATERIALS (POTASSIUM TESTS)*
- *ESTABLISH PROBABLE BLADE SPEED (WATER & POTASSIUM TESTS)*

0C220

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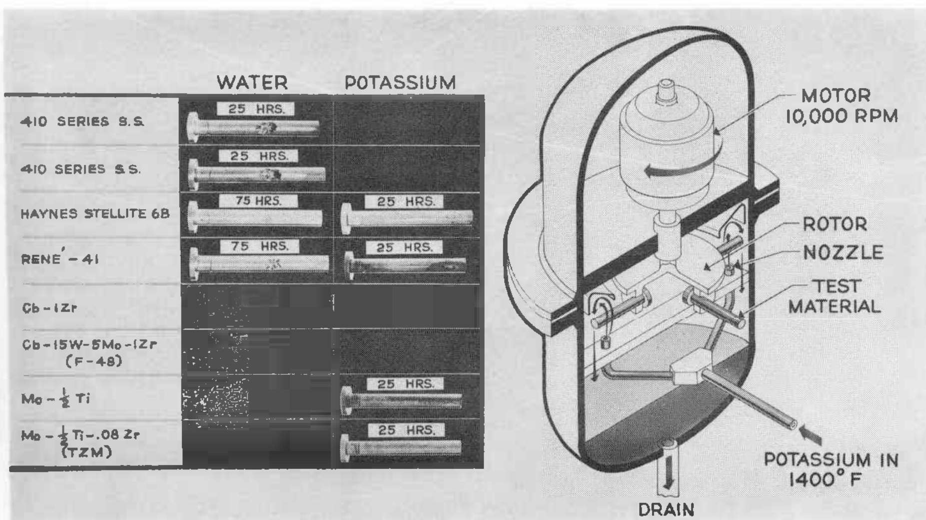
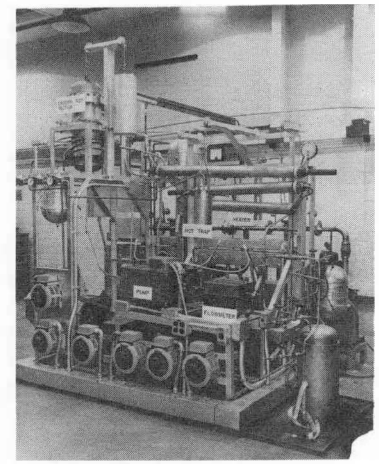
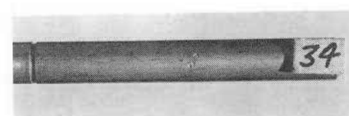


EROSION TEST RIG

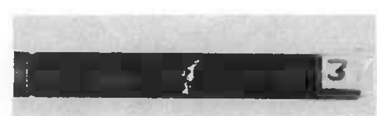


EROSION LOOP

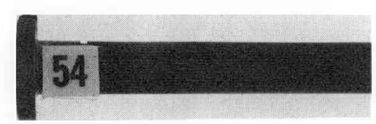


STELLITE 6B

TZC
Mo (1Ti-.15Zr-.15C)



F-48
Cb (15W-5Mo-1Zr)



T111
TA (8W-2Hf)

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PHOENIX, ARIZONA

One of the key factors influencing the design of the turbine and its efficient operating life, is erosion of the turbine blades due to impingement of slow moving liquid droplets on the moving turbine blades. It has been observed that erosion of steam turbine blading with saturated steam conditions does not occur at blade speeds below approximately 600 fps. For speeds greater than this, the blade is protected by inserting Stellite strips into the blades. With these inserted strips, a service life of many years is common.

Since a satisfactory method of extrapolating steam turbine experience to new materials under entirely new and different conditions is not available, a limited test program was initiated to evaluate materials under low temperature water and high temperature potassium conditions. The rig design, shown above consists of a rotor holding four specimens which rotate normal to the flow direction of two 1/4 inch diameter fluid jets.

With this type of test rig, erosion occurs rapidly for many materials and more slowly for others provided the size of the fluid jet and specimen speed are such that erosion will occur.

The relative weight loss is shown of several common materials used for steam and gas turbines after testing in water for periods up to 75 hours (76×10^6 impacts). The following figure shows the relative weight loss of various refractory metal alloys based on testing for periods up to 50 hours and the results indicate that the molybdenum base alloys are more resistant to erosion in potassium than Stellite 6B, (HS6B). However, the next figure shows that 1400°F potassium produces approximately twice the weight loss that water does for the same number of impacts.

[redacted]

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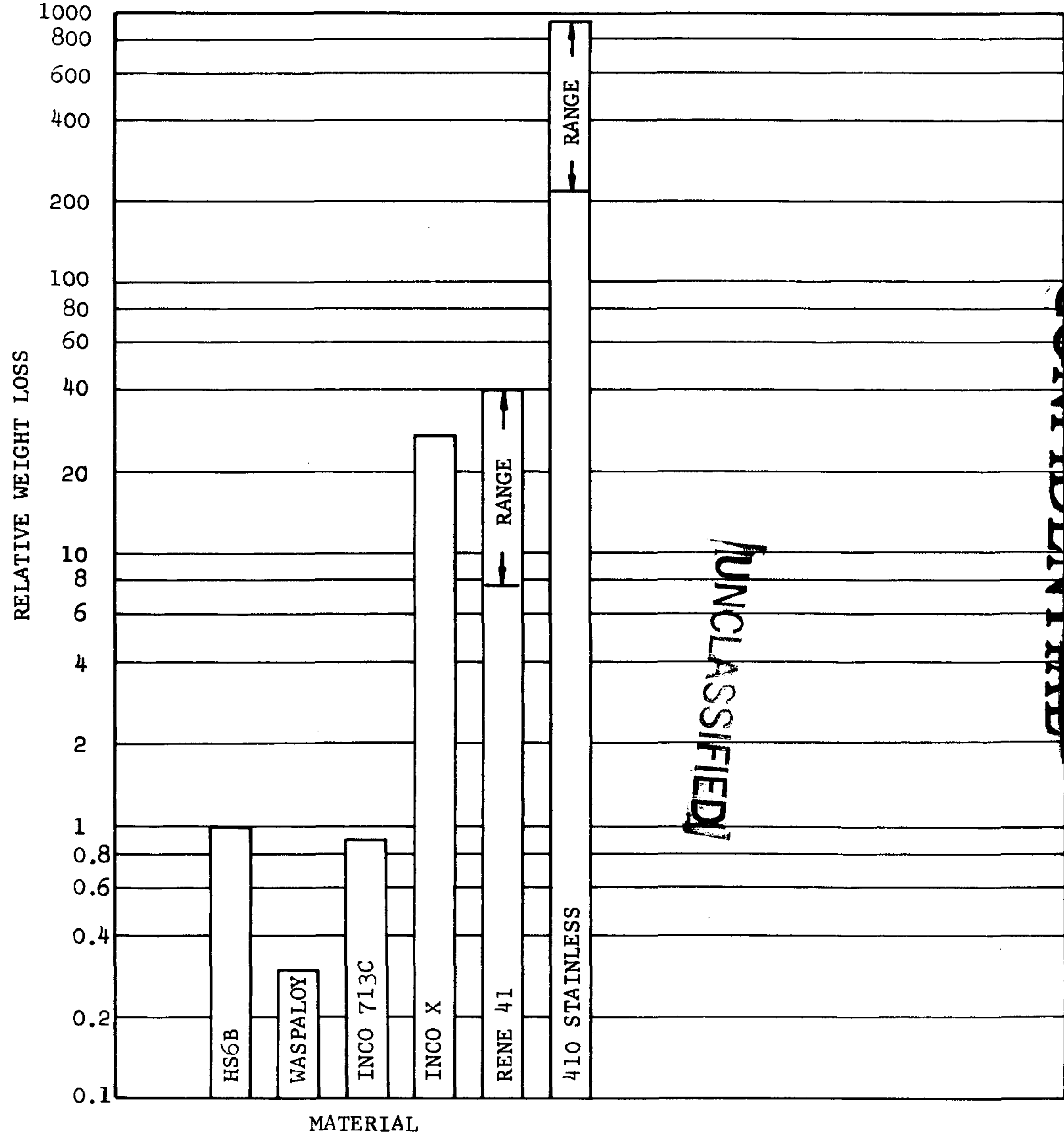
WATER EROSION TEST RESULTS



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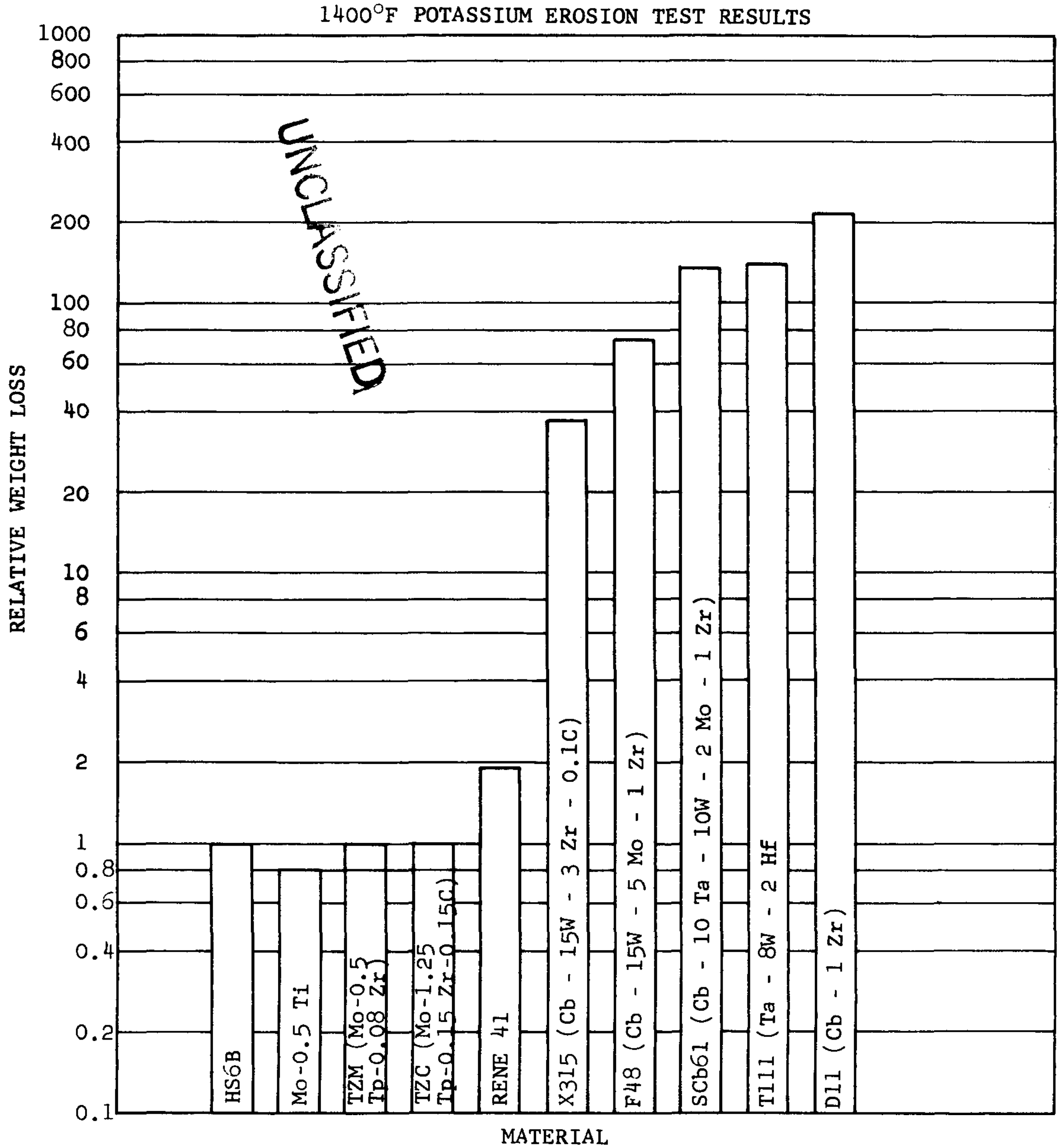
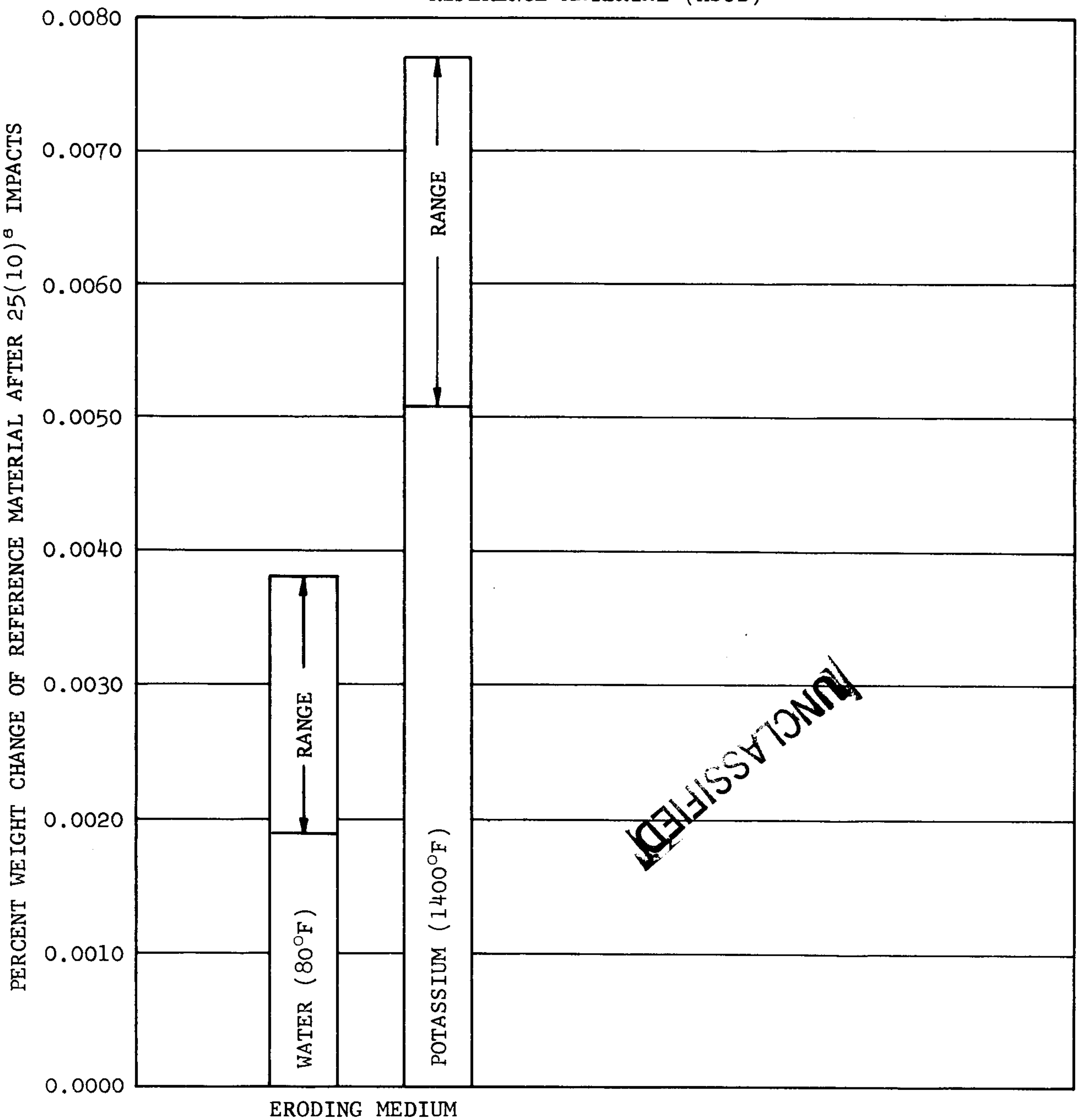
The effect of velocity on erosion rate is not precisely known and various investigations have reported that erosion damage is a linear function of velocity. (Peter DeHaller DTMB Report) However, the same investigators have stated that below certain velocities no erosion takes place. This agrees with turbine experience and indicates that velocity is not a linear effect. Other investigators have reported values as high as the eighth power. Other agencies have initiated additional work in this area and the results should be available shortly.

For the SNAP 50/SPUR design, the current value being considered for blade tip speed is 1,000 feet, per second--just slightly higher than an extrapolation of the above data indicates is acceptable. This value however is considered acceptable for initial testing, considering the differences in operating lifetimes between the SNAP 50/SPUR turbine and that of present day large steam turbine power systems.



COMPARISON OF WATER AND POTASSIUM EROSION OF
REFERENCE MATERIAL (HS6B)



00223

TURBINE - ROTOR MATERIAL AND CONSTRUCTION (Mo ALLOYS)

- ***OBTAIN MECHANICAL PROPERTIES OF CANDIDATE MATERIALS***
  - -INERT GAS AND VACUUM, CREEP/RUPTURE & FATIGUE (50; 2000 F, 1000 HRS)
- ***DETERMINE EFFECT OF POTASSIUM ON MECHANICAL PROPERTIES***
  - -POTASSIUM VAPOR, CREEP/RUPTURE & FATIGUE (50, 2000 F, 1000 HRS)
- ***ESTABLISH BEST FORGING -MACHINING PROCEDURES***
- ***OBTAIN MECHANICAL PROPERTIES OF ROTOR FORGINGS***
  - -CREEP/RUPTURE, IMPACT, OVERSPEED

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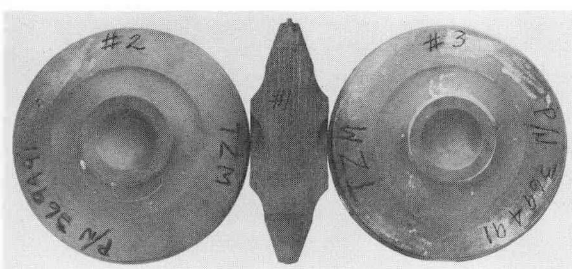
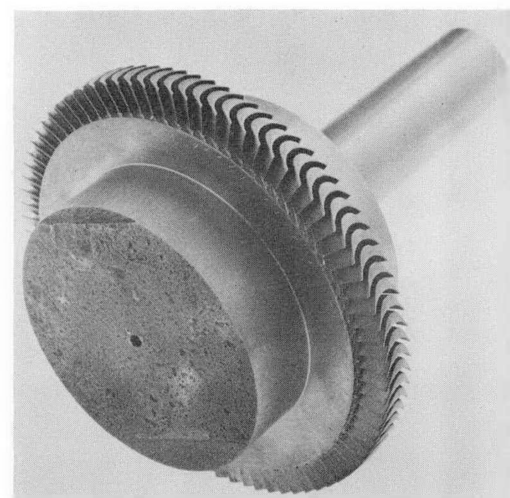


TZM FORGINGS


TZM TURBINE WHEEL

The molybdenum alloy, Mo-0.5 Ti, was initially selected as the principal candidate rotor material. Selection over other potentially suitable high temperature alloys (F-48, D-31, D-41 etc.,) was based on available data, on the following:

(1) Good creep resistance at 2000°F

(2) Good fabricability and reproducability

(3) Compatibility with working medium-potassium

To support the selection of this alloy, tests were initiated to determine the long time creep and fatigue properties and to ascertain what affect, if any, potassium had on these properties. These tests were performed earlier in the program and reported in several Topical Reports. A representative portion of the fatigue test data is shown in the following figure.

As a result of technological advancements in casting and fabrication, test data for the TZM alloy (Mo-0.5 Ti-0.08 Zr) indicated superior mechanical properties to those of the Mo-0.5 Ti alloy. The TZM alloy therefore became the candidate turbine rotor material and tests were initiated to further evaluate the alloy. Specimens, machined from bar stock, were tested to determine what effect, if any, potassium had on the creep-rupture properties. Upon completion of the tests, it was concluded that the 1000-hour creep-rupture behavior of this material is not influenced significantly by exposure to potassium during creep testing as is shown in the following figures.

Early limitations in forging the TZM alloy necessitated the initiation of a forging development program that would result in a material to satisfy the requirements listed above in addition to the following:

(1) Low ductile-to-brittle transition temperature

(2) Long time stability (10,000 hours) under stress at 2000°F

(3) Good forgeability with property uniformity

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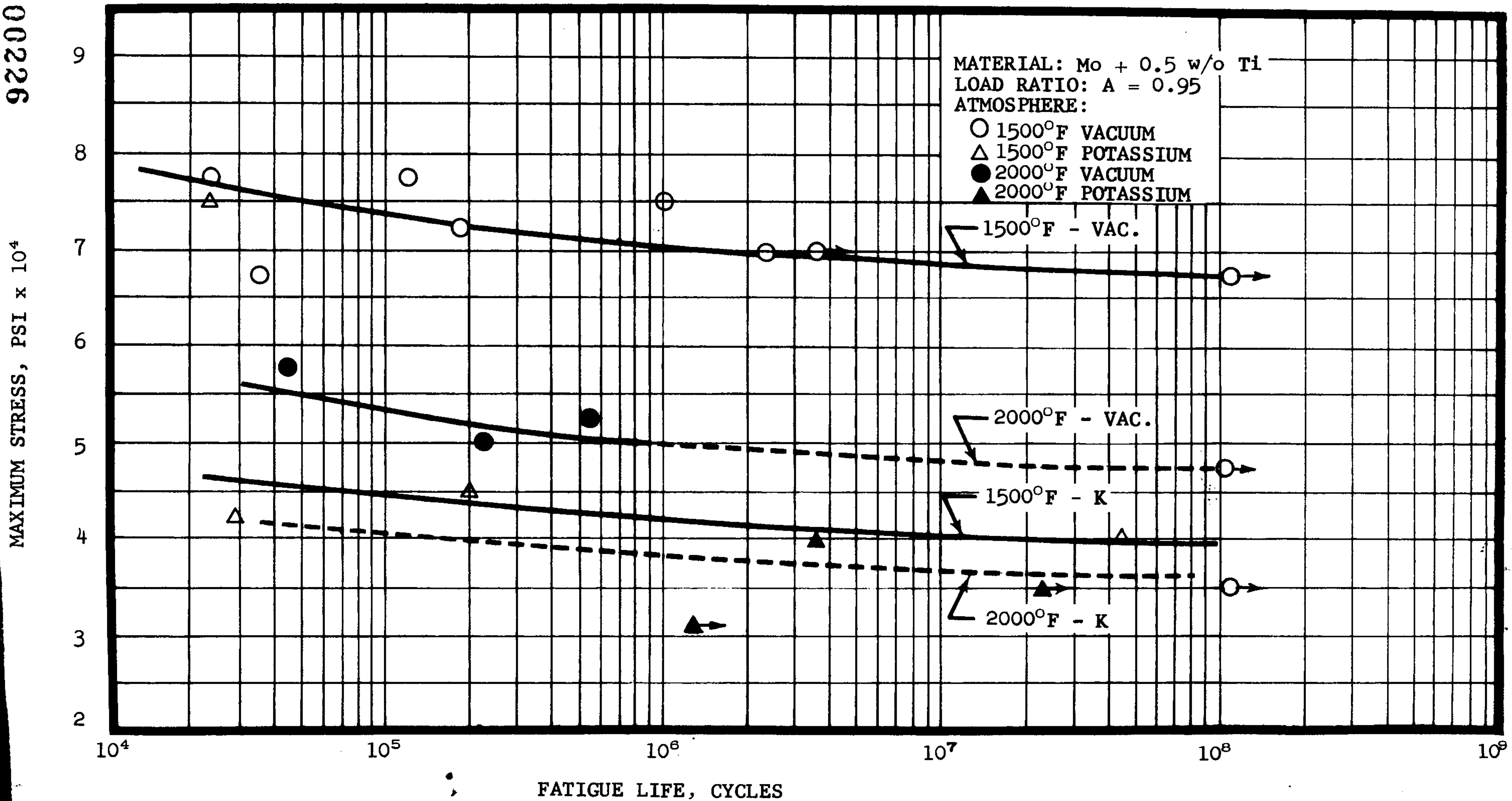

RESULTS OF AXIAL LOAD FATIGUE TESTS ON Mo + 0.5 w/o Ti
IN HELIUM-VACUUM ATMOSPHERE, OR POTASSIUM-VACUUM
ATMOSPHERE AT 1500°F AND 2000°F

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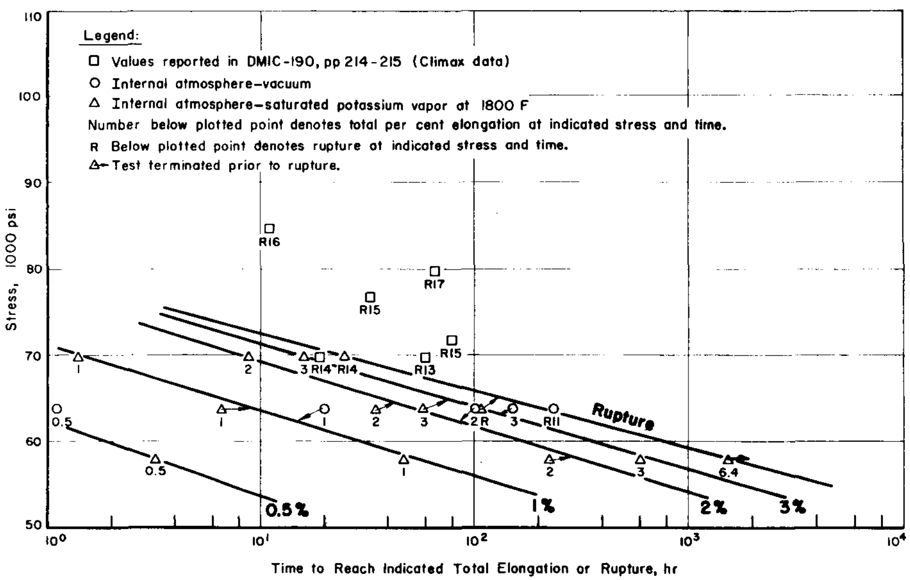
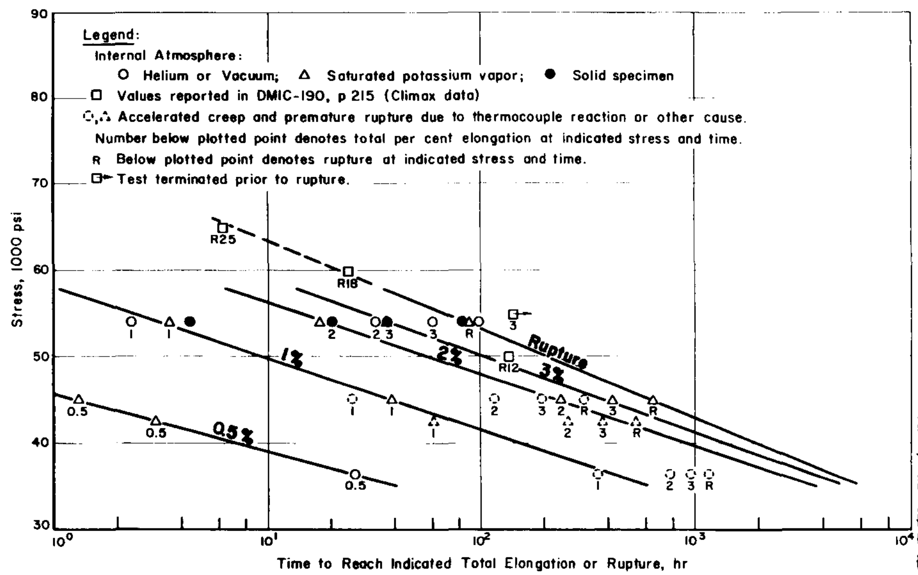


DESIGN CURVES FOR STRESS-RELIEVED TZM ALLOY AT 1800 F



DESIGN CURVES FOR STRESS-RELIEVED TZM ALLOY AT 2000°F

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The 4-inch diameter billets, of varying carbon composition, were specially processed as shown in the following table. Specimens from the 3/4-inch thick x 9-inch diameter forgings in the photograph shown above were tested to obtain hot hardness, ductile-to-brittle transition temperature and 1000-hour creep data at 2000°F. Representative portions of the data are shown in the tabulations in subsequent pages, which also show the predicted stress levels in the turbine rotor discs and blades. The maximum permissible creep for a 10,000 hour life turbine is considered to be about 0.5 percent in 10,000 hours based on tip clearance requirements for good turbine performance.

Work being performed by others in being surveyed and will be utilized where applicable. Of primary importance is the long time creep data on refractory alloys at elevated temperatures being conducted by TAPCO. The turbine alloys being investigated are as follows:

TZM	disc forging
T2C	plate
ST-222	plate
Cb-132M	plate
AS-30	plate

Tests of 1000 hours duration at 2200°F are being conducted for screening purposes. The most promising alloys will be tested for 10,000 hours.

TAPCO is also engaged in a program to determine the fatigue properties of several refractory materials (to be selected later) at frequencies of 3 to 15,000 cps. None of these tests are being performed in liquid metal environments.

DISC No.	INGOT DIA. (in.)	BREAKDOWN: EXTRUSION DIA.	RECRYSTALIZATION TEMP., °F/HR	FORGE/ROLL	RECRYSTALIZATION TEMP., °F/HR	COMPOSITION W/O: C	Ti	Zr	O_2	N_2	H_2
1	12 →	6 inches → 2200°F		/4 inches → 2200°F	2900/2	0.022	0.46	0.09	0.0012	0.0008	0.0002
2	12 →	6 inches → 2200°F		/4 inches → 2200°F	2900/2	0.014	0.49	0.114	0.0018	0.0010	0.0001
3	12					0.024	0.48	0.114	0.0018	0.0002	0.0002
4	12 →	6 inches → 2250°F	2800/4 →	4 inches / → 3400/2800°F	2900/2	0.024	0.48	0.114	0.0018	0.0002	0.0002
5	12	———	———	——— ———	———	0.024	0.48	0.114	0.0018	0.0002	0.0002
				Extrusion	Stress Relief						
6	12					0.014	0.43	0.10	0.0003	0.0001	0.0001
7	12 →	7 1/4 inches → 2000°F	3000/1 ↓ Sol. Ann. → 7 in. → 3800/1	4 1/4 in. → 2250°F	2250/1	0.014	0.43	0.10	0.0003	0.0001	0.0001
8	12				*	0.014	0.43	0.10	0.0003	0.0001	0.0001

*NOTE: Disc No. 8 received a 2-hour age at 2700°F after the 2250°F, 1-hour stress relief

PROCESSING HISTORY AND COMPOSITION OF TZM DEVELOPMENT FORGINGS

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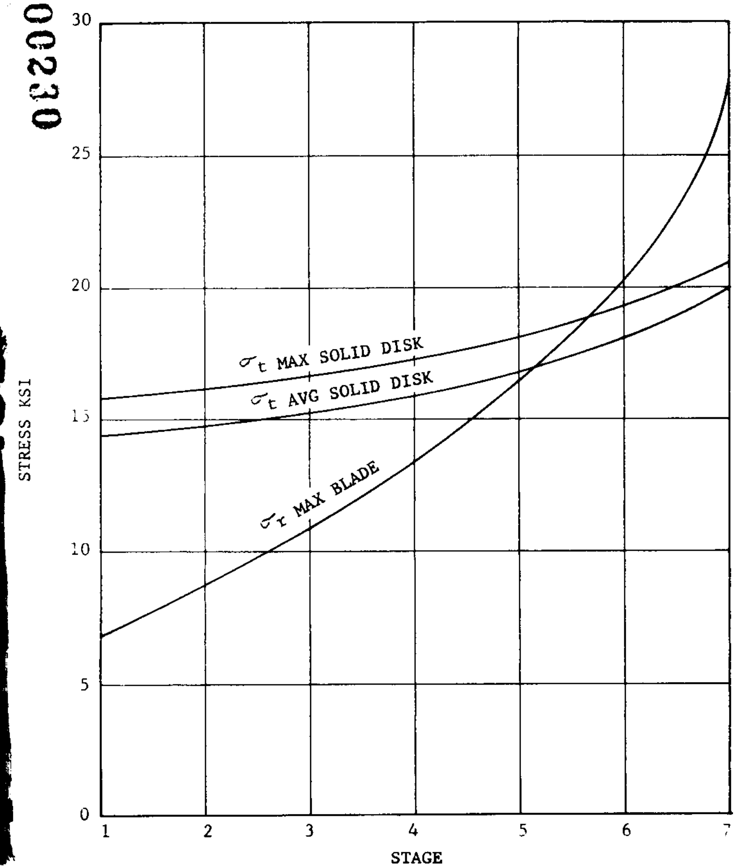
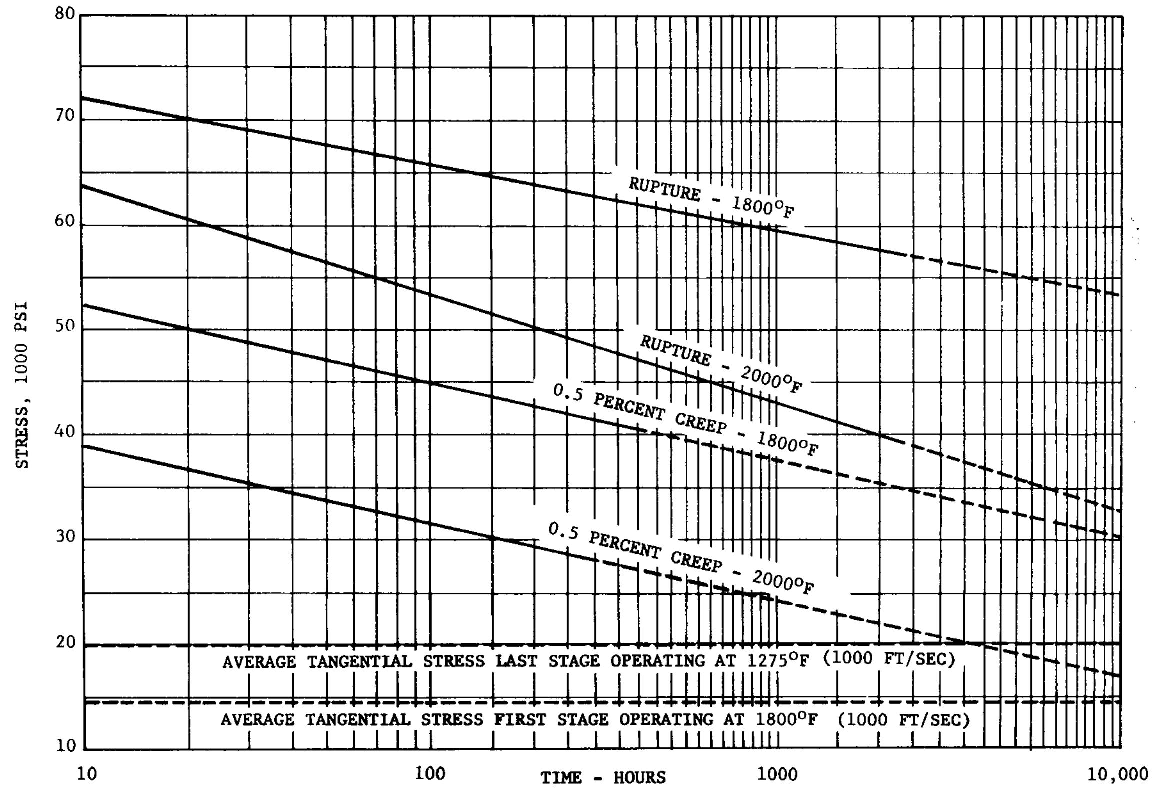


CONSTANT HUB DESIGN WITH INSERTED BLADES
7 STAGE POTASSIUM VAPOR TURBINE INLET
TEMPERATURE 1850°F
EXHAUST TEMPERATURE 1275°F
TURBINE SPEED 24,000 RPM
ROTOR AND BLADE MATERIAL TZM



TZM ARC CAST, HOT ROLLED,
STRESS RELIEVED (1 HOUR
AT 2200°F) BAR STOCK

[redacted]

[redacted]

AIRESEARCH MANUFACTURING COMPANY
A DIVISION OF THE GARRETT CORPORATION
PHOENIX, ARIZONA

HARDNESS AND PERCENT RECRYSTALLIZATION OF TZM FORGINGS

Forging No. /	1		2		3		4		5		6		7		8	
Condition	DPH	% Rec.	DPH	% Rec.	DPH	% Rec.	DPH	% Rec.	DPH	% Rec.	DPH	% Rec.	DPH	% Rec.	DPH	% Rec.
As-Received	273	0	267	0	311	0	270	0	305	0	294	0	359	0	362	0
1 Hr-2500 F	257	5	253	10	303	1	262	0	280	1	285	1	350	1	325	3
1 Hr-2600 F	216	98	191	98	230	25	243	1	238	20	196	99	224	90	213	95
1 Hr-2650 F	-	-	-	-	-	-	-	-	-	-	202	100	221	98	211	99
1 Hr-2700 F	192	99	188	99	213	90	242	3	230	75			211	100	214	100
1 Hr-2750 F	195	99+	192	99+	-	-	-	-	-	-						
1 Hr-2800 F	193	100	196	100	201	99	235	8	209	90						
1 Hr-2900 F					197	99	222	15	205	98						
1 Hr-2950 F					197	100	-	-	-	-						
1 Hr-3000 F							202	60	190	99						
1 Hr-3050 F							--	-	187	100						
1 Hr-3100 F							198	90								
1 Hr-3150 F							172	100								

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UNNOTCHED CHARPY IMPACT TEST RESULTS

Absorbed Energy, ft-lbs

Forging No. / Test Temp., F	1	2	3	4	5	6	7	8
-50	7-1/2					16-1/2		
-25	2-1/2					22-3/4		
0	12-1/2	12-1/4	11-1/2		8	16		
25	21-3/4							
50	30+					17-3/4		
75	30+	16-1/4	8	3-3/4	15-3/4	30+		8
100						30+		
150		9-3/4	9-3/4		10-1/2		11-1/4	
175		30+					13	
200		30+	14-1/2	10	10-1/4		30+	
225			30+		30+			
250		30+	30+	2-1/4	30+			11-1/2
275				2-1/4				
300				30+			30+	
400							30+	
450								16-1/4
475							30+	30+
525								30+
600								30+

00231

TENSILE TEST DATA OF TZM DEVELOPMENT FORGINGS
(DUPLICATE TESTS)

Disc No.	Room Temperature Tests: UTS x10^{-3} psi	0.2% YS x10^{-3} psi	Elong %	Red. in Area
1	120.2	112.7	18.6	46.0
	113.1	110.2	16.8	51.1
2	116.4	111.6	19.7	48.9
	112.5	107.0	19.4	45.8
3	129.9	121.3	14.7	34.0
	132.5	122.3	15.1	34.8
4	112.5	100.7	13.0	21.9
	111.5	100.5	20.5	47.1
5	129.8	119.8	13.8	22.1
	118.3	115.3	1.6	0.9
6	127.9	121.9	17.4	48.4
	125.6	119.1	14.1	44.7
7	148.6	141.3	3.1	5.4
	147.3	142.0	1.6	3.6
8	146.8	143.5	2.2	3.8
	144.4	141.3	0.7	0.8

Disc No.	2000°F Tests (in Vacuum, ≤2x10^{-5} torr): UTS x10^{-3} psi	0.2% YS x10^{-3} psi	Elong %	Red. in Area	Hardness, R_A Before	Hardness, R_A After	Decrease in Average UTS x10^{-3} psi
1	72.3	70.9	12.7	74.4	61.3	61.5	43.8
	73.4	72.4	11.9	73.4	60.4	60.6	
2	66.6	65.6	13.8	81.8	59.9	59.8	48.6
	65.1	63.7	10.3	71.9	60.9	60.0	
3	88.3	87.5	12.2	79.6	62.4	63.5	44.4
	85.3	83.6	10.1	65.3	61.4	61.0	
4	63.4	61.6	13.0	78.6	60.1	58.0	47.6
	65.3	64.4	11.0	71.9	58.8	58.6	
5	81.0	80.1	12.8	82.2	61.4	61.2	41.4
	84.3	82.7	12.8	74.7	62.4	61.1	
6	78.8	77.1	11.5	77.6	61.4	61.8	47.4
	80.0	78.3	12.3	76.7	63.0	60.6	
7	102.1	98.3	--a	--a	65.1	64.6	47.4
	99.0	97.4	6.9	40.2	66.2	64.8	
8	100.4	98.4	9.0	49.4	66.7	63.4	44.8
	101.1	96.0	10.0	60.1	66.4	64.3	

a - Physical setup prevented specimen from elongating to failure.

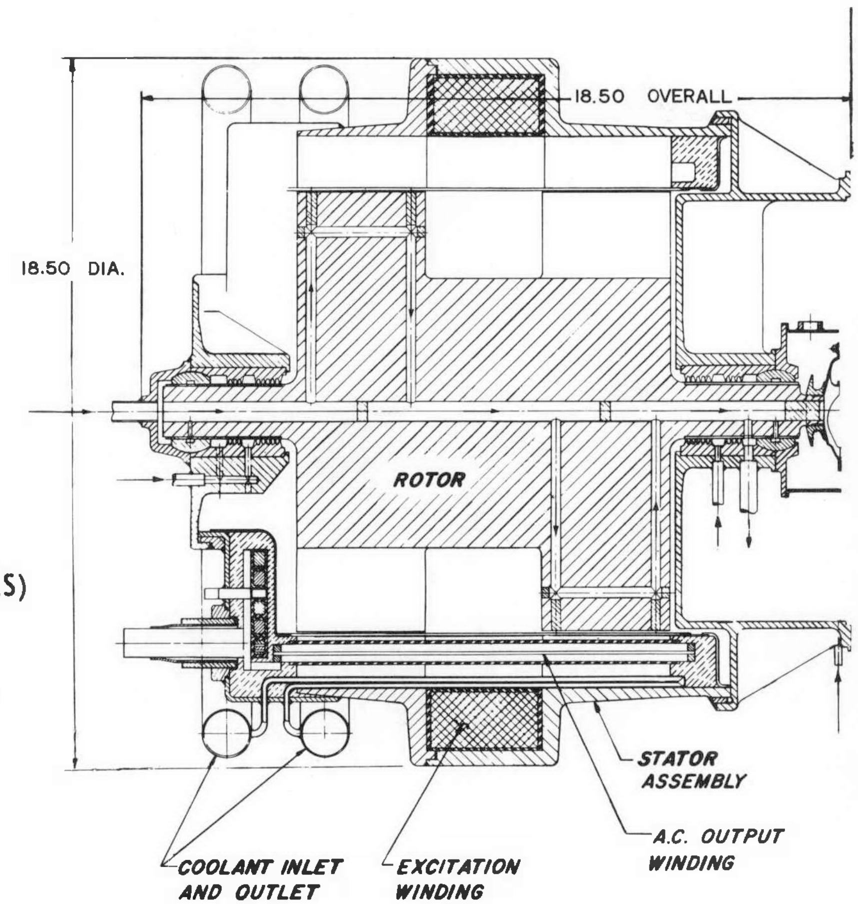
Strain rate: 0.005 inch/min to 0.6% offset then
0.05 inch/min to failure

[redacted]

SNAP 50/SPUR GENERATOR

DEVELOPMENT AREAS

- **ELECTRICAL PERFORMANCE**
  - -GOOD EFFICIENCY (3200 CPS AT TEMP)
- **THERMAL CHARACTERISTICS**
  - -STATOR & ROTOR COOLING
- **STATOR INSULATION SYSTEM**
  - -SEALED FROM POTASSIUM (BORE SEAL)
  - -DIELECTRIC DEGRADATION (TEMP., TIME, VACUUM OR INERT GAS)
- **ROTOR MATERIAL SELECTION AND FABRICATION**
  - -MAGNETIC Vs. CREEP PROPERTIES
  - -POTASSIUM COMPATIBILITY
  - -POLE FACE LOSS
- **ROTOR CAVITY EVACUATION**
  - -WINDAGE LOSS
  - -CAVITY CONDENSATION



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MS-744-25

The radial magnetic gap, homopolar inductor generator was chosen since no rotating windings or brushes are required; the rotor is one piece and symmetrical with low stresses, and electromagnetic unbalance is least affected by rotor-stator misalignment. Seals are provided at each end of the generator rotor in order to prevent liquid potassium from entering the rotor cavity. 24,000 RPM was selected as the highest speed consistent with practical design limits for both the turbine and generator. The unit is cooled by liquid potassium, at approximately 600°F, circulated through the rotor and stator.

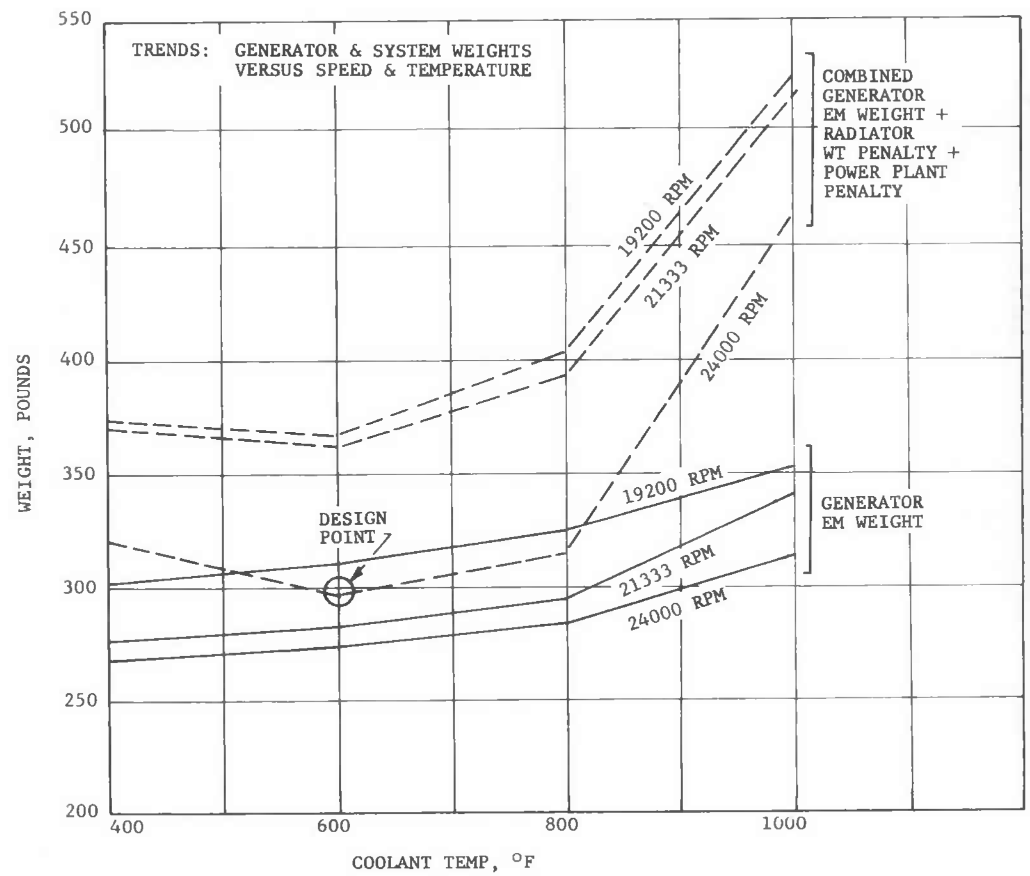
The generator design point selection was based on results of studies which showed that as the coolant temperature is increased, above 600°F, overall powerplant weight tends to increase due primarily to reduced generator efficiency that arises from lowered magnetic properties of the rotor and stator materials and from higher electrical resistance in the conductors. Reduced generator efficiency at higher coolant temperatures requires a larger and heavier boiler, turbine, etc. Higher generator temperatures result in a lighter and smaller auxiliary radiator but this saving in weight is offset by the increases in the system weight penalty and therefore, leads to an optimization of generator temperature at 600°F, as shown on the adjoining figure.

Development of this generator can be achieved with existing materials. The generator materials programs discussed on the following pages consist of defining material characteristics and properties under specific design conditions.

Early attempts to find good individual wire insulation that was compatible with potassium were unsuccessful and the bore seal concept was adopted. The use of a ceramic bore seal completely isolates the electrical wiring from potassium and therefore alleviates any potassium compatibility problems with the electrical insulators and conductors. Ceramic material must be used in the air gap since its low electrical conductivity will minimize induced eddy currents resulting from magnetic flux pulsations.

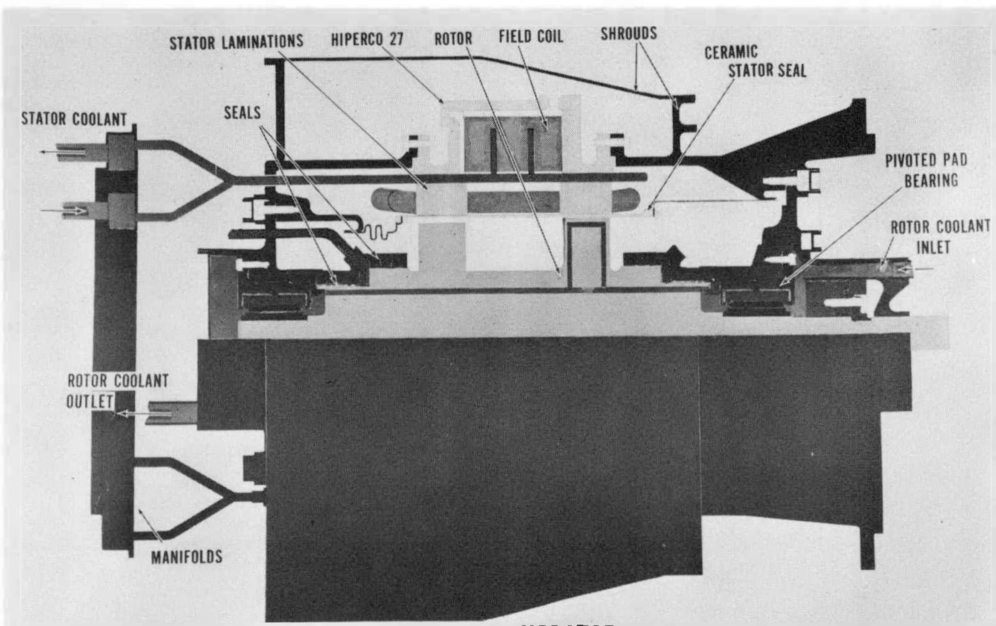
Rotor material must have good creep strength in potassium at operating temperature and must have good magnetic permeability since the rotor is a link in the magnetic circuit of the generator. The motion of the tips of the rotor poles passing the slots in the stator stack results in eddy current losses in the rotor pole faces. The magnitude of these losses has been estimated to be 2 kw, thus requiring passages in the rotor for the potassium coolant. Tests are being conducted to determine the actual value of these losses.

Windage losses of the generator are minimized by designing hydrodynamic seals to assure that no liquid potassium enters the rotor cavity. Potassium vapor is present in the rotor cavity at 0.1 psia.

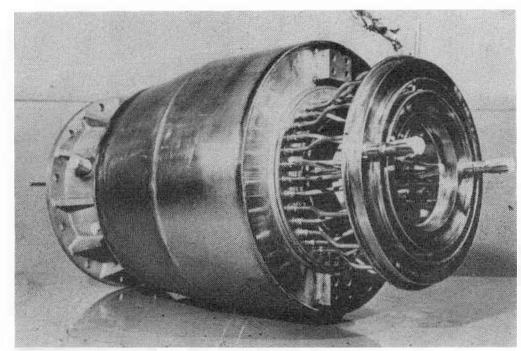


GENERATOR - LMCD II TESTING

- **50 KW POTASSIUM COOLED GENERATOR (3200 CPS)**
  - -OBTAIN PRELIMINARY PERFORMANCE DATA (ELECTRICAL, THERMAL)
  - -INITIAL DATA INSULATION INTEGRITY
  - -INITIAL DATA CAVITY EVACUATION
  - -FABRICATION AND ASSEMBLY TECHNIQUES



LMCD II GENERATOR
LIQUID POTASSIUM COOLED
50 KVA AND LUBRICATED 24,000 RPM



The purpose of the LMCD-II generator is to (1) define fabrication and assembly techniques for the high-temperature inductor generator; (2) verify design principles; (3) obtain preliminary performance data on electrical and thermal characteristics; and (4) identify development problem areas.

Two of the 50-kw model SNAP 50/SPUR generators, shown above, have been fabricated. This generator is a 3200-cps, 24,000-rpm unit with potassium bearings and seals and duplicates the SNAP 50/SPUR machine as far as rotor and stator construction techniques and operating environment are concerned.

Initial checkout of these machines has been accomplished up to 16,000 rpm using conventional liquids as lubricants. Short-duration potassium tests up to 8500 rpm have been run on one unit with potassium cooling temperature of 400°F.

Preliminary data on the insulation integrity, ceramic bore seal assembly, and cavity evacuation have been obtained.

GENERATOR BORE SEAL

- ***TO PROTECT STATOR INSULATION FROM POTASSIUM VAPOR IN CAVITY***
- ***MINIMUM AIR GAP THICKNESS; NONMAGNETIC; NON CONDUCTING***
- ***COMPATIABLE WITH POTASSIUM VAPOR***
- ***MECHANICAL AND THERMAL INTEGRITY***

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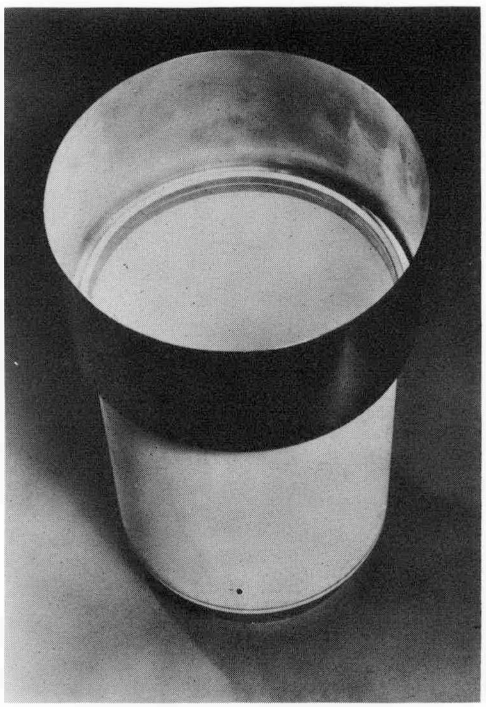
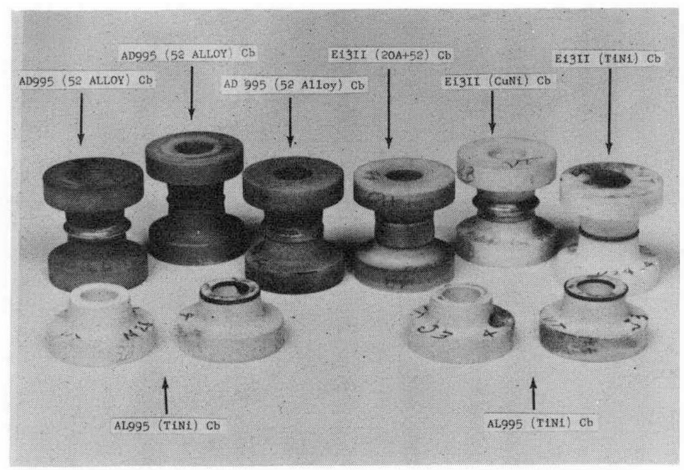


ALUMINA BORE SEAL



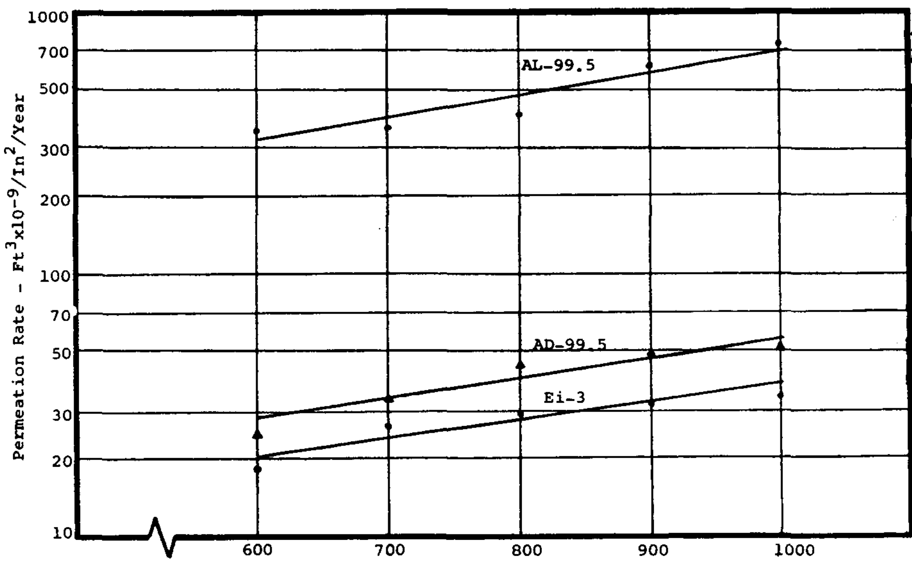
ALUMINA TEST SAMPLES

The purpose of the bore seal is to protect the stator windings and insulation from potassium vapor in the rotor cavity. A material with nonmagnetic and nonconductive properties, along with compatibility with potassium vapor is required. Alumina was selected for this application as a result of potassium compatibility tests of various materials.

The general feasibility of the alumina bore seal concept has been demonstrated in the testing of LMCD-I and LMCD-II generators. An additional verification of the alumina bore seal concept has been demonstrated in the 500-kva, 3200-cps, 250°F oil-cooled UTE generator developed by AiResearch; this machine uses a 7-inch diameter alumina bore seal fabricated from 0.100-inch thick AD-99.5.

Attachment of the ceramic bore seal cylinder to the housing of the generator with a leak-tight joint is required. The development of a braze joint for this application has been a major part of the bore seal program. Other requirements which must be met in the bore seal development are low permeability to leakage of potassium out of the rotor cavity or to leakage of inert gases into the rotor cavity. Mechanical integrity of the bore seal must be maintained in the vibration, shock and thermal environment imposed on the unit.

In order to determine the most practical bore seal design for the SNAP 50/SPUR generator various grades of alumina were brazed to Kovar and pure columbium and tested in 900°F and 1100°F vapor and liquid potassium. Basic tensile tests in air and at room temperature supplemented the above data. Different high-temperature brazing alloys were used and evaluated in the above tests for both compatibility and strength. High density alumina was found to be more compatible with potassium and the brazing alloys. Coast Metal 52, a nickel-titanium alloy, and copper-nickel had the best brazing integrity. However, metallographic examinations revealed that columbium parts must use a protective coating to avoid attack from nickel in brazing alloys. As a result, several active metal brazes have also been tested. Based on early tests, three grades of alumina, Ei-3III (WESGO), AD-99.5 (Coors) and Ei-3IV (Coors) were selected for further testing in potassium environment. Tests at 900°F in potassium vapor for 1,000 hours and 5,000 hours indicated that the Ei-3 material showed the least attack by the potassium. Tensile tests and metallography were performed on these test specimens. Additional 5000- and 10,000-hour potassium tests are in progress.



HELIUM PERMEATION THROUGH ALUMINUM OXIDE

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Leak testing of brazed tubular assemblies fabricated from AL-99.5, AD-99.5, and Ei-3 alumina was performed at 600, 700, 800, 900, and 1000°F. The test was conducted with a helium atmosphere on the outer wall of the ceramic tube and a vacuum of 10^{-5} torr on the inside of the tube. The helium permeation, as measured with a mass spectrograph calibrated with a leak rate of 5.0×10^{-8} cc per sec., was obtained. The above figure shows that Ei-3 material has the lowest permeation rate of the materials investigated.

A full-size ceramic cylinder, 11 inches in diameter with a 0.090-inch thick wall, was fabricated from the AD-99.5 (Coors) material. This cylinder was successfully subjected to a vibratory load on a conventional vibration shake fixture.

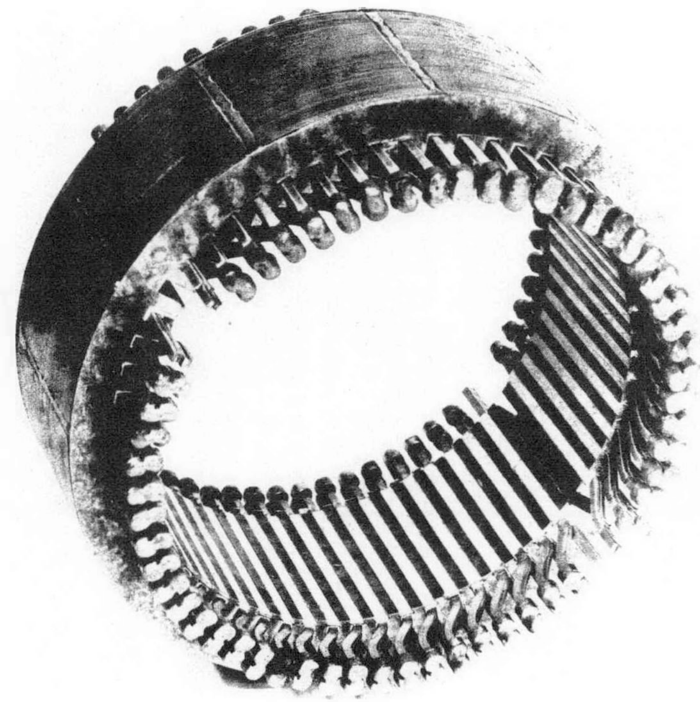
In addition, three ceramic cylinders, shown in the previous photograph, 7 inches in diameter with 0.070-inch thick walls have been fabricated for the LMCD-II generators.

Development of the SNAP 50/SPUR bore seal assembly has been complemented by NASA's program under Contract NAS 3-4162. Construction materials for bore seal assemblies are being evaluated at 1000 and 1600°F for operation in alkali-metal environments. Ceramic materials under test include high-purity alumina, beryllia and rare earth oxides. Refractory metals being evaluated for the metallic section of the assembly include columbium 1 percent zirconium, D-43 and tantalum T-111. Brazing techniques include active-metal brazing and brazing to metallized ceramics. While metallizing with refractory metal can be performed with commercial ceramics, active-metal brazing appears superior for ultra-high purity ceramics. Major tests being performed with brazed ceramic assemblies include tensile tests, peel tests and leak-tightness tests.

Electrical windings of the SNAP 50/SPUR generator will operate as high as 900°F. Because this temperature is approaching the upper limit for winding materials, two statorettes, such as shown in the adjoining figure, are undergoing 10,000-hour aging in a 600°F ambient with winding temperatures of 900°F. One statorette has the winding cavity evacuated to 10^{-5} torr; the second statorette has the winding cavity pressurized with 5 psig argon. Tests will provide data of material degradation with time and of relative heat transfer between pressurized and evacuated stator cavities.

Vacuum aging of Hiperco 27 alloy discs and magnetic ring specimens with coatings of interlaminar insulation have been performed for 1000 hours.

Two USAF programs, AF33(657)10701 and AF33(615)1360, are evaluating electrical conductors and electrical insulation systems for direct exposure to potassium vapor. Success in these programs may alleviate the need for a 10,000-hour, hermetic-tight ceramic bore seal assembly for the SNAP 50/SPUR generator.



ROTOR DEVELOPMENT

- ***OPTIMUM CREEP AND MAGNETIC PROPERTIES AT TEMPERATURE -***
  - MATERIAL AND PROCESS SELECTION
- ***COMPATIBLE WITH POTASSIUM (ROTOR COOLANT) -***
  - EFFECT ON CREEP PROPERTIES; MASS TRANSFER
- ***POLE FACE LOSS AND COOLANT PASSAGE CONFIGURATION***

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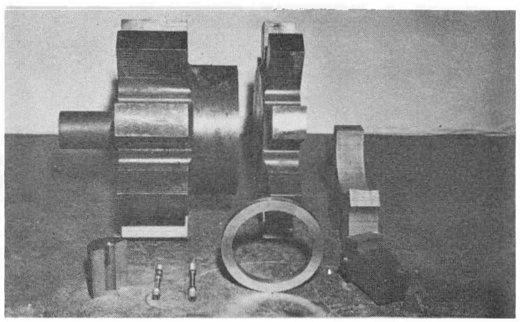
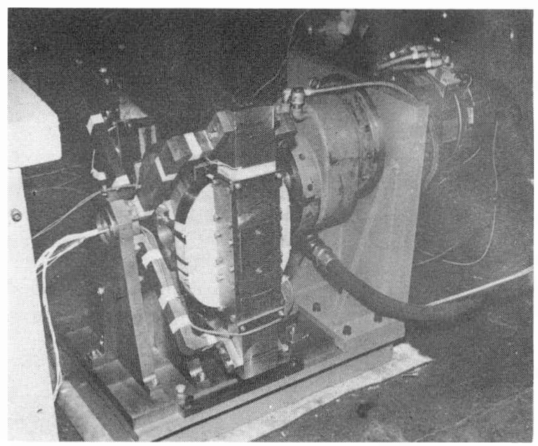
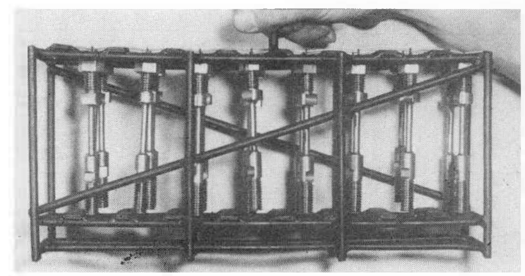


ROTOR PROCESS EVALUTION



POLE FACE LOSSES



H-11 CREEP TESTS IN LIQUID POTASSIUM

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00239

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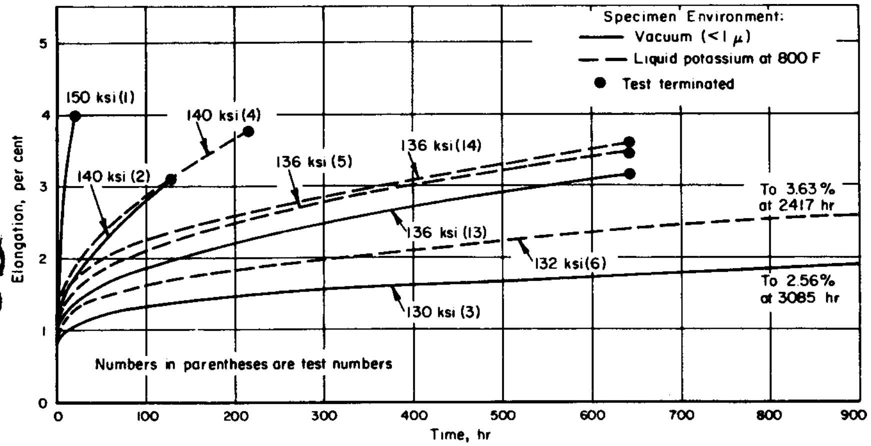
Physical and magnetic properties have been measured, and manufacturing processes evaluated, for the candidate generator rotor materials such as 4340, H-11, and maraging steels. Pole face loss experiments have been conducted to determine the magnitude of the rotor tooth cooling problem. Creep tests of the various materials have been performed in air, potassium, and vacuum to evaluate the effects of potassium on these properties.

The selection of the generator rotor material was based upon the following requirements: (1) magnetic properties, (2) strength and creep resistance to withstand applied stresses during 10,000 hours of operation, and (3) resistance to potassium corrosion at 800°F and 1000°F. The alloy that was initially selected for the generator rotor was SAE 4340. Other candidate materials, AISI H-11 and the nickel maraging steels, were also considered. Therefore, initial material evaluations such as magnetic determinations and creep-rupture strength in both air and potassium environments were made with the SAE 4340 alloy. However, since the AISI H-11 has higher strength properties and magnetic properties at elevated temperatures than SAE 4340, the present rotor development investigation is being performed with the H-11 alloy. Material surveys were also made with the 15, 18, and 20 percent nickel maraging steels, however, the magnetic characteristics of H-11 at elevated temperature was higher than the maraging steels. D.C. magnetization tests were conducted on H-11 to determine the proper heat treatment of the material to yield optimum magnetic properties. Hardness measurements were used as a means of identifying the material condition after heat treatment. From these tests, it was found that the heat treatment which resulted in a material hardness of Rc-45 gave the superior magnetic properties over the operating temperature range of the generator as shown in the adjoining figure. In addition, the effect of exposure time at temperature on the magnetic properties was determined.

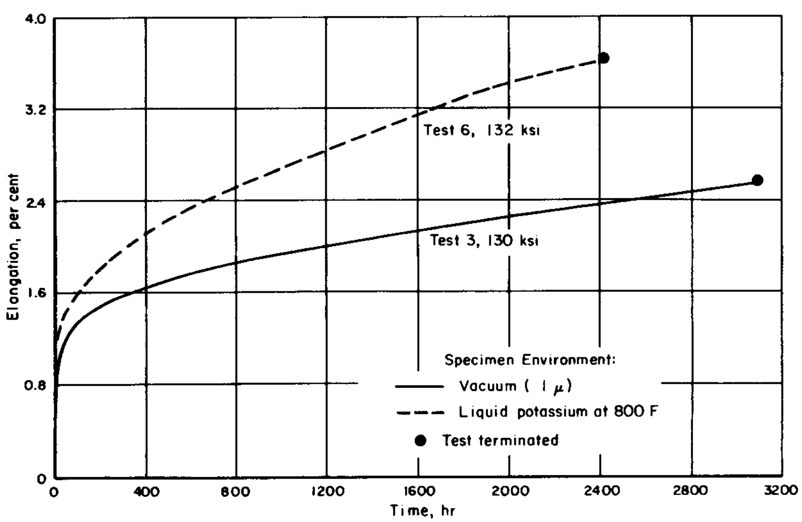
When the heat treatment for superior magnetic properties was established, the mechanical properties of the H-11 in that condition were determined. Tensile and creep tests in air were conducted on H-11 material heat treated to a hardness of Rc-45. Charpy V-notch tests showed no indications of the material being notch sensitive.

After the mechanical properties of the material, heat treated to Rc-45, were obtained, a study was initiated to determine the effect of the potassium environment on the creep properties of H-11 at elevated temperatures. Comparative tests were made in vacuum and in potassium. It was found that at 800°F and 1000°F the minimum creep rate of H-11 was not significantly affected by the potassium. Surface roughness of the test specimens indicates a surface reaction with the potassium environment. Metallography of the test specimens are presently being conducted to supplement the mechanical data. Representative results of magnetic and mechanical property tests are shown on the following pages.

Evaluation of magnetic materials for the SNAP 50/SPUR generator rotor has been complemented under Contract NAS 3-4162. Electrical, magnetic, mechanical and thermophysical properties of numerous magnetic materials have been obtained for operating temperatures which have been tested include H-11, Nivco, Hiperco 27, Hiperco 50, Cubex, nickel maraging steels and Cubex. These material tests have emphasized the higher temperature characteristics and have been coordinated with SNAP 50/SPUR rotor material tests to provide design data.



DEFORMATION-TIME CURVES FOR CREEP TESTS OF H-11 STEEL AT 800 F



DEFORMATION-TIME CURVES FOR LONG-TERM CREEP TESTS OF H-11 STEEL AT 800 F

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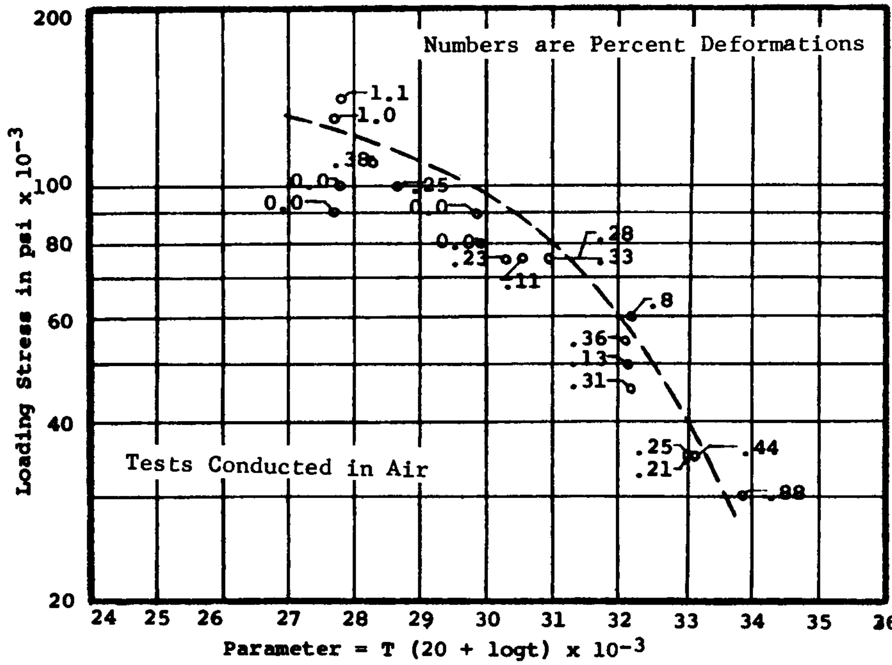
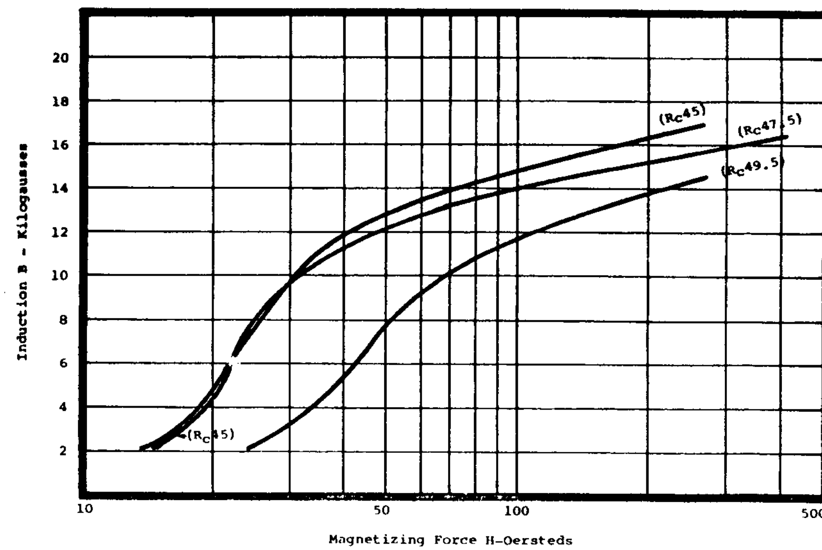


CREEP DEFORMATION FOR H-11 STEEL - R_c 45



COMPARISON OF 700F D-C MAGNETIZATION CURVES FOR H-11 STEEL

GENERATOR DEVELOPMENT DATA - OTHER SOURCES

DEVELOPMENT AREA	SOURCE
● ELECTRICAL DESIGN	STATE-OF-ART HIGH FREQ. HIGH SPEED INDUCTOR GENERATOR 467KVA
● INSULATION SYSTEM	SNAP 50/SPUR NASA SUPPORTED MATL. RESEARCH (WEC.) RTD SUPPORTED (OPEN WIRE, stc) (WEC)
● BORE SEAL DEVELOPMENT	SNAP 50/ SPUR NASA SUPPORTED (WEC) OTHER
● ROTOR DEVELOPMENT	SNAP 50/SPUR NASA MATL. RESEARCH (WEC)
● ROTOR CAVITY EVACUATION	SNAP 50/SPUR

AIRESEARCH MANUFACTURING COMPANY
A DIVISION OF THE GARRETT CORPORATION
PHOENIX, ARIZONA

This figure indicates the sources of data that are being utilized to assist in resolving the major development areas that have been described.

Two development programs at Westinghouse, in addition to the SNAP 50/SPUR generator program, are contributing technology toward successful development of the SNAP 50/SPUR generator.

The first program is sponsored by NASA under Contract NAS 3-4162. The main objective of this program is to define design data for electromagnetic materials as needed for development of advanced, high-temperature electric power systems for space applications. Definition of this design data has first been achieved by correlating existing data, and secondly by performing material tests to fill in the gaps in existing technology.

Many of the planned SNAP 50/SPUR generator construction materials are being tested in this program. However, in general the test program emphasis is on higher operating material temperatures. This is evidenced by the program's test temperature range of 500-1600°F for magnetic materials and 1000-1600°F for ceramic bore seal materials.

Electrical, magnetic, mechanical and thermophysical properties of numerous magnetic materials have been obtained. Materials being tested include H-11, Nivco, Hiperco 27 and nickel maraging steels.

Electrical, mechanical, and thermophysical properties for numerous conductor materials are being obtained. Materials under test include austenitic stainless steel clad zirconium copper, austenitic stainless steel clad silver, Inconel clad dispersion-strengthened copper, Inconel clad silver, nickel clad copper, thorium oxide dispersion-strengthened nickel and beryllium oxide dispersion-strengthened copper.

Two classes of electrical insulation materials are being tested over wide temperature ranges. The first class is organic materials which are being evaluated between -65 and 1000°F. The second class is the inorganic materials which are being evaluated between 500 and 1600°F. Wire insulation, flexible sheet, rigid sheet, molded rigid parts, encapsulation compounds and interlaminar insulations are being tested.

Construction materials for bore seal assemblies are being evaluated at 1000 and 1600°F for operation in potassium, sodium-potassium eutectic and lithium environments.

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Ceramic materials under test include high-purity alumina, beryllia and one rare earth oxide. Refractory metals being evaluated for the metallic section of the assembly include columbium 1% zirconium, D-43 (Cb-10W-1Zr) and tantalum T-111 (Ta-8W-2Hb). Brazing techniques include active-metal brazing and brazing to metallized parts. While refractory metal metallizing can be performed with commercial ceramics, the active-metal brazing appears superior for ultra-high purity ceramics.

The second program at Westinghouse is being performed under USAF Contract AF33(615)-1360 and is directed toward providing an electrical insulation system for withstanding the corrosive attack of alkali metal vapors. Experimental tests will be performed on a single-phase transformer in 1100°F potassium vapor. This contract was preceded by Contract AF33(657)-10701 which evaluated insulated electrical conductors for direct exposure to alkali-metal vapors.

Success in both of these USAF programs may alleviate the need for a 10,000-hour. hermetic-tight ceramic bore seal assembly for the SNAP 50/SPUR generator.

These material programs are producing advanced technological data. Applicable results will be factored into the future SNAP 50/SPUR generator development where improvements can be made.

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VI COMMITTEE QUESTIONS

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VI. ANSWERS TO COMMITTEE QUESTIONS

The preparation of the preceeding presentation has been guided by the topics of interest supplied by the Space Technology Panel. This section is included to provide answers to the questions on the SNAP-50/SPUR program which are not covered in the basic presentation. For those questions answered in the basic presentation the appropriate sections of interest are indicated.

QUESTION 1-b

A weight breakdown by subsystem for SNAP-50, SNAP-8, SNAP-2, and the ORNL Medium Power System, all scaled to the same electric power output - 30 Kw.

ANSWER

The question is answered both for a powerplant based on the SNAP-50 technology which was optimized for operation at 30 Kwe and also for derating the reference powerplant to 30 Kwe.

The weight breakdown and flow schematic for a powerplant designed specifically for a power level of 30 Kwe and utilizing the SNAP-50/SPUR technology are shown in Figs. 87 and 88. A more detailed weight breakdown is given below. The resulting unshielded powerplant weight is 1750 pounds or 58.3 lbs/Kwe. It is also of significance that the total main and auxiliary radiator area is only 150 feet^2.

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	35 Kwe U^{235}	Weight, Lb.		~~35 Kwe U^{233}~~
Reactor	760		760	500
Core		180		
Pressure Vessel and Structu		112		
Reflectors		308		
Control Drives		160		
Lithium Pump	88		75	
Lithium Piping and Auxiliaries	41		35	
Power Conversion System	400		340	
Boiler	47	40		
Condenser	41	35		
Turboalternator	128	110		
Condensate and Jet Pump	87	75		
Static Frequency Converter	47	40		
Piping and Auxiliaries	47	40		
Main Heat Rejection System	263		225	
Radiators	175	150		
Pump	64	55		
Piping and Auxiliaries	23	20		
Auxiliary Heat Rejection System	76		65	
Radiators (Ft^2)	58	50		
Piping and Auxiliaries	18	15		
Control Equipment	175		150	
Structure	117		100	
Unshielded Powerplant Weight	1920		1750	1620
Powerplant Specific Weight, lbs/l	55		58.3	46
Shield: $5 \times 10^{11} - 10^{13}$ nvt in 10,000 hours			2200-1300	

The weight and performance of major components were calculated on the same basis as for the initial reference design. Potential weight reductions may be realized in the reactor by reducing the side reflector thickness from the reference design value of 4 inches due to relaxing the shutdown reactivity restrictions. Further improvement in the heat rejection system through the use of beryllium and the direct condensing concept do not appear to offer significant weight savings for a small system of this type.

00246

Rather than designing a powerplant specifically for the 30 Kwe power level the 300 Kwe reference powerplant could be operated at a derated condition of 30 Kwe. Based upon preliminary studies of powerplant off-design performance the operation of the 300 Kwe powerplant components at this power level appears feasible. For operation specifically at this power level, it would be possible to reduce the size of the radiators and piping and to install a smaller alternator. It is estimated that these units could be decreased in weight by about 1500 pounds from those of the reference design. As shown in Fig. 89, the resulting unshielded powerplant specific weight is about 192 lbs/Kwe. Furthermore, as shown in the schematic, Fig. 90, operation at this derated condition results in a reactor temperature reduction from 2000F to 1577F and a total radiation area less than half that of the reference system. Although the specific weight is greater than for a SNAP-50/SPUR type system specifically designed for this power or SNAP-8, for that matter, it does serve to indicate the versatility of the reference SNAP-50/SPUR design in providing electric power for both the higher and lower power requirements.

30Kwe POWERPLANT BASED ON SNAP-50/SPUR TECHNOLOGY

00248

REACTOR	760
PRIMARY SYSTEM	110
POWER CONVERSION SYSTEM	340
MAIN HEAT REJECTION SYSTEM	225
AUXILIARY HEAT REJECTION SYSTEM	65
CONTROLS & STRUCTURE	250
TOTAL POWERPLANT WEIGHT	1750
POWERPLANT SPECIFIC WEIGHT	58.3 LB/Kwe
SHIELD	
$5 \times 10^{11} - 10^{13}$ nvt IN 10.000 HRS	1300 - 2200

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FIG 87

30 Kwe SNAP-50 FLOW SCHEMATIC

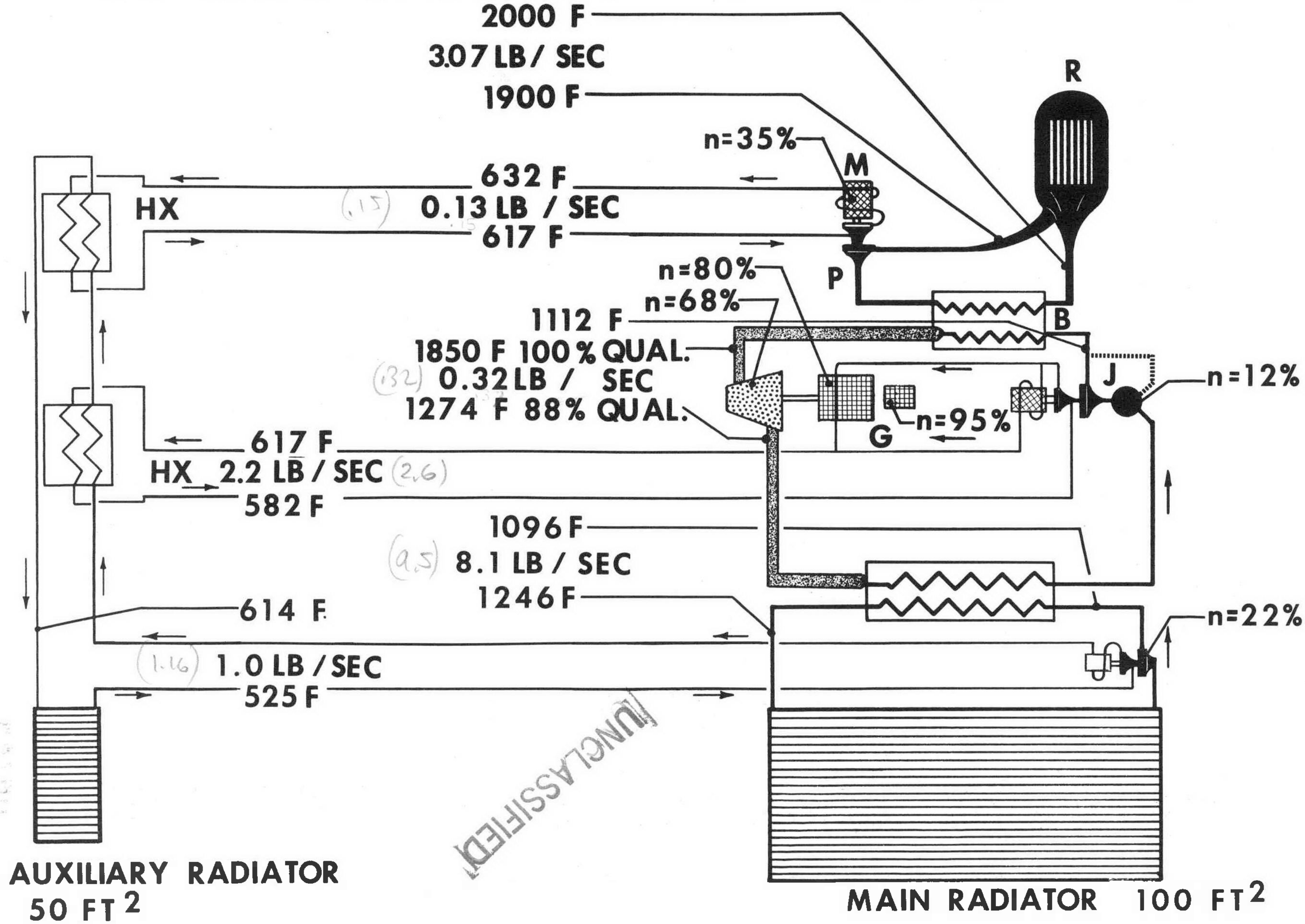

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FIG 88

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300 Kwe SNAP-50/SPUR DERATED TO 30 Kwe

	REFERENCE DESIGN - 300 Kwe	DERATED DESIGN - 30 Kwe
REACTOR	1800	1800
PRIMARY SYSTEM	440	440
POWER CONVERSION SYSTEM	1710	1360
HEAT REJECTION SYSTEM	2615	1440
CONTROLS & STRUCTURE	925	725
TOTAL POWERPLANT WEIGHT	7490 LBS.	5765 LBS
POWERPLANT SPECIFIC WEIGHT	25 LB/Kwe	192 LB/Kwe

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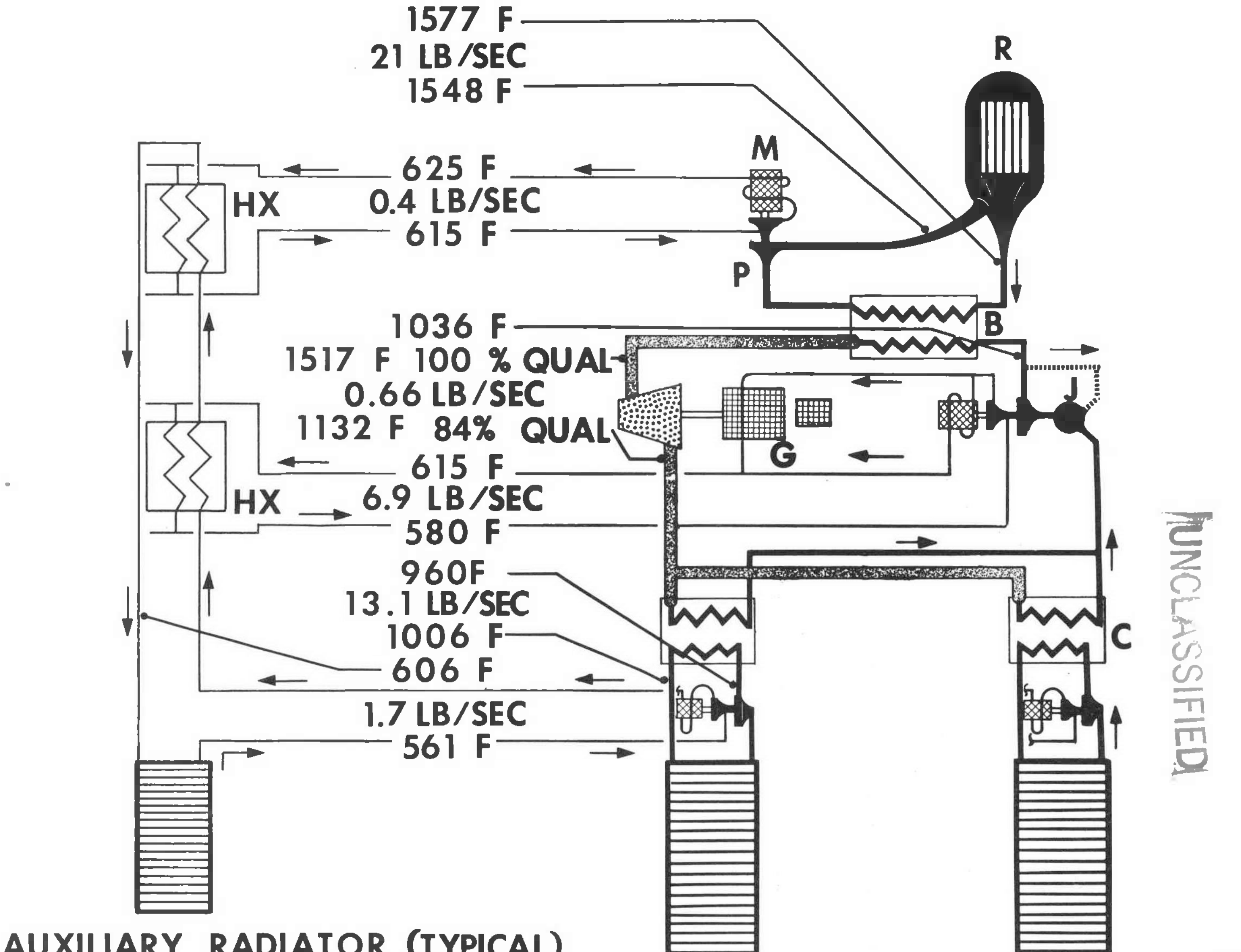
FIG 89

300 Kwe SNAP-50 DERATED TO 30 Kwe



AUXILIARY RADIATOR (TYPICAL)
170 FT ^{2} TOTAL

MAIN RADIATOR 331 FT ^{2}

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FIG 90

00251

QUESTION 1-d

The state of the technology of critical materials (especially tungsten) to include operation temperatures, pressures, life and fabricability.

ANSWER

Regarding the use of tungsten, it is used in the reference reactor design as a compatibility barrier between the fuel and cladding and is discussed in Section V-B on the Reactor and Shield. However, because of its excellent high temperature strength, the alloys of tungsten are of much interest and are being studied as an advanced structural material. The present status of fabricability and availability in useful shapes and quantities precludes the extensive use of tungsten in the reference design. However, if the development of tungsten is successful, it would be considered for improving the SNAP-50/SPUR design in such areas as fuel cladding.

The state of technology of critical materials required by the PWAR-20 reactor is generally well established. Information on this subject has been covered in Section IV - Technological Basis and in Sections V-A, B, C, and D of the Current Program.

QUESTION 4-a

The power ranges over which SNAP-50 type reactor systems can be designed for space operation. The limitations on the high and low end of the power scale.

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ANSWER (Fig. 91)

Although the SNAP-50/SPUR powerplant concept is usually thought of in the power range of 300 Kwe to 1000 Kwe, there appear to be no technical limitations on the lower limit of output power at which this technology is applicable. In general, there is a loss in cycle efficiency as components become smaller due to such factors as bypass leakages and fabrication limits. There is no reason to expect these effects to be more severe for the SNAP-50/SPUR type of system than for any other Rankine cycle powerplant. Furthermore, as reactor power levels are reduced, the reactor size becomes limited by criticality considerations rather than fuel burnup and fission gas containment considerations. For the criticality limited cores, a fast core can be as small as a moderated core, therefore, there is no inherent reactor weight penalty associated with a fast reactor of the SNAP-50/SPUR type. Low powerplant specific weight and especially the reduced radiator area of a SNAP-50/SPUR type of system are of concern for most any space application, even at the lower power levels.

As power levels are extended beyond the current reference design value of 300 Kwe, certain limitations are reached within the present concept. Preliminary reactor studies are currently in progress over the range of 2 Mwt to 40 Mwt which corresponds to electrical outputs of 300 Kwe to excess of 5 Mwe. From these studies, certain conditions such as side reflector cooling, reflector reactivity worth and fuel burnup capabilities which affect the present reactor concept have been observed.

The reference design utilizes side reflectors located external to the reactor pressure vessel and cooled by thermal radiation to space. However, as reactor power is increased, the temperature of the inner region of the side reflectors become excessive using radiation cooling. However, it is possible to extend the power level at which radiatively-cooled side reflectors can be used by altering the reflector heating through larger core sizes as influenced by fuel burnup and length-to-diameter ratio. When this condition occurs, it is necessary to provide forced liquid metal cooling and rotating control drums as were used in the LCRE. The transition from a radiatively to a forced convection cooled side reflector results in a weight increase of about 1500 to 2000 pounds.

The reactivity worth of the side reflector is reduced as core size increases. As shown in Fig. 51 for the larger cores, a condition is reached at which the total side reflector worth is not adequate for start-up and burnup reactivity requirements. For this condition, it is necessary to alter the reactor design concept by providing either poison control rods in the core or movable fuel.

The allowable burnup capability of the fuel material is significant as it affects reactor and shield weights at the higher powers. As has been shown in the Reactor Discussions (section V.B), the UC/UN fuel with a burnup limit of one percent is acceptable for the 2 Mwt reactor design and increasing the burnup limit beyond 1.5 percent does not produce further weight reductions since criticality limits are reached. However, for the 8 Mwt reactor, Fig. 48, weight reductions of roughly one-third can be achieved through extending the burnup limit from 1 percent to 1.5 percent and further extension up to the criticality limit of 4.9 percent results in another reduction of about one third. For power levels beyond 8 Mwt, it is possible to still use UC/UN fuel with burnup limits of one to one and a half percent, however, the weight penalties associated with this fuel over improved fuels are large. Increasing the burnup capability of the fuel through additives to the UC/UN, the use of other fuel materials such as cermets or low density fuel or by employing different design concepts to provide fission gas venting are desirable to reduce the weight of the higher powered reactors.

In addition to limitations in the reactor, there are also size considerations for other parts of the powerplant. Based upon the current 300 Kwe power conversion module size and the use of multiple modules to achieve higher powers, there is a limit to the number of these modules which can be used due to system complexity. We currently believe this limit to be about four. However, one would most likely redesign to a higher power conversion module size. Preliminary studies have been performed on power conversion module sizes up to 1 Mwe and no size limitations have been found over this range.

The limitations on the use of non-deployable radiators at higher powers are dictated by the particular launch vehicle characteristics. The higher power systems which have radiator areas exceeding the spacecraft area as indicated by the booster will require the development of deployment mechanisms and flexible liquid metal piping.

In general, there appear to be no inherent limitations in the SNAP-50 technology for higher powers, however, additional development and changes in design concept are needed in certain areas.

LOW & HIGH POWER LIMITS

LOW POWER:

NO TECHNICAL LIMIT

REDUCED COMPONENT EFFICIENCY

HIGH POWER:

NO TECHNICAL LIMIT

LIQUID METAL-COOLED REFLECTORS

SUPPLEMENT REFLECTOR CONTROL WITH CORE CONTROL

DEPLOYABLE RADIATORS

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FIG 91

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QUESTION 4-b

The problems and advantages that are realized by designing for reactor temperatures over 1600F, over 1800F, over 2000F.

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ANSWER

The effects of operating temperatures are answered in terms of both reactor design considerations and over-all powerplant performance at 300 Kwe and are summarized in Fig. 92.

For operation below 2000F it has been estimated that the reference reactor redesigned for operations at 1600F could be decreased in weight from 1800 to about 1000 pounds. This is based on increasing fuel pin size to permit a greater temperature vairation in the fuel. The reference design reactor can also be used with the outlet temperature reduced to 1600F. The corresponding powerplant weight comparison is shown in Fig. 93. The lower temperatures associated with the derated reference design reactor results in over a threefold increase in allowable strength of the columbium alloy.

For operation above 2000F a reactor at 2200F was studied. Using UC/UN of 95 percent theoretical density and columbium alloy fuel cladding, the reactor weight is estimated to be about 1300 pounds heavier than the reference reactor weight of 1800 pounds. This is due to about a 50 percent reduction in cladding allowable strength (1500 to 800 psi) and an assumed fuel burnup capability of 0.5 percent. At 2200F it appears desirable to use a new fuel concept and cladding material to reduce reactor weight. A reactor weight of about 1650 pounds is estimated for a system using low density UC/UN (80 percent of theoretical) fuel material which can accommodate solid fission products with no external swelling and a stronger fuel cladding. However, due to the low fuel density, nearly all the gaseous fission products are released, thereby requiring a stronger cladding material at this temperature such as a tungsten or tantalum alloy.

A reduction of the reactor exit temperature from 2000 to 1600F would be expected to result in some decrease in development costs, however an increase in temperature to 2200F would increase the development costs. At 2200F, it is doubtful that columbium-1 Zr alloy could be used as the structural material nor any columbium alloy as fuel element cladding. This will require the development of higher strength alloys such as tungsten or tantalum.

It is unlikely that development of the reflector and shield would be significantly affected by changes in either direction from the present 2000F system. Their operating conditions are more directly controlled by reactor power than by coolant temperature. Also, the control drive development currently in the 500-800F range would be unaffected.

The temperature sensors in the reactor system would be strongly affected by higher operating temperatures. At 2000F, the development of W-Re thermocouples is being pursued as a backup to chromel/alumel. The ability to measure temperatures in the 2200F range is presently required in fuel development. At 1600F, it appears that chromel/alumel will meet all requirements. However, at 2200F, candidate materials for temperature levels to 2400F are not available or well known and a significant development effort increase would be required.

The effects of reactor outlet temperature on powerplant performance are shown in Fig. 94. Specific weight is shown both for the reference 300 Kwe powerplant derated in both temperature and power and for SNAP-50 type systems specifically optimized for operating temperatures from 1600 to 2200F. The specific weight penalties for reduced temperature operation are significant for both cases.

REACTOR OUTLET TEMPERATURE EFFECTS

BELOW 2000F

- FUEL BURNUP CAPABILITY EXPECTED TO BE INCREASED
- GREATER CLADDING STRENGTH
- HIGHER SPECIFIC WEIGHTS

ABOVE 2000F

- FUEL BURNUP CAPABILITY EXPECTED TO BE LOWER
- ADVANCED ALLOYS REQUIRED
- SPECIFIC WEIGHT DEPENDENT ON MATERIALS AVAILABILITY

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FIG 92

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POWERPLANT WEIGHT VARIATION

		REF DES.
REACTOR OUTLET TEMPERATURE	1600F	2000F
REACTOR	1800	1800
PRIMARY SYSTEM	425	435
POWER CONVERSION SYSTEM	1700	1660
HEAT REJECTION SYSTEMS	4640	2670
CONTROLS & STRUCTURE	1545	925
TOTAL POWERPLANT WEIGHT. LBS.	10,110	7490
POWERPLANT SPECIFIC WEIGHT. LBS/K_{we}	33.7	25.0

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FIG 93

TEMPERATURE EFFFECT ON POWERPLANT SPECIFIC WEIGHT

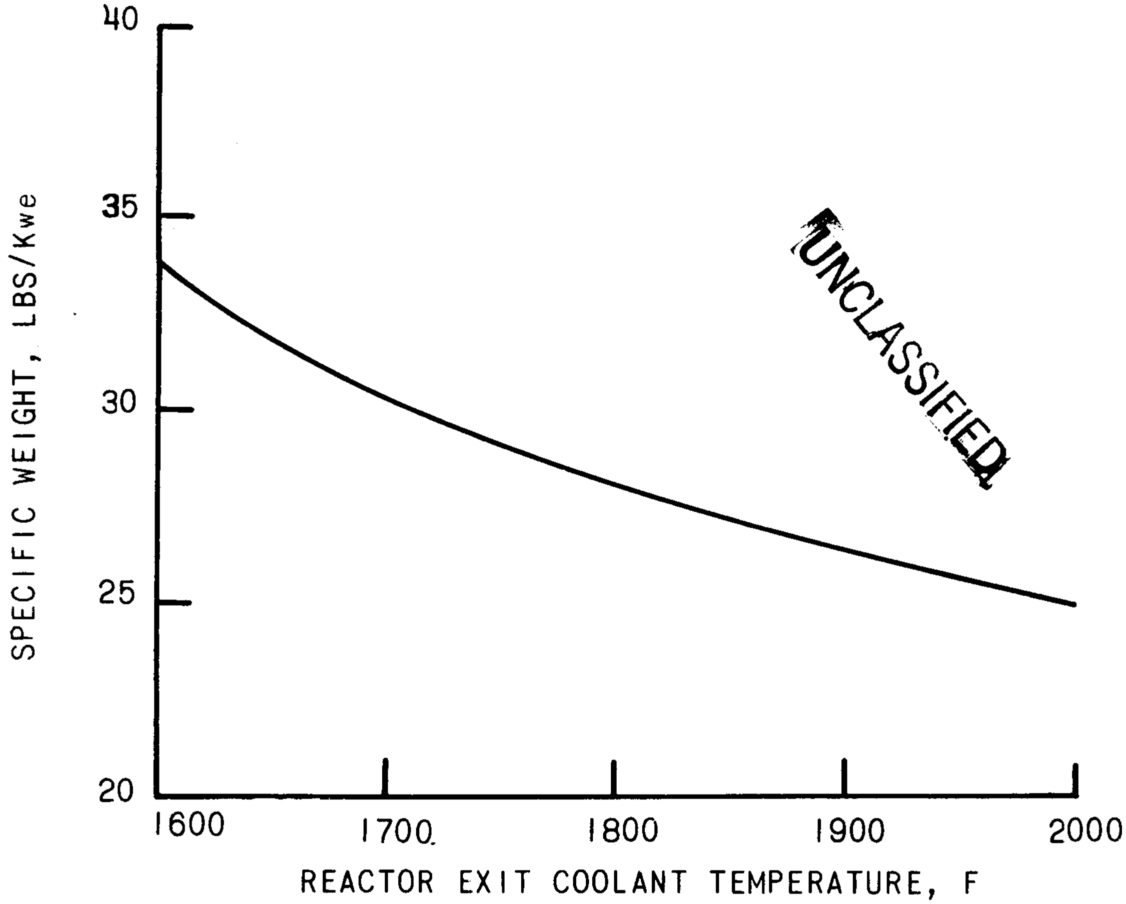

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FIG 94

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QUESTION 4c

The use of the compact potassium condenser and NaK radiator system vs. direct condensing radiator for potassium.

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ANSWER (Fig. 95)

In comparing the indirect-condensing and direct-condensing radiators for application to the SNAP-50/SPUR powerplant, many factors must be evaluated. The major factors to be considered for each of the concepts are the relative weights, the materials requirements, the relative complexity and reliability, and operating requirements.

Weight

The weight of a direct condensing radiator powerplant has been estimated to be approximately 1100 pounds lighter than an indirect condensing radiator powerplant for the 300 Kwe powerplant. These savings come about primarily through the elimination of the NaK system and are contingent upon the use of a "dry" startup and stainless steel radiators in the direct condensing powerplant as is discussed below.

Materials

In selecting a material for either type of radiator, the preference is to use stainless steel as the major radiator material rather than a refractory alloy. Preliminary tests of a stainless steel loop with columbium alloy tab inserts, circulating NaK, indicates that significant corrosion problems are not encountered at the conditions of the SNAP-50/SPUR radiator system. This is due to the relatively low operating temperature level (1096 to 1246F) and small temperature differences (150F). A more elaborate test, simulating the SNAP-50 reference powerplant heat rejection system is currently under test at ORNL and will provide more data on the suitability of stainless steel in the indirect condensing radiator system.

The use of stainless steel in the direct condensing radiator could result in a corrosion problem in the system. There is a larger temperature gradient in the direct system (1274 to 1850F) which provides a greater driving force for mass transfer. Iron, nickel, chromium, carbon, oxygen and nitrogen in a stainless steel radiator are dissolved in the liquid potassium and are expected to come out of solution in the boiler. At this point the carbon, oxygen and nitrogen may react with the columbium in the boiler and the iron, nickel, and chromium either plated on the boiler tubes or carried as particulate matter into the turbine. A calculation, based on solubility data, indicates that the mass transfer will be an order of magnitude greater in a stainless steel radiator than would be experienced in a columbium alloy radiator. Preliminary results from a natural convection boiling potassium corrosion loop will be available later this year to assist in evaluating the extent of this problem.

Segmentation

The relative simplicity of the direct condensing radiator appears attractive and a reliability advantage can be inferred by eliminating the condenser, pumps and NaK accumulators required in the indirect condensing system. However, one consideration to be taken into account is the feasibility of providing segmentation in the heat rejection system to permit part power operation after a failure in a radiator circuit. The indirect condensing system can be readily segmented by splitting the turbine exhaust flow so that it enters separate condenser units and the heat of condensation from each condenser unit is then rejected to space by a separate radiator loop. In addition redundancy can also be readily incorporated into a segmented system such that full power operation can be maintained after a failure in any of the radiator loops.

Two possible approaches to accommodating radiator system failures in the direct condensing system are: 1) separate power conversion machinery modules each coupled to a condensing radiator, or 2) a leak detection system and valving arrangement to isolate radiator segments after failure. The first approach

RADIATOR COMPARISON

RADIATOR TYPE	ADVANTAGES	DISADVANTAGES
INDIRECT CONDENSING	SEGMENTATION POSSIBLE	REQUIRES LARGER AREA
	ALLOWS RADIATOR SEPARATION FROM POWERPLANT	ADDITIONAL LOOP REQUIRED
	REDUCED RADIATOR MATERIAL RESTRICTIONS	
	MORE STARTUP MODES	
DIRECT CONDENSING	LOWER POWERPLANT WEIGHT	LIMITED MATERIALS CHOICE
	FEWER COMPONENTS	SEGMENTATION DIFFICULT
		OFF-DESIGN FREEZING PROBLEM

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FIG 95

would result in a heavier and considerably more complex powerplant. The second approach involves a detection and isolation network that would be extremely complex and could result in substantial weight penalties. The indirect condensing system is more readily adaptable to segmentation and permits operation following failures in the radiator system and therefore is felt to be a distinct advantage for this type system.

Startup and Off-Design Operation

The indirect radiator system can use a "wet" startup (circulation of potassium in the liquid phase though the power conversion loop, initiating boiling by flow and pressure reduction) without a significant weight penalty. The quoted weight advantage of the direct condensing system is realized if a "dry" startup (initiating boiling by injection of the potassium directly into the boiler) is used. The added weight of the potassium required to fill the radiator and radiator piping in a direct condensing system and the large capacity potassium accumulator required with a "wet" startup reduce this weight advantage and indicate a "dry" startup be considered.

With a "dry" startup the reactor control system must be designed to bring the reactor to some predetermined temperature and power level without the benefit of the potassium circuit. Since the power generated in the reactor will be equal to the heat losses from the primary system, the reactor will be operating at a low power level. A large reactor power increase is therefore required during injection of the potassium and initiation of boiling.

Admission of lubricant to the turboalternator and condensate pump bearings must be closely sequenced with the initiation of boiling and turbine rotation in a "dry" startup to prevent filling of the alternator and pump cavities and leakage into the rest of the system. Dynamic seals in these units will prevent leakage of liquid potassium during operation.

In addition the "dry" startup requires that a valve be located at the condenser exit to hold up the working fluid until the condenser pressure has risen to a sufficient level to prevent vapor through flow to the pump. This requires the additional complexity of a sensing and valve actuating system not required with a "wet" start. The "dry" startup which may be required in the direct condensing system is therefore felt to be more complex than the indirect condensing system with the "wet" startup.

During off-design operation, as powerplant electrical power is decreased, and startup of the powerplant system temperature levels are reduced due to the over-capacity of the various powerplant components. The heat rejection rates for the radiators in both a direct and indirect powerplant are approximately equal for any given net electrical power output. Analysis indicates that the potassium leaving the condensing radiator approaches its freezing point as the power output is reduced to close to an idle "condition". This is due to the over capacity of the direct radiators which permit condensation in a relatively small portion of the radiator with the remaining area available for subcooling the relatively low flow of potassium. In the indirect radiator system the NaK radiator temperatures are strongly determined by turbine exhaust temperatures and the much larger NaK flow restricts the subcooling of the potassium. This consideration may limit the minimum alternator output power level obtainable with a direct condensing radiator system.

Vehicle Integration and Testing

Since the radiator is one of the powerplant components which is expected to change configuration with launch vehicle and specific mission requirements, it is advisable to minimize these effects on the rest of the powerplant. To simplify this interface between the powerplant and vehicle contractors the liquid filled indirect radiator system is preferred to the direct condensing radiator utilizing the potassium working fluid. Furthermore, during the development phase of the power conversion system the compact indirect condensers are preferred to reduce facility size and cost.

Conclusions

The current reference powerplant design employs the compact potassium condenser with indirect radiators in preference to the direct condensing concept. Although a weight saving can be shown for the direct condensing concept there are many other factors as indicated above which must be considered. One of the more important factors is the suitability of stainless steel in a direct condensing radiator. Until this is demonstrated experimentally or until the details of columbium radiator construction can be demonstrated without undue weight penalites, it is felt that the reference design should incorporate the indirect radiator.

Answer to Question 4c.;

Use of compact condenser and NaK radiator (indirect system) vs. direct condensing system for potassium (direct system)

Introduction

There is general agreement that, for a given reliability in the 0.90 to 0.98 range, a direct-condensing system offers a substantial potential weight saving over an indirect system. This being the case, it is informative to establish the magnitude of this weight saving, and then make direct-indirect comparisons in terms of the advantages peculiar to indirect systems vs. the problems peculiar to direct systems. This is done in brief form in the following paragraphs.

It is concluded on the basis of the studies summarized herein that:

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1. Use of a direct-condensing radiator in the 300-KWE SNAP 50/SPUR system offers potential weight savings of about 6 lbs per KWE over indirect systems for the same reliability.

2. There are several unknowns about the direct-condensing radiator; no problems are presently known to be unsolvable, but the feasibility remains uncertain due to lack of data, particularly in the area of compatibility of nonrefractory radiator materials with the columbium-alloy conversion loop.

3. The great potential advantage in weight of the direct-condensing radiator provides strong incentive for accelerated development effort to determine its feasibility.

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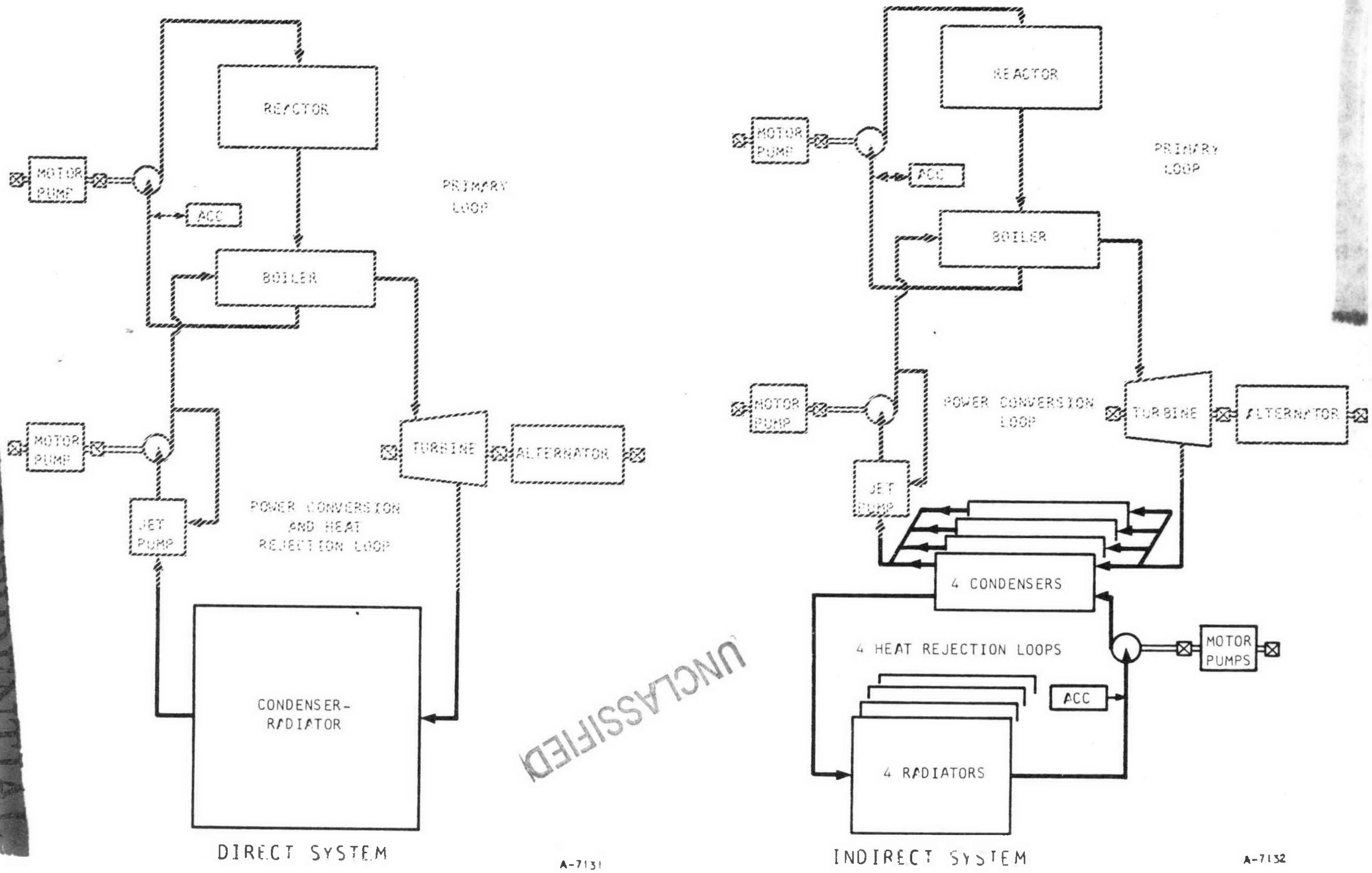
Weight/Reliability Comparison

1. Basis of Comparison

A comparison was made between a direct-condensing radiator system and an indirect four-loop system which has separate liquid loops to transfer the latent heat of condensation from an intermediate condenser to a liquid-filled radiator. A sodium-potassium eutectic (NaK-78) was chosen as the coolant for the indirect system because of its low melting point (10°F vs 357°F for lithium) and because of its compatibility with conventional nonrefractory structural containment materials. The use of a lithium coolant in the indirect system could potentially lead to a weight saving of about 65 lbs; it would, however, require the use of columbium in the radiator, and would increase the complexity of the preheat and startup system.

Schematics of the two systems considered are shown in Figure 4c-1 with auxiliary cooling circuits omitted for simplicity. Ideally, a weight comparison between the direct and indirect systems would involve a comparison of the total weight and reliability of a power system employing a direct radiator with the total weight and reliability of a power system with an indirect radiator. Such a comparison is beyond the scope of this study; however, the comparison made includes all equipment (shown in bold-face on Figure 4c-1) that is directly concerned with the primary heat-rejection function. This equipment includes:



SCHEMATICS OF DIRECT AND INDIRECT HEAT REJECTION SYSTEMS

FIGURE 4c-1

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- Direct radiator system. Weight includes the entire radiator--tube, fin, armor, wet manifolds, tube end reinforcements, ring stiffeners, and miscellaneous hardware.

- Indirect radiator system. Weight includes the entire radiator, as above. In addition, the following auxiliary equipment is included:

	Weight, Lb
Condenser	200
Part of 350 lb associated with NaK accumulator and piping	200
Pump power, 20 kw at 10 lb/kw	200
Increased auxiliary generator weight	100
Pump cooling	70
Pumps	300
Total	1,070

2. Weight and Area Comparison

A detailed weight breakdown of the direct and indirect heat rejection systems is shown in Table 4c-1. The direct system is 1,002 lbs lighter than the indirect for meteoroid nonpuncture probabilities of 0.9 for the direct and 0.54 for the indirect. The effect of reliability on the comparisons is discussed below.

The design details of the direct and indirect radiators are shown in Table 4c-1. The most important physical differences between the two units are in the size and area density, or weight per sq ft. The indirect system requires 690 sq ft, or

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TABLE 4c-1

WEIGHT COMPARISON
(5% MARGIN)

	Indirect System Weight (Lbs)	Direct System Weight (Lbs)
Tubes	293	318
Armor	204	411
Fins (including cladding)	474	430
Manifolds	119	119
Tube end reinforcements	140	205
Ring stiffeners	425	190
Miscellaneous hardware	25	25
Coolant	174	174
Auxiliaries	1,070	0
Total heat rejection subsystems	2,924	1,872

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TABLE 4c-2

DESIGN DETAIL COMPARISON

	Indirect System	Direct System
Area, ft^2	690	563
Area density of tubes, fins, and armor, lb/ft^2	1.41	2.06
Radiator diameter, ft	10	10
Radiator length, ft	22	18
Number of tubes (total, four panels)	1000	1000
Tube ID, in.	0.131	0.185
Maximum tube wall (armor) thickness, in.	0.065	0.110
Minimum tube wall thickness, in.	0.030	0.030
Fin thickness, in. (copper)	0.011	0.013
Fin thickness, in. (stainless steel)	0.004	0.004
Fin effectiveness	0.85	0.85
Probability of at least 3/4 of area surviving for 1 year		
Including meteoroid-induced failure only	0.90	0.90
Including meteoroid and pump failures	0.84	0.90
Probability of full area surviving for 1 year		
Including meteoroid-induced failures only	0.54	0.90
Including meteoroid and pump failures	0.44	0.90

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22 percent more area than the direct, while the tube-fin-armor weight is 16 percent less than for the direct. This leads to an area density, for tubes, fins, and armor, of 1.41 lbs per sq ft for the indirect, compared with 2.06 lbs per sq ft for the direct. This lighter area density, combined with the higher loads resulting from the increased length of the indirect system, lead to stress levels in the indirect radiator during launch about twice as high as in the direct system.

3. Reliability Comparisons

Required Reliability Level

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A detailed reliability analysis of both the SNAP 50/SPUR power plant and the heat rejection system is required to establish reasonable reliability design goals, and to direct the radiator design effort. A previous study by AiResearch (Report SY-5544-R) showed a heat rejection system reliability requirement in the range of 0.95 to 0.98. Table 4c-3 is taken from this report. The first column of numbers in Table 4c-3 lists what were thought to be reasonable, achievable relative failure rates of the components of the system. While all the individual numbers are open to argument, the conclusions that failures in the control system will encompass about half of all the allowable system unreliability, and that failures in the turbogenerator unit approximately an additional one-third, lead to the conclusion that the heat-rejection system reliability must be quite high. The second and third columns of Table 4c-3 give numerical examples of typical reliability levels required for the heat-rejection subsystem. Note that Table 4c-3 is based on a [redacted]p, while the present system uses an electric-motor-driven pump. Accounting for this change would increase slightly the required heat rejection system reliability.

[redacted]

TABLE 4c-3

POWER CONVERSION LOOP (PCL) RELIABILITY

Subsystem or Component	Relative Failure Rate	Required Subsystem or Component Reliability (For 10,000 hr) For PCL Reliability = 0.50	For PCL Reliability = 0.80
Boiler	0.05	0.965	0.989
Separator and Regenerator	0.005	0.996	0.999
Turbine	0.125	0.917	0.972
Generator (w/o voltage regulator)	0.225	0.856	0.951
Heat rejection subsystem (condensing)	0.075	0.948	0.983
Heat rejection subsystem (subcooling)	0.025	0.983	0.994
Pump (shaft driven)	0.005	0.996	0.999
Plumbing	0.02	0.986	0.996
Controls (frequency voltage regulator, etc., but without start-up)	0.47	0.722	0.900
System	1.00	0.50	0.80

[redacted]

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If 7.5 percent of all failures are allotted to the heat-rejection system, then to achieve a power conversion loop reliability of 0.5, the heat rejection reliability, as shown in Table 4c-3, must be 0.948; to achieve power conversion loop reliability of 0.8, the heat rejection subsystem must have a reliability of 0.983. While the numbers in Table 4c-3 are not definitive by any means, they do indicate that allowing for a meteoroid reliability of only 0.9 is not sufficient. The heat-rejection subsystem will have to have a reliability considerably in excess of 0.9, including modes of failure other than meteoroid damage. Further work is required to assess more accurately the modes of failure other than those caused by meteoroids, and to establish more reasonable radiator reliability goals. This will lead to a higher required meteoroid reliability, and thus to increased radiator weight.

Sources of Failure

Failure modes of the direct radiator include the following:

1. Meteoroid puncture of tubing

2. Meteoroid impact on tubes leading to spalling with subsequent damage to pumps or fluid lines

3. Structural failures

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The heat rejection system in the indirect radiator system includes the above-named radiator failures, plus the following additional modes:

1. Failure of pumps to circulate coolant to one or more of the radiator sections.

2. Failure of the electrical power supply to the pumps, leading to the loss of one or more pumps.

3. Failure on the shell side of the condenser, causing loss of radiator coolant in a single panel.

4. Failure on the tube side of the condenser, causing loss of potassium working fluid.

For the purposes of this report, radiator failures due to causes other than meteoroids will be neglected, as they are expected to be of considerably less importance than meteoroids. For the auxiliary equipment of the indirect system, condenser and power supply failures will be neglected. Pump failure rates will be treated parametrically; these "pump failure rates" may be thought of as including other auxiliary equipment in the indirect system.

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Considerable data on reliability experience with liquid metal pumps operating in the 1000-1300°F range at Oak Ridge National Laboratory have been published by Fraas.* These data indicate that pump reliability of about 0.95 for 10,000-hour operation may be achieved. This pump reliability is used as a basis for comparing direct and indirect systems.

Bases of Comparison

Since segmentation, or part-power capability, is apparently not applicable to direct radiators, the direct radiator reliability applies to full-power operation, with the reliability equal to the meteoroid nondamage probability. The indirect systems will be compared assuming both full-radiator and 3/4-radiator survival for one year. The relationship between the probability of all segments surviving and 3 of 4 segments surviving is derived as follows:

If r = reliability of a single loop
f = failure probability of a single loop

Then, $r + f = 1$

If there are 4 loops,

$$(r + f)^4 = r^4 + 4r^3f + 6r^2f^2 + 4rf^3 + f^4 = 1$$

*"Reliability as a Criterion in Nuclear Space Powerplant Design," by Arthur P. Fraas, SAE-ASME paper No. 861A-64, Air Transport and Space Meeting, New York, April 1964.

[redacted]

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The first term in the above expansion, r^4, represents the probability of all four loops surviving, and the second term represents the probability of three of the four loops surviving. The sum of the first two terms is P(3/4), the probability of at least three loops surviving the mission. If the sum of the first two terms is equal to 0.9, the $r = 0.857$, and $r^4 = 0.54$. Figure 4c-2 shows a plot of P(0) required for a given value of P(3/4). In order for the indirect system to have a reasonably high probability of full-power operation for one year, it will be necessary to supply additional (redundant) area, so that full-power operation may be maintained after one segment is lost. Therefore five-segment systems, of which four are required for full power, will be compared with the other indirect and direct systems.

Weight-Reliability Comparison

Figure 4c-3 is a plot of heat-rejection-system weight vs pump failure rate for four types of heat-rejection systems. The lightest is the direct system that utilizes stainless steel and copper. Since this system has no pump, its weight is of course independent of pump reliability. The direct-cycle columbium radiator, while heavier than stainless-steel copper, is lighter than any indirect system. The indirect radiators are the heaviest, independent of reliability. As the reliability requirement increases, so does the weight advantage of the direct system. Table 4c-4 summarizes the heat-rejection system weights for the various approaches, using a pump reliability of 0.95. If the direct radiator at full power is compared with the indirect at three-fourths

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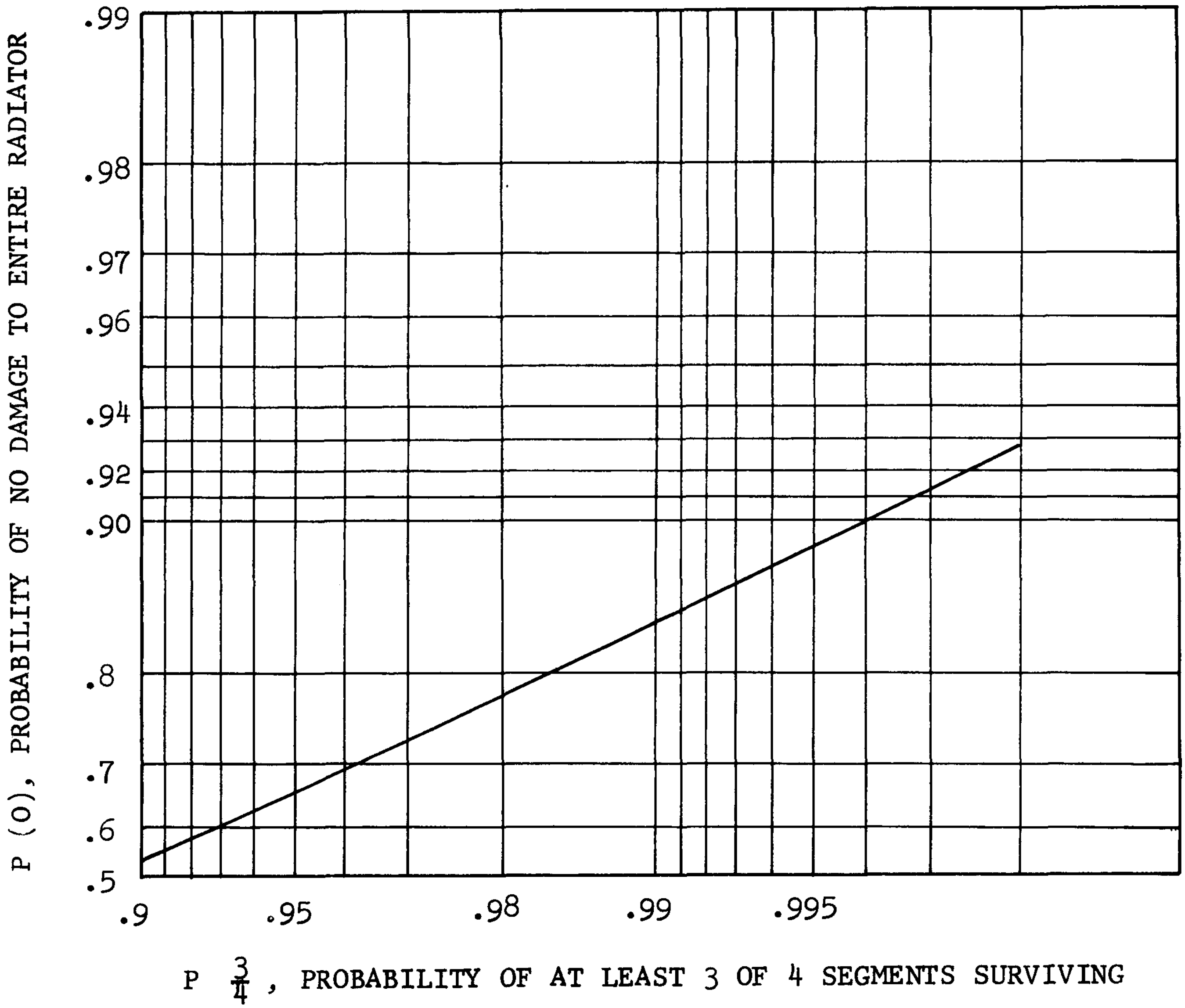

FIGURE 4c-2 [redacted]

PROBABILITY OF NO DAMAGE REQUIRED FOR
A GIVEN PROBABILITY
OF 3 OF 4 SEGMENTS SURVIVING

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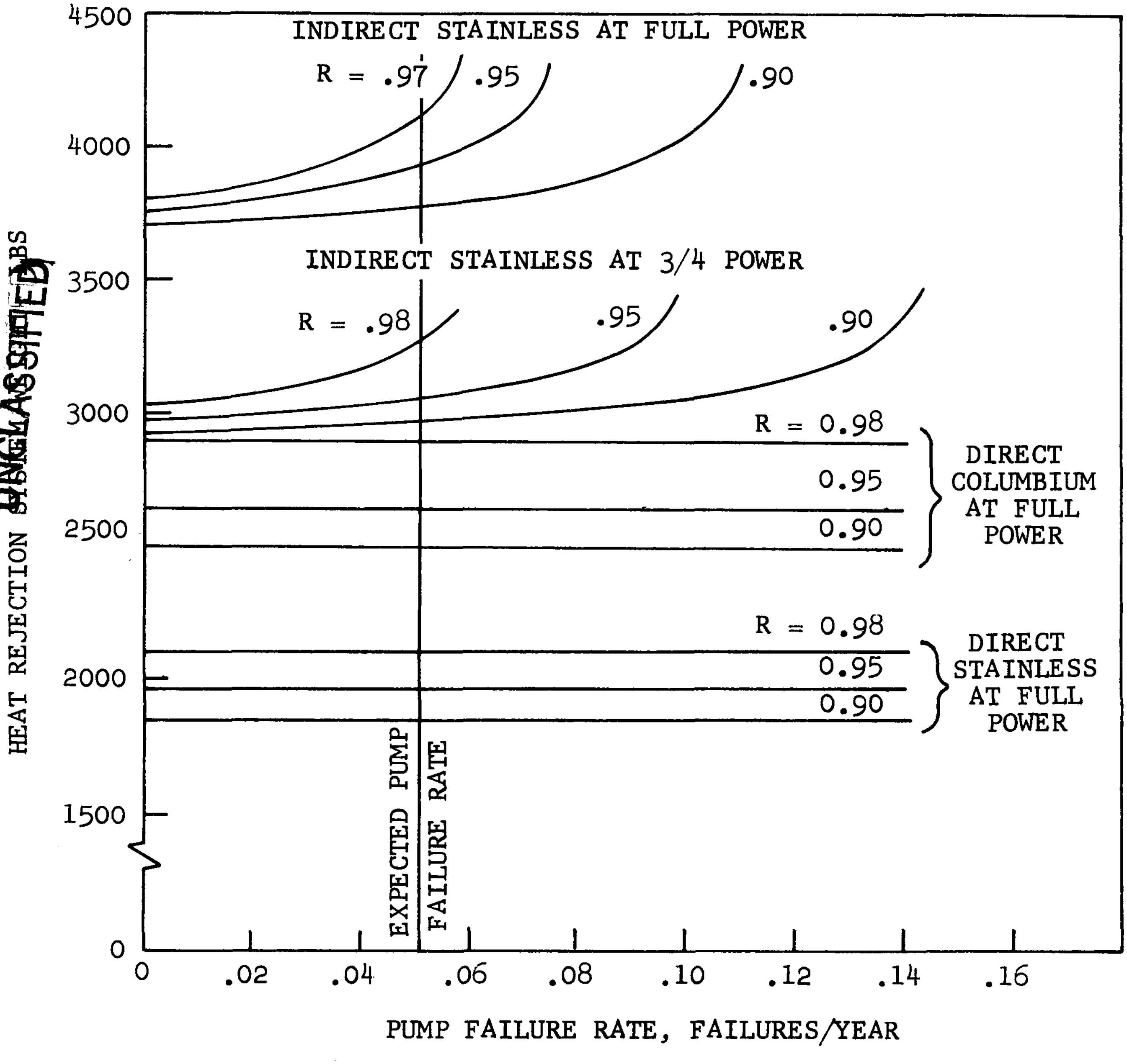

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FIGURE 4c-3

HEAT REJECTION SYSTEM WEIGHT VS PUMP (OR AUXILIARY EQUIPMENT) FAILURE RATE FOR FOUR DIFFERENT TYPES OF HEAT REJECTION SYSTEMS

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TABLE 4c-4

WEIGHT RELIABILITY COMPARISON

Reliability	Hazards Considered	Weight of Direct Heat Rejection System	Weight of Indirect Heat Rejection System	Difference
0.9-full power	Meteoroids only	1872	3695	1823
0.9-3/4 power	Meteoroids only	1872	2924	1052
0.95-full power	Meteoroids and pumps*	1973	3925	1952
0.95-3/4 power	Meteoroids and pumps*	1973	3050	1077
0.98 full power	Meteoroids and pumps*	2115	4110	1995
0.98 3/4 power	Meteoroids and pumps*	2115	3255	1140

*Failure rate of pumps, or of auxiliary equipment, is 0.05 failures/year.

power, the direct is lighter by 1050 to 1150 lb, depending on the reliability. Comparing the two systems, both on the basis of full power capability, the direct is lighter from 1800 to 2000 lb. The absolute value of reliability surprisingly has little effect on either the comparison between, or the absolute value of, the weights of the two systems.

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Discussion of Advantages and Disadvantages

Indirect-system advantages tending to offset its higher weight:

1. Ability to develop the closed power conversion loop independent of the large radiator.

   It appears that much preliminary development work can be done on an indirect system without a radiator, cooling the compact condenser with an auxiliary heat dump system. Eventual ground testing with a radiator will be needed; however, greater flexibility in locating the radiator outside the system vacuum chamber is provided.

2. Greater flexibility in adapting the radiator to various vehicle geometries, because of insensitivity of conversion loop performance to radiator detail design.

[redacted]

The use of an indirect system provides a convenient break between the compact reactor/conversion loop system and the large radiator. This may permit vehicle integrators greater freedom in incorporating the radiator into the vehicle design.

3. Ability to incorporate redundancy of condensor radiator loops as a means of increasing reliability.

As was shown in the preceding section, the use of redundancy in the indirect system does not provide a better weight/reliability picture; in fact, the redundancy is needed to offset the greater unreliability introduced by the pumps. Previous studies have shown that only for very high reliabilties (i.e., 0.999) is the use of redundant loops advantageous. Thus, this apparent advantage is not a real advantage for the SNAP 50/SPUR system.

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Direct-system problems tending to offset its lower weight:

1. More complex fluid management and inventory problems.

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The fluid management comparison has been studied in detail, and is discussed in AiResearch Report L-9443, "Radiators for SNAP 50/SPUR." It is concluded that there are no substantial problems in this area for either direct or indirect systems, and no significant advantage of one over the other in this respect.

The inventory requirement, for a startup scheme which begins with a liquid-filled conversion loop, requires a larger initial inventory in the direct system, with displacement of a larger volume of liquid to a reservoir on startup. This requires a larger potassium reservoir, which has been allowed for in the weight comparisons presented above.

2. Limitations on materials imposed by requirements for compatibility with columbium-alloy conversion loop components.

The choice of radiator materials may be much more limited in a direct system than in an indirect. In the indirect system with a non-refractory radiator, a liquid bimetal loop exists between the radiator and the relatively cool condenser. In the direct system, a two-phase bimetal loop exists between the radiator and the high-temperature boiler, and mass transfer and corrosion problems are expected to be much more severe. Very little is known about this problem; development is required to determine the

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magnitude and nature of the problem, and to evaluate the feasibility of solutions such as use of refractory-alloy-lined tubes in the radiator.

3. Possible freezing of potassium liquid in the radiator at low power levels.

Analysis indicates that condensed liquid in the direct radiator would be cooled to temperatures as low as 200°F at idle power operation; therefore, a possibility of freezing exists. It is expected that this would not be a problem; if some tubes freeze, the remainder will run hotter. When power level is increased, heat conducted through the continuous fin from active tubes will remelt the potassium in the frozen tubes.

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4. More expensive development, due to larger facilities required for testing system with condensing radiator.

It is not necessary to put a full-size cylindrical radiator in a vacuum chamber; chamber size can be minimized by "folding" the direct radiator into a compact package. It is expected that testing a direct system would require a vacuum chamber up to 50 percent larger than that required for the power conversion loop of an indirect system.

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4d "The technological status of the potassium turbine system and plans for further development."

A description of the design, performance objectives, and development areas of the turbogenerator was given in the briefing document. A summary was included describing the turbogenerator development program, and some of the analytical and experimental results generated to date, leading to solution of major development problems. Sources of data - other than the SNAP 50/SPUR program - that are being monitored, and utilized in specific development areas was also described.

In summary the briefing document indicated that the technological status of the bearings, turbine, and seals is presently as follows:

A. Bearings:

1. Data has been generated (principally by Rocketdyne, AiResearch and ORNL) that demonstrates the feasability of operation of potassium lubricated journal bearings at temperatures, speeds, and loads required in the SNAP 50/SPUR turbogenerator.

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2. Data has also been generated (Na, NaK, and K) that indicates the general compatibility of several bearing materials for multiple start/stop cycles under boundary lubrication. Additional data (potassium) will be generated over the next year at AiResearch, NASA and ORNL.

3. The stability characteristic of potassium journal bearings, suitable for the turbogenerator, have not been thoroughly demonstrated.

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However, the orbital computer and verification test program, and the turbodynamic test program at AiResearch, and the conventional fluid test program (NASA) are expected to yield satisfactory solutions to this problem within approximately the next year.

4. Tests to evluate the suitability of bearing designs and material selections for endurance testing in the turbogenerator under oscillatory loads (fatigue) and general bearing wear have not been conducted. It is anticipated, however, that tests will be initiated within the next year in the AiResearch turbodynamic test rig.

B. Turbine

The feasibility of operating turbines on potassium vapor has been recently verified by 3 separate experiments. AiResearch, in August of this year, operated a 20 hp impulse turbine for 5 hours at speeds to 20,000 rpm. ORNL has operated a small potassium turbine intermittently since April of this year. GE, in August of this year made initial runs of their large turbine rig. Early in 1965 AiResearch will begin operation of the SNAP 50/SPUR turbine test rig using potassium bearings and driving a potassium cooled generator.

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1. Thermodynamic property data of potassium is being measured by ONR and BMI. This data will be utilized to improve the accuracy of the aerodynamic designs of the turbine, which have-to date-been based on the calculated data in WADD TR-61-96.

2. Aerodynamic tests of potassium turbine stages are being initiated presently by GE (NASA).

The SNAP 50/SPUR potassium turbine tests are scheduled to be initiated at AiResearch within several months. Results from the turbine program are expected to yield general data on aerodynamic performance and blade erosion of potassium vapor turbines. Also, condensate removal systems, if required to meet performance objectives and minimize erosion, will be developed in this program.

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3. Extrapolated results of elevated temperature creep tests conducted on the molybdenum alloy TZM indicates that a satisfactory stress margin (50 percent of the stress for .5 percent creep) exists for 10,000 hours operation. No significant effects of potassium on the mechanical properties of TZM have been found at elevated temperatures

[redacted]

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(and 1,000 hours). This alloy is considered satisfactory for further evaluation of mechanical properties over extended test periods (5,000 to 10,000 hours)

4. Forging and machining procedures are being evaluated for the TZM alloy in order to establish the suitability of this material for application to the turbine rotor. To data a number of satisfactory forgings have been obtained.

5. Erosion tests conducted in potassium at (1400°F) on a number of candidate refractory alloys indicate that the molybdenum alloy TZM, potentially offers the highest erosion resistance as compared to alloys of Columbium and Tantalum. Acceptable tip speeds for operation in the turbine without potential erosion will not be established until actual turbine tests are conducted, however, analytical extrapolation of erosion theories indicate present design tip speeds to bepotentially satisfactory.

6. Funding limitations have restricted the scope of the turbine material evaluation program summarized above. However, creep tests of several refractory alloys are to be conducted by TAPCO (NASA) at elevated temperatures in vacuum from which an alternate turbine material may be selected in the future.

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C. Seals

Development and research work, applicable to SNAP 50/SPUR turbogenerator seal requirement, is being conducted at various facilities as follows:

1. AiResearch has conducted development work in hydrodynamic seals (SNAP 50/SPUR program) using conventional fluids. Successful sealing against pressure differences of 20 psi, speeds above 24,000 rpm, with acceptable power loss was accomplished utilizing a specially developed shrouded vaned slinger impeller. Tests in potassium have been initiated with these seals in the turbodynamic test program. Further evaluations will continue during this next year.

2. Research and Development work is being conducted at BMI (USAF) to develop long-life contact seals for high speed rotating shafts in liquid metal systems. This work is being monitored, but has not advanced sufficiently such that feasible seal designs and materials can be evaluated for the SNAP 50/SPUR turbogenerator.

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3. A program is being conducted by GE (USAF) for the development of high speed shaft seals for liquid metal systems. Tests have been conducted in water of several feasible designs, and a test system is under construction for evaluations of these designs in potassium.

[redacted]

4. In house development has been conducted by Westinghouse on seals (Holwek type) intended to provide evacuation of the generator rotor cavity.

At the present time, operation of a seal design suitable for the SNAP 50/SPUR turbogenerator, has not been demonstrated in potassium under simulated conditions. However the feasibility of developing satisfactory seals has been demonstrated in conventional fluids, and it is expected that development of satisfactory seals will evolve from the above programs.

QUESTION 4-e [redacted]

The test experience with other critical components and materials, to include problems encountered, probable causes and solutions. A comparison of conditions under which tests were performed with conditions expected in system operation.

ANSWER

The test experience with reactor fuels, reactor structures, corrosion loops, pumps and heat exchangers is summarized in Section IV of this document. A detailed discussion of the reactor fuels, fuel cladding, structural materials and control drive testing is contained in Section V-B. Pump, liquid metal heat exchanger, valve and accumulator experience is presented in Section V-C. The controls and instrumentation work is discussed in Section V-D. The materials experience is discussed in detail in Section V-E.

All the reactor component testing has been accomplished at or near 2000F and is directly applicable to the current reactor design effort. Pump and heat exchanger work carried out under the LCRE program covered a temperature range of 1000F to 2000F. The specific design of these components is not directly applicable to SNAP-50/SPUR; however the technology developed is.

QUESTION 4-f [redacted]

An evaluation of the experience and state-of-the-art concerning high temperature liquid metal operation (especially with respect to corrosion, availability, desired alloys with good quality control.)

ANSWER

Section IV of this document contains a summary of the high temperature liquid metal experience at CANEL a detailed discussion of the structural capability of columbium alloys and its corrosion behavior with the working fluids is contained in Section V-E.

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QUESTION 4-g

The advantages and/or disadvantages of replacing lithium by potassium and columbium by stainless steel in SNAP-50.

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ANSWER

The effects of substituting reactor coolant and structural materials are summarized in Fig. 96.

The replacement of lithium by potassium does not appear to introduce any major changes in the overall concept. Compatibility with the reactor circuit materials and overall nuclear characteristics are essentially unaltered. Some of the less favorable aspects of potassium relative to lithium are:

1. Relative pumping power for a given heat removal rate using potassium is about 44 times that for lithium.

2. The system pressure to suppress boiling and pump cavitation is greater since the vapor pressure of potassium at 2000F is 150 psia compared to 2 psia for lithium.

3. Dose rates from activation of the reactor coolant are nearly comparable for both lithium and potassium however, the decay of activity after shutdown is rapid with lithium due to a less than one second half-life for activated lithium. Although the activation levels can be reduced to negligibly low levels through the use of the potassium-39 isotope, no facility currently exists for separating this material in the quantities required. Admittedly, separation of lithium to 99.9 percent lithium-7 is required for the SNAP-50 reactor system to provide a sufficiently large negative coefficient for the loss of coolant. However, sufficient quantities of this material can be obtained from existing facilities at the Oak Ridge National Laboratory.

Some of the more favorable aspects of potassium include:

1. Reduced system preheat requirements due to the lower melting point of potassium (150F versus 354F for lithium).

2. The compatibility with the fuel materials in the event of a defective fuel element cladding is expected to be marginally improved with potassium.

The direct replacement of columbium alloy by stainless steel is not feasible if lithium is used as the reactor coolant. Stainless steel will not contain lithium for appreciable periods of time at temperatures above 1000F. However, if potassium is used in place of lithium, it is possible to consider stainless steel at a reduced reactor exit temperature.

The use of stainless steel would be favored for the following reasons:

1. Stainless steel alloys' resistance to oxidation and contamination at elevated temperatures permits operation in air. Columbium alloys operating under the same conditions require elaborate oxidation protection.

2. Fabricated stainless steel alloy components would cost less than columbium alloy components because of the lower ingot cost and reduced manufacturing costs. However, it is not expected to result in a major reduction in total development costs.

However, the disadvantages of stainless steel include:

CHANGE OF MATERIALS

LITHIUM TO POTASSIUM

- COMPATIBILITY ESSENTIALLY UNCHANGED
- HIGHER PUMPING POWER
- HIGHER VAPOR PRESSURE
- LOWER MELTING POINT

COLUMBIUM ALLOY TO STAINLESS STEEL

- CANNOT USE LITHIUM ABOVE 1000F; MUST USE POTASSIUM
- REACTOR OUTLET TEMPERATURES LIMITED TO 1400F FOR 10,000 HOURS
- COLUMBIUM ALLOY FUEL CLAD NOT USEABLE
- COMPATIBLE WITH AIR
- LOWER COST

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1. Although the stainless steel can be used in service at temperatures up to 1600F, it is doubtful whether reactor outlet temperature in excess of 1400F for 10,000 hours can be achieved when hot channel effects and film temperature drops are realistically considered. The effects on the powerplant and specific weight, in particular, of reducing reactor exit temperatures to this range can be inferred from the answer to question 4-b.

2. Carbon migration and mass transfer at temperatures in the range of 1200F to 1500F can be expected based on the results of extensive experience with sodium, NaK, (78 K)* and limited experience with boiling potassium** in stainless steel systems.

3. It does not appear advisable to use columbium alloys in a portion of the system, such as fuel element cladding, where high temperature strength is required, and stainless steel in the remainder of the system where strength requirements are less stringent. The columbium will be significantly degraded by absorption of impurities removed from the stainless steel and transported by the coolant.

* Sodium Mass Transfer Test Results GEAP-3726
General Electric Company - Atomic Power Equipment Division, San Jose, California

** High Temperature Bi-Monthly Progress Report CF-63-1-45
Oak Ridge National Laboratory, Oak Ridge, Tennessee

[redacted]

4h. (A) The SNAP 50/SPUR work being done on power conversion system components from the standpoint of how it is related to or differs from the work on higher power conversion components being support be Lewis.

(B) A comparison of the SNAP 50/SPUR and Lewis work from the standpoint of risk, potential performance, and development time.

(A) In our understanding, the SNAP 50/SPUR and NASA/Lewis programs are similar in ultimate objective, since both are aimed at evolving high temperature Rankine Cycle conversion components utilizing potassium in refractory metal systems. These programs differ in philosophy, scope, and timing.

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The SNAP 50/SPUR program is aimed primarily at developing a 300-kw_e powerplant, the overall design of which has been reasonably well established by a reference powerplant design. Thus, SNAP 50/SPUR development work is aimed specifically at solving the problems required for development of this powerplant: general-technology data and data in areas not needed for the SNAP 50/SPUR powerplant development are produced only as by-products.

This approach has the advantage of assuring proper distribution of development effort to the various components, the relevance of the development effort to system requirements and provides additional assurance that all aspects of the system and its operation are being considered in the program.

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By contrast, we understand the NASA/Lewis work as being a series of general component technology programs, related only by the general concept of application to high-temperature megawatt-level powerplants. Based on this approach, the NASA/Lewis programs appear to be directed toward developing a broad base of technology which will support the initial design and tests of components in the future.

The following activities can be utilized to illustrate some of the specific differences in the SNAP 50/SPUR and Lewis programs.

(1) Boiler: In the SNAP 50/SPUR program a boiler design was evolved from the reference powerplant design using available and extrapolated data. Major heat transfer development areas were derived and work initiated in these areas, principally heat transfer in high quality flows, and boiling stability in parallel tubes. The parallel tube stability problem is not included in Lewis programs, and the heat transfer data at high quality flow was either not available or not adequate because of the size and configuration of the test sections.

Therefore, SNAP 50/SPUR boiler development is concentrated in these two areas by analysis and tests in the freon visualization, single tube boiling potassium, and 400 kw HS25 boiler test programs as described in the briefing document.

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(2) Radiator: The SNAP 50/SPUR program has included analytical and supporting activities required to verify the design which is described in the briefing document. This design utilizes state-of-the-art data in the areas of tube selection, fin and structural design. Material selections, such as beryllium meteoroid armor, which may offer some overall powerplant improvement, but which introduce significant technological problems, were specifically excluded in the SNAP 50/SPUR program. The Lewis programs do, however, include programs to evaluate these materials, which if successful, can be easily incorporated into the SNAP 50/SPUR program. No work is included in the SNAP 50/SPUR program to evaluate materials for meteoroid armament or radiator emissive coatings, since work in these areas is felt to be either adequately covered by Lewis, or otherwise availabe, and is being factored as needed into SNAP 50/SPUR.

Another comparison is that the SNAP 50/SPUR radiator concept has evolved as a cylindrical shell with structural capability, based on minimizing system weight, while NASA/Lewis is testing radiator concepts which do not provide structural capability.

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(3) Turbine: The SNAP 50/SPUR and Lewis programs do not essentially differ with respect to definition of problem areas or to test approach in the areas of aerodynamic performance and blade erosion, except for the blade material erosion program in the SNAP 50/SPUR program, as described in the briefing document.

In the area of rotor development the SNAP 50/SPUR program has included tests to determine the suitability of materials (creep and fatigue) for application to the turbine rotor, including the effects of potassium on critical properties. This work was extended in programs, as described in the briefing document, intended to establish the feasibility of forging and machining the selected material into complex shapes required for the turbine rotors, and for determining the mechanical properties of these forgings. The Lewis program has initiated programs to evaluate creep and fatigue properties in vacuum of several refractory alloys selected from plate stock (except a TZM alloy) for the turbine rotor. No manufacturing (forging and machining)studies and no mechanical property evaluations of forgings are included in these programs. Recognizing that differences exist between the mechanical properties of forgings and plate or bar stock, it will be necessary to conduct some retest of selected forged materials. The effects of

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potassium on the creep and fatigue of selected alloys is not being conducted on these materials.

(4) The SNAP 50/SPUR bearing development includes an extensive analytical program to evaluate stability margins of bearings and of bearing rotor systems, supported by a conventional fluid verification test program.

The Lewis programs have included stability tests of various bearing designs. However, the Lewis programs have not included sufficient analytical effort to support the experimental program adequately. Wear (erosion and fatigue) of selected bearings is to be evaluated under simulated conditions in the turbodynamic test rig as part of the SNAP 50/SPUR program. These are not included in the Lewis program.

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(5) The SNAP 50/SPUR program presently incorporates motor driven pumps for the lithium, potassium, and NaK circuits. Development of EM type pumps will require significant advances in technology before these types can be incorporated into SNAP 50/SPUR as a potential powerplant improvement. Development activity on these types is not therefore included in the SNAP 50/SPUR program. An EM pump technology is supported by NASA/Lewis and results will be utilized where possible in the SNAP 50/SPUR powerplant.

(B) A comparison of the SNAP 50/SPUR and Lewis work from the standpoint of risk, potential performance, and development time.

(1) The Lewis programs have no true risk because these programs at this time are strictly investigative in nature.

(2) It is the belief of Lewis that risk exists in the SNAP 50/SPUR program to the extent that a satisfactory containment material (Cb-1 Zr) is not available for a lithium-potassium powerplant.

It is our conviction that, as an example, the demonstration of 9,600 hours of operation of the Cb-1Zr, lithium - NaK loop at CANEL confirms the suitability of this material for SNAP 50/SPUR.

(3) The "risk" in the SNAP 50/SPUR program is therefore simply the degree to which the predicted component performance can be attained. The financial aspect of this risk is being minimized by development of the conversion loop components in non-refractory test systems to obtain test data on a number of important items (bearings, seals, generator, turbine rotor, etc.) under conditions duplicating those of the final refractory alloy components. This approach simplifies test procedures and provides the bulk of the data needed for finalizing the design of refractory alloy components. The

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non-refractory metal component development provides additional benefits, and less overall finicial risk since the resulting test data and experience can be applied directly to system of lower output (i.e. MPRE) or of lower operating temperatures.

(4) No risk is believed to exist in the SNAP 50/SPUR program, that by adopting a Reference Powerplant Design of 300-kw output, requirements for powerplants of greater output cannot be met.

The conversion loop components of the Reference Design can be applied to powerplants in the megawatt class since it is generally agreed that large powerplants will consist of multiple conversion loops operating in parallel, instead of a single component system--principally for reliability, integration, and weight improvements. As an example, in the Lewis supported program at GEMSD, (fourth quarterly report of Contract NAS3-2533, Page 6-46), important advantages for the multiple component approach are shown for 1.2 mw, and larger, systems.

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Therefore, the components that evolve from the 300-kw SNAP 50/SPUR conversion loop development program can be utilized essentially without change for powerplants to over 1200-kw output.

(5) The SNAP 50/SPUR conversion loop development program is aimed at achieving component performance goals derived from a reference powerplant design. As the major emphasis is being placed on solving development problems required to meet these goals, it is expected that the performance and development time required to achieve, or establish, a usable powerplant are inherently better in the SNAP 50/SPUR program.

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QUESTION 4-i [redacted]

The planned future program.

ANSWER

This information is provided for the overall program in Section III, Over-all Program Parts A and B. The future program for each of the powerplant components is discussed throughout Section V, Current Program.

QUESTION 4-j

The structure of costs for various operations at CANEL.

ANSWER

This information is provided in Section III-C, Over-all Program.

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Question 9. Space Radiator Development

a. Magnitude of the development effort underway:

No SNAP 50/SPUR radiator development is presently funded. Approximately $20 - 30,000 per year is currently being invested in analytical studies and small-scale experiments to provide system design verification data.

b. Manner in which work is factored into power system design and development.

Results of studies noted above are used in system optimization studies.

c. Major problem areas.

The radiator is a very large heat exchanger designed for minimum weight; therefore, it has substantial development problems in fabrication, structural design, thermal stress, etc. of the type common to any large high-performance heat exchanger. In addition there are several problem areas peculiar to the SNAP 50/SPUR radiator:

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1. Evaluation of direct vs indirect heat rejection systems. As explained in the answer to question 4c, this hinges largely on the problem of material compatibility in a two-phase bimetallic potassium loop. Investigation of this problem is required to resolve current uncertainty in system weight of up to 6 lb/kw, resulting from uncertainty as to whether, and in what form, a direct-condensing radiator can be used.

2. Evaluation of the meteoroid hazard and the selection of meteoroid armor material.

Presently-existing variation among reported meteoroid flux velocity, and density characteristics; among penetration theories, and among armor material evaluations, imply uncertainty of about 1 lb per kw in SNAP 50/SPUR system weight.

3. Condensing stability and fluid dynamics.

Although preliminary indications are encouraging with regard to the inherent stability of the condensing process, more data is required. In addition, means must be developed to make the operation of the condenser, whether a compact unit or a condensing radiator, insensitive to gravity, so that ground testing will simulate

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space operation. This must be done with minimum increase in condenser pressure drop, as the system is sensitive to pressure losses in the low-pressure part.

4. Fabrication

This general area includes development of improved techniques for applying stainless clad to copper fins, of bonding fins, armor, and tubes together, of applying emissive coatings, of uncorporating tube liners if required, and of making tube/manifold joints.

5. Structural Design

Development is required to maximize the effectiveness of the cylindrical radiator matrix as a structural support for the power system and launch vehicle nose cone.

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d. Relative developability of liquid metal and condensing metal vapor radiators.

This question is answered under Question 4c, which concludes that there is no known reason why either cannot be developed and that the large weight advantage of a direct-condensing radiator provides strong incentive for its development.

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e. Materials most promising for radiators and their compatibility with NaK and potassium.

As discussed more fully in the briefing document and under Question 4c, the SNAP 50/SPUR reference design materials for either liquid NaK or condensing potassium is a stainless-clad copper fin, stainless armor, and stainless tube. The tube material, Type 318, is expected to be quite compatible with liquid Nak in a bimetal system at the anticipated temperatures. Its compatibility with a bimetallic (stainless plus columbium alloy) two-phase potassium system is more questionable, and requires investigation. If problems exist here, the use of columbium or columbium-alloy tube liners is a promising possibility.

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L-9391	SNAP 50/SPUR CONDENSER COMPUTER PROGRAM

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CANEL FACILITIES

The Connecticut Advanced Nuclear Engineering Laboratory (CANEL) is located on the west bank of the Connecticut River five miles south of Middletown (Figs. A and B). It is a wholly Government-owned facility which is operated by the Pratt & Whitney Aircraft division of United Aircraft Corporation for the U. S. Atomic Energy Commission. It was constructed in 1957 by the U. S. Army Corps of Engineers for the Aircraft Nuclear Propulsion program. Today CANEL is a $70, 000, 000 integrated complex which includes 17 major laboratories and buildings totaling nearly 750, 000 square feet and occupying more than 1, 000 acres of land (Fig. C).

1. Administration Building

   This building provides space for the majority of the office personnel connected with the program, including engineering groups, design groups, library, AEC offices, cafeteria, purchasing, personnel and employment offices, administrative offices, accounting and treasury offices, and the photographic laboratory.

2. General Laboratory

   This facility houses the chemistry, metallurgy, health physics, instrumentation, electronics, and computing laboratories. It is constructed in a manner to provide space for research and small scale technical and development work of a relatively clean and low hazard nature requiring only small amounts of power.

3. Shop Laboratory

   This facility is designed for the fabrication and testing of a wide variety of propulsion system and reactor components and materials. The building is generally open and, therefore, quite flexible with respect to function and change. Rigs can be set up in a relatively short time, tests conducted, and rigs modified or dismantled to make way for other tests.

4. Fuel Element Laboratory

   In this building fuel materials are prepared and processed in batches and are hand assembled and welded. This facility has been designed with careful attention to the consideration of health and nuclear safety, contamination control, scrap recovery, accountability, and security.

5. Machine Shop

   This facility houses the varied machining, fabricating, mechanical inspecting, and assembling functions associated with the development program. Included in the building are facilities for close-tolerance machining of test rig, reactor, and engine components, experimental plating, welding, sheet metal working, heat treating, and assembling. A walk-in inert gas welding and annealing facility and a 7, 000 square foot clean room afford unique capability for the fabrication of large scale refractory metal components and assemblies. Also included are provisions for X-ray and constant temperature inspection.

6. Nuclear Physics Laboratory

   The Nuclear Physics Laboratory is a facility designed and equipped for conducting critical experiments and associated functions. The building consists of two high bay test cells, control rooms, assembly rooms, storage vaults, counting room, laboratories, locker rooms, and office area.

7. Hot Laboratory

   The Hot Laboratory is a facility whose primary function is to carry out post-irradiation tests and examinations of materials and components that are essential to the current research and development program. Provision is also made for processing for disposal of all radioactive waste from the CANEL site.

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CONNECTICUT ADVANCED NUCLEAR ENGINEERING LABORATORY

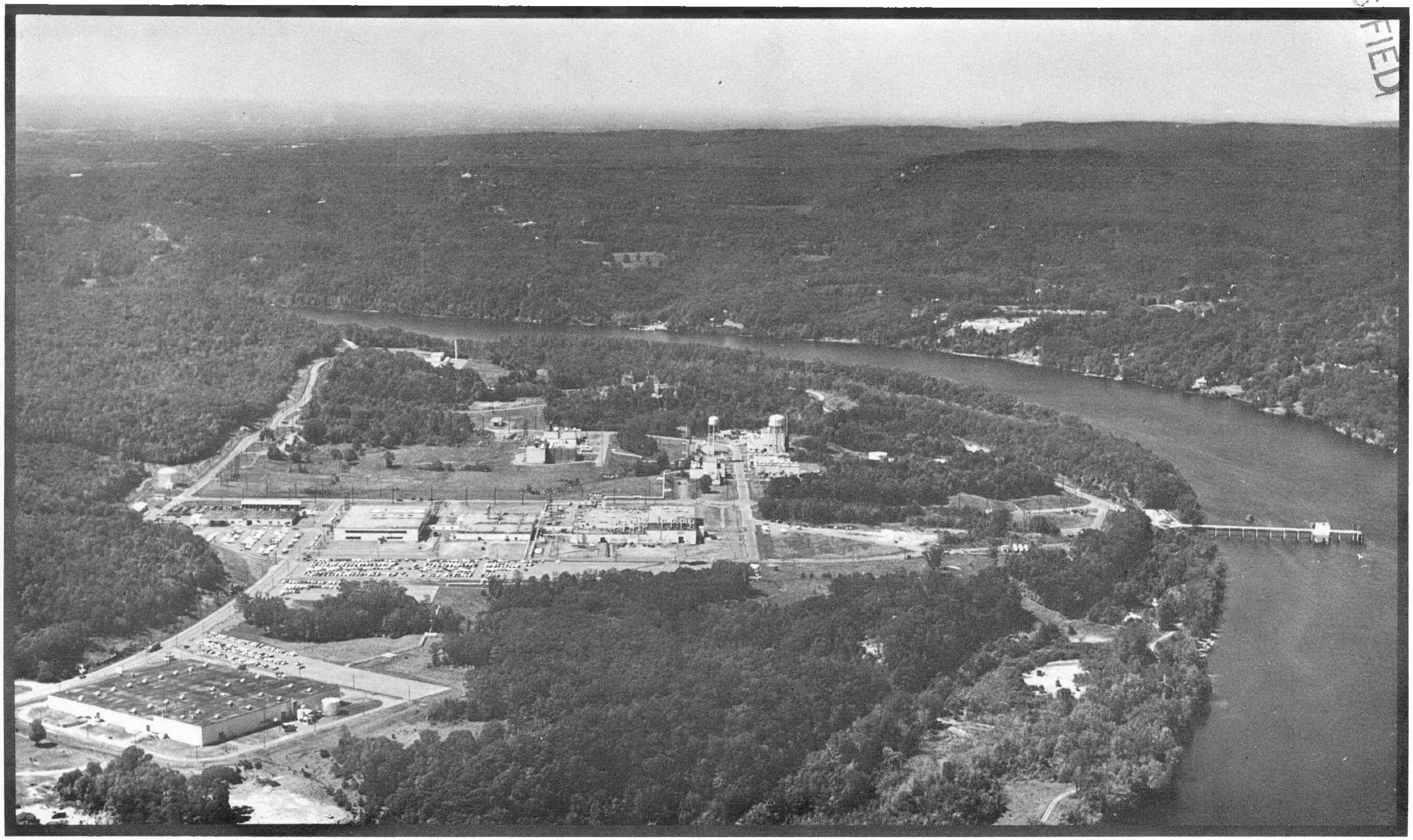

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FIG A

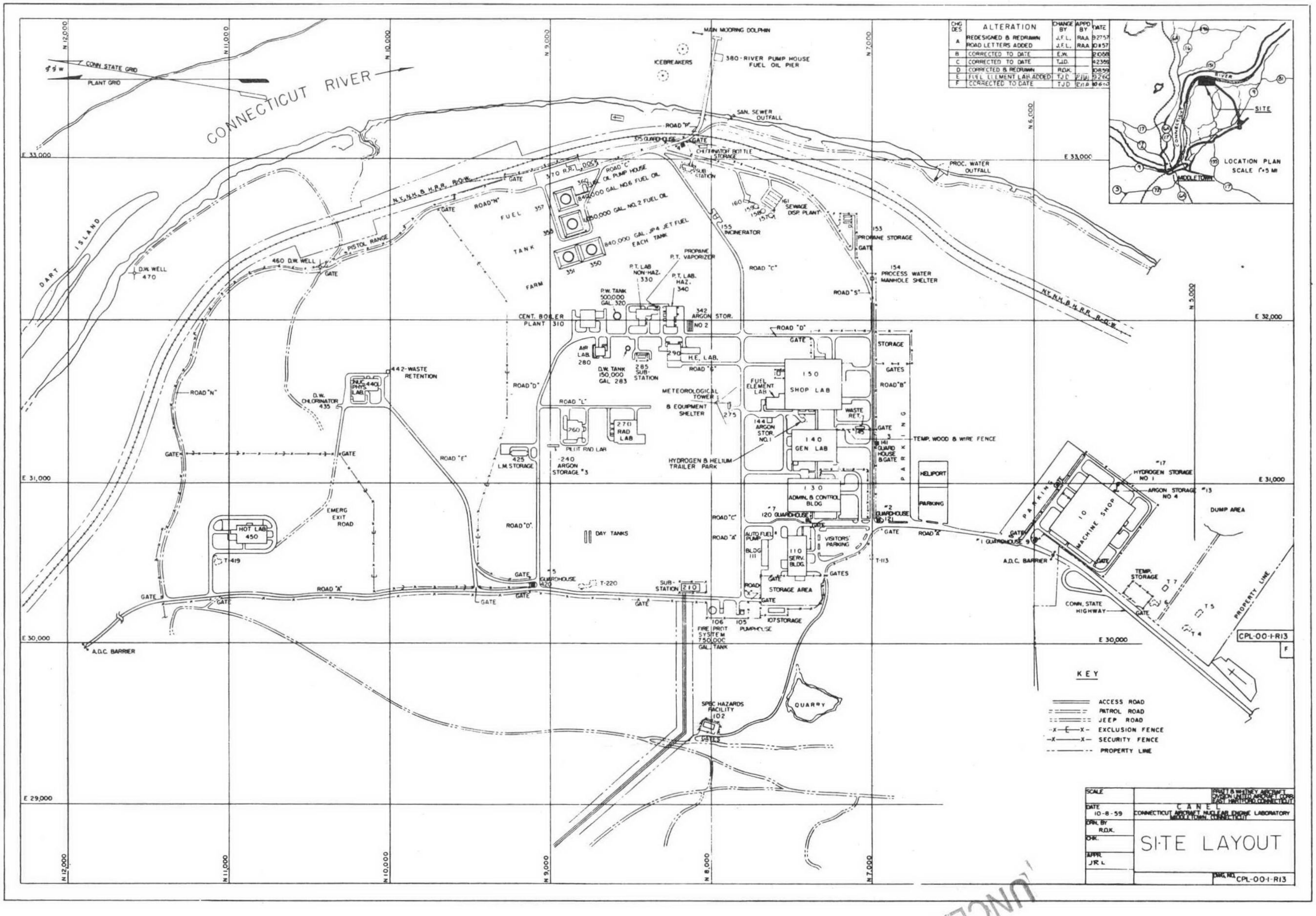

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FIG B

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FIG C

CANEL COST SUMMARY

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Facility	Real Property	Capital Equipment	Total	Floor Area Square Feet
Site (1, 027 Acres)	$ 2, 734, 220 (1)	$ 13, 137	$ 2, 747, 357	
Administration	2, 534, 314	1, 424, 729	3, 959, 043	151, 200
General Laboratory	2, 087, 796	3, 096, 871	5, 184, 667	57, 376
Shop Laboratory	3, 456, 937	5, 575, 168	9, 032, 105	115, 784
Fuel Element Laboratory	501, 010	633, 927	1, 134, 937	19, 130
Machine Shop	2, 614, 407	6, 458, 420	9, 072, 827	154, 983
Nuclear Physics Laboratory	1, 567, 778	398, 876	1, 966, 654	31, 600
Hot Laboratory	1, 785, 783	656, 863	2, 442, 646	34, 392
Pump Turbine (Hazardous) Laboratory	873, 985	169, 528	1, 043, 513	9, 527
Pump Turbine (Non-Hazardous) Laboratory	1, 270, 663	306, 875	1, 577, 538	12, 482
Air Laboratory	1, 014, 805	223, 869	1, 238, 674	7, 213
Heat Exchanger Laboratory	1, 582, 636	1, 189, 386	2, 772, 022	17, 747
Pilot Radiator Laboratory	5, 284, 729	368, 449	5, 653, 178	13, 414
Radiator Laboratory	4, 875, 299	373, 014	5, 248, 313	34, 643
Meteorological Laboratory	61, 650	34, 350	96, 000	604
Special Hazard Facility	49, 198	34, 233	83, 431	1, 470
Central Power Plant	2, 924, 915 (2)	130, 086	3, 055, 001	24, 276
Service Building	392, 816	714, 331	1, 107, 147	33, 216
Automotive Equipment Shelter	29, 518	126, 867	156, 385	5, 184
Liquid Metal Storage Building	148, 051	28, 552	176, 603	3, 420
CAFFEE Building	9, 563	600	10, 163	1, 536
Welding & Annealing Facility	989, 761	---	989, 761	8, 176 (3)
Clean Room Facility	390, 928	---	390, 928	7, 708 (3)
Computer Facility	---	848, 505	848, 505	1, 613 (4)
Fuel Farm & Fuel Oils System	795, 481	994	796, 475	1, 104
Primary Electrical System	3, 085, 303	132, 294	3, 217, 597	---
Communications System	40, 771	---	40, 771	---
Fuel Oil Pier & Pump House	1, 614, 270	3, 669	1, 617, 939	1, 561
Domestic Water & Fire	1, 329, 322	1, 024	1, 330, 346	1, 274
Process Water System	1, 635, 525	210	1, 635, 735	---
Sanitary Sewage System	299, 907	1, 640	301, 547	375
Miscellaneous Construction (In Process)	539, 000	---	539, 000	---
Equipment (Undelivered)	---	287, 000	287, 000	---
Other (5)	505, 808	181, 262	687, 070	920 (6)
	$47, 026, 149	$23, 414, 729	$70, 440, 878	734, 485

(1) Includes Access Road Construction Cost at $683, 982
(2) Includes Steam Distribution System
(3) Facility Located in the Machine Shop - Floor Area Not Added to Total
(4) Facility Located in the General Laboratory - Floor Area Not Added to Total
(5) Installation, Rehabilitation, Freight, Cancellation Charges, Plant Clearance Costs, Write-Offs and Equipment Located at Laboratories Other than CANEL
(6) Guard Houses

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8. Pump Turbine Laboratories

The Pump Turbine Laboratories have been designed to accommodate the endurance testing of full scale liquid metal pumps. The buildings include two 50 x 30 foot cells for testing full scale liquid metal pumps, three 35 x 20 foot cells for research testing of reduced scale liquid metal pumps, and four test cells for water testing of reduced scale and full scale liquid metal pumps.

9. Heat Exchanger Laboratory

The Heat Exchanger Laboratory is a facility for development testing of liquid metal-to-liquid metal heat exchangers. It has been designed to accommodate a program of steady state performance and endurance tests as well as transient thermal cycling and shock tests. A 6 Mw electrical power supply is presently available in this laboratory for lithium-columbium heat exchanger and liquid metal boiler tests.

10. Pilot Radiator Laboratory and Radiator Laboratory

These laboratories were designed to test full scale liquid metal-to-air radiators and sections of radiators for the Aircraft Nuclear Propulsion Program.

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