32
5-11

Report No. IDO-28653
Contract AT(10-1)-880

MASTER

AGN

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

Progress Report
1 July through 31 October 1965

January 1966

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

AEROJET-GENERAL NUCLEONICS
SAN RAMON, CALIFORNIA

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

PAGES i to ii
WERE INTENTIONALLY
LEFT BLANK

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

Report No. IDO-28653
TID-4500 (47th Edition)
UC-80, "Reactor Technology"

CFSTI PRICES

H.C. \$ 7.00; MN .75

AGN

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRESS REPORT
1 JULY THROUGH 31 OCTOBER 1965

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Published

January 1966

Approved by: *R. H. Chesworth*
R. H. Chesworth
Supervising Representative
Contract AT(10-1)-880

AEROJET-GENERAL NUCLEONICS

A DIVISION OF AEROJET-GENERAL CORPORATION

Printed in U.S.A. Price \$3.00.
Available from the Clearinghouse
for Federal Scientific and
Technical Information, National
Bureau of Standards, U.S. Department
of Commerce, Springfield, Va.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRESS REPORT
1 JULY THROUGH 31 OCTOBER 1965

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 for the development of a mobile, low-power nuclear power plant for Military field operation.

Testing of the ML-1 reactor in the GCRF was initiated and 214.6 Mwh of operation were accumulated. The reactor was shut down early in August to investigate a leak in the calandria. The total exposure as of this date was 7674.8 Mwh.

Analyses were completed to determine the transient temperatures in the ML-1 reactor following a scram of the ML-1/GCRF system. This work indicated that the transient temperatures following a normal scram were the most severe of all the cases analyzed but were probably acceptable for limited reactor testing.

The fabrication of 20 replacement fuel elements for the ML-1-I core loading was completed.

The performance evaluation testing of the Stewart-Warner precooler was essentially complete. An analysis of the test data indicates that the pressure drop is higher than predicted and that the effectiveness is higher than the design value, although the completion of the reduction of data concerning the air-side cooling flow test is required before a final determination of the effectiveness can be made. The effect of the installation of power conversion skid components under the precooler was determined; a significant change in precooler performance was observed when the Griscom-Russell recuperator was installed.

System performance analysis of the ML-1 system including the CSN-1A(M) t-c set, the Stewart-Warner precooler and the improved recuperator indicated that the net electrical output of the plant at 100°F ambient temperature would be 319.7 kw.

The ML-1/GCRF scram response time analysis was completed; it was concluded that the time response characteristics of the ML-1/GCRF scram circuits were adequate for protection of the reactor.

The CSN-1A redesign was essentially completed by G.E. Testing of the alternator was completed by Fairbanks-Morse, the unit was delivered to San Ramon and preparations for additional testing were completed at the end of the report period. Testing of the improved lubrication system components was complete and the system was assembled and ready for system tests at the end of the report period.

The Hastelloy X corrosion, creep and rupture strength investigations being performed under the ML-1 Technology Program were suspended at the end of September pending resolution of the FY 66 program.

All AGCRSP activity under Contract AT(10-1)-880 was terminated on 31 October 1965. The termination phase activities were initiated on 1 November with the scheduled completion date of 31 March 1966. This report is the last in a series of periodic technical progress reports covering AGCRSP activities.

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRESS REPORT

1 JULY THROUGH 31 OCTOBER 1965

CONTENTS

	<u>Page</u>
<u>ABSTRACT</u>	v
I. <u>PROGRESS TO 30 JUNE 1965 - SUMMARY</u>	1
II. <u>ML-1 PROJECT</u>	7
1.0 ML-1 TEST OPERATIONS	7
2.0 ML-1 OPERATIONS ENGINEERING SUPPORT	11
3.0 ML-1 DEVELOPMENT AND IMPROVEMENT	30
III. <u>ML-1 TECHNOLOGY PROGRAM</u>	39
4.0 FUEL ELEMENT TECHNOLOGY	39
IV. <u>FACILITY ENGINEERING</u>	57
5.0 GDF DESIGN AND CONSTRUCTION LIAISON	57
<u>REFERENCES</u>	59
<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.	ML-1 Reactor Control System Scram Response Time Nuclear Channels 5, 6 and 7 (linear)	10
2.	ML-1A Coolant Loss Accident Results	15
3.	Worth of ML-1 Regulating Rod	16

CONTENTS (Continued)

<u>Table Number</u>	<u>Title</u>	<u>Page</u>
4.	Fuel Pin Processing Losses	18
5.	Summary of Fuel Processed	18
6.	Pressure Loss Summary	19
7.	Calculated Design Point Performance of the ML-1 System with the CSN-1A(M) T-C Set	25
8.	Calculated Limiting Parameters for Selected ML-1 Operating Points with the CSN-1A(M)	26
9.	ML-1 Influence Coefficients	27
10.	Elevated Temperature Tensile Properties of As-Received Hastelloy X Heat X14593	36
11.	Average Maximum Depth of Oxide Penetration Into Hastelloy X Sheet Exposed in 1900°F Air, Inches	39
12.	Average Depth of Internal Oxidation of Hastelloy X Sheet Exposed in 2000°F Air, Inches	39
13.	Average Microhardness of Hastelloy X Sheet Samples Exposed in 2000°F Air (KHN)	40
14.	Ratio of Major X-ray Diffraction Peak Heights for Spinel and Cr ₂ O ₃ Oxides on Hastelloy X Sheet Materials Exposed 500 Hr at 1900°F in Air	40
15.	X-ray Fluorescence Peak Intensities	41
16.	Room Temperature Tensile Properties of Hastelloy X Exposed in 1900°F Air	42
17.	Room Temperature Tensile Properties of Hastelloy X Sheet Materials Exposed in 2000°F Air	43
18.	Elevated Temperature Tensile Properties of Hastelloy X Sheet, As-received, and As-exposed in Air for 5000 Hr at 1900 or 2000°F	44
19.	Elevated Temperature Tensile Properties of Hastelloy X Sheet after High Temperature Exposure in Air	45
20.	Tensile Properties of Hastelloy X Tubing after 5000 Hour Air Exposure	45
21.	Elevated Temperature Tensile Properties of Hastelloy X Heat E9517 Exposed in Air	46
22.	Lattice Parameter of M ₆ C Precipitates in Hastelloy X Sheet Specimens Before and After Aging for 100 Hr at 1300°F	47
23.	Room Temperature Tensile Properties of Hastelloy X Aged in Inert Gas at 1300°F	50

CONTENTS (Continued)

<u>Table Number</u>	<u>Title</u>	<u>Page</u>
24.	Room Temperature Tensile Properties of Hastelloy X Aged in Inert Gas at 1850°F	50
25.	Oxidation Rate Constants, Diffusion Constants, and Activation Energy Constants for Hastelloy X	51
26.	Oxidation Penetrations of Hastelloy X Tubing After 3998 Hour Exposure in 1850°F Air	53
27.	Microhardness of As-received and Tubing Specimens Exposed to 1850°F Air for 3998 Hours	55

FIGURES

<u>Figure Number</u>	<u>Title</u>	
1.	ML-1 Reactor Control System Scram Response Time Nuclear Channel 1	8
2.	ML-1 Reactor Control System Scram Response Time Nuclear Channel 4	9
3.	ML-1 Pressure Vessel Disassembly - Inspection Program	14
4.	Stewart-Warner Precooler Nitrogen Side Pressure Drop Characteristics	20
5.	Stewart-Warner Precooler Performance	21
6.	Predicted CSN-1A(M) Compressor Performance Map	23
7.	CSN-1A(M) Turbine Map Closed Cycle, Flange to Flange Performance with Nitrogen	24
8.	Temperature Rise from Ambient to S.E. Precooler Fan Intake Plenum, ML-1	28
9.	Schematic, ML-1/GCRF System Analog Simulation	29
10.	CSN-1A Intermediate Case Design	32
11.	Precooler Test Facility	34
12.	Rotating Machinery-Alternator Test	35
13.	Hastelloy X, Heat X-4381 After 1000 Hour Exposure to 1900 Air Followed by a 1000 Hour 1300°F Argon Exposure	48
14.	Hastelloy X, Heat X-4381 After 1009 Hour Air Exposure at 2000°F Followed by a 1000 Hour 1300°F Argon Exposure	49
15.	Comparison of Weight Gain of High and Low Cobalt Hastelloy X at the Indicated Temperatures	52

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRESS REPORT*

1 JULY THROUGH 31 OCTOBER 1965

I. PROGRESS TO 30 JUNE 1965 - 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 (GCRF) was completed 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 completed by Aerojet under contract with the USAEC. This test reactor was provided to investigate the operational

*Published by Aerojet-General Nucleonics, San Ramon, California. This report, the last in the series of quarterly progress reports, covers a four month period from 1 July 1965 to the date of the initiation of the AGCRS Program Termination.

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 lifetime testing of fuel elements. The heterogeneous, water-moderated, 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 GCRF was placed in standby condition. The GCRF was reactivated during late 1965 and early 1965 to permit test operation of the ML-1 reactor skid without the normally associated power conversion system (see 14 below).

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 the GTTF early in 1960. This testing was performed by Aerojet under contract with the Department of the Army, APCDB, until May 1963.

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 improve the ventilation for the power plant, and to improve the 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 2.3.

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 but the initiation of significant activity was postponed in consideration of the decision to extend the operating lifetime of the first core loading by providing a replacement partial loading (ML-1-IM). Activities under this task are discussed in Section 3.3.

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. Evaluation of test data generated in late 1964 and early 1965 led to the decision to modify the unit to improve the aerodynamic performance and a contract for this work was awarded to the General Electric Co. in April 1965. Activity relating to the CSN-1 t-c set is discussed in Section 3.2.

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 damage to the CSN-1 turbine which was repaired. Test operations were resumed in September 1964 and continued in the conduct of endurance plant performance evaluation tests until late December. A leak at the lower plenum cap was detected during preparations to relocate the reactor skid and control cab to the modified GCRF (see 14 below) for continued testing while the development of power conversion components was being completed. (Activity relating to the reactor skid testing in the current period is discussed in Sections 1.0 and 2.1 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 the conceptual design for a second generation gas-cooled mobile, nuclear power plant (ML-2) were completed 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 11 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.

14) The design of modifications to the GCRF (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 USAED-ID and was completed in April 1965.

This report is organized under four major headings: Summary of Progress to 30 June 1965, ML-1 Project, ML-1 Technology Program, and Facility Engineering. Significant areas of activity are identified by numbers 1.0 through 5.0 (second order identification) and details are presented under the appropriate second order identification. Figures and tables 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.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

II. ML-1 PROJECT

1.0 ML-1 TEST OPERATIONS

At the beginning of the report period, the final preparations for high power operation of the ML-1 reactor skid in the GCRF test loop were in progress. These preparations were completed and the reactor was operated at 2.0 Mw intermittently during late July and early August 1965. A total of 214.6 Mwh were accumulated; the total exposure on the ML-1-I core at the end of the report period was 7674.8 Mwh.

High power reactor operation began on 28 July 1965. The plant was shut down on 2 August for repairs to the emergency diesel generator. During the attempt to resume power operation on 9 August 1965, evidence of a gas leak in the pressure vessel was observed and a program to determine the location and nature of the leak was initiated.

Initial investigations indicated that the leak was in the lower part of the tube section of the calandria. Subsequent tests of the fueled reactor failed to develop data from which a definitive conclusion concerning the location and nature of the leak could be reached. As a consequence, the decision was made to defuel the reactor and proceed with the investigations of the leak and the subsequent disassembly of the pressure vessel. The fuel elements were transferred to the TAN facility, the reactor skid was removed from the reactor pit and the pit was flooded to serve as a shielded facility for the pressure vessel disassembly operation. At the end of the report period, the partial disassembly of the reactor skid required for pressure vessel removal had been completed and the installation of fixtures in the reactor pit to accept pressure vessel was also completed (see Section 2.1-g). All further work associated with reactor skid disassembly or pressure vessel inspection was suspended in accordance with instructions received from USAEC-ID on 28 October 1965.

The measurement of the response times of the ML-1/GCRF scram systems was completed during the report period. Typical results are shown in Figures 1 (for startup Channel 1) and 2 (for Log N Channel 4). Typical response time data for linear power channels are shown in Table 1. A report of this test was published.^{(1)*}

*Numbers in parenthesis identify references listed at the end of the report.

FIGURE 1. ML-1 REACTOR CONTROL SYSTEM SCRAM RESPONSE TIME
NUCLEAR CHANNEL 1 (LOG COUNT RATE)

FIGURE 2. ML-1 REACTOR CONTROL SYSTEM SCRAM RESPONSE TIME
NUCLEAR CHANNEL 4 (LOG-N)

TABLE 1 - ML-1 REACTOR CONTROL SYSTEM SCRAM RESPONSE TIME
NUCLEAR CHANNELS 5, 6 & 7 (LINEAR)

Test Signal (% Power)	Response Time (Milliseconds)		
	Channel 5	Channel 6	Channel 7
0-116 Step	0 - 2.5	1 - 11	0
100-116 Step	0 - 7.5	0 - 2	0 - 2.5

A critical experiment was conducted on 8 September with the core in a flooded condition to compare the reactivity data with that obtained during a similar experiment conducted in 1964. This test indicated an increase in core reactivity of 0.105% $\Delta K/K$; ⁽²⁾ this increase was attributed to the loss of silver from the reactivity shims as a result of corrosion and mechanical abrasion. ⁽³⁾

2.0 ML-1 OPERATIONS ENGINEERING SUPPORT

2.1 Reactor and Auxiliaries

a. Reactor Pressure Vessel Repair: The summary report, covering the repair of the failed seal weld between the lower plenum and lower plenum cap of the ML-1 pressure vessel was published.⁽⁴⁾ This report described the laboratory and theoretical analyses performed to determine the cause of failure, the development and proof testing of the new omega-type seal and the repair operations carried out in the high radiation environment.

b. Aluminum Corrosion Studies: The evaluation of Type 6061 and 3003 aluminum specimens exposed in the ML-1 reactor shield tank for 6120 hr was completed. Examination of the specimens revealed minor general corrosion and a stripping of the protective oxide film. Localized pitting attack, which resulted in pits of a maximum depth of 0.001 in., was also observed. Grain boundary preferential voiding was observed in many of the specimens which contained welds but it is not possible to determine whether this condition was the result of exposure to the shield solution or of shrinkage porosity which occurred at the time of welding.

c. Stress-Corrosion Cracking Studies: Metallurgical evaluation of specimens removed from the defective stainless steel ion chamber pressurization line^{(a)*} revealed that the cracks resulted from a stress corrosion mechanism. The cracking was attributed to the presence of chlorides which are known to have been present in the boric acid shield solution early in the operating history of the reactor.

Stress corrosion cracking specimens were fabricated for installation in the ML-1 shield tank and for parallel testing in the laboratory to determine if the ML-1 shield solution, when mixed in accordance with current specifications, would promote stress corrosion cracking. Laboratory testing was initiated during the report period but the field test specimens were not installed. No results had been obtained from the laboratory work at the time of termination of the AGCRSP.

*Letters in parenthesis identify references listed at the end of the report.

d. Maintenance of Control Rod Actuators: The program for overhaul of the ML-1 control rod actuators continued until the middle of August at which time all work was suspended pending resolution of the contract funding problems. At the time work was suspended, replacement components for all actuators had been procured and the test actuators had been rebuilt with modified clutches, radiation resistant motors, new wire harnesses and new bearings. The test unit had been cycled through 30,000 withdrawal-scrum cycles in a humid atmosphere at 180°F; examination of the assembly at the completion of this test did not reveal any deterioration or evidence of failure of new components.

Preparations for inspection of the ML-1 control rod drive shafts were completed. Two replacement drive shafts and bearing housings and a gear puller set were procured for use in this work. A test was performed at San Ramon under conditions simulating those anticipated in the reactor. During this test, a drive shaft was removed and the hole was plugged and subsequently the plug was removed and the replacement drive shaft was inserted. This test indicated that the tooling and fixtures were satisfactory and that the work could be done rapidly with a negligible loss of water.

e. Shield Solution Gas Removal System: During the preparations for reactor operation in July 1965, it was observed that a gas leak existed in the outlet gas duct flange beneath the reactor. Although provisions had been made to vent the lower side of the lower plenum cap to the suction side of the shield solution circulating pump (to assure that the lower surface of the plenum cap was wet with shield solution at all times), the leak rate from the outlet gas duct flange (1 - 2 scfh) exceeded the capacity of the vent system. A supplementary gas removal system was designed and installed in the reactor in late July. This system consisted of two small, variable speed self-priming pumps which maintained a suction on the existing lower plenum cap vent lines. The pumps discharged to a gas disengager; the gas was vented to the building exhaust system and the shield solution returned to the shield tank.(5)

f. Reactor Shielding: The evaluation of radiation data obtained during ML-1 power operation with 5% boric acid solution was completed during the quarter.⁽⁶⁾ It was found that the gamma dose rates during operation with 5% boric acid solution were about 20% greater than those measured during operation with 10% boric acid solution. No correlation was observed between the fast neutron dose rates and the concentration of boric acid. It was also noted that the gamma dose rate observed shortly after shutdown was essentially the same following operation with 2% and 5% shield solutions.

g. ML-1 Reactor Disassembly and Inspection: In early October 1965, it was established that a leak path existed in the ML-1 pressure vessel between the gas system and the moderator system. Tests revealed that the leak rate increased with increasing reactor coolant gas pressure (2270 cc/min at 220 psig and 640 cc/min at 100 psig) and that the location of the leak was in the region of the lower tube sheet. The reactor skid was uncoupled from the gas piping, the gas coolant passages in the reactor core were flooded with demineralized water and a program of diagnostic investigation was undertaken to determine the location and nature of the leak.

It was observed that gas bubbles were consistently released to the flooded interior of a peripheral pressure tube (7-G) when the moderator water was

drained and the moderator system pressurized with nitrogen to 35 psig. Borescopic examination revealed that the bubbles emerged in the space between the lower tube-to-tube sheet weld and the insulation retainer ring. The fact that bubbles appeared in position 7-G did not conclusively indicate that the seal weld on this pressure tube had failed; it was possible that the bubbles were being formed in some other location and traveling along the lower face of the tube sheet to emerge at the 7-G position.

It appeared that the only technique which would definitely locate the position of the leak was a direct examination of the lower surface of the lower tube sheet. Accordingly, a program was formulated for the disassembly and inspection of the pressure vessel.⁽⁷⁾ This program involved the defueling of the reactor, the transfer of the fuel elements to the TAN storage facility, conversion of the GCRF reactor pit to a pool facility, removal of the pressure vessel from the reactor skid, removal of the reflectors from the pressure vessel and the underwater examination of the "bare" pressure vessel in the GCRF pit.

The fuel elements and reactivity shims were removed from the reactor and a series of experiments were conducted in an attempt to further define the location of the leak. When the reactivity shim was removed from position 6-E, a leak was observed in the pressure tube immediately above the upper surface of the lower tube sheet and, at the same time, evidence of the leak at position 7-G disappeared and did not reappear for the balance of the test. These findings did not alter the earlier conclusions regarding the requirement for pressure vessel disassembly but suggested that certain disassembly and inspection steps be performed prior to the removal of the lower plenum to expose the lower tube sheet. A revised plan for the pressure vessel disassembly and inspection program is shown schematically in Figure 3.

Preparations for removal of the pressure vessel began early in October and, at the end of the month, all such preparations were completed. The GCRF reactor pit had been converted to a pool facility and filled with demineralized water. All tooling and fixtures required for the support and inspection of the pressure vessel had been installed in the pool. All work on this task was suspended by direction of the USAED-ID on 28 October 1965.

2.2 Instruments and Controls

a. Off-Normal Scram System: The fabrication of a breadboard assembly of the off-normal scram system chassis was completed and all fixed coefficient input and range resistors were selected. A revised 6 kw scram circuit was developed and tested and incorporated into the breadboard chassis. A 60 cps simulated flow rate sweep circuit was added to the system to improve the alignment accuracy of the function segment intercept and gain adjustments. The functional and environmental tests of the breadboard chassis were completed; these tests indicated that the design was satisfactory. As of the end of October, the final design was completed and fabrication of the final chassis was approximately 30% complete. The completion of the fabrication and testing of this system was authorized as part of the AGCRSP termination program; the results of this activity will be documented in a topical report published late in 1965.

Report No. IDO-28653

FIGURE 3. ML-1 PRESSURE VESSEL DISASSEMBLY - INSPECTION PROGRAM

b. Emergency Power Transfer Switch: The design and procurement of a replacement emergency power transfer switch for the ML-1 control cab was completed and the parts were stored pending resolution of the FY 66 program. The new switch eliminates the automatic reset feature incorporated in the original unit to increase the reliability of the transfer function.

c. Scram Circuit Switch Rewiring: The design of modifications to the ML-1 scram circuits to relocate switches from the neutral side to the "hot" side of the circuit was completed. This change will eliminate the possibility of an undetected short circuit to ground which could disable a scram circuit without providing an indication of a failure.

2.3 Fuel Elements

a. Thermal and Neutronic Analysis: The ML-1A coolant loss accident was analyzed with the ANASIM computer code. It was assumed, that the fuel element was operating at a hot spot temperature of 1800°F just prior to the accident and that the reactor shutdown margin was -4% $\Delta K/K$. The results of this analysis are presented in Table 2.

TABLE 2 - ML-1A COOLANT LOSS ACCIDENT RESULTS

<u>Flow Coastdown Time, sec.</u>	<u>Peak Transient Temperature, °F</u>
0	2143
4	1848
20	1695

The temperature of the ML-1-I fuel elements after 30 and 60 days of decay were computed to support the ML-1 defueling and fuel element transfer operations. These calculations indicated that all component temperatures were less than the normal operating values; the peak fuel pin temperature after 30 days of decay was computed to be 940°F.

Calculations were performed to predict ML-1-I fuel element temperatures during storage in the dry condition in the ARA hot cell. These calculations indicated that the natural convective heat transfer, which occurs on both the inner and outer surfaces of the fuel element, will limit the component temperatures to less than 900°F after a decay period of 30 days.

The reduction and evaluation of control rod recalibration data was completed. The conclusions of this work were as follows:

- o The worth of shim rod 2 is 1.58% $\Delta K/K$. Measured values ranged from 1.40 to 1.82% $\Delta K/K$; the standard deviation of the quoted value is 0.11% $\Delta K/K$. The previous quoted worth based on 1961 data(8) was 1.48% $\Delta K/K$.

- The worth of the regulating rod is a function of the positions of the shim rods is as shown in Table 3.

TABLE 3 - WORTH OF ML-1 REGULATING ROD

<u>Position of Shim Rods</u>			<u>Worth of Regulating Rod</u>
<u>#1</u>	<u>#2</u>	<u>#3</u>	
40°	56°	40°	0.45 ± 0.01% ΔK/K
60°	17°	60°	0.51 ± 0.01% ΔK/K

- The shutdown margin of the ML-1 reactor in the cold condition (moderator water at 68°F) is $5.4 \pm 0.3\%$ ΔK/K. This is the worth of the rods inserted into the reactor from a "just-critical" condition. In the just-critical condition (10 June 1965), 2.0% ΔK/K worth of rods are already in the reactor; the total worth of all rods is, therefore, 7.4% ΔK/K.

- The comparison of results obtained from the rod drop and period methods of rod calibration with the preliminary results of the on-line analog computer method do not indicate reasonable agreement.

b. ML-1 Engineering Support: Analyses were performed to predict ML-1-I fuel element temperatures and stress levels during transient operation. The thermal characteristics of the ML-1/GCRF system immediately following a scram are significantly different from those of the ML-1 system, primarily because of the characteristics of the ML-1/GCRF coolant system which operates essentially independent of the reactor. The coolant system operation after a ML-1/GCRF scram can follow any one of the following patterns:

- In the event of a normal scram, compressor operation (coolant flow) and the reactor coolant inlet temperature remain essentially unchanged.
- In the event of a power failure scram with normal operation of the auxiliary diesel power supply, the coolant flow is reduced to and maintained at 50% of full flow and the reactor coolant inlet temperature decreases as a result of automatic shutdown of the heater.
- In the event of a power failure scram without auxiliary diesel power, coolant flow decays to zero in about 9 seconds and the coolant inlet temperature decays as the heater is automatically shut down.
- In the event of a coolant loss scram, the coolant flow stops immediately.

The thermal analysis performed to determine transient fuel element temperatures for the above cases assumed that, at the time of the scram, the reactor was operating at 2.2 Mw and that the fuel elements were operating at a hot spot cladding temperature of 1800°F. The temperatures derived by these analyses(9)

were then used to establish fuel pin cladding stresses resulting from the transient thermal conditions. These analyses showed that the thermal transients are greatest for a normal scram when the maximum cladding cooling rate is 64°F/sec if the nominal as-built 0.001 in. radial gap exists between the fuel and cladding and 80°F/sec if the cladding has collapsed on to the fuel. In the case where a gap exists between the fuel and the cladding, the calculated stresses are less than those associated with the normal, steady state, full-power operation. However, in the case where the cladding has collapsed on to the fuel, a compressive stress of about 102,500 psi develops in the cladding; this stress will plastically deform the cladding about 0.00016 in. (diametrical increase). This change in the pin diameter is acceptable from both the coolant and fuel pin structural integrity considerations.

A power failure scram with auxiliary power will result in a maximum fuel pin cladding cooling rate of 40 to 58°F/sec . If the cladding was in contact with the fuel, the stresses associated with this cooling will result in a plastic deformation of the cladding of about 0.00004 in. A power failure scram without auxiliary power will also result in a maximum cooling rate of 40 to 58°F/sec with the associated cladding deformation discussed above. In this case, the fuel pin temperature will rise after the coolant flow decays to zero; a peak temperature of 1600°F will be reached 250 seconds after the scram. In the event of coolant loss accident, no cooling of the fuel pin cladding occurs. The maximum cladding temperature reaches 2000°F approximately 40 seconds after the accident; the stresses associated with this temperature increase are significantly smaller than those discussed above for the other shutdown cases.

The results of the analysis discussed above indicate that the normal scram case results in the least desirable conditions. In consideration of the fact that the IB-8T-2 test element was subjected to identical scram conditions 77 times during the successful 10,000 hr test in the GETR, it is considered that the normal scram is an acceptable but not desirable situation, especially if 70 or more such scrams are projected during ML-1/GCRF operation. A more desirable mode of operation following a scram would be to permit the coolant flow rate to decay to zero.

c. Fuel Element and Reactivity Shim Inspection: During the defueling of the ML-1 reactor in connection with the pressure vessel leak investigation, five selected fuel elements and one reactivity shim were inspected.⁽¹⁰⁾ The purpose of this inspection was to provide general qualitative information concerning the condition of the components to aid in the planning of subsequent hot cell examinations. The inspection revealed that the fuel elements were in good condition structurally but what appeared to be the initial stages of a reaction between the silver from the reactivity shims and the stainless steel outer liners of the fuel elements was observed. Considerable corrosion was observed at the interface between the stainless steel and the silver material in the reactivity shims and some evidence of separation of the silver from the stainless steel and spallation of the silver was noted.

d. ML-1-IM Fuel Element Fabrication: The assembly of 20 ML-1-IM fuel elements, including four instrumented elements, was completed during the report period. The only significant deviation from the design^(a, 11) was in the fuel loading; the design fuel loading was 736.8 ± 7.4 grams U^{235} per element and the average as-built loading was $727.8 \pm 4.4, - 4.9$ grams U^{235} per element. This deviation

resulted from the lower than anticipated bulk density of the fuel associated with the concave ends and small edge chips in the pellets. This deviation will not affect either the thermal or neutronic performance of the fuel.

The losses incurred in processing the fuel pin tubing are summarized in Table 4; the yields represented by these data are quite high because special care was exercised to minimize losses of this critical component. The losses associated with fuel processing are summarized in Table 5; the yield in this process is somewhat lower than anticipated because of problems associated with the development of production-type techniques for the centerless grinding of the fuel pellets.

TABLE 4 - FUEL PIN PROCESSING LOSSES

<u>Operation</u>	<u>Number Fuel Pin Tubes</u>	<u>% of Original Tubes</u>
1. Final Machining Losses	2	0.5
2. Lower Plug Welding Losses	3	0.8
3. Fuel Loading Losses	5	1.3
4. Upper Plug Welding Losses	3	0.8
5. Unloaded Spares	11	2.8
6. Loaded Acceptable Fuel Pins	<u>376</u>	<u>93.8</u>
7. Original Tubes (Acceptable)	400	100.0

TABLE 5 - SUMMARY OF FUEL PROCESSED

<u>Description</u>	<u>U-235 (grams)</u>	<u>% U-235 of Total</u>
1. Oversize Reject Pellets	1,126.32	5.5
2. Undersize Reject Pellets	1,900.22	9.2
3. Pellets Rejected During Loading	471.02	2.3
4. Acceptable Ground Pellets	<u>14,696.05</u>	<u>71.4</u>
Total Sintered Pellets	18,193.61	88.4
5. Sample Pellets	285.54	1.4
6. Scraps, Fines, and Losses	<u>2,104.28</u>	<u>10.2</u>
Total UO ₂ as Powder (Start)	20,583.43	100.0

2.4 System Performance Analysis

a. ML-1 Cycle Optimization: The study to determine the optimum turbine-compressor set characteristics for the ML-1 system was completed.⁽¹²⁾ A special computer program was developed to perform the calculations required for this study. The reference system evaluated was characterized by the most recent performance characteristics of the ML-1 reactor, the Stewart-Warner precooler and the improved recuperator.⁽¹³⁾ The study revealed that the turbine and compressor efficiencies have the most important effect on the power output of the system and that, for any combination of turbine and compressor efficiency, an optimum mass flow rate exists at which maximum output power may be realized. The study also indicated that the compressor pressure ratio has very little effect on output power in the range of pressure ratios from 2.60 to 2.80.

b. Stewart-Warner Precooler Performance Analysis: The reduction and analysis of data generated during the precooler test at PCTF were completed. Pressure drop and effectiveness values were developed for nitrogen flow rates from 12 to 28 lb/sec and inlet temperatures from 250 to 500°F with the cooling system fans operating at half and full speed for several configurations of major equipment on the power conversion skid. The flange-to-flange pressure drop as a function of coolant flow rate is shown in Figure 4. The observed pressure drop is somewhat higher than the design value; major differences between the design and test values occur in the precooler core and outlet headers and duct as shown in Table 6. The effect of the increase in pressure drop is to reduce the ML-1 net system power by approximately 16 kw.

The precooler effectiveness as a function of the nitrogen flow rate and skid configuration is shown in Figure 5. At the design flow rate of 27 lb/sec, the measured effectiveness ranged from 95.7% (with all skid components removed) to 91.7% (with the Griscom-Russell recuperator and other major skid component mockups installed). The reduction of the air-side flow data was in progress at the end of the report period. Since the predicted design point effectiveness was based on an air-side flow rate of 68.8 lb/sec, the comparison of the precooler heat transfer performance with design predictions cannot be made until the evaluation of the air-side flow data is completed.

TABLE 6 - PRESSURE LOSS SUMMARY

<u>Element</u>	<u>Design Estimates</u>	<u>Test Values</u>
Inlet headers and duct	0.82 psid	0.79 psid
Precooler core	1.07 psid	1.71 psid
Outlet headers and duct	<u>0.15 psid</u>	<u>0.55 psid</u>
Total	2.04 psid	3.05 psid

FIGURE 4. STEWART-WARNER PRECOOLER NITROGEN SIDE PRESSURE DROP CHARACTERISTICS

FIGURE 5. STEWART-WARNER PRECOOLER PERFORMANCE

c. CSN-1A(M) Turbine-Compressor Set Analysis: Predicted performance maps for the CSN-1A(M) turbine and compressor were transmitted by the General Electric Company in August 1965.⁽¹⁴⁾ These maps provided performance data in the form of overall blading performance and did not include inlet and outlet losses. The maps were converted to include the inlet and outlet losses to make the format consistent with that normally used in ML-1 systems analysis. The converted performance maps are shown in Figures 6 and 7.

d. CHOP Program Modification: The updated performance characteristics of the ML-1 system components⁽¹³⁾ were prepared in the input format for the CHOP ML-1 systems analysis program.⁽¹⁵⁾ The CSN-1A(M) turbine and compressor performance maps shown in Figures 6 and 7 were also converted to CHOP format to complete the revision of the code to reflect the latest information on performance characteristics of the ML-1 system components.

e. System Performance Analysis: A series of CHOP runs was made to predict the performance of the ML-1 system with the CSN-1A(M) t-c set. The precooler characteristics of Reference 13 were adjusted to agree with the most recent test data and the analytical model included the improved recuperator. The results of the performance calculations at design conditions with an ambient temperature of 100°F indicated that 319.7 kw (net) electrical output power would be obtained as shown in Table 7. The net electrical power output reflects the parasitic power loss of 67.5 kw, a 3600:1800 rpm gear set efficiency of 97%, a lube oil pump load of 6 kw, a generator efficiency of 93% and a 70 kw auxiliary power load. In addition, a 3% temperature rise was assumed from the precooler intake air due to system heat losses.

A series of CHOP runs was made to define the operating limitations during off-design performance of the various system components. Particular interest was focused on the reactor inlet temperature, precooler inlet temperature and fuel element hot spot temperature, since these operating parameters had been found to be limiting in earlier analyses. The results of these calculations are shown in Table 8; these data clearly indicate the desirability of re-orificing the existing ML-1-I core to obtain a more optimum coolant flow distribution.

f. ML-1 Influence Coefficients: A revised set of ML-1 influence coefficients were developed from the test data generated during the conduct of ANSOP 16650 and from the analytical evaluation of the CSN-1A(M) system described above.⁽¹⁶⁾ The major influence coefficients resulting from this work are shown in Table 9.

FIGURE 6. PREDICTED CSN-1A(M) COMPRESSOR PERFORMANCE MAP

11.90-66-142

FIGURE 7. CSN-1A(M) TURBINE MAP CLOSED CYCLE, FLANGE TO FLANGE PERFORMANCE WITH NITROGEN

11.90-66-143

TABLE 7 - CALCULATED DESIGN POINT PERFORMANCE
OF THE ML-1 SYSTEM WITH THE CSN-1A(M) T-C SET

<u>State Points</u>	<u>100°F Ambient Temperature</u>	
	<u>°F</u>	<u>psia</u>
<u>Compressor</u>		
In	123.5	119.0
Out	362.0	325.4
<u>Recuperator (HP)</u>		
In	362.0	324.9
Out	819.4	320.5
<u>Reactor</u>		
In	819.4	320.5
Out	1206.0	304.2
<u>Turbine</u>		
In	1200.0	303.5
Out	907.1	124.9
<u>Recuperator (LP)</u>		
In	907.1	124.9
Out	453.2	122.2
<u>Precooler</u>		
In	451.4	121.9
Out	123.5	119.0
Net Electrical Power, Kw	319.7	
Reactor Power, Kw	2948.6	
Compressor Flow, lb/sec	27.39	
Turbine Flow, lb/sec	26.84	
Control Bypass, lb/sec	0	
Compressor Efficiency (ISEN), %	79.76	
Recuperator Effectiveness, %	83.90	
Turbine Efficiency (ISEN), %	85.06	
Precooler Effectiveness, %	94.10	

TABLE 8 - CALCULATED LIMITING PARAMETERS FOR SELECTED ML-1
OPERATING POINTS WITH THE CSN-1A(M)

Ambient Temperature °F	Compressor Inlet Pressure psia	Turbine Inlet Temperature °F	Reactor Inlet Temperature °F	Precooler Inlet Temperature °F	F.E. Hot Spot Temperature Non-Optimum Orificing* °F	F.E. Hot Spot Temperature Optimum Orificing** °F	Reactor Power To Gas Mw	Net Elec- trical Output Power kw
100	119	1200	819.4	451.4	1812	1678	2.95	319.7
80	119	1200	804.9	439.5	1836	1697	3.18	389.6
40	119	1200	773.6	416.0	1885	1734	3.69	559.9
100	119	1150	783.9	443.2	1739	1610	2.84	262.1
80	119	1150	769.5	431.4	1764	1629	3.06	329.6
40	119	1150	738.9	407.2	1811	1666	3.55	491.8
100	119	1100	749.0	434.9	1665	1541	2.72	200.5
80	119	1100	737.9	423.0	1689	1560	2.94	264.4
40	119	1100	705.3	398.3	1736	1596	3.41	417.2

* Fuel cladding hot spot temperature of center ring of six fuel elements in the ML-1-I core as currently orificed.

** Fuel cladding hot spot temperatures, exclusive of center ring of six elements, in the ML-1-I core as currently orificed.

TABLE 9 - ML-1 INFLUENCE COEFFICIENTS

<u>Parameter</u>	<u>Units</u>	<u>ANSOP 16650 System</u>	<u>Improved System with CSN-1A(M)</u>
Turbine inlet temperature	kw/ ^o F	0.8	1.3
Ambient temperature	kw/ ^o F	-2.0	-3.6
Total pressure drop, $\Delta P/P$	kw/%	-17.0	-19.6
Turbine efficiency	kw/%	17.0	26.0
Compressor efficiency	kw/%	17.2	22.7
Precooler effectiveness	kw/%	10.8	14.5
Recuperator effectiveness	kw/%	1.4	1.3
Bypass fraction	kw/%	-14.5	-24.0

g. Precooler Air Temperature Rise: A study was initiated to investigate the temperature rise of the precooler intake air due to heat losses from the ML-1 system components. A 3^oF rise has normally been used in the ML-1 system analyses based on an earlier evaluation.⁽¹⁷⁾ However, the configuration used during the testing of the ML-1 power plant during ANSOP 16650 and ANSOP 16635 (precooler exhaust air duct) provided data from which a more accurate assessment of the air temperature rise could be made. Figure 8 illustrates the results of the evaluation of the test data; it appears from this work that the temperature rise to the precooler may be greater than the 3^oF normally used in the calculations but additional experimental and analytical work will be required to confirm this conclusion.

h. Scram Response Time Analysis: Equations describing the ML-1/GCRF system were written and programed on an analog computer. A second computer was programed to provide a detailed simulation of the reactor core temperature distribution. A schematic diagram of the computer arrangement is shown in Figure 9. A series of problems were run from which the allowable scram circuit delay time (as controlled by the fuel element hot spot cladding temperature upper limit) was determined for each of the scram circuits for a variety of abnormal operational conditions. A report describing the analysis, assumptions and development of the simulation equations was published,⁽¹⁸⁾ as was the final report describing the results of the analog runs and the evaluation of the adequacy of the scram circuits.⁽¹⁹⁾ The general conclusion reached was that the response times of the scram circuits in the ML-1/GCRF system were sufficiently short to provide adequate protection against all abnormal operational conditions investigated.

FIGURE 8. TEMPERATURE RISE FROM AMBIENT TO S.E. PRECOOLER FAN INTAKE PLENUM, ML-1

FIGURE 9. SCHEMATIC, ML-1/GCRF SYSTEM ANALOG SIMULATION

3.0 ML-1 DEVELOPMENT AND IMPROVEMENT

3.1 Reactor and Auxiliaries Development

a. Gas Generation and Storage Skid Mockup: All of the major components for the gas generation and storage skid mockup were procured in 1964. In July 1965, a program was initiated to assemble these components into a mockup of the system and to perform functional tests. A layout drawing was prepared showing the location of the components on the ML-1 gas storage skid base, a physical inventory of all materials on hand was completed and system piping and instrumentation diagrams were modified to reflect the instrumentation required for system evaluation. All work on this task was suspended early in August pending resolution of the FY 66 program.

b. ML-1 Pressure Vessel Development: The report describing the conceptual design of an advanced ML-1 pressure vessel capable of operation at 1400⁰F reactor outlet temperature and 500 psia coolant pressure was published in July.(20)

The report describing the limited developmental effort undertaken in support of the advanced pressure vessel design was also published in July.(21) This report included a discussion of the photoelastic experimental program performed to determine the effective elastic constants for stress analysis of ML-1-type tube sheets, the modifications to improve the precision and flexibility of the AGN-TSA computer code for stress calculations and the preliminary evaluation of the properties of candidate materials for use in the advanced pressure vessel.

3.2 Power Conversion Equipment Development

a. Clark Bros. Turbine-Compressor Set (CSN-1A): At the end of the prior report period, the aerodynamic design of the CSN-1A(M) had been completed by the General Electric Company and reviewed by Aerojet technical personnel.(a) However, the detailed design of the compressor discharge duct had not been defined at that time.

During the current report period, model flow tests were conducted by General Electric to establish a satisfactory configuration for the compressor discharge duct. The configuration selected for incorporation in the CSN-1A(M) had a total pressure loss at design conditions of 1.75% $\Delta P/P$ and a uniform distribution of flow around the circumference of the compressor blading discharge annulus. Model flow tests were also conducted by General Electric to evaluate the performance of the CSN-1A(M) reference design compressor inlet duct. These tests revealed that the pressure loss in the inlet at design conditions would be 0.54% $\Delta K/K$, that the flow distribution was quite uniform and that no flow separation occurred. At the conclusion of the model testing, General Electric Company completed the aerodynamic design report and transmitted this document to Aerojet for review and comment.⁽¹⁴⁾ General Electric also requested that the ML-1 cycle performance be re-evaluated on the basis of the turbine and compressor performance characteristics presented in the design report; the results of this analysis are discussed in Section 2.4.

The detailed inspection of the parts of the disassembled CSN-1A machine was completed and, based on the results of this inspection, a list of parts which would require replacement prior to buildup of the CSN-1A(M) was prepared. The procurement of these parts was initiated by Aerojet. The majority of the parts consisted of standard fasteners, fittings and connectors. The largest single item is a replacement intermediate case assembly; cracks between the inside diameter and the inner bolt circle were observed at 16 locations and between the inner and outer bolt circle in 3 locations in this assembly (Figure 10). The decision to provide a replacement intermediate case assembly was based on the fact that this component provides the structural support for and precisely locates the entire turbine end of the turbine-compressor set. General Electric recommended that stress relieving scallops be provided at the inside diameter of the replacement part as shown in Figure 10. The Aerojet analysis revealed that, under certain operating conditions, the temperature gradients in the assembly result in stresses which exceed the allowable design values for the 17-7 PH material and, as a consequence, the use of A-286 was specified for the replacement part. The fabrication of the replacement assembly was in progress at the end of the report period.

Aerojet procured the services of a DeLaval field service engineer to disassemble, inspect and reassemble the CSN-1A gear set. According to the report submitted by this engineer, the gear set was in excellent condition.

At the request of General Electric, Aerojet reviewed the design of the CSN-1A(M) test facility lubrication and seal gas systems and recommended several modifications. Several items of equipment associated with the lubrication and seal gas systems were loaned to General Electric for use during the test. In addition, the CSN-1A rotor assembly balance tooling was shipped to General Electric for use during the CSN-1A(M) buildup.

b. CSN-2 Turbine-Compressor Set: The design and fabrication of the CSN-2 open-cycle test facility at the PCTF was initiated. The procurement of several components was completed and the preparation of mechanical design layout drawings and instrumentation and control designs was completed by the end of September at which time all work was suspended by direction of the USAEC.

PRESENT INTERMEDIATE CASE DESIGN

STRESS RELIEVING SCALLOPS

FIGURE 10. CSN-1A INTERMEDIATE CASE DESIGN

11.90-66-146

c. Precooler: The preparations for the heat transfer performance tests of the precooler were completed. The fabrication of the test facility, including a gas-fired heater capable of heating 27 lb/sec of nitrogen at 130 psia from 125°F to 500°F, a gas blower capable of delivering 27 lb/sec at 120 psia compressor inlet pressure with a pressure rise of 16 psi, a drive system powered by a 765 hp d.c. motor and a water-calibrated venturi-type flowmeter, was completed (Figure 11). A report describing the design of the test facility was published.(22)

A test program was conducted to establish that the precooler satisfied the performance requirements of the procurement specification, to evaluate the effects of operating the assembly in the compact configuration and to obtain performance data for power plant systems analysis. The results of these tests are discussed in Section 2.4. Core and header metal temperature and air flow velocity profiles were obtained as a function of nitrogen and atmospheric temperatures; the evaluation of these data was in progress at the end of the report period. A 100 hr endurance run was completed with no evidence of deterioration of the precooler core or cooling air fan assemblies.

d. ML-1 Alternator: An endurance test of the 600 kw alternator was completed by Fairbanks-Morse in July and the unit was delivered to San Ramon. The results of this test indicate that the alternator is capable of developing the electrical power and voltage specified for the ML-1 power plant but the power quality and the voltage regulation and control characteristics of the exciter-regulator could not be demonstrated at the Fairbanks-Morse facility. The fabrication of an alternator test facility at the PCTF, including a 1000 hp d.c. motor and the appropriate speed reduction units, was completed and the alternator was installed in preparation for testing (Figure 12). A design report for the alternator test facility was published.(23) Alternator testing was deferred at the direction of the USAEC pending resolution of the FY 66 program.

e. GDF Lubrication System: The program of functional checkout and adjustment of the components of the GDF lubrication system was completed. These tests established that the absorption capacity of the desiccant system was sufficient to maintain the hydrocarbon concentration in the return buffer gas at less than 1 ppm for approximately 125 hr at conditions simulating a refrigerator failure. The mechanical oil/gas separator was flow tested and operated satisfactorily. The sump tank level indicators, level alarms, heaters and temperature controls were satisfactorily checked out. A functional check and adjustment of all valves, regulators and pressure switches was completed. The power requirements were determined and the performance and reliability of the standby lube pump and startup compressor were verified. The startup compressor was used as a gas circulator for the desiccant system tests during which time 400 hr of operation were logged on the unit.

Preparations were initiated for the system demonstration and evaluation testing but this activity was suspended by direction of the USAEC pending resolution of the FY 66 program. The report describing the design of the GDF lubrication system was published.(24)

FIGURE 11. PRECOOLER TEST FACILITY

FIGURE 12. ROTATING MACHINERY-ALTERNATOR TEST

f. ML-1 Hydraulic Starting System: The power requirements for starting the CSN-1A(M) t-c set were determined and final design drawings for a hydraulic starting system capable of delivering 300 ft-lb breakaway torque and 30 kw driving power up to rated t-c set were prepared. A test description, outlining the test program for evaluation of the starting system, was published.(25) The procurement of system components and the continuation of the design were deferred in September by direction of the USAEC.

3.3 ML-1-II Core Elements

a. IB-17R Out-of-Pile Support: The elevated temperature tensile tests of as-received IB-17R tubing were completed. These data, shown in Table 10, provide baseline information for comparison with the high temperature properties of aged and irradiated IB-17R tubing.

TABLE 10 - ELEVATED TEMPERATURE TENSILE PROPERTIES
OF AS-RECEIVED HASTELLOY X HEAT X14593

<u>Sample No.</u>	<u>Tensile Test Temp., °F</u>	<u>Ultimate Tensile Strength, psi</u>	<u>Yield Strength, psi</u>	<u>% Elong. in 1-in.</u>	<u>% Reduction in Area</u>
FKA-34-2	1300	66,000	32,200	43	35
FKA-34-1*	1300	68,900	33,100	54	39
FKA-8-1	1450	38,300	30,300	78	53
FKA-8-2	1600	23,500	22,500	55	41
FAK-13-2	1750	15,400	14,800	38	32
FAK-13-1	1750	15,800	15,700	36	32
FAK-1	1800	11,500	11,400	31	27

*Test interrupted by malfunction; continued later.

Samples of ML-1 and IB-17R tubing exposed in the laboratory at 1750°F for 10,000 hr were transmitted to a commercial laboratory for microprobe analysis. This work had been completed by the end of the report period but the results had not been received by Aerojet. All activity under this task was suspended at the end of September; specimens which had not been evaluated were appropriately marked and stored in the event of a decision to continue the work at a later date.

b. IB-17R-3 Test: An analysis was performed to compare the quantity of fission product release by the failure of the cladding of fuel pin 1 in the IB-17R-3 test element with that released in the IB-7T defective fuel pin test.(26) It was found that the IB-17R release was comparable with that of the IB-7T and that this release in the ML-1 system, under the most adverse weather

conditions, would result in fission product concentrations below the MPC at the control cab. Accordingly, it was concluded that the ML-1 power plant could continue normal full power operation with a fuel cladding failure of the type and nature which occurred in the IB-17R-3 test element.

The decision was made to defer indefinitely the hot cell destructive evaluation of the IB-17R-3 test element.

**THIS PAGE
WAS INTENTIONALLY
LEFT BLANK**

III. ML-1 TECHNOLOGY PROGRAM4.0 FUEL ELEMENT TECHNOLOGY4.1 Hastelloy X Cladding Evaluation

a. Hastelloy X Corrosion Studies: Metallographic evaluation of Hastelloy X sheet specimens exposed for 2500 and 5000 hours at 1900°F air was completed. Internal oxidation was found to be the primary mode of attack although carbide depletion was observed in varying degrees in some of the samples. The results of this evaluation are presented in Table 11.

TABLE 11 - AVERAGE MAXIMUM DEPTH OF OXIDE PENETRATION INTO
HASTELLOY X SHEET EXPOSED IN 1900°F AIR, INCHES

<u>Heat No.</u>	<u>100 hr</u>	<u>1000 hr</u>	<u>2500 hr</u>	<u>5000 hr</u>
X4762	0.0012	0.0015	0.0030	0.0039
X9505	0.0014	0.0018	0.0035	0.0045
X34381	0.0012	0.0020	0.0040	0.0050

Hastelloy X sheet samples exposed for 1000 and 2500 hr at 2000°F air were examined metallographically to determine the extent of oxidation; the results are shown in Table 12.

TABLE 12 - AVERAGE* DEPTH OF INTERNAL OXIDATION OF HASTELLOY X
SHEET EXPOSED IN 2000°F AIR, INCHES

<u>Heat No.</u>	<u>Exposure Time</u>			
	<u>100 hr</u>	<u>1000 hr</u>	<u>2500 hr</u>	<u>5000 hr</u>
X4762	0.0018	0.0045	0.0078	0.0112
X9505	0.0020	0.0053	0.0078	0.0135
X34381**	0.0020	0.0056	0.0084	0.0119

* Average of three samples.

** Also subject to some intergranular attack.

The microhardness of Hastelloy X sheet specimens exposed for 100, 1000 and 2500 hr in 2000°F air is shown in Table 13. As indicated in these data, the edge of the specimen was consistently less hard than the center, probably as a result of the internal oxidation and decarburization observed near the edge.

TABLE 13 - AVERAGE* MICROHARDNESS OF HASTELLOY X SHEET SAMPLES EXPOSED IN 2000°F AIR (KHN)

Heat No.	Exposure Duration, Hr					
	100		1000		2500	
	Center	Edge	Center	Edge	Center	Edge
X4762	221	219	242	237	243	222
X34381	212	207	241	228	235	196
X9505	229	211	233	232	231	219

* Average of three coupons.

X-ray diffraction analyses were performed on the surface of Hastelloy X sheet specimens exposed for 500 hr in 1900 and 2000°F air with the oxide scale in situ to characterize the scale and determine if any differences resulted from differences in alloy composition and exposure temperatures. The most significant finding was that, after exposure at 2000°F, only a Cr_2O_3 oxide was observed while the presence of a spinel (nominally NiCr_2O_4) was also observed on the specimens exposed at 1900°F. The range of spinel concentration in the oxide is shown by the relative intensities of the major peaks in the X-ray diffraction pattern, as summarized in Table 14.

TABLE 14 - RATIO OF MAJOR X-RAY DIFFRACTION PEAK HEIGHTS FOR SPINEL AND Cr_2O_3 OXIDES ON HASTELLOY X SHEET MATERIALS EXPOSED 500 HR AT 1900°F IN AIR

Heat No.	Spinel to Cr_2O_3 Peak Ratio
X4762	70:20
X34381	55:29
E9505	26:48

A reasonable correlation exists between the presence of the spinel and the degree of oxidation corrosion as evidenced by the weight change data. This correlation suggests that the spinel is instrumental in forming a tenacious, protective oxide. If the alloy chemistry could be altered to promote spinel formation at 2000°F, the suitability of Hastelloy X for use at this temperature would be significantly improved.

Microconstituents were electrolytically extracted from two Hastelloy X sheet samples and, after reduction in hydrogen for 100 hr at 2100°F, were subjected to fluorescent analyses to determine whether the M_6C particles could be decarburized and the form of the alloy in the event such conversion occurred. One of the samples, identified as 65-70, contained both M_6C particles and Mu phase before the hydrogen reduction treatment; the other sample (65-58) contained only M_6C particles. The results of the fluorescent analyses, shown in Table 15, indicate that no significant change occurred in the material constituents but that the sample which previously contained the Mu phase was richer in molybdenum and poorer in nickel, chromium and cobalt than the material containing the M_6C particles. X-ray diffraction analysis indicated that sample 65-58, after treatment, consisted of M_6C and austenite while 65-70, after treatment, consisted of M_6C particles and a delta phase. The delta phase was identified as that reported by Rideout and Back in Ni-Cr-Mo alloys at 1200°C. These results indicate that the Mu phase in this alloy is apparently unstable at 1100°C.

TABLE 15 - X-RAY FLUORESCENCE PEAK INTENSITIES
(Mo = 100)

Before H ₂ Reduction		After H ₂ Reduction	
	65-70	65-70	65-58
Mo	100	100	100
Ni	50	42	98
Fe	15	14	16
Cr	10	8	68
Co	0	0	3

Room temperature tensile tests of Hastelloy X sheet specimens exposed for 2500 and 5000 hr at 1900 and 2000°F were completed. The results of these tests, along with the data for the shorter duration exposures, are shown in Tables 16 and 17.

The elevated temperature tensile tests of Hastelloy X sheet specimens exposed for 5000 hr at 1900 and 2000°F were completed. These results, some of which have been reported earlier, are presented in Table 18. These data indicate that the specimens aged for 5000 hr have poorer 1300 and 1600°F strength than the as-received material but, at 1900 and 2000°F, the aged alloys are slightly stronger.

TABLE 16 - ROOM TEMPERATURE TENSILE PROPERTIES OF
HASTELLOY X EXPOSED IN 1900°F AIR

<u>Heat No.</u>	<u>Exposure Time, hr</u>	<u>Ultimate Tensile Strength, psi</u>	<u>Yield Strength, psi</u>	<u>% Elongation in 1-inch</u>
X4762	As-Received	114,400	60,000	40.4
E9505	As-Received	109,700	59,000	35.5
X34381	As-Received	123,000	77,800	35.8
X4762	100	104,400	---	40.3
E9505	100	102,700	---	43.3
X34381	100	107,900	---	40.1
X4762	1000	103,400	---	39.6
E9505	1000	98,900	---	43.0
X34381	1000	101,600	---	24.0(?)
X4762	2500	92,000	29,300	44.5
E9505	2500	91,100	38,600	40.8
X34381	2500	91,500	34,900	37.5
X4762	5000	88,600	34,700(?)	42.8
E9505	5000	89,600	41,300(?)	43.7
X34381	5000	88,600	41,800(?)	33.5
Ave	As-Received	115,700	65,600	37.2
Ave	100	105,000	---	41.2
Ave	1000	101,300	---	41.3
Ave	2500	91,500	34,300	40.9
Ave	5000	88,900	39,300(?)	40.0

TABLE 17 - ROOM TEMPERATURE TENSILE PROPERTIES OF HASTELLOY X
SHEET MATERIALS EXPOSED IN 2000°F AIR

<u>Heat No.</u>	<u>Exposure Time, hr</u>	<u>Ultimate Tensile Strength, psi</u>	<u>Yield Strength, psi</u>	<u>% Elongation in 1-inch</u>
X4762	100	98,500	---	40.8
E9505	100	100,200	---	40.0
X34381	100	97,200	---	40.6
X4762	1000	86,500	---	20.3
E9505	1000	96,300	---	38.0
X34381	1000	96,900	---	38.0
X4762	2500	82,900	34,600	33.7
E9505	2500	83,200	35,600	33.1
X34381	2500	81,500	34,200	28.0
X4762	5000	74,500	31,100	26.4
E9505	5000	74,900	28,900	27.4
X34381	5000	73,900	37,000	22.3
Ave	100	98,600	---	40.4
Ave	1000	93,200	---	25.7
Ave	2500	82,500	34,800	31.6
Ave	5000	74,400	32,300	25.3

TABLE 18 - ELEVATED TEMPERATURE TENSILE PROPERTIES OF
HASTELLOY X SHEET, AS-RECEIVED, AND AS-EXPOSED IN
AIR FOR 5000 HR AT 1900 OR 2000°F

<u>Sample Number</u>	<u>Heat Number</u>	<u>Exposure Temp. °F</u>	<u>Tensile Test Temp. °F</u>	<u>Ultimate Strength, psi</u>	<u>Yield Strength, psi</u>	<u>% Elongation</u>
R-7	X4762	As-Rec'd.	1300	73,300	43,100	47
R-3	X4762	As-Rec'd.	1300	67,700	43,000	44
R-2	X34381	As-Rec'd.	1300	71,460	47,000	63
R-8	E9505	As-Rec'd.	1300	66,100	37,600	18.3
19-14	X4762	1900	1300	55,600	23,900	40.5
14	E9505	1900	1300	58,500	25,200	49
19-8	X4762	2000	1300	50,700	22,900	27.5
19-8	E9505	2000	1300	50,700	21,500	34
R-9	X4762	As-Rec'd.	1600	33,900	29,600	66.4
R-4	E9505	As-Rec'd.	1600	32,900	29,300	66.0
R-9	X34381	As-Rec'd.	1600	30,750	26,600	98.5
13	E9505	1900	1600	28,200	25,500	75.5
19-14	X34381	1900	1600	29,600	21,400	54
19-7	E9505	2000	1600	27,800	16,800	48
19-8	X34381	2000	1600	28,200	17,500	38.5
R-12	X4762	As-Rec'd.	1900	11,630	9,280	56.4
R-11	E9505	As-Rec'd.	1900	12,500	11,800	48.4
R-10	X34381	As-Rec'd.	1900	9,530	8,200	60.8
19-13	X4762	1900	1900	15,600	14,800	39
19-13	X34381	1900	1900	11,140	8,000	24
R-11	X4762	As-Rec'd.	2000	8,370	7,600	49
R-12	E9505	As-Rec'd.	2000	8,900	7,500	42.5
R-11	X34381	As-Rec'd.	2000	8,650	7,730	26.7
19-7	X4762	2000	2000	8,995	6,940	21.5
19-7	X34381	2000	2000	8,790	6,450	31.4

The results of elevated temperature tests of Hastelloy X sheet specimens aged in air at 1900 and 2000°F for shorter durations are shown in Table 19.

TABLE 19 - ELEVATED TEMPERATURE TENSILE PROPERTIES OF HASTELLOY X
AFTER HIGH TEMPERATURE EXPOSURE IN AIR

<u>Heat No.</u>	<u>Exposure Temp., °F</u>	<u>Time, hr</u>	<u>Tensile Temp., °F</u>	<u>Ultimate Tensile Strength, psi</u>	<u>Yield Strength, psi</u>	<u>% Elongation</u>
E9505	1900	100	1300	72,900	28,600	51
X4762	1900	100	1300	74,300	34,400	46
X4762*	1900	1000	1300	71,000	33,200	57
X34381*	1900	1000	1300	69,500	27,100	40
E9505*	1900	100	1600	35,700	27,100	84
X34381	1900	100	1600	38,000	24,600	72
E9505	2000	100	1300	69,700	29,400	42
X4762	2000	100	1300	70,900	26,800	36
E9505*	2000	100	1600	36,800	26,500	69
X34381	2000	100	1600	39,900	24,500	64
E9505*	2000	1000	1600	36,000	23,500	45

* New data points.

The room and elevated temperature tests of Hastelloy X tubing exposed for 5000 hr at 1900 and 2000°F are summarized in Table 20.

TABLE 20 - TENSILE PROPERTIES OF HASTELLOY X TUBING
AFTER 5000 HOUR AIR EXPOSURE

<u>Sample No.</u>	<u>Heat No.</u>	<u>Exposure Temperature °F</u>	<u>Test Temp., °F</u>	<u>Ultimate Tensile Strength, psi</u>	<u>Yield Strength, psi</u>	<u>% Elong.</u>	<u>% Reduction in Area</u>
FKA-112-B	X14593	1900	70	88,800	40,500	38.2	31.8
KKA-112-I	X14593	2000	70	88,600	40,700	46.4	32.8
A-2136-A	E9517	1900	70	80,000	39,100	27.6	22.1
A-2136-H	E9517	2000	70	84,100	42,500	38.4	26.5
A-2136-K	E9517	1900	1300	57,500	29,200	23.3	19.3
A-2255-D	E9517	2000	1300	54,600	27,100	29.2	30.0
A-2255-K	E9517	1900	1600	27,300	24,100	30.4	28.9
A-2255-H	E9517	2000	1600	27,600	23,000	32.2	30.6
A-2255-A	E9517	1900	1900	20,400	17,000	43.5	29.5

The results of elevated temperature tensile tests of Hastelloy X tubing exposed for 10,000 hr in air at various temperatures is presented, along with the previously reported as-received and 7500 hr data, in Table 21.

TABLE 21 - ELEVATED TEMPERATURE TENSILE PROPERTIES OF HASTELLOY X
HEAT E9517 EXPOSED IN AIR

Sample Number	Tensile Test Temp. °F	Exposure		Strength, psi		% Elongation in 1-in.	% Reduction In Area
		Time, hr	Temp. °F	Ultimate Tensile	0.2% Yield		
A-2306-1	1300	As Received		64,500	27,300	68	51
A-2348-H	1300	7,500	1300	64,300	30,400	45	32
A-2348-D	1300	7,500	1750	62,000	30,000	29	24
A-2348-E	1300	7,500	1800	60,900	28,200	41	32
A-6972-I	1300	10,000	1300	70,400	37,700	44	35
A-6970-E	1300	10,000	1750	61,500	29,200	45	31
A-2306-4	1450	As Received		50,200	33,800	72	50
A-2348-B	1450	7,500	1450	74,600	31,300	40	33
A-6970-G	1450	10,000	1450	47,400	28,600	17	16
A-2306-3	1600	As Received		29,100	25,200	64	44
A-2348-G	1600	7,500	1600	27,500	23,700	49	37
A-2558-F	1600	7,500	1750	27,600	23,600	47	33
A-6972-G	1600	10,000	1600	29,300	21,200	37	30
A-6971-D	1600	10,000	1750	26,600	23,300	43	35
A-230602	1750	As Received		19,200	16,300	61	39
A-2558-B	1750	7,500	1300	18,600	16,200	48	36
A-2558-H	1750	7,500	1450	16,500	15,700	47	34
A-2558-C	1750	7,500	1600	17,500	15,500	35	27
A-2558-D	1750	7,500	1750	17,200	15,100	50	30
A-6971-B	1750	10,000	1300	18,100	16,000	41	36
A-6971-H	1750	10,000	1450	18,300	16,000	38	32
A-6971-C	1750	10,000	1600	17,100	14,300	46	31
A-6971-F	1750	10,000	1750	17,200	15,700	43	32
A-2558-E	1800	7,500	1800	13,600	13,400	43	27
A-6971-E	1800	10,000	1800	13,900	13,700	44	27

The metallographic evaluation of several samples of Hastelloy X material which had been aged for long durations in 1850, 1900 and 2000°F air and subsequently exposed in a 1300°F inert gas atmosphere for 100 and 1000 hr was completed. The results indicate that the subsequent aging at 1300°F drastically alters the microstructure from that observed following the higher temperature exposures. The spherical carbide precipitates formed along the grain boundaries are altered by the precipitation of a fine microconstituent (either carbide or sigma phase) along the twin boundaries of the alloys. This change would be expected to be highly detrimental to the ductility of the material. The structure observed in specimens after initial exposure at 1900 and 2000°F and subsequent exposure at 1300°F is shown in Figures 13 and 14. Microconstituents were electrolytically dissolved from these samples and subjected to X-ray diffraction analysis in an attempt to identify the microconstituent; the only microconstituent was found to be M_6C resulting from a secondary precipitation caused by the lower carbon solubility at the lower exposure temperature. The lattice parameters for the newly formed M_6C were found to be essentially identical with those formed as a result of higher temperature exposure, as shown in Table 22.

TABLE 22 - LATTICE PARAMETER OF M_6C PRECIPITATES IN HASTELLOY X SHEET SPECIMENS BEFORE AND AFTER AGING FOR 100 HR AT 1300°F

Previous Exposure Heat No.	1000 Hours 2000°F		1000 Hours 1900°F		5000 Hours 1850°F	
	Before	After*	Before	After*	Before	After*
X4762			10.916	--	10.936	10.847
X34381	10.918	10.927	10.935	10.926	--	--
E9505	10.945	10.916	10.950	10.918	--	--
E9517	--	--	--	--	10.943	10.921

* 100 hour exposure.

Room temperature tensile tests of Hastelloy X specimens aged in inert gas at 1300°F for 100 and 1000 hr were completed. The results of this activity are summarized in Tables 23 and 24.

All activity in this area was discontinued at the end of September; samples which had not yet been evaluated were properly identified and stored for possible future use and a final report was published.(b)

b. Continuous Weighing Experiment: The evaluation of data generated in the 1850°F continuous weight change experiment was completed and the oxidation rate constants, diffusion constants and activation energies were calculated (Table 25). A plot of weight gain data from the 1850°F experiment and the 1750°F experiment is shown in Figure 15.

ETCHED

500X

HASTELLOY X, HEAT X-4381 - AFTER 1000 HOUR
EXPOSURE TO 1900

ETCHED

500X

HASTELLOY X, HEAT X-4381 - AFTER 1000 HOUR
EXPOSURE TO 1900 AIR FOLLOWED BY A 1000 HOUR,
1300°F ARGON EXPOSURE

FIGURE 13.

ETCHED
HASTELLOY X, HEAT X-4381 - AFTER 1090 HOUR
AIR EXPOSURE AT 2000°F

500X

HASTELLOY X, HEAT X-4381 - AFTER 1909 HOUR AIR
EXPOSURE AT 2000°F FOLLOWED BY A 1000 HOUR
1300°F ARGON EXPOSURE

500X

FIGURE 14.

TABLE 23 - ROOM TEMPERATURE TENSILE PROPERTIES OF HASTELLOY X
AGED IN INERT GAS AT 1300°F

<u>Sample Number</u>	<u>Heat No.</u>	<u>Exposure Time, hr</u>	<u>Ultimate Tensile Strength, psi</u>	<u>Yield Strength, psi</u>	<u>% Elongation in 1-inch</u>
R-16	X4762	100	108,200	60,000	11.4
R-22	X34381	100	107,100	58,700	11.2
R-9	E9522	100	114,500	61,600	11.3
R-22	E9505	100	97,500	58,200	6.8
II-I	E9517*	100	113,000	73,200	44
III-I	X14592*	100	113,000	64,900	40
R-25	X4762	1000	106,700	43,500	21.5
R-25	X34381	1000	112,800	45,400	21.2
R-14	E9522	1000	107,300	46,400	24.3
R-25	E9505	1000	105,900	44,300	20.1
1	E9517*	1000	113,400	53,700	12.2
2	X14593*	1000	110,900	54,300	13.7

* Tubing samples.

TABLE 24 - ROOM TEMPERATURE TENSILE PROPERTIES OF HASTELLOY X
AGED IN INERT GAS AT 1850°F

<u>Sample Number</u>	<u>Heat No.</u>	<u>Exposure Time, hr</u>	<u>Ultimate Tensile Strength, psi</u>	<u>Yield Strength, psi</u>	<u>% Elongation in 1-inch</u>
R-31	X4762	100	108,200	43,900	41.0
R-31	X34381	100	107,300	41,300	39.1
R-12	E9522	100	107,300	43,400	40.5
R-32	E9505	100	104,800	40,600	45.2
IV-1	E9517*	100	112,300	66,000	39
IV-1	X14593*	100	112,000	66,700	39
R-15	X4762	1000	100,000	41,400	29.9
R-16	X34381	1000	98,800	32,900	24.3
R-7	E9522	1000	100,000	47,350	25.8
R-15	E9505	1000	92,600**	38,200	24.3
3	E9517*	1000	101,400	42,400	21
2	X14593*	1000	98,200	44,800	16

* Tubing samples

** Broke in radius.

TABLE 25 - OXIDATION RATE CONSTANTS, DIFFUSION CONSTANTS, AND
ACTIVATION ENERGY CONSTANTS FOR HASTELLOY X

Specimen Number	Test Temp.		Avg. Temp. K	n' (1)	n' Average	W _g mg/cm ²	K ⁽²⁾ (g/cm ²)/sec n'	K (average) (g/cm ²)/sec n'	Q ⁽³⁾ (cal/mole)		A ⁽⁴⁾ (g/cm ²)/sec n'
	F	K							For Separate Tests	For Combined Tests	
LOW COBALT:											
2 - 6	1743	1224.6	1231.3	0.333	0.328	4.00	8.064x10 ^{-6*}	9.87x10 ^{-6*}	6.278x10 ⁴	6.885x10 ⁴	3.840x10 ⁵
4 - 8	1767	1238.0		0.323		5.15	1.067x10 ^{-5*}				3.843x10 ⁵
2 - 6	1839	1276.5	1281.4	0.318	0.318	5.19	2.700x10 ^{-5**}	2.94x10 ^{-5**}	6.408x10 ⁴	6.885x10 ⁴	2.374x10 ⁶
4 - 8	1856	1285.3		0.318		6.10	3.180x10 ⁻⁵				2.462x10 ⁶
Calculated from log K vs $\frac{1}{T}$ plot (Figure 7)										6.954x10 ⁴	2.132x10 ⁷
HIGH COBALT:											
1 - 5	1752	1.2296	1230.1	0.326	0.325	4.20	1.060x10 ^{-5*}	1.08x10 ^{-5*}	9.881x10 ⁴	5.804x10 ⁴	1.112x10 ⁶
3 - 7	1754	1.2307		0.324		4.25	1.100x10 ^{-5*}				1.085x10 ⁶
1 - 5	1841	1278.1	1282.0	0.385	0.385	5.20	2.747x10 ^{-5**}	2.821x10 ^{-5**}	6.290x10 ⁴	5.838x10 ⁴	1.395x10 ⁶
3 - 7	1855	1286.0		0.385		5.61	2.896x10 ^{-5**}				1.425x10 ⁶
Calculated from log K vs $\frac{1}{T}$ plot (Figure 7)										5.838x10 ⁴	2.542x10 ⁵

(1) Exponent of t, calculated from slope of log-log curves (Figure 4).

(2) Oxidation rate constant, calculated from equation 5.

(3) Activation energy constant, calculated from equation 2.

(4) Diffusion constant, calculated from equation 9.

* For t > 3 hours (1750°F)

** For t > 2 hours (1850°F)

FIGURE 15. COMPARISON OF WEIGHT GAIN OF HIGH AND LOW COBALT HASTELLOY X AT THE INDICATED TEMPERATURES

Metallographic evaluation of the tubing samples from the 1850°F test revealed extensive, semi-continuous surface oxide films on all samples. The extent of attack was greater on the outer surface than on the inner surface of the tubing. Although considerable difference in the extent of attack was apparent between samples, in general a distinctive and different mode of attack was observed on each of the two types (cobalt content) of tubing. In the normal cobalt material, a uniform band of internal oxide formed beneath the exposed surface and the area affected by this internal oxidation was depleted in carbides. In the low cobalt material, the band of internal oxides was much less uniform and no evidence of carbide depletion was observed.

The extent of oxide penetration in the various samples is summarized in Table 26.

TABLE 26 - OXIDATION PENETRATIONS OF HASTELLOY X TUBING
AFTER 3998 HOUR EXPOSURE IN 1850°F AIR

<u>Sample Number</u>	<u>Tube Number</u>	<u>Heat Number</u>	<u>Max. Penetration Inches</u>	<u>Comments</u>
1	FK-95	X14593	.0060	Internal oxidation to 0.0055 in.
2	A-1796	E9517	.0040	Little intergranular oxidation.
3	FK-17	X14593	.0060	Few intergranular penetrations.
4	A-1796	E9517	.0065	Massive oxides to 0.0035 in.
5	FK-95	X14593	.0045	Intergranular penetrations 0.004 in.
6	A-1796	E9517	.0050	Bulky oxide penetrations.
7	FK-17	X14593	.0050	Separated oxide sites to 0.008 in.
8	A-1796	E9517	.0060	Massive oxides to 0.0035.

The microhardness data on the various continuous weight change experiment samples is summarized in Table 27.

An evaluation of the results of the test led to the following conclusions:

- The scaling rate was greater for the low cobalt alloy than for the higher cobalt material when thermal cycled at 1750°F.

The weight gain/unit surface area was slightly higher for the low cobalt material than for the higher cobalt alloy although the rate of oxidation after the initial three hours of exposure was nearly the same for both.

- The scaling rate and unit weight gain at 1850°F were essentially the same for both materials.

- The unit scaling rate and unit weight gain of the low cobalt alloy were approximately 200% and 120%, respectively, greater at 1850°F than at 1750°F.

- The unit scaling rate and unit weight gain of the high cobalt alloy were approximately 400% and 120%, respectively, greater at 1850°F than at 1750°F.
- Oxidation penetrations in low cobalt alloy were less severe than those observed in the high cobalt alloy in the range of temperatures investigated. Some laminar type of attack was observed in the low cobalt specimens. The penetration in the high cobalt alloy resulted, predominantly, from intergranular oxidation; this was probably associated with the uniform band of internal oxides beneath the exposed surface and the depletion of the M_6C carbide particles which occurred in this band. No depleted band was observed in the low cobalt material.
- The oxidation penetration increased by about 80% in both alloys when the test temperature was increased from 1750 to 1850°F.
- A general softening was observed in both alloys when tested at 1850°F; this effect was more pronounced in the high cobalt alloy.
- X-ray diffraction analysis indicated that two principal oxides are formed in the spallation products; these are Mn_3O_4 and Cr_2O_3 . The ratio of Mn_3O_4 and Cr_2O_3 increased with time, indicating that the amount of Mn migrating from the Hastelloy X matrix increased as the exposure time increased.
- The plots of weight gain vs. time indicate that no change occurs in mechanism of oxidation for the low cobalt alloy between 1750 and 1850°F.
- The plots of weight gain vs. time indicate that a change occurs in the mechanism of oxidation of the high cobalt alloy between 1750 and 1850°F.
- The rate of weight gain for low cobalt Hastelloy X is nearly the same from $t =$ approximately 2 hours to $t = 4000$ hours at 1750 and 1850°F.
- The rate of weight gain for the high cobalt material increases significantly with time, indicating a change in the diffusion mechanism(s).
- The activation energy constant (Q) appeared to be independent of the test temperature but dependent on the alloy composition as demonstrated by both the analytical and graphical method of analysis.
- The oxidation rate constant (k) was shown to be temperature dependent, as anticipated.
- The values for the constants derived from this test agree well with those reported by Wlodek^(c); reported values of 1.02×10^{-5} and 1.52×10^{-5} for the oxidation rate constant (k) at temperatures of 1700 and 1800°F, respectively, compared favorably to test values of 9.87×10^{-6} and 2.94×10^{-5} for the high cobalt Hastelloy X at 1750 and 1850°F, respectively.

• The oxidation curves have an exponential temperature dependency and a cubic time dependency (slope 1/3) rather than a square dependency characteristic of oxidation processes. This means that the oxidation rate is lower than predicted by diffusion theory, that the formation of more than one oxide is likely to have occurred, and that oxidation in this alloy is a composite reaction produced by the diffusion of the several components.*

• Analyses of the oxide layers and oxide/metal interface and the development of data over a wider range of temperatures are necessary to determine the exact oxidation mechanism which applies in this case.

TABLE 27 - MICROHARDNESS OF AS-RECEIVED AND TUBING SPECIMENS
EXPOSED TO 1850°F AIR FOR 3998 HOURS

Sample No.	Tube No.	Condition	Heat No.	Microhardness, KHN			
				Location of Measurement-Tubing Wall			
				Inner Surface	Center	Outer Surface	Average
(Ave 7 tubes)		As-received	E9517	250.6	245.2	251.3	249.0
(Ave 4 tubes)		As-received	X14593	254.9	250.3	249.8	251.6
1	FK-95FK	Exposed	X14593	206.5	253.0	211.6	223.7
2	A-1796	Exposed	E9517	235.8	249.0	229.2	238.0
3	FK-17	Exposed	X14593	235.8	235.8	224.6	232.0
4	A-1796	Exposed	E9517	241.9	241.9	222.3	235.3
5	FK-95	Exposed	X14593	220.1	249.0	222.3	230.4
6	A-1796	Exposed	E9517	271.4	266.0	248.6	260.0
7	FK-17	Exposed	X14593	232.1	228.0	222.3	227.4
8	A-1796	Exposed	E9517	266.4	273.0	211.6	250.3

c. Hastelloy X Creep Studies: The 100 hour 1900°F creep test was completed and 700 hr of operation under the 1000 hr 1900°F test were completed prior to suspension of all activity under this task at the end of September. No significant progress was made in the analysis of the 1800°F test data by the time of AGCRS Program termination.

d. Hastelloy X Stress Rupture Studies: Tubing specimens for the stress rupture experiment had been prepared and the preparation of furnaces for aging the test specimens was completed prior to termination of all activity under this task.

*Wlodek states that after about 3 hr, the oxidation was parabolic with slope of 1/2 (square dependency) but curves plotted from his experimental data in the range of temperature of this experiment actually show a slope of about 1/2.8, much closer to the cubic dependency determined in this experiment. The smaller oxidation rate of the AGN specimens was due to the passive protective coating present on the as-received material used in the testing. Had the actual surface area of the AGN specimens (larger, due to surface roughness) been measured and taken into consideration, the resulting calculated oxidation rate would have been even lower than that reported; this supports the cubic rather than the square dependency.

**THIS PAGE
WAS INTENTIONALLY
LEFT BLANK**

IV. FACILITY ENGINEERING

5.0 GDF DESIGN AND CONSTRUCTION LIAISON

The decontamination, rehabilitation and stabilization of the GDF site was completed on 2 September 1965. The Title II design for the closed cycle test facility was completed by Norman Engineering on 21 September 1965.

Requests for quotation were transmitted to qualified vendors of the major equipment items for the test facility. These requests were cancelled at the time of cancellation of the AGCRS Program.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

REFERENCES WITH GENERAL DISTRIBUTION

- a. AGCRSP Quarterly Progress Report, 1 April through 30 June 1965, 13 August 1965.
- b. Studies of Hastelloy X Corrosion in Air at 1900 and 2000°F, AGN TM-414, December 1965.
- c. Wlocek, S. T., "The Oxidation of Hastelloy X," Transactions of AIME, Vol. 20, 177, February 1964.

REFERENCES WITH IN-CONTRACT DISTRIBUTION

1. OID-1516, "ML-1 Reactor Scram Response Measurements - Phase II Nuclear Instrumentation," 15 September 1965.
2. OID-1525, "Wet Critical Experiment Results," 17 September 1965.
3. OID-1508, "Weight Loss of ML-1 Reactivity Shim Liners," 7 September 1965.
4. AN-AGCR-930, "Summary Report of ML-1 Reactor Pressure Vessel Lower Cap Seal Weld Repair," transmitted with sr-1114, 10 September 1965.
5. AN-AGCR-921, "Design Report, ML-1 Shield Solution Gas Removal System," transmitted with sr-1100, 28 July 1965.
6. AN-AGCR-936, "Test Report on the ML-1 5% Boric Acid Shield Solution Experiment," transmitted with sr-1128, 14 October 1965.
7. AN-AGCR-934, "ML-1 Pressure Vessel Disassembly and Inspection Program," transmitted with sr-1116, 14 September 1965.
8. AN-IDOP-34, "Final Control Rod Calibration, Shield Water and Tungsten Baffle Reactivity Worth for ML-1," May 1961.
9. AN-AGCR-937, "ML-1 Fuel Element Temperatures and Stresses During ML-1/GCRF Transient Operation," transmitted with sr-1128, 14 October 1965.
10. AN-AGCR-939, "ML-1 Fuel Element and Reactivity Shims Inspection, 13-15 September 1965," transmitted with sr-1125, 11 October 1965.
11. AN-AGCR-958, "ML-1-IM Design Report," transmitted with sr-1144, 21 December 1965.

12. AN-AGCR-931, "ML-1 Cycle Optimization," transmitted with sr-1113, 8 September 1965.
13. AN-AGCR-844, "ML-1 System Performance Characteristics," transmitted with sr-1025, 12 April 1965.
14. Component Aerodynamic Designs for CSN-1A(M) Turbine-Compressor Set, R 65 FPD 207, General Electric Company, Cincinnati, Ohio.
15. AN-AGCR-935, "Advanced Recuperator Study - March 1965," transmitted with sr-1004, 19 March 1965.
16. AN-AGCR-928, "ML-1 Systems Influence Coefficients," transmitted with sr-1115, 13 September 1965.
17. AN-AGCR-610, "ML-1 Reference System Analysis," transmitted with sr-559, 1 November 1963.
18. AN-AGCR-944, "ML-1/GCRF Analog Model for Scram Response Time Analysis," transmitted with sr-1132, 27 October 1965.
19. AN-AGCR-961, "ML-1/GCRF Scram Circuit Response Study," transmitted with sr-1145, 30 December 1965.
20. AN-AGCR-911, "Conceptual Design Report Advanced ML-1 Pressure Vessel," transmitted with sr-1103, 30 July 1965.
21. AN-AGCR-910, "Summary Report - ML-1 Advanced Pressure Vessel Development," transmitted with sr-1098, 26 July 1965.
22. AN-AGCR-938, "ML-1 Precooler Test Loop Design Report," transmitted with sr-1124, 1 October 1965.
23. AN-AGCR-933, "Design Report, Alternator Test Facility," transmitted with sr-1119, 22 September 1965.
24. AN-AGCR-964, "CSN Lubrication System Design Report," transmitted with sr-1146, 6 January 1966.
25. AN-AGCR-941, "Start System Description Test," transmitted with sr-1128, 14 October 1965.
26. AN-AGCR-918, "A Comparative Analysis of the IB-17R-3 Test Element Leak," transmitted with sr-1108, 11 August 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_2 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 a 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 switchgear, 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
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

Report No. IDO-28653

<u>Document No.</u>	<u>Title</u>	<u>Classification</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, 22 February 1961</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
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

<u>Document No.</u>	<u>Title</u>	<u>Classification</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
IDO-28641	<u>AGCRSP Quarterly Progress Report, 1 October Through 31 December 1964, 15 February 1965</u>	U
IDO-28645	<u>AGCRSP Quarterly Progress Report, 1 January Through 31 March 1965, 14 May 1965</u>	U
IDO-28649	<u>AGCRSP Quarterly Progress Report, 1 April Through 30 June 1965, 13 August 1965</u>	U

APPENDIX BML-1 PLANT CHARACTERISTICS1. 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	6,500 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 ²
Maximum heat flux	140,000 Btu/hr/ft ²
Average heat flux	78,200 Btu/hr/ft ²
Maximum to average heat flux ratio	
Axial	1.41
Radial	1.27
Maximum fuel center temperature (including hot spot factors)	2160°F (BeO-UO ₂) 2650°F (UO ₂) ²
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)	1.9×10^{12} neut/cm ² -sec
Average fast neutron flux (fuel)	1.7×10^{13} neut/cm ² -sec
Maximum to average thermal flux ratio	3.9
Hydrogen to U-235 atom ratio	40

Core buckling*	0.0047 cm ⁻²
Fermi age	60 cm ²
Square of thermal diffusion length, L ²	2.05 cm ²
Thermal utilization, f*	0.66
Infinite multiplication factor, k	
Without shims	1.54
With shims*	1.36
Neutron lifetime	1.9 x 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	3.5% in 10,000 hr
Maximum	6.5%
Prompt temperature coefficient, $\Delta K/K$ -°C	
at 0°C	+ 0.3 x 10 ⁻⁶
at 90°C	-0.5 x 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°F (max.)

*Indicates item revised from previous tabulation.

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 ₂ O; 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 x 10 ⁵ Btu/hr
Maximum power density	360 Btu/hr-in. ³

6. BIOLOGICAL SHIELDING

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

7. CORE (EXCLUDING REFLECTOR)

Diameter*	21.54 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 ₂
Outer 12 pins	31 vol% UO ₂ , 93% enriched U-235, 69 vol% BeO
Core composition	
Materials	<u>Volume %</u>
UO ₂	4.3
BeO	3.3
Stainless steel	3.6

*Indicates item revised from previous tabulation.

Materials (continued)	<u>Volume %</u>
Hastelloy X	7.0
H ₂ 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 ₂ (outer pins); UO ₂ (inner pins)
Number of pins per element*	19 (12 with 70 wt% UO ₂ in BeO; 6 with 100 wt% UO ₂ ; 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	0.5% 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 9 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

*Indicates item revised from previous tabulation.

Reactivity worth of control elements:

Safety and shim rods (each)*	1.58% $\Delta K/K$
Regulating rods*	0.45% $\Delta K/K$
Total (6 rods)*	7.4% $\Delta K/K$

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	$10^5 \times 10^6$ Btu/hr

11. REACTOR WORKING FLUID FLOW

Working fluid	99.5 vol% N_2 + 0.5 vol% O_2
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 ΔP	15 psi
Reactor ΔP	22 psi

*Indicates item revised from previous tabulation.

12. POWER CYCLE

Type	Brayton cycle with regeneration
Total volume of working fluid system	120 ft ³
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

	<u>Stratos T-C Set</u>	<u>Clark T-C Set</u>
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 ϕ , 60 cycle
Voltage	2400/1600 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.
Material	
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 ΔP	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 ΔP	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

L E G A L N O T I C E

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or,

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

- - - - -

NOTE: This is the 110th in a series of Research and Development Progress Reports. The preceding reports in this series are Nos. IDO-28501, 28502, 28504, 28515, 28516, 28517, 28518, 28519, 28520, 28521, 28523, 28524, 28525, 28526, 28527, 28528, 28529, 28531, 28532, 28533, 28535, 28536, 28538, 28541, 28542, 28543, 28544, 28545, 28546, 28548, 28549, 28551, 28553, 28554, 28556, 28557, 28558, 28559, 28562, 28563, 28565, 28566, 28567, 28568, 28569, 28570, 28571, 28572, 28573, 28574, 28575, 28576, 28577, 28579, 28580, 28581, 28583, 28585, 28586, 28587, 28588, 28589, 28590, 28592, 28593, 28594, 28596, 28599, 28601, 28602, 28603, 28604, 28606, 28607, 28608, 28610, 28611, 28612, 28613, 28614, 28615, 28617, 28618, 28619, 28620, 28621, 28622, 28623, 28624, 28626, 28627, 28629, 28630, 28632, 28633, 28635, 28636, 28637, 28638, 28641, 28642, 28643, 28644, 28645, 28648, 28647, 28648, and 28649. The information herein is regarded as preliminary and subject to further checking, verification and analysis.

AEROJET-GENERAL NUCLEONICS