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## ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

Quarterly Progress Report  
1 October through 31 December 1964

15 February 1965

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

QUARTERLY PROGRESS REPORT

1, OCTOBER THROUGH 31 DECEMBER 1964

Published

15 February 1965

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Approved by: *RH Chesworth*

R. H. Chesworth  
Supervising Representative  
Contract AT(10-1)-880

## AEROJET-GENERAL NUCLEONICS

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

QUARTERLY PROGRESS REPORT\*

1 October Through 31 December 1964

ABSTRACT

This document summarizes the technical progress of the Army Gas-Cooled Reactor Systems Program under Contract AT(10-1)-880 between the U. S. Atomic Energy Commission and Aerojet-General Corporation to develop a mobile, low-power, nuclear power plant for Military field operation.

The ML-1 power plant was operated for 1509.53 hr during the report period; the overall on-stream factor was 68.36%. The thermodynamic performance test, several shielding evaluation experiments and a preliminary afterheat evaluation experiment were completed. Analysis of the thermodynamic test data indicates the net power plant output was about 30 kw less than during earlier runs.

Tests to evaluate the stability and lifetime of CSN-1A type bearings were initiated. The final design of the improved ML-1 precooler was approved and fabrication of this unit was initiated in the vendor's shop. The fabrication of the 600 kw alternator was initiated and the gear set for use with the alternator was delivered.

Metallurgical evaluation of the IB-17R-2 test element was completed. The in-pile test of a prototype of the ML-1-II fuel element (IB-17R-3) was terminated after 9501 hr of operation; fission product activity in the test loop indicated the possibility of cladding failure. The initiation of fabrication of the ML-1-II core loading was delayed pending investigation of the IB-17R-3 failure.

Under the ML-1 Technology program, the evaluation of Hastelloy X cladding continued and a program to define the limits of the existing pressure vessel technology was initiated. A shielding optimization study to supplement the ML-1A Preliminary Design was begun. Modification of the GCRE facility continued; completion of this work is scheduled for March 1965.

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

QUARTERLY PROGRESS REPORT

1 OCTOBER THROUGH 31 DECEMBER 1964

CONTENTS

	<u>Page</u>
<u>ABSTRACT</u>	v
I. <u>PROGRESS TO 30 SEPTEMBER 1964 - SUMMARY</u>	1
II. <u>ML-1 PROJECT</u>	5
1.0 ML-1 TEST OPERATIONS	5
2.0 ML-1 OPERATIONS ENGINEERING SUPPORT	10
3.0 ML-1 DEVELOPMENT AND IMPROVEMENTS	39
III. <u>ML-1 TECHNOLOGY</u>	55
4.0 FUEL ELEMENT TECHNOLOGY	55
5.0 ADVANCED PRESSURE VESSEL TECHNOLOGY	64
IV. <u>ML-1A PROGRAM</u>	65
6.0 ML-1A PRELIMINARY DESIGN	65
V. <u>GCRE FACILITY</u>	67
7.0 GCRE FACILITY MODIFICATION	67
<u>REFERENCES</u>	69
<u>APPENDIX A - AGCRSP BACKGROUND INFORMATION</u>	A-1
<u>APPENDIX B - ML-1 PLANT CHARACTERISTICS</u>	B-1

TABLES

<u>Table Number</u>	<u>Title</u>	
1-1	Summary of ML-1 Power Plant Shutdowns During ANSOP 16650	6
2-1	Calculated Equilibrium Temperatures of Reactor Core Components with 0.5 lb/sec Gas Cooling Flow	14

(Continued)

CONTENTS - Continued

<u>Table Number</u>	<u>Title</u>	<u>Page</u>
2-2	Estimated Rate of Temperature Rise in Uncooled Reactor Components	15
2-3	Nominal Temperature of Reactor Components	15
2-4	Typical DRML-1 Printout	24
2-5	DRML-1 Printout of Special Parameters	25
3-1	Summary of ML-1 Improved Precooler Burst Test Program	42
3-2	Mechanical Properties of Irradiated Hastelloy X Tubing From the IB-17R-2 Experiment	53
4-1	Composition of Hastelloy X Specimens for High Temperature Corrosion Tests	55
4-2	Oxidation of Hastelloy X Exposed in Air at 1900°F for 100 Hours	56
4-3	Comparison of Weight Gained at 1750°F and 1850°F	56

FIGURES

<u>Figure Number</u>	<u>Title</u>	
1-1	ML-1 Test Operations	7 & 8
2-1	Effect of Shield Solution Height on ML-1 Radiation Dose Rate During Operation	12
2-2	Predicted ML-1A Dose Rates	13
2-3	ML-1 Reactor Operating Envelope - 1750°F Hot Spot	18
2-4	ML-1 Reactor Operating Envelope - 1800°F Hot Spot	19
2-5	Hot Spot Temperatures of Inner Ring Elements During Control Rod Switch Incident	21
2-6	Hot Spot Temperatures of Non-Inner Ring Elements During Control Rod Switch Incident	22
2-7	Parametric Performance Diagram - CSN-1A T-C Set	26
2-8	CSN-1 Turbine Performance	27
2-9	CSN-1 Turbine Work and Efficiency	28
2-10	CSN-1 Compressor Map	30
2-11	CSN-1A Compressor Work and Efficiency	31
2-12	ML-1 Operational Gamma Dose Rates	32
2-13	ML-1 Operational Fast Neutron Dose Rates	33
2-14	ML-1 Shutdown Neutron Dose Rates	34

(Continued)

CONTENTS - Continued

<u>Figure Number</u>	<u>Title</u>	<u>Page</u>
2-15	ML-1 Shutdown Gamma Dose Rates	35
2-16	Temperature Distribution - Recuperator Tube Bundle Assembly	37
2-17	CSN-1A Overspeed Performance	38
3-1	ML-1 Gas Consumption	40
3-2	Burst Section, ML-1 Improved Precooler Core	43
3-3	Braze Joint, ML-1 Improved Precooler Core	43
3-4	ML-1-II Instrumented Fuel Element Pattern	46
3-5	Hastelloy X Structure, IB-17R-2 Fuel Pin 1 at X/L = 1.0	48
3-6	Inside Surface of Pin 3, IB-17R-2 Element at Selected Locations	49
3-7	Unusual Structure of UO <sub>2</sub> Particle Boundaries, IB-17R-2, Pins 1 and 3	50
3-8	Irregular UO <sub>2</sub> Particle Surface Present in IB-17R-2 Fuel	51
3-9	IB-17R-2 Fuel Pin 2 Showing Attack by Nitric Acid	52
4-1	Microstructure at Outer Surface of Hastelloy X Tubing Exposed 100 Hours in 1900°F Air	57
4-2	Microstructure in Interior of Hastelloy X Tubing Exposed 100 Hours in 1900°F Air	58
4-3	Weight Gain vs Time of Hastelloy X at 1750°F and 1800°F in Air	60
4-4	Centrifugal Creep Testing Machine	61
4-5	Head of Centrifugal Creep Testing Machine with Speci- mens Attached	62
4-6	Centrifugal Creep Test Specimens	63



ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

QUARTERLY PROGRESS REPORT\*

1 October Through 31 December 1964

I. PROGRESS TO 30 SEPTEMBER 1964 - SUMMARY

The Army Gas-Cooled Reactor Systems Program evolved from studies conducted at ORSORT in 1954 and by Sanderson-Porter Company in 1955 to evaluate the feasibility of the development of a mobile, nuclear power plant for military use. These studies indicated the feasibility of such a concept and established the basic objective of the Program. This objective was the development of specifications for a mobile, low-power, nuclear power plant capable of extended operation under military field conditions, based on the design and performance of a demonstration plant. The programs to develop the reactor and power conversion equipment for the plant began in late 1956 and Aerojet was selected as the systems contractor to integrate all Program activity in 1959. (A bibliography of major reports under the Program is given in Appendix A.) The following major projects have been undertaken:

1) The design of a reactor test facility (GCRE) was performed by Aerojet under contract with the USAEC. The construction of the test facility at the NRTS was supervised by the USAEC-ID. The design work began in mid-1957 and construction was completed in late 1959.

2) A turbine-compressor (t-c) set test facility (GTTF) at Fort Belvoir, Virginia, was completed by Aerojet in 1959 by installing equipment in accordance with the design provided by, and under contract with, the Department of the Army, APCDB.

3) The design, fabrication and test operation of a test gas-cooled reactor (GCRE-I) was performed by Aerojet under contract with the USAEC. This test reactor was provided in the program to investigate the operational and control characteristics of the reactor concept chosen for the power plant, to provide information on system transients for use in designing the plant, and to permit developmental and life-time testing of fuel elements. The heterogeneous, water-moderated,

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nitrogen-cooled reactor operated at nominal thermal power of 2.2 Mw. The reactor first achieved criticality in February 1960 with plate-type fuel elements and operated with these and with a replacement core of pin-type (prototype for the power plant) elements until April 1961 when the experimental program was interrupted for investigation of a failure in the calandria. This investigation continued through the remainder of 1961. The decision was made to deactivate the GCRE-I in early 1962 and the GCRE facility was placed in standby condition (Section 7.0 summarizes the GCRE modification and reactivation program).

4) The design and fabrication of a developmental t-c set (TCS-560) was accomplished by the Stratos Division of Fairchild Engine and Aircraft Co. under contract with the Department of the Army, APCDB. This unit was delivered in late 1959 and evaluation testing began in early 1960 in the GTTF. This testing was performed by Aerojet under contract with the Department of the Army, APCDB until May 1963 when Aerojet was relieved of cognizance for the developmental program.

5) The design and construction of a test facility for the ML-1 power plant at the NRTS, Idaho, supervised by the USAEC-ID, was completed in late 1960. Modifications to the test facility to increase the working space in the auxiliary control building, to provide improved ventilation for the power plant, and to provide improved facilities in the test building for handling shield solution were completed in 1963 and 1964. The modifications were based on designs prepared by Aerojet; the construction was supervised by the USAEC-ID.

6) The design, development and fabrication of two core loadings for the GCRE-I were completed by Aerojet under contract with the USAEC. The first core loading consisted of 75 plate-type fuel elements and was completed in the fall of 1959. The second core loading, consisting of 75 pin-type elements, was completed in mid-1960. Neither of these core loadings was operated long enough to demonstrate long-term reactivity or lifetime characteristics because of the failure of the GCRE-I reactor calandria.

7) The design, development and fabrication of the first core loading for the reactor of the demonstration power plant (ML-1) was completed by Aerojet under contract with the USAEC. This core loading, consisting of 61 elements (plus spares), was completed early in 1961. The core loading has been in operation in the ML-1 reactor since the spring of 1961; activities relating to this operation are discussed in Section 2.4.

8) The design and development of a second core loading for the ML-1 reactor was completed by Aerojet under contract with the USAEC in mid-1964. Aerojet was authorized to proceed with the fabrication of this core loading in July 1964; activities under this task are discussed in Section 3.4.

9) The design and fabrication of two t-c sets for the demonstration power plant was completed by subcontractors under a contract between Aerojet and the Department of the Army, APCDB. The activities relating to the t-c sets are summarized below:

a) TCS-670 - This set was designed and fabricated by the Stratos Division of Fairchild Engine and Aircraft Co. to specifications prepared by Aerojet. The unit was delivered early in 1961 and, after preliminary testing revealed that the machine did not satisfy the design specifications, modification was undertaken. During open-cycle tests of the modified unit in 1963, the set failed because of insufficient internal clearance to accommodate thermal expansion. Additional modifications were made and, although the set operated satisfactorily in the open-cycle configuration, seizure occurred during subsequent closed-cycle tests. Evaluation of this failure resulted in the decision in 1963 to defer further modification or testing of the TCS-670. Cognizance for completion of the development of this unit was assigned to APCDB in mid-1964.

b) CSN-1 - This t-c set was designed and fabricated by Clark Bros. Co. to specifications developed by Aerojet. The unit was delivered in March 1961 and, after preliminary testing, installed on the power conversion skid which was delivered to the NRTS for testing with the demonstration power plant in June 1962. The set performed acceptably during ML-1 testing in September 1962 and February 1963 although the power output was less than the design value. Inspection following the February operation revealed abnormal bearing wear and some cracking in the turbine blades. A new bearing design was developed jointly by Clark Bros. and Aerojet, and the turbine blade design was modified to improve the blade strength. The t-c set with new bearings and turbine blades was returned to the NRTS and operated satisfactorily during April and May 1964. Inspection of the machine following this operation revealed damage to the first stage turbine blades; the necessary repairs were completed in mid-1964 and mechanical operation of the unit has been satisfactory since that date. The status of activity relating to the CSN-1 t-c set is discussed in Section 3.2a.

10) The design, fabrication and test operation of a demonstration power plant (ML-1) were performed by Aerojet under contract with the USAEC. The design and fabrication of the ML-1 control cab and reactor skid were completed early in 1961 and these components were delivered to the NRTS. The reactor achieved initial criticality on 30 March 1961. Operational tests to verify predictions of control rod worth, reactivity, temperature coefficients and shielding effectiveness, and to develop general core physics data were conducted from April 1961 to June 1962. After delivery of the power conversion skid to the NRTS in June 1962, final power plant checkouts were completed and initial operations were conducted in September 1962. Test operation was resumed in January 1963 following a shutdown for modification and maintenance. During these tests, the ability of the reactor to operate at full design power (3.3 Mw) was demonstrated and 247 kw of shaft output power was measured. At the conclusion of this test run, evidence of a leak in the reactor pressure vessel was observed. After confirmation of the leak, the reactor skid was partially disassembled, the leak repaired, and the skid reassembled. The CSN-1 t-c set was modified during this period (see

Item 9-b above). Test operation of the power plant was resumed in mid-April 1964. The plant operated for more than 660 hr during a limited endurance test and was shut down at the end of May. Inspection at that time revealed the damage to the CSN-1 turbine which was repaired. Test operations were resumed in September 1964. (Test operation of the power plant in the current quarter is discussed in Sections 1.0 and 2.0 of this report.) The ML-1 Plant Characteristics are presented in Appendix B.

11) The development of performance specifications for a field-operable, gas-cooled, nuclear power plant (ML-1A) based on the ML-1 design was completed by Aerojet under contract with the USAEC on 30 June 1963.

12) A design study and the development of conceptual designs for a second generation gas-cooled, mobile, nuclear power plant (ML-2) were performed by Aerojet under contract with the USAEC. Preliminary feasibility studies of advanced concepts were completed in early 1962, at which time a more detailed evaluation was initiated. The goal of this evaluation was to define a 500 kw(e) power plant with minimum weight, maximum reliability and maintainability, minimum logistic requirements, and minimum startup and relocation times. The final report of the study was published in October 1962. At the direction of the USAEC, a limited evaluation of a reactor concept not fully considered in the basic study was performed in May and June 1963.

13) The preparation of the preliminary design of a field-operable, gas-cooled, nuclear power plant (ML-1A), based on the ML-1 design and the ML-1A performance specification (Item 12 above), was completed by Aerojet under contract with the USAEC. This work was initiated in mid-1963 and the preliminary design report was published in June 1964. The limited follow-on activity in this area is discussed in Section 6.0.

14) The design of modifications to the GCRE facility (see Item 1 above) to permit testing of the ML-1 reactor skid in that facility was performed by Aerojet under contract with the USAEC. The design work was initiated in the fall of 1963 and completed early in 1964. Construction to implement the design was begun in April 1964 under the supervision of USAEC-ID and was in progress on 31 December 1964 (Section 7.0 discusses the status of this activity).

This report is organized under five major headings: Summary of Progress to 30 September 1964, ML-1 Project, ML-1 Technology Program, ML-1A Program and GCRE Facility. Significant areas of activity are identified by numbers 1.0 through 7.0 (second order identification) and details are presented as decimals of the appropriate second order identification. Figures and tables are identified with the second order identification and are included in the text close to the point of reference. Two types of references are cited, numerical designations refer to in-contract reports; alphabetical designations refer to reports which have received general distribution.

## II. ML-1 PROJECT

### 1.0 ML-1 TEST OPERATIONS

The "ML-1 Thermodynamic Performance Test" (ANSOP 16650), in progress at the beginning of October, was completed on 30 November 1964 after 1774.7 hr of operation at rated speed (see Figure 1-1 for a graphic summary of test operations from 1 October through 31 December 1964). Two major shield evaluation experiments were conducted at the same time. One experiment provided data for the operational and shutdown radiation dose rates with a 2 wt% boric acid shield solution and the other determined the radiation dose levels during operation as a function of the amount of the shield solution above the reactor core.

The power plant was shut down for one week for maintenance and installation of special test equipment following the completion of ANSOP 16650. Operation was resumed on 6 December 1964 to perform the "ML-1 Power Plant Endurance Test" (ANSOP 16635). The first period of operation in the endurance test ended on 17 December 1964 when, without a power plant shutdown, performance of the "ML-1 Preliminary Afterheat Evaluation" (ANSOP 16656) was initiated. The plant operated for 93.2 hr at 3.0 Mw(t), following which the power level was reduced to 2.2 Mw on 16 December for the 24-hr period immediately preceding the afterheat evaluation. The reactor was manually scrammed and the t-c set allowed to coast to a stop to develop data concerning the afterheat temperature transients in the reactor. The start motor was energized shortly after the shutdown was initiated; this emergency action resulted from misinterpretation of the fuel element temperature data which indicated that these temperatures were approaching the predetermined limit. The start motor was de-energized after approximately 160 seconds of operation; the indicated maximum fuel element temperature was approximately 1225°F (the limit established for this test was 1600°F).

The ML-1 power plant operated at rated speed for a total of 1509.53 hr during the quarter; the overall on-stream factor was 68.36%. Operation was interrupted several times for minor maintenance or by reactor scrams; these interruptions are summarized in Table 1-1 on the following page.

TABLE 1-1 - SUMMARY OF ML-1 POWER PLANT SHUTDOWNS  
DURING ANSOP 16650

<u>Date</u>	<u>Shutdown Type</u>	<u>Reason</u>
1 Oct	Normal	To repair defective coolant gas supply valve
3 Oct	Automatic	Operator inadvertently bumped Channel 4 nuclear drawer
8 Oct	Automatic	Operator inadvertently bumped Channel 4 nuclear drawer
12 Oct	Automatic	Operator error during load transfer from dynamometer to start motor
20 Oct	Normal	Installation of shield tank extension for shielding evaluation
24 Oct	Automatic	Failure of NRTS power
28 Oct	Normal	To install a remotely operated shield tank drain valve
28 Oct	Manual	Failure of control rod switch
9 Nov	Normal	To refill shield tank
13 Nov	Normal	Failure of dynamometer bearing
25 Nov	Normal	To refill shield tank
25 Nov	Automatic	Moderator flow transient resulting from demineralizer valving operation
30 Nov	Normal	Test terminated
<u>ANSOP 16635</u>		
11 Dec	Normal	To install special test instrumentation
14 Dec	Automatic	Moderator water sensing line ( $\Delta P$ gauge) froze and ruptured
17 Dec	Normal	Failure of emergency generator
17 Dec	Normal	Apparent loss of shield water
22 Dec	Normal	Test terminated

The only significant incident during the quarter occurred during a power plant startup on 28 October 1964. The t-c set had been accelerated to half speed by the start motor in accordance with normal startup procedures and reactor power was being increased on a 40 to 60 second period. When the console operator attempted to insert the Shim 1 control rod to terminate the power increase at approximately 2.0 Mw, the rod would not drive in. Repeated attempts to insert all three shim rods resulted in incremental control rod movements sufficient to limit the power to 2.5 Mw and, shortly thereafter, to stabilize reactor power at 2.2 Mw. However, since both the reactor outlet temperature high annunciator and the fuel element temperature high annunciator had sounded and since the control rod system was not functioning normally, the console operator initiated a manual scram.

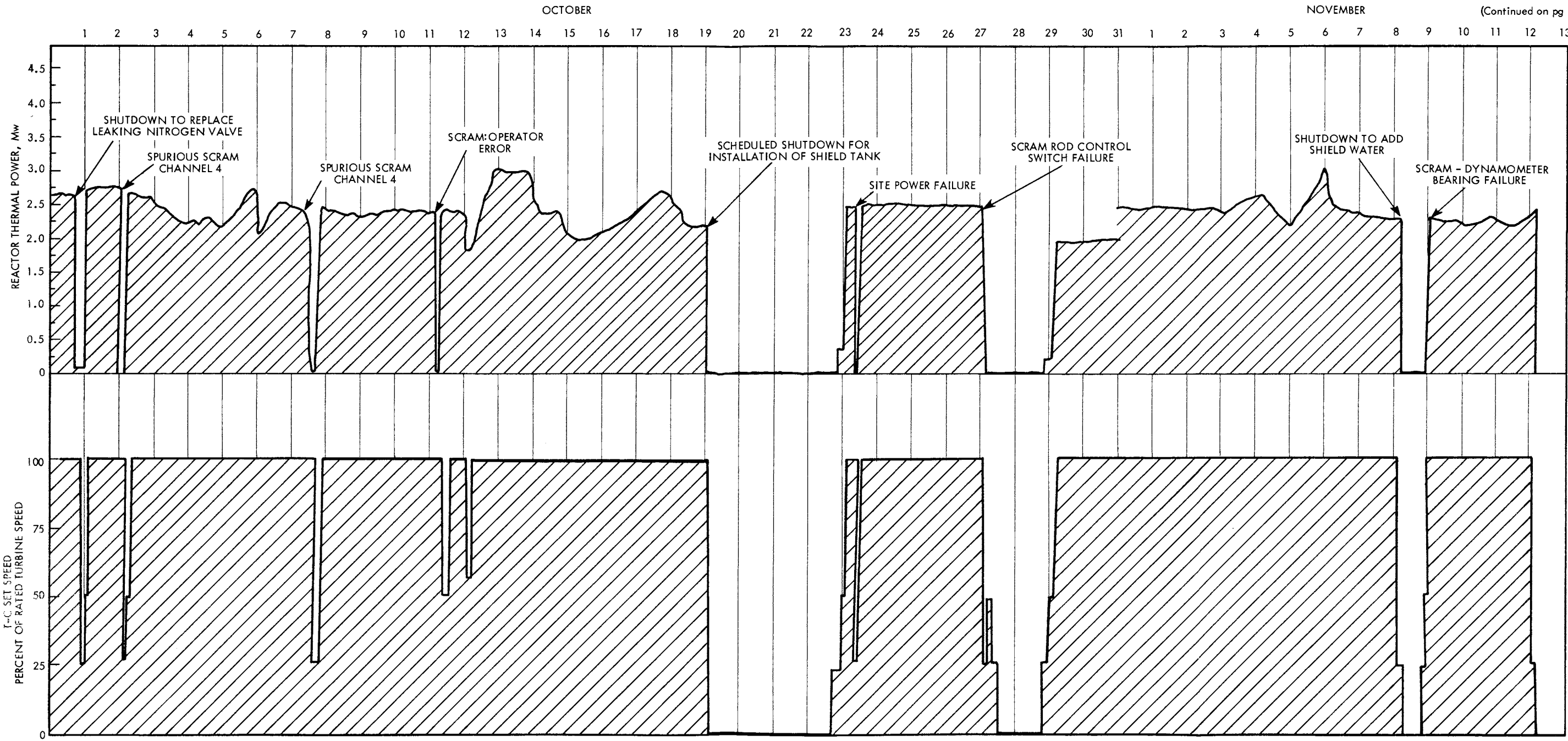


FIGURE 1-1. ML-1 TEST OPERATIONS

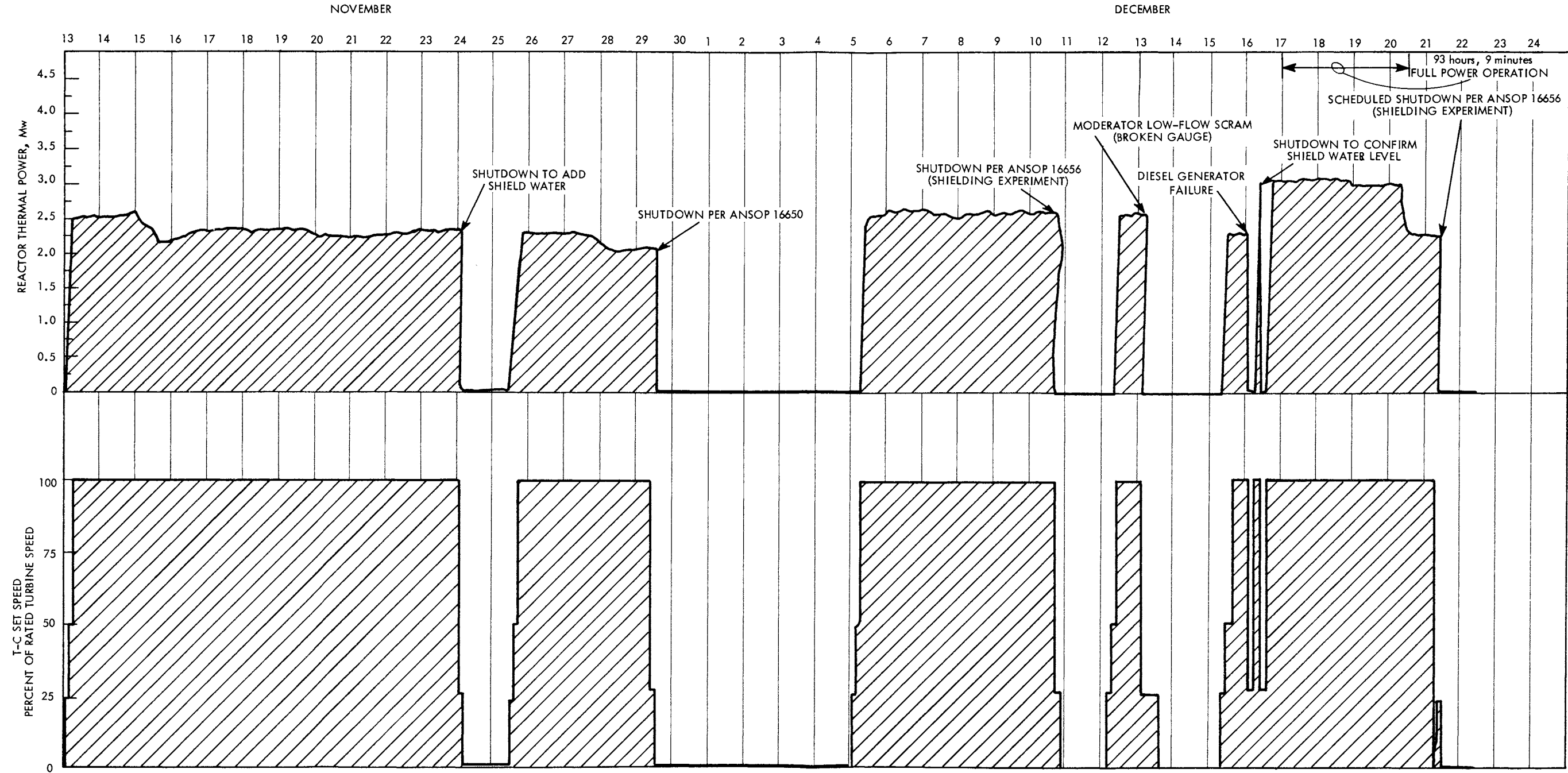


FIGURE 1-1. ML-1 TEST OPERATIONS

An analysis of the effects of this incident (Ref. 1)\*indicated that the temperatures of the fuel pin cladding in the central ring of six fuel elements approached 1990°F (for less than one minute) and the hot spot cladding temperature of the remaining 55 fuel elements was approximately 1775°F. It was concluded that these temperatures did not result in damage to the fuel elements. An examination of the control rod switch revealed that wear in the mechanical portion of the switch had permitted overtravel and the consequent interruption of certain of the electrical circuits. A replacement switch was installed and a program to develop and test an improved switch was initiated (see Section 2.0).

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\*References are listed at the end of the main body of text, pp 69 and 70. Numerical designations apply to reports given only in-contract distribution; alphabetical designations refer to reports that were given general, TID-4500 distribution.

## 2.0 ML-1 OPERATIONS ENGINEERING SUPPORT

### 2.1 Reactor and Auxiliaries

a. Corrosion of Aluminum: A series of laboratory tests was initiated to determine the corrosion rate of Type 3004 aluminum alloy when exposed to de-mineralized water and aqueous solutions of boric acid as an extension of the earlier aluminum corrosion work (Ref. a) and to confirm the ML-1A Preliminary Design. The tests being performed are similar to those conducted earlier (Ref. 2): the aluminum specimens are exposed to test solutions in glass beakers and removed after specified time intervals for evaluation. By the end of December, test specimens exposed for 150, 450, and 1350 hr were being evaluated; other specimens were being exposed.

b. Control Rod Actuators: A report describing the design and testing of modifications to the control rod actuator clutches was published (Ref. 3). The modified clutches operated in the ML-1 throughout the quarter without evidence of slippage or other malfunction.

c. Reactor Shielding: A series of experiments was conducted at the ML-1 during the quarter to develop data for verification of the ML-1A shielding analysis. These experiments provided data for the following evaluations:

- Comparison of radiation dose rates during reactor operation with both 10 and 2 wt% boric acid shield solutions.
- Comparison of radiation dose rates during operation of the reactor with and without a wood expedient shield over the reactor and with 2 wt% boric acid shield solution.
- Radiation dose rates during reactor operation as a function of the amount (depth) of 2 wt% boric acid shield solution over the top of the reactor.
- Comparison of the radiation dose rates during reactor operation with and without a radial polyethylene expedient shield on the power conversion skid side of the reactor.

The results of the experiment to determine the dependence of the radiation dose rate on the shield solution level are shown in Figure 2-1. The estimated dose levels for the ML-1A Preliminary Design (Ref. b), based on these experimental data, are shown in Figure 2-2. (See Section 2.5 for further discussion of this experimental program.)

d. Engineering Support: Engineering support for the operation of the reactor and auxiliaries at the NRTS during the quarter included an improvement in the response time of the moderator level indicator achieved by relocation of the sensing head, completion of the design for ML-1 fuel element storage canisters, the temporary repair of a cracked weld in the upper moderator surge drum, and the continued revision of the ML-1 as-built drawings to document modifications.

e. ML-1 Reactor Skid Relocation: The procedure proposed for use during relocation of the ML-1 reactor skid was analyzed to evaluate the safety aspects of the proposed move. (The ML-1 reactor skid will be moved to the GCRE facility to permit continued operation while the components of the power conversion skid are being developed.)

Results of the analysis indicated no problem would be experienced and that no credible combination of circumstances would produce a nuclear incident during the movement, since the ML-1 was designed specifically for this kind of handling. However, the analysis revealed that a nuclear incident could result from a malfunction of the handling equipment during the transfer of the reactor skid from the floor of the GCRE facility to the reactor pit. The hypothesized situation and the calculated results are as follows.

If the reactor skid were dropped 30 feet onto the floor of the reactor pit, the skid structure would be extensively damaged; however, such damage would not result in, or contribute in any significant way to, a nuclear incident. The fuel elements would attain a speed of 44 ft/sec at the time of impact. Assuming that the entire deceleration would be absorbed by a 0.55 in. deformation of the spiders (conservatively ignoring deceleration which would occur during crushing of the bottom components of the reactor skid), the average deceleration load on the fuel pins would be 655 g and the peak deceleration load would be about 2000 g. The peak load would result in a compressive stress in the fuel pin cladding of approximately 35,400 psi. The calculated allowable stress for bending in unirradiated Hastelloy X tubing of the configuration used in the fuel pins is 64,400 psi but the mechanical properties of the tubing have been degraded by irradiation to the point where bending under the conditions described above is probable. It was concluded that a major dislocation of the fuel in the reactor would not result from the hypothesized 30 ft drop of the reactor skid, but that the pins probably would bend and that the damage to the fuel elements would probably prevent further use.

The possible extent of mechanical damage to the control blades as a result of the hypothesized incident was determined. In this analysis, the heaviest blade (the upper shim-scam blade weighing 6.04 lb) was considered and it was assumed that the three "Spirol" pins would deform in perfect double shear at the manufacturer's rated loading. It was determined that the hypothesized 30 ft drop would result in failure of the control blade

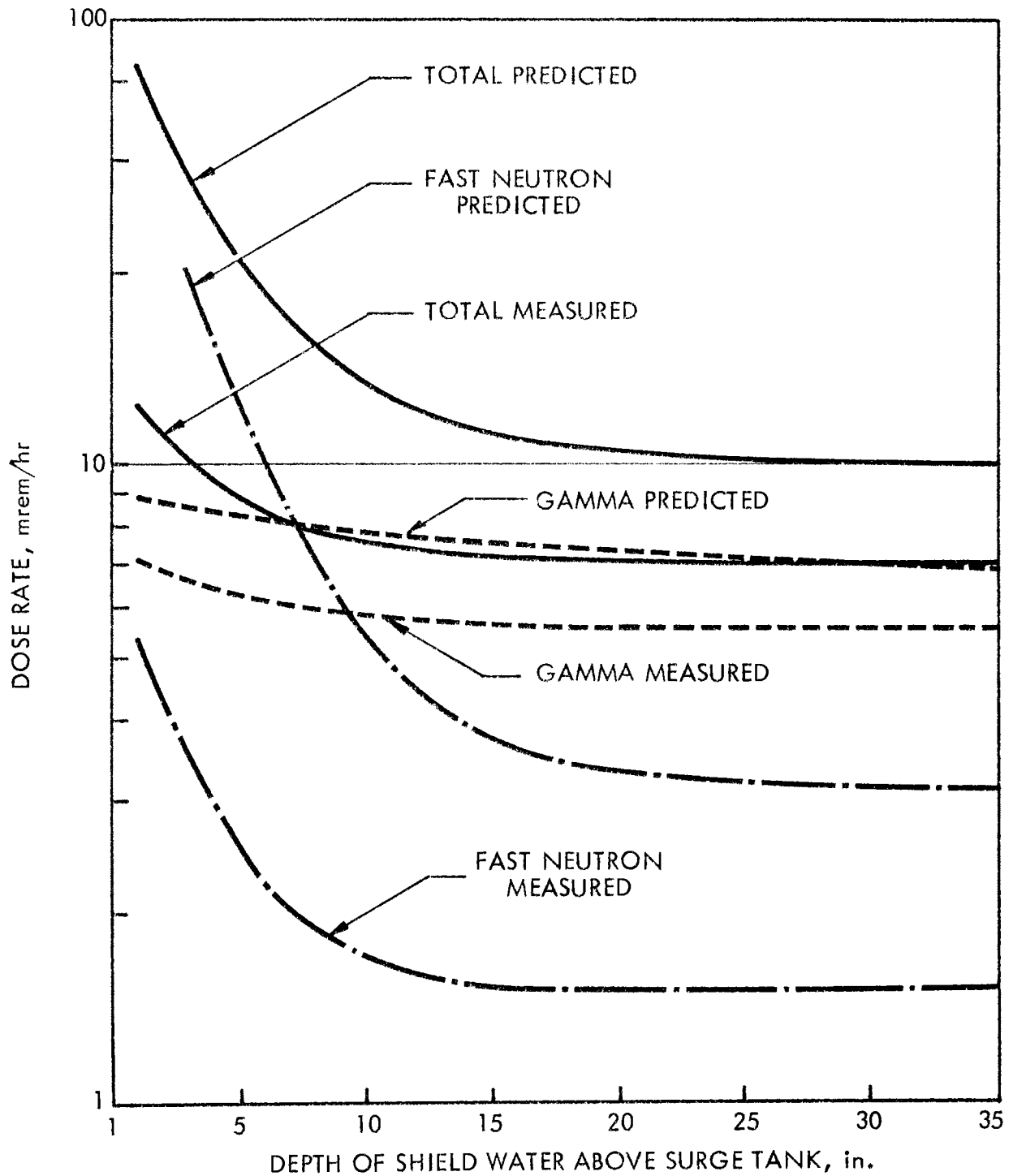


FIGURE 2-1. EFFECT OF SHIELD SOLUTION HEIGHT ON ML-1 RADIATION DOSE RATE DURING OPERATION

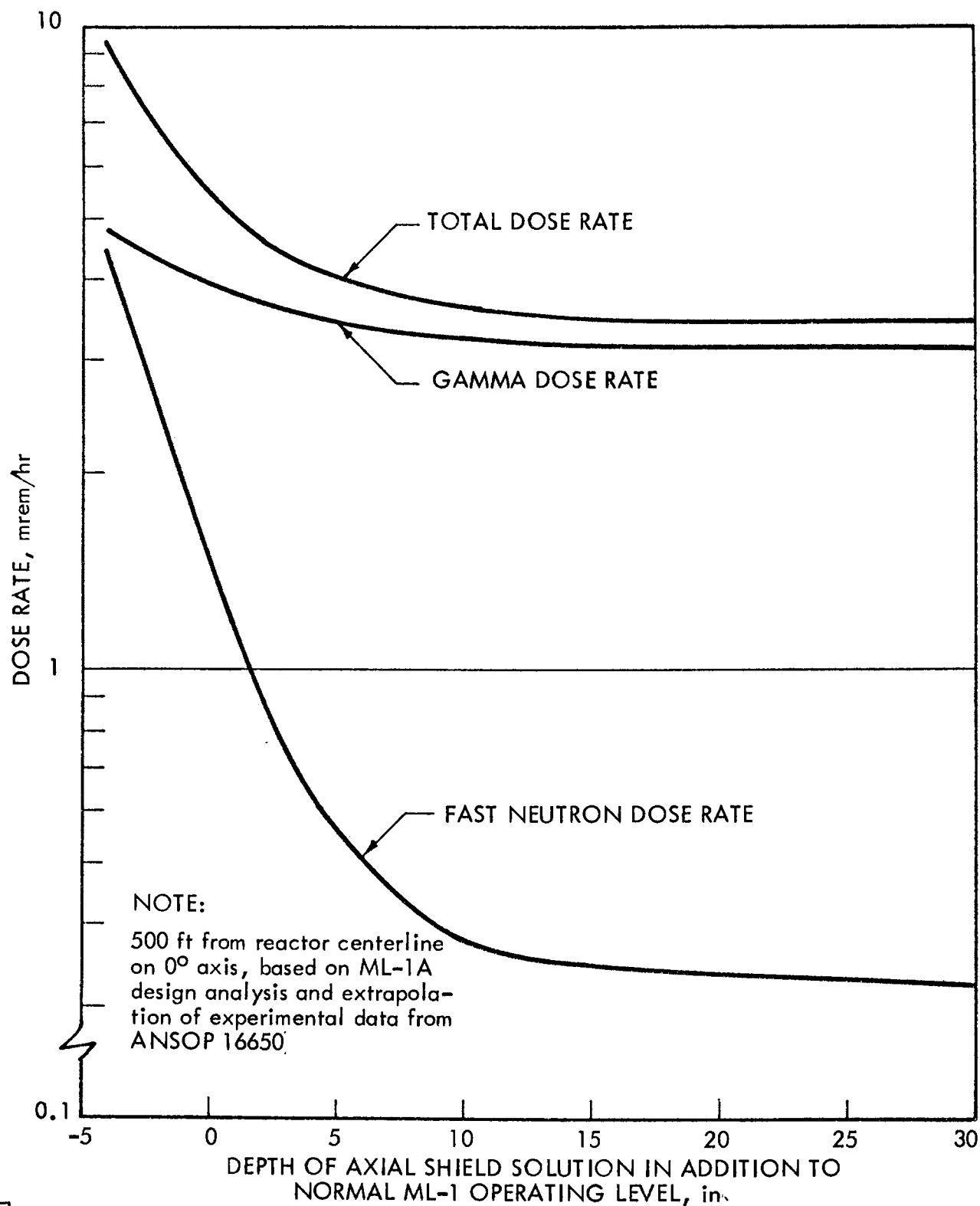


FIGURE 2-2. PREDICTED ML-1A DOSE RATES

support pins and that the blades would come to rest on the upper surface of the lower tube sheet in a position where the neutron absorption would be effectively the same as if the blades had been removed from the reactor core.

The nuclear characteristics of the ML-1 core following the hypothesized 30 ft drop were determined, considering that the fuel pins had bent and were displaced downward and that the control blades were "removed" from the core structure. This analysis revealed that criticality was possible in the normally moderated core configuration, but that criticality could not be attained if the moderator water were removed (metal must be removed from the fuel elements and the fuel pellets compacted to attain criticality with the 49 kg load in the reactor core in the absence of moderation).

As a result of this analysis, it was decided to remove the moderator water from the ML-1 reactor during the movement of the reactor skid from the floor of the GCRE facility to the bottom of the reactor pit to protect against the consequences of an unlikely failure of the crane. Inasmuch as the moderator water absorbs afterheat during periods of reactor shutdown, this decision required the provision of an alternate means of heat absorption during the transfer operation. The afterheat generation rate was calculated for several probable situations, and the rate of heat deposition in the various reactor components with the moderator drained was calculated for decay periods up to 45 days to define the requirements of the special cooling system.

The concept selected provides a blower to circulate air through the reactor core. A flow rate of 0.5 lb/sec was calculated to produce acceptable temperatures in the critical components of the reactor core. The calculated equilibrium temperatures for typical decay times are shown in Table 2-1 and the estimated rate of temperature rise of these same components with neither cooling flow nor moderator are presented in Table 2-2. By way of comparison, the nominal temperatures experienced by these components during power plant operation at 3 Mw is shown in Table 2-3.

TABLE 2-1 - CALCULATED EQUILIBRIUM TEMPERATURES  
OF REACTOR CORE COMPONENTS WITH 0.5 lb/sec GAS COOLING FLOW

<u>Reactor Component</u>	<u>Temperature, °F</u>		
	<u>Decay Time, Days</u>		
	<u>7</u>	<u>14</u>	<u>21</u>
Fuel elements	<500	<500	<500
Tube sheets	<150	<150	<150
Reflectors	<160	<160	<160
Control blades	550	440	360

TABLE 2-2 - ESTIMATED RATE OF TEMPERATURE RISE  
IN UNCOOLED REACTOR COMPONENTS

<u>Reactor Component</u>	<u>Rate of Temperature Increase, °F/hr</u>		
	<u>Decay Time, Days</u>		
	<u>7</u>	<u>14</u>	<u>21</u>
Fuel elements	1020	710	570
Tube sheets	16	10	7
Reflectors	58	38	26
Control blades	710	496	345

TABLE 2-3 - NOMINAL TEMPERATURE OF REACTOR COMPONENTS

<u>Reactor Component</u>	<u>Normal Operation at 3 Mw</u>	<u>24 hr After Shutdown*</u>
Fuel pins	1750	780
Fuel element outer liner	500	<500
Tube sheets	500	335
Control blades and mounts	200	200
Reflectors	250	250
Moderator seals	<200	<200

\*With moderator water circulated by the standby pump

Examination of these data revealed that the control blades are the only components that will exceed operating temperatures during the transfer of the skid to the reactor pit. At the end of the quarter, detailed analyses were in progress to define the effect of this temperature on the control blades and the control blade mounts. The investigation includes evaluation of the different rates of thermal expansion of the silver alloy blades and the 17-4 PH gears, and the load imposed on the bearings by this differential expansion, and the possible problems associated with the blade-to-pressure tube clearances.

## 2.2 Power Conversion

Power conversion engineering support during the quarter was limited to liaison between engineering at San Ramon and operations at the NRTS, primarily because of the high on-stream factor and the absence of any major problems. Assistance was also provided for the preliminary evaluation data on the performance of the power conversion equipment.

### 2.3 Instruments and Controls

a. ML-1 Dynamometer: At the time of the failure of the bearings in the ML-1 dynamometer (13 November 1964), there was an apparent malfunction of the dynamometer automatic speed control system. After the bearings had been replaced, the electrical and speed control systems of the dynamometer were thoroughly checked; no evidence of malfunction was observed. It was concluded that the apparent loss of speed control was probably caused either by the momentary opening of the flow switch in the cooling water system or by inadvertent actuation of the load dump circuitry. Either of these conditions could have resulted from the shock and vibration in the system at the time of bearing failure. No evidence of any irregularity in the dynamometer speed control system was observed during the subsequent operations.

b. Replacement Electrical Power and Control Cables: The temporary repair of the electrical power and control cable (W-1304) completed during the prior quarter (Ref. a) resulted in satisfactory performance during this quarter. Four separate smaller replacement cables were ordered; two of these assemblies had been received by the end of December and were being subjected to quality control examination, and the other two assemblies are scheduled for delivery in January.

The replacement cables incorporate the concept specified in the ML-1A Preliminary Design (Ref. b) to improve the moisture resistance of the connectors. The connector backshells are completely filled with epoxy compound and the junction of the backshell with the neoprene sheath is totally enclosed in polyurethane. The replacement cables thus will not only improve the reliability of the ML-1 but will demonstrate the ML-1A design concept.

c. ML-1 Analysis Instrumentation: A six-point chromel-alumel thermocouple rake, fabricated from thermocouple extension wire and connected to existing reference junctions and readout equipment, was installed in the recuperator low pressure outlet duct to evaluate the temperature variations there. A high speed indicating and recording system, consisting of a multichannel oscillograph and four solid state driver amplifiers, was provided for use during the afterheat experiment. This system provides continuous indication and recording of selected fuel element temperatures, the reactor inlet and outlet coolant temperatures, the coolant pressure, the t-c speed and outputs generated by nuclear instrumentation channels 3 and 6.

d. ML-1 Turbine Speed Calibration Oscillator: The fabrication of the turbine speed calibration oscillator was completed. Testing indicated that the clips used to mount the sealed nickel-cadmium batteries needed minor redesign; this work was in progress at the end of December. This oscillator provides a portable source of 25, 50, 100, 110 and 125% turbine speed calibration signals that are derived from a tuning fork-regulated, solid state oscillator circuit.

e. Rod Control Switch: The redesign and testing of the control rod actuation switch was undertaken as a result of the malfunction which occurred on 28 October (see Section 1.0). The mechanical stop assembly in the switch was replaced with one of improved design, and improved shaft bearings were also installed (Ref. 1). Mechanical cycling tests of the modified switch

were performed and examination after 30,000 cycles indicated excessive wear in the Teflon shaft bearings. Replacement bearings were fabricated from Nylatron (MoS<sub>2</sub> impregnated nylon) and testing was resumed. Examination after 300,000 cycles was in progress at the end of December. One of the two different models of Blue Line rotary switches, procured for evaluation, was being tested at the end of December.

f. ML-1/GCRE Operations Support: All modifications to the ML-1 dry critical box were completed and checked out to assure the compatibility of this component with ML-1/GCRE operation. A study to define the requirements for transport of the ML-1 control cab to the GCRE facility was initiated.

g. As-Built Drawings: The revision of approximately 100 engineering drawings was completed in the continuing program to develop current instrumentation and control drawings for the ML-1 power plant.

## 2.4 Fuel Elements

a. ML-1-I Thermal and Neutronic Analysis: Analyses of data generated during ANSOP 16625A and ANSOP 16650 indicated that the nominal maximum fuel element hot spot temperature was 1745°F. This temperature occurred in the cladding in the inner ring of six fuel elements; the computed nominal maximum hot spot temperature for the remaining 55 elements was 1668°F. Although the data are badly scattered, an apparent minor decrease in the core pressure drop occurred between ANSOP 16625A and ANSOP 16650; the best estimate of the decrease is approximately 2%.

Operating envelopes for the existing (non-optimum orificing) configuration of the ML-1-I core loading were developed. Since it is doubtful that the fuel element thermocouples will remain operable throughout the 10,000-hr lifetime of the core, the operating envelope was established in terms of parameters other than the fuel element temperature (reactor coolant gas inlet and outlet temperatures and coolant gas flow). Two operating envelopes were developed which limit the cladding hot spot temperature at the hottest location in the core (the six elements in the inner ring) to 1750°F and 1800°F (see Figures 2-3 and 2-4); a summary of the analysis was published (Ref. 4).

Work was completed on a method to determine the annular neutron flux within the ML-1 fuel cell (Ref. 5). It was found that acceptable results could be obtained from the solution of the integral equation relating the annular flux to the scalar flux. A machine code, FLU, was written to perform this calculation, and specific cases for typical ML-1 fuel cells were solved with the code. These solutions indicate that the annular flux has only a minor dependence on the value of the outer radius of the cell. Having developed an acceptable technique for the estimation of the annular flux entering the pin bundle, the intra-element flux and power distributions were recalculated by the method of successive generations. The computer code prepared to perform these calculations (BOUNCE) was reviewed and extensive modifications were made to reduce the time needed to run individual problems.

The reduction and evaluation of reactivity data generated during ANSOP 16650 was completed. This results indicate that, after 5800 Mwh of operation,

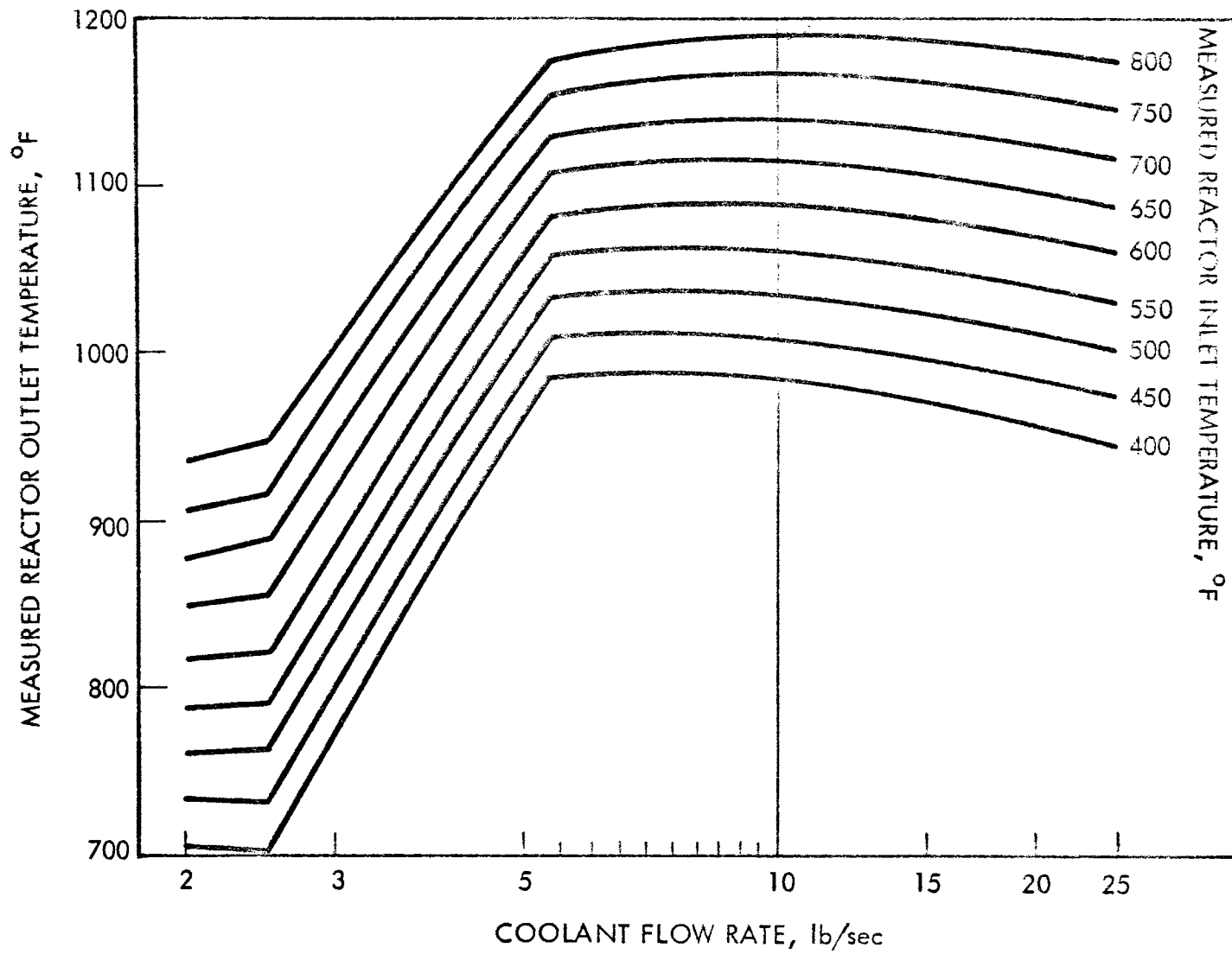


FIGURE 2-3. ML-1 REACTOR OPERATING ENVELOPE - 1750°F HOT SPOT

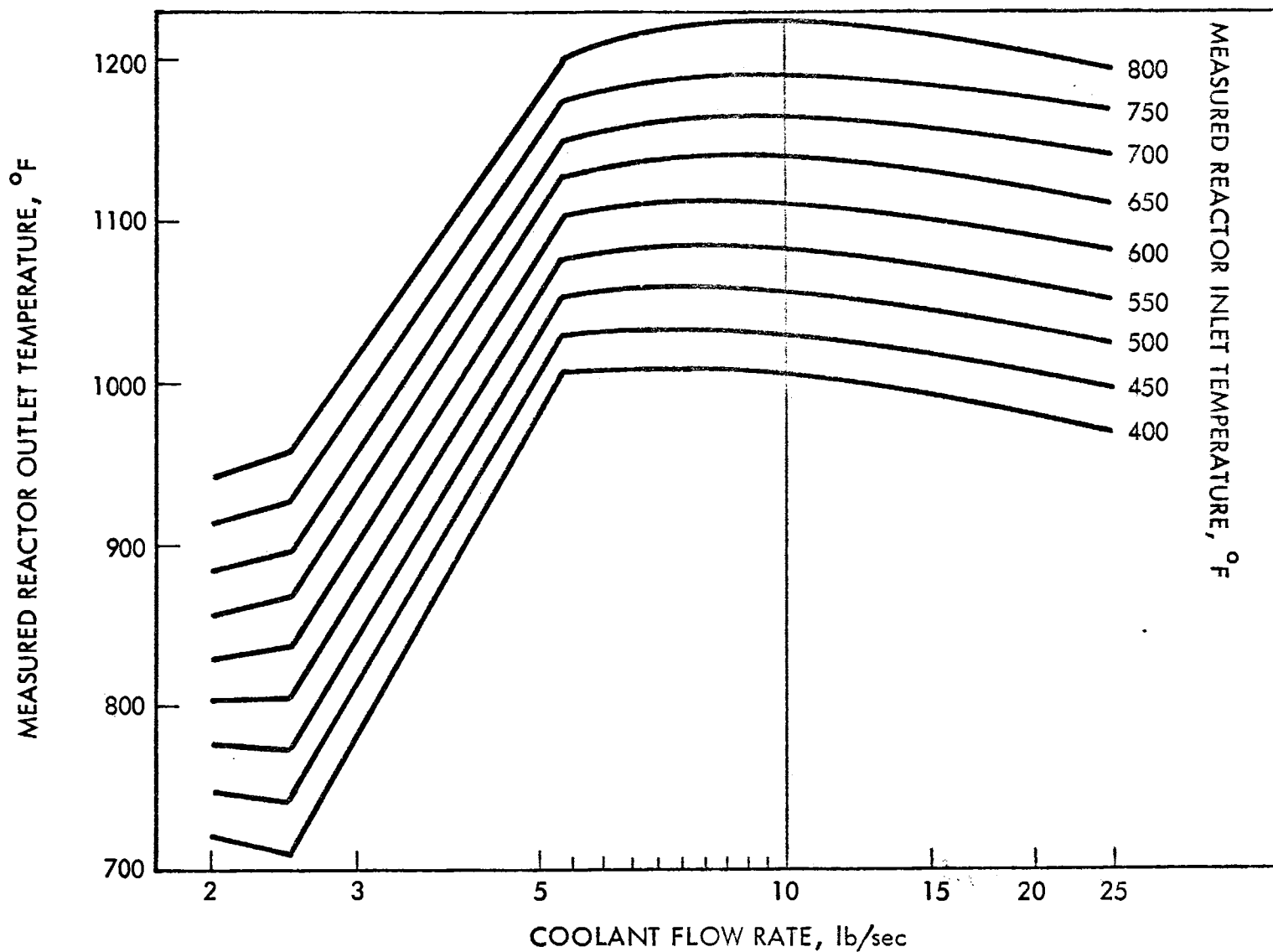


FIGURE 2-4. ML-1 REACTOR OPERATING ENVELOPE - 1800°F HOT SPOT

the reactivity change of the core is less than the experimental error (approximately  $\pm 0.1\% \Delta K/k$ ) and less than the theoretically predicted change ( $+ 0.15\% \Delta K/k$ ).

b. ML-1-I Engineering Support: The out-of-pile capsule experiments to investigate the mechanism of formation of  $CO_2$  in the fuel pin and the reaction rates of this gas with the fuel and cladding (Ref. a) were completed. The capsules, containing ML-1-I type fuel and various mixtures of gases, were exposed for varying periods over a wide range of temperatures. The capsules were leak checked after the exposure (no leaks were found) and shipped to the NRTS hot cells for extraction of and analysis of the internal gases. The procedure for performance of the work at NRTS was published (Ref. 6).

The analysis of the operating data generated during the control rod switch incident on 28 October 1964 (see Section 1.0) was completed. The calculated peak hot spot temperature of the inner ring elements was between 1900 and 1990°F 45 seconds before the reactor was scrammed. At the time of the reactor scram, the maximum calculated hot spot temperatures were 1933°F for the inner ring elements and 1733°F for other elements in the core. Figures 2-5 and 2-6 show plots of the calculated hot spot temperature transients during the incident. Examination of these figures indicates that the total time in excess of 1750°F was approximately 90 seconds for the inner ring elements and 35 seconds for the other elements in the core. Details of this analysis were published (Ref. 1). Based upon the analyses, it was conservatively predicted that no damage occurred to the fuel elements during the incident.

c. IB-8T In-Pile Test: Irradiated cladding from the IB-8T-2 test element was analysed at BMI to determine if the "sticking" of the fuel to the cladding was the result of uranium migration. No uranium was detected chemically in the cladding. This activity concluded the evaluation of the IB-8T-2 in-pile test.

d. Metallurgical Support of ML-1-I Air-Cycle Operation: The exposure of low cobalt (0.07 wt%) Hastelloy X tubing, identical to that used in the ML-1-I core, at high temperatures (1300-1800°F) continued under the ML-1 Technology Program (see Section 4.0) as a result of a decision to defer indefinitely consideration of air cycle operation of the ML-1. Data from this experiment will contribute to the basic understanding of the long term exposure characteristics of Hastelloy X.

## 2.5 System Performance Analysis

a. Data Reduction: The DRML-1 code was used to reduce about 220 data points obtained during ANSOP 16650. The code was modified for this work as follows:

- A pressure balancing routine was incorporated which statistically determines the state point pressures in the same fashion as balanced temperatures are determined. This modification eliminated a hand calculation.
- Provision was made to use temperature input generated by the rakes installed in the turbine and compressor inlet and outlet flanges.

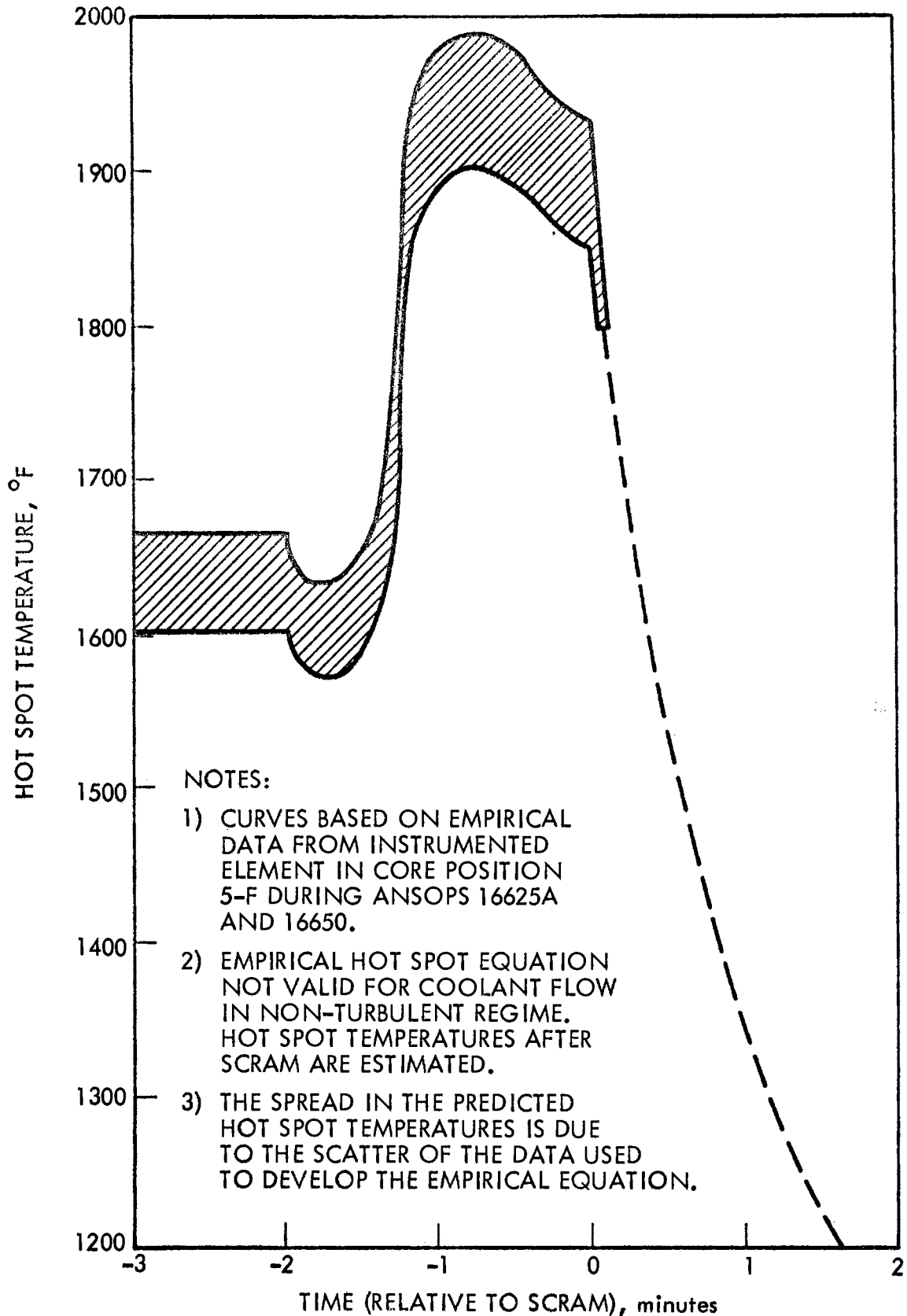


FIGURE 2-5. HOT SPOT TEMPERATURES OF INNER RING ELEMENTS DURING CONTROL ROD SWITCH INCIDENT, 28 OCT 64

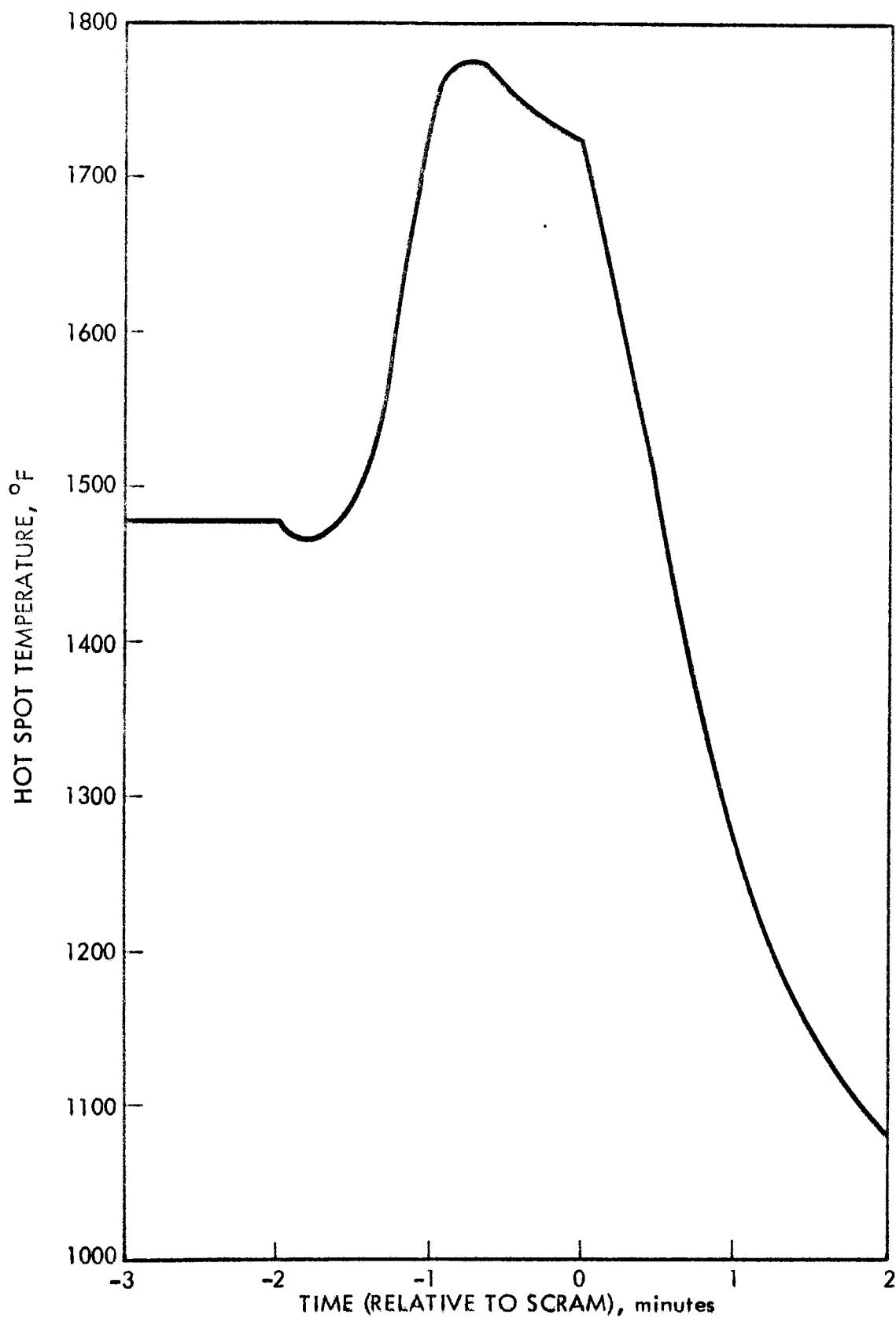


FIGURE 2-6. HOT SPOT TEMPERATURES OF NON-INNER RING ELEMENTS DURING CONTROL ROD SWITCH INCIDENT, 28 OCT 64

- Provision was made to compute parasitic power losses from lubricating oil temperatures and flow rates and to print out the results of this computation.
- Provision was made to compute and print out selected system performance parameters required for analysis and curve plotting, significantly reducing the number of hand computations normal to this work.

A typical DRML-1 output sheet is shown in Table 2-4. Table 2-5 shows a typical print out of the selected parameters required for analysis and curve plotting as developed by the improved DRML-1 routine.

b. Aerothermodynamic Performance Analysis: Analysis of ANSOP 16650 data continued throughout the quarter. The significant results of this activity are:

- Net power plant output during this experiment was approximately 30 kw less than that at comparable conditions during ANSOP 16625A.
- The effective turbine flow area during this experiment was approximately 3% larger than during ANSOP 16625A.
- The compressor efficiency was apparently about 2.5% lower than during ANSOP 16625A.
- The turbine performance appeared to be essentially identical with that observed during ANSOP 16625A.
- The performance of all plant components (except the CSN-1A t-c set) appears to be essentially identical with that observed during ANSOP 16625A.

The decrease in net plant output power is illustrated in Figure 2-7. The solid lines represent output of the system analysis computer code (CHOP) for ANSOP 16625A which were found to be in good agreement with ANSOP 16625A test results. The decrease in thermodynamic output power becomes apparent, as plotted in this figure, when the ANSOP 16650 data points are compared with the CHOP results for the same  $T_7/T_1$  temperature ratio.

Figure 2-8 shows plots of the indicated increase of effective flow area in the CSN-1A turbine. This increase may be attributable to changes in geometry resulting from replacement of the first stage rotor blades and from the elimination of the first stage nozzle inner shim; these modifications were made following the conduct of ANSOP 16625A.

The indicated increase in effective flow area does not appear to have significantly affected the CSN-1A performance. Figure 2-9 shows the plots of turbine work and efficiency developed from ANSOP 16625, 16625A and 16650 data. These curves indicate that the turbine work and efficiency was, within the limits of data scatter, essentially unchanged during the three tests. It is significant to note, however, that the extensive data developed during ANSOP 16650, which covered a large portion of the t-c set operating range, for the first time has permitted a reasonable definition of the detailed shape

TABLE 2-4 - TYPICAL DRML-1 PRINTOUT

Data Point 140 ANSOP 16650

		<u>DEG. F*</u>	<u>PSIA</u>	<u>BTU/LB*</u>
1	Compressor in	79.3 ( 78.1)	80.5	132.09 (131.79)
2	Compressor out	293.0 ( 294.7)	199.6	186.09 (186.53)
3	Recuperator (hp) in	292.9 ( 294.6)	199.3	186.06 (186.50)
4	Recuperator (hp) out	704.0 ( 707.1)	194.6	291.08 (291.90)
5	Reactor in	704.0 ( 707.1)	194.6	291.08 (291.90)
6	Reactor out	1123.5 (1119.6)	184.7	402.91 (401.85)
7	Turbine in	1116.5 (1112.6)	183.8	401.00 (399.94)
8	Turbine out	878.2 ( 887.3)	83.4	336.86 (339.27)
9	Recuperator (Lp) in	878.2 ( 887.3)	83.4	336.86 (339.27)
10	Recuperator (Lp) out	388.0 ( 479.8)	82.8	210.10 (233.40)
11	Precooler in	384.5 ( 476.3)	82.1	209.21 (232.50)
12	Precooler out	79.4 ( 78.2)	80.5	132.11 (131.81)

Thermodynamic power, kw	87.32
Reactor power to gas, kw	1935.97 (1903.36)
Recuperator shell loss, kw	376.54 ( 8.66)
Compressor flow lbs/sec	16.68
Turbine flow lbs/sec	16.41
Compressor efficiency	0.7353 (0.7240)
Turbine efficiency	0.8047 (0.7631)
Recuperator effectiveness	0.7024 (0.6961)
Precooler effectiveness	0.9134 (0.9349)
Precooler inlet air temperature	50.5
Precooler Cmin/Cmax	0.2907 (0.2910)
T-C set speed, rpm	22049.

\*Values in parentheses are the statistically probable values that satisfy recuperator and t-c set energy balances.

TABLE 2-5 - DRML-1 PRINTOUT OF SPECIAL PARAMETERS

Data Point 140 ANSOP 16650

Compressor pressure ratio	2.480
Compressor referred speed	950.497
Compressor flow function	4.806
Compressor work parameter	0.102
Compressor power coefficient	0.516
High pressure loop pressure drop product	3153.820
High pressure recuperator pressure drop product	947.100
Reactor pressure drop product	1929.917
Turbine expansion ratio	2.204
Turbine flow function	3.541
Turbine referred speed	556.009
Turbine work parameter	0.039
Turbine power coefficient	0.146
Overall temperature ratio	2.922
Compressor power	963.069 kw
Turbine power	1050.398 kw
Compressor pressure coefficient	16.378
Compressor flow coefficient	2.293
Turbine pressure coefficient	17.669
Turbine flow coefficient	2.888
Low pressure loop pressure drop product	242.208
Low pressure recuperator pressure drop product	48.442
Precooler pressure drop product	131.127
Parasitic power	54.482 kw
Precooler heat rejection	1771.575 kw



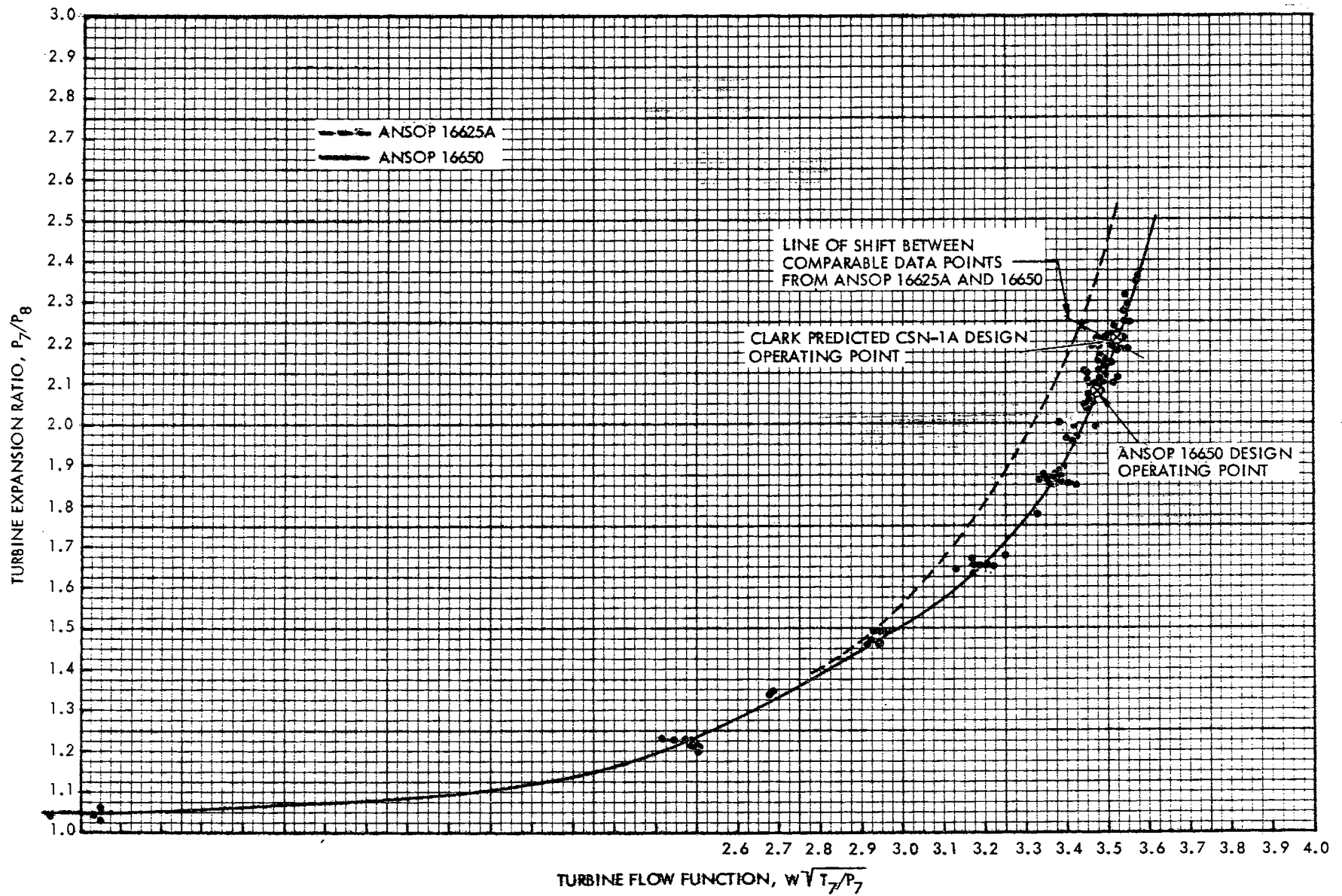


FIGURE 2-8. CSN-1A TURBINE PERFORMANCE

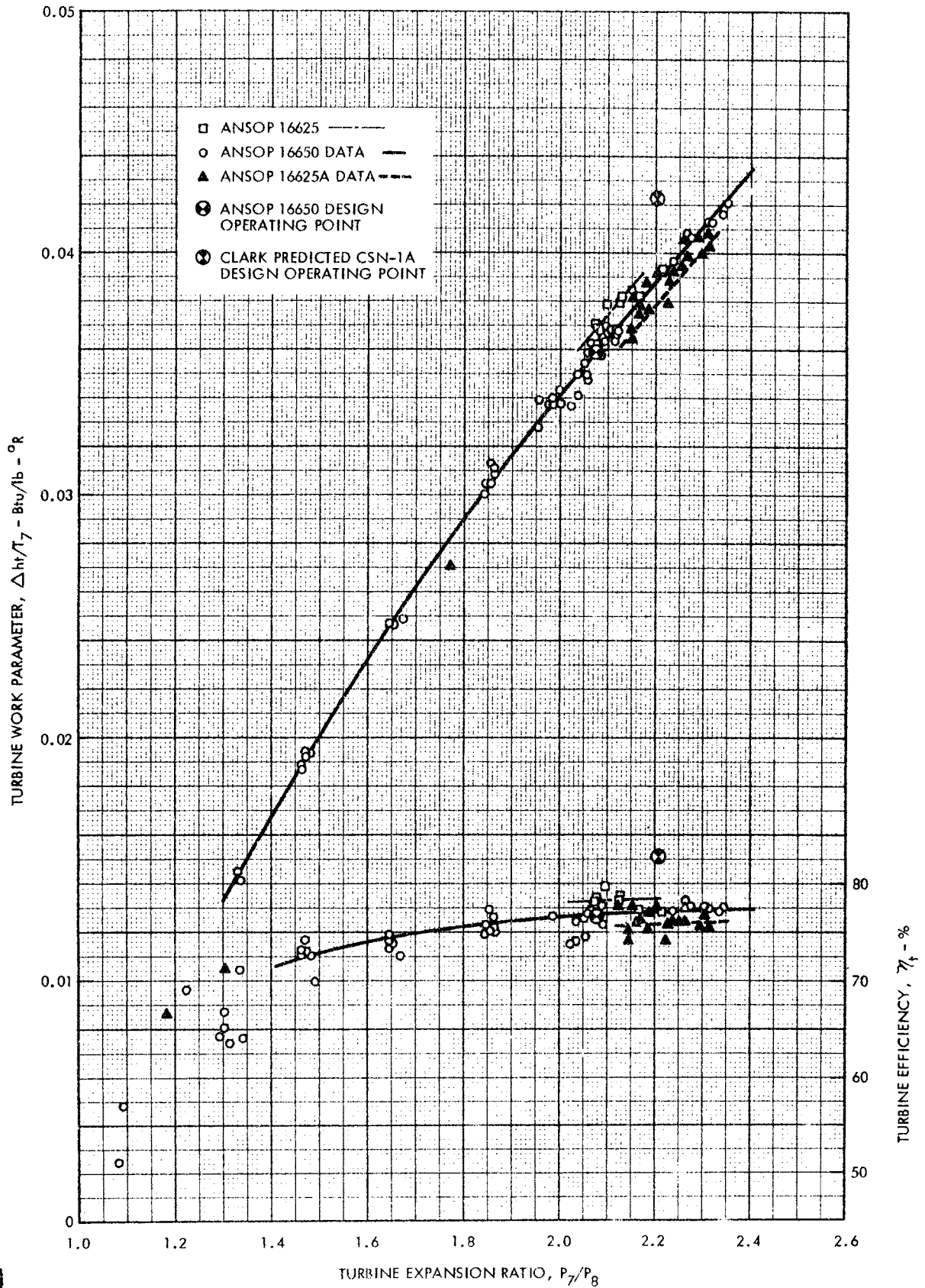


FIGURE 2-9. CSN-1A TURBINE WORK AND EFFICIENCY

of various performance parameter curves. Previous data had been obtained over a relatively limited range of turbine expansion ratios, thus reducing the confidence with which the performance curves could be interpreted.

The CSN-1A compressor performance during ANSOP 16650 is shown in Figures 2-10 and 2-11. Figure 2-10 shows the ANSOP 16650 data superimposed on the basic compressor map generated during the initial open-cycle testing at the vendor's shop and subsequent closed-cycle testing at Azusa. The overall temperature ratio ( $T_7/T_1$ ) lines shown were developed from cross plots of the ANSOP 16650 data which, for the first time, had sufficient range to permit this treatment. Comparison between the ANSOP 16625A operating line and that developed from the ANSOP 16650 data indicates that a reduction in pressure ratio of approximately 1% (at  $N\sqrt{T} = 905$  and  $T_7/T_1 = 2.85$ ) has occurred. This change is consistent with the indicated 3% increase in the effective flow area of the turbine (discussed earlier). It also appears from the compressor map that the cleaning after ANSOP 16625A may have slightly increased the compressor capacity.

Compressor work and efficiency during ANSOP 16650 are shown in Figure 2-11 with similar data from ANSOP 16625A and 16625. These plots indicate that compressor work apparently increased between ANSOP 16625A and ANSOP 16650 with a corresponding 2.5% reduction in efficiency. Although the reason for this change is not apparent from the data, the change is consistent with the observed degradation in net power output and turbine performance discussed earlier.

c. Shield Analysis: Experiments were performed concurrent with the conduct of ANSOP 16650 to permit evaluation of the performance of a 2 wt% boric acid shield solution. The results, shown on Figure 2-12, indicate that dose rates with the 2 wt% solution were greater by about 1.4 than with the 10 wt% solution; the analytical prediction was that the dose rates would be a factor of 1.6 greater with the less concentrated solution. The effect of the expedient wood shield on the fast neutron dose rate with 2 wt% shield solution is shown in Figure 2-13. Comparison of these data with those developed during the experiment to evaluate the effect on radiation dose rates of increasing the amount of shield solution over the reactor (see Section 2.1) indicates significant streaming out of the air gap between the wood shield and the top of the shield solution. As indicated in Section 2.1, these data were extrapolated to develop improved estimates of the ML-1A operational radiation dose rates; the revised estimated total ML-1A dose rate at 500 feet on the axis through the power conversion skid is 3.4 mrem/hr.

Preliminary analysis of data on the shutdown radiation dose rate at the conclusion of ANSOP 16650 (Figures 2-14 and 2-15) indicates that the effect of operation with 2 wt% boric acid solution may be less serious than anticipated; however, additional data are required to confirm this conclusion.

d. Test Planning: Test plans covering all ML-1 test operations subsequent to ANSOP 16650 were published, including the following:

- The ML-1 endurance test and manual speed control evaluation (Ref. 7)

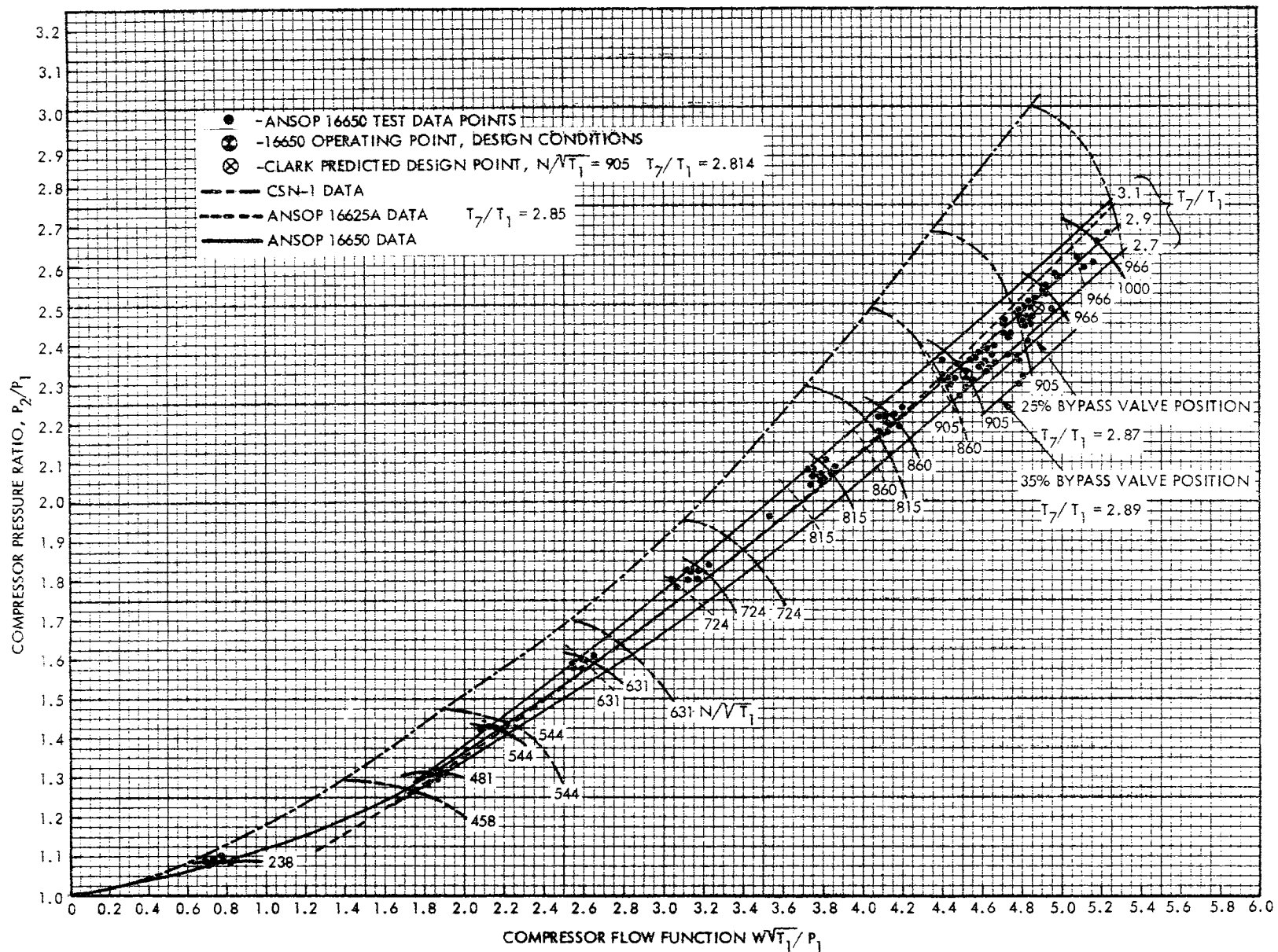


FIGURE 2-10. CSN-1 COMPRESSOR MAP

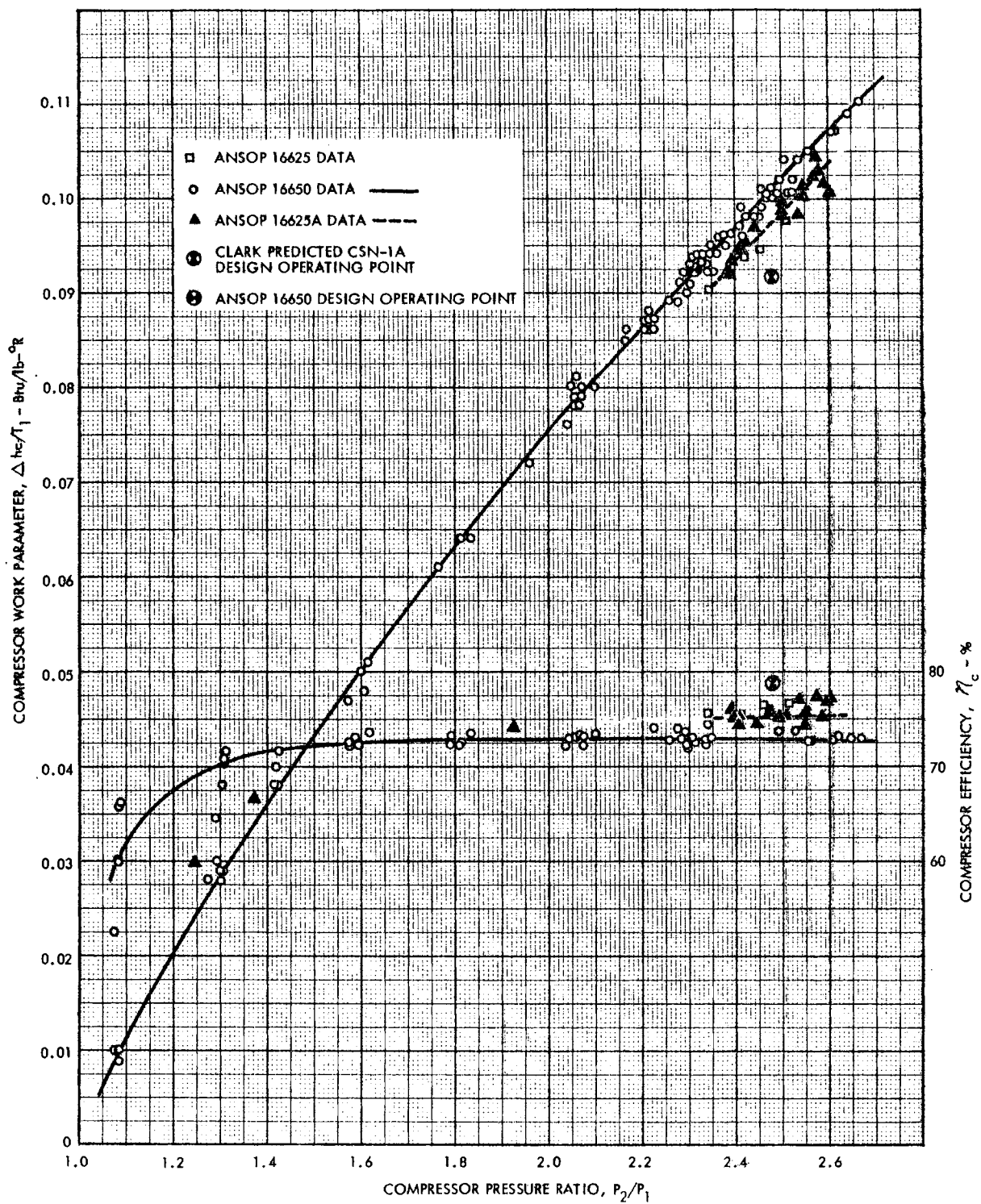


FIGURE 2-11. CSN-1A COMPRESSOR WORK AND EFFICIENCY

11.2-65-2.99

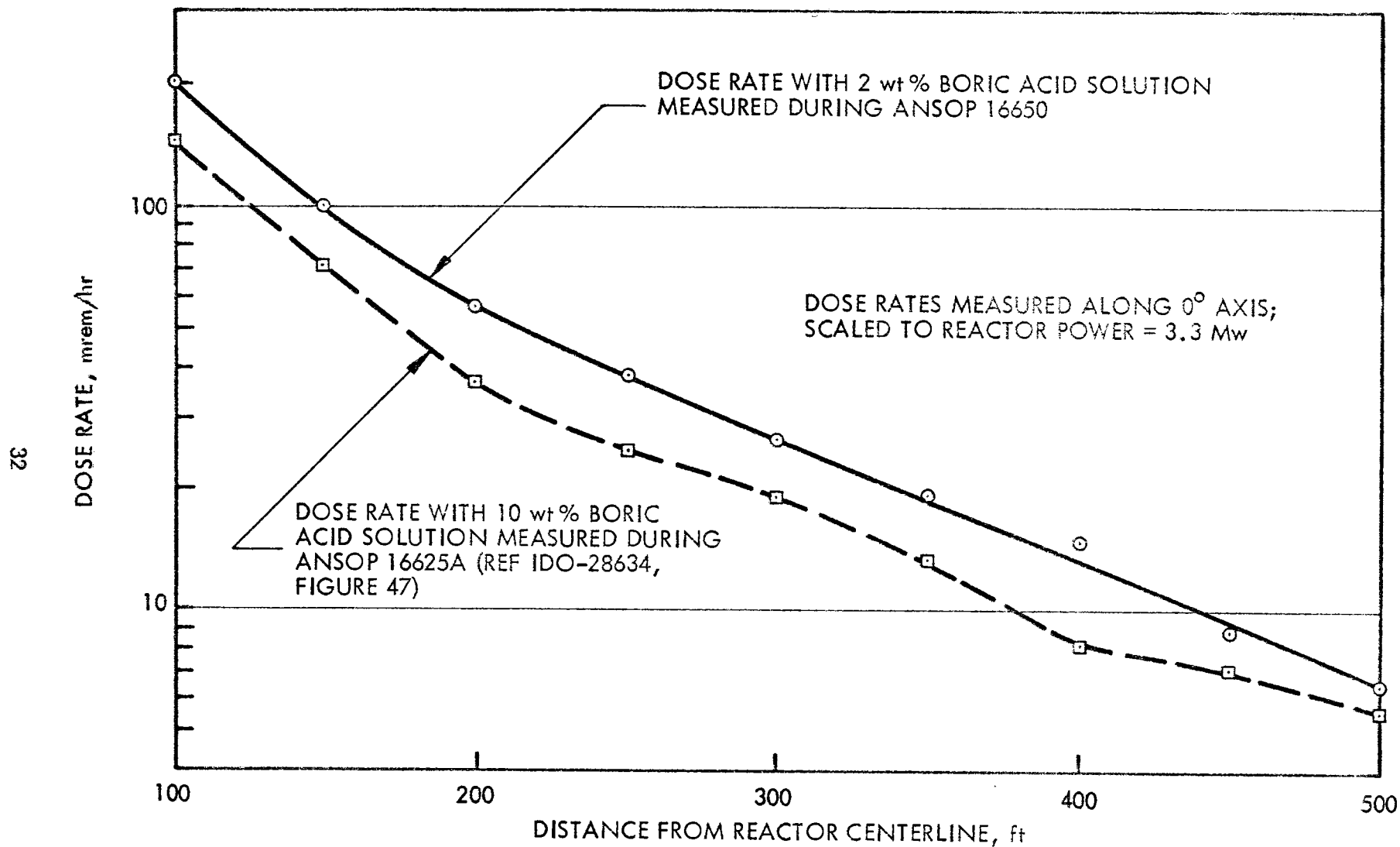
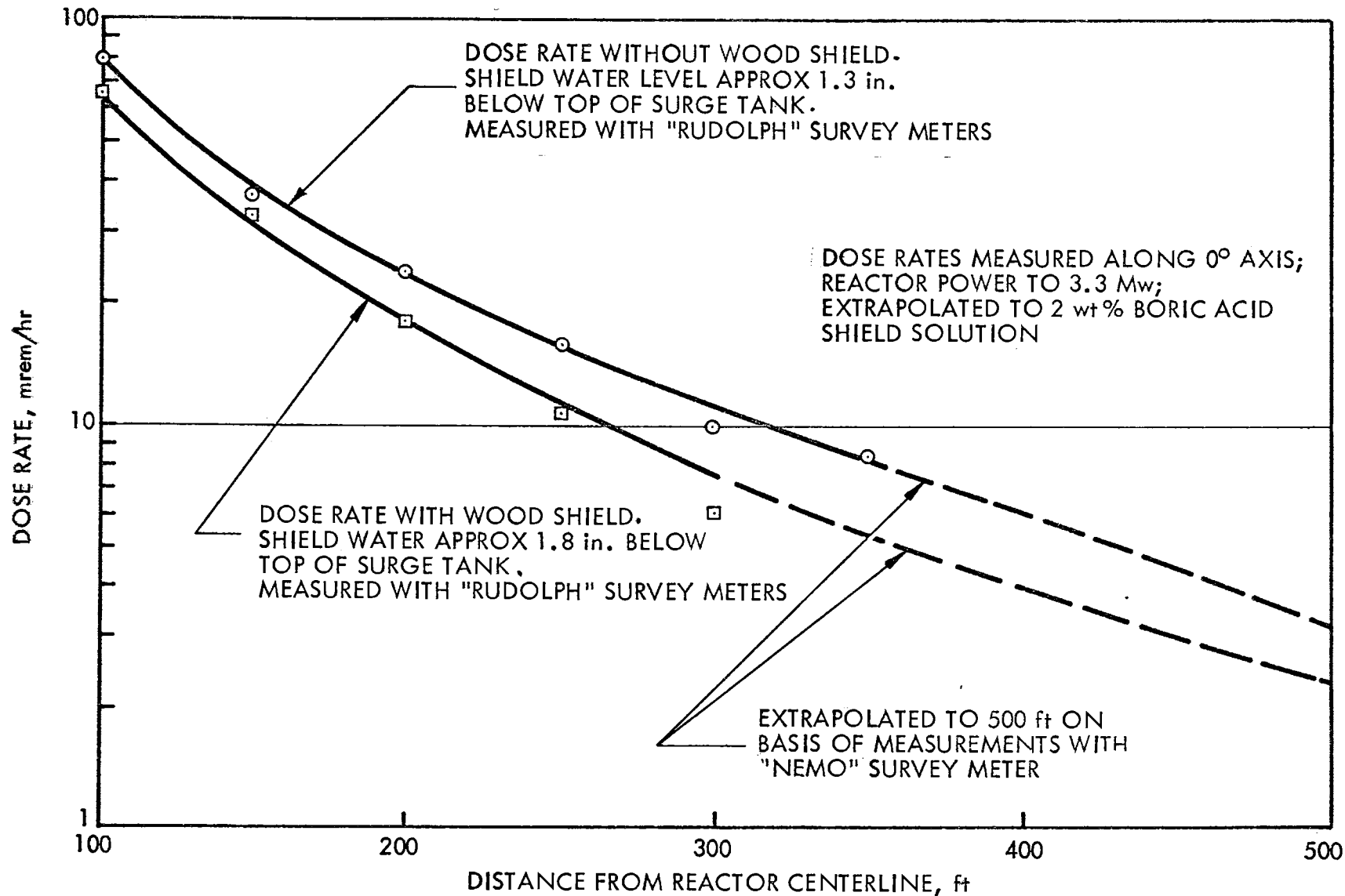


FIGURE 2-12. ML-1 OPERATIONAL GAMMA DOSE RATES

11.2-65-301

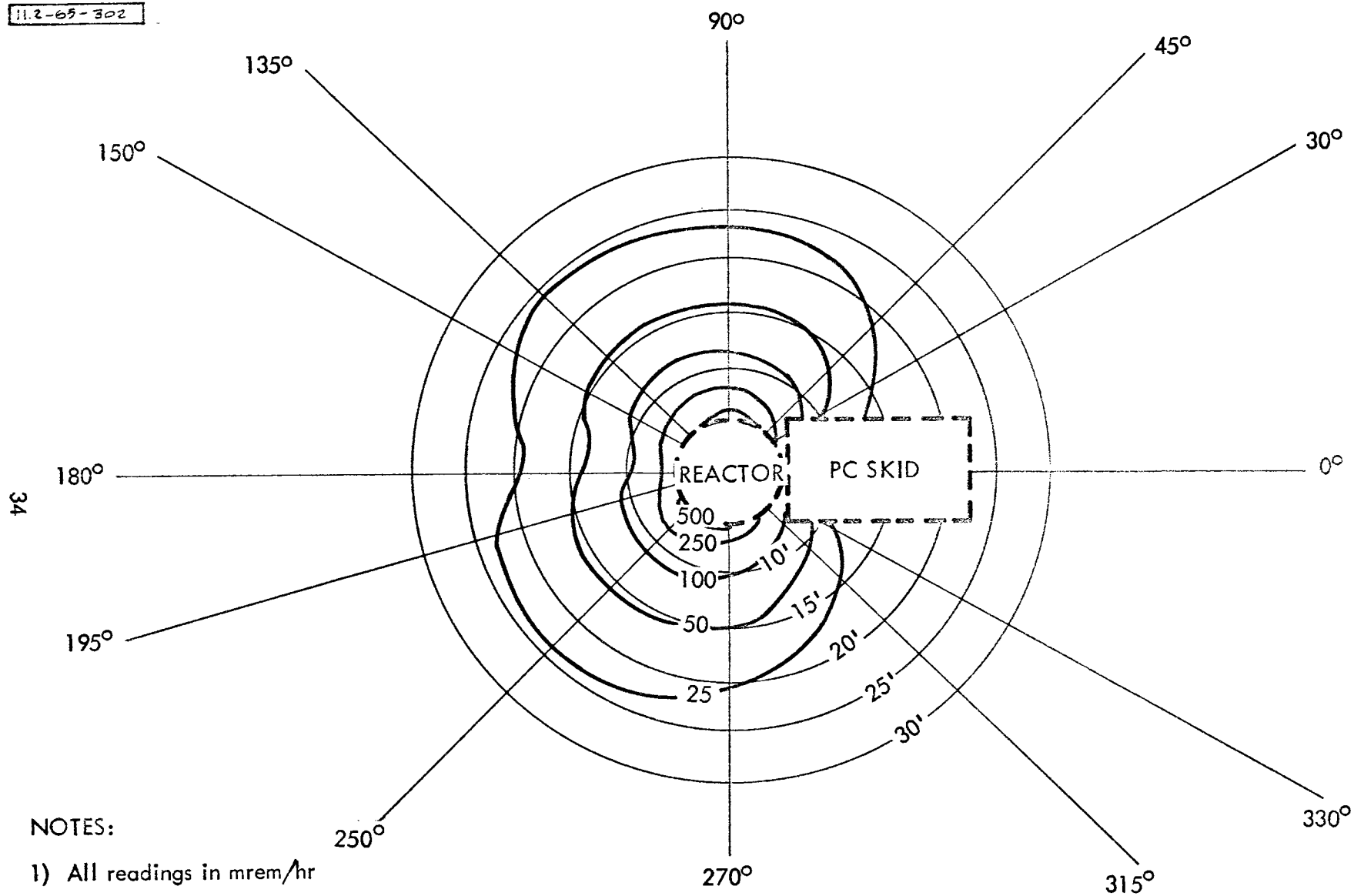
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Report No. IDO-28641

FIGURE 2-13. ML-1 OPERATIONAL FAST NEUTRON DOSE RATES

11.2-65-302



NOTES:

- 1) All readings in mrem/hr
- 2) Data taken 26 1/2 hr after shutdown from ANSOP 16650, with shield tank empty.

FIGURE 2-14. ML-1 SHUTDOWN NEUTRON DOSE RATES

11.2-65-303

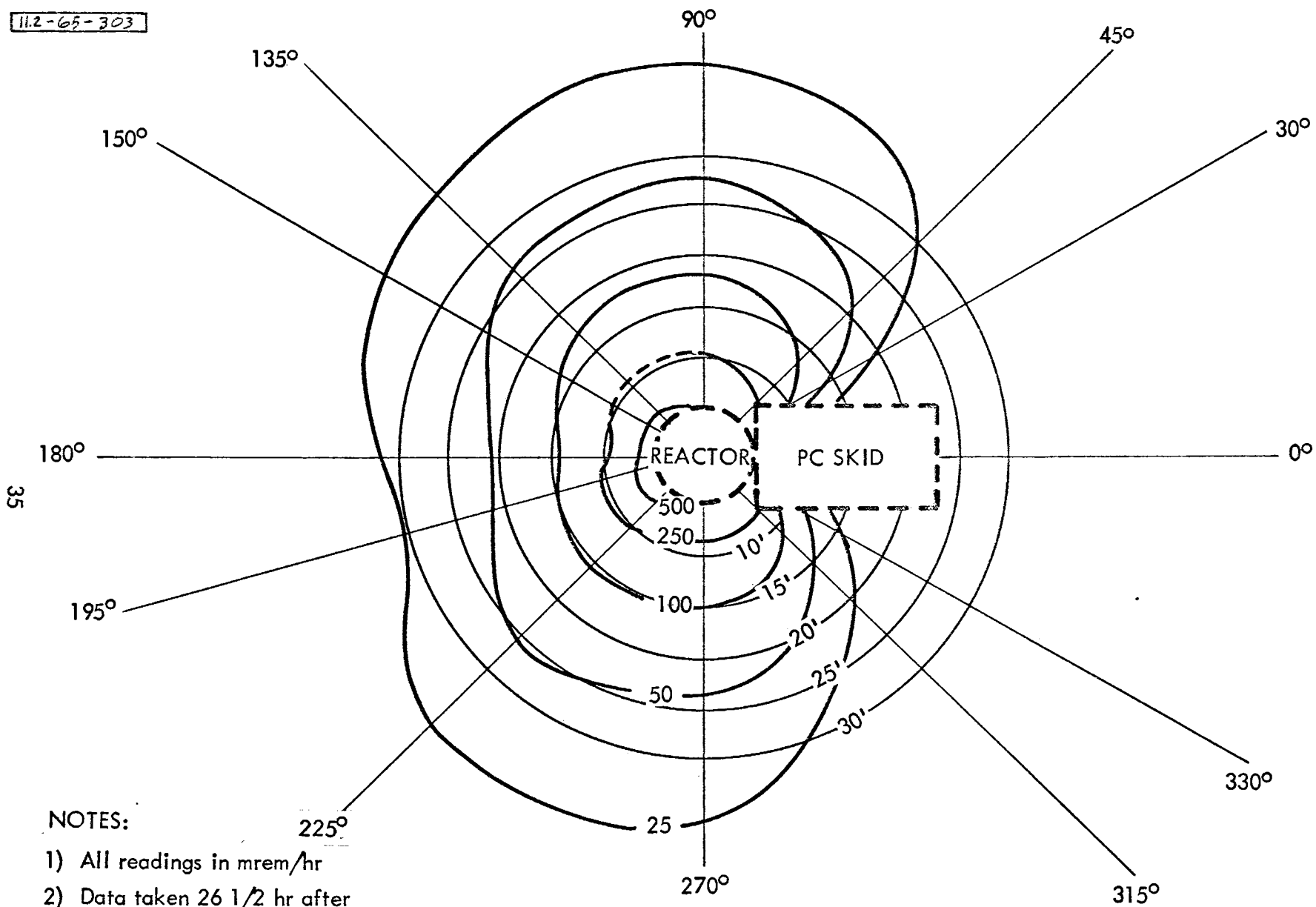


FIGURE 2-15. ML-1 SHUTDOWN GAMMA DOSE RATES

- The sustained full (reactor) power test and preliminary afterheat evaluation (Ref. 8)
- General shielding evaluations (Ref. 9 and 10)
- A special shield experiment to determine the effect of increased depth of shield solution over the reactor (Ref. 11)
- An experiment to determine the effect of increased radial shielding (Ref. 12)
- An experiment with 5 wt% boric acid shield solution to provide data for use in optimizing the concentration of the shield solution boric acid (Ref. 13)
- Specification of the shutdown radiation dose rate measurements to be taken during movement of the ML-1 reactor skid to the GCRE facility (Ref. 14)

e. Recuperator Low Pressure Outlet Temperature Study: The discrepancy between the measured and calculated temperatures (based on an energy balance across the recuperator) at the low pressure outlet of the recuperator was studied. The measured temperature is historically 50 to 100°F lower than the calculated temperature; the average difference is approximately 70°F. This anomaly is apparent in Table 2-4 where the measured temperature is shown as 388.0°F and the balanced temperature is 479.8°F. The possibility that the temperature difference was the result of gas leakage from the high pressure to the low pressure pass of the recuperator was evaluated. The calculations indicated that a bypass flow of approximately 20% would account for the 70°F difference; such a bypass flow is not consistent with the aerodynamic performance of the plant.

A temperature rake was installed at the low pressure recuperator outlet flange to determine if a large temperature gradient existed across the 13 in. duct which would not be sensed by the single thermocouple previously installed in this location. The preliminary data developed by the temperature rake are shown in Figure 2-16; the temperature gradient suspected is apparent from these data and the indicated average (mixed) gas temperature of 480°F is quite reasonable. Note that the temperature at the top of the duct (the location of the single thermocouple previously used for this measurement) is the lowest of all the indicated temperatures.

f. Overspeed T-C Set Operation: An overspeed test was conducted during ANSOP 16650 to determine the effect on the shaft power output of operating the CSN-1 t-c set in excess of the rated speed (22,000 rpm). The preliminary results of this test are shown on Figure 2-17; the circles are plots of the data, including minor perturbations as the result of slight changes in operating conditions; the triangles represent the overspeed data normalized to constant  $P_1$ ,  $T_1$  and  $P_7$  conditions. The normalized data indicate a slight gain in power with speed change which would not be reflected in net power plant output, since these data do not account for the effect of the degradation of pre-cooler performance with increased mass flow. It is concluded that operation of the t-c set in excess of the rated speed will have little effect on overall net plant output power.

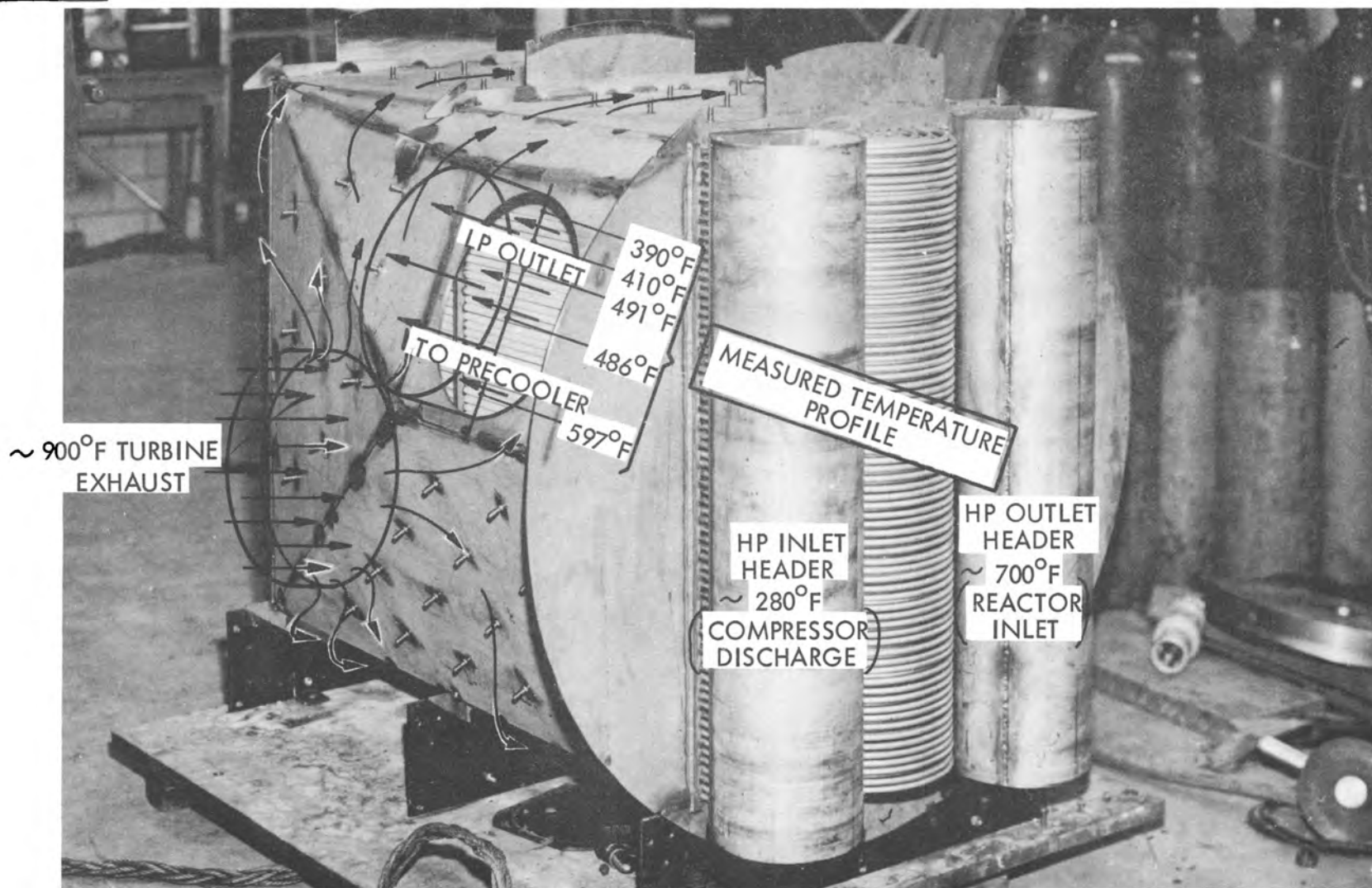


FIGURE 2-16. TEMPERATURE DISTRIBUTION GRISCOM-RUSSELL RECUPERATOR TUBE BUNDLE ASSEMBLY

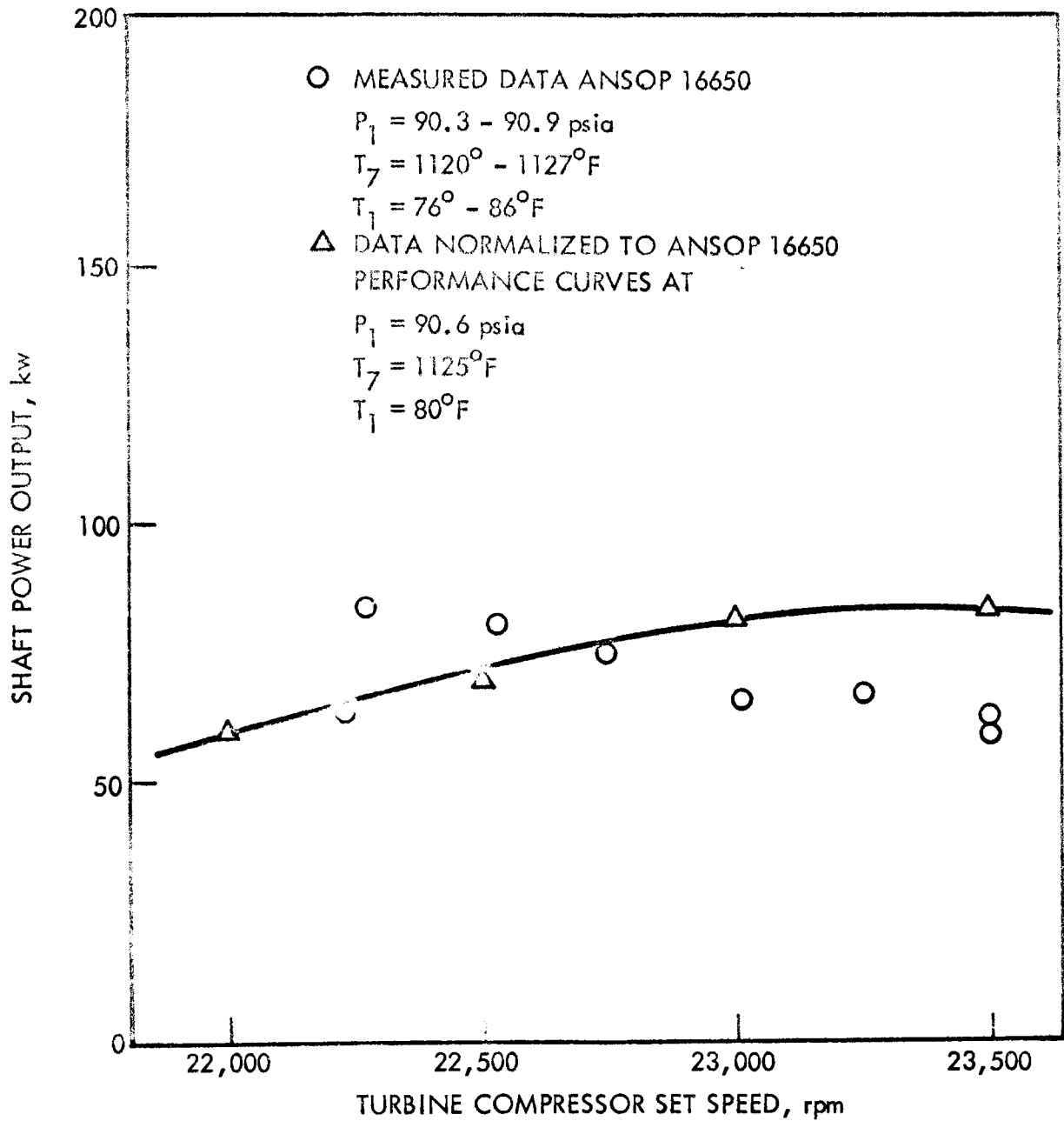


FIGURE 2-17. CSN-1A OVERSPEED PERFORMANCE

112-68-305

### 3.0 ML-1 DEVELOPMENT AND IMPROVEMENTS

#### 3.1 Reactor and Auxiliaries

The procurement of the major equipment for the fabrication of a mockup of the ML-1A gas generation and storage skid was 90% complete by the end of December. This equipment will be stored until early FY 66. The gas skid was designed to produce an average of 2.3 scfm of oxygenated nitrogen at a ambient temperature of 70°F. The gas consumption rates at the ML-1 power plant have been monitored continuously to validate the capacity of the skid (Figure 3-1). The consumption of the ML-1 is greater than would be anticipated for a field plant because approximately 1 scfm is consumed for sampling and because changes in the loop pressure to accommodate data requirements and numerous plant shutdowns and startups to accommodate experimental requirements result in significant purging of the gas. The steady state leakage of the ML-1 is estimated at 1.5 scfm. It is anticipated that improvements in the makeup system, the precoolers and the gas duct joints will reduce this to 0.5 scfm.

#### 3.2 Power Conversion

a. Clark Turbine-Compressor Set (CSN-1): The analysis of the performance of the CSN-1 t-c set during ANSOP 16650 is reported in Section 2.5. The additional evaluations are discussed below:

- The interstage instrumentation installed during ANSOP 16650 indicates that the static pressure loss is approximately 60% higher in the compressor inlet at near design flow conditions than predicted. This significant difference clearly shows that redesign of the compressor inlet configuration probably would improve t-c set performance.
- The measured static pressure rise in the CSN-1A compressor discharge (from the eleventh stage to the compressor discharge flange) was 6.4 psid. Under similar conditions, a 100% efficient diffuser would have a static pressure rise of 13 psid. It is apparent from these data that modification of the compressor discharge configuration could significantly improve the diffuser efficiency with a corresponding increase in compressor efficiency.

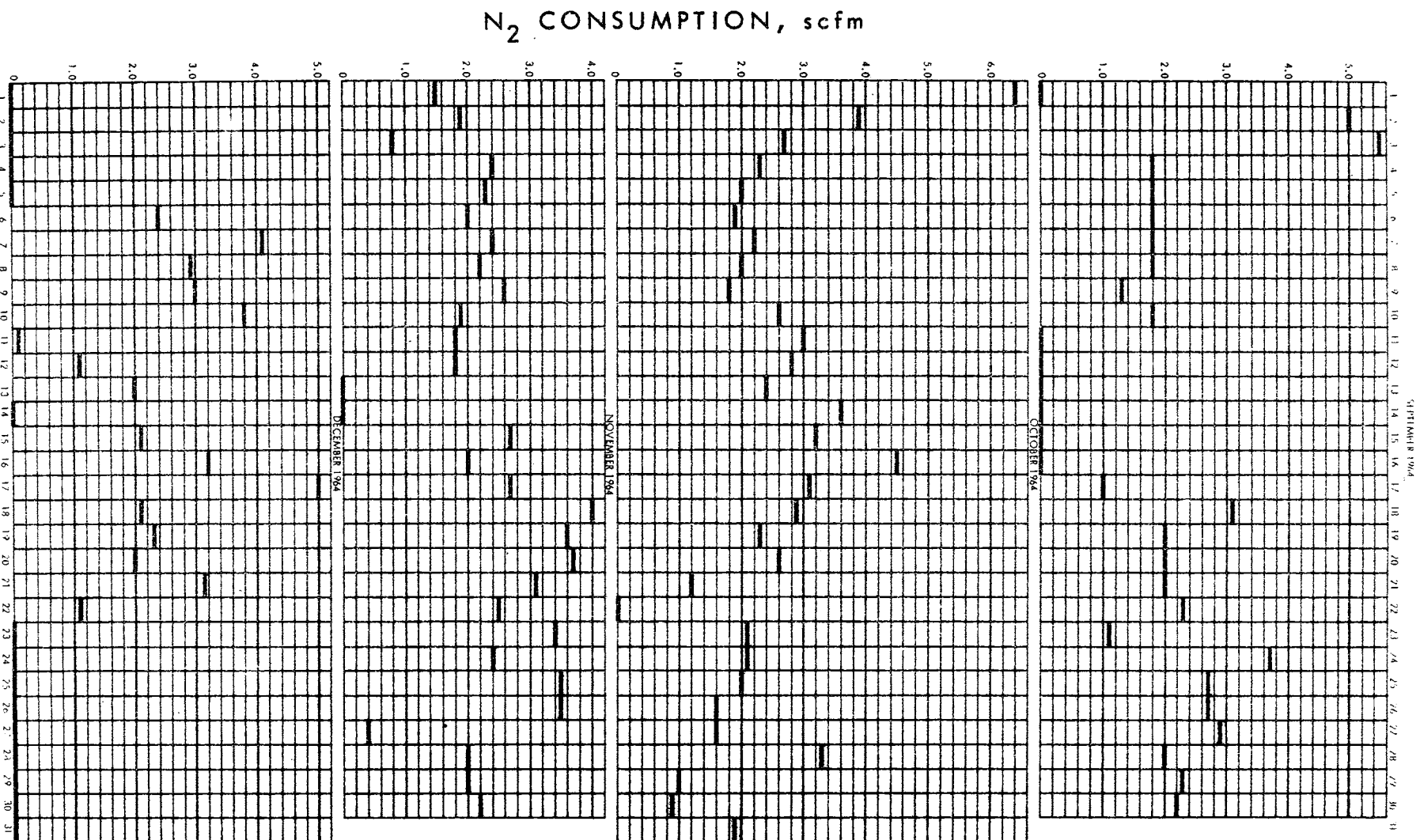


FIGURE 3-1. ML-1 GAS CONSUMPTION

11.2-65-306

- A detailed study of the CSN-1A turbine interstage performance and turbine velocity diagrams indicated that there is an unsatisfactory flow pattern within the turbine. This study indicated that the gas inlet angles to each blade row fall in a region of high loss coefficient which has a very detrimental effect on turbine performance.

The results of the preliminary evaluation were published in detail (Ref. 15).

b. Stratos Turbine-Compressor Set (TCS-670A): During discussions at USAEC-HQ late in November 1964, it was agreed that responsibility for the completion of the development of the TCS-670A would be assigned to the Advanced Power Conversion Development Branch at Ft. Belvoir, Virginia.

c. Dash-Two Turbine Compressor Sets (CSN-2 and TCS-670-2): The revised aerothermodynamic analysis of the CSN-2 compressor was published (Ref. 16).

d. Bearings: Testing of the CSN-1A-type bearings was initiated during the quarter in accordance with the Phase I test plan (Ref. 17). The initial 25-hr shakedown test was satisfactorily completed on 31 December after two previous attempts were terminated due to excessive rotor vibration. The maximum vibration level recorded during the 25-hr test was less than 0.0002 in. The excessive vibration noted earlier was the result of imbalance in the test rotor which was successfully balanced (after two attempts by a vendor) in the Aerojet facilities at Sacramento.

e. Seals: The summary report of the literature search to establish the state-of-the-art of high speed seal concepts suitable for application in the AGCRSP was published (Ref. 18).

f. Equipment Testing: The bearing test fixture was assembled and the test program outlined above (Section d) was performed. As stated earlier, the test rig was disassembled, repaired and reassembled twice during this program because of imbalance of the rotor assembly.

The load and calibration test of the open cycle prime mover was initiated; this test was interrupted to correct minor deficiencies and had accumulated 10.7 hr under load by the end of December.

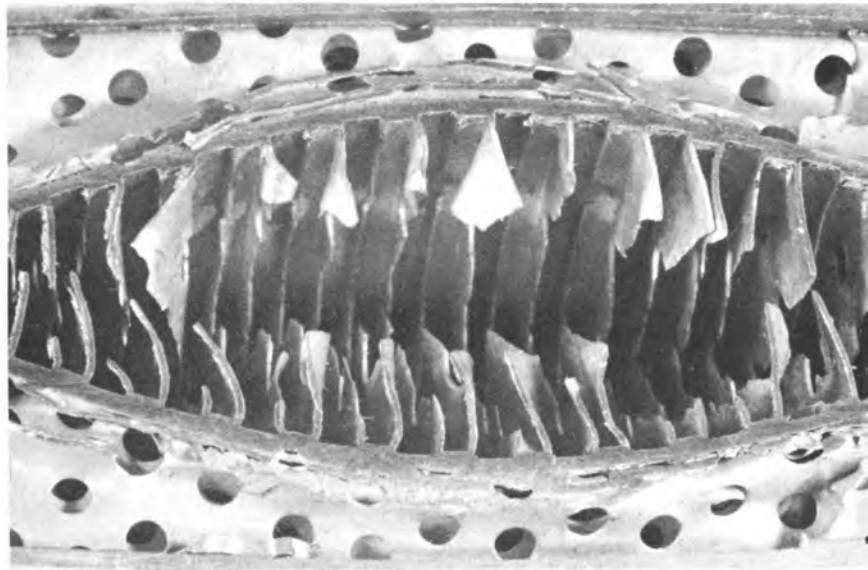
g. Precooler: The fabrication of burst test samples of the various components of the improved plate fin precooler was completed by Stewart-Warner and tests were conducted to verify that the design satisfied the requirements of the ASME Unfired Pressure Vessel Code. Table 3-1 summarizes the results of the burst test program and indicates that all components conform to the requirements of the Code. Stewart-Warner published a preliminary design report (Ref. 19). The review and evaluation of this report revealed that the information on pressure drop, heat transfer, and stress calculations was inadequate. This information was provided in an addendum to the preliminary design report (Ref. 20). The final design was approved (Ref. 21) and the vendor authorized to fabricate the assembly. By the end of December, Stewart-Warner had completed the brazing and leak checking of two of the five modules of the precooler core.

TABLE 3-1 - SUMMARY OF ML-1 IMPROVED PRECOOLER BURST TEST PROGRAM

<u>Component Verified by Test</u>	<u>Operating Pressure (psig) at Maximum Temperature (°F)</u>	<u>Test Article</u>	<u>Burst Pressure (psig) at Temperature (°F)</u>
Inlet collector	105.3 @ 525	06468-X-10-2	665 @ 525
Weld joint, inlet collector to core	105.3 @ 525	06468-X-10-2	665 @ 525
High temperature N <sub>2</sub> side fins	105.3 @ 525	06468-X-43	730 @ 525
Intermediate tempera- ture N <sub>2</sub> side fins	105.3 @ 450	06460-X-44	880 @ 450
Low temperature N <sub>2</sub> side fins	105.3 @ 350	06468-X-47 (Unit #1)	790 @ 350
Outlet end N <sub>2</sub> side fins	105.3 @ 215	06468-X-47 (Unit #2)	980 @ 215
Weld joint, outlet collector to core	105.3 @ 215	Outlet header test sample	660 @ 215
Outlet collector	105.3 @ 215	Outlet header test sample and 06468-X-47 (Unit #2)	660 @ 215 and 980 @ 215

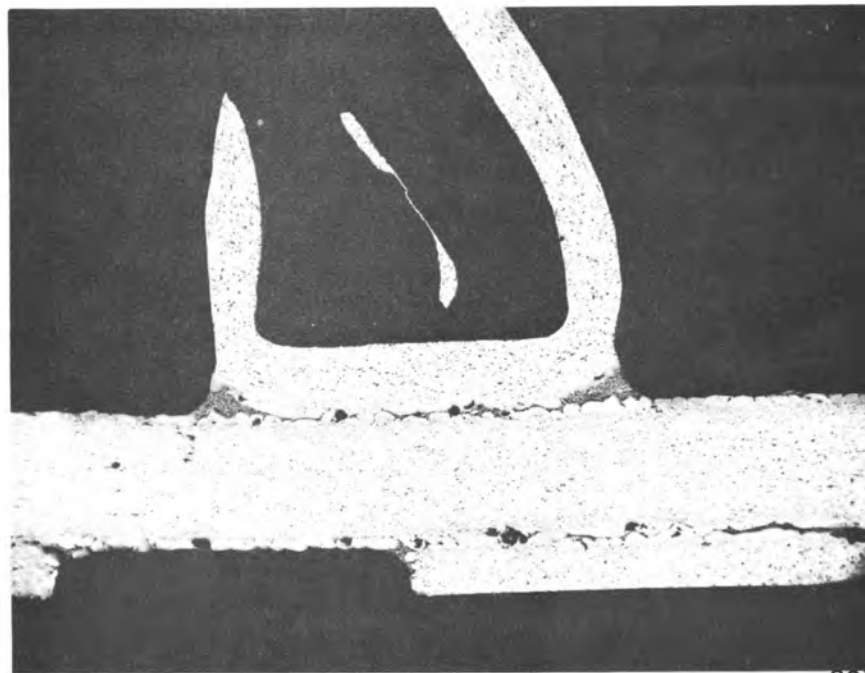
A sample of the core burst test assembly was investigated to determine the mechanism of failure and the adequacy of the braze technique used in fabrication. The sample evaluated was removed from the high temperature nitrogen side fin specimen (06468-X-43) which failed at 730 psig at 525°F. Figure 3-2, a slightly enlarged view of the section at which failure occurred, shows that the rupture resulted from tensile failure of the fins rather than from failure of the brazed joint between the fins and separator plate. Figure 3-3 is an enlarged view of the joint between a typical fin and separator plate. The dark line between the fin section and separator plate and the dark fillet at the corners of the fin section are braze material. It is evident from this photograph that the quality of the braze joint was unaffected by the pressure test. The ductile failure of the fin is also shown by the necked-down appearance of the fin section.

h. Precooler Air-Side Cooling Assembly: The fabrication of the fan assembly support structure for the improved precooler air-side cooling assembly was completed; dummy frames were installed to mock-up the fan assemblies which had not been received by the end of December. The procurement of the fan assemblies was initiated and delivery is anticipated in late January 1965. The fan assembly consists of a seven-bladed, cast aluminum fan mounted on a totally-enclosed, two-speed motor which is supported in an aluminum frame.



4X

FIGURE 3-2. BURST SECTION, ML-1 IMPROVED PRECOOLER CORE  
SAMPLE 06468-X-43



30X

FIGURE 3-3. BRAZE JOINT, ML-1 IMPROVED PRECOOLER CORE  
SAMPLE 06468-X-43

11.2-65-307

i. ML-1 Alternator: The fabrication of the ML-1 alternator was initiated by Fairbanks-Morse in November and delivery is anticipated in April 1965. This unit is a four-pole, air-cooled machine rated at 600 kw at 4160/2400 volt at 0.8 power factor and an ambient temperature of 125°F.

j. Alternator Gear Set: The final design of the 3600/1800 rpm gear set for the alternator was completed by Western Gear Corporation and fabrication was approved early in the quarter. The completed unit was delivered in mid-December 1964 and inspection revealed that the machine conformed to the drawings and specifications. Delivery of the tooling, operating instructions and parts list had not been completed by the end of December.

An engineering evaluation of the lifetime of the gear set indicated that the gears were capable of operating for more than 50,000 hr at rated power but that a credible accident involving a three-phase dead short of the alternator could result in failure of the alternator drive gear. As a consequence, a spare set of gears was ordered; these components were fabricated at the same time as the original parts which minimized cost and maximized assurance of compatibility of the spare parts and the original equipment.

An analysis was completed which verified that the critical speeds of the rotating machinery, including the alternator, gear set and CSN-1 t-c set, do not occur near the t-c set rated speed.

k. Power Conversion Skid Modification: The modification of the recuperator to provide for compatibility with the inlet duct of the improved precooler was completed. The unit was hydrostatically tested to verify the mechanical integrity of the pressure vessel and modified connecting ducts and flanges. The modification of the turbine inlet duct to incorporate a flow measuring venturi was completed. Both the recuperator and turbine inlet duct were temporarily stored as the modification program for the power conversion skid No. 1 was brought to an orderly conclusion in conformance with the revised power conversion equipment development program.

A preliminary layout and piping and instrumentation diagram was completed for the improved lubrication system for the skid. The improved design significantly simplifies the system by providing an unpressurized sump tank, increased gas-oil separation capacity and increased oil flow capacity to satisfy the lubrication requirements of the alternator gear set.

l. Starting System: The final design of a hydraulic starting system for the ML-1 power plant, capable of providing 300 ft-lb of breakaway torque and 25-30 kw of driving power up to rated speed, was brought to an orderly conclusion in conformance with the revised program for development of the power conversion equipment. The completed final design layout defines the method of installation of the starting system on the skid and lists all the components.

m. GDEF Conceptual Design: A conceptual design for a power conversion equipment closed-cycle test loop was completed at the request of the USAEC-ID (Ref. 22). This test loop, tentatively designated as the Gas Dynamics Experimental Facility, is intended for use in further development of t-c sets as well as in evaluating the performance of other power conversion components.

### 3.3 Instruments and Controls

a. Overspeed Scram Chassis: The fabrication, functional testing and environmental testing of the improved ML-1 overspeed scram chassis were completed (Ref. 23 and 24). The chassis was temperature cycled in an environmental test chamber between  $-25^{\circ}\text{F}$  and  $150^{\circ}\text{F}$  and, simultaneously, the line voltage was varied from 105 to 125 v a.c. The maximum error in the indicated turbine speed was less than 1% under these test conditions.

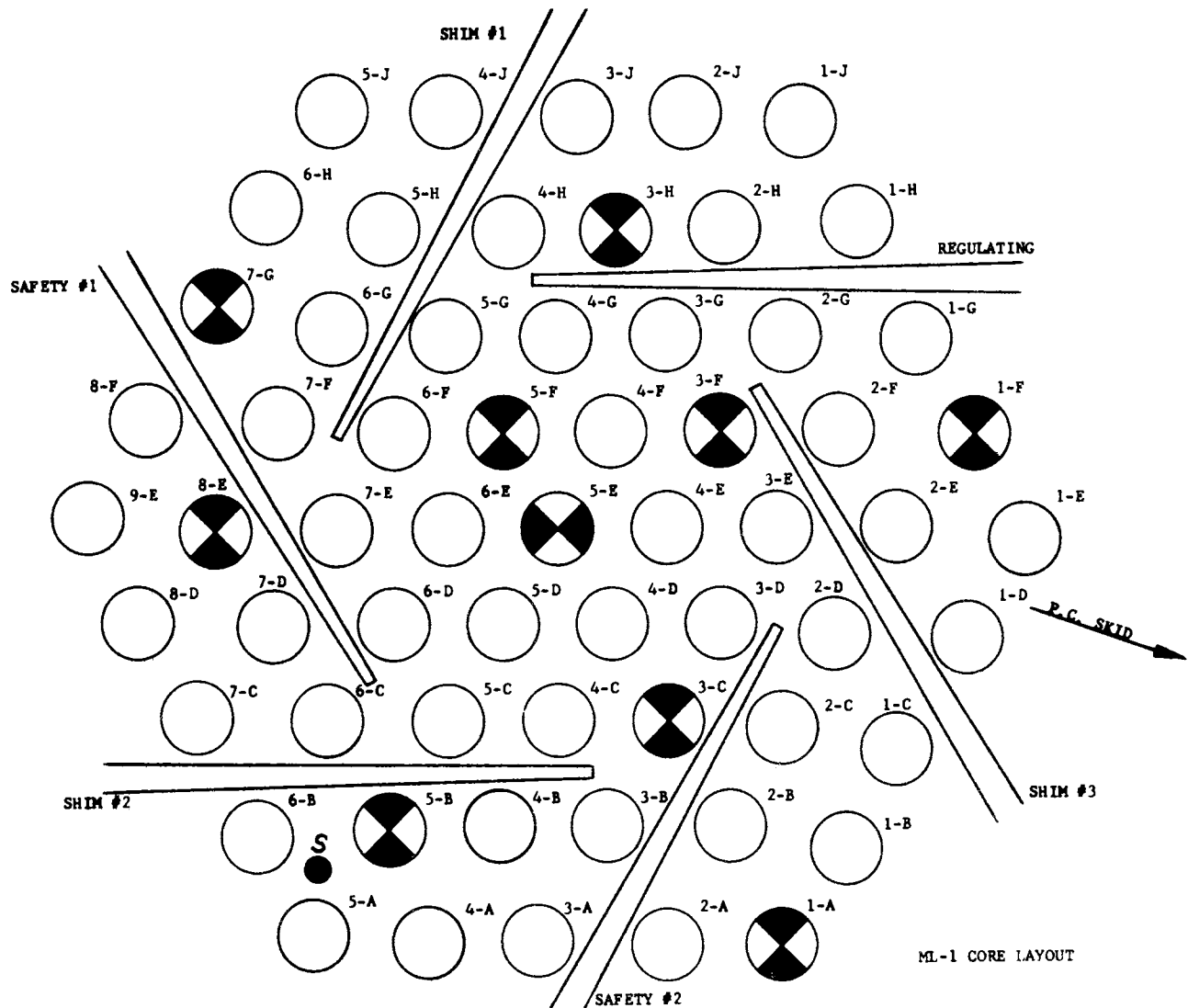
b. Improved ML-1 Speed and Temperature Controller: A summary of the present status of the speed and temperature controller equipment and the anticipated test program for this system was published (Ref. 25). A series of tests was conducted at the ML-1 during December 1964 to familiarize power plant operators with the manual mode of speed control.

### 3.4 ML-1-II Core Elements

a. ML-1-II Thermal and Neutronic Analysis: In consideration of the existence of a non-optimum flow distribution pattern in the ML-1-I core (with an associated elevation in operating temperature of the inner ring of fuel elements) and of the possibility that such a condition could occur in the ML-1-II core, the instrumentation plan for the latter core loading was reviewed. The reference ML-1-II instrumentation plan provides for three instrumented fuel elements. As a result of the review discussed above, it was recommended that the number of instrumented elements in the ML-1-II core loading be increased to ten. Nine of these elements will be assembled with the thermocouple hot junction at  $X/L = 1.0$ ; these elements will be loaded into the core so that at least one instrumented element is loaded in each ring of fuel elements. The tenth element will be assembled with the thermocouple junction at  $X/L = 0.5$  (peak power location); this element will be loaded into the central core position. The proposed instrumented fuel element pattern for the ML-1-II core is shown in Figure 3-4. The design of the instrumented elements will be similar to that used in the ML-1-I core loading; the central pin will be unfueled and provision will be made for removing and replacing the thermocouple in this pin. The reduction of 355 grams of contained U-235 associated with the 10 unfueled pins was calculated to be equivalent in reactivity worth to about half a peripheral fuel element.

b. IB-17R Out-of-Pile Support: Coolant/cladding and cladding/fuel compatibility tests are being performed out-of-pile to develop control data for evaluation of the IB-17R in-reactor test elements. These tests completed 10,000 hr of exposure during the quarter. Specimens were removed for metallographic evaluation and for the determination of room and elevated temperature mechanical properties. These evaluations were in progress at the end of December. The experiment is continuing; 11,600 hours of exposure had been accumulated on the remaining samples by the end of December.

c. IB-17R-2 Metallurgical Evaluation: The results of examination of IB-17R-2 pins 1 and 3 in the NRTS Hot Cells were discussed in the previous report (Ref. a). In that discussion, the formation of a second phase in certain locations on the inner surface of the cladding of pins 1 and 3 was mentioned and the need for additional evaluation of this situation was stated. During this



LEGEND:

THERMOCOUPLE AXIAL LOCATION (X/L):

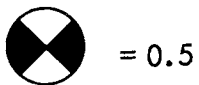
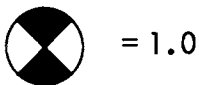


FIGURE 3-4. ML-1-II INSTRUMENTED FUEL ELEMENT PATTERN

112-65-308

quarter, two sections from each pin were transferred to the AGN Hot Cell and detailed investigation of the apparent second phase formation was initiated. The examination of pin 1 confirmed the presence of the second phase (pi phase as identified in the IB-8T-1 program (Ref. c)) in the colder regions of the pin. The penetration of the second phase as far as 0.012 in. from the inner surface of the cladding at  $X/L = 0.5$  (Figure 3-5) was significant. The detailed examination of pin 3 confirmed the earlier findings that the second phase existed at all locations on the inner surface of the fuel pin cladding. Figure 3-6 reproduces a photomicrograph of the cladding at several locations in this pin.

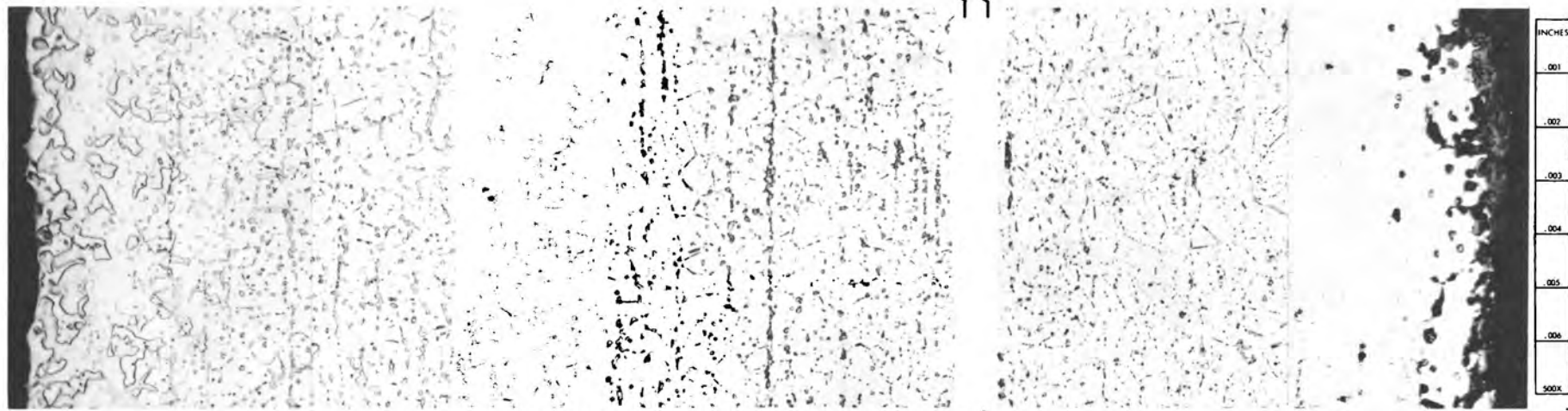
The layer phase observed in the fuel pin cladding interface of some specimens from both pins 1 and 3 (Ref. a) was apparent in the work done during this report period. The identity of this phase has not been successfully established; it might consist of a higher uranium oxide formed by reaction between the  $UO_2$  fuel and air which had entered the pin through porous thermocouple welds.

The detailed examination of the fuel from pins 1 and 3 revealed that the cracking of the  $UO_2$  bodies was not as severe in these samples as had been reported earlier (Ref. a). The peculiar structure observed at the interface between the  $UO_2$  particles and the beryllium matrix (Figure 3-7) in some fuel pins was investigated thoroughly. This investigation revealed that the edge of the  $UO_2$  body was irregular, probably as the result of excessive sintering temperatures or migration of fine  $UO_2$  particles toward larger fuel particles during exposure in the reactor, and that the polishing and etching operations created a false structure which had the appearance observed earlier. Carefully polished and etched samples of the fuel did not reveal the unusual formation observed earlier but clearly showed the irregular edge of the  $UO_2$  body and the dispersion of fine particles of  $UO_2$  in the beryllium matrix (Figure 3-8). Comparison of the irradiated fuel specimens with unirradiated control samples indicated that no significant increase in the grain size of the BeO matrix had occurred as a result of the IB-17R-2 irradiation.

The preparation of cladding samples for tensile testing at BMI was completed. These samples were prepared by removing the fuel from the fuel pins in boiling nitric acid solution. In some cases, the time required for removal of the fuel was quite long and, as a consequence of the microstructural condition of the alloy (associated with the exposure temperature) or of contamination by fuel or fuel oxidation products, the cladding was severely attacked in some locations. Figure 3-9 shows a cross section of a typical fuel pin which was severely attacked by the nitric acid.

Eleven fuel pin cladding specimens were subjected to room and elevated temperature tensile testing. Table 3-2 summarizes the results of this work. Note that specimens 11, 12 and 16 were tested twice; the second test was performed on the longer section of the broken original specimen. Although comparative values from tubing specimens exposed in the laboratory are not available at this time, comparison with the mechanical properties of unexposed sheet Hastelloy X indicate that the room temperature yield and ultimate strengths of the irradiated material is higher than unexposed material but that the elevated temperature strength of the irradiated material is lower than unexposed material.

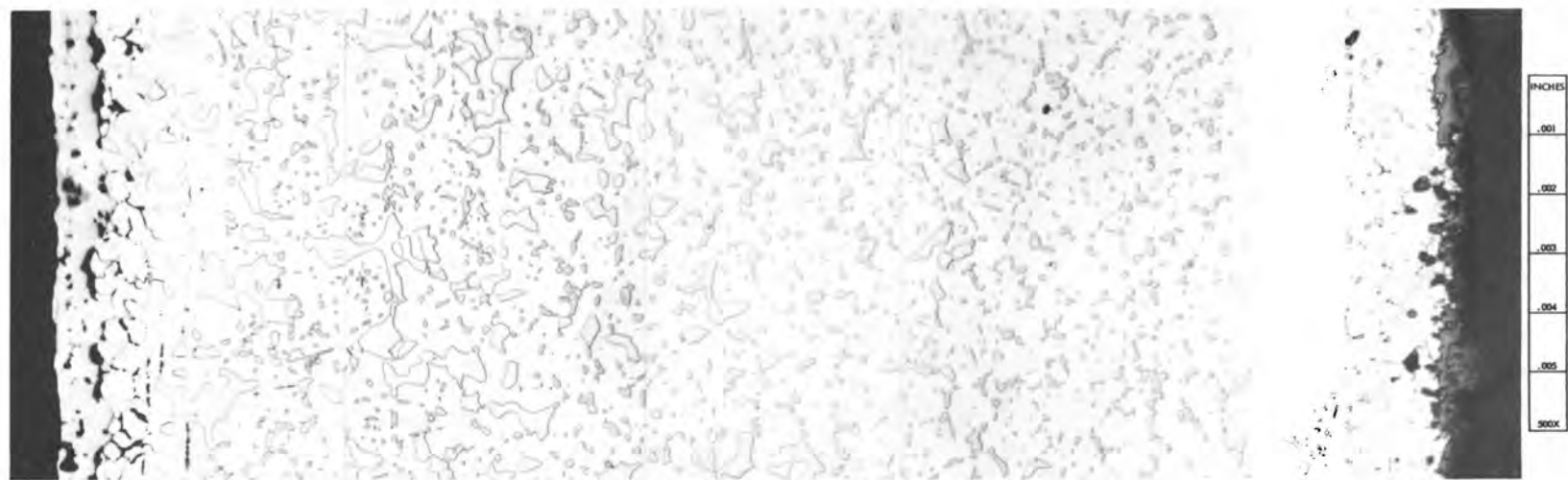
11.2-65-309



STRUCTURE AT  $X/L = 0.25$ , SURFACE TEMPERATURE  $1260^{\circ}\text{F}$

Report No. IDO-28641

48



INNER SURFACE

STRUCTURE AT  $X/L = 0.5$ , SURFACE TEMPERATURE  $1385^{\circ}\text{F}$

OUTER SURFACE

FIGURE 3-5. HASTELLOY X STRUCTURE, 1B-17R-2 FUEL PIN 1 AT  $X/L = 1.0$

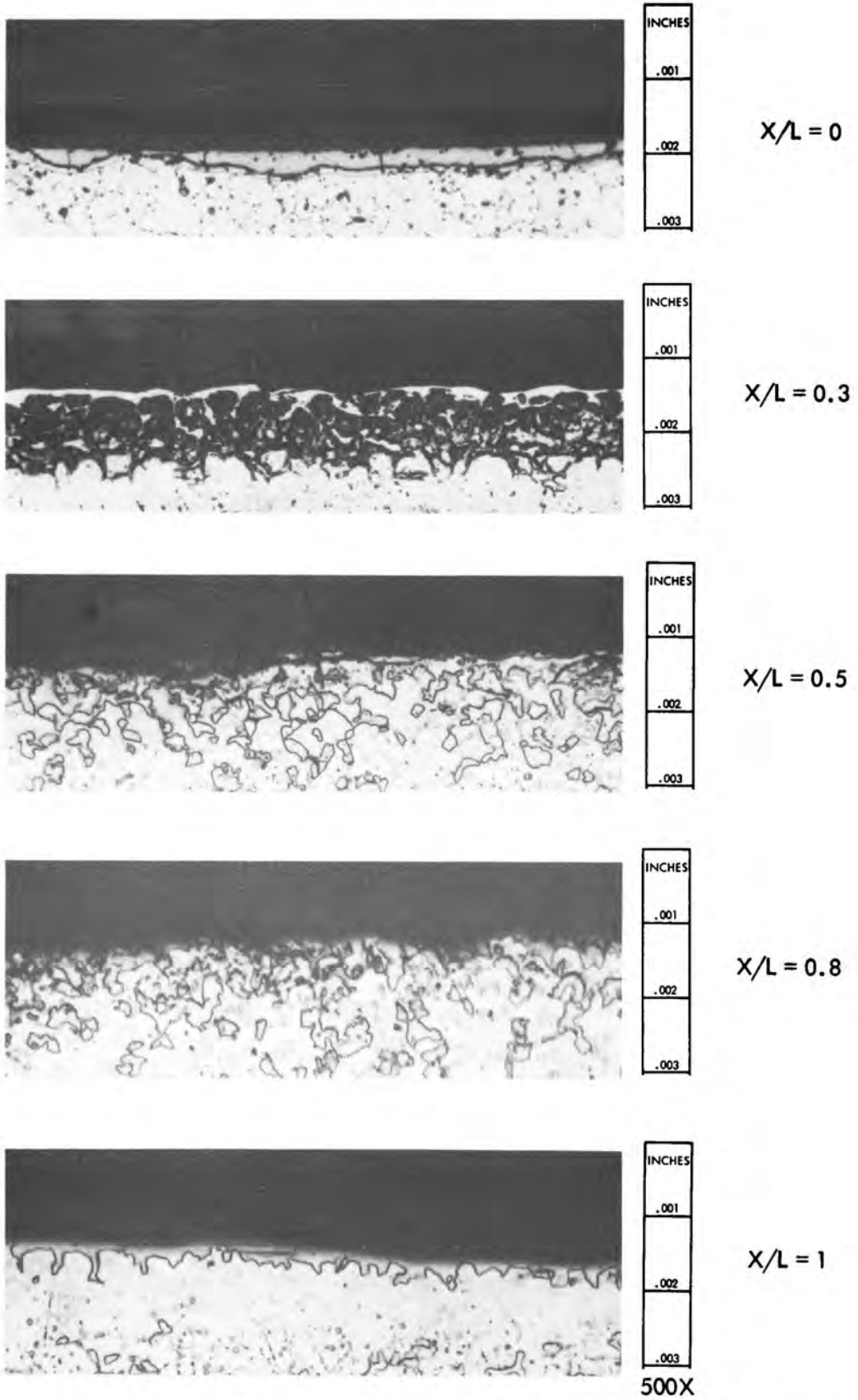


FIGURE 3-6. INSIDE SURFACE OF PIN 3, IB-17R-2 ELEMENT, AT SELECTED LOCATIONS

11.2-69-310

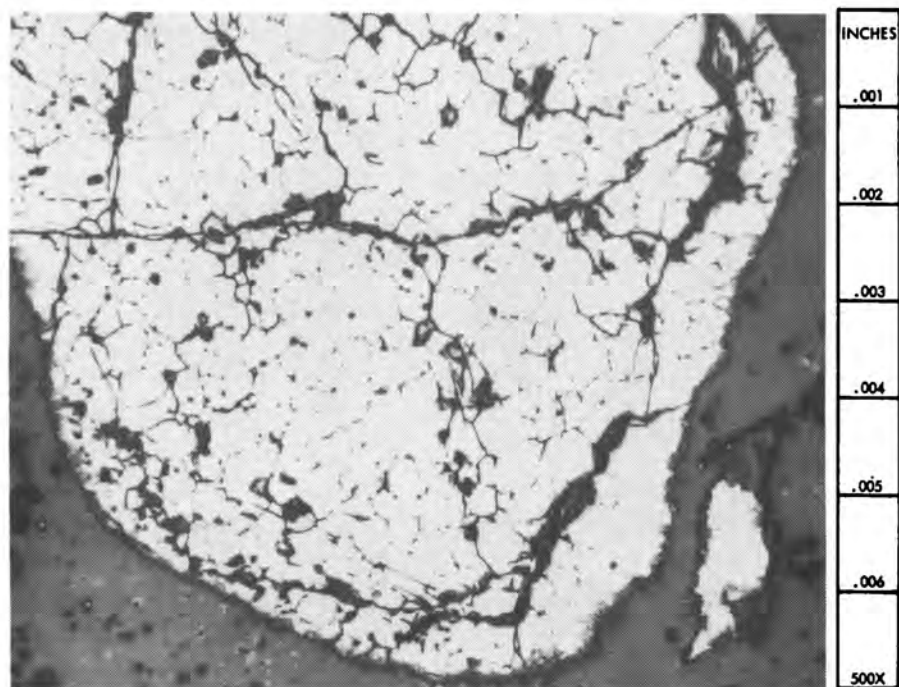
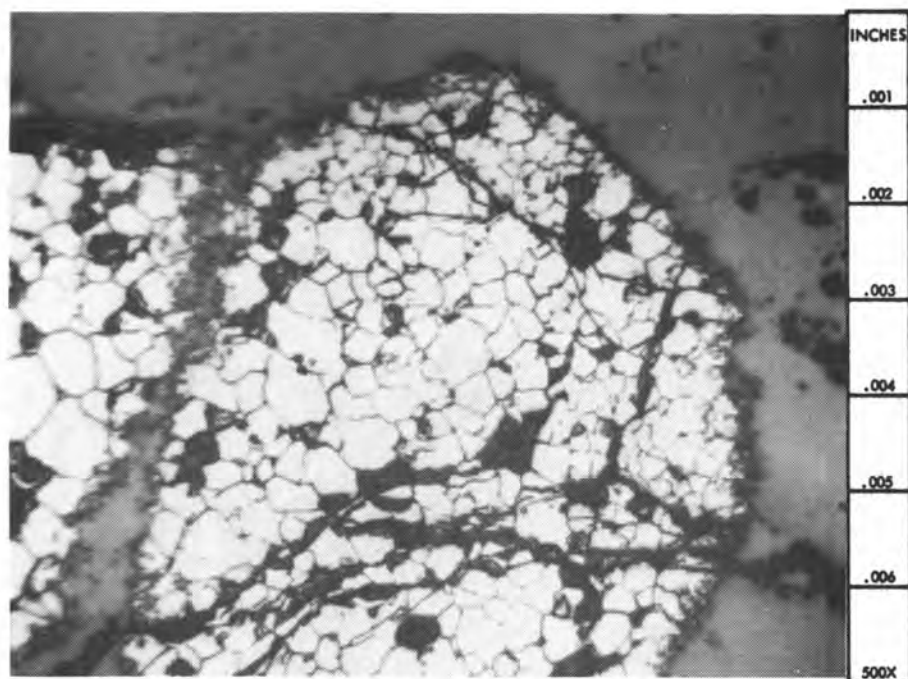


FIGURE 3-7. UNUSUAL STRUCTURE AT  $\text{UO}_2$  PARTICLE BOUNDARIES, IB-17R-2, PINS 1 AND 2

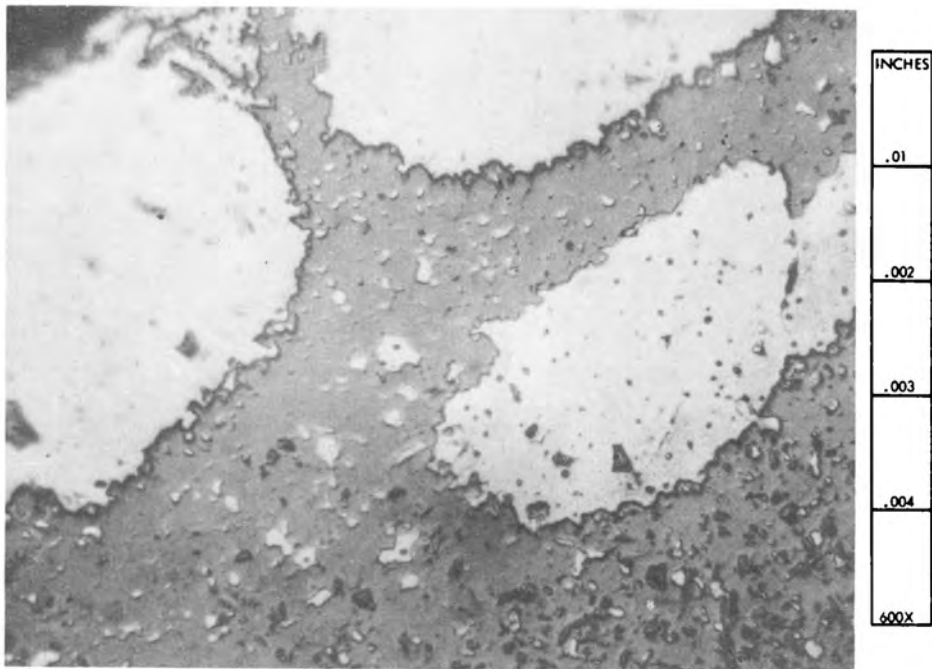
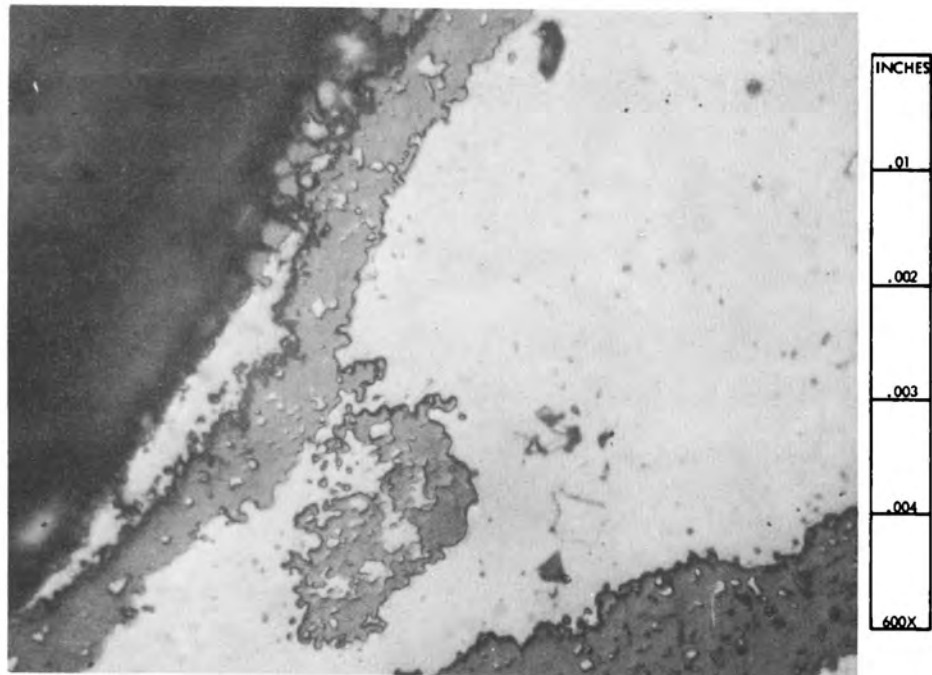


FIGURE 3-8. IRREGULAR UO<sub>2</sub> PARTICLE SURFACE PRESENT IN IB-17R-2 FUEL

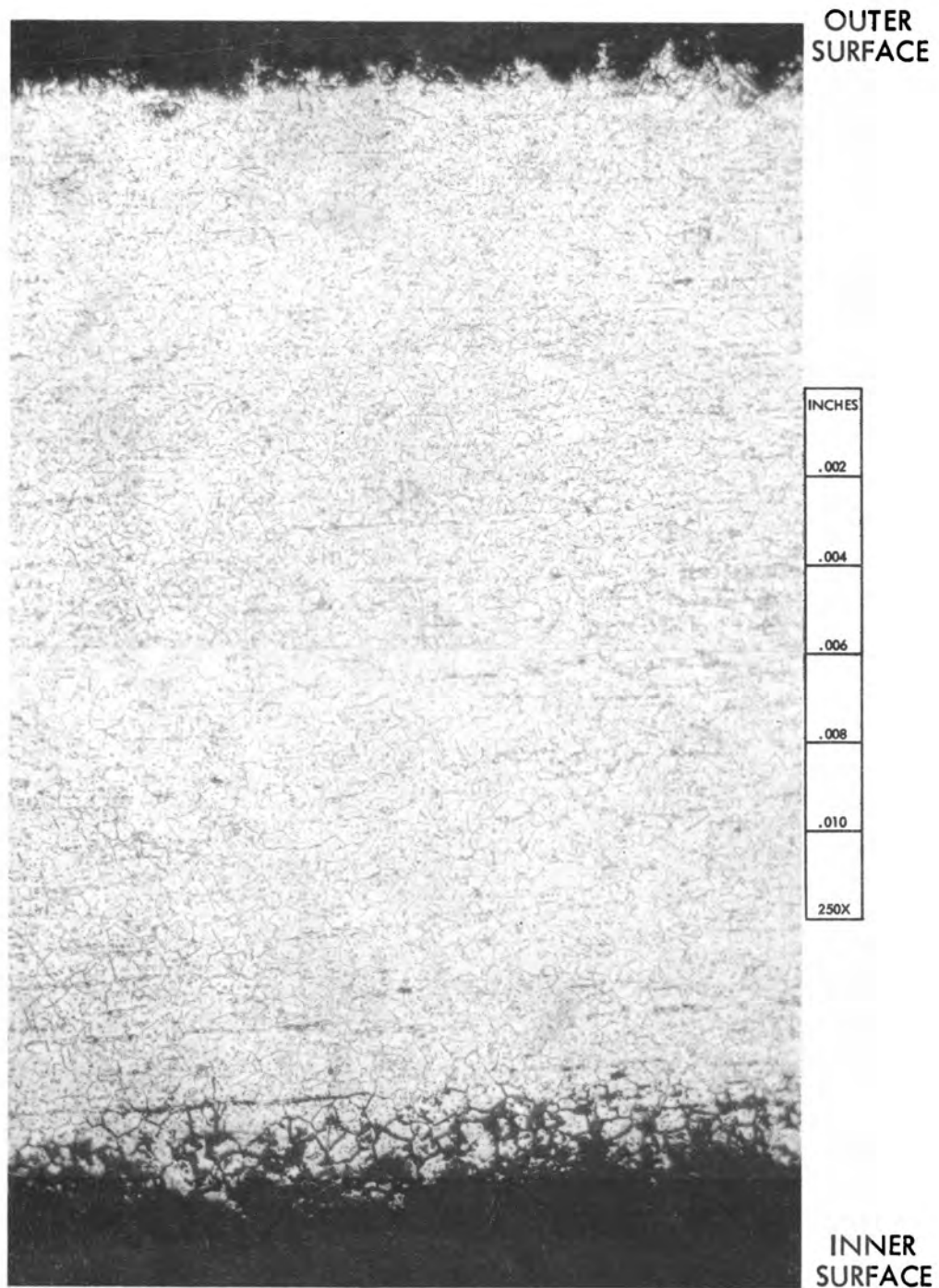


FIGURE 3-9. IB-17R-2 FUEL PIN 2 SHOWING ATTACK BY NITRIC ACID AT SURFACE

11.2-65-313

TABLE 3-2 - MECHANICAL PROPERTIES OF IRRADIATED  
HASTELLOY X TUBING FROM THE IB-17R-2 EXPERIMENT

<u>Pin Number</u>	<u>Testing Temperature, °F</u>	<u>0.2% Offset Yield Strength x 10<sup>3</sup> psi</u>	<u>Ultimate Strength x 10<sup>3</sup> psi</u>	<u>Total Elongation, %</u>	<u>Remarks</u>
6	RT	61.8	113	7.63 <sup>(a)</sup>	b
18	RT	83.0	118	2.75	c
11a	RT	63.0	-	-	d
11b	RT	78.4	-	-	d
1	1500	25.8	30.3	2.35	e
16a	1500	25.5	26.3	0.90	e
16b	1500	32.5	33.0	0.75	f
10	1750	15.5	16.5	0.80	e
15	1750	16.2	16.3	1.00	f
3	1750	11.4	11.4	0.25	e
7	1850	5.6	5.6	0.85	f
12a	1850	7.4	7.4	0.80	e
12b	1850	8.8	8.8	0.35	e

(a) Elongation measured on specimen

(b) Slipped, then broke

(c) Broke in top section of pin

(d) Slipped, never broke

(e) Broke in grip at ferrule

(f) Broke at top of extensometer

RT = Room temperature

At the end of December, BMI was preparing cladding specimens for micro-probe analysis and was extracting particles of the second phase for subsequent X-ray diffraction analysis.

d. ML-1-II Fuel Pin Tubing Burst Testing: The burst tests of smooth tubing, to obtain data for comparison with the finned-tube tests reported earlier (Ref. a) were completed. Evaluation of the test data indicated little difference in rupture strength between the smooth and finned tubing and that, as reported earlier (Ref. a), the finned tubing appeared to have sufficient strength to withstand the "worst case" loss-of-coolant accident situation.

e. Fuel Pin Tip Testing: The final report of the experiments conducted in the previous quarter (Ref. a) was being prepared at the end of December.

f. IB-17R-3 In-Pile Test: The test irradiation of the uninstrumented ML-1-II prototype fuel element (IB-17R-3) in the GETR loop continued without incident until 24 November 1964, by which time 9501 hr of nuclear operation had been accumulated. A release of fission product activity into the test loop and GETR containment enclosure occurred on 24 November and, as a consequence, the reactor was shut down and the test element removed. By the end of December, the IB-17R-3 element had been transferred from the test loop section to the GE-VAL hot cell for disassembly and leak check of the fuel pins.

Throughout the IB-17R test program erosion of the test elements was monitored by radiochemical analysis of samples of material removed from the filters in the gas loop. The technique used involved swabbing the surface of the filter and analyzing the material collected on the swab for the metallic elements expected to be present. Although this technique ignored variations in the size of the sample, it was felt that changes could be observed by comparing the ratio of the elements to the indicated quantity of the iron in the sample. This approach was demonstrated to be unsatisfactory when reduction of the data from cycles 52 through 57 revealed that the Cr:Fe ratio varied by a factor of 1,000 (0.015 to 15.0) and the Co:Fe ratio varied by a factor of 8.5 (0.002 to 0.017). This monitoring was terminated at the end of November when the IB-17R-3 test element was removed from the GETR.

g. ML-1-II Core Fabrication: The delivery of the ML-1-II core was postponed until November 1966 to permit evaluation of the cause of the apparent failure of the IB-17R-3 test element and the effect of the findings of this investigation on the reference design for the ML-1-II core. A 60-day "hold" was placed on the Hastelloy X fuel pin tubing order which had been let in mid-November. The vendors for the spider castings were also advised to delay the production of the final castings pending the results of the IB-17R-3 evaluation. All other long lead procurement was postponed until 1 July 1965.

The evaluation of sample upper and lower spider castings continued throughout the quarter. The upper spider sample was found to be dimensionally acceptable but Zyglo testing of representative metallographic specimens revealed several unacceptably porous sections. This situation was being discussed with the vendor at the end of December. The dimensional evaluation of the lower spider sample revealed that minor changes in the casting molds will be required. This problem was discussed with the vendor and new samples will be provided for evaluation after the necessary modification.

The specifications for assembly of ML-1-II fuel element were completed and the assembly fixture drawings were revised to conform to the latest fuel element design.

III. ML-1 TECHNOLOGY4.0 FUEL ELEMENT TECHNOLOGY4.1 Hastelloy X Cladding Evaluation

a. Hastelloy X Laboratory Corrosion Tests: The laboratory exposure of low cobalt Hastelloy X fuel pin tubing in air at temperatures from 1300 to 1800°F (initiated under the ML-1 air cycle program) was completed during the quarter when 10,000 hr of exposure was reached. The metallographic evaluation and tensile testing of specimens is scheduled to be initiated early in the next quarter.

Laboratory exposure of Hastelloy X tubing and sheet specimens in air at 1900 and 2000°F was begun during the quarter. Specimens of material from five heats are being exposed. The identification and composition of the various specimens is shown in Table 4-1.

TABLE 4-1 - COMPOSITION OF HASTELLOY X  
SPECIMENS FOR HIGH TEMPERATURE CORROSION TESTS

Heat No.	Form	Composition, wt%										
		Cr	W	Fe	C	Si	Co	Ni	Mn	Mo	P	S
E9517*	Tubing	22.37	.31	18.41	.12	.61	0.07	Ba1	.75	8.73	.010	.005
X14593**	Tubing	22.16	.44	18.27	.10	.57	1.94	Ba1	.64	8.80	.012	.006
E9505	Sheet	22.48	.50	17.96	.10	.56	0.28	Ba1	.40	8.53	.006	.004
X4762	Sheet	21.52	.45	18.15	.10	.70	2.07	Ba1	.62	9.00	.010	.005
X34381	Sheet	22.21	.67	18.40	.10	.82	0.84	Ba1	.58	8.83	.017	.004

\* Used for ML-1 fuel element cladding.

\*\*Used for IB-17R fuel element cladding.

During the quarter, sheet and tubing specimens which had reached 100 and 1,000 hr of exposure at 1900°F were removed for evaluation. The metallographic examination of the 100-hr specimens was completed (see discussion below) and similar evaluation of the 1,000-hr specimens was in progress at the end of December.

Metallographic evaluation of the 100 hr 1900°F tubing material revealed the following:

• The low cobalt tubing (heat E9517) was characterized by a uniform oxide scale, an internal band of oxide and relatively few intergranular penetrations. On the other hand, the high cobalt tubing (heat X14593) contained frequent intergranular penetrations and accompanying internal oxidation sites (Figure 4-1).

• Semicontinuous precipitates were observed along the grain boundaries of both materials as was the presence of small spherical precipitates within the grains. In general, the commercial cobalt material appeared to have a smaller grain size than the low cobalt alloy (Figure 4-2).

The oxidation penetration measurements are presented in Table 4-2.

TABLE 4-2 - OXIDATION OF HASTELLOY X EXPOSED IN AIR  
AT 1900°F FOR 100 HOURS

<u>Heat No.</u>	<u>Form</u>	<u>Average Maximum Depth, in.</u>		<u>Typical Thickness of Surface Oxide, in.</u>
		<u>Intergranular Oxidation</u>	<u>Internal Oxidation</u>	
E9517	Tubing	0.0012	0.0007	0.0005
X14953	Tubing	0.0016	0.0009	0.0004
E9505	Sheet	0.0014	0.0010	0.0003
X4762	Sheet	0.0012	0.0008	0.0002
X34381	Sheet	0.0012	0.0009	0.0002

b. Continuous Weighing Test: A laboratory experiment utilizing a multiple channel continuous weighing apparatus to evaluate the oxide film growth and scaling characteristics of Hastelloy X at 1750°F in air was terminated after 5460 hr. At the end of December, the data had been reduced and X-ray diffraction analysis of the spalled oxide film was in progress. A second test, using the same apparatus and specimens of the same materials at 1850°F, was initiated during the quarter. By the end of December, this test had accumulated 1206 hours of operation. The data developed in the 1850°F test to date are compared with data generated in the 1750°F test in Table 4-3.

TABLE 4-3 - COMPARISON OF WEIGHT GAINED AT 1750°F and 1850°F

<u>Sample Cobalt Content</u>	<u>Temperature of Sample Exposure, °F</u>	<u>Weight Gain (x 10<sup>-3</sup> mg/cm<sup>2</sup> hr)</u>
High	1750	1.92
High	1850	2.12
Low	1750	1.75
Low	1850	2.39

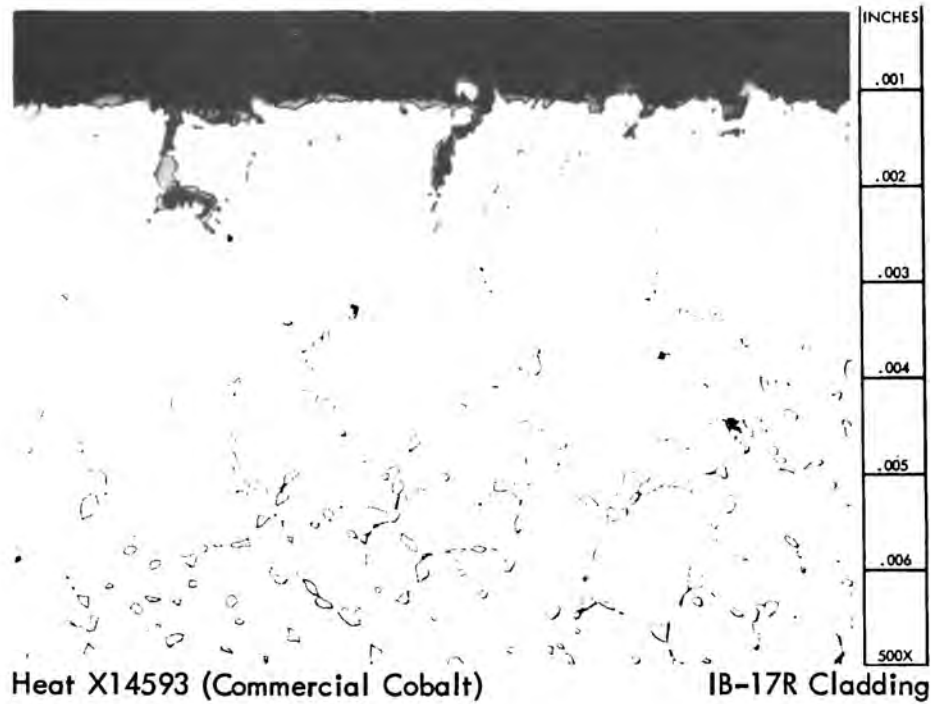
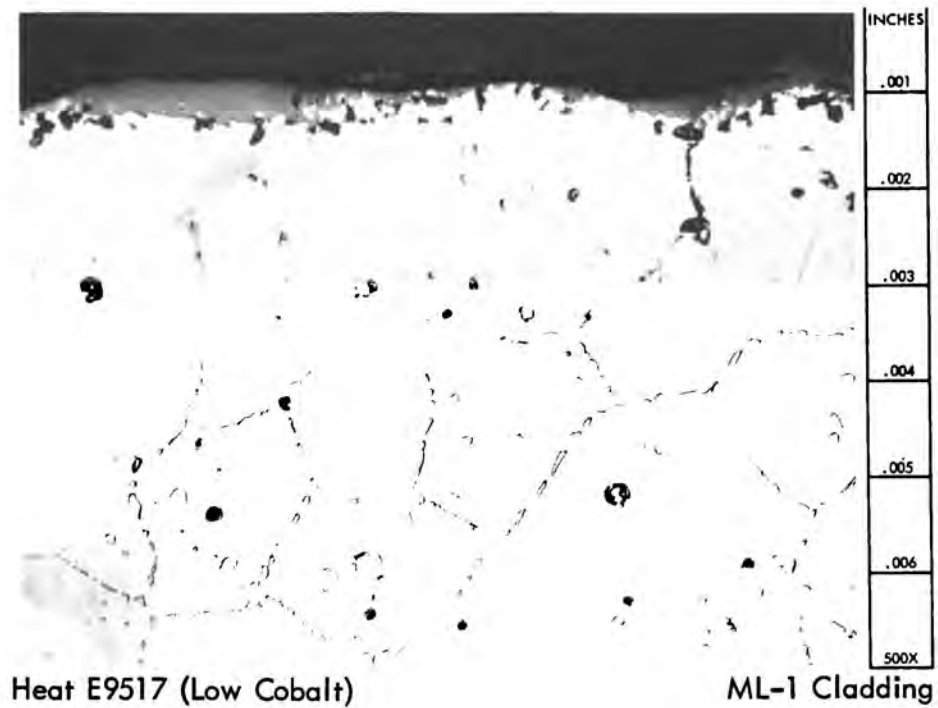


FIGURE 4-1. MICROSTRUCTURE AT OUTER SURFACE OF HASTELLOY X TUBING EXPOSED 100 HOURS IN 1900°F AIR

11.2-65-3/4

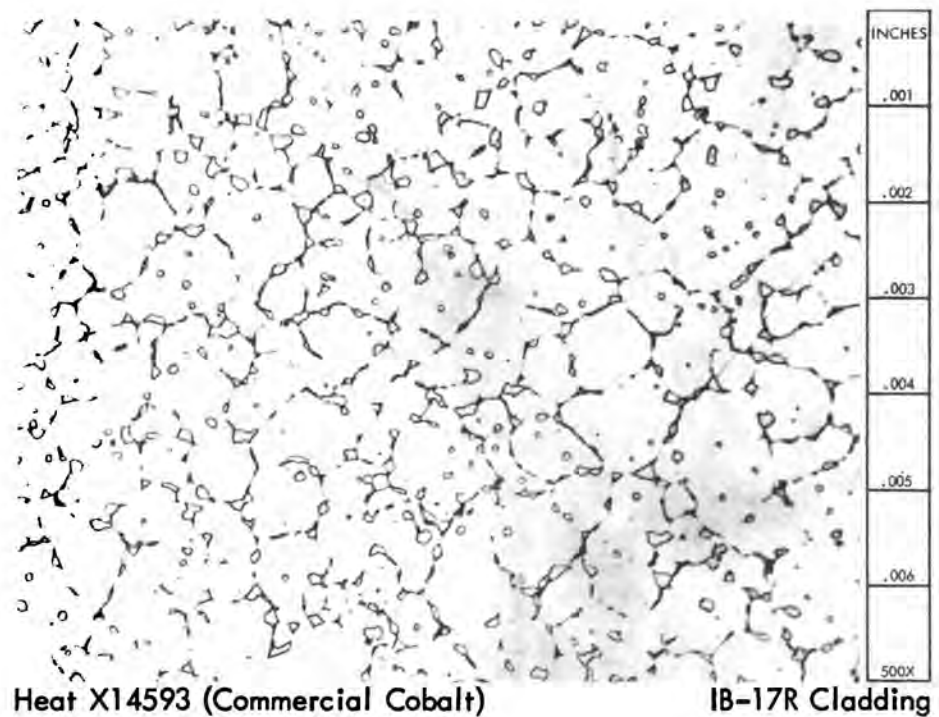
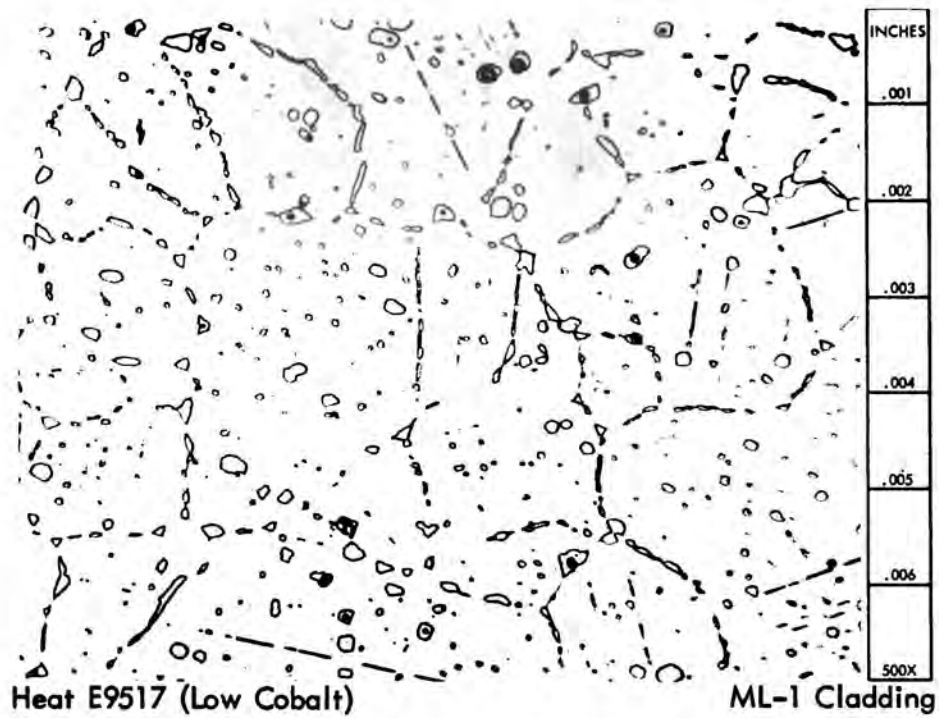


FIGURE 4-2. MICROSTRUCTURE IN INTERIOR OF HASTELLOY X TUBING EXPOSED 100 HOURS IN 1900°F AIR

11.2-65-315

The weight gained by the two alloys at the two test temperatures is compared in Figure 4-3.

c. Hastelloy X Creep Property Evaluation: The objective of this test is to develop creep data on low and commercial cobalt content Hastelloy X at temperatures above 1750°F. The test will be performed in the AGN-developed centrifugal creep test machine which is capable of exposing 60 specimens simultaneously to identical environmental conditions.

The assembly of the centrifugal creep test machine was completed during the quarter and the machine was being calibrated and the operating characteristics of this machine were being determined at the end of December. Figure 4-4 reproduces a photograph of the machine and identifies the major external components. Figure 4-5 shows the test head viewed from above with typical test specimens in place. Figure 4-6 illustrates the configuration of test specimens which will be exposed during this test. Each specimen consists of a gripping section (by which the specimen is attached to the head), a bending section (in which the data relating to creep characteristics is determined) and a loading section to apply stress to the bending section as a result of centrifugal motion to the test head. Variations in the size of the loading section permit the application of a variety of loads to the bending section. Some specimens have reduced cross-sections machined into the loading section to permit the subsequent determination of tensile data.

A 1000-hr test at 1800°F is scheduled to start in mid-January 1965.

d. Hastelloy X Stress Rupture Properties: The objective of this task is to determine the 20,000-hr stress-rupture properties of Hastelloy X at temperatures above 1800°F. Some data on the stress rupture strength of unexposed Hastelloy X are available from commercial sources but these data generally cover only sheet materials of inapplicable thicknesses and information is not available on the characteristics of tubing or on the effects of long duration aging or of irradiation on any of the forms.

A detailed scope of activity for this program was developed during the quarter; it was anticipated that the experimental program would begin in January 1965.

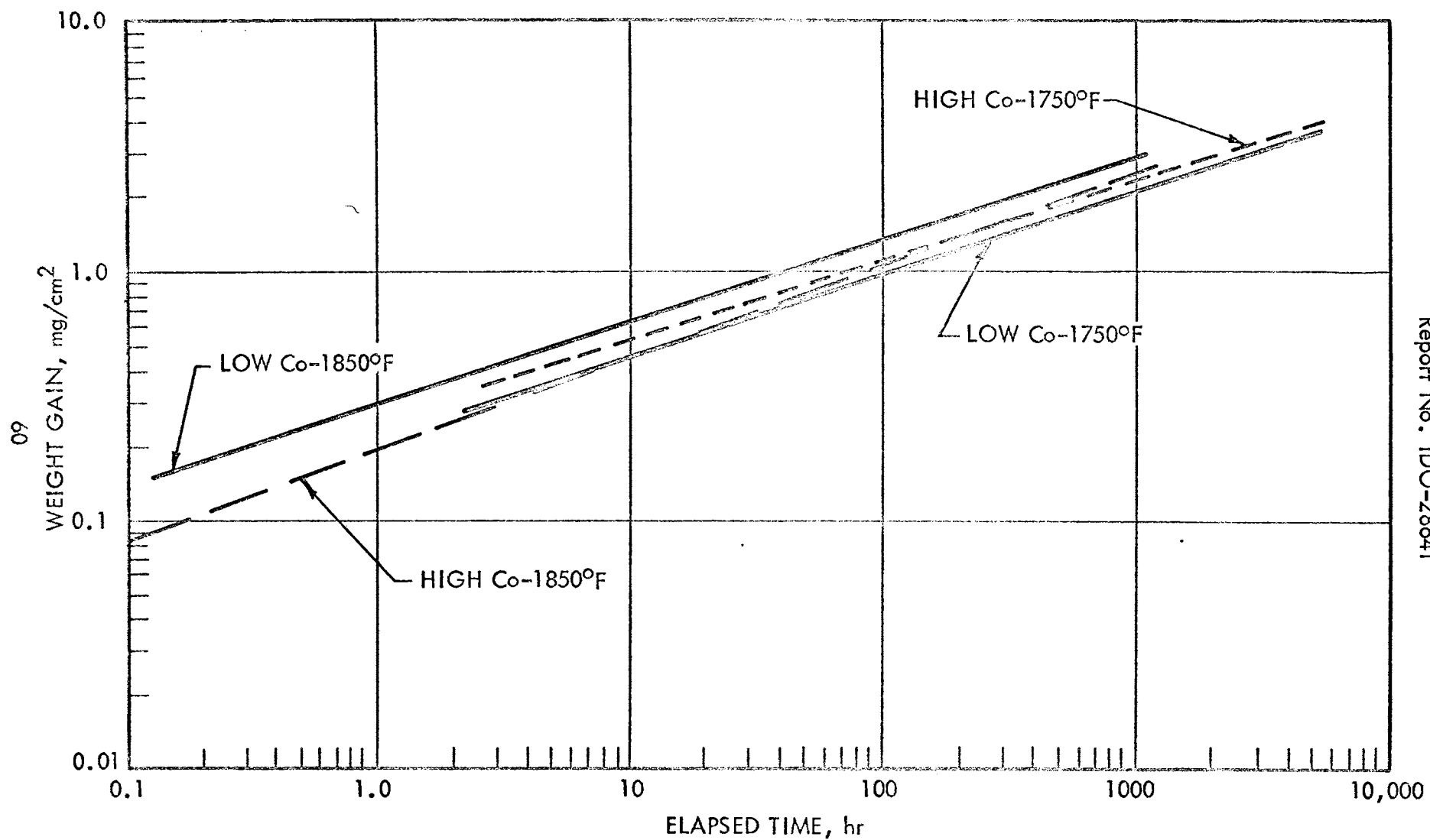


FIGURE 4-3. WEIGHT GAIN VERSUS TIME OF HASTELLOY X AT 1750°F AND 1850°F IN AIR

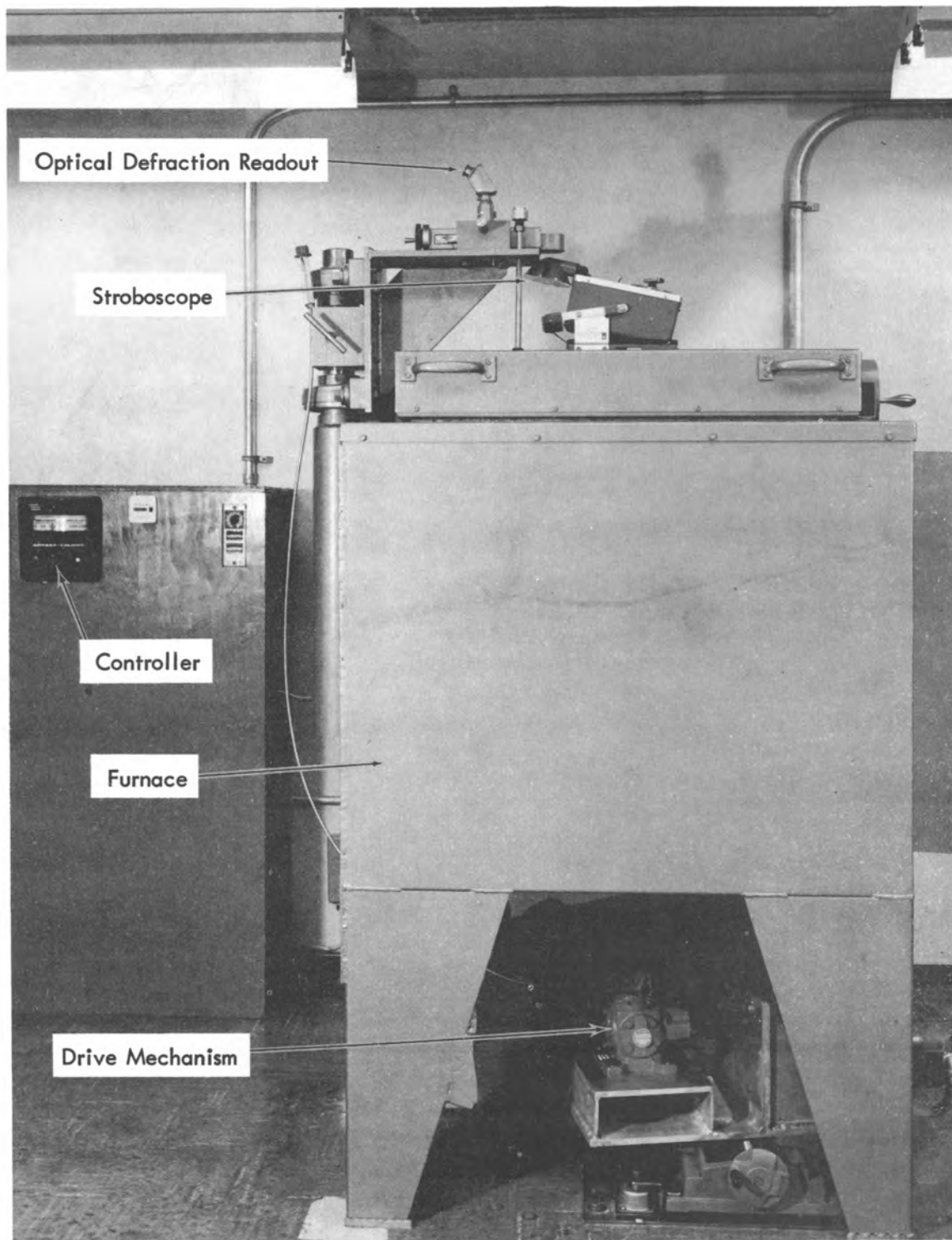


FIGURE 4-4. CENTRIFUGAL CREEP TESTING MACHINE

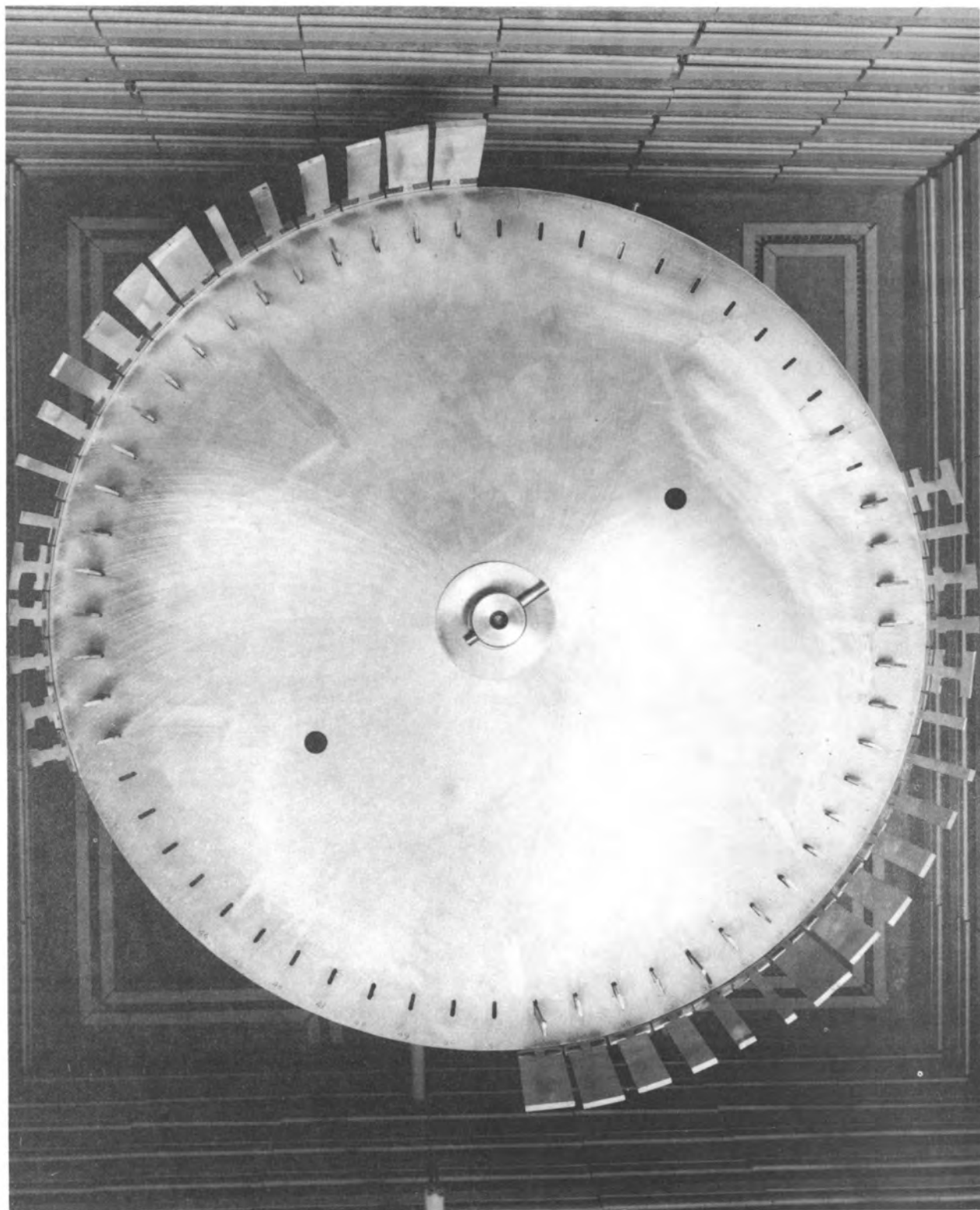


FIGURE 4-5. HEAD OF CENTRIFUGAL CREEP TESTING MACHINE WITH SPECIMENS ATTACHED

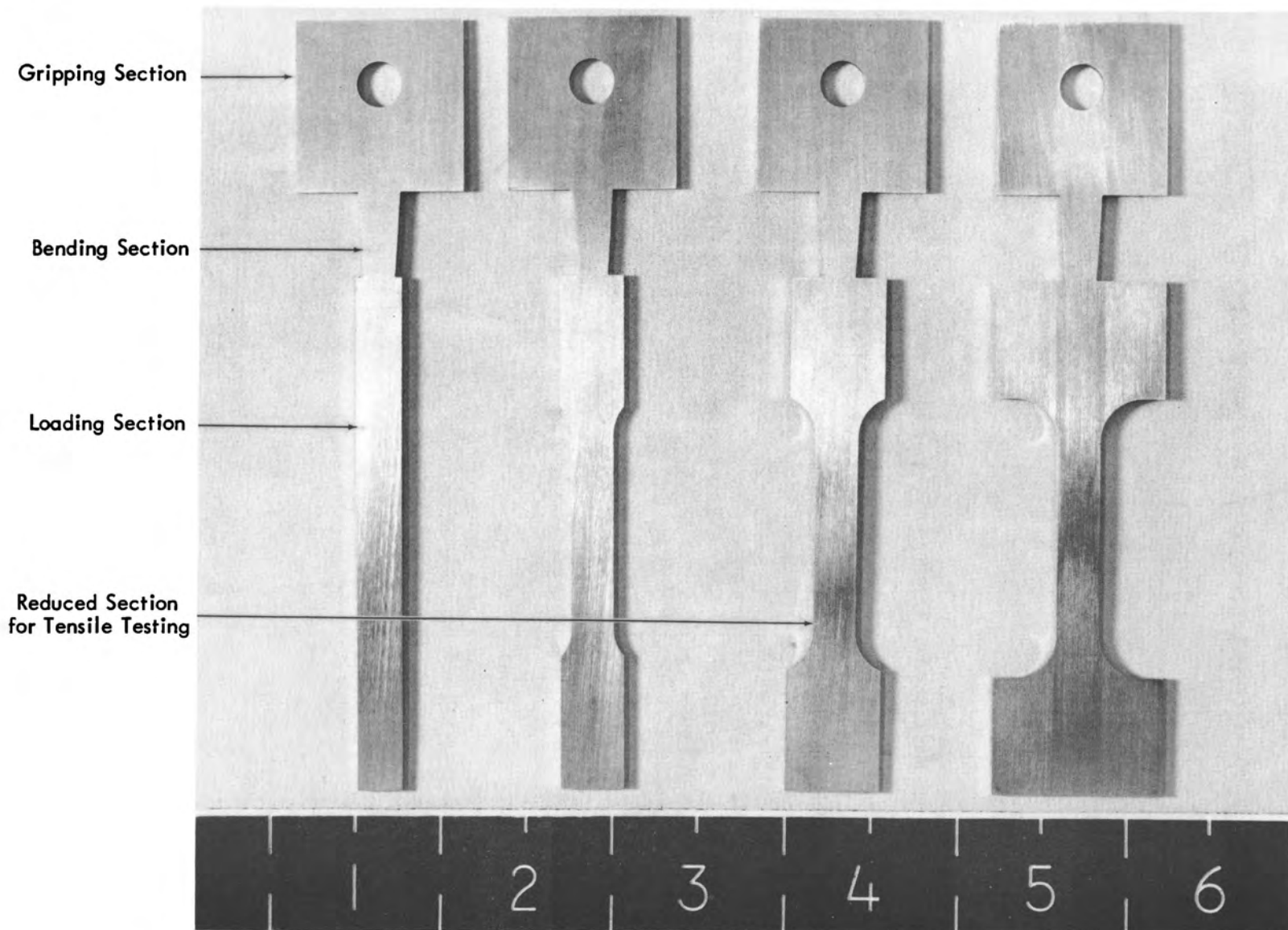


FIGURE 4-6. CENTRIFUGAL CREEP TEST SPECIMENS

### 5.0 ADVANCED PRESSURE VESSEL TECHNOLOGY

A development program was initiated to define the limits of the ML-1 pressure vessel technology and to determine methods by which this technology might be extrapolated for the design of an advanced pressure vessel assembly. The goal of this program is to design a pressure vessel that can operate at outlet gas temperatures up to 1400°F at pressures up to 500 psia in a radiation environment that is up to 25% more severe than in the ML-1 reactor.

The stress analysis code for the calandria (TSA-1) was modified to incorporate the equations of Langor and O'Donnell (Ref.d) for tube sheet stresses and pressure tube discontinuity stress calculations and to present the output parameters in the form required for analysis consistent with the techniques of Langor and O'Donnell. A photoelastic experiment program was defined to evaluate the properties of perforated plates of the type employed in the ML-1 tube sheets. The data will be compared with published design data to determine the adequacy of the procedures used in calculations for the modified design (including the TSA-1 code) and thus obtain the means to accurately predict the stresses and deflections in ML-1-type tube sheets. This program was initiated under the supervision of R. C. Sampson, who conducted the photoelastic experiments on which the design data published by Langor and O'Donnell were based.

Near the end of the quarter, the USAEC directed AGN to bring the pressure vessel technology program to an orderly conclusion. As a consequence, the TSA-1 code modification work and photoelastic experimental program will be completed, but all other activity under this program will be deferred. It is anticipated that a summary report of the activity conducted under the program will be published about 1 May 1965.

#### IV. ML-1A PROGRAM

##### 6.0 ML-1A PRELIMINARY DESIGN

During discussion of the AGCRS Program at USAEC Headquarters, Germantown, on 23 and 24 November 1964, Government spokesmen made the following points:

- 1) The reference ML-1A exclusion area (500 ft in the control cab direction and 1100 ft in the opposite direction) reduced dose levels to the 5 mrem/hr specified in the draft QMR
- 2) This large exclusion area significantly reduced the attractiveness of the ML-1A as a mobile electrical power plant for the Army
- 3) The 5 mrem/hr dose rate at the control cab could result in the exposure of the operating crew exceeding the established dose limits
- 4) A 4 mrem/hr dose rate (maximum) at the control cab was highly desirable

As a consequence, Aerojet was directed to evaluate various concepts of operational shields for means of effecting a reduction in the dose rate during operation of the ML-1A power plant. In addition, Aerojet was requested to review and modify the shutdown shielding design so that the dose rate would be reduced during power plant transportation. The following criteria were established for the shielding optimization study:

- The dose rate during reactor operation at 250 ft (in any direction) from the reactor shall not exceed 4 mrem/hr
- The dose rate during reactor operation at 100 ft (in any direction) from the reactor shall not exceed 4 mrem/hr when field expedient shielding is employed
- The dose rate 24 hr after reactor shutdown at 25 ft from the reactor in the direction of the transport vehicle cab shall not exceed 12 mrem/hr (maximum of 10 mrem/hr is desirable)

The shielding design optimization study was initiated. The following approaches are being investigated:

- Modification of the basic ML-1 internal shield design
- Provision of detachable shield components to be transported as part of the basic power plant
- The use of various arrangements of field expedient shielding

Maximum use is being made of the shielding analysis and analytical techniques developed during the ML-1 and ML-1A shield design, and of the experimental data generated during ML-1 testing at NRTS. The results of the study will be:

- A series of conceptual designs which satisfy the criteria
- The analyses supporting these designs
- Evaluation of the tradeoffs between, and the desirability of, the various design approaches
- The proposed experimental programs to verify the analyses of the shield concepts selected

## V. GCRE FACILITY

### 7.0 GCRE FACILITY MODIFICATION

Title II Architect-Engineer services were provided throughout the quarter in support of the construction at the GCRE facility site. Nine specification changes, five design changes and eleven requests for Architect-Engineer Approvals were processed during the report period. The facility modification is currently scheduled for completion early in March 1965.

In December 1964, the USAEC-ID assigned to Aerojet the responsibility for the completion of the design, fabrication and testing of the expansion joint assemblies to be incorporated in the gas ducts. The design of the existing assemblies was reviewed with Pathway Bellows Inc., and a proposal was requested from this organization for the redesign, fabrication and testing of the assemblies. Receipt of this proposal was anticipated early in January 1965.



GENERAL REFERENCES\*

- a. Quarterly Progress Report, 1 July Through 30 September 1964, IDO-28637, 13 November 1964
- b. Preliminary Design Report for the ML-1A Nuclear Power Plant, AGN TM-408, June 1964
- c. In-Pile Test of Prototype ML-1 Fuel Elements, IDO-28616, June 1964
- d. Design of Perforated Plates, O'Donnell, W. J. and Langor, B. F., ASME Paper No. 61-WA-115

IN-CONTRACT REFERENCES\*\*

1. Failure of ML-1 Rod Control Switch, AN-AGCR-795, transmitted with sr-942, 30 December 1964
2. ML-1 Reactor Aluminum Corrosion Study, AN-AGCR-752, transmitted with sr-845, 11 September 1964
3. ML-1 Control Rod Clutch Modification, AN-AGCR-775, transmitted with sr-919, 25 November 1964
4. ML-1 Reactor 1750°F and 1800°F Hot Spot Temperature Operating Envelopes for the Present Orificed Pattern, AN-AGCR-788, transmitted with sr-940, 21 December 1964
5. Calculation of Angular Neutron Fluxes Within ML-1 Cells, AN-AGCR-774, transmitted with sr-910, 10 November 1964
6. Fuel Pin Laboratory Capsules Proposed Gas Analysis at NRTS, AN-AGCR-790, transmitted with sr-952, 13 January 1965
7. ML-1 Endurance Test and Manual Speed Control Evaluation, AN-AGCR-776, transmitted with sr-930, 9 December 1964
8. ML-1 Sustained Full Power Test and Preliminary Afterheat Evaluation Experiment, AN-AGCR-780, transmitted with sr-930, 9 December 1964

\*All published by Aerojet-General Nucleonics, San Ramon, California, under Contract AT(10-1)-880 (AGCRSP).

\*\*All published by Aerojet-General Nucleonics, San Ramon, California, and distributed only within the AGCRSP.

9. Shutdown Shielding Measurements Following ANSOP 16656 Full Power Run, AN-AGCR-798, transmitted with sr-952, 13 January 1965
10. Shielding Measurements for ANSOP 16635 and 16656, AN-AGCR-778, transmitted with sr-930, 9 December 1964
11. ML-1 Shielding Experiment-Variable Shield Water Level, AN-AGCR-744, transmitted with sr-910, 10 November 1964
12. Radial Shield Tank Extension Experiment, AN-AGCR-783, transmitted with sr-952, 13 January 1965
13. ML-1 Shielding Experiment 5% Shield Water Solution, AN-AGCR-800, transmitted with sr-952, 13 January 1965
14. Shutdown Shielding Measurements During ANSOP 16295, AN-AGCR-789, transmitted with sr-952, 13 January 1965
15. ANSOP 16650 Preliminary T-C Set Performance Report, AN-AGCR-810, transmitted by sr-959, 25 January 1965
16. CSN-2 Compressor Analysis Report, Dr. M. H. Vavra, VA-29, transmitted with sr-941, 23 December 1964
17. CSN-Type Bearing Development Test Plan, Phase I, AN-AGCR-749, transmitted with sr-840, 4 September 1964
18. An Evaluation of the State-of-the-Art of High-Speed Seals, AN-AGCR-759, transmitted with sr-945, 31 December 1964
19. Addendum A, Preliminary Design Report, Model 8453-A, Precooler, Southwind Division, Stewart-Warner Corporation, 11 November 1964
20. Addendum A, Preliminary Design Report, ML-1 Improved Precooler, AN-AGCR-782, transmitted with sr-925, 3 December 1964
21. Design Report, Model 8453A Precooler, Revision A, Southwind Division, Stewart-Warner Corporation, transmitted with sr-946, 31 December 1964
22. Conceptual Design - Gas Dynamics Experimental Facility, AN-AGCR-791, transmitted 22 December 1964
23. Improved Overspeed Scram Chassis, AN-AGCR-768, transmitted with sr-866, 30 September 1964
24. Improved Overspeed Scram Chassis, AN-AGCR-794, transmitted with sr-948, 4 January 1965
25. Improved ML-1 Speed and Temperature Control System, AN-AGCR-802, transmitted with sr-949, 5 January 1965

APPENDIX A

AGCRSP BACKGROUND INFORMATION

This background of the Army Gas-Cooled Reactor Systems Program includes a short history of the Program, a description of the ML-1 power plant, and a selected bibliography.

A. HISTORY

The purpose of the Army Gas-Cooled Reactor Systems Program (AGCRSP) is to develop a mobile nuclear power plant for military field use. The current primary goal of the Program is the fabrication and test operation of a demonstration model of such a plant.

In 1955, at the request of the USAEC Division of Reactor Development, the Oak Ridge School of Reactor Technology performed a study which established the feasibility of the concept of a mobile, closed-cycle, gas-cooled nuclear power plant. Following this work, the Corps of Engineers Nuclear Power Field Office authorized the Sanderson and Porter Company to evaluate power conversion equipment and to prepare a conceptual design for the projected plant. At the conclusion of the Sanderson and Porter work, responsibility for development of the power conversion equipment for the plant was assigned to the Corps of Engineers and the development of the reactor was assigned to the USAEC.

As a result of the above arrangement, parallel programs were undertaken as follows:

- 1) The Corps of Engineers directed the Stratos Division of Fairchild Engine and Aircraft Corporation to develop a turbine-compressor set suitable for use in the projected plant. The construction of a test facility (GTTF) to evaluate the power conversion equipment was assigned to Aerojet-General Corporation. (The design of the facility was completed by Sanderson-Porter.)
- 2) Under the direction of the USAEC, Aerojet studied the feasibility of several concepts for the reactor to be incorporated in the power plant. The water-moderated concept was selected as the basis for development because of the modest extrapolation of technology required. Aerojet was

awarded the contract for the design and testing of a reactor based on this concept. At the same time, Aerojet was assigned responsibility for the design and construction of a test facility (GCRE) at NRTS.

Aerojet was designated as systems contractor and performance specifications for the demonstration power plant were evolved in June 1959. Fabrication of the reactor skid was completed in April of 1961, fabrication of the power conversion skid was completed in June 1962, and the power plant first operated as a unit (ML-1) in September 1962. Following a modification and checkout, the ML-1 power plant operated successfully for 101 hours in February and March 1963.

#### B. THE ML-1

The ML-1 is a closed cycle, gas-cooled nuclear power plant developed under the AGCRSP to demonstrate the feasibility of such a plant for military field use. During the design and construction, every reasonable effort was made to incorporate features into the plant which would be directly usable in the design of a field unit. However, since evaluation of the performance of the plant was a major requirement for the ML-1, a significant amount of additional instrumentation was provided. The physical arrangement of the equipment is such that the "prototype" components are easily identified as the:

- 1) Reactor Skid - a 15 ton unit containing the nuclear reactor and associated shielding and controls;
- 2) Power Conversion Skid - a 15 ton unit containing the power conversion equipment; and,
- 3) Control Cab - a 2-1/2 ton unit containing all the instruments and controls for operation of the field plant.

The ML-1 reactor consists of a calandria-type pressure vessel with appropriate inlet and exit gas ducts and plenums. Sixty-one pin-type BeO-UO<sub>2</sub> fuel elements are located in the tubes of the calandria. The demineralized water moderator surrounds the calandria and the six semaphore-type control rods operate in the moderator spaces between the calandria tubes. The entire reactor structure is supported inside a nine-foot diameter tank which contains heavy metal shields to permit relocation of the reactor within 24 hours after shutdown from extended operation, and a drainable (borated water) shield to attenuate radiation during reactor operation.

The plant working fluid (99.5 vol% nitrogen, 0.5 vol% oxygen) enters the reactor at 800°F and approximately 300 psia. The gas is heated to 1200°F in a single pass over the hot surfaces of the fuel elements. The moderator water is maintained at a temperature of 180°F; energy deposited in the moderator is removed in an water-to-air heat exchanger mounted on top of the power conversion skid. Provision is made for circulation, filtration and demineralization of the moderator water and for circulation and cooling (by exchange with the moderator water) of the shield water.

The hot gas leaving the reactor is expanded in a gas turbine which drives the compressor and alternator. The gas leaving the turbine passes through a regenerative heat exchanger (recuperator), through the system heat sink (air-to-air precooler) and to the compressor suction. The compressor discharges through the recuperator to the reactor inlet, thus completing the closed (modified Brayton) cycle. A lubrication system (including provision for recovery and removal of lubricating oil from the working fluid which leaks past the turbine compressor seals), the electrical switch gear, and miscellaneous power conversion hardware are mounted on the power conversion skid. An a-c, two-speed motor is coupled to the turbine compressor shaft to provide starting power to the set.

The following auxiliary systems are provided for the ML-1:

- 1) A deoxygenation system which removes dissolved oxygen from a bypass stream to maintain the moderator system oxygen content below 0.7 ppm.
- 2) An emergency cooling system which automatically injects a supply of coolant gas into the reactor in the event of a complete stoppage of working fluid flow.
- 3) A working fluid makeup system to compensate for normal leakage and to provide for initial charging of the system.
- 4) Waste gas storage facilities to accommodate the charge of radioactive gas in the loop in the event of an emergency.

### C. BACKGROUND BIBLIOGRAPHY

The following bibliography provides information on sources of additional background to the material presented in this report. These documents trace the technical evolution of the AGCRSP from inception but do not, in general, document programmatic decisions. Such activity may be inferred from the technical approaches pursued and from the general background information presented in the reports.

The following reports were published by Aerojet-General Nucleonics, San Ramon, California under the Army Gas-Cooled Reactor Systems Program.

<u>DOCUMENT NO.</u>	<u>TITLE</u>	<u>CLASSIFICATION</u>
IDO-28505	<u>GCRE Semiannual Report, 1 November 1956 Through 30 June 1957, 20 February 1958</u>	CRD
IDO-28506	<u>GCRE-I Hazards Summary Report, December 1958, with three addenda, March 1959, February 1960, May 1960</u>	U
IDO-28519	<u>GCRE Semiannual Progress Report, 1 July Through 31 December 1957, 26 June 1958</u>	CRD
IDO-28526	<u>GCRE Semiannual Progress Report, 1 January Through 30 June 1958, 17 October 1958</u>	CRD

DOCUMENT NO.	TITLE	CLASSIFICATION
IDO-28533	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1958, 28 February 1959</u>	CRD
IDO-28537	<u>Preliminary Hazards Summary Report for the ML-1 Nuclear Power Plant, 22 April 1959</u>	U
IDO-28542	<u>AGCRSP Semiannual Progress Report, 1 January Through 30 June 1958, July 1958</u>	U
IDO-28549	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1959, 22 December 1959</u>	U
IDO-28550	<u>The ML-1 Design Report, 16 May 1960</u>	U
IDO-28555	<u>ML-1 Transportability Studies, 23 March 1960</u>	U
IDO-28558	<u>AGCRSP Semiannual Progress Report, 1 January Through 30 June 1960, 11 July 1960</u>	U
IDO-28560	<u>Final Hazards Summary Report for the ML-1 Nuclear Power Plant, with four supplements, 5 August 1960</u>	U
IDO-28567	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1960, 17 December 1960</u>	U
IDO-28573	<u>AGCRSP Semiannual Progress Report, 1 January Through 30 June 1961, 10 August 1961</u>	U
IDO-28581	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1961, 31 January 1962</u>	U
IDO-28590	<u>AGCRSP Semiannual Progress Report, 1 January Through 30 June 1962, 24 August 1962</u>	U
IDO-28597	<u>AGCRSP, Study of the GCRE Tube Bundle Failure, 14 December 1962</u>	U
IDO-28602	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1962, 22 February 1963</u>	U
IDO-28607	<u>AGCRSP Quarterly Progress Report, 1 January Through 31 March 1963, 15 May 1963</u>	U
IDO-28612	<u>AGCRSP Quarterly Progress Report, 1 April Through 30 June 1963, 15 August 1963</u>	U
IDO-28617	<u>AGCRSP Quarterly Progress Report, 1 July Through 30 September 1963, 15 November 1963</u>	U

<u>DOCUMENT NO.</u>	<u>TITLE</u>	<u>CLASSIFICATION</u>
IDO-28621	<u>AGCRSP Quarterly Progress Report, 1 October Through 31 December 1963, 27 January 1964</u>	U
IDO-28626	<u>AGCRSP Quarterly Progress Report, 1 January Through 31 March 1964, 15 May 1964</u>	U
IDO-28632	<u>AGCRSP Quarterly Progress Report, 1 April Through 30 June 1964, 15 August 1964</u>	U
IDO-28637	<u>AGCRSP Quarterly Progress Report, 1 July Through 30 September 1964, 13 November 1964</u>	U



APPENDIX BML-1 PLANT CHARACTERISTICS

Note: Items marked with a single asterisk (\*) indicate changes made since 31 March 1964. Items marked with a double asterisk (\*\*) indicate entries added since 31 March 1964.

1. GENERAL

Design performance at 100°F

Gross electrical output	420 kw*
Net electrical output	350 kw*
Reactor thermal power	2.98 Mw to gas; 3.41* Mw total
Cycle efficiency $\left(\frac{\text{Thermal output}}{\text{Power to gas}}\right)$	17.2%*
Plant thermal efficiency $\left(\frac{\text{Gross elect. pwr}}{\text{Total reactor pwr}}\right)$	13.2%*
Net plant efficiency $\left(\frac{\text{Net elect. output}}{\text{Total reactor pwr}}\right)$	10.3%**
Coolant flow (compressor inlet)	92,500*
Dose rate at control cab @ 500-ft during full power operation	5 mr/hr (with expedient shielding as needed)
Dose rate at 25 ft, 24 hr after shutdown (direction of transport vehicle driver with P-C skid in place)	15 mr/hr
Overall plant dimensions	279 x 113 x 93 in. high
Overall plant weight and dimensions	Weight      Dimensions (in.)
Reactor package	30,000 lb    111 x 110 x 93 high (plus ion exchange column on end)
Power-conversion package	30,000 lb    168 x 113 x 93 high
Control cab	6500 lb    145 x 82 x 81 high
Auxiliary equipment	15,000 lb    - - - - -
Operating supplies (startup and 90 day operation):	
Demineralized water	2900 gal
Nitrogen (with 0.5 vol% oxygen)	2400 scf
Oxygen	200 scf

Anhydrous boric acid ( $B_2O_3$ )	1200 lb
Mixed bed ion exchange resin	900 lb max.
Lubricating oil	60 gal
Filter elements	7
Plant startup time	12 hr
Auxiliary power requirements	.
Pre-startup	30 kw max.
Normal startup	75 kw max.
Normal shutdown	45 kw max., 3 kw ave
Emergency shutdown	none
Reactor drying	36 kw max.

## 2. REACTOR THERMAL CHARACTERISTICS

Power density	700 kw/ft <sup>2</sup>
Maximum heat flux	140,000 Btu/hr/ft <sup>2</sup>
Average heat flux	78,200 Btu/hr/ft <sup>2</sup>
Heat transfer surface	126.5 ft <sup>2</sup>
Maximum to average heat flux ratio	
Axial	1.41
Radial	1.27
Maximum fuel center temperature (including hot spot factors)	2160°F (BeO-UO <sub>2</sub> ) 2650°F (UO <sub>2</sub> )
Maximum moderator temperature	190°F
Maximum surface temperature of fuel cladding (nominal, average)	1500°F
Maximum surface temperature of fuel cladding (including hot spot factors), reference	1650°F

## 3. REACTOR NUCLEAR CHARACTERISTICS

Average thermal neutron flux (fuel)	$1.9 \times 10^{12}$ neut/cm <sup>2</sup> -sec
Average fast neutron flux (fuel)	$1.7 \times 10^{13}$ neut/cm <sup>2</sup> -sec
Maximum to average thermal flux ratio	3.9
Hydrogen to U-235 atom ratio	40
Core buckling	0.0059 cm <sup>-2</sup>
Fermi age	60 cm <sup>2</sup>
Square of thermal diffusion length, L <sup>2</sup>	2.05 cm <sup>2</sup>
Thermal utilization, f	0.75

Infinite multiplication factor,  $k$ 

Without shims	1.54
With shims	1.47
Neutron lifetime	$1.9 \times 10^{-5}$ sec
$k_{eff}$ , cold, clean core; no shims or burnable poison	1.067
Operating $k_{eff}$ , cold, clean core, with shims and burnable poison	1.018
Core life, full power	3000 hr min; 10,000 hr design
Burnup (U-235), average	3.6% in 10,000 hr
Maximum	6.5%
Prompt temperature coefficient, $\Delta k/k-^{\circ}C$	
at $0^{\circ}C$	$+0.3 \times 10^{-6}$
at $90^{\circ}C$	$-0.5 \times 10^{-6}$

4. REACTOR VESSEL

## Materials

Tube sheet	Stainless Steel, Type 304, 2.94 in. thick
Pressure tubes	Stainless Steel, Type 321
Source tube	Stainless Steel, Type 321
Gas ducts, plenums	Stainless Steels, Types 304-L, 321 and 347
Baffle	Stainless Steel, Type 321; Tungsten; and Inconel X (springs)
Outside diameter	30.960 in. max. (exclusive of upper flanged connection)
Overall height	79.5 in.
Pressure tube length	24 in. between inside surfaces of tube sheets
Design pressure	345 psia (gas)
Design temperature	$525^{\circ}F$ (max.)
Wall thicknesses	Tubes 0.020 in.; plenum 2.12 in. min
Source tube	0.020 in. wall thickness; 6.500 in. OD

5. REFLECTOR

Composition, top	2 in. $H_2O$ ; 4.5 - 5.0 in. stainless steel; 1.5 in. W
bottom	3-4 in. stainless steel; 3 in. W

radial	1.8 in. Pb; 2 in. W; 180° segment 4 in. Pb; 180° segment
Total heat generation	$6 \times 10^5$ Btu/hr
Maximum power density	360 Btu/hr-in. <sup>3</sup>

6. BIOLOGICAL SHIELDING

Composition	3-1/2 to 4 in. lead and tungsten plus 30 in. of borated water (2 wt% boric acid)*
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7. CORE (EXCLUDING REFLECTOR)

Diameter	22 in. equivalent
Height	22 in.
Number of fuel elements	61
Number of coolant passages	61
Number of coolant passes	1
Type of geometry of fuel elements	Cluster of 19 pins (18 fueled)
Cold, clean critical mass, U-235 no shims, no burnable poison	37 kg
U-235 loading	49 kg
Enrichment, inner 6 pins	93% U-235 as UO <sub>2</sub>
outer 12 pins	31 vol% UO <sub>2</sub> , 93% enriched U-235 69 vol% BeO

## Core composition

Materials	<u>Volume %</u>
UO <sub>2</sub>	4.3
BeO	3.3
Stainless steel	3.6
Hastelloy X	7.0
H <sub>2</sub> O	58.6
Insulation	7.0
Gas void	<u>16.2</u>
Total	100.0

8. FUEL ELEMENT

Dimensions	1.72 in. OD x 32 in.
Fuel material	BeO-UO <sub>2</sub> (outer pins); UO <sub>2</sub> (inner pins)

Number of pins per element	19 (12 wt% BeO-UO <sub>2</sub> ; 6 wt% UO <sub>2</sub> ; 1 empty)
Pin outside diameter	0.241 in.
Pin cladding material	Hastelloy X
Pin cladding wall thickness	0.030 in.
Pin spacer	0.040 in. OD Hastelloy wire
Heat transfer material (pin internal)	He
Pellet diameter	0.176 in. (nominal)
Type burnable poison	Cadmium
Reactivity worth of burnable poison	0.6% at startup

#### 9. CONTROL ELEMENTS

Type	Tapered blades
Location	Moderator
Number: Shim blades	3 pairs (3 actuators)
Safety blades	2 pairs (2 actuators)
Regulating blades	1 pair (1 actuator)
Absorber material: Safety and shim blades	5 wt% Cadmium-
	15 wt% Indium-
	80 wt% Silver
Dimensions (each blade)	4 x 10.5 x 0.25 to 0.62 in.
Regulating blades	Stainless steel
Dimensions (each blade)	4 x 9 x 0.25 to 0.62 in.
Cladding material	none
Reactivity worth of control elements:	
Safety and shim blades	0.058 $\Delta k/k$
Regulating blades	<u>0.004</u> $\Delta k/k$
Total	0.062 $\Delta k/k$
Actuating time for regulating blade:	
Drive	13.3 sec for full insertion or withdrawal
Scram	0.35 sec (max.) for full insertion from signal
Safety and shim actuator:	
Drive	4.0 min for full insertion or withdrawal
Scram	0.35 sec (max.) for full insertion from scram signal

10. MODERATOR

Type	Water
Reactor inlet temperature	180°F
Reactor outlet temperature	190°F
Pressure	30 psi max.
Flow rate	300 gpm
Type of flow circulation	Forced
Purity:	
Total solids	1 ppm
Resistivity	$10^5$ to $10^6$ $\Omega$ -cm
Total heat removal rate	$1.5 \times 10^6$ Btu/hr

11. REACTOR WORKING FLUID FLOW

Working fluid	99.5 vol% N <sub>2</sub> + 0.5 vol% O <sub>2</sub>
Reactor inlet temperature	800°F nominal
Reactor mixed-mean outlet temperature	1200°F max.
Average velocity in core	160 ft/sec
Maximum velocity	180 ft/sec
Inlet pressure	315 psia (max.)
Core $\Delta P$	15 psi
Reactor $\Delta P$	22 psi

12. POWER CYCLE

Type	Brayton cycle with regeneration
Total volume of working fluid system	120 ft <sup>3</sup>
Total system working fluid inventory full load at 100°F	52 lb
Working fluid transit time	2.0 sec
Cycle characteristics (100°F ambient temp)*	
Net power, kw	350*
Reactor inlet, °F	781*
Turbine inlet, °F	1193*
Compressor inlet, °F	133*
Compressor inlet, psia	116*
Compressor outlet, psia	321*
Reactor inlet, psia	314*

13. TURBINE-COMPRESSOR SET

	<u>Stratos T-C Set</u>	<u>Clark T-C Set</u>
Speed, rpm	18,338	22,000
Turbine stages	2	2
Turbine rotor material	Incoloy 901	A-286 (first stage)* AISI 422 (second stage)*
Turbine blade material	Inco 713	M-252*
Turbine stator blade material	Inconel	N 155 or 19-9 DL
Expansion ratio	2.38	2.42*
Compressor stages	2	11
Compressor material	AL 355 T71	403 stainless steel
Rotor shaft	SAE 4340	SAE 4340
Compressor ratio	2.72	2.765*
Case material	304 stainless steel	304 stainless steel
Seals		
at journals	Buffered labyrinth	Buffered labyrinth
interstage	Plain labyrinth	Plain labyrinth
shaft	Buffered labyrinth	Double "L" ring seal oil buffered
Bearings		
journal	Tilting pad	Plain babbitt
thrust	Kingsbury type	Kingsbury type (in low pressure area)
Support	Overhung turbine	Turbine and compressor supported between bearings

14. ALTERNATOR

Output	
Rating	750* KVA 3 Ø, 60 cycle
Voltage	2400/4160 V
Rotor shaft speed	1800*
Diameter, maximum	36 in.*
Length	54 in.*
Weight	5000 lb**

15. RECUPERATOR

Length (including insulation)	81 in.
Outside diameter (including insulation)	49.25 in.
Headers	
High pressure inlet	8 in.
High pressure outlet	8 in.
Low pressure inlet	20 in.
Low pressure outlet	14 in.
Effectiveness	78.4*
Pressure loss	
High pressure $\Delta P/P$	2.1%*
Low pressure $\Delta P/P$	1.25%*
Type	Shell and tube regenerator
Tubes	4 passes x 840 tubes
Shell	1 pass
Surface	External fins
Materials	300 series stainless steel

16. PRECOOLER, MODERATOR COOLER AND OIL COOLER ASSEMBLY

Dimensions:

Length, overall	166 15/16 in.
Precooler	122 5/16 in.
Moderator cooler	32 1/8 in.
Oil cooler	11 5/16 in.
Width	113 in.
Thickness, overall	32 in.
Core	15 in.
Fans and plenums	17 in.
Materials	
Tubes and fins	Series 1100 aluminum
Headers	Series 2219 aluminum
Weight	6500 lb

Precooler:

Header, inlet	One, 14 in.
Header, outlet	One, 10 in.
Effectiveness	92.2%*
Total $\Delta P/P$	1.69%*
Air flow	247,500 lb/hr
Type	Fin fan air-to-gas exchanger
Tubes	1105 tubes, single pass
Surface	Internal and external fins

Moderator cooler:

Headers, inlet and outlet	4 in.
Total $\Delta P$	2.77 psi
Water temperature	
In	190°F
Out	180°F
Air flow	73,250 lb/hr
Type	Fin fan air-to-water exchanger
Tubes	88 tubes per pass, three passes
Surface	External fins

Oil cooler:

Connections, inlet and outlet	1 1/2 in.
Total $\Delta P$	9.38 psi
Oil temperature	
In	180°F
Out	150°F
Oil flow	18,900 lb/hr
Air flow	27,500 lb/hr
Type	Fin fan air-to-oil exchanger
Tubes	45 tubes, 2 passes
Surface	Internal and external fins



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**AEROJET-GENERAL NUCLEONICS**