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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRAM FINAL REPORT

March 1966

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

AEROJET-GENERAL NUCLEONICS

SAN RAMON, CALIFORNIA

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRAM FINAL REPORT

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRAM FINAL REPORT

ABSTRACT

The technical accomplishments under the Army Gas-Cooled Reactor Systems Program are summarized. A brief chronology of the evolution of the program, including the reactor development activity, the design, fabrication and subsequent testing of the ML-1 nuclear power plant and related technological and engineering activities are discussed. The summary is supplemented by a comprehensive bibliography which includes a brief abstract of each referenced report or document.

JUL 15 1966

8 July 1966

TO: HOLDERS OF IDO-28655 *encl.*
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SUBJECT: ERRATA SHEET - REPORT NO. IDO-28655

*See list
7-18-66*

Please make the following corrections to the subject report:

Page 5B - Change reference to "Report IDO-17160"
to "Report IDO-17190"

Page 165 - Change reference in footnote from
"IDO-17159" to "IDO-17190"

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRAM FINAL REPORT

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

PROGRAM FINAL REPORT

I. INTRODUCTION

A. BACKGROUND

The activity which ultimately evolved into the Army Gas-Cooled Reactor Systems Program (AGCRSP) was initiated in 1954 by a study at the Oak Ridge School of Reactor Technology (ORSORT) which defined the basic concepts of a mobile gas-cooled nuclear power plant suitable for deployment to support armed forces in the field. In 1955, the Sanderson-Porter Co. conducted a feasibility evaluation of the ORSORT concepts and concluded that the development of such a plant was practical. A joint Army-U. S. Atomic Energy Commission (USAEC) program was formally initiated in 1956 as the AGCRSP; the former agency was assigned responsibility for the development of the power conversion system while the latter accepted responsibility for the development of the reactor and controls. The Army program involved the procurement of a gas turbine-compressor set, heat exchangers and other power conversion components and the construction of a test facility at Ft. Belvoir, Virginia, (GTTF) to evaluate the performance of and complete the development of the power conversion system.

The USAEC issued a request for proposals for the design, development, fabrication and testing of a compact gas-cooled reactor with the associated test facility and controls (GCRE). Aerojet-General Corporation (AGC) was selected as the contractor for this activity and cognizance for the work was assigned to the Commission's Idaho Operations Office. A contract (AT(10-1)-880) was negotiated between the USAEC and AGC in November 1956 and work on the Gas-Cooled Reactor Experiment was initiated.

In June 1959, the Army and AEC developed the performance specifications (Military Characteristics) for the mobile nuclear power plant, assigned systems contractor responsibility to Aerojet, and directed that work proceed immediately to design, develop and fabricate a prototype power plant (ML-1) based, insofar as was possible, on the information developed at the GCRE and GTTF. The goal of the program, as specified by the Government in June 1959, was the development of procurement specifications for the production version of the plant (based on demonstration operation of the ML-1) by May 1962.

B. REPORT SCOPE AND ORGANIZATION

This summary report of AGCRSP activity has been prepared to document, primarily, the technical accomplishments under the program rather than a history of the program. As such, the presentation does not include a comprehensive discussion of programmatic matters. Major technical accomplishments are properly placed in time, although not necessarily presented in a chronological manner, to facilitate reference to the appropriate technical progress reports (see discussion below).

The report is organized in nine major sections identified with Roman numerals. Major topics in each section are identified by capital letters; details are presented under numerical and alphabetic identification in the normal fashion.

References are provided throughout the report to topical reports published during the course of the work. These references fall into two categories; those given standard TID-4500 distribution and, as a consequence, available in most Government and Government Contractors document control centers and those given only limited, in-contract, distribution. The latter are available in the AGCRSP Technical Data File at the USAEC Idaho Operations Office. Lists of references (bibliographies) are presented at the end of each significant section of the report. These bibliographies, in addition to providing a complete identification of the reference document, also include a brief abstract of the material in the document. Periodic technical progress reports (monthly, quarterly and semiannual) are not referenced in text; a bibliography of these reports is presented in Appendix A.

In consideration of the fact that the design, development, fabrication and test operation of the ML-1 power plant constituted the major activity under the AGCRSP, the majority of this report is devoted to that subject. Brief discussions of other aspects of the program, as indicated in the Table of Contents, are presented to provide a comprehensive summary of the work.

C. CLASSIFICATION

The GCRE and AGCR programs were, at their inception, classified but the entire AGCRS Program was declassified in the spring of 1959. At that time all classified documents generated prior to the declassification action were reviewed and formal action was taken to declassify those which, it was felt, would be used frequently in the continuing work. All documents which are referred to in this report have been declassified.

II. SUMMARY

The Army Gas-Cooled Reactor Systems Program (AGCRSP) was conducted, in major part, under Contract AT(10-1)-880 between the Idaho Operations Office of the USAEC and the Aerojet-General Corporation. Technical cognizance for the work was vested, largely, in Aerojet-General Nucleonics, a subsidiary (and, later, a division) of Aerojet-General Corporation.

The work began in the fall of 1956 with the initiation of the program to develop a compact gas-cooled reactor suitable for use as the heat source in a mobile, nuclear power plant intended for use in the field by military personnel. The performance specifications for the power plant were finalized by the DOD in mid-1959, at which time the design, development and fabrication of a prototype plant was initiated. The reactor skid and control cab were delivered to the National Reactor Testing Station (NRTS) for testing in the spring of 1961. Problems with the development of compact, high-performance heat exchangers and with the development of the gas turbine and compressor unit delayed the completion of the power conversion equipment until mid-1962.

Testing of the prototype ML-1 power plant at the NRTS proceeded intermittently from September 1962 until August 1965. The work was interrupted by minor (but difficult to repair) failures in the reactor, the requirement to design and install additional instrumentation and support systems (required to assure safe operation of the developmental plant or to provide detailed data for the evaluation of the performance of plant components), and by sub-standard performance of the t-c set (since no suitable spare unit existed, testing was interrupted for each t-c set modification).

In October 1965, the decision was made by the USAEC to terminate the AGCRSP by 30 June 1966 (in fiscal year 1966). At that time an unresolved, and potentially significant, leak existed in the ML-1 reactor pressure vessel and power conversion system testing had been discontinued for several months while a program for the development of improved components was undertaken. The development of a 10,000 hr fuel element was considered essentially complete although some problems with the prefabrication inspection of components remained to be resolved. The development of the instrumentation and control systems had proceeded by an iterative process to the point where it was considered that the work was essentially complete.

At termination, it was considered that, in the absence of a specific mission for the ML-1 or of a potential exclusive application of the gas-cooled technology, the completion of the development of the ML-1 was not justified.

The activity completed under the AGCRSP is considered to have made significant contributions in the following technological areas:

- The analysis and understanding of the neutronic behavior of highly-heterogeneous, water-moderated, gas-cooled, compact nuclear reactors.
- The analysis, design and fabrication of pin-type, ceramic-fueled, gas-cooled fuel elements.
- The analysis and understanding of closed-cycle gas power systems, including the complex interactions between components and the characteristics required of the control system for a reactor-heated gas power plant.
- The analysis, design, fabrication and performance evaluation of closed-cycle, gas turbine-compressor sets.
- The design and fabrication of high-performance, gas-to-gas, compact heat exchangers.
- The analysis and design of high-performance, composite radiological shields, with attenuation characteristics which satisfy stringent dose rate criteria both during reactor operation and following shutdown.
- The physical properties of and irradiation stability of Hastelloy X and high-beryllium content UO_2 -BeO ceramic fuel bodies.

III. CONTRACT CHRONOLOGY

Contract AT(10-1)-880, as executed by the Idaho Operations Office of the USAEC and AGC in November 1956, authorized a literature search to define the current status of gas-cooled reactor technology, the preliminary design of the GCRF (including all test support equipment), the research and development required for the design of the GCRE-I reactor (including a critical experiment), the fabrication and installation of the GCRE-I reactor (including a safety analysis), and the testing and operation of the GCRE-I in the GCRF.

The modification to this contract, effective in November 1957, provided more explicit definition of portions of the above scope and authorized the conduct of studies to develop a master plan for the Army Reactor Experimental Area at the NRTS during the 1957-58 contract year (1 November 1957 - 30 September 1958).

The contract modification for 1958-59 designated the overall program as the Gas-Cooled Systems Program and directed Aerojet to coordinate the power conversion and reactor development activities (the USAEC had provided this coordination up to this date) and to initiate studies and preliminary designs for the GCSE. The initiation of GCRE-II work was also authorized.

The contract for 1959-60 redesignated the GCSP as the AGCRSP and directed Aerojet to serve as systems contractor. The continuation of the GCRE activity was authorized. The GCSE work was canceled in favor of the initiation of the ML-1 program; under this part of the contract, AGC was authorized to initiate all aspects of ML-1 design, development and construction, as well as the preparation of production drawings and specifications for the ML-1A. A modification made late in the contract year authorized the initiation of design and development of the ML-1 second core loading.

In the 1960-61 contract, the only significant change in the work scope was the authorization to initiate test operations at the ML-1 test site. At this time, however, the format of the contract was altered to provide for the presentation of detailed technical scope as an exhibit to the basic contract. This exhibit was designated as the program letter.

The contract for 1961-62 recognized and authorized the requirement for continued engineering support of the ML-1 test operation, and authorized the

initiation of an improvements program and evaluations of advanced (ML-2) systems. Modifications made during this contract year authorized open-cycle testing and subsequent rework of the TCS-670 t-c set, the design, fabrication and installation of the ML-1 emergency cooling system, the deoxygenation system and ultimate shutdown device, and directed the suspension of GCRE-I test operations.

The only new scope authorized in the 1962-63 contract covered several specific items under the improvements program, including a study of the modifications necessary to convert the CSN-1A t-c set for air cycle operation.

Beginning with the 1963-64 contract, the inclusion of specific items of scope was eliminated and a general research and development contract was written. Quarterly estimates of costs for a mutually agreeable scope of work served as the basis for fee negotiation. The basic program work-scope was defined in the annual Project Proposals (Form 189).

Funds in the amount of \$4.1 million were requested from Congress by the USAEC and authorized by the JCAE for continuation of the AGCRSP during FY 1966. The House Appropriations Committee appropriated only \$2 million for the program, following which the Senate Appropriations Committee recommended appropriation of the full amount. In joint conference, the Senate withdrew the recommendation for the \$4.1 million appropriation and, in effect, agreed to the amount appropriated by the House. Following this action, in consideration of the status of the ML-1 development and the absence of intent on the part of the Army to procure ML-1-type systems in the near future, the USAEC decided to terminate the program in FY66 with the funds authorized. Aerojet was officially notified by the USAEC Idaho Operations Office to initiate termination activities as of 1 November 1965.

As of the close of business for Government fiscal year 1965, the total costs under Contract AT(10-1)-880 were as summarized in Table 1. The estimated costs for GFY 1966 are approximately \$1,600,000.

TABLE 1 - SUMMARY OF CONTRACT AT(10-1)-880 COSTS THROUGH 30 JUNE 1965

<u>Activity</u>	<u>Costs, Millions of \$</u>
<u>GCRE Program</u>	
Research and Development	4.4
Fuel Fabrication	1.0
Test Operations	2.5
Reactor Experiment Fabrication	<u>4.4</u>
Subtotal	12.3
<u>ML-1 Program</u>	
Research and Development	19.0
Fuel Fabrication	1.8
Test Operations	4.0
Irradiation Charges (In-Pile Test Loops)	.7
In-Pile Loop Operation	3.9
Capital Equipment	1.1
Construction	<u>5.8</u>
Subtotal	36.3
<u>ML-1A Program</u>	1.2
TOTAL	49.8

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IV. GCRE PROGRAM

A. REACTOR CONCEPT SELECTION

The initial task under Contract AT(10-1)-880 between the USAEC-ID and Aerojet was the performance of a series of studies of reactor concepts suitable for use in the mobile nuclear power plant which was ultimately designated ML-1. This work was performed between November 1956 and March 1957.

Preliminary evaluations of several reactor systems were completed as shown in Table 2.^{(1)(2)(3)(4)(5)(6)*} A summary of the techniques used to calculate the neutronic characteristics of each system was published⁽⁷⁾ as well as the summary of the evaluation of candidate coolant gases.⁽⁸⁾

In March 1957, the USAEC and Aerojet selected the water-moderated concept for the GCRE-I reactor. This decision was based primarily on the fact that the characteristics of the selected concept were acceptable under the design criteria and that a minimum of extrapolation of existing technology would be required during the design and development of such a reactor.

Selection of the test facility concept was also undertaken during this period; a pool-type facility was selected because of the versatility offered by this concept.

*Superscripts in text identify references cited at the end of each significant section of the report.

TABLE 2 - REACTOR CONCEPTS EVALUATED FOR GCRE-I

	<u>Concept Characterization</u>			<u>Configuration</u>	<u>Reference</u>
	<u>Fuel</u>	<u>Moderator</u>	<u>Coolant</u>		
1.	UO ₂ ⁽¹⁾	H ₂ O	Oxygenated ⁽²⁾ nitrogen	Heterogeneous	IDO-28500
2.	UO ₂ ⁽¹⁾	Be	Oxygenated ⁽²⁾ nitrogen	Heterogeneous	IDO-28503
3.	UO ₂ ⁽¹⁾	BeO	Oxygenated ⁽²⁾ nitrogen	Heterogeneous	IDO-28503
4.	UO ₂ ⁽¹⁾	ZrH	Oxygenated ⁽²⁾ nitrogen	Heterogeneous	IDO-28507
5.	UC ⁽¹⁾	Graphite	Oxygenated ⁽²⁾ nitrogen	Homogeneous	IDO-28508
6.	UO ₂ ⁽¹⁾	ZrH	Helium or nitrogen	Homogeneous	IDO-28509
7.	UO ₂ ⁽¹⁾	Be	Nitrogen	Homogeneous	IDO-28510

(1) Fully enriched

(2) ~ 0.5 vol% O₂

REACTOR CONCEPT SELECTION BIBLIOGRAPHY

1. IDO-28500*, "Gas-Cooled Reactor Experiment Preliminary Design," 10 March 1957

The preliminary design of a water-moderated, gas-cooled reactor system is presented along with a discussion of the development aspects of the concept.

2. IDO-28503*, "Preliminary Evaluation of Beryllium and Beryllium Oxide Heterogeneous Reactors," 19 December 1957
3. IDO-28507*, "Preliminary Design of Heterogeneous, Zirconium-Hydride-Moderated Reactor Systems," 31 October 1958
4. IDO-28508*, "Preliminary Design of a Homogeneous, Graphite-Moderated Reactor," June 1958
5. IDO-28509*, "Preliminary Design of a Homogeneous, Zirconium Hydride-Moderated Reactor," September 1958
6. IDO-28510*, "Preliminary Design of a Homogeneous Beryllium Reactor," June 1958

The above documents (2 through 6) present a preliminary design for each of the indicated reactor systems and an evaluation of the advantages and disadvantages of the system in terms of performance and development requirements.

7. IDO-28511*, "Reactor Systems Neutronics," 7 November 1958

The technique used to analyze the neutronic characteristics of the various reactor systems studied (Ref. 1 through 6) in the preliminary phase of the GCRE evaluations are presented.

8. IDO-28512*, "Gas-Cooled Reactor Experiment - Heat Transfer Considerations," 24 November 1958

The chemical, thermodynamic and activation characteristics of several candidate (for GCRE-I reactor coolant) gases are evaluated. The methods of evaluation are discussed and it is concluded that helium and nitrogen are the most suitable.

*Indicated documents with TID-4500 or M-3679 distribution.

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B. GCRE-I REACTOR DESIGN AND FABRICATION

The development and design of the GCRE-I reactor was initiated in the spring of 1957. The major elements of the work involved the development of the fuel elements⁽¹⁾ (including analyses⁽²⁾), a critical experiment⁽³⁾, an in-pile test program⁽⁴⁾, fabrication development⁽⁵⁾ and design⁽⁶⁾, of the reactor (including a full scale engineering test of the core⁽⁷⁾ and development of the control rods⁽⁸⁾), and of the supporting instrumentation system⁽⁹⁾. In addition, dynamic systems analyses were performed to provide data for the design of the test facility and a detailed safety analysis was completed.⁽¹⁰⁾

All design and development work was completed by mid-1959 and the reactor and instrumentation system were delivered to the NRTS for installation in the Gas-Cooled Reactor Facility (GCRF). The reactor consisted of an aluminum calandria with 3 in. thick tube sheets and 73 pressure tubes. Plenums were provided at each end of the calandria to distribute and collect the coolant gas and a radial lead neutron reflector surrounded the core (Fig. 1). Control blades entered the core region horizontally in the space between rows of calandria tubes. The entire assembly, as shown in Figure 2, was immersed in the GCRF pool; the demineralized water in the pool served as the moderator and primary radiological shield. The reactor characteristics are summarized in Table 3.

TABLE 3 - GCRE-I REACTOR DESIGN CHARACTERISTICS

	<u>IZ Core</u>	<u>IB-2L Core</u>
1. Operating Characteristics		
a. Power, thermal, maximum, Mw	2.2	2.08
b. Coolant inlet temperature, °F	800	800
c. Coolant outlet temperature, °F	1200	1185
d. Maximum fuel temperature (including hot spots), °F	1650	1750
2. Nuclear Characteristics		
a. Active section diameter, in.	24	24
b. Active section length, in.	28	24
c. Number of fuel elements (operating core loading)	61	56
d. Neutron flux, thermal, avg, n/cm ² -sec	3×10^{12}	3.5×10^{12}
e. Neutron flux, fast, avg, n/cm ² -sec	1.9×10^{13}	1.3×10^{13}
f. Critical mass, cold clean, kg U-235	13.8	36.3

FIGURE 1. GCRE-I REACTOR

1. REACTOR SUPPORT STAND
2. REACTOR (INLET PLENUM)
3. LEAD REFLECTOR
4. ROD GUIDE ASSEMBLY
5. STAND SUPPORT RODS
6. STAND SAFETY SUPPORTS
7. INLET COOLANT GAS DUCT
8. OUTLET COOLANT GAS DUCT
9. ROD ACTUATOR (TYP)
10. ROD ABSORBER BLADE (TYP)
11. NEUTRON DETECTOR (TYP)

15

REACTOR ASSEMBLY

REACTOR IN REACTOR POOL
(Top cover removed and pool
partially filled with water)

FIGURE 2. GCRE-I REACTOR ASSEMBLY

The fuel elements were suspended in the calandria tubes. Two types of fuel element were designed and fabricated for testing in the GCRE-I; the plate-type (IZ) element (Table 4 and Figure 3) and the pin-type (IB-2L) which was prototype for the ML-1 reactor (Table 4 and Figure 4).

The instrumentation and control system designed for use with the GCRE-I (Figure 5) provided for the operation of the reactor and supporting test equipment from the remotely located control room as well as for the accumulation of all pertinent data required for performance evaluation.

TABLE 4 - GCRE-I FUEL ELEMENT DESIGN DATA

	<u>IZ Element</u>	<u>IB-2L Element</u>
Total length, in.	44	44
Active length, in.	28	22
Outside diameter, in.	1.72	1.740
Fuel composition	Fully enriched UO_2	Inner pins; fully enriched UO_2 cold pressed sintered pellets. Outer pins; fully enriched UO_2 in BeO , cold pressed sintered pellets.
Fueled section	UO_2 dispersion in AISI Type 316 stainless steel	--
Fuel loading, wt%	30	60 in outer pins, 100 in inner pins
Fuel loading, gm (nominal)	315.0	800
Pellet diameter, in.	--	0.175
Length, in.	--	0.230
Cladding		
Material	AISI Type 318 stainless steel	Hastelloy X (< 1% Co)
Wall thickness, in.	0.006	0.030
OD, in.	--	0.241
Central spine		
Material	AISI Type 316 stainless steel	--
OD, in.	0.499	--
Wall thickness, in.	0.032	--
Inner liner		
Material	AISI Type 316 stainless steel	AISI Type 316 stainless steel
OD, in.	1.446	1.446
Wall thickness, in.	0.010	0.010
Insulation		
Material	Thermoflex	Thermoflex
Thickness, in.	0.215	0.28
Outer liner		
Material	Reactor grade zirconium	AISI Type 316 stainless steel
OD, in.	1.720	1.740
Wall thickness, in.	0.030	0.010
Burnable poison		
Location	Inside of outer liner, center of active length	Inside outer liner; bottom half of active length
Material	Samarium	Cadmium
Quantity, gm	0.163	--
Configuration	0.005 in. thick foil, 6 in. long	11 in. long foil
Spacer wire		
Material	--	Hastelloy X
Diameter, in.	--	0.040

FIGURE 3. GCRE-I PLATE-TYPE FUEL ELEMENT

FIGURE 4. GCRE-I PIN-TYPE FUEL ELEMENT

LEGEND FOR FIGURE 5 - GCRE CONTROL ROOM LAYOUT

RACK	PANEL	INSTRUMENT(S) OR CONTROL(S) FUNCTION	RACK	PANEL	INSTRUMENT(S) OR CONTROL(S) FUNCTION
1	---	Control Relays	10	F	Safety Rod Housing Pressure Indicator
2	---	Control Relays	11	A	Annunciator Panel
3	A	Wind Speed and Direction Indicators	11	B	Main Loop Flow Indicator
3	B	Main Compressor Controls	11	C	Reactor Outlet Temperature Indicator
3	C	Cooling Water and Lube Oil Pump Controls	12	A	Annunciator Panel
3	D	Main Compressor Controls	12	B	Reactor ΔP Indicator
3	E	Heater Controls	12	C	Reactor Inlet Pressure Indicator
3	F	Remote Operable Valve Controls	13	A	Annunciator Panel
4	A	Shutdown Cooling Loop Temperature Indicator	13	B	Reactor ΔT Indicator
4	B	Shutdown Cooling Loop Flow Indicator	13	C	Heater Outlet Temperature Indicator
4	C	Fire Pump and Test Building Exhaust Fan Controls	13	D	Main Compressor Outlet Temperature Indicator
4	D	Well, Sump and Wash Pump Controls	13	E	Compressor Inlet and Outlet Pressure Indicator
4	E	Stack Blower, Pool Water Pump and Wash Pump Controls	14	A	Gas-to-Water Heat Exchanger Outlet Gas Temperature Indicator
4	F	Reactor Flooding Water Level Indicator	14	B	Gas-to-Water Heat Exchanger Outlet Gas Temperature Controller
4	G	Water-in-Rod Housing Indicators	14	C	Compressor Seal Bleed Flow Controller
5	A	Annunciator Panel	14	U	Compressor Seal Pressure Indicator
5	B	Water-to-Water Heat Exchanger Bypass Control (Temperature)	15	A	Btu Recorder
5	C	Cooling Tower Bypass Control (Temperature)	15	B	Pitot-Static Flow Meter Controls
5	D	Pool Water Heat Exchanger Bypass Control (Temperature)	15	C	Gas-to-Water Heat Exchanger Inlet Water Temperature Indicator
5	E	Heater Bypass (Reactor Inlet Temperature) Control	15	D	Gas-to-Water Heat Exchanger Outlet Water Temperature Indicator
5	F	Make-up Gas Flow Indicator	15	E	Demineralized Water Flow Indicator
6	A	Annunciator Panel	15	F & G	Power Supplies
6	B	Main Loop Gas Dew-Point Indicator	16	A	Fission Product Sampler Count Rate Indicator
6	C	Main Loop Oxygen Concentration Indicator	16	B & C	Fission Product Monitor System
6	D	Filter/Dryer Loop Flow Indicator	16	D	Loop Purge Flow Control
6	E	Waste Gas Holder Level Indicator	16	E	Power Supply
7	A	Log Count Rate Recorder	17	A, B, & C	Fuel Element Temperature Indicators
7	B	Scaler	17	D, E, & F	Remote Area Radiation Monitors
7	C	Log Count Rate (Channel I) System	18	A & B	Fuel Element Temperature Indicators
7	D	Log Count Rate (Channel II) System	18	C	Recording Oscillograph
7	E & F	Linear Amplifier	18	D & E	Miscellaneous Temperature Meters and Selector Switches
8	A	Period Recorder	19	A	Fuel Element Outlet Gas Temperature Indicator
8	B	Log-N Period (Channel III) System	19	B	Fuel Element and Outlet Gas Thermocouple Patch Panel
8	C	Log-N Period (Channel IV) System	C-1	----	Intercom and Emergency Signals
8	D & E	Period Amplifier (Channels III and IV)	C-2	----	Shutdown and Emergency Cooling Controls
8	F & G	Compensated Ion Chamber Power Supply	C-3	----	Interlock Indicators
9	A	Log N Recorder	C-4	Upper	Nuclear Instrument Indicators
9	B	Linear Power Recorder (Channel IX)	C-4	Lower	Safety and Shutdown Rod Controls
9	C	Linear Amplifier (Channel IX)	C-5	----	Main Console Power Switch and Control Circuit Indicators
9	D	Micro-micro Ammeter (Channel IX)	C-6	Upper	Nuclear Instrument Indicators
9	E	Compensated Ion Chamber Power Supply	C-6	Lower	Shim, Setback and Fine Rod Controls
10	A	Power Level Recorder	C-7	----	Log Count Rate Detector Chamber Shield Operating Controls
10	B	Safety Channel Amplifier and Trip Circuit (Channels V and VI)	C-8	----	Main Coolant Loop Parameter Controls and Instruments
10	C	Safety Channel Amplifier and Trip Circuit (Channels VII and VIII)	C-9	----	Temperature Display Oscillograph
10	D & E	Scram Timer

11.90-66-480

FIGURE 5. GCRE-I CONTROL ROOM

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GCRE-I REACTOR DESIGN AND FABRICATION BIBLIOGRAPHY

1. Fuel Element Development

- a. AGN TM-301, "GCRE Fuel Element Fabrication Proposal,"
17 July 1957

Outlines the program for the development of the concentric cylinder-type fuel element for the GCRE-I, including the evaluation of the gas corrosion of cladding materials, fuel element component forming, joining and assembly techniques, the thermal evaluation of fuel element mockups in the laboratory and in in-pile tests and the development of fabrication quality control standards.

- b. AGN TM-308, "GCRE Fuel Element Interim Report," 30 August 1957
- c. AGN TM-310*, "GCRE Fuel Element Task Interim Report,"
2 September 1957

The above reports document the interim results of the several fuel element developmental programs. Of major significance is the justification presented for selection of the concentric tube fuel element concept.

2. Fuel Element Analysis

- a. IDO-28534*, "Heat Transfer in an Annulus with Asymmetric Heating," February 1959

Reports the results of studies of the validity of using tube flow correlations for the calculation of pressure drop and heat transfer characteristics in thin annuli; of the validity of using Stein's heat flux asymmetry correction for parallel planes to calculate heat transfer in asymmetrically heated thin annuli; and of the effects of eccentricity on pressure drop, heat transfer and temperature distribution in concentric ring fuel elements.

- b. AGN TM-317, "Computing Routine for No-Flow Problem Distribution,"
17 February 1958

Presents an IBM 650 computer program for the calculation of transient temperatures in the GCRE-I reactor.

- c. AGN TM-318, "Computer Program for Calculation of Steady State Fuel Element Temperatures," January 1958

Presents an IBM 650 computer program for the calculation of steady state temperatures in the GCRE-I reactor.

- d. AGN TM-327, "GCRE Description of Heat Transfer Experimental Program and Presentation of Initial Results," 5 June 1958

The heat transfer experimental program to determine the thermodynamic characteristics of the GCRE-I fuel element is discussed along with the preliminary results from the program.

- e. AGN TM-332, "A Computing Program for Calculation of Temperatures in a Two, Three or Four Plate Reactor Fuel Element," 21 August 1958

Presents a generalized IBM 650 program for the computation of temperatures in two, three and four concentric tube fuel elements.

3. Critical Experiment

- a. AGN TM-303, "Facility Requirements and Experimental Program for GCRE," July 1957

Documents the design criteria and proposed experimental program for the GCRE-I critical experiment.

- b. AGN TM-323, "GCRE Critical Experiment," 6 May 1958

Describes the design (by AGN) of the GCRE-I critical experiment, and the conduct (by BMI) of the experimental program and the interpretation of the program results.

4. In-Pile Test Program

- a. AGN TM-322, "GCRE Description of Mark II In-Pile Experiment," 6 May 1958

A description of the Mark II (later designated I-1T) test element and the in-pile test facility in the Battelle Research Reactor, and a discussion of the projected test and evaluation program are presented.

- b. IDO-28539*, "In-Pile Loop Tests of GCRE Elements (I-1T and I-1'T)," July 1959

A description of the I-1T test (1180 hrs at an average thermal flux of 2.7×10^{12} n/cm²-sec) of the I-1'T test (393 hrs of thermal cycling) in the BRR-GCR test loop is presented along with the results of the post-irradiation examination of the test elements. No distortion, corrosion or other deleterious effects were observed.

- c. AGN TM-345, "AGCRSP Description of the IB-1T Fuel Element In-Pile Experiment," 29 January 1959

The objectives of this test of the first pin-type element are discussed, a description of the test element and test loop are

presented, the proposed test is outlined and a preliminary safety analysis is included.

- d. IDO-28552*, "Interim Report - In-Pile Tests of GCRE-IB Test Elements," December 1960

The results of short term (less than 500 hr) irradiations of two prototype ceramic-fueled, pin-type fuel elements in the BRR-GCR loop are presented.

- e. AGN TM-334, "GCSP In-Pile Loop Testing Program," September 1958

The results of a survey of existing gas-cooled in-pile test loops available for development of fuel elements for the Gas-Cooled Systems Program is presented. It is recommended that a new loop be fabricated and installed in the General Electric Test Reactor at Vallecitos, California.

5. Fabrication Development

- a. IDO-28547*, "The Fabrication of Modified GCRE Fuel Plates Using Spherical UO_2 Dispersion," 15 March 1960

The results of a program to evaluate the use of spherical UO_2 (instead of hydrothermally-grown UO_2) in GCRE fuel plates are presented. It is concluded that the use of spherical material results in an improved plate fabrication yield, a more uniform fuel dispersion, less fractioning of UO_2 particles and improved plate bending ductility.

- b. AGN TM-343, "Evaluation of Extruded Fuel Elements," January 1959

The results of a program to evaluate the feasibility of extruding full cylindrical fuel plates for GCRE elements (as opposed to roll-forming 120° sectors of cylindrical fuel plates) are discussed. It is concluded that the extrusion technique would require about one year of development before acceptable parts could be produced.

6. Fuel Element Design

- a. AGN TM-309, "GCRE Fuel Element Thermocouple Proposal," 13 September 1957

The proposed program for development and installation of thermocouples to monitor temperatures in selected GCRE fuel elements is discussed.

- b. AGN TM-311, "Preliminary Fuel Plate Specifications," December 1957

The specifications for fabrication and inspection of the GCRE fuel plates are presented.

- c. AGN TM-333, "Trip Report for July 10-25 and August 5, 1958," September 1958 (In two sections)

The results of a comprehensive survey of the state-of-the-art of ceramic-fueled, pin-type fuel elements is presented. This information led to the decision to use this fuel element concept for the IB-2L core for the GCRE.

7. Engineering Core Test

- a. AGN TM-302, "Proposal for GCRE Core Test," 19 July 1957

The objectives, design and test program for the full scale GCRE reactor core engineering test are presented.

- b. AGN TM-344, "AGCRSP Engineering Core Test," 30 January 1959

The results of the engineering core test are presented.

- c. AGN TM-377, "The Thermoelastic Analysis of the GCRE-I Tube Sheet and Plenum Chamber," May 1960
- d. AGN TM-384, "The Thermoelastic Analysis of the ML-1 Tube Sheet and Plenum Chamber," February 1961

The above reports present iterations of the development of analytical techniques for the estimation of stress distribution in the GCRE-I reactor calandria tube sheet and plenum along with results of the stress computations.

8. Control Rod Development

- a. AGN TM-307, "Reactor Control System for the Gas-Cooled Reactor Experiment," Undated

The proposed system of twelve horizontally-operating control rods for the GCRE-I reactor is presented and the developmental areas are identified.

9. Instrumentation System

- a. AGN TM-306, "Fission Product Monitoring System Experimental Mixer for Sampling Section," 23 August 1957
- b. AGN TM-312, "Summary of Experiments with Gas Sampling Mixers," December 1957

These documents outline and present the results of an experimental program to evaluate various configurations of mixers to be installed at the outlet end of a fuel element to assure that fission products in the outlet gas stream will be detected by a single, centrally-located sampling tube. It is concluded that a venturi design mixer will perform satisfactorily and will have the minimum pressure drop.

10. Safety Analysis

- a. AGN TM-300, "Preliminary Report of Health and Safety Aspects," July 1957

A preliminary evaluation of the health and safety aspects of GCRE-I operation are presented as an aid in the GCRE site selection.

- b. IDO-28506*, "GCRE-I Hazards Summary Report," December 1958
- c. IDO-28506*, Addendum I, "GCRE-I Hazards Summary Report - Addendum I," March 1959
- d. IDO-28506*, Addendum II, "GCRE-I Hazards Summary Report - Addendum II," February 1960
- e. IDO-28506*, Addendum III, "GCRE-I Hazards Summary Report - Addendum III," May 1960

The above listed documents summarize the analysis of the safety of the operation of the GCRE-I reactor in the GCRF at the NRTS. A brief description of the site, the reactor and test facility design and the projected mode of operation is included. Addendum I was published to document the response to certain questions raised by the USAEC; Addendum II was published to report the results of the safety analysis of the GCRE-I system after modification (substitution of pin-type fuel elements for the plate-type and replacement of the aluminum calandria with a stainless steel assembly); Addendum III was published to report the results of the safety analysis of minor modifications (primarily to the scram system) of GCRE-I equipment.

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C. GCRF DESIGN AND CONSTRUCTION

The design⁽¹⁾⁽²⁾ of the Gas-Cooled Reactor Facility (GCRF) was initiated by Aerojet in mid-1957 and construction began (under the supervision of the USAEC-ID) in mid-1958. The facility was located in the Army Reactor Area of the National Reactor Testing Station (NRTS) in Idaho.

The GCRF consisted of the Control and Test Building and several supporting and service buildings shown in Figure 6. The office and laboratory spaces and the test facility control room were provided in the one-story control building section while the high bay area housed the reactor pool and the reactor test equipment.

The major test equipment consisted of a gas circulation system with a compressor (blower), gas heater and heat exchangers to circulate coolant through the reactor. Excess heat was dissipated through a demineralized water cooling system to a cooling tower (Figure 7). The system was capable of delivering 17.5 lb/sec of 800°F gas (oxygenated nitrogen) to the reactor and, subsequently, of cooling the 1200°F gas exiting from the reactor to approximately 140°F at the suction of the blower.⁽³⁾⁽⁴⁾

The construction of the GCRF was completed in the early fall of 1959.

- 1. GATE HOUSE
- 2. SERVICE BUILDING
- 3-4. CONTROL AND TEST BUILDING
- 5. VENTILATION STACK
- 6. FUEL OIL STORAGE TANK
- 7. COOLING WATER PUMP HOUSE
- 8. COOLING TOWER

- 9. UTILITY WATER STORAGE TANK
- 10. RADIOACTIVE WASTE PUMP HOUSE
- 11-12. LIQUID RADIOACTIVE WASTE TANKS
- 13. WASTE GAS HOLDER
- 14. WASTE GAS STACK
- 15. BULK NITROGEN STORAGE
- 16. LEACHING PIT

FIGURE 6. GCRE FACILITY LAYOUT

FIGURE 7. GCRE-I FLOW DIAGRAM

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GCRF DESIGN AND CONSTRUCTION BIBLIOGRAPHY

1. Technical Specifications, Contract AT(10-1)-945
2. Technical Specifications, Contract AT(10-1)-999

These documents present the detailed technical specifications for the construction of the GCRF. As indicated by the two contract numbers, the construction was undertaken in two phases; the basic facility construction was completed under Increment I and the control room installation and certain modifications and refinements to the test loop were performed under Increment II.

3. IDO-28506*, "GCRE-I Hazards Summary Report," December 1958 with Addendum I, March 1959; Addendum II, February 1960; and Addendum III, May 1960
4. IDO-28598*, "Final Summary Report of the Gas-Cooled Reactor Experiment-I," October 1963

These documents include a description of the GCRF test facility and operating conditions.

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D. GCRE-I REACTOR OPERATION

After a delay of several months to correct deficiencies* in the control rods,⁽¹⁾ the initial critical experiment with the GCRE-I reactor was completed on 23 February 1960. A series of low power nuclear tests with plate-type fuel elements in the aluminum calandria was conducted between this date and mid-April 1960.⁽²⁾ At this time the aluminum calandria was replaced with a stainless steel assembly⁽³⁾ because the design of the ML-1 specified a stainless steel unit and because of repeated failures of the tube-to-tube sheet seal welds in the aluminum assembly. Reactor operation resumed in early June and, following the conduct of nuclear experiments, design power (2.2 Mw(t)) was attained in July.⁽²⁾

The replacement core loading of the pin-type design (see Section V-C-2) was received in August 1960 and, following replacement of the plate-type fuel elements with the new loading, a comprehensive series of nuclear and engineering tests was performed.⁽²⁾ The maximum power which could be attained was 1.8 Mw(t); the coolant flow distribution was such that the operating temperature of the seven central fuel elements reached the allowable limit at this power (1250°F outlet gas temperature).

The fuel element flow control orifices were changed during December 1960 (to redistribute coolant flow and reduce the temperature in the central fuel elements) and operation resumed to accumulate experience during extended periods at steady state conditions. This phase of the test program was interrupted on 3 January 1961 when the plant was shut down as a precautionary measure because of an incident at the SL-1. The GCRE facility served as a communications and decontamination headquarters during the early phases of the SL-1 incident investigation and, following decontamination and a comprehensive safety review, operation was resumed on 14 February 1961.

Test operations of the GCRE-I reactor were terminated on 6 April 1961 when a gas leak developed in a single calandria tube.⁽⁴⁾ The leak resulted from a horizontal crack in the area of the upper tube sheet internal coolant passage and occurred as a result of corrosion fatigue initiated from the outside of the tube when the tube sheet coolant flow was inadvertently discontinued. As a part of the program of investigation of this leak, the failed tube was plugged and the assembly thermal cycled. Pressure tests following this cycling revealed several new leaks and, in January 1962, the decision was made to deactivate the GCRE project. The deactivation work was essentially complete by July 1962.

A total of 2989 Mwh was accumulated on the GCRE-I reactor during the period of test operations, including approximately 1000 hr at full power. A summary of the major experiments conducted is presented in Table 5.

*The control rods would not operate reliably; in general, rods would scram satisfactorily but often could not be withdrawn because of rapid deterioration of the electrical or mechanical components.

TABLE 5 - SUMMARY OF GCRE-I EXPERIMENTAL PROGRAM

Experiment	Purpose	Results
<u>A. Plate-Type Fuel Elements in Aluminum Calandria</u>		
1. Initial (Wet) Critical (9200)*	Define critical loading with plate-type (IZ) elements in flooded core	36 plate-type elements required for criticality
2. Full Flooded Core (9201)	Demonstrate subcriticality of fully loaded core in flooded condition with two control rods inoperable	61 element plate-type core subcritical flooded with two strongest rods out
3. Reactor Drying (9204)	Demonstrate effectiveness of drying procedures	Loop dew-point; minus 40°F after 84 hr
4. Dry Critical (9202)	Define critical loading with plate-type elements in operating (dry) condition	59 plate-type elements required for criticality
5. Operating Core Loading (9203)	Define operating core loading	61 plate-type elements; $k_{\text{excess}} = \sim 0.7\% \Delta K/K$
6. Preliminary Instrument Calibration (9205)	Establish actual reactor operating power	Satisfactorily completed
7. Control Rod Calibration (9206)	Check procedures, calibrate fine rod, determine other rod worths	Completed; procedures generally acceptable
8. Moderator Temperature (9208)	Determine moderator temperature coefficient of reactivity	$+ 0.9 \times 10^{-4} \Delta K/K-^{\circ}F$
9. Flux Mapping (9209)	Check procedures for radial and axial flux and power distribution determinations	Completed; procedure and equipment generally satisfactory
<u>B. Plate-Type Fuel Elements in Stainless Steel Calandria</u>		
1. Initial (Wet) Critical (9500)	Define critical loading with plate-type elements in flooded stainless steel calandria	52 plate-type elements required for criticality
2. Full Flooded Core (9501)	Same as 9201 except with stainless steel calandria	72 element plate-type core subcritical with two strongest rods out
3. Dry Critical (9502)	Define critical loading of plate-type elements in stainless steel calandria in operating (dry) condition	66 plate-type elements required for criticality
4. Operating Core Loading (9503)	Define operating core loading in stainless steel calandria	71 plate-type elements
5. Flux Mapping (IB-90)**	Define reactivity worth of pin-type elements	Completed; satisfactory
6. Power Ascension (9210)	Raise reactor power to design value	1.85 Mw(t) limited by single element outlet gas temperature of 1250°F
7. Photoneutron Experiment (9510)	Establish magnitude of photoneutron source	Unsuccessful
8. Power Operation	Evaluate plate-type elements in stainless steel calandria	850 Mw-hr accumulated on plate-type elements
<u>C. Pin-Type Fuel Elements in Stainless Steel Calandria</u>		
1. Initial (Wet) Critical (9600)	Define critical loading of pin-type (IB-2L) elements	45 pin-type elements required for criticality
2. Full Flooded Core (9601)	Same as 9501 except with pin-type elements	57 element IB-2L core subcritical with two strongest rods out
3. Dry Critical (9602)	Define critical loading of pin-type elements in operating (dry) condition	53 pin-type elements required for criticality
4. Operating Core (9603)	Define operating core loading	56 pin-type elements; $k_{\text{excess}} = \sim 1.6\% \Delta K/K$
5. Reactivity Worth (9606)	Calibrate fine rod and establish fuel element reactivity worths	Completed successfully
6. Flux Distribution (9609)	Determine axial and radial power and flux distributions	Completed successfully
7. Moderator Temperature Coefficient (9608)	Determine moderator temperature coefficient of reactivity	$+ 0.9 \times 10^{-4} \Delta K/K-^{\circ}F$ at $\sim 100^{\circ}F$
8. Reactivity Coefficients (9605, 9607)	Determine temperature and pressure coefficients of reactivity	Temperature: $-1 \times 10^{-7} \Delta K/K-^{\circ}F$ Pressure: $-7 \times 10^{-7} \Delta K/K-psi$
9. Power Ascension (9610)	Raise reactor power to design value (uniform orificing)	1.8 Mw(t) limited by outlet gas temperature of 7 central elements
10. Rod Shadowing (9622)	Determine effect on fuel element temperature of rod shadowing	Completed successfully; little effect noted
11. Shutdown Cooling (9604)	Predict reactor temperature behavior in event of coolant flow stoppage	Maximum temperatures observed did not exceed normal operating values
12. Reactivity Coefficients (9630)	Same as 9605-9607	Temperature: $+ 0.34(\pm 0.21) \times 10^{-6} \Delta K/K-^{\circ}F$ Pressure: $-3.0(\pm 1.1) \times 10^{-6} \Delta K/K-psi$
13. Photoneutron Experiment (9629)	Same as 9510	Ratio of photoneutron power to full power = $7(\pm 2) \times 10^{-5}$

*Refers to the standard operating procedure (ANSOP) which specified experimental procedure.

**Refers to test element used for flux mapping.

GCRE-I REACTOR OPERATION BIBLIOGRAPHY

1. AGN TM-387, "The GCRE Control Rod System: Operational Difficulties and Design Modifications," August 1960

Describes the design modifications, resulting from testing and operational experience, of the GCRE-I control rod system.

2. IDO-28598*, "Final Summary Report of Gas-Cooled Reactor Experiment I," October 1963

A summary of the GCRE-I experimental program. Included also is a discussion of the operating organization and philosophy.

3. AGN TM-386, "Tube Bundle Replacement in the Gas-Cooled Reactor Experiment," July 1960

Describes the design and testing of the aluminum tube bundle as well as the operating experience with this calandria. The design, fabrication and installation of the stainless steel calandria is also discussed.

4. IDO-28597*, "Study of GCRE Tube Bundle Failure," December 1962

The history of the fabrication, testing and operation of the stainless steel calandria is reviewed and the post-failure examination and metallurgical investigations are discussed. The failure was identified as a horizontal crack in one tube at a location corresponding to the lower edge of the upper tube sheet internal cooling passage. The evaluation of the metallurgical examination data resulted in the conclusion that the failure was caused by an inadvertent interruption of the cooling water (moderator) flow in the tube sheet internal cooling passage. This condition resulted in boiling in the passage which initiated severe corrosion in the highly (mechanical) stressed portion of the tube wall which "bridged" the cooling passage.

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E. GCRE-II PROGRAM

The GCRE-II program was initiated in July 1958. The work undertaken involved the preliminary evaluation of a solid-moderated, homogeneous, gas-cooled reactor concept to serve as a backup for the GCRE-I in the event of serious technological problems with the reference design. A feasibility study of a 4 Mw(t) reactor fueled with UO_2 or UC dispersed in a graphite moderator and operating with an outlet temperature of 1400°F was completed as was the conceptual design of such a reactor.⁽¹⁾ A series of materials investigations were undertaken which resulted in the conclusion that the UC fuel was superior to UO_2 for this application and that refractory or metal carbide coatings were probably optimum for use with the fueled graphite to contain gaseous fission products.⁽²⁾ All work under this program was suspended in June 1960.

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GCRE-II BIBLIOGRAPHY

1. IDO-28530*, "Conceptual Design and Feasibility Study for the Gas-Cooled Reactor Experiment-II," October 1958; revised October 1959

The results of a feasibility study of a solid-moderated, homogeneous, gas-cooled reactor and the conceptual design of such a reactor are presented. The conceptual design describes a 4 Mw(t), UO_2 or UC-fueled, graphite-moderated, system with a 1400°F outlet gas temperature. Discussions of materials, neutronics, heat transfer and safety are included and the areas requiring significant development are identified.

2. IDO-28564*, "Summary Report on Materials for GCRE-II," October 1960

A summary of the results of investigations of materials for use in the GCRE-II is presented. It is concluded that UC fuel is superior to UO_2 for this application and that a refractory or metal carbide coating of the graphite is probably optimum for gaseous fission product retention.

3. AN-ENGR-40, "Carburization Barrier Coatings on Graphite," October 1961

Reports the results of an 832 hr test of Hastelloy X clad graphite specimens at 1750°F in which the cladding was badly carburized.

4. AN-ENGR-42, "Silicon-Carbide Coatings on Graphite," October 1961

Reports the results of a program to develop methods for applying Si-C coatings on graphite for high temperature service in nuclear systems. The laboratory testing of coated specimens indicated that development of a Si-C capable of preventing oxidation of the graphite for 10,000 hr in 1000°C air appeared feasible.

V. ML-1 PROGRAM

A. AGCR/PGCR

During the period when responsibility for the development of the major power plant components was divided between the USAEC (reactor) and the U. S. Army (power conversion system), the reactor was identified as the Army Gas-Cooled Reactor (AGCR) and characterized as a 25 ton trailer-mounted assembly with a fuel cycle of 3000 hr.⁽¹⁾ Some early studies by Aerojet of the problems to be solved in coupling a nuclear heat source to a closed-cycle gas-turbine system⁽²⁾ resulted in the recommendation that the AGCRSP be integrated under a single systems contractor to minimize the interface problems associated with the then current arrangement.⁽³⁾

In July 1958, the mobile power plant was identified as the Portable Gas-Cooled Reactor (PGCR) and defined as a marriage of a modified GCRE-type reactor with a modified GTTF (see later discussion under Section IX) power conversion system. The resulting assembly was to produce 400 kw (net) of electrical power; the maximum package weight was specified as 18 tons and the total plant weight as 40 tons. Aerojet developed a preliminary design for such a plant and laid out a development schedule which had as its goal the assembly of all components for a systems test by December 1960.⁽⁴⁾ A supplemental study was performed which established the feasibility of using waste heat from the PGCR for space heating.⁽⁵⁾

In June 1959, Aerojet was directed to proceed directly with the development, design and fabrication of a prototype of a field-operable mobile gas-cooled nuclear power plant (ML-1), eliminating, in effect, the engineering test model represented by the PGCR. This direction was necessitated by a DOD request in August 1958 to the USAEC to revise the goals of the AGCRSP to accomplish test operation of a prototype plant by 30 June 1962. At the same time, Aerojet was designated as systems contractor with overall responsibility for the completion of the work, operating under contracts with the USAEC Idaho Operations Office and with the Corps of Engineers at Ft. Belvoir, Virginia.

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AGCR/PGCR BIBLIOGRAPHY

1. AGN TM-315, "A Tentative Schedule and Budget for Development of the AGCR Program for FY 58-64," January 1958

An integrated program for the development of the AGCR (25 ton trailer-mounted, gas-cooled reactor for use in a nuclear power plant) is presented along with projected schedules and cost estimates.

2. AGN TM-314, "Combining the Closed-Cycle Gas Turbine and the Nuclear Reactor," March 1958

The modifications required to existing methods of cycle optimization for gas-turbine systems to include equations characteristic of a gas-cooled reactor are discussed. The relationship between reactor fuel element hot spot temperature, reactor coolant outlet temperature and coolant (working fluid) flow rate is derived and the effect of these variables on power cycle optimization is reviewed.

3. AGN TM-316, "An Integrated Program for the Development of the Army Gas-Cooled Reactor Power Plant and the Army Reactor Experimental Area," January 1958

An integrated program is presented and it is recommended that a systems contractor be designated to coordinate all facets of the work.

4. AGN TM-341, "PGCR - Preliminary Design Report," January 1959

The design criteria and preliminary design for the combination of a GCRE-I-type reactor with a GTTF type power conversion system is presented. The specifications describe a 400 kw (net) electrical power output plant with a maximum package weight of 18 tons and a total plant weight of 40 tons. Transportability characteristics are also specified and an evaluation of projected plant performance and reactor safety is included.

5. AGN TM-347, "PGCR Space Heater Feasibility Study," February 1959

Presents the results of a study of the feasibility of using PGCR waste heat for space heating and recommends a pressurized hot water system (300°F, 120 psia) to make available ~ 400 kw of space heat with the addition of 2500 lb of weight to the plant.

B. MILITARY CHARACTERISTICS

As previously indicated, in June 1959 the decision was made to proceed immediately with the development, design, fabrication and test operation of a prototype mobile gas-cooled nuclear power plant to be designated ML-1. As a corollary to this work, plans and specifications for a military prototype (ML-1A) of the plant were also to be prepared, based on ML-1 design and testing experience, which could serve as the basis for fixed-price procurement. All this work was to be complete by May 1962.

The performance specifications for the ML-1 plant were transmitted to Aerojet as a set of Military Characteristics (Appendix B). These specifications differed from those of the PGCR in several areas but the most significant variation was in the maximum package weight which was reduced from 18 tons to 15 tons. Since the radiation environment criteria were not changed, the change in the weight specification required that the reactor core size be reduced to reduce the volume and weight of shielding. An evaluation of the consequences of reducing the core size revealed that the plate-type fuel element concept would have to be abandoned (the existing design incorporated as much fuel as was feasible in this configuration and some mechanical problems were not completely solved) in favor of a ceramic-fueled pin-type fuel element.

The development and design of the ML-1 plant based on the Military Characteristics was initiated immediately. At this point in time, GCRE-I had not attained criticality and the first t-c set had not yet been delivered from the vendor.

C. ML-1 DESIGN AND DEVELOPMENT

1. General

The significant activities undertaken during the design and development of the ML-1 power plant are discussed in this section. The final plant characteristics are presented in Appendix B and a list of the major drawings describing the plant may be found in Appendix D. Typical views of the major power plant components are shown in Figures 8, 9, and 10 and the P&I diagrams and plant layout drawings are presented in Appendix F.

2. Fuel Element Design and Development

Three core loadings were designed and developed for the ML-1 reactor. The basic design criteria for all fuel elements are summarized in Table 6.

A pin-type fuel element, consisting of 1 center pin, 6 inner ring pins, and 12 outer ring pins and incorporating ceramic $\text{UO}_2/\text{UO}_2\text{-BeO}$ fuel, was selected as the basic concept. The 19 pins were supported by an upper spider and were free to expand thermally into but were guided radially by a lower spider. Fuel pin cladding material evaluations performed early in the design and development stage served as the basis for selecting Hastelloy X as the cladding material. The bundle of 19 fuel pins was surrounded by a liner assembly consisting of an inner and an outer liner between which was placed the thermal insulation and the burnable neutron poison. The liner assembly served

OUTBOARD END

1. Shield Solution Pump
2. Moderator Pump
3. Demineralizer

POWER CONVERSION SKID END

1. Inlet Gas Duct
2. Outlet Gas Duct
3. Socket for Power Conversion Skid Fit-up
4. Moderator Line from Cooler
5. Moderator Line to Cooler
6. Control Rod Actuator Well Cover Plate
7. Main Support Point

FIGURE 8. ML-1 REACTOR SKID

1. Alternator (Dummy)	4. Moderator Cooler
2. Precooler	5. Recuperator
3. Lube Oil Cooler	6. Turbine Compressor Set

FIGURE 9. ML-1 POWER CONVERSION SKID

11-90-66-484

Interior

Exterior

FIGURE 10. ML-1 CONTROL CAB

to insulate the reactor pressure tubes from the hot coolant gas and act as an axial support member for the fuel element assembly. The three core loadings⁽¹⁾ developed for the ML-1 power plant are identified and characterized in Table 7; Figures 11 and 12 show the first and second core fuel elements.

TABLE 6 - BASIC DESIGN CRITERIA FOR ML-1 FUEL ELEMENTS

<u>Parameter</u>	<u>Criteria</u>
Coolant inlet temperature, °F	781
Coolant outlet temperature, mixed mean, °F	1200
Coolant pressure, nominal, psi	300
Thermal power, gross, Mw	3.3
Thermal power to coolant, Mw	2.98
Design lifetime, hr*	10,000
Coolant	99.5 vol% N ₂ with 0.5 vol% O ₂

*The first core loading was expected to have a lifetime of approximately 3000 hr because of cladding corrosion and fuel burnup characteristics. This estimate was later revised upward to 10,000 hr based on in-pile test and ML-1 operating experience.

The iterative process involved in the design of the fuel elements included consideration of the results of critical experiments⁽²⁾, neutronic and thermal analysis⁽³⁾ and the extensive developmental test program summarized below:

- a. Cladding Selection - Extensive laboratory tests were conducted which resulted in initial selection of Hastelloy X as the fuel cladding.⁽⁴⁾ Subsequent laboratory tests and evaluation of in-pile experiments have verified the suitability of this material for use at temperatures up to 1750°F for 10,000 hr.⁽⁵⁾
- b. BeO-UO₂ Fuel Evaluation - The series of capsule experiments performed to evaluate fuel stability, radiation effects and fuel/cladding interactions and to determine burnup and fission gas release characteristics⁽⁶⁾ are summarized in Table 8. Information gained from the in-pile tests, described below and summarized in Table 9, complemented and expanded the results of the capsule experiments in the area of BeO-UO₂ fuel evaluation.
- c. Mechanical Tests - The strength and mechanical integrity of the spider design was demonstrated.⁽⁷⁾ Thermal shock and cycling tests were conducted to evaluate the effectiveness of the design to accommodate differential thermal expansion.⁽⁸⁾ Tests were performed to demonstrate the ability of the element to withstand rapid depressurization.⁽⁹⁾ Shock and vibration tests were performed to demonstrate compliance with the requirements of the Military Characteristics.⁽¹⁰⁾

TABLE 7 - ML-1 CORE LOADING CHARACTERISTICS

PARAMETER	ML-1-I	ML-1-IM	ML-1-II
Design reference	IDO-28550 AGN TM-378	AN-AGCR-958	AGN TM-415
General			
Purpose	Initial loading	Partial replacement loading	Second loading
Design, Development & Fabrication Period	6/59-2/61	3/65-9/65 ⁽¹⁾	9/60-12/62 ⁽²⁾
Design lifetime, hr	3000 ⁽³⁾	10,000	>10,000
No. of fuel elements/core loading	61	19	61
Nuclear			
K _{eff} (cold, clean; w/o reactivity shims, burnable poison or control rods)	1.067	1.066 ⁽⁴⁾	1.103
Burnable poison, % $\Delta K/K$	-0.6	-0.6	-5.2
Reactivity shims, % $\Delta K/K$	-4.6	-4.6	-3.9
Temperature coefficient, cold to operating condition, % $\Delta K/K$	-0.2	-0.2	-0.2
Control margin, % $\Delta K/K$	-1.3	-1.2	-1.0
Thermal			
Heat transfer area, core, ft ²	127	133 ⁽⁴⁾	130
Heat flux, average, Btu/hr-ft ²	7.93×10^4	7.68×10^4 ⁽⁴⁾	7.74×10^4
Heat transfer coefficient, average, Btu/hr-ft ² -°F	310	310	220-390 ⁽⁵⁾
Maximum nominal cladding surface temperature, °F	1510	1470	1240-1410 ⁽⁶⁾
Maximum hot spot cladding surface temperature, °F	1650	1620	1550
Power distribution, radial, P_{max}/P_{ave}	1.51	1.60	1.25
Power distribution, axial, P_{max}/P_{ave}	1.26	1.31	1.26
Fuel center temperature at hot spot location, °F	2380-2100	2000	2000
Coolant flow rate, core, lb/sec	25.2	25.2	25.2
Flow area/element, nominal, in. ²	0.71	0.73	0.85
Reynolds number, average	4.0×10^4	3.84×10^4 ⁽⁴⁾	3.78×10^4
Coolant velocity, average, ft/sec	1.56×10^2	1.52×10^2 ⁽⁴⁾	1.3×10^2
Pressure drop through element (orificed), psid	12.5	12.5	~ 10
Physical			
Core loading, kg U-235 ⁽⁸⁾	49.41	44.94 ⁽⁴⁾	41.18
Pin loading, gm U-235 ⁽⁸⁾	75.0-30.0 ⁽⁷⁾	38.8	35.5
Fuel material	100 wt% UO ₂ (6 pins) 70 wt% UO ₂ in BeO (12 pins) ⁽²⁾	71.8 wt% UO ₂ in BeO (19 pins)	74.9 wt% UO ₂ in BeO (19 pins)
UO ₂ grain size (microns)	5-20	5-20	5-20
Active length, in.	22	22	22
Pellet dimensions, in.	0.325 x 0.177 dia. (inner ring) 0.230 x 0.176 dia. (outer ring)	0.240 x 0.197 dia.	0.200 x 0.181 dia.
Cladding material	Hastelloy X (0.07% Cu)	Hastelloy X (0.10% Cu)	Hastelloy X (1.8% Cu)
Cladding OD, in.	0.241	0.264	0.250
Cladding wall thickness, in.	0.030	0.032	0.033 (smooth section), 0.022 (fin section) min.
Spacer wire diameter, in.	0.040	0.029	0.045
Inner liner material	Hastelloy X	Hastelloy X	Hastelloy X
Inner liner inside diameter, in.	1.426	1.509	1.518
Inner liner thickness, in.	0.010	0.010	0.010
Outer liner material	Type 316L stainless	Type 316 ELC stainless	Hastelloy X
Outer liner inside diameter, in.	1.668	1.668	1.668
Outer liner thickness, in.	0.012	0.010	0.011
Burnable poison location	lower 11 in. of fuel length	(same as -I)	covers full 22 in. of fuel length
Burnable poison material	1% Cd in Cu, clad in Type 316 stainless	0.5 gm Cd in Cu, clad with nickel	2.15 gm europium as Eu ₂ O ₃ -2TiO ₂ clad with Type 304L stainless
Burnable poison thickness, in.	0.012	0.018	0.010
Insulation material	Thermax (RF-1800)	Refrasil (TF-1005)	Refrasil (TF-1005)
Insulation thickness	0.091	0.030	0.030
Hanger material	Type 316 SS	Hastelloy X	Hastelloy X
Upper spider material	Type 316L stainless casting	Hastelloy X casting	Hastelloy X casting
Lower spider material	Hastelloy X (machined)	Hastelloy X casting	Hastelloy X casting
Spider inner support ring diameter, in.	0.567	0.602	0.605
Spider outer support ring diameter, in.	1.097	0.170	1.160

(1) Limited development; design based on development work performed in connection with ML-1-I and II designs.

(2) No fuel elements of this design were fabricated.

(3) Subsequently increased to 10,000 hr based on materials evaluation and in-pile tests.

(4) For mixed core of 19 new ML-1-IM and 42 partially exposed ML-1-I elements.

(5) Two values represent smooth and finned section, respectively.

(6) Quoted value is for ML-1-IM elements when operating in mixed ML-1-I/ML-1-IM core; see corresponding items for values for ML-1-I elements.

(7) 75 gms in 6 pins; 30 gms in 12 pins.

(8) Fully enriched.

TABLE 8 - ML-1 CAPSULE IRRADIATION EXPERIMENTS

Capsule Description		Purpose of Test	Duration, hr	Test Completion Date	Results		Reference
Model	Fuel Characteristics ⁽¹⁾				Fission Gas Release, %	Burnup, % U-235	
BRR-3CR-2	56 wt% UO ₂ (small grain) in BeO ⁽²⁾	Determine U-235 burnup, fission gas release, extent of radiation effects on the integrity of the fuel body, and extent, if any, of fuel/cladding reactions.	1152	May 1960	Data unreliable; probably because of leaks	1.4-2.3	BMI-1484
BRR-3CR-3	100 wt% UO ₂	Same as BRR-GCR-2 except for a 100% UO ₂ fuel body.	3312	May 1961	14-24	3.2	IDO-28582
MTR-3CR-3	70 wt% UO ₂ (small grain) in BeO	Determine fuel body properties as in BRR-GCR-2 for 70 and 80 wt% UO ₂ fuel loading and fuel material stability at these loadings.	3912	April 1961	0.6-2.2	6.9-8.5	IDO-28600
	80 wt% UO ₂ (small grain) in BeO				2.7 ⁽⁴⁾	4.8-5.6	
MTR-3CR-4	75 wt% UO ₂ (large grain) ⁽³⁾	Determine fuel body properties for large grain fuel for comparison with previous (small grain) capsule experiments	2570	February 1963	(NaK heat transfer medium was lost during the test and all 6 specimens melted; no evaluation could be made)		AN-AGCR-719

(1) Fuel cladding in all cases was Hastelloy X. Heat transfer medium around the cladding was NaK.

(2) The grain size of small grain UO₂ particles is 5-20 microns.

(3) The grain size of large grain UO₂ particles is 150 to 200 microns.

(4) One of the 2-80wt% UO₂ specimens failed due to fuel melting, swelling, and subsequent cladding rupture.

FIGURE 11. ML-1-I FUEL ELEMENT

FIGURE 12. ML-1-II FUEL ELEMENT

- d. Heat Transfer - A number of tests were conducted to determine the friction factor and heat transfer characteristics of various fuel pin and pin bundle designs.⁽¹¹⁾ These tests employed both the capacitance discharge and naphthalene loss techniques.
- e. Fabrication Process Development - An extensive developmental effort was undertaken to specify the process for fabrication of the BeO-UO₂ fuel pellets⁽¹²⁾ and the parameters for the various welds and assembly processes required in the assembly of the element.⁽¹³⁾
- f. In-Pile Tests - The results of several in-pile tests⁽¹⁴⁾ performed to provide design data or to demonstrate the adequacy of the designs are summarized in Table 9. In addition, these tests provided excellent data for the evaluation of the UO₂-BeO fuel under actual operating conditions and significantly expanded the data obtained in the capsule irradiation experiments.

One full core loading (ML-1-I), consisting of 61 fuel elements and 8 spares, and a partial core loading (ML-1-IM), consisting of 20 elements to be used for spiking the ML-1-I core, were fabricated. At the time of termination of the AGCRSP activities, the ML-1-I core loading had operated successfully for approximately 2500 equivalent full-power hours. There were no indications of problems with the fuel, although a hot cell evaluation would be required to determine the exact condition of the elements and the potential for continued operation. The results of the in-pile irradiations of a prototype ML-1 type test element indicated that the element was fully capable of operation beyond the 10,000 hr goal. The in-pile ML-1-II prototype (IB-17R) irradiation was terminated after 9501 hr when a fuel pin defect allowed fission gases to leak into the test loop. Hot cell evaluation of the element indicated that only one fuel pin leaked and that the failure was caused by a defect originally present in the fuel pin tubing.

The results of metallographic examination of the area of the failure indicated that normal oxidation had penetrated the thin surface of the cladding which remained after machining of the turbulence promoters and connected to small internal defects to cause the failure. These defects were not detected by the extensive quality control inspections conducted prior to the machining operation and could not have been detected by similar inspections conducted following machining because of the perturbations introduced by the varying tube wall thickness associated with the turbulence promoter configuration (dye penetrant examination after machining did not reveal any porosity).

It appears that the ML-1-type fuel element design is capable of 10,000 hr of full power operation provided improved inspection techniques can be developed to assure that all cladding tubing used during fabrication is sound.

During the handling of the ML-1 first core fuel elements for reprocessing in early 1966 (the elements were removed from the reactor in October 1965 and stored in demineralized water), it was observed that the outer liner of several of the elements was cracked. This evidence of

TABLE 9 - ML-1 IN-PILE TEST PROGRAM

Element Model	Description Fuel Characteristics	Purpose of Test	Test Condition		Testing Dates		Results		Remarks	Reference
			Coolant	Duration, hr	Start	Complete	Fission Gas Release (max), %	Burnup (max), % U ²³⁵		
IS-7T-1	UO ₂	Determine fission product release through a cladding defect, fission product deposition.	99.5% N ₂ - 0.5% O ₂	648	5/1/61	6/4/61	0.06 (release to coolant through defect)	--	a. 1 ³¹ I plate-out at end of fuel pin = 25.8 d/sec-cm ² b. Fuel oxidized at defect; no swelling.	AGN TM-404
-2	70 wt% UO ₂ (small grain) -BeO		100% N ₂	553	9/23/61	10/22/61	Small since defect became plugged with oxide.	--	a. No detectable plate-out. b. Fuel oxidized; no swelling.	
-3	UO ₂ and 70 wt% (small grain) UO ₂ -BeO		Air	400	4/23/61	5/10/62	Small since defect became plugged with oxide.	--	a. No detectable plate-out. b. Fuel oxidized; no swelling.	
IS-8T-1	100% UO ₂ inner pins 60 wt% UO ₂ (small grain) in BeO outer pins unfueled center pin.	Confirm neutronic, heat transfer, and hydraulic analysis of ML-1-I mechanical and materials information.	99.5% N ₂ - 0.5% O ₂	6415	12/2/60	1/1/62	0.73 (UO ₂ -BeO fuel) 1.02 (UO ₂ fuel)	5.8 1.8	a. Cladding was embrittled by nitrogen introduced through leaks in thermocouple sheathing. b. Cracking was noted in both UO ₂ and UO ₂ -BeO fuel. c. Fuel pin spiralling was noted, particularly on hotter pins. d. Performance of element was as predicted.	IDO-28616
-2	Same as IS-8T-1	Obtain lifetime mechanical and materials information	99.5% N ₂ - 0.5% O ₂	6415	12/2/60	10/7/62	0.75 (UO ₂ -BeO fuel) 5.6 (UO ₂ fuel)	9.5 2.9	a. Cladding corrosion = 0.0044". b. Minor cracking of UO ₂ -BeO observed. c. Severe cracking of UO ₂ fuel observed. d. Fuel pins were spiralled. e. Performance of element was as predicted. f. Element appeared capable of continued operation.	
			Air	3736						
				10151 (Total)						
IS-14R	65 wt% UO ₂ (small grain) in BeO (19 fuel pin uniform loading)	Confirm preliminary design analysis and out-of-pile test results on turbulence promoted fuel pins.	Air	3735	1/15/62	10/7/62	0.31	1.65	a. Performance of turbulence promoters could not be evaluated because of poor thermocouple readouts. b. Pressure drop was 1.5 psi greater than IS-8T elements.	AN-AGCR-655
IS-17R-1	75 wt% UO ₂ (large grain) in BeO (19 fuel pin uniform loading)	Confirm neutronic, thermal, and hydraulic analysis of the ML-1-II design and obtain 1000 hr data on materials properties.	Air	1179	2/4/63	4/1/63	1.77	0.5	a. Turbulence promoter performance was somewhat poorer than predicted; however thermal performance evaluation was somewhat clouded by instrumentation failures and data scatter. b. Some fuel pin spiral twisting was noted. c. Cladding corrosion (max) = 0.0005". d. Fuel cracking was greater than seen in previously irradiated UO ₂ -BeO fuel.	AN-AGCR-697
-2	Same as IS-17R-1	Obtain long term heat transfer data on ML-1-II design and obtain 6000 hr data pertaining to materials and mechanical behavior.	Air	5819	2/4/63	1/6/64	Gas leaks through thermocouple/fuel pin brazings did not allow a meaningful release fraction to be calculated.	2.6	a. Turbulence promoter tended to increase heat transfer coefficients such that a 35°F to 50°F reduction in maximum nominal temperature was indicated compared to smooth wall tubing. b. Pressure drop was about 1 psi less than design predictions. c. Fuel pin spiral twisting noted in hotter pins. d. Cladding corrosion (max) = 0.003". e. Excessive fuel cracking was noted. f. Eutectic-stainless steel burnable poison was unaffected.	AGN TM-409
-3	Same as IS-17R-1	Obtain lifetime, materials and mechanical information on the ML-1-II fuel element design	Air	9501 ⁽¹⁾	4/1/63	11/24/64	No analysis performed	No analysis performed.	Terminated due to fission gas release from fuel pin No. 1. Cladding defect apparently present in the as-built condition. See text for details of failure mechanism and post-test evaluation.	AN-AGCR-892 AN-AGCR-950

(1) Terminated due to fission gas release from a fuel pin.

embrittlement was being evaluated at the time of publication of this report; the findings of this investigation will be published in Report IDO-17160.

ML-1 FUEL ELEMENT DESIGN AND DEVELOPMENT BIBLIOGRAPHY

1. Core Design

- a. IDO-28550*, "ML-1 Design Report," 12 January 1960

Describes the design of the ML-1 system including the original design of the ML-1-I fuel elements.

- b. AGN TM-378, "ML-1 Components Design Report," July 1960

Describes the final detail design of the ML-1-I fuel elements as delivered to the test site, in addition to the detailed design of other ML-1 system components.

- c. AGN TM-415*, "ML-1-II Design Report," January 1966

Describes the design and the results of the development and testing of the ML-1-II fuel element.

- d. AN-AGCR-662, "ML-1-I Fuel Element - Lower Spider Modification," June 1964

Presents the results of the ML-1-I lower spider failure analysis, and describes the design, development, and field installation of the modified lower spider.

- e. AN-AGCR-958, "ML-1-IM Design Report," December 1965

Describes the design of the ML-1-IM fuel elements intended to spike the ML-1-I core.

2. Critical Experiment

- a. BMI-1501*, "ML-1-IB Critical Assembly Studies," 1961

Presents the results of the ML-1 critical experiments performed in the critical facility at Battelle Memorial Institute. These results were used for the prediction of the ML-1-I core loading.

- b. IDO-28631*, "Summary - Nuclear Physics Experiments," January 15, 1965

Presents a comprehensive summary of the results of the nuclear physics experiments performed in support of the ML-1 core analysis. Details of experimental procedure and data analysis are not given but a detailed bibliography is included which references specific test reports. Experiments were conducted on both the concentric plate-type element (GCRE) and the 19-pin type element typical of the GCRE-IB-2L and ML-1 type elements. The

experiments were conducted to obtain the following general information for each type of fuel element:

- Critical core configuration (dry and flooded) and the worth of peripheral elements.
- Worths of various types of burnable poisons.
- Worth of reflectors.
- Worth of fuel and inert material additions.
- Worth of non-uniform fuel element spacing.
- Worth of control blade configurations and materials.
- Worth of various reactivity shims.
- Values of temperature and void coefficients.
- Flux and power distributions (both full core and fuel element radial and axial profiles).
- Effects of control blades on power and flux distribution.
- Pin-to-pin power ratios in pin-type elements with various fuel loadings and pin sizes.

3. Neutronic and Thermal Analysis

- a. AGN TM-355, "COP (Core Optimization Program), A Heat Transfer and Fluid Flow Code," 28 April 1959

Describes the programming and operation of a Fortran computer code which calculates temperatures and pressure drop for a cylindrical core and optimizes core cooling design by a combination of design choices and orificing.

- b. AGN TM-364*, "Temperature Distribution in a Fuel Pellet and Surrounding Metal Wall," October 1959

Describes the general analytical solution for the temperature distribution in a UO_2 fuel pellet and in the surrounding metal cladding for a typical pin-type fuel element.

- c. AGN-TM-369, "MYSTIC - An IBM 704 Nuclear Code to Obtain Multigroup Diffusion and Transport Theory Constants," 1 October 1959

Describes the programming and operation of a Fortran computer code which calculates multigroup and transport theory constants used in neutronic core analysis for a given neutron spectrum.

- d. AGN TM-381*, "HECTIC - An IBM 704 Computer Program for Heat Transfer Analysis of Gas-Cooled Reactors," May 1961

Describes the programming and operation of the HECTIC (Fortran) computer code for the calculation of axial and radial temperature distribution in gas or liquid-cooled reactor passages. (See also IDO-28595 - HECTIC-II.)

- e. AGN TM-407*, "AGN-GAM, An IBM-7090 Computer Code to Calculate Spectra and Multigroup Constants," April 1965

Describes the programming and operation of a Fortran code to calculate spectra and multigroup constants for use in neutronic core analysis. Broad group constants, weighted by the flux quantities, are derived for use in diffusion, isotropic transport or anisotropic transport calculations. The method of Adler, Hinman, and Nordheim is used to calculate resonance absorption and fission cross sections.

- f. AGN TM-411*, "BOUNCE: An IBM-7090 Code to Calculate Flux Distribution Within Multiple-Pin Fuel Elements by the Method of Successive Generations," November 1965

Describes the programming and operation of a Fortran code to determine the neutron flux distributions within the pin bundle of a multiple-pin fuel element by using the method of successive generations. Computer code results compared to experimental results show good agreement.

- g. IDO-28595*, "HECTIC II - An IBM 7090 Fortran Computer Program for Heat Transfer Analysis of Gas-or Liquid-Cooled Reactor Passages," December 1962

Describes the programming and operation of a Fortran computer program used to calculate radial and axial temperature distribution in reactor coolant passages. This code allows greater flexibility in the selection of boundary conditions than those of the original HECTIC code. (See AGN TM-381.)

- h. AN-AGCR-1, "BURP, An IBM 7090 Machine Code Which Solves for Reactor Burnup by Using the Two-Group Perturbation Equations," March 6, 1962

Describes the programming and operation of a Fortran computer code which estimates core lifetime.

- i. AN-AGCR-20, "Axial Temperature Gradient and Thermal Stress at Smooth and Finned Transition of Fuel Pin," March 17, 1962

Presents the analysis performed to estimate the thermal gradient and stresses present in the transition zone between the smooth portion and finned portion of the ML-1-II fuel pin cladding. Results indicated that the axial bending stress is 2800 psi while the tangential stress is 1300 psi.

- j. AN-AGCR-30, "II-C-Replacement Reactivity Measurements in the ML-1 Reactor," March 23, 1962

Describes the experimental work and data reduction of the ML-1-IIC experiment, the results of which were the basis of the ML-1-II neutronic design.

- k. AN-AGCR-31, "Fuel Pin Temperature During ML-1-I Core Flooding," March 26, 1962

Presents the analysis to estimate fuel pin temperature and thermal gradients resulting from post-operation core flooding; alternate methods of core flooding are suggested.

- l. AN-AGCR-60, "ML-1-I Fuel Pin Thermal Stresses During Core Flooding," April 9, 1962

Presents the analysis and results of the estimation of fuel pin thermal stresses resulting from core flooding operation performed 2 days after reactor shutdown; calculated axial bending stress and tangential stress are 9,200 psi and 11,500 psi, respectively.

- m. AN-AGCR-283, "Thermal Stresses in the ML-1 Core Upon Actuation of the Emergency Coolant Gas System," July 30, 1962

Presents the results of the fuel pin thermal stress analysis for the case where the emergency coolant gas is injected into a hot core; resulting stresses were less than those which could cause cladding failure.

- n. AN-AGCR-294, "ML-1 Coolant Loss Temperatures," 14 August 1962

Presents the results of analyses performed to estimate the maximum fuel cladding temperatures associated with a coolant loss accident; sets the design requirements for the emergency cooling system.

- o. AN-AGCR-299, "Pin Bowing Versus Reactivity Change for Step Change in Power," 16 August 1962

Presents the analysis to determine the dependency of reactivity on the radial displacement of fuel pins as limited by the mechanical design of the assemblies. Considers two extreme cases: all pins in each element bowing toward the center of the element and all elements bowing toward the center of the core; maximum reactivity changes for each case were 0.05% $\Delta K/K$ and 0.01% $\Delta K/K$, respectively.

- p. AN-AGCR-300, "ML-1 Reactivity Variations Caused by Pin and Pressure Tube Bowing," 17 August 1962

Describes the analysis used to develop the general expression relating reactivity and fuel pin displacement in Reference o above.

- q. AN-AGCR-335, "IIC-Pin-to-Pin Power Reactor Measurements in the ML-1 Reactor," August 31, 1962

Describes the experimental procedure, data reduction, and results of in-reactor testing to determine pin-to-pin power ratios for the ML-1-II and similar fuel elements.

- r. AN-AGCR-388, "Two Dimensional Steady State Heat Transfer Code for the IBM-7090--2 DRZ," November 30, 1962

Describes the programming and operation of a Fortran computer code which calculates steady state temperatures in a cylindrical coordinate system considering conductive and convective heat transfer mechanisms.

- s. AN-AGCR-392, "ML-1 Kinetics," 4 December 1962

Presents the results of a series of calculations performed to determine the neutron flux changes and stable reactor periods resulting from a step increase or decrease in reactivity.

- t. AN-AGCR-403, (403A), "Experimentally Determined Axial Power Distributions of Individual Fuel Pins in ML-1 Fuel Elements Under the Influence of Control Blades," December 28, 1962

Describes the experimental procedure, data reduction, and results of in-reactor testing to determine the effect of control blade position on the axial power profile of ML-1-I fuel pins.

- u. AN-AGCR-431, "Shim Liner Effects on Individual Pin Power and Pin-to-pin Power Ratios as Measured in the ML-1 Reactor," 27 December 1962

Describes the experiment, data reduction, and results of tests performed to determine the effect of reactivity shims on power ratios and profiles. Results indicate that pin power is decreased in the area of the shims but increased in unshimmed areas; pin-to-pin power ratios did not vary between shimmed and unshimmed areas.

- v. AN-AGCR-446, "ML-1 Fuel Element Center Temperatures," January 25, 1963

Compares the calculated cladding and ceramic fuel temperatures of the ML-1-I and ML-1-II fuel element designs and presents the associated assumptions and calculations; peak fuel centerline temperature is shown to be 2800°F.

- w. AN-AGCR-461, "ML-1-I Upper Plenum Pressure Distribution,"
15 February 1963

Presents the analysis performed to estimate pressure losses and subsequent coolant flow redistribution resulting from the incorporation of fuel element holddown hardware in the upper plenum region. Results indicate that an increased pressure drop of 1.5 psi and an increase in fuel pin hot spot temperature from 1700°F to 1760°F would be the maximum expected from the flow redistribution; subsequent reactor operation indicated these estimates to be conservatively high.

- x. AN-AGCR-471, "The ML-1-IØ Flux and Power Mapping Experiment,"
28 February 1963

Presents the experimental procedure, data analysis, and results of experiments performed to determine the radial and axial power profiles of the ML-1-I core for various control blade positions; results indicate that both the radial and axial power profiles are heavily dependent upon control blade position.

- y. AN-AGCR-531, "Pin-to-Pin Power Ratios in the Flooded ML-1 Core,"
17 May 1963

Presents the experimental procedure, data analysis and results of experiments performed to determine the pin-to-pin power ratios of the ML-1-I core in the flooded condition and incorporating 28 reactivity shims. The results indicate that the ratio of power produced in an outer ring pin to that produced in an inner ring pin is approximately 20% higher in the flooded condition than in the dry condition.

- z. AN-AGCR-582, "Helium Production in ML-1-I Fuel Pins,"
September 3, 1963

Presents the physics describing the production of helium in ML-1-I fuel pins resulting from beryllium ($n, 2n$) and (n, α) reactions; results indicate that a maximum of 2.9×10^{19} He atoms/cc of UO_2 -BeO fuel would be formed after 10,000 hours at 3.4 Mw power. Also see AN-AGCR-584.

- aa. AN-AGCR-584, "Helium Production in the ML-1-I Fuel Pins,"
September 5, 1963

Describes the analysis performed to estimate the release of transmuted helium from the fuel pellets based on calculated total helium production and fission gas release in the IB-8T fuel elements. Results indicate that if all produced helium was released, the helium partial pressure in the fuel pin would be 166 psi; if 10% of the produced helium was released, the helium partial pressure would be 57 psi (see AN-AGCR-582).

- bb. AN-AGCR-618, "Re-Shimming of ML-1-I Core," December 4, 1963

Presents the analysis used in setting the reactivity patterns in the ML-1-I core, the resulting power profiles and the expected core reactivity variations as a function of operating time.

- cc. AN-AGCR-622, "The Effect of the Fuel Element Nose Pieces Modification on ML-1 Lower Tube Sheet Temperatures," December 13, 1963

Presents the results of computer analysis performed to estimate the ML-1 lower tube sheet temperature distribution resulting from the incorporation of lateral ports in the fuel element nose piece. It is shown that the ported nose piece results in lower tube sheet temperatures which are 10 - 15°F higher than those obtained when nose piece ports are not present.

- dd. AN-AGCR-627, "ML-1 Reactivity Changes: March 1963 to November 1963," January 24, 1964

Presents the results of ML-1 reactivity measurements during March 1963 and November 1963 which indicate a decrease of 0.5% $\Delta K/K$ reactivity. Analyses show that this is primarily the result of an increase in the weight of silver on the new reactivity shims.

- ee. AN-AGCR-639, "ML-1-I Re-Orificing," January 10, 1964

Describes the analysis performed to size the fuel element flow control orifices after the lower spider had been modified.

- ff. AN-AGCR-642, "ML-1-I Fuel Element Safety Analysis Summary," January 17, 1964

Presents the safety analysis for hypothesized accidents unique to the ML-1-I fuel element with a modified lower spider assembly.

- gg. AN-AGCR-679, "Review of ML-1-I Hot Spot Temperature Effects," April 6, 1964

Summarizes previous hot spot analyses and updates these results where applicable using the most recent experimental data. The revised hot spot adjustment (above the maximum nominal cladding temperature) was $63 \pm 87^\circ\text{F}$. At reference ML-1 operating conditions, this resulted in an estimated maximum hot spot temperature of 1650°F , 50°F less than the previous prediction.

- hh. AN-AGCR-774, "Calculation of Angular Neutron Fluxes Within ML-1 Cells," 15 October 1964

Presents the analytical method used to estimate angular fluxes in ML-1 type cells. Results indicate that the angular flux is strongly dependent upon the presence of burnable poison.

- ii. AN-AGCR-788, ML-1-I Reactor 1750°F and 1800°F Hot Spot Temperature Operating Envelopes for the Present Orificed Pattern," December 14, 1964

Updates previous operating envelopes to reflect recent experimental data.

- jj. AN-AGCR-827, "Fuel Element Transient Temperatures During Reactor Shutdown: Experimental Data Analysis," March 9, 1965

Describes the analysis of ML-1-I afterheat experiment data and formulates a general mathematical model which closely approximates the experimental results.

- kk. AN-AGCR-927, "ML-1 Coolant Loss Accident Analysis," August 18, 1965

Presents the results of analysis performed to determine peak fuel element shutdown temperatures as a function of coolant flow decay, operating time, and shutdown reactivity. Uses the mathematical model described in AN-AGCR-827 and updates all previous analytical work in this area. Results indicate that the peak fuel pin temperature following a simultaneous reactor scram/coolant loss would be about 2150°F occurring about 76 seconds after scram. A reactor scram followed by 4 seconds of coolant flow would result in a peak temperature of 1880°F, occurring 240 seconds after the scram. Peak temperatures below 2000°F would be maintained if at least 2 seconds of post-scram coolant flow were available to the fuel pins.

- ll. AN-AGCR-937, "ML-1-I Fuel Element Temperatures and Stresses During ML-1/GCRF Transient operation," October 1, 1965

Describes the unique thermal and mechanical behavior of ML-1-I fuel elements operating in the ML-1/GCRF configuration during scram and shutdown conditions.

- mm. AN-FDD-210, "ML-1 Fuel Element Maximum Operating Temperature - Summary Report," August 26, 1961

Describes the analysis performed to summarize and combine previously independent hot spot effects into one hot spot factor. This work is the basis of all subsequent hot spot analysis.

- nn. AN-FDD-217-(217A), "ML-1-I Orificing," September 7, 1961

Presents the analysis of fuel element flow control orificing; includes basic assumptions, power profiles, elemental power skewing data, and the selection of the original ML-1-I orificing pattern.

4. Cladding Selection

- a. IDO-28605*, "Evaluation of Long Term Gas Corrosion of Heat Resistant Alloys for Use as Gas-Cooled Reactor Fuel Cladding," April 1963

Describes the results of the experimental work performed to screen candidate cladding materials for use in the ML-1 fuel pin environment. Presents the justification for the selection of Hastelloy X as the cladding material. Hastelloy X, Inconel 600, Inconel 702, Inconel 604, Hastelloy R-235, G-157, Nichrome V, Nichrome V-Cb, and AISI Type 310 stainless steel were evaluated to determine gas corrosion effects and internal microstructural changes after exposure for periods up to 10,000 hrs at 1750°F (4000 hrs at 1850°F) in the reference ML-1 gas environment (99.5% N₂-0.5% O₂). The oxide scale associated with Hastelloy X was seen to be quite adherent and spalling was minimal. Oxide penetration in the Hastelloy X after 10,000 hrs at 1750°F was approximately 0.002 in. All other candidate materials were penetrated more than 0.006 in. except Inconel 604 which was comparable to Hastelloy X. Hastelloy X exhibited approximately 20% greater post-exposure strength than the Inconel 604. Based on these evaluations, Hastelloy X was chosen as the reference cladding material.

- b. AGN TM-361*, "AGCRSP Materials Evaluation of Nickel-Base Alloys," January 1961

This document covers the results of tests performed on candidate cladding materials exposed at 1750°F in 200 psi reference gas (99.5% N₂ + 0.5% O₂) and 300 psi air for durations up to 5000 hrs. Materials evaluated were Hastelloy X, Hastelloy R-235, INOR 8, Inconel, Inconel X and Inconel 750. Screening parameters included corrosion penetration, short-time room temperature tensile tests, and short-term creep rupture tests. (See AN-ENGR-19 which presents the results of the continuation of this materials program through 10,000 hr exposure at 1750°F in reference gas and air.)

The results of gas corrosion testing indicate that Hastelloy X when exposed to reference gas (200 psi) at 1750°F for 5000 hrs can be expected to lose 0.4 mg/cm² (0.002 in. deep oxide penetration); the best of other candidate materials exhibited a net weight gain of about 4 mg/cm² (0.009 in. deep oxide penetration) which indicates that Hastelloy X has superior oxidation resistance. The results of 1000 hr creep testing indicated that the elongation of Hastelloy X at 1650°F and 1850°F and under a 1000 psi stress

was less than 50% of that of the other materials. Short-term tensile properties of Hastelloy X were similar to other materials and satisfied the basic strength requirement.

5. Demonstration of Cladding Acceptability

- a. AGN TM-401*, "Compatibility Tests of Hastelloy X in Oil-Contaminated Gases," January 1963

Describes the results of tests performed to determine the effect of oil, carried in the coolant gas, on the Hastelloy X cladding material at ML-1 operating conditions. Results indicate that Hastelloy X is not adversely affected when exposed to oil vapors carried by the oxygenated nitrogen reference coolant gas at ML-1 operating conditions.

- b. AGN TM-413*, "Isothermal Oxidation of Hastelloy X at 1750°F and 1850°F in Air," December 1965

Describes the experimental procedure, data analysis, and results of work performed to obtain the temperature and alloy composition dependence of the Hastelloy X oxidation rate; activation energies and oxidation rate constants are derived for different alloy compositions from the experimental results.

- c. AGN TM-414*, "Studies of Hastelloy X Corrosion in Air at 1900 and 2000°F," December 1965

Presents an evaluation of the corrosion resistance of Hastelloy X sheet and tubing materials at 1900 and 2000°F in air to establish limiting lifetime/temperature characteristics of the reference cladding material. Results of the evaluation of sheet specimens indicated that oxidation and oxide spalling were accelerated between 1900 and 2000°F after exposure for 5000 hr. The maximum weight losses during this period were 4.2 mg/cm² and 17.2 mg/cm² for the 1900 and 2000°F specimens respectively. Average depth of oxide penetration was 0.0045 in. (1900°F) and 0.0135 in. (2000°F). The ultimate tensile strength of both the 1900 and 2000°F specimens was essentially the same for all exposure periods from 100 to 5000 hr (10,000 to 11,000 psi for 1900°F specimens and 8,500 to 10,000 psi for 2000°F specimens). The evaluation of tubing specimens was not complete at the time of program termination.

- d. AN-ENGR-19, "Evaluation of Metal Alloys for Cladding the Fuel Elements in a High Temperature Nitrogen-Cooled Reactor," June 1961

Presents a general summary of materials evaluation activity in conjunction with fuel cladding selection. (Presented at the American Nuclear Society Meeting in Pittsburgh, Pa., 5-7 June 1961) Materials initially screened were Hastelloy X, Inconel, Inconel 702, Inconel X, INOR 8, and Hastelloy R-235, all of which are nickel-based alloys and cover the spectrum of solid solution

strengthened and precipitation-hardened alloys. Initial screening included tests to determine corrosion resistance, metallurgical stability, and creep strength for 100 hrs and 2500 hrs at 1750°F in both reference gas (99.5% N₂ + 0.5% O₂) and air. Based on the results of these tests, Hastelloy X was tentatively selected as the reference cladding with Inconel and Inconel 702 selected for backup. Continued exposure at 1750°F for up to 10,000 hr confirmed the suitability of Hastelloy X. Results obtained show 10,000 hr reference gas oxide penetration depths of 0.0020 in. and 0.0075 in. for Hastelloy X and Inconel, respectively. Inconel 702 was found unacceptable because of susceptibility to nitriding, resulting from the aluminum and titanium constituents. Hastelloy X exhibited the best room temperature tensile properties of the three materials after 10,000 hr aging at 1750°F.

- e. AN-AGCR-48, "Preliminary Evaluation of Europium Oxide-Stainless Steel Burnable Poison Foil," February 5, 1962

Presents the results of preliminary studies and tests performed to define potential problem areas in the fabrication and operation of ML-1-II type burnable poison. Establishes the requirements for additional developmental and experimental efforts.

- f. AN-AGCR-58, "Materials Considerations Associated with Flooding and Transporting ML-1 Fuel Elements," 6 April 1962

Presents the theoretical and experimental evidence that the Hastelloy X fuel cladding can withstand the thermal shocks and stresses associated with post-operative core flooding; suggests some out-of-pile tests to determine the thermal shock resistance of the copper-cadmium burnable poison to similar conditions. Specifically, it was shown that the fuel pin temperature will be 420°F (max.) at the time of quenching with 212°F water. Two modes of possible failure could result from these conditions: spalling of the protective oxide film, which would accelerate future corrosion, and excessive thermal stresses, which would deform the fuel pin. Laboratory tests which quenched the fuel pins from 1850°F to 1000°F within 3 seconds (more severe than flooding condition) did not reveal an increase in oxide spalling nor an increase in oxidation rate. The fuel pin thermal stress due to flooding was calculated to be about 11,500 psi. This is well below the room temperature yield strength of Hastelloy X (49,000 psi) after 10,000 hrs exposure at 1750°F.

- g. AN-AGCR-216, "Long Term Air Corrosion of Several Nickel-Based Alloys," 18 May 1962

Summarizes the results of out-of-pile materials tests performed at AGN through 1961; discusses microstructural changes and oxide penetration of several nickel based alloys at elevated temperatures in an air environment. (Presented at the 11th Annual AEC Corrosion Symposium, May 23-25, 1962, Brookhaven National Laboratory.)

- h. AN-AGCR-433, "Trip Report: Inspection of Selected ML-1-I Fuel Elements at NRTS," January 7, 1963

Describes the in-field fuel element inspection after the first sustained reactor operation (ANSOP 16615). Nine elements were inspected. The condition of all visible components, upper spider, hanger, and outer liner, appeared to be satisfactory and only minor discolorations were noted.

- i. AN-AGCR-434, "Inspection of ML-1 Reactivity Shims: Trip Report for Visit to ML-1 Site, NRTS (3 and 4 January 1963)," January 8, 1963

Describes the in-field inspection of one silver reactivity shim after the first sustained reactor operation (ANSOP 16615). The appearance of the shim indicated good silver adhesion, some scratches in the silver, and some discoloration over both the silver and stainless steel areas of the shim.

- j. AN-AGCR-607, "Trip Report - Inspection of ML-1 Fuel Elements," October 18, 1963

Describes the condition of the 61 fuel elements after the second period of sustained reactor operation (ANSOP 16625). Five of the elements were seen to have failed lower spiders; all 61 lower spiders were heavily corroded.

- k. AN-AGCR-620, "Trip Report: Metallurgical Evaluation of ML-1-I S/N-11 Spiders and Outer Liner, IB-17R-1 and IB-14R Samples," December 6, 1963

Describes the results of metallurgical evaluation of a typical, failed ML-1 lower spider and indicates that the primary mode of failure was due to a sub-standard alloy composition. (See also AN-AGCR-662, Section 1.) The cladding of the IB-17R-1 and IB-14R fuel pins after irradiation (1179 hrs and 3700 hrs, respectively) was seen to be in excellent condition with no indication of embrittlement or excessive oxidation.

- l. AN-AGCR-628, "Material Selection, Procurement, and Evaluation of ML-1-I Upper and Lower Spiders," December 16, 1963

Presents a detailed review of the selection and procurement of the ML-1 lower spiders which failed during reactor operation due to sub-standard material; also reviews the upper spider selection, procurement and evaluation activities which indicate this component to be acceptable for continued reactor operation.

- m. AN-AGCR-675, "ML-1 Reactivity Shims, March 26, 1964

Describes the hot cell evaluation of reactivity shims, both silver plated and unplated, which had operated in the ML-1 reactor for approximately 100 full power hours. Presents the program for design and development of a modified reactivity shim.

- n. AN-AGCR-717, "ML-1-II Burnable Poison Development - Final Report," June 30, 1964

Presents the results of development and testing of the Eu_2O_3 -2 TiO_2 /stainless steel cermet burnable poison for use in ML-1-II elements. Results indicate that the titania stabilized europia/Type 304L cermet core clad by Type 304L stainless steel using the picture frame technique is completely acceptable for use in the ML-1 core; techniques for production inspection are included.

- o. AN-AGCR-723, "ML-1 Fuel Element Inspection, June 1964," July 6, 1964

Describes the in-reactor inspection of 2 of 6 fuel elements which had operated approximately 50°F hotter than predicted. The exterior of the elements were seen to be in excellent condition; internal inspection was not performed. The condition of the elements did not suggest a cause of or adverse effects from the higher than expected temperature.

- p. AN-AGCR-764, "Operation of ML-1-I Fuel Elements at 1800°F Cladding Hot Spot Temperatures," September 14, 1964

Describes the results of laboratory tests performed on Hastelloy X tubing in air for 7500 hr at 1800°F; mechanical properties and gas corrosion results indicate that fuel cladding temperatures of 1800°F would not degrade the integrity of the cladding. Specifically, it was shown that the 5000 hr maximum oxide penetration would be about 0.001 in. greater when exposed at 1800°F than when exposed at 1750°F. The room temperature ultimate and yield strengths of Hastelloy X aged for 5000 hrs is about 1% to 4% less when exposed at 1800°F than when exposed at 1750°F.

- q. AN-AGCR-880, "Creep-Rupture Tests of Finned and Smooth-Walled Hastelloy X Pin Tubing," May 21, 1965

Describes the results of elevated temperature rupture tests conducted on "as-received" Hastelloy X tubing, both finned and smooth walled; comparison is made with other experimental and theoretical work. Finned tubing stress concentration was not detected. Tests were conducted on unaged smooth and finned tubing at temperatures of 2000, 2100, and 2200°F; internal fuel pin gas pressure was varied between 100 psi to 500 psi. The 10-hour rupture strength for 2000, 2100, and 2200°F were 1600 psi, 1000 psi, and 750 psi, respectively; data fell within 10% of straight line correlation. No differences in the rupture characteristics of finned and smooth tubes were detected.

- r. AN-AGCR-939, "ML-1-I Fuel Element and Reactivity Shim Inspection, 13-15 September 1965," October 4, 1965

Describes the condition of selected fuel elements and reactivity shims visually inspected after approximately 3000 full power hours of operation. The fuel elements appeared to be capable of continued operation while the silver shims appeared to be corroding.

- s. AN-AGCR-948, "Aging of Hastelloy X in an Inert Gas Atmosphere," October 1965

Presents the partial results of a prematurely terminated experimental program to separate the effects of oxidation and microstructural changes on the mechanical properties of Hastelloy X sheet and tubing at temperatures of 1300 and 1850°F exposed for up to 5000 hours.

- t. AN-AGCR-952, "ML-1-I Fuel Pin Fill Gas Dilution Studies," 2 November 1965

Presents the results of laboratory tests performed to determine the mechanism of fill gas dilution seen in some ML-1-I fuel pins operated for about 100 hours under reactor conditions. Presence of CO₂ in the fill gas is shown to be temperature dependent and probably introduced by the thermal decomposition of acetone and alcohol used as cleaning agents during fabrication.

- u. AN-AGCR-953, "Evaluation of Hastelloy X ML-1 Tubing (Heat E-9517) Exposed in Air for Durations up to 10,000 Hour," November 4, 1965

Describes the results of tests performed to evaluate the capability of the low cobalt content Hastelloy X fuel cladding material used in ML-1-I fuel elements to operate for up to 10,000 hr in air at temperatures up to 1800°F. Results indicate that the maximum oxide penetration is 0.0043 in. after 10,000 hrs at 1800°F; 10,000 hr exposure at 1750°F results in maximum oxide penetration of 0.0037 in. confirming earlier data which indicated only a slight oxidation increase between 1750°F and 1800°F. Microhardness tended to decrease with increasing temperatures. Room temperature ultimate and yield strengths after 10,000 hr exposure were reduced by about 15% between exposure temperatures of 1300 and 1800°F.

- v. AN-AGCR-956, "Centrifugal Creep Testing of Hastelloy X; Status Report," November 18, 1965

Presents the preliminary results of centrifugal creep tests to determine the creep properties of Hastelloy X at temperatures up to 1900°F for time durations up to 5000 hours; stress levels ranged from 640 psi to 1460 psi. Summarizes the status of the experiment at the time of program termination.

- w. AN-AGCR-959, "Fuel Pin Tubing Control Tests," November 1965

Presents the results of laboratory tests performed on ML-1-II type Hastelloy X cladding material in support of the ML-1-II prototype fuel element irradiation program. Includes results of oxidation aging, and mechanical properties evaluation for durations up to 10,000 hr in air at temperatures from 1300 to 1800°F. Maximum oxidation penetration was 0.0031 in. for 5820 hrs at 1750°F. This compares quite favorably with a maximum observed oxide penetration on the IB-17R-2 of 0.003 in. (about 1700-1750°F for 5820 hrs, in-pile). The 10,000 hr oxide penetration was 0.0045 in. at 1750°F and 0.0050 in. at 1800°F. As in previous tests, the ultimate and yield strengths tended to assume a single value at exposure temperatures of about 1800°F and higher. The ultimate strength of as-received material was about 15,600 psi at 1750°F and about 11,500 psi at 1800°F.

6. Fuel Evaluation

- a. AN-ENGR-39, "Effect of High Temperature In-Pile Irradiation Upon High Density UO_2 Bodies," undated

Presents the results of the capsule irradiation of 100% UO_2 ceramic fuel in the Battelle Research Reactor; burnup of the BRR-GCR-3 capsule was equivalent to 8500 full power ML-1 hours. The post-irradiation condition of the fuel showed cracking which did not appear to lessen the operational capabilities. (Also see IDO-28582.)

- b. BMI-1484*, "A Study of the Radiation Stability of UO_2 Dispersions in BeO," 1960

Summarizes the results of the BRR-GCR-2 capsule irradiation experiment in which 56 wt% UO_2 in BeO was irradiated for 1152 hrs.

- c. IDO-28582, "Capsule Irradiation of UO_2 Fuel for the ML-1," June 1962

Presents the results of the 100% UO_2 capsule irradiation experiment designated BRR-GCR-3. The fuel was severely cracked after the equivalent of 8,500 full power ML-1 hours but appeared fully capable of continued operation. Values for fission gas release and burnup are presented.

- d. IDO-28600*, "High Temperature Irradiation of UO_2 -BeO Bodies," January 1963

Presents the results of the MTR-GCR-4 capsule irradiation of 70 wt% and 80 wt% UO_2 in BeO ceramic bodies. The 70 wt% bodies exhibited excellent irradiation stability while central melting occurred in one of the 80 wt% bodies. Values for fission gas release and burnup are presented.

- e. AN-AGCR-719, "MTR-GCR-4 Capsule Test Final Report," July 1964

Describes the MTR-GCR-4 capsule irradiation of 75 wt% UO_2 in BeO which failed due to the loss of the NaK heat transfer media. No results pertaining to the radiation stability or fission gas release of the fuel were obtained due to the failure of the auxiliary equipment.

7. Spider Tests

- a. AN-AGCR-255, "ML-1-II Spider Load Test," July 5, 1962

Presents the results of mechanical load tests performed on the ML-1-II upper and lower spiders. Results indicate that the ML-1-II spiders are more than twice as strong as the ML-1-I spiders; failure occurred at approximately 2700 lb, uniformly distributed over all 19 points of loading.

- b. AN-FDD-108, "Upper Spider Load Tests - ML-1, IB-2L," October 5, 1960

Presents the results of mechanical load tests performed on both ML-1-I and IB-2L upper spider assemblies. Initiation of failure was noted at uniform loadings of 1250 lb and 640 lb for the ML-1-I and IB-2L upper spiders, respectively.

8. Thermal Tests

- a. AN-AGCR-464, "ML-1-II Fuel Pin Thermal Cycle Tests," February 1963

Presents the results of thermal cycle tests performed on ML-1-II fuel pins and upper and lower spiders. The components were subjected to 250 thermal cycles from 200°F to 1650°F at 250 psig reference ML-1 coolant gas; maximum quench rate was 15°F/sec. Mechanical tests performed on the components after cycling indicated that no component damage or strength reduction had occurred.

- b. AN-AGCR-580, "Thermal Shock Tests, ML-1-II Element," August 30, 1963

Describes the experimental procedure and presents the results of laboratory thermal shock tests performed on a prototype ML-1-II fuel element. One series of tests imposed 285 thermal shocks on the element from a peak temperature of 1400°F; maximum fuel pin quench rate was 38°F/sec. A second series of tests quenched the element from 1585°F at a maximum quench rate of 100°F/sec; 10 such cycles were performed. Component discolorations and fuel pin bowing were the only effects noted as a result of test conditions. (Also see AN-AGCR-586.)

- c. AN-AGCR-585, "Measurement of ML-1-II Fuel Element Cooldown Rate," September 11, 1963

Presents the experimentally obtained results of cladding cooldown rates as experienced by the IB-17R-2 in-pile ML-1-II fuel element prototype in the AGN-GETR test loop. Maximum cooling rate was 40°F/sec from an initial steady state temperature of 1400°F. (See also AN-AGCR-595.)

- d. AN-AGCR-586, "Thermal Shock Tests, ML-1-II Element, Supplemental Report," September 24, 1963

Presents a detailed mechanical analysis to explain and correct the fuel pin bowing observed during the examination of the fuel element used in the thermal shock experiments. (See also AN-AGCR-580.)

- e. AN-AGCR-595, "Measurement of ML-1-II Fuel Element Cooldown Rate (Additional Data)," October 4, 1963

Presents additional experimentally obtained results of cladding cooldown rates as experienced by the IB-17R-2 in-pile test element in the AGN-GETR test loop. Maximum cooling rate was 75°F/sec from an initial steady state temperature of 1600°F. (See also AN-AGCR-585.)

- f. AN-FDD-106, "Thermal Ratcheting Tests," September 29, 1960

Presents the results of laboratory thermal cycling tests on ML-1-I fuel pin assemblies. Components were subjected to 290 thermal cycles between 200 and 1650°F under 300 psig reference ML-1 coolant gas. The integrity of the fuel pins assembly was not affected by the test conditions; fuel cracking, but not displacement, was observed.

9. Depressurization Tests

- a. AN-FDD-64, "Depressurization Test on IB-10M' and IB-10M Models," June 21, 1960

Presents the results of laboratory tests performed to determine the effect of rapid depressurization on the integrity of ML-1 type fuel elements. The results indicate that pressure equalizing holes should be placed in the inner liner to prevent liner collapse during maximum depressurization, about 190 psi/sec.

10. Shock and Vibration Tests

- a. AN-AGCR-91, "ML-1-I Fuel Element Drop Test," April 19, 1962

Presents the results of laboratory tests conducted to determine the response to and the effects of various accidental impact loads imposed on the ML-1-I fuel element. Results indicate that

the element is fairly susceptible to minor damage due to vertical shock loads greater than a square wave equivalent of 150 g's at about 0.1 milliseconds.

- b. AN-AGCR-505, "ML-1-II Shock and Vibration Tests," April 1963

Describes the test procedure and results of a series of tests designed to prove the integrity of the ML-1-II fuel element under conditions of transportational and operational shock and vibration. All aspects of the test were successful. Resonant frequencies and responses to given vibratory and impact loads are given. Resonant frequencies were found at 220 cps (longitudinal) and at 74 and 450 cps (transverse); amplification factors at each resonant frequency were found to be 1, 12, and 23, respectively. The element was vibrated at each resonant frequency for 30 minutes with a 5 g input; no damage was sustained. The element was subjected to impact loading in both axis up to 35 g's (0.050 sec duration) with no damage.

- c. AN-AGCR-529, "Transportability of Irradiated ML-1 Fuel Elements," May 1963

Summarizes all available shock and vibration data on ML-1 type fuel elements, both irradiated and unirradiated, and concludes that irradiated ML-1 type fuel elements can be transported within the reactor without the possibility of mechanical damage to the elements.

- d. AN-AGCR-654, "ML-1-II Fuel Element - Mechanical Testing," July 1964

Presents a summary of all ML-1-II fuel element mechanical testing; includes spider loading, fuel pin and spider thermal cycling, inner liner tensile/compressive shock and vibration, and fuel element thermal shock tests.

- e. AN-FDD-109, "Shock and Vibration Test: ML-1 Prototype Fuel Element," 5 October 1960

Describes the test procedure and results of shock and vibration testing performed on the ML-1-I fuel element. Test conditions were those as specified in the Military Characteristics for the ML-1 system and those analytically shown to exist as a result of fuel element operation. Resonant frequencies and responses to given vibratory and impact loads are given.

- f. AN-FDD-125, "ML-1 Test Program: Summary of Results," 14 December 1960

Presents a summary of all ML-1-I fuel element mechanical tests; includes depressurization, thermal ratcheting, shock and vibration and spider load tests.

11. Heat Transfer Tests

- a. AN-AGCR-223, "Local Film Coefficients on the ML-1 Fuel Elements Due to Boundary Layer Turbulence Promoters," June 8, 1962

Presents the results of laboratory heat transfer tests to determine the local film coefficients due to rectangular fins on fuel pins using the mass/heat transfer analogy. (Presented to ANS 8th Annual Meeting, June 1962, Boston, Mass.)

- b. AN-AGCR-235, "Heat Transfer and Fluid Friction Characteristics of Tube Clusters with Boundary Layer Turbulence Promoters," 20 July 1962

Presents the results of laboratory heat transfer and hydraulic tests performed on simulated ML-1 type 19 pin clusters using various turbulence promoter finning configurations. Colburn and friction factors are calculated from the experimental results. (Presented at ASME Annual Meeting, November 1962, N.Y.)

- c. AN-AGCR-383, "CROC-90; A Fluid Flow Data Reduction Code for the IBM-7090 Computer," 26 November 1965

Describes the programming and operation of a Fortran computer code for reducing out-of-pile ML-1 type fuel element fluid flow data. Friction factors, entrance and exit coefficients, and orifice coefficients are calculated from experimental data.

- d. AN-AGCR-391, "Flow Control Orifice Calibration for the ML-1-II Core," 4 December 1962

Presents the results of laboratory hydraulic tests conducted on an ML-1-II prototype to calibrate the flow control orifices.

- e. AN-AGCR-637, "Calibration of Flow Control Orifices for ML-1-I Core with Scalloped 18 Hole Spiders," 6 January 1964

Presents the results of laboratory hydraulic tests conducted on an ML-1-I fuel element model to calibrate the flow control orifices used with the modified lower spider.

- f. AN-AGCR-919, "Mass Transfer Studies of Shaped Turbulence Promoter Performance," July 1965

Presents the results of heat transfer tests using the mass/heat transfer analogy to determine the optimum turbulence promoter fin configuration. Summarizes previous work performed on rectangular fins and presents the latest results of ramped rectangular fins and various shapes of grooves. Fins with downstream ramps about four fin heights long produced the optimum heat transfer coefficient.

- g. AN-FDD-135, "Flow Control Orifice Calibration of the ML-1 Prototype Element with Redesigned Orifice Fixture," 3 January 1961

Presents the results of laboratory tests performed to determine the orifice coefficients of the "as-built" ML-1-I fuel elements.

12. Fuel Pellet Fabrication

- a. AN-FDD-102, "Fabrication of UO_2 -BeO Ceramic Fuel Pellets for the IB-2L Core," 8 September 1960

Describes the development and production of the UO_2 -BeO fuel for IB-2L core. Presents correlations between processing parameters and final product. (The same process was used in the ML-1-I core.)

- b. FAB-59, "Fabrication of UO_2 -BeO Ceramic Fuel Pellets for the ML-1 First Core," June 7, 1961

Describes the production of the ML-1-I ceramic fuel. Correlates process variables and final product; discusses production problems and rates.

13. Fuel Element Assembly

- a. AN-AGCR-21, "Fuel Pin Helium Back-Fill Procedure, " 27 February 1962

Presents the analysis which is the basis for the technique used to helium back-fill the ML-1 type fuel pins prior to final closure welding.

- b. AN-AGCR-29, "Fuel Element Assembly Studies," 16 March 1962

Presents the results of laboratory tests to determine the feasibility of assembling the ML-1-II fuel element in two independent steps. Results of this work aided in the selection of fabrication tolerances.

- c. AN-AGCR-97, "ML-1-II Insulation Friction Tests," 24 April 1962

Presents the result of laboratory tests performed to determine the friction buildup between the inner and outer liners during operational induced differential thermal expansion. Presents the basis for the insulation wrapping technique specified for the ML-1-II element.

- d. AN-AGCR-98, "Fuel Pin Spacer Wires," 25 April 1962

Presents the recommendation for allowing slack in the spacer wires to eliminate spiral fuel pin distortion due to fuel pin/spacer wire differential expansion during operation. (See also AN-AGCR-100.)

- e. AN-AGCR-100, "Spiral Bending of the IB-8T-1 Fuel Pin,"
27 April 1962

Describes the analysis performed to understand fuel pin spiral deformation observed in the IB-8T-1 in-pile prototype fuel element. Basis for placing slack in the spacer wire is to eliminate fuel pin/spacer wire interaction. (Also see AN-AGCR-98.)

- f. AN-AGCR-463, "Joint Designs Between the Structural Members of the ML-1 Second Core Fuel Elements," 18 February 1963

Summarizes the process development work performed in the design and application of joints and joining techniques used in the ML-1-II fuel element.

- g. AN-AGCR-888, "Fuel Pin Tubing Quality Control Inspection Techniques," 28 May 1965

Summarizes previous and current methods of inspecting fuel pin tubing at AGN and compares the various NDT methods available for tubing inspection.

- h. AN-FDD-242, "IB-14R. Welding and Brazing Process Development,"
25 October 1961

Summarizes the work performed in the development of welding and brazing of instrumented fuel pin end plugs. Presents the process parameters for brazing thermocouple sheaths to end plugs and for welding end plugs to fuel pin tubing; rework parameters are also described.

14. In-Pile Tests

- a. IDO-28616*, "In-Pile Test of Prototype ML-1 Fuel Elements,"
June 1964

Presents the results and describes the test program and equipment design of the IB-8T-1, -2 prototype ML-1-I element irradiated in the AGN-GETR loop. Includes the results of operation and post-irradiation hot cell disassembly and metallurgical evaluation of components exposed to up to 10,151 hours of ML-1 full power operation.

- b. AGN TM-404*, "Defective Fuel Element Testing for the ML-1-I,"
September 1963

Describes the equipment, procedure, and results of in-pile irradiation tests performed to determine the effect of defective fuel cladding on the stability, integrity and fission gas release characteristics of ML-1-I fuel pins. These tests, designated IB-7T were conducted in the GETR facility. Predicted fission gas release rates based on theoretical considerations show reasonably good agreement with actual measured release rates.

- c. AGN TM-409, "IB-17R-2 In-Pile Test," 26 August 1964

Presents the results of the IB-17R-2 in-pile ML-1-II prototype fuel element irradiated for 5819 full power hours. Includes a description of test equipment, operation, and post-irradiation hot cell disassembly and metallurgical evaluation of all fuel element components.

- d. AN-ENGR-43, "A Theory for Calculating the Release of Fission Gases from Defective Sheathed Ceramic Fuel Elements," undated

Presents a theory for predicting fission gas release rates due to defective fuel cladding material. Calculated and experimentally determined release rates show good agreement.

- e. AN-AGCR-63, "Permissible Number of Defective Fuel Pins in the ML-1 Core Based on Allowable Radioactive Leakage," 2 March 1962

Presents the analysis which indicates that the minimum number of defective fuel pins which could be radiologically tolerated would be 455 during plant operation. The analysis is based upon IB-7T defective cladding release data and expected ML-1 system leak rates.

- f. AN-AGCR-655, "IB-14R In-Pile Test Final Report," March 1964

Presents the results of in-pile testing of a prototype element of the ML-1-II preliminary design. The element was irradiated for 3700 full power hours and was designed to obtain basic heat transfer and hydraulic data to be used in the final design of the ML-1-II element. Problems encountered with the design of the hot junction of the fuel pin thermocouples prohibited a detailed thermal evaluation of the turbulence promoters. The thermocouple hot junction was redesigned for use in the IB-17R test series. Cursory thermal analysis indicated that the turbulence promoters increased heat transfer, compared to smooth tubing, but not as much as predicted. Fuel element pressure drop was 1.5 psi greater than observed in the IB-8T test element.

- g. AN-AGCR-697, "IB-17R-1 In-Pile Test," June 1964

Presents the results of an 1179 full power hour irradiation of a prototype ML-1-II fuel element in the AGN-GETR loop. Heat transfer and hydraulic characteristics obtained during irradiation are summarized. Results indicate that the turbulence promoters decrease the maximum cladding temperature when compared to smooth tubing. The amount of the decrease was somewhat smaller than predicted. The thermal analysis was made difficult by thermocouple failures, instrumentation problems, and extreme data scatter. The pressure drop through the element remained constant throughout the test although it was slightly greater than predicted. This may have been due to a faulty pressure instrument which failed completely shortly before the end of the test. The fuel was cracked

more severely than observed in previous UO_2 -BeO fuel irradiations. The fuel cladding and neutronic performances were in excellent agreement with pre-test predictions.

- h. AN-AGCR-892, "Post-Irradiation Metallurgical Evaluation of the IB-17R-3 Defective Fuel Pin," May 1965

Describes the hot cell evaluation of the failed fuel pin from the IB-17R-3, ML-1-II prototype, fuel element which failed after 9501 full power hours in the AGN-GETR loop. Concludes that a defect, originally present in the fuel pin tubing, was the primary mechanism for cladding failure.

- i. AN-AGCR-950, "Summary and Status Report: IB-17R-3 In-Pile Test," 30 November 1965

Summarizes the operation, failure analysis, and present condition of the IB-17R-3 in-pile ML-1-II prototype test element which failed due to a defective fuel pin which allowed fission gases to escape into the AGN-GETR loop after 9501 full power hours of testing.

3. Reactor Design

a. General

The reactor for the ML-1 was enriched UO_2 -fueled, water-moderated and gas-cooled. The major components of the reactor were the calandria-type pressure vessel, the reflectors and solid shielding, the control rods, the moderator water system and the liquid shielding system. Figure 13 shows the elevation section of the reactor skid; the general design characteristics are presented in Appendix C.

b. Pressure Vessel Assembly

The ML-1 pressure vessel consisted of a calandria with gas inlet and outlet plenums which were integral with the calandria tube sheets (Figure 14). The calandria consisted of 61 AISI Type 321 stainless steel pressure tubes rolled and seal-welded into AISI Type 304 stainless steel tube sheets and plenums. The pressure vessel was completed by a permanently-installed AISI Type 304L stainless steel lower cap and a removable (to allow refueling) AISI Type 304 stainless steel upper cap (Figure 15). Both caps accommodated ducts to route the coolant gas between the reactor and the power conversion equipment. Stainless steel clad tungsten axial shielding baffles were supported in each plenum. Stainless steel clad lead and tungsten reflectors and silver-alloy control rods were mounted outside of the pressurized envelope. Fuel elements were supported in the pressure tubes on machined seats in the upper tube sheet. The entire pressure vessel assembly was submerged in a liquid shield (see later discussion).

The design of the ML-1 pressure vessel was based on the experience gained in the design and testing of the GCRE reactor calandria. The major developmental work involved the selection of materials and the development of suitable stress analysis techniques⁽¹⁾ to permit the design of an assembly which satisfied all operational requirements and, at the same time, weighed as little as possible. Some development was also required for the design of the tube-to-tube sheet joint and for the closure of the removable upper cap.

The major modifications made to the pressure vessel are summarized below:

- During initial power plant testing, it was observed that the temperature in the upper tube sheet was unacceptably high. An evaluation of the problem led to the conclusion that hot (800°F) inlet gas was bypassing the fuel element through imperfect seals at the fuel element-to-upper tube sheet seat. A modification made to the seal and the insertion of spring-loaded holddown fuel element devices in the upper baffle corrected the problem.⁽²⁾

- A leak in the pressure vessel developed as a result of the failure of the seal weld on a single (of 18) shear pin which connected the lower plenum cap to the lower plenum. The design of the seal weld was modified to prevent recurrence and this modification was completed on all 18 welds.⁽³⁾

FIGURE 13. ML-1 REACTOR SKID (Section)

SCALE
Dimensions in feet

FIGURE 14. ML-1 REACTOR CALANDRIA

11.90-66-489

FIGURE 15. ML-1 REACTOR PRESSURE VESSEL

- The seal weld between the lower plenum cap and lower plenum wall failed. A modified omega seal was installed over the failed seal weld.⁽⁴⁾
- A small leak developed in the flanged joint in the outlet gas duct; no attempt was made to repair this leak because, although the repair consisted only of replacing a gasket and retorquing the flanged connection, the interruption of the test program was unjustified. A special gas removal system was installed to prevent the leaking gas from interfering with the cooling (by shield solution) of the plenum cap.⁽⁵⁾
- A very small leak developed in the lower tube sheet region in the summer of 1965. The program⁽⁶⁾ to identify the location and nature of this leak was interrupted by the AGCRSP termination.

The design of an improved pressure vessel based on the experience gained during fabrication and test operation of the ML-1 was developed.⁽⁷⁾

c. Water Moderator System

The moderator water system consisted of the piping, pumps, heat exchangers, surge tanks, and demineralization equipment required to circulate the demineralized moderator water to remove heat and maintain water purity. The main circulation pump discharged water through a small water-to-water heat exchanger to cool the shield water and then to the reactor core region where it flowed upward in the voids surrounding the pressure tubes and was forced through the internal cooling passages in the upper and lower tube sheets (see Figure 14). The water was collected near the top of the calandria and flowed through the moderator cooler on the power conversion skid where it was cooled and returned to the pump. A continuous bypass flow was circulated through a mixed-bed demineralizer to maintain water purity at greater than 500,000 ohms/cm³ resistivity (in this condition, if all the impurities were chlorides, the chloride concentration would be less than 1 ppm). During periods of standby operation, a 15 gpm pump maintained flow to remove core afterheat.

Although no provision was made to deoxygenate the moderator water in the original ML-1 design, experience with the GCRE and on other programs indicated that such processing was desirable until the results of a test program to evaluate the necessity for deoxygenation were available. Consequently, after suitable investigation, a "non-prototype" deoxygenation system was designed, fabricated and installed as an auxiliary to the ML-1 reactor skid.⁽⁸⁾ Meanwhile, a test program was conducted (from August 1962 until June 1964) to provide quantitative data which resulted in the determination that no requirement existed for deoxygenation of the moderator water.⁽¹⁰⁾

d. Shield Solution System

The entire ML-1 pressure vessel was submerged in a liquid shield which was contained in an aluminum tank. The shield tank was the basic reactor skid structure; it contained and supported the pressure

vessel and gas ducting assembly, the solid shielding, piping for the moderator and shield solution systems, the moderator surge tank, the shield water heat exchanger, the nuclear instrumentation ion chambers and the control rod drives. The moderator and shield solution pumps, the moderator water demineralizer, the shield water pump, the control actuators and the ultimate shutdown device were mounted exterior to the shield tank. The shield solution was (nominally) 10% boric acid in demineralized water. The solution was drained before transport of the power plant. During operation, the boric acid solution was circulated and heat was transferred to the moderator water in a heat exchanger mounted inside the shield tank.

e. Ultimate Shutdown Device

A shutdown device was provided for the ML-1 reactor to permit reactor shutdown from normal operation in the event of a malfunction of all control rods.⁽¹¹⁾ A poison, in the form of boric acid, was available for injection into the core region water moderator from a pressurized container mounted on the outside of the shield tank by actuation of an explosive-type valve.

REACTOR DESIGN BIBLIOGRAPHY

1. AGN TM-384, "The Thermoelastic Analysis of the ML-1 Tube Sheet and Plenum Chamber," February 1961

Presents the results of the development of analytical techniques to predict stresses in the ML-1 pressure vessel and the results of evaluation of the pressure vessel tube sheets, plenum walls and pressure tubes.

2. Tube Sheet Modification

- a. AN-AGCR-477, "Bypass Flow Reduction Design Report," March 1963

Describes the tube sheet temperature problem, the results of investigations to determine causes of the heating and the design and development of the corrective modification.

- b. AN-AGCR-517, "Completion of Thermoelastic Analysis of the ML-1 Modified Upper Baffle," 26 April 1963

Presents a complete analysis of stresses in the upper baffle, as modified to accommodate the bypass flow modification (see AN-AGCR-477). This more sophisticated stress analysis shows the stresses to be generally lower and more uniformly distributed than determined by the preliminary analysis.

3. AGN TM-405*, "ML-1 Pressure Vessel Repair Summary Report," December 1963

Describes the work performed to disassemble the reactor, repair the lower cap shear pin seal welds and reassemble the reactor. The pressure vessel was removed from the reactor skid, the leak path between the gas and moderator system was identified, the defective welds were repaired and the reactor was reassembled with modifications made to permit subsequent replacement of the pressure vessel, if necessary. The repair work was performed between March and September of 1963.

4. Lower Cap Seal Weld Repair

- a. AN-AGCR-922, "ML-1 Lower Cap Failure Analysis," 12 July 1965

Presents temperature and stress profiles calculated for the lower cap and plenum wall to determine the cause of the seal weld failure and to provide a basis for design of modifications. The failure resulted from mechanical fatigue after 45 cycles of 0.015 in. movement between the lower cap and plenum wall. Thermal and stress analysis of the modified design are included.

- b. AN-AGCR-930, "Summary Report of ML-1 Reactor Pressure Vessel Lower Cap Seal Weld Repair," September 1965

Describes the repair of the ML-1 pressure vessel lower cap seal weld. An "omega-type" seal membrane was demonstrated to withstand twice as many thermal cycles as the "trepan-type" joint by laboratory testing and was installed with the reactor in the fully-fueled condition. Modifications to improve the heat transfer between the lower cap and the shield solution to reduce the movement between the lower cap and plenum wall are described.

5. Lower Gas Duct Leak

- a. AN-AGCR-909, "ML-1 Lower Gas Duct Flange Leak," 7 July 1965

Summarizes the evaluation of a gas leak in the ML-1 pressure vessel-to-lower gas duct flange. The leak rate was in the order of 1-2 scfh and originated in the lower half of the flexitallic gasketed joint.

- b. AN-AGCR-921, "Design Report, ML-1 Shield Solution Gas Removal System," 26 July 1965

Presents the design of a system to remove gas from the lower gas duct flange leak to prevent accumulation on the lower surface of the pressure vessel lower cap. A pump continually removes shield solution and gas from below the lower cap; a gas disengager separates and vents the gas and the shield solution is returned to the shield tank.

- 6. AN-AGCR-960, "Interim Report - ML-1 Reactor Disassembly and Inspection Program," December 1965

The program for disassembly and inspection of the ML-1 pressure vessel to determine the location and nature of the leak in the lower tube sheet region is presented along with a report of the status of the work at the time of termination of the AGCRSP.

- 7. AGN TM-398, "Design Report for an Improved ML-1 Pressure Vessel," September 1962

Presents a review of the ML-1 pressure vessel tube bundle and possible improvements in corrosion resistance, stress concentrations and fabricability were evaluated. A near-optimum cooled tube sheet conceptual design is included.

8. Moderator Water Demineralization

- a. AN-ENGR-53, "The ML-1 Moderator Demineralizer," February 1962

Presents the design of the ML-1 moderator system demineralizer. One and one-half cubic feet of mixed-bed demineralizer resin are used to remove corrosion products and maintain resistivity at

≥ 1 megohm. A testing program to provide data for the selection of the resin is included.

- b. AN-AGCR-226, "Stability Tests of Mixed Bed Demineralizer Resins," 25 May 1962

Describes the problem of reduced demineralizer life when operating at 140 to 180°F with dissolved oxygen in the water. Laboratory tests to determine the effects of temperature and oxygen on the resin performance are described.

9. Deoxygenation

- a. AN-AGCR-66, "Deoxygenation of the ML-1 Moderator Water," 16 April 1962

Presents designs for deoxygenation of the ML-1 moderator by H₂ blanket, vacuum deoxygenation, or ion exchange resin systems. The methods are compared and P&I diagrams, schedules and costs are presented.

- b. AN-AGCR-253, "ML-1 Moderator Water Deoxygenation System Facility Space, Power and Water Requirements," 2 July 1962

Presents the space, electrical power, instrumentation and water requirements for the deoxygenation system.

- c. AN-AGCR-404, "ML-1 Deoxygenation System Design Report," December 1962

Presents the design of the ML-1 deoxygenation system for removal of oxygen from the circulating moderator water to maintain the dissolved oxygen concentration at a value of less than 0.5 cc/liter (0.7 ppm). A 20 gpm moderator bypass stream is processed through a deoxygenation column and appropriate heat exchangers, pumps, piping and valves.

- d. AN-AGCR-455, "Errata Sheet - ML-1 Deoxygenation System Design Report - AN-AGCR-404," 4 February 1963

Adds backwash pump data (section IV.B) and modifies Appendix A.

10. Tube Bundle Mockup Test Program

- a. AN-AGCR-327, "ML-1 Mockup Tube Bundle Testing," 18 August 1962

Presents a summary of the evolution of the 7-tube mockup testing program. The mockup was used for fabrication development in 1960, as a boiler to provide steam for testing the afterheat condenser, to assist in the failure analysis of the GCRE pressure vessel, as a test bed for development of the fuel element bypass flow modification and for long term testing at non-nuclear ML-1 conditions with and without deoxygenated moderator.

- b. AN-AGCR-330, "Metallurgical Inspection Seven-Tube Mockup Tube Bundle," 22 August 1962

Presents the results of the metallurgical inspection of the 7-tube bundle following testing at more severe conditions than those experienced by the ML-1 pressure vessel. No significant corrosion was observed that was judged to limit service to less than 50,000 hr.

- c. AN-AGCR-542, "1744 Hour Inspection, ML-1 Pressure Vessel Mockup," 24 June 1963

A pressure tube was removed from the 7-tube mockup for examination. Metallographic examinations of the removed tube and non-destructive examinations of the tube bundle indicated the bundle was in good condition. A replacement tube was installed and testing continued.

- d. AN-AGCR-650, "7-Tube Bundle Test - Pressure Vessel Tube #4, Metallographic Examination," 21 February 1964

Presents results of metallurgical examinations of two pressure tubes removed from the 7-tube pressure vessel after 1744 and 4256 hr of testing. The pressure tubes were found to be sound with no evidence of crevice or stress corrosion.

- e. AN-AGCR-551, "Seven-Tube Bundle Test - Pressure Vessel Tube No. 5 - Metallurgical Evaluation," 17 July 1963

Presents results of the metallurgical examinations of a pressure tube removed from the 7-tube bundle after 5500 hr of testing with oxygenated water including 56 hr without moderator flow during which time moderator boiling was experienced in the upper tube sheet coolant passages and below the upper tube sheet. No evidence of corrosion was observed.

- f. AN-AGCR-760, "Seven-Tube Mockup Test Program Summary Report," September 1964

Presents a summary of the 7-tube mockup test program. The 7-tube bundle was exposed to the non-nuclear ML-1 pressure vessel conditions for 5500 hr to evaluate corrosive effects of oxygen in moderator water.

11. Ultimate Shutdown Device

- a. AN-AGCR-93, "ML-1 Ultimate Shutdown Device, Design Report," 26 April 1962

Presents the design of the ML-1 ultimate shutdown device. The device provides emergency reactor shutdown independent of normal reactor controls. Nuclear poison (2.5 wt% boric acid) is injected from a pressurized container by an explosive operated valve into the core moderator region.

- b. AN-AGCR-245, "ML-1 Ultimate Shutdown Device, Design Review and Modification Proposal," 13 June 1962

Presents discussions of credible events leading to actuation of the ultimate shutdown device and the existing safeguards provided to prevent unplanned injection of the poison solution into the moderator.

- c. AN-AGCR-293, "ML-1 Ultimate Shutdown System," 9 August 1962

Presents the test plan for demonstration of performance of the ML-1 ultimate shutdown system and for instruction of operators.

- d. AN-AGCR-306, "ML-1 Ultimate Shutdown Device Checkout, OETP-452," 5 September 1962

Discussion of the checkout of the ML-1 ultimate shutdown device. The "boric acid poison" (simulated with demineralized water) was released by detonation of the explosive valve and the "poison" was injected into the moderator system in 9 minutes, 20 seconds with a tank pressure drop from 120 to 37 psig.

- e. AN-AGCR-486, "Ultimate Shutdown Device," 22 March 1963

Discusses use of an Army "10 Cap Blasting Machine" to activate the Conax valve on the ultimate shutdown device. Use of the Army machine is not recommended.

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4. Control Rod Design

The ML-1 reactor control rods are semaphore-type with neutron absorbing blades which operate in the moderator (water)-filled spaces between rows of calandria tubes. The neutron absorber is a silver-cadmium-indium alloy compounded specially to be most effective in the ML-1 under-moderated neutron spectrum. The blades are actuated by a drive system which is housed in a dry cavity at the outer surface of the shield tank and connected to the blades by a drive shaft which penetrates the shield solution area (Figure 16). Six control rods are provided; five are designated as shim-safety rods and one is designed as a regulating rod. The reactivity and mechanical characteristics of the rods are presented in Appendix C and the details of the design are available in the ML-1 design reports.⁽¹⁾

The performance of the ML-1 control rods during the 4½ year period of reactor and power plant operation was essentially trouble free. A minor modification was made to the dashpots to reduce the sensitivity of these assemblies to foreign matter in the moderator water⁽²⁾ and the electromagnetic clutches were modified slightly to improve the withdrawal characteristics.⁽³⁾ An evaluation performed after more than three years of operation indicated that, despite earlier laboratory results to the contrary, some corrosion of the silver-cadmium-indium alloy was occurring.⁽⁴⁾ This corrosion did not significantly alter the neutronic characteristics of the ML-1 reactor but operation and maintenance were severely affected by the contamination of the moderator water and reactor components with radioactive silver. It was recommended that the ML-1A control blades be canned in stainless steel to prevent this undesirable condition.

FIGURE 16. ML-1 CONTROL ROD

CONTROL ROD DESIGN BIBLIOGRAPHY

1. Control Rod Design

- a. AN-AGCR-385, "ML-1 Design Data Report," 3 transmittals - 12/14/62, 4/19/63 and 2/21/64
- b. AGN TM-378, "ML-1 Components Design Report," July 1960

Present the details of the design of the ML-1 power plant.

2. AN-ENGR-54, "Design Report: Modification to ML-1 Control Rod Dashpot," undated

Describes the modification made to the ML-1 control rod dashpots (substitution of a harder material in the vanes and an increase in certain internal clearances) to reduce the sensitivity of this device to foreign material in the moderator water.

3. AN-AGCR-775, "ML-1 Control Rod Clutch Modification Program," November 1964

Describes the malfunction (increase in withdrawal time) observed and the investigations, modifications and proof testing completed to eliminate the problem.

4. AN-AGCR-874, "Silver-Indium-Cadmium Control Rod Corrosion," 26 April 1965

Reports the results of an experiment conducted to determine the amount of neutron absorber lost from the ML-1 control blades. Test data indicates that 0.7% of the original weight of the material has corroded away and concludes that this loss has no appreciable effect on reactivity of the system.

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5. Reactor Shielding

The design of the operational and shutdown shielding for the ML-1 reactor evolved in two distinct stages. The first design was based upon the performance criteria set forth for the Portable Gas-Cooled Reactor (PGCR). Preliminary PGCR shielding design began in 1958 with the evaluation of several concepts and materials. A conceptual shutdown shield was developed which incorporated a cooled, lead reflector/primary radial shield surrounded by a secondary lead shield. Tungsten plates were provided in the inlet and outlet coolant plenums to attenuate axial streaming and the entire reactor/shield assembly was surrounded by a layer of canned ammonium pentaborate salt. Operational shielding was provided by immersing the reactor/shutdown shield assembly in a tank of borated water. This shield concept resulted in a reactor package which weighed 18 tons.

The second stage of shield design⁽¹⁾ began with the receipt of the ML-1 Military Characteristics in June 1959. A shield design different from that developed for the PGCR was required to meet these specifications because of the 15 ton weight specified for the reactor package. The pertinent shield performance criteria for the ML-1 and PGCR plants are compared in Table 10. The final ML-1 shield design was the product of two preliminary configurations⁽¹⁾; modifications to these configurations resulted from the materials evaluation and experiment performed at the Lid Tank Shielding Facility (LTSF) at ORNL.⁽²⁾

The final design⁽¹⁾ of the ML-1 shield consisted basically of 2-1/2 in.-thick lead inner and 1-1/4 in.-thick outer lead radial shields placed outside of the cylindrical stainless steel core support structural member (see Figure 13).

TABLE 10 - COMPARISON OF PGCR AND ML-1 SHIELD PERFORMANCE CRITERIA

	<u>PGCR</u>	<u>ML-1</u>
<u>Plant Characteristics</u>		
Maximum reactor package weight (for transport)	18 tons	15 tons
Time interval between major overhauls	10,000 hr	10,000 hr
Operating lifetime	50,000 hr	50,000 hr
<u>Radiation Criteria</u>		
<u>Shutdown</u> ⁽¹⁾		
Front (toward p/c shield)	5 mrem/hr	15 mrem/hr
Top	500 mrem/hr	1500 mrem/hr
Sides and bottom	50 mrem/hr	150 mrem/hr
<u>Operating</u> ⁽²⁾		
At control cab	5 mrem/hr	5 mrem/hr ⁽³⁾

(1) 24 hours after shutdown, following 10,000 hr full power operation, 25 ft from reactor, with reactor package in transport condition.

(2) At control cab, 500 ft from reactor, during full power operation.

(3) With expedient shielding as required.

A 6 in.-thick cylindrical reflector assembly was placed between the outer ring of pressure tubes and the stainless steel support structure; the reflector was lead in the non-cab direction and lead/tungsten in the cab direction. A 2 in.-thick polystyrene shield was located outside the outer radial shield in the direction of the cab. Axial radiation was attenuated by 3 in.-thick tungsten baffles clad in stainless steel placed in both the upper and lower plenums. Additional axial shielding was provided at both the top and bottom by a conical-shaped extension of the outer radial shield. The reactor core/shield package was positioned inside the shield solution tank which was filled with borated water. The shield system was designed to satisfy the shutdown dose criteria set forth in Table 10 with a minimum weight. Analysis indicated that the operational dose criteria could be satisfied at the control cab by use of expedient shielding.

The performance of the final ML-1 shield design was evaluated during a series of experiments performed with the ML-1 reactor⁽³⁾; a summary of the test objectives, test conditions and primary results of the major shield performance evaluations is presented in Table 11. The underlying purpose of all these tests was to demonstrate the capability of the shield to meet the design criteria during and after a 10,000 hr, full power run. The lower power tests were designed to obtain attenuation data for each shield component to permit a prediction of full power performance. The higher power (up to full power) tests were designed to determine overall operational and shutdown dose rates, the effect of air scattering, the effect of various concentrations of boric acid in the shield solution, and to establish the requirement for operational expedient shielding, where necessary. Extrapolation of data from the higher power tests to design conditions indicated that the shield system generally satisfied the performance criteria set forth in the Military Characteristics.

As the development of plans and requirements for ML-1-type systems proceeded, it became apparent that the lower dose rates during reactor operation were highly desirable. This concept was reflected in the Qualitative Materiel Requirements (QMR) issued for ML-1 type plants in the summer of 1963 which specified an operational dose rate of 5 mrem/hr at 500 ft without the use of expedient shielding (Appendix E). Although the schedules did not permit the implementation of programs to modify the ML-1 shielding to satisfy this specification, many of the tests conducted in 1964 and 1965 had as an objective the development of data which would provide a basis for the design of such modifications and the more stringent criteria was considered in the ML-1A preliminary design (see Section VII).

TABLE 11 - SHIELDING PERFORMANCE EVALUATION - ML-1 REACTOR

Experiment (ANSOP)	Shielding Experiment Objective	Power (watts)	Time (hours)	Boric Acid Conc. (wt%)	(1) Radiation Levels (neutron & gamma), Extrapolated to 3.3Mw Operation, mrem/hr					
					Operational Radiation			Shutdown Radiation (Extrapolated to 10,000 hr, 24 hr decay)		
					Cab;R=500ft	θ=0;R=500ft	θ=180;R=500ft	θ=0;R=25ft	θ=180;R=25ft	R=0;Z=25ft
Design Criteria					<5.0	-	-	<15.0	<150	<1500
16240	Fast neutron attenuation due to dense shielding, integrity of dense shielding, axial core leakage, and the worth of axial shield components	≤50	<10	2	-	-	-	-	-	-
16241A	Fast neutron attenuation due to dense shielding, gamma and neutron penetration of the upper tube sheet, gamma distribution in core and tube sheets, preliminary evaluation of air scattering.	≤250	<15	10	-	-	-	-	-	-
16241B	Determine gamma and neutron dose rates, both external to and within the reactor shield system, in the axial and radial directions using the reference dense shielding components.	≤6 x 10 ³	<15	2,10	-	-	-	-	-	-
16615	Determine gamma and neutron dose rates, both external to and within the reactor shield system, in the axial and radial directions using the reference dense shielding components during operation and to obtain shutdown dose rates at various locations. (2)	≤1.6 x 10 ⁶	<25	10	2.75 ⁽³⁾	7.3 ⁽³⁾	31.0 ⁽³⁾	9.2	125	332
16625	Confirm and extend the dose rate results from ANSOP 16615 with particular emphasis placed on the effect of the wood shield on the horizontal dose rate due to air scattering. (2)	<3.0 x 10 ⁶	<110	10	<2.9 ⁽³⁾	8.2 ⁽³⁾	31.4 ⁽³⁾	18.8	198	475
16635		(2.2 x 10 ⁶ ave)			7.0 ⁽⁴⁾	19.6 ⁽⁴⁾	42.8 ⁽⁴⁾			
15625A	Determine dose rates at various radial positions using a wood shield (2) and without using a wood shield.	<3.0 x 10 ⁶ (2.2 x 10 ⁶ ave)	<700	10	-	7.5 ⁽³⁾ 11.0 ⁽⁵⁾	30.0 ⁽⁵⁾	3.7 ⁽⁶⁾ 10 ⁽⁷⁾	37 ⁽⁶⁾	There is some doubt as to the accuracy of these measurements
15650	Determine dose rates with and without the wood shield (2) and with a borated water shield in place of the wood shield.	≤3.0 x 10 ⁶ (2.2 x 10 ⁶ ave)	<2000	2	-	8.4 ⁽³⁾ 8.9 ⁽⁵⁾ 7.9 ⁽⁸⁾ 7.2 ⁽⁹⁾	30.0 ⁽⁵⁾	11.6 ⁽¹⁰⁾	51 ⁽¹⁰⁾	

(1) Extrapolated values are for the noted boric acid concentration in the shield solution; shutdown doses are for drained shield tank; cab is behind an earth berm.

θ = angular position relative to a line drawn through reactor and control cab.

R = radial position relative to the center of the core.

Z = vertical position relative to the center of the core.

(2) A wood expedient shield was positioned directly above the reactor to reduce the air scattered contribution to the horizontal dose rate.

(3) Extrapolated values with wood shield in place (see Note 2).

(4) Calculated values with wood shield removed based on experimental results with wood shield in place.

(5) Extrapolated values based on experimental data obtained without wood shield and at the reference shield water height.

(6) Dose rate as measured during ANSOP 16625A, 24 hours after shutdown; shield tank drained.

(7) Crudely estimated for 10,000 hr @ 3.3 Mw based upon degree of saturation observed during actual measurement

(8) Extrapolated values based on experimental data obtained with shield water height increased by 3 in.

(9) Extrapolated values based on experimental data obtained with shield water height increased by 12 in.

(10) Dose rate as measured during ANSOP 16650, 26 hours after shutdown following 1000 hr operation at 2.2 Mw; shield tank drained.

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REACTOR SHIELDING BIBLIOGRAPHY

1. ML-1 Shielding Design

- a. IDO-28609*, "ML-1 Shielding Design Report," October 1963

Describes the evolution of the shielding design from that of the PGCR through the final ML-1 configuration. Presents a detailed summary of the mechanical, metallurgical, and neutronic analyses and evaluations performed in support of the shielding design effort. Presents the results of shielding experiments performed at ORNL and the shielding performance evaluation conducted with the ML-1 reactor during the conduct of ANSOP 16625 and 16635.

- b. AN-ENGR-15, "The Development and Design of a Biological Shield for the ML-1, A Mobile Gas-Cooled Nuclear Power Plant," undated

Presents a concise summary of the ML-1 shielding development and design activities. Discusses the results of initial lid tank (ORNL) and foil activation experiments.

2. Shield Experiments

- a. AGN TM-360, "Recommendations for ML-1 Shielding Experiments," August 1959

Discusses the proposed initial ML-1 shielding experiments and surveys possible facilities where the experiments could be performed. Recommends that the experiments be performed at AGN, San Ramon, based on schedule, cost and flexibility.

- b. AN-ML-1-44, "Lid Tank Shield Facility Experiment No. 77 for the ML-1," undated

Presents the results of the lid tank experiments performed at ORNL to determine shield component activation and efficiency. The results of these experiments led to major shield design alterations including the replacement of the canned borated salt blanket with borated water, lead, and polystyrene; replacement of stainless steel canning materials with aluminum wherever possible; and the addition of more lead shielding for the stainless steel plenums and gas ducts.

3. Shielding Performance

- a. AN-AGCR-599, "Effect of Shield Water Boron Content on ML-1 Dose Rates," 10 September 1963

Presents the results of activation data from lid tank shield experiments and activation and operational survey data obtained at ML-1 during ANSOP 16241B as a function of ML-1 shield solution

boron concentration. Results indicate that the gamma dose rate with a 2 wt% boric acid shield solution will be about twice that of a 10 wt% boric acid solution.

- b. AN-AGCR-669, "The Effect on Radiation Dose Rates of Changing Shield Water Boric Acid Concentration from 10 wt% to 2 wt%," 18 February 1964

Presents the analysis, based on experimental results, to determine the effect of using a 2 wt% boric acid shield solution instead of the reference 10 wt% boric acid solution. Results indicate that the operational dose rates will increase by about 30% but can be kept below the maximum allowable at the control cab by expedient shielding. Shutdown gamma dose rates are shown to increase by about a factor of two.

- c. AN-AGCR-936, "Test Report - ML-1 5% Boric Acid Shield Solution Experiment," 5 October 1965

Reports the results of an experiment conducted at the ML-1 to evaluate the radiation dose rates with a 5% boric acid shield solution. The evaluation of the test data indicates that gamma dose rates during reactor operation were about 20% greater than those measured with a 10% boric acid shield solution but that no obvious change in neutron dose rates could be observed. The gamma dose rate following shutdown after operation with 5% boric acid shield solution was essentially the same as that observed after operation with 2% boric acid shield solution.

- d. AN-AGCR-752, "ML-1 Reactor Aluminum Corrosion Study," September 1964

Presents the results of in-reactor inspection and out-of-pile laboratory tests performed to determine the extent and mechanism of aluminum corrosion of reactor components exposed to borated shield solution and establishes controls to minimize such corrosion.

- e. IDO-28634*, "Final Report, Full Power and Limited Endurance Test of the ML-1 Nuclear Power Plant (ANSOP 16625A)," September 1964

Summarizes the operation of the ML-1 plant during ANSOP 16625A. Of particular interest is the evaluation of the performance of the operational and shutdown shielding.

- f. AGN TM-410, "Final Test Report, ML-1 Thermodynamic Performance Test (ANSOP 16650)," May 1965

Summarizes the operation of the ML-1 plant during ANSOP 16650. Includes a section on the performance of the operational and shutdown shielding.

6. Heat Exchangers

The compact, high-performance heat exchangers (precooler and recuperator) for the ML-1 power plant were procured under an R&D subcontract with Griscom-Russell Company.

The concept selected by Griscom-Russell for the precooler (based on specifications developed by Aerojet⁽¹⁾) was a finned-tube unit constructed exclusively of aluminum. The external tube fins were brazed to the tube wall and internal finning was provided by inserting an extruded section into the tubes. The tubes were seal welded into the tube sheets which formed a portion of the distribution (inlet) and collection (outlet) headers. The design effectiveness of the unit was 92% at a nitrogen flow rate of 26.3 lb/sec.⁽²⁾

An evaluation of the performance of the precooler based on ML-1 test data revealed the following deficiencies:

- The aluminum tubes sagged and bowed⁽³⁾⁽⁴⁾, changing the tube spacing and generally degrading the performance by perturbation of the cooling air flow and distribution.
- The variation in cooling air flow and internal flow resulted in non-uniform temperatures from tube-to-tube; the resulting non-uniform stresses caused leaks in the tube-to-tube sheet joints.⁽³⁾⁽⁴⁾
- The effectiveness of the unit was approximately 88%; this is 4% less than the design value.⁽⁵⁾

The design of an improved precooler, incorporating modifications to correct the mechanical and performance deficiencies of the Griscom-Russell design, was completed by Aerojet.⁽⁶⁾⁽⁷⁾ This unit was of the finned-tube concept with aluminum tubes rolled into a stainless steel tube sheet. The tubes were bowed upward to minimize the effect of non-uniform temperature distribution. This design was evaluated during the ML-1A preliminary design work (Section VII-B) and it was found that the size of the assembly was not compatible with the ML-1A p/c skid envelope. As a consequence, discussions were initiated with heat exchanger manufacturers which resulted in the procurement of a new fin-plate type precooler (see V-G below).

The ML-1 recuperator was a shell-and-finned tube type consisting of four externally finned high-pressure tube passes and one low-pressure shell pass fabricated from AISI 300 series stainless steel. The unit was considered to be a four-pass, cross-counterflow heat exchanger designed to have an effectiveness of 79% at a flow rate of 26.3 lb/sec.⁽⁸⁾

The evaluation of the performance of the recuperator was very difficult because of the compact configuration of equipment on the p/c skid. Special instrumentation was installed and sufficient data were collected to indicate that the actual effectiveness of the unit was about 70%.⁽⁹⁾ Analysis indicated that this condition was consistent with the assumption that the exchanger operated as a combination cross-counterflow and parallel-counterflow unit because of gas leakage where the tubes penetrate the internal flow-directing baffles in the low-pressure (shell side) pass.⁽¹⁰⁾

recuperator⁽¹¹⁾ A performance specification was prepared for an improved and transmitted to eleven heat exchanger manufacturers with requests for proposals to design and fabricate such an assembly. Three acceptable responses were received and evaluated. The proposal from Stewart-Warner, specifying a plate-fin, three-pass, cross-counterflow unit, was judged to be superior.⁽¹⁰⁾ A fixed price quotation for the work was solicited and received from Stewart-Warner but the contract had not been awarded as of the time of termination of AGCRSP activity.

HEAT EXCHANGER BIBLIOGRAPHY

1. AGN TM-359, "Specifications for the Combined Precooler and Moderator Cooler for ML-1," 28 September 1959

Presents the performance specifications and physical dimensions for the precooler assembly to serve as the basis for design and fabrication of the unit.

2. "Final Report, Precooler/Moderator/Oil Cooler, ML-1 Project," Griscom-Russell Co., undated

This final design report published by Griscom-Russell describes the design of the precooler as well as the engineering and development activities which supported this design.

3. AN-AGCR-319, "Precooler Tube History," 1 November 1962

Tabulates two sets of data taken approximately three months apart to illustrate the extent of precooler tube sag.

4. AN-AGCR-422, "ML-1 Precooler and Recuperator Inspection," 14 May 1963

Presents the results of an inspection of the precooler and recuperator. The physical configuration of the precooler was essentially the same as that determined a year prior to these measurements but indications of significant leakage at the precooler tube-to-tube sheet joint was observed. No evidence of internal leakage in the recuperator was detected.

5. IDO-28634*, "Final Report Full Power and Limited Endurance Test of the ML-1 Nuclear Power Plant (ANSOP 16625A)," September 1965

The effectiveness of the precooler is determined to be approximately 88% based on data developed during the conduct of ANSOP 16625A.

6. AN-AGCR-533, "Improved ML-1 Precooler Interim Design Report," June 1963

The interim status of the design of an improved finned tube precooler is presented.

7. AN-AGCR-643, "Final Design Report Improved ML-1 Precooler," February 1964

The final design of an improved ML-1 precooler of the finned type concept is presented. In this design the tubes are bowed to minimize the effect of non-uniform temperature distribution. The aluminum tubes are sleeved and rolled into the stainless steel tube sheets; the inlet and outlet headers are also fabricated of stainless steel.

8. "Final Report - Recuperator - ML-1 Project," Griscom-Russell Co.
undated

Describes the final design of the ML-1 recuperator and summarizes the engineering and development activities which supported this design.

9. AN-AGCR-869, "ML-1 (Griscom-Russell) Recuperator Performance,"
8 April 1965

Describes the analysis of test data which resulted in the determination of a 70% effectiveness for the Griscom-Russell recuperator.

10. AN-AGCR-835, "Advanced Recuperator Study," March 1965

Presents an analysis of the reasons for the deficient performance of the Griscom-Russell recuperator as well as the specifications developed for an improved recuperator and the responses of qualified heat exchanger manufacturers to request for proposals for the design and fabrication of such a unit.

11. AGC-60263, Aerojet-General Nucleonics Specification, "Heat Exchanger, Gas-to-Gas, Recuperator," 9 September 1964

This document presents the performance specifications and physical dimensions for the improved recuperator.

7. Turbomachinery Development

The ML-1 turbomachinery was designed specifically for use in the closed, Brayton cycle power extraction system and consisted basically of a gas turbine, a gas compressor and a starting motor. The turbine and compressor were mounted on a common shaft. The turbine extracted energy from the 1200°F, 300 psia reactor coolant/power conversion cycle working fluid (99.5% N₂-0.5% O₂). The compressor accepted precooled gas at 132°F, 117 psia and discharged 366°F, 320 psia gas to the recuperator for preheating prior to entry into the reactor (see Figure 19 in Section V-C-12). A speed reduction unit was mounted on the compressor end of the turbine-compressor (t-c) set to reduce the alternator drive shaft speed to 3600 rpm.

The design of the t-c set was initiated late in 1959.⁽¹⁾ Two subcontractors, the Stratos Division of Fairchild Engine and Aircraft Co.* and Clark Bros. Co., were selected under a contract between Aerojet and the Department of the Army, APCDB, to conduct preliminary design studies, prepare the final design, fabricate, and test one t-c set each. During this early design period, as well as throughout the entire design effort, general techniques were developed to analytically determine the stability, lifetime and overall performance characteristics of ML-1 type t-c sets.⁽²⁾ The design and performance characteristics of the two t-c sets are compared in Appendix C. In general, the Stratos machine incorporated an overhung axial turbine and a two stage axial compressor. In the Clark unit, both the axial turbine and eleven stage centrifugal compressor were supported between the main journal bearings.

The results of the Stratos t-c set preliminary design study⁽³⁾ were reviewed by Aerojet early in 1960 and recommendations for design changes, particularly in the area of sealing arrangement and the aerodynamic design (based on the estimate of the t-c set performance) were made. Preliminary open-cycle testing of the TCS-670 and subsequent data analysis indicated that the unit did not need design specifications.⁽⁴⁾ In particular, it was concluded that the turbine flow rate was greater than the design value, which resulted in an inefficiently matched turbine-compressor system. The unit was modified (TCS-670A) and during subsequent open-cycle testing the turbine inter-stage seal failed due to thermal expansion which caused the seal to rub against the first stage rotor.⁽⁵⁾ Additional modifications to the set were completed in 1963 which proved successful during open-cycle testing⁽⁶⁾ but subsequent closed-cycle tests with this unit again resulted in seal failure.⁽⁷⁾ Teardown inspection revealed that the pressure loading associated with closed-cycle operation had resulted in the failure of certain internal support structures which permitted axial motion of seal components and a pressure area diaphragm resulting in interference between moving and stationary parts. Further testing and modification of this unit was deferred and cognizance for subsequent developmental efforts for the TCS-670 was assigned to APCDB in 1964.

The preliminary and final design of the Clark Bros. Co. t-c set (CSN-1) was reviewed by Aerojet during 1960 and resulted in the conclusion that the proposed system was mechanically sound but that the performance specification could probably not be satisfied without an aerodynamic modification.⁽⁸⁾ The proposed design for the lube oil seals was tested and found to be acceptable⁽⁹⁾ and, after the successful completion of a preliminary open-cycle test, *Now Stratos Division of Fairchild-Hiller Corp.

the CSN-1 was installed in the ML-1 power conversion skid and delivered to the NRTS for power plant demonstration in June 1962.

The CSN-1 t-c set operated successfully during ML-1 power plant operations in September 1962 and February 1963 (ANSOP 16615 and 16625, respectively; see Section V-F). However, an inspection after the February operation indicated that abnormal bearing wear and some turbine blade cracking had occurred. Aerodynamic performance during both runs appeared to be poorer than design predictions. A new bearing design was developed, jointly by Aerojet and Clark, and the turbine blade design was modified to improve blade strength. The modified unit, CSN-1A, was successfully open-cycle tested⁽¹⁰⁾, installed in the ML-1 power conversion skid and tested under power plant conditions during the conduct of ANSOP 16625A (April/May 1964). The aerodynamic performance of the unit was poorer than design predictions.⁽¹¹⁾ ANSOP 16625A was terminated when emergency power to the plant was lost; this "hot-stop" shutdown resulted in turbine bearing temperatures approaching 400°F. Subsequent t-c set disassembly and inspection indicated that the bearings had not been damaged by this exposure but that extensive damage had occurred (probably during operation) to the first stage turbine blades.⁽¹²⁾ Inspection of the set revealed that a spot-welded internal part had broken free and, moving axially under the force exerted by the flowing gas, had worn a groove in the leading edge of the blades of the turbine first stage rotor. Corrective modifications and repairs were made and the CSN-1A unit was again installed in the ML-1 system in mid-1964. Mechanical operation of the t-c set during late 1964 and early 1965 (ANSOP 16650 and 16656) was completely satisfactory although the aerodynamic performance continued to be poorer than predicted.⁽¹³⁾ The results of CSN-1A disassembly and inspection subsequent to the early 1965 test indicated that the mechanical condition of the unit was satisfactory.⁽¹⁴⁾

A summary chronology of the experience with the two turbine-compressor sets is presented in Table 12.

During the course of CSN-1A performance evaluation and testing, it became evident that two rather basic problems prevented the Clark turbine-compressor set from completely satisfying the design specification. These were the inability of the set to develop the power output specified and the problems associated with the mechanical integrity of bearings, seals and turbine blades. The mechanical problems associated with bearings and seals appeared to be solved, based on the excellent results of the power plant operation in late 1964 and early 1965 and the results of the analytical and experimental evaluations of CSN-1A type bearings and seals.⁽¹⁵⁾ These evaluations covered a wide range of operating conditions, including those associated with operation of the ML-1 power plant using air as the working fluid, and indicated good stability and potential lifetime for the bearing/rotor system.

The inability of the CSN-1A aerodynamic design to develop design power was evident from power plant test data as early as November 1962.⁽¹⁶⁾ Subsequent minor aerodynamic modifications to the unit during the completion of mechanical repairs proved to be ineffective. As a result, the decision was made that further evaluation of the machine in the compact configuration required for power plant testing was not the optimum method for completion of the development. The best available data on t-c set performance were submitted to several qualified manufacturers with a request for proposals

TABLE 12 - TURBINE-COMPRESSOR SET DEVELOPMENT CHRONOLOGY

<u>EVENT</u>	<u>DATE</u>	<u>COMMENT</u>
<u>1. TCS-670 (Stratos)</u>		
Preliminary Design (based on TCS-560)	February 1960	
Shipped to AGC/Azusa	February 1961	Installed in power conversion skid for testing.
Initial Closed-Cycle Test	June 1961	Unit failed to operate in closed-cycle configuration. Removed from skid and installed in open-cycle test facility.
Open-Cycle Test	Nov 1961 - Mar 1962	Successfully open-cycle tested. Compressor mapped. Data for modification to TCS-670A obtained.
Modification	April-Dec 1962	Aerodynamic modification by Stratos. Redesignated TCS-670A.
Open-Cycle Test	January 1963	Interstage seal failed.
Investigation and Modification	Jan-June 1963	Clearances modified by Stratos to allow for thermal expansion.
Open-Cycle Test	June-July 1963	Successfully open-cycle tested.
Closed-Cycle Test	August 1963	Interstage seal failed at full speed and design conditions.
Transferred to cognizance of U.S. Army	November 1963	
<u>2. CSN-1 (Clark)</u>		
Shipped to AGC/Azusa	November 1961	
Installation on P/C Skid	Nov-Dec 1961	Preparation for initial closed-cycle testing.
Closed-Cycle Test	Dec 61 - Feb 62	Successfully tested in closed-cycle configuration, including 50 hour endurance run.
Teardown and Inspection	February 1962	Insulation blanket on recuperator failed during closed-cycle run. No discrepancies noted in t-c set.
Closed-Cycle Test	March 1962	T-C set performed satisfactorily. Seal gas reconfiguration system froze and oil entered primary loop.
Teardown and Inspection	April 1962	Unit cleaned, inspected, and reassembled. No discrepancies noted.
Closed-Cycle Test	May-June 1962	Successfully closed-cycle tested, including 51 hour endurance run.
Shipment to NRTS	June 1962	
ANSOP 16615	September 1962	Initial closed-cycle operation with ML-1 reactor as heat source.
ANSOP 16625	Feb-Mar 1963	Successful closed-cycle operation, including 100 hour endurance run.
Teardown and Inspection	March 1963	Revealed excessive bearing wear.
Modification	Apr 1963-Feb 1964	Bearing development program (AGN and Clark Bros.) and aerodynamic modification by Clark. Redesignated CSN-1A.
ANSOP 16625A	Apr-May 1964	Successful closed-cycle test, including 664 hours self-sustained operation. Power failure resulted in hot stop.
Teardown and Inspection	June 1964	Revealed first stage turbine blades damaged due to filler ring movement.
Rotor Repair	June-July 1964	New turbine blades installed, rotor balanced. Unit reassembled with new bearings. Static seals removed.
ANSOP 16650	Sept 64-Jan 65	Successful closed-cycle test, including 2560 hours of rotation. Power output deficient.
Teardown and Inspection	February 1965	Revealed unit to be in excellent condition.
Shipment to G.E.	April 1965	To be modified to improve aerodynamic performance.

for modification of the unit to permit the attainment of the design power output. The contract for this modification was awarded to General Electric Co. in April 1965, and at the termination of the prime contract, the modification of the CSN-1A was about 50% complete.⁽¹⁷⁾ The contemplated program included testing in a specially designed test facility to characterize the performance of the modified machine prior to reassembly of the ML-1 p/c skid.

The starting motor designed for the CSN-1 t-c set was tested but never actually used since it did not have adequate power.⁽¹⁸⁾ During power plant testing, the t-c set was started with a 150 hp General Electric 3-speed motor. The design of an improved starting motor, compatible with the CSN-1A startup requirements, was completed but the procurement and testing of this unit had not been initiated at the time of AGCRSP termination.⁽¹⁹⁾

TURBOMACHINERY DEVELOPMENT BIBLIOGRAPHY

1. AGN TM-353, "Turbine-Compressor Set Specifications," September 1959

This document was supplied by AGN as the technical specification for the subcontracts for a design study of a closed-cycle gas turbine set. Includes performance and environmental requirements for operation, shutdown and transportation in addition to design, development, fabrication and testing activity requirements.

2. T-C Set Performance Prediction and Analysis

- a. AN-ENGR-25, "Determination of Critical Speed of Arbitrary Shafts Supported by Two Bearings," November 1959

Presents a simplified method for the calculation of the critical speed of rotors of turbomachines which are supported by two bearings.

- b. AN-ENGR-27, "Stresses and Deflections of Cylindrical Shells as Used for Design Elements in Turbomachines," February 1960

Presents a method for calculating stresses and deflections of cylindrical shells with constant wall thickness, having finite axial length, which are subjected to external forces and moments at their extremities. The analysis is based on a mathematical model of the Stratos TCS-670 turbine-compressor set.

- c. AN-ENGR-35, "A Graphical Solution of the Matching Problem of Closed-Cycle Gas Turbine Plants," September 1961

Describes a simple graphical method for the determination of operating points of the turbomachines in a closed-cycle gas turbine power plant. Results of the method as used for ML-1 or GTTF plants indicate that any leakage from the high pressure to the low pressure side of the loop has the same effect as reducing the turbine inlet pressure.

- d. IDO-28591*, "Effects of Gas Corrosion and Aging on ML-1 Turbine Alloys," August 1962

Gas corrosion resistance and changes in mechanical properties of alloys to be used in the ML-1 gas turbine are evaluated at the anticipated maximum operating conditions for up to 5000 hr in 99.5% N_2 -0.5% O_2 , air and 100% N_2 . Alloys evaluated were Inconel, Inconel 713c, Incoloy 901, AISI Types 347, 304 and 422 stainless steels and Multimet Alloy N-155.

- e. AN-AGCR-829, "Turbine Blading Loss Coefficient Computer Program, LOSCO," January 4, 1965

Presents the analytical background and operation of a 7094 Fortran computer code for the calculation of loss coefficients of the blading in a two-stage axial flow turbine.

- f. AN-AGCR-830, "Turbine Performance Evaluation Computer Program, TURBO-A," 26 January 1965

Describes the analytical background and operation of a 7094 Fortran computer program to evaluate the performance of a two-stage axial-flow turbine with unchoked flow. The program can be easily modified to analyze axial turbines having more than two stages.

- g. AN-AGCR-882, "T-C Set Stage Characteristics Computer Program (CATSEP)," 6 May 1965.

Describes the use of an IBM-7094 Fortran computer code for the prediction of interstage and overall performance based on turbine or compressor test data.

3. TCS-670 Design Review

- a. AN-ENGR-28, "Method for the Comparison of the Performance of Centrifugal Compressors," February 1960

Presents a method for comparing the performance of centrifugal compressors of different designs at various operating conditions. Results of the analytical and experimental efforts indicate that the predicted performance of the Stratos TCS-670 compressor is in error. Concludes that either the compressor map supplied by Stratos was constructed inaccurately or that the impellers of the compressor were completely mismatched in the preliminary design. Test results are described in AN-ENGR-34.

- b. AN-ENGR-29, "Comments to the Design of the Stratos TCS-670 Turbine-Compressor Set of Preliminary Design Report SRC-103," February 1960

Presents a design review of the Stratos TCS-670 t-c set. Recommends basic design changes to both the turbine and compressor. Examines in detail the sealing arrangement and estimates net power output.

- c. AN-ENGR-32, "Tests of Seal #I," February 1961

Presents the results of tests performed to determine the effectiveness of the labyrinth type oil seals arranged on the compressor side of the Stratos TCS-560 and TCS-670 turbine-compressor sets. Evaluates the effect of lubricating oil pressure drop and flow rates on oil leakage into the compressor for various rotor speeds. Leakage was observed under some conditions.

4. TCS-670 Open Cycle Test Analysis

- a. AN-ENGR-34, "Performance of TCS-670 Compressor from Tests with Air at the Stratos Division," September 1961

Analyzes the results of tests performed on the Stratos TCS-670 compressor. Results indicate that the compressor is sound and that there does not exist a stage mismatch as concluded from the analysis of the preliminary design. See AN-ENGR-28 for results of preliminary analysis.

- b. AN-ENGR-38, "Estimated Performance of TCS-670," September 1961

Presents the results of analysis performed to determine the matching characteristics of the Stratos turbine-compressor set. Concludes that the turbine flow rate is greater than estimated and results in a plant output of about 80 kw less than design due to decreased compressor efficiency at the higher flow rates.

5. TCS-670A Modifications

- a. AN-AGCR-484, "Proposal for Repair of TCS-670A," March 1963

Presents a program for the repair of the TCS-670A turbine-compressor set damaged during open-cycle testing, including the necessary redesign of the unit to correct the deficiency at the interstage seal and procurement of replacement parts or repair of existing parts.

- b. AN-AGCR-498, "TCS-670A Failure Analysis and Recommendations," 5 April 1963

Presents the results of the failure analysis of the TCS-670A which failed during open-cycle testing. Results indicate the primary mode of failure was interference between rotating and stationary turbine components resulting in failure of the turbine interstage seal causing shaft seizure. Recommendations to eliminate a recurrence are included.

6. AN-AGCR-558, "Open Cycle Test Report," July 1963

Presents the results of TCS-670A, a modified version of the TCS-670 turbine-compressor set, open-cycle testing. The set was operated for more than 20 hours including more than 7 hours of hot rotation. Performance of the machine was in good agreement with turbine redesign predictions. High breakaway torques were experienced early during the testing but diminished with increasing operation, indicating that complete seal wear-in had been accomplished. The lubrication system functioned satisfactorily throughout the test program; all oil flows and bearing temperatures remained well within the specified operating limits. Turbine inlet temperature was maintained at 1200°F at rated speed; power output for the open-cycle tests (taken at the reduction gear output shaft)

was approximately 20-25 kw. The compressor efficiency at the design point for open-cycle operation was 80%. Closed-cycle performance, predicted from open-cycle test data, indicated that the t-c set would self-run at 71% of rated speed (80°F ambient temperature, 90 psia system pressure, and 1000°F turbine inlet). Compressor pressure and turbine expansion ratios at the design inlet conditions were predicted to be 2.84 and 2.47, respectively. Extrapolation of open-cycle test data indicated that closed-cycle operation should deliver 300 kw at 100°F ambient temperature.

7. AN-AGCR-624, "Stratos TCS-670A: Turbine-Compressor Set Closed-Cycle Test Report," January 10, 1964

Presents the results of closed-cycle testing of the ML-1 power conversion skid using the Stratos TCS-670A turbine-compressor set. Closed-cycle testing was prematurely terminated when the turbine inner stage seal rubbed on the turbine wheel causing failure of the inner stage seal. At the time of failure, almost 21 hours of operation (about 12½ hours above 50% rated speed) had been achieved. The power output at the t-c set gearbox was calculated as 152kw, based on dynamometer data for 72°F ambient and 1128°F turbine inlet temperatures. Measured parasitic power losses were 36 kw higher than those measured during open-cycle testing. The limited duration of testing did not allow data to be collected for all operating conditions. Extrapolation of existing data to 100°F ambient and 1200°F turbine inlet temperatures indicated that 250 kw (shaft power) could have been realized. At these conditions the turbine and compressor efficiencies would have been 79% and 82.5%, respectively.

8. CSN-1 Design Review

- a. AN-ENGR-30, "Review of Mechanical Design and Stress Conditions of ML-1 Turbine-Compressor Set," May 1960

Reviews the ML-1 turbine-compressor set design resulting from a design study performed by Clark Bros. Approves of general design philosophy and concept. Recommends certain design changes to increase efficiency and reduce weight.

- b. AN-ENGR-31, "Review of Aerodynamic Design of Clark Compressor," October 1960

Presents an analytical review of the aerodynamic design of the ML-1 turbine-compressor set as presented in the Clark Bros. final design report. Concludes that the design point performance predicted by Clark Bros. could not be met. Recommends design changes to increase efficiency.

9. AN-ENGR-33, "Tests with Seal #II," February 1961

Presents the results of tests performed to determine the effectiveness of the labyrinth-type oil seals of the Clark Bros. t-c set for ML-1. The seals were found to be fully capable of retaining the oil at specified and off-design oil pressure differences.

10. AN-AGCR-711, "Final Summary Report: Vibration Results and Observations During CSN-1A Open Cycles Nos. 5, 6, 7, and 8," 4 June 1964

Presents the results of open-cycle CSN-1A vibration and lifetime tests. Recommends the use of Bently probes for vibration monitoring based on test experience. The CSN-1A successfully completed the test program.

11. CSN-1A Performance Evaluation

- a. AN-AGCR-702, "Preliminary Performance Evaluation of CSN-1A T-C Set," 27 May 1964

Presents the preliminary performance analysis of the CSN-1A t-c set as operated during ANSOP 16625A. Performance is shown to be no better (45 kw output at design conditions) than that observed prior to the CSN-1 modification. Poor performance of the t-c set was attributed to a partial stall condition felt to be present in the last four stages of the compressor. Such a stall condition is shown analytically to be possible and is due to the effective flow area in the last few stages of the compressor being too large. The installation of additional interstage instrumentation is recommended to confirm the stall condition; it is also suggested that the turbine flow area be opened to alleviate the condition.

- b. AN-AGCR-754, "CSN-1A Performance Improvement Modifications," 1 September 1964

Summarizes the CSN-1A performance deficiencies and the changes to the compressor inlet and discharge required to improve performance. It is suggested that the t-c set output would have been 212 kw based on previous calculations had the turbine nozzle shim not failed during ANSOP 16625A. It is proposed that a series of tests and analyses be performed to optimize the compressor inlet configuration. A possible 80 kw output power increase is identified with such an optimum inlet configuration. A similar program is outlined for optimizing the compressor outlet configuration with a projected resulting increase in power output of 30 kw. Descriptions and layouts of the various proposed changes to inlet and outlet configurations are included.

12. CSN-1A Inspection

- a. AN-AGCR-716, "CSN-1A Disassembly Inspection Report; June 1964 Disassembly," 19 June 1964

Describes the results of the CSN-1A t-c set disassembly and inspection at the ML-1 test site, NRTS, by Clark Bros. technicians and engineering representatives from Clark and AGN. Presents the description of the condition of 22 t-c set components following the conduct of ANSOP 16625A. It was noted that major damage had occurred in the turbine. The inlet nozzle shim had moved downstream and contacted the first stage turbine rotor blades at the leading

edge. Prime cause of shim failure was felt to be the failure of 3 tack welds used to position the shim. The blades exhibited cuts approximately 0.250 in. deep by 0.060 to 0.070 in. long. Minor wear was noted in the seals and bearings. Foreign deposits and fouling were seen in the compressor stator assembly and, to a lesser degree, on the rotor blades. The metallurgical and mechanical evaluation of these conditions is presented in AN-AGCR-756 and AN-AGCR-886.

- b. AN-AGCR-756, "Metallurgical Evaluation of CSN-1A Journal Bearings," 3 September 1964

Presents the results of metallurgical evaluation of CSN-1A journal bearings after an emergency shutdown operation during ANSOP 16625A wherein the bearings experienced a temperature of 400°F for about one hour. Concludes that the bearings were not degraded by this exposure.

- c. AN-AGCR-886, "CSN-1A Mechanical Evaluation Following ANSOP 16625A," May 1965

Presents the results of a detailed assembly and inspection of the CSN-1A t-c set after it had experienced a turbine bearing temperature of 400°F due to an emergency shutdown condition during the conduct of ANSOP 16625A. The following discrepancies were noted and were corrected as indicated.

- Damaged first stage turbine blades were replaced.
- The turbine inlet liner was modified to eliminate the requirement for the shim.
- The galled surface on the rotor shaft, damaged during disassembly, was repaired.
- A new turbine face labyrinth seal was fabricated to replace the partially failed bond between turbine interstage honeycomb seal backing material and the stator ring member.
- Pressure and temperature sensing rakes were modified to eliminate interaction with the rotor.

In addition, white deposit, analyzed as AgCl, was found on the compressor stator and was determined to be the reaction product of the silver in the braze filler and chloride from flux used to fabricate the compressor stator. Compressor and turbine journal bearings did not show excessive wear.

13. CSN-1A Performance Evaluation

- a. AN-AGCR-810, "ANSOP 16650 Preliminary T-C Set Performance Report," January 1965

Presents a preliminary analysis of the CSN-1A t-c set performance during ANSOP 16650. Results indicate that the overall t-c set power output was decreased approximately 20 kw to 30 kw from that obtained during ANSOP 16625A. Compressor efficiency was about 2.5% lower while the turbine efficiency remained unchanged from that observed during ANSOP 16625A.

- b. AN-AGCR-887, "Calculated CSN-1A Compressor Performance Map," 15 May 1965

Presents the analytical method used to construct the CSN-1A compressor map as correlated from the test results of ANSOP 16650. The calculated compressor map indicates those areas extrapolated from actual test data.

- 14. AN-AGCR-898, "CSN-1A Disassembly Report, April 1965 Disassembly," 3 June 1965

Describes the results of a preliminary inspection of the CSN-1A prior to shipment of the set to the General Electric Co. for modification. Gross decontamination of the set was also performed at this time. The general condition of the CSN-1A was satisfactory although the unit was extensively fouled with rust deposits, presumably the result of the shield solution which had entered the unit subsequent to shutdown.

15. CSN Bearing/Rotor System Evaluation

- a. AN-AGCR-647, "Bearing Development Program," 26 June 1964

Presents the outline of an analytical and experimental program to develop and demonstrate long-term reliability of the CSN-1A type bearings and to develop a reduced pressure differential bearing for the CSN-type machines.

- b. AN-AGCR-664, "Proposed Seal and Bearing Development Program for Air-Cycle Operation," 28 February 1964

Describes a proposed program to develop seals and bearings for the CSN-1 t-c set operating in an air environment.

- c. AN-AGCR-759, "An Evaluation of the State-of-the-Art of High Speed Seals," December 1964

Presents the results of a literature search conducted to determine and evaluate the various types of seals available for ML-1 air cycle operation. Summarizes the design criteria of the seals and presents a theoretical evaluation of the six seals selected in the literature search.

- d. AN-AGCR-803, "Test Report - Initial CSN-1A Type Bearing Test 6 Through 16 October 1964," January 1965

Presents the results of CSN-1A bearing tests. The test was terminated after 3.3 hrs at speeds from 0 to 18,200 rpm when excessive vibration was detected. Post-test disassembly indicated that the bearings had failed, the rotor was out of balance, and the rotor unbalance was felt to be the primary cause of the excessive vibration and subsequent bearing failure.

- e. AN-AGCR-808, "Test Report - Second CSN-1A Type Bearing Test (30 October through 8 December)," January 1965

Presents the results of the second attempt to complete a 25 hr initial bearing test. This test was terminated after 9.8 hr of testing from 0 - 22,000 rpm (full speed) when excessive vibration was encountered. Earlier attempts to decrease vibration by lowering lubrication oil temperatures from 130°F to 112°F did not significantly reduce vibration at full speed. Post-test disassembly indicated that both bearings were damaged beyond repair but not to the extent of those of the initial bearing test. Rotor unbalance was again seen to be the primary cause of vibration and subsequent bearing failure even though the rotor had been balanced prior to testing.

- f. AN-AGCR-818, "Test Report - Initial 25 Hour CSN-1A-Type Bearing Test - 29 December through 31 December 1964," February 1965

Presents the results of the successful 25.6 hours of bearing testing from 0 to 22,000 rpm (full speed). Post-test disassembly indicated that the condition of the bearings was satisfactory. Success of the tests was due to the near perfect balance of the rotor which minimized vibration induced bearing wear.

- g. AN-AGCR-821, "Report from Professor D. D. Fuller on Application of CSN-1A Bearing for Low Supply Pressure and Ambient Pressure," 16 February 1965

Presents the results of analysis performed to determine the operating characteristics of the CSN-1A bearing under conditions similar to those expected during ML-1 air operation or the proposed ML-1A system.

- h. AN-AGCR-823, "Bearing and Seal Test Facility Design Report," March 1965

Describes the design of the facility used to test CSN-type bearings and seals. The facility is designed to provide the necessary equipment to explore experimentally the lifetime and stability of the CSN-1A bearings over a wide range of operating conditions.

- i. AN-AGCR-848, "Test Report - CSN-1A Type Bearing Stability Evaluation Test," March 1965

Presents the results of a 98.1 hour bearing test performed to determine the stability characteristics of the CSN-1A bearings

as a function of bearing differential pressure, inlet oil temperature, and rotor speed. Bearing stability was shown to be satisfactory over the full range of rotor speeds although the bearing stability did show a sharp dependency upon rotor speed particularly at the theoretical synchronous whirl speed of the bearings. The vibration amplitude also was somewhat affected by the inlet oil temperature (100°F-175°F); bearing differential pressure (40 - 70 psid) did not affect vibration. Post-test inspection revealed no damage to the bearings.

- j. AN-AGCR-849, "Test Report - 500 Hour CSN-1A Type Bearing Test," March 1965

Presents the results of a 514.3 hour bearing lifetime test conducted at 22,400 rpm rotor speed. Rotor vibration amplitudes were low and relatively constant. Post-test disassembly indicated that both bearings were in satisfactory condition.

- k. AN-AGCR-901, "Test Report CSN-1A Type Bearing Stability Evaluation Test - Addendum A," June 1965

Presents the results of bearing stability tests conducted at 22,400 rpm rotor speed for 94.8 hours. Results indicate that bearing stability characteristics were not adversely affected when the bearing lubricating oil differential pressure was lower than reference (20-40 psid) nor when bearing oil temperature was varied (94-265°F). Overall vibration amplitudes were low throughout all testing. Post-test inspection revealed that the condition of both bearings was satisfactory.

- l. AN-AGCR-902, "Report from Professor D. D. Fuller on Results of CSN-1A Type Bearing Stability Evaluation Test, Addendum A," 9 June 1965

Presents an analysis of bearing stability based on bearing lubricating oil temperatures and pressures used during bearing stability tests. Results of the analytical work compare favorably with test results indicating the calculational method used to predict bearing stability is valid.

16. AN-AGCR-377, "CSN-1 Modification Proposal," November 1962

Outlines a program to modify the CSN-1 t-c set to improve the performance of the unit and to permit operation of the unit with air as the working fluid. Proposed modifications include both aerodynamic redesign and mechanical seals and bearings redesign and development.

17. AN-AGCR-963, "CSN-1A Turbine-Compressor Set Modification (Interim Report)," December 1965

Summarizes activity relevant to CSN-1A modification to improve t-c set performance through December 1965. Includes the selection of General Electric Co. to perform the required modification and the status of that modification at the time of contract termination.

18. Starting Motor Evaluation

- a. AN-AGCR-468, "ML-1 Start Motor," 21 February 1963

Presents the results of initial tests performed on the ML-1 start motor. The results showed that the performance did not satisfy the design specifications and that the motor would not be acceptable as a prototype unit.

- b. AN-AGCR-338, "ML-1 Start Motor Power," 6 September 1962

Presents the results of analytical efforts used to extrapolate start motor power requirements from the results of test data through a wide range of ambient temperatures. Results indicate that start motor power requirements decrease with increasing ambient temperatures.

19. AN-AGCR-550, "Final Design Report ML-1 Start Motor," July 1963

Presents the design of the ML-1 start motor for the CSN-1 t-c set. Includes the start motor performance and design specifications and the results of performance testing.

8. Alternator Design and Development

The reference design of the ML-1 power plant contemplated the use of a brushless alternator because of the space limitations on the p/c skid. This concept had been successfully developed by Aerojet in smaller machines and it was anticipated that extrapolation of the technology to the 400 kw size required for the ML-1 would be feasible. The design of the larger machine was completed, incorporating what were felt to be the significant characteristics of the smaller machine. The alternator was fabricated and testing was initiated. It was apparent that the voltage generated was only about 50% of the design value. Subsequent investigation revealed that the deficiency was the result of excessive leakage between the rotor poles and an unsatisfactory flux distribution across the inactive air gap.

A wooden model of the rotor was fabricated and sectioning of this model revealed a magnetic short circuit between the poles and other paths of probable high flux leakage. The rotor was redesigned and the performance of the new configuration was evaluated in static tests of two scale models of different sizes. These tests indicated that a 32 in. rotor would produce 101% of the design flux at 0.8 pf⁽¹⁾ with 45 KVAR capacitance per phase connected in parallel with the machine.

A redesigned alternator was fabricated incorporating the new rotor configuration. During a series of tests, it was shown that the machine would develop rated power (400 kw at 0.8 pf) but that two major deficiencies existed.⁽²⁾ A magnetic unbalance of the rotor prevented satisfactory mechanical performance and losses, especially eddy current losses, were excessive (approximately 20% of output). In the absence of well-defined solutions for these problems, a decision was made to abandon further development of this concept. In addition, it was recognized that a larger machine (600 kw) would be required to utilize the total t-c set shaft power output at ambient temperatures less than 100°F.

During testing of the ML-1 power plant at the NRTS (see I-F), the output of the t-c set was absorbed and measured by an eddy current dynamometer since an alternator was not available. ML-1A preliminary design study (see VII-B) indicated that a commercial alternator could be accommodated on the p/c skid and such a unit was procured under the ML-1 Skid Optimization Program (see V-G-4).

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ALTERNATOR DESIGN AND DEVELOPMENT BIBLIOGRAPHY

1. AN-AGCR-225, "ML-1 Alternator Design Scale Rotor Tests," June 1962

Reports the results of static tests of two scale model brushless alternator rotors which established the final design configuration for the rotor assembly.

2. AGN TM-406, "Alternator Final Design Report," June 1964

The development and testing of a demonstration brushless alternator (with one heteropolar and one homopolar axial air gap) for the ML-1 power plant are described. The development was discontinued before solutions to the problems of low efficiency and magnetic unbalance of the rotor could be found. The machine could not be operated for significant periods of time without overheating and bearing damage. Completion of the development of a 400 kw brushless alternator appeared feasible but would require an increase in the size and weight of the machine which was not compatible with the power conversion skid design which existed at the time.

9. Instrumentation and Controls Design

The design of the ML-1 instrumentation and controls was initiated in mid-1959. The original design effort was completed by mid-1960.⁽¹⁾

The basic objectives of the design effort were to provide a compact, light-weight, easily-maintained, reliable system which would control reactor temperatures and turbine-compressor set speed while monitoring all critical plant parameters and including provisions for controlling and monitoring electrical power distribution. In addition to the basic operational controls, reactor scram circuitry was provided to protect the plant in the event of conditions which would compromise the safety or continued operability of the plant. An extensive annunciator system was provided to warn the operator of conditions requiring corrective action (Table 13). A source of power was provided to operate critical components of the power plant during emergency shutdown conditions. All controls and instrumentation were designed for fail-safe operation.

The control and instrumentation system was housed in a one man, air-conditioned control cab (modified S-141 shelter) which included a graphic display of the primary and critical auxiliary systems of the power plant. The control cab was designed for transport in the bed of an M-35 truck and weighed approximately $2\frac{1}{2}$ tons. In addition to the basic instrumentation, controls and emergency power supply required for power plant operation, a significant number of recording and indicating instruments were provided for accumulating data necessary for detailed analyses of the performance of the power plant. These instruments, together with closed-circuit television receivers to permit observation of power plant equipment during operation, were housed in an auxiliary control room immediately adjacent to the control cab.⁽²⁾

The power plant control system consisted of three automatic servo-loops as shown schematically in Figure 17. The control system was based on the concept of maintaining a constant inventory of working fluid in the system (constant mass). During normal steady state power plant operation, control was automatically accomplished by the speed control servo-loop, the reactor outlet temperature servo-loop, and the moderator water temperature control servo-loop. Direct manual control was used during startup and shutdown operations. The monitoring instrumentation included a display of nuclear, process loop, and electrical load parameters in logical groupings in the control cab.

The control cab, complete with all control components and instrumentation, was delivered to the ML-1 test site early in 1961 and used to control the reactor during the initial criticality and low-power physics experiments. This operation permitted the accomplishment of "de-bugging" tests and extensive, but minor in nature, wiring modifications were made on much of the ML-1 instrumentation during this period. Studies of the process instrumentation safety and reliability were performed in 1962.⁽³⁾ Early in 1963, an extensive program was initiated to update and standardize all process controls and instrumentation components in the ML-1 control cab to reflect the results of operating experience, bench testing, studies made by the Human Engineering Laboratories and advancements in the state-of-the-art.⁽⁴⁾ The performance of the instrumentation and control system was evaluated continuously during the

TABLE 13 - ML-1 ANNUNCIATORS

Annunciator Designation	Function	Normal Set-Point
1-A	Fast Period Channel 1 Scram	10 sec.
1-B	Fast Period Channel 2 Scram	10 sec.
1-C	Fast Period Channel 3 Scram	10 sec.
1-D	Fast Period Channel 4 Scram	10 sec.
1-E	High Power Channel 5 Scram	110%
1-F	High Power Channel 6 Scram	110%
2-A	High Power Channel 7 Scram	110%
2-B	High Reactor Outlet Temperature Scram	1250°F
2-C	Moderator Low Flow Scram	3.8 psig
2-D	Moderator ΔP Low Flow Scram	10.0 psid
2-E	Power Failure Scram	Actuated by loss of power to precooler fan circuit breakers
2-F	Manual Scram	Actuated by depression of manual scram button
3-A	Turbine Speed High Scram	24,200 rpm
3-B	Loop Pressure Low Scram	50 psia
3-C	Fast Pressure Loss Scram	45 psi/10 sec.
3-D	Scram Circuit Bypassed	
3-F	Reactor Outlet Temperature High	1225°F
4-A	Reactor Outlet Temperature High	1225°F
4-B	Compressor Inlet Temperature High	145°F
4-C	Compressor Inlet Pressure High	130 psia
4-D	Compressor Inlet Pressure Low	55 psia
4-E	Cab Radiation High	7.5 mR/hr
4-F	Skid Area Radiation High	800 R/hr
5-A	Moderator Water Inlet Temperature High	180°F
5-B	Moderator Water Inlet Temperature Low	140°F
5-C	Moderator Outlet Temperature High	190°F
5-D	Moderator Water Level Low	-7 inches
5-E	Moderator Water Conductivity High	2.0 μmhos
6-A	Bearing ΔP Low	24 psid
6-B	Bearing Temperature High	220°F (Radial); 170°F (Thrust)
6-C	Lubrication Oil Sump Level Low	30 gallons
6-E	Oil Filter ΔP High	30 psid
6-F	Oil Separator Overflow	
7-A	Seal ΔP Low	12.5 inches water
7-B	Shield Water Level Low	-4.0 inches
7-C	Shield Water Temperature Low	140°F
7-D	Shield Water Conductivity Low	150 μmho
7-E	Dynamometer Malfunction	140°F cooling water temperature 2.0 psig (flow monitor)
7-F	Inverter Battery Voltage Low	125 volts
8-A	Start Motor Overload	
8-B	Sump Equalizing Compressor Overload	
8-C	Startup Compressor Overload	
8-D	Moderator Pump Overload	
8-E	Moderator/Lubricating Oil Fan Overload	
8-F	Precooler Fan Overload	
9-A	Count Rate Low - Channel 1	12 cps
9-B	Channels 1 and 2 "ON/OFF" Interlock	5×10^3 - 8×10^4 nv
9-C	Automatic Temperature Control Permissible Interlock	3×10^8 nv
9-D	Regulating Rod Travel at Limit	70 degrees
A-1	Waste N ₂ Storage Tank Pressure High	14.5 psig
A-2	Ion Chamber Pressure Low	3.0 psig
A-3	Ion Chamber Pressure High	7.5 psig
A-4	N ₂ Bank Pressure Low	500 psig
B-1	Turbine Seal ΔP Low	.5 psid
B-2	Compressor Seal ΔP Low	.5 psid
B-3	Fuel Element Temperature High	1400°F
B-4	Hydrocarbon Analysis High	93-97%
C-1	100 kw Diesel-Generator Failure	
C-2	60 kw Diesel-Generator Failure	
C-3	Moderator Vent Flow High	5.0 inches water
C-4	Fission Products High (B ¹⁰ Chambers)	5.0 R/hr
D-1	Dynamometer Sump Level Low	-10 inches
D-2	Compressor Bearing Oil ΔP Low	25 psid

* 100% = 11 ppm Hexane @ 100 psia.

FIGURE 17. ML-1 CONTROL SYSTEMS

conduct of all reactor and power plant operation. Analytical studies, based on the results of operational testing, were performed to uncover design deficiencies and to improve the performance of the control and instrumentation components.⁽⁶⁾

As a result of these studies, improved components were designed, fabricated and installed.⁽⁶⁾ The last of these improvements was the off-normal scram system which protected the fuel elements from overheating during startup and similar transient conditions; this system was bench tested but not installed prior to contract termination.⁽⁷⁾

Health physics instrumentation provided for the ML-1 plant monitored radioactive system effluent (precooler cooling air outlet) and radiation dose levels at the control cab and power conversion skid. In addition, a number of instrumentation systems were provided to monitor miscellaneous plant conditions not necessarily directly required for plant safety, control, or performance evaluation.⁽⁸⁾

A program was undertaken to evaluate the adequacy of the plant protective (scram) circuits. Analytical studies, based upon experimentally derived component characteristics, were performed to determine the response behavior during and following abnormal operating conditions.⁽⁹⁾ These studies indicated that adequate protection was provided by the ML-1/GCRF scram circuits.

INSTRUMENTATION AND CONTROLS DESIGN BIBLIOGRAPHY

1. Instruments and Controls Design

- a. AN-ENGR-17, "Control System for the ML-1 Nuclear Power Plant," May 1961

Presents the design of the ML-1 instrumentation and control systems including the design philosophy and criteria and the final design. Includes block diagrams of the main servo-loops and P&I diagrams of the ML-1 system.

- b. AGN TM-378, "ML-1 Components Design Report," July 1960

Presents a detailed design description of all ML-1 components including the instrumentation and controls of the ML-1 power plant.

- c. IDO-28550*, "ML-1 Design Report," 16 May 1960

Presents the overall design description of the ML-1 power plant.

2. AN-AGCR-683, "A Review of Analysis Instrumentation," 24 March 1964

Presents a detailed summary of the type, location, calibration, and expected accuracy of the various instrumentation components and systems used in system performance analysis. The instruments discussed in this document are located in the auxiliary control building.

3. Process Instrumentation Review

- a. AN-AGCR-210, "Safety and Reliability Review of ML-1 Process Instrumentation," 10 May 1962

Presents the results of a study performed to assess the safety and reliability of ML-1 instrumentation. The study covers only the control cab-mounted instrumentation and not those instruments required for system performance evaluation. Suggests some additional and modified instruments.

- b. AN-AGCR-16, "ML-1 Emergency Power Paralleling System," 6 March 1962

Presents a review of the emergency power paralleling system and recommends some testing and circuitry modifications to improve the reliability of the system.

- c. AN-AGCR-246, "Safety and Reliability of ML-1 Process Instrumentation," June 1962

Itemizes the action to be taken on the reinstrumentation of the power plant as suggested in AN-AGCR-210.

4. Process Instrumentation Improvements Studies

- a. AN-AGCR-676, "ML-1 Scram Logic Circuit," 24 March 1964

Describes a modification to the ML-1 scram logic circuit which improves plant safety and operational reliability by providing circuit redundancy. Recommends additional modifications to the circuit; these modifications were performed under ANSOL 0803-328.

- b. AN-AGCR-511, "ML-1 Fast Pressure Loss Scram System Design Report," May 1963

Presents the design and results of calibrational testing of the modified fast pressure loss scram system. The original system scrambled the reactor if the total pressure loss exceeded 45 psi in 10 seconds; the improved system scrams the reactor if the pressure loss rate exceeds 4.5 psi/sec and if the total pressure loss exceeds 40 psi.

- c. AN-AGCR-513, "Improved ML-1 Pressure Instrumentation," 1 May 1963

Presents a proposed improved and standardized system to sense and display various plant pressures including compressor inlet and outlet, lube oil sump, and bearing inlet oil pressure.

- d. AN-AGCR-522, "Improved Spare Nuclear Drawers," 10 May 1963

Presents a proposed program for modification of nuclear instrumentation system to improve performance and interchangeability and for the fabrication of spare nuclear drawers.

- e. AN-AGCR-525, "Improved ML-1 Process Temperature Instrumentation," 13 May 1963

Presents a proposed program for improvement of the process temperature instrumentation including standardization of meter scales, use of interchangeable components, consolidation of chassis with similar functions and improvements in the stability of the circuitry.

- f. AN-AGCR-563, "Improved ML-1 Speed and Temperature Controller," August 6, 1963

Presents the preliminary design for an improved ML-1 speed and temperature controller. Includes an analysis of the performance of the original controller which points out the deficiencies of that design. The proposed improved system uses: 1) modular construction to improve maintainability, 2) standardized components to simplify spare parts inventory, and 3) tested circuitry to

assure reliability and stability; performance is improved in terms of response time and positional accuracy of the controls.

- g. AN-AGCR-489, "Improvements to ML-1 Emergency Power System," 28 March 1963

Presents the results of investigations performed to determine the ML-1 emergency power requirements based on operating experience and proposes to incorporate these results in the design and procurement of a new system which is more compact and reliable than the original system. The proposed system separates the regulated power and emergency power busses to avoid the use of a voltage regulator and emergency power source of greater capacity than would be required to supply only the respective loads. In addition, this configuration would provide emergency power to only such equipment as is required during an emergency situation, thereby reducing the power requirement to approximately 1440 va (previous total was 2630 va).

- h. AN-AGCR-328, "ML-1 Speed Control Valve History," 21 August 1962

Summarizes the test and operational experience with the speed control valve and presents a description of the valve operating mechanism and performance specifications (see AN-AGCR-371 also).

- i. AN-AGCR-371, "ML-1 Turbine Speed Controller Test and Operational History," November 1962

Presents the design criteria, results of analog computer analysis, bench test results and operational experience of the turbine speed controller system and outlines a program to improve this system.

- j. AN-AGCR-378, "ML-1 Turbine Speed Control Valve Modifications," 8 November 1962

Presents a summary of steps taken to correct deficiencies in the turbine speed control valve.

5. Instruments and Controls Performance Analysis

- a. AN-AGCR-703, "ML-1 Turbine Speed and Nuclear Instrumentation Performance - ANSOP 16625A," 2 June 1964

Reviews the various control and instrumentation malfunctions associated with the turbine speed and nuclear instrumentation noted during the conduct of ANSOP 16625A. Summarizes the failure analysis and indicates where corrective action has or should be made.

- b. AN-AGCR-729, "ML-1 Nuclear Instrument Status," 17 July 1964

Supplements the instrumentation performance evaluation presented in AN-AGCR-703 using additional field data. Some of the original conclusions reached regarding instrument malfunction are altered.

- c. AN-AGCR-795 (795A), "Failure of ML-1 Rod Control Switch," December 1964

Presents the failure analysis and summarizes the operational effects of the ML-1 rod control switch malfunction. The immediate corrective action taken is discussed and a recommended long-range program for correction is presented.

6. Process Instrumentation Improvement

- a. AN-AGCR-768, "Improved Overspeed Scram Chassis," September 1964

Presents the improved overspeed scram system design criteria, summarizes the design and evaluation program and presents the design specifications and assembly and schematic drawings.

- b. AN-AGCR-794, "Improved Overspeed Chassis," December 17, 1964

Presents a critique of the improved overspeed chassis design and includes some corrections to the original design report (see AN-AGCR-768).

- c. AN-AGCR-870, "Design Review, ML-1 Improved Automatic Speed/Temperature Controller," 4 May 1965

Presents a design review of the "as-built" automatic speed/temperature controller system which controls reactor outlet temperature by adjusting reactor power (i.e., position of regulating rod) and controls turbine-compressor speed by bypassing a portion of the compressor outlet gas to the precooler inlet.

7. Off-Normal Scram System

- a. AN-AGCR-631, "ML-1 Startup Power Level Scram System," 20 December 1963

Reports the results and conclusions of a study performed to evaluate the various methods by which an automatic scram and/or alarm system could be provided to prevent violation of the fuel element hot spot temperature limits during plant startup where automatic protective circuitry used during steady state operation is inadequate. Recommends a program to optimize the various systems studies and the preparation of a preliminary design.

- b. AN-AGCR-897, "Preliminary Design Report: Off-Normal Scram System," June 1965

Describes the preliminary design, including design criteria, technical description, and component specifications, of the off-normal scram system. The system essentially computes the fuel element hot spot temperature from reactor inlet and outlet temperature and mass flow rate and then compares the calculated temperature with the preset limit.

- c. AN-AGCR-962, "Final Design Report: Off-Normal Scram System," December 1965

Presents the final design of the off-normal scram system. Includes the design criteria and analysis together with component specifications and the results of environmental testing of the reference system.

8. Miscellaneous Instrumentation Systems

- a. AN-AGCR-414, "ML-1 Fission Product and Effluent Air Monitoring System," 14 February 1963

Describes the design and operation of systems for the collection and analysis of radioactive gases from the primary gas loop.

- b. AN-AGCR-281, "ML-1 Hydrocarbon Detection System," 27 July 1962

Presents the results of a study performed to evaluate various methods of hydrocarbon detection in the primary gas loop. Recommends the checkout, calibration and installation of two candidate detectors at the ML-1 for subsequent comparative evaluation.

- c. AN-AGCR-376, "ML-1 Radioactive Effluent Monitoring," November 1962

Presents the design of the monitoring and sampling system included in the ML-1 to control atmospheric releases of radioactive effluents. Includes design criteria, derivation of analytical equations, and computer programs used in the analysis of the system.

- d. AN-AGCR-567, "Reliability Study of Beckman Model 109 Hydrocarbon Analyzer," 29 October 1963

Presents the results of field testing of the Beckman Model 109 Hydrocarbon Analyzer as recommended in AN-AGCR-281 to determine lifetime gas consumption, and calibration drift of the instrument. Concludes that this instrument is satisfactory and reliable for field use.

9. Scram System Response Analysis

- a. AN-AGCR-312, "High Temperature Scram Response Times,"
1 October 1962

Presents the results of experimental work performed to determine the response times of a resistance bulb and a chromel/alumel thermocouple used to sense reactor outlet temperature as part of the high temperature scram system. Concludes that the thermocouple system is generally superior, although in some cases the resistance bulb system shows an advantage.

- b. AN-AGCR-315, "Nuclear Instruments Scram Response Time,"
12 November 1962

Presents the results of tests performed on ML-1 nuclear instrument circuitry to determine scram response times both before and after instrument modification to minimize spurious scrams caused by rapid, microsecond-duration power transients.

- c. AN-AGCR-424, "Nuclear Detector Response (Experiment 16607),"
5 June 1963

Summarizes the results of experiments performed to determine nuclear detector response time base data to be compared to data obtained during subsequent operational instrument checkout.

- d. AN-AGCR-687, "Nuclear Detector Response-II (Experiment 16607),"
29 April 1964

Presents the results of tests performed to determine the response of the nuclear detectors after the extensive reactor repair in 1963. Comparison of these results with previously obtained data indicates that the response times were essentially unchanged.

- e. AN-AGCR-903, "ML-1/GCRF Scram Response Time Analysis,"
10 June 1965

Proposes and outlines a program to determine the ML-1 reactor scram response times when operated in the GCRF by using analog computational techniques.

- f. AN-AGCR-944, "ML-1/GCRF Analog Model for Scram Response Time Analyses," 1 October 1965

Presents the analog model of the ML-1 reactor operating in the GCRF test loop. The model is based on experimentally derived system and component characteristics obtained during power plant and loop testing.

- g. AN-AGCR-961, "ML-1/GCRF Scram Circuit Response Study,"
December 1965

Presents the final results of the ML-1/GCRF scram system response study based on experimentally derived system characteristics using analog computational techniques. Abnormal and steady state conditions were analyzed and the allowable response times were derived. These were compared with measured response times and it was concluded that the response of the scram circuitry was adequate for protection against damage resulting from the abnormal operating conditions investigated.

10. Auxiliaries Design

The development of the auxiliaries required to support the field operation of the ML-1 power plant is summarized in this section.

a. Gas Storage Skid

Design, fabrication and testing of the gas storage skid was completed late in 1961.⁽¹⁾ The skid was designed to supply gas for the initial filling of the ML-1 gas system and sufficient makeup gas for four system changes. In addition, the skid included a vacuum pump and related equipment required to evacuate and store contaminated gas from the power plant loop. The major skid components were a two-stage diaphragm compressor, a rotary, mechanical vacuum pump, and two high pressure gas storage spheres. Suitable piping, valves, instruments and controls were provided to permit remote operation from the power plant control cab. The equipment was skid-mounted in a configuration suitable for transport in a military 2½ ton truck; the overall dimensions were 80" x 94" x 76" and the unit weighed approximately 6000 lb.

b. Cable Reel Assembly

A cable reel skid was designed and fabricated for storage (during transport) of power and control cables used for connecting the power plant control cab to the power conversion skid. Two 24 x 64 in. diameter drums 30 in. wide were mounted horizontally on a common shaft and driven by a 2 hp electric motor. The cable reel equipment was skid-mounted in a configuration suitable for transport on a 2½ ton military truck.

c. Water Treatment Equipment

Equipment was designed and fabricated to provide high quality water for makeup for the moderator system and for shield solution preparation. The design contemplated processing water from a standard Army purification unit to remove dissolved solids. Such water could then be added directly to the moderator system or used to mix the boric acid shield solution. The equipment, including a demineralizer, pump, collapsible tank and hoses, was skid-mounted in a configuration suitable for transport in a standard military 2½ ton truck. The overall dimensions were 78" x 90" x 82" and the skid weighed approximately 1400 lb (provision was made for the storage of 2800 lb of auxiliary supplies on the skid).

d. Refueling Equipment

The development of equipment required for field refueling of the ML-1 reactor was partially completed. Experimental hardware for the remote handling of the radioactive pressure vessel cap and fuel elements was designed, fabricated and demonstrated. This developmental effort was terminated before the design of a transfer cask and cask handling equipment was initiated because of a change in program concept; it was decided that all reactor refueling would be accomplished at a rear echelon facility equipped for this function and that only the capability to remove and replace single fuel elements was required in the field. The packaging of the refueling equipment, consisting of the shield tank extension and tank cover (400 lb), the personnel

bridge (1900 lb) and the bridge support structure (2100 lb), was not optimized but it appeared that the packaging would consist of three 5' X 5' X 12' boxes.

Included as a part of the refueling equipment was a reactor drying skid. This assembly was capable of supplying 200 lb/hr of electrically-heated (to 600°F) air to the reactor (after uncoupling from the p/c skid) to evaporate the water left in the gas passages after the flooded reactor was drained following a refueling operation.

e. Auxiliary Power System

The design objective for the ML-1 specified that the plant be capable of startup with no more than 45 kw of auxiliary electrical power, provided by a diesel-generator unit. However, the aerodynamic characteristics of the t-c set were such that, for most of the operational testing, a 60 kw unit was connected directly to the starting motor and a 40 kw unit provided auxiliary power for the balance of the plant. Although neither set was loaded to capacity, this arrangement was used during testing to minimize the possibility of damage to plant components in the event of an emergency shutdown.

f. Emergency Power System

The emergency power system for the ML-1 power plant consisted of a battery bank with a static inverter. The performance of this system was evaluated in 1963,⁽²⁾ and the design of an improved system was initiated. An interim system, consisting of ten 12 volt, 50 ampere-hour batteries, was installed in the ML-1 analysis building equipment room. The battery output was routed to a 2 KVA rotary inverter which supplied power to the control cab emergency bus. The development of the improved system was in progress at the time of termination of the AGCRS Program.

g. Emergency Cooling System

The design, fabrication, installation and checkout of an emergency reactor cooling system for the ML-1 power plant were completed in August 1962.⁽³⁾ Nitrogen coolant was provided to remove heat generated by the fuel elements in the event of coolant loss during power plant operation. The system was capable of providing a flow of nitrogen until such time as required for the amount of afterheat generated to decrease to that amount that could be removed by the moderator system. A nitrogen storage farm was located outside the ML-1 test building, the cooling skid was located inside the test building and the controls were mounted in the auxiliary control building and control cab. Tests performed late in the ML-1 program indicated that an emergency cooling system was not required except in the improbable case where less than 4 seconds of t-c set coastdown existed.⁽⁴⁾

AUXILIARIES DESIGN BIBLIOGRAPHY

1. Gas Storage Skid

- a. AN-AGCR-363, "ML-1 Gas Handling Skid," 19 October 1962

Presents a description of the gas handling skid as well as the development program for conversion to operation of the ML-1 with air as the working fluid.

- b. AGN TM-393, "Design Report of the ML-1 Gas Storage Skid," January 1962

Describes the gas storage skid design, fabrication and checkout. The design criteria, component design specifications, testing, operating instructions and fabrication drawings are presented.

2. AN-AGCR-583, "ML-1 Emergency Power System," 26 August 1963

Describes the improved ML-1 emergency power system and also presents the description of an interim system (see Section V-10-f) for use until such time as required for installation of the improved system. Both systems use batteries connected directly into the control cab circuitry to provide a completely independent power supply for operation of the ML-1 equipment required during times of power failure. The proposed improved system consists of a silver cadmium battery bank of approximately 1.9 kwh capacity weighing less than 100 lbs. Battery charging would be accomplished by a relatively simple, automatic battery charger tailored to the size of the battery bank. The batteries would operate a 2 KVA rotary inverter to supply power to the cab emergency power bus. This design would allow the entire emergency power system to be housed in one rack at a weight saving of over 250 lbs.

3. AN-AGCR-350, "Design Report - ML-1 Emergency Cooling System," 10 October 1962

Describes the design of the emergency cooling system to prevent excessive fuel element temperatures in the event of a coolant loss accident during operation of the ML-1 power plant. A high-pressure nitrogen supply was provided, along with the associated injection valves, piping, instrumentation and controls. The system could be actuated automatically by a loss of system pressure or manually from the control cab. The equipment was skid-mounted suitable for transport on a 2½ ton military truck. Drawings of the system are included in this report.

4. AN-AGCR-927, "ML-1 Coolant Loss Accident Analysis," August 18, 1965

Presents the results of analysis performed to determine peak fuel element shutdown temperatures as a function of coolant flow decay, operating time, and shutdown reactivity. Uses the mathematical model des-

cribed in AN-AGCR-827 and updates all previous analytical work in this area.

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11. Packaging

Conformance with the requirements for mobility represented one of the greatest challenges in the design and development of the ML-1 power plant. For example, the mobility criteria specified in the Military Characteristics were primarily responsible for the abandonment of the PGCR plate-type fuel element concept in favor of the pin-type design (which resulted in a more compact core) and for the decision to attempt the development of a 400 kw brushless alternator. Throughout the design phase, the weight and size of each component was as important a consideration as the capability of the component to perform the design function. Structural assemblies were designed specifically for minimum weight consistent with adequate performance. The equipment on the major skids was arranged to fit within the overall envelopes specified in the Military Characteristics while, at the same time, providing for maximum maintainability.

The mobility criteria also required that a shock isolation system be provided to protect the major plant skids from the shocks normally associated with loading and transport. An elastic polymer shock mount was developed and tested⁽¹⁾ and, subsequently, incorporated in the design of the structure of the skids. Each shock mount consisted of a top fitting, an elastomer core and a cylindrical retaining ring, all bonded into a single unit. The elastomer core was molded in the form of a cylinder with a tapered hole. A void at the bottom of the hole accommodated vertical deflections and the large bearing area of the elastomer absorbed lateral deflections by placing the elastomer in compression against the container. Satisfaction of the loading criteria for the various contemplated modes of transport was confirmed by actual loading and off-loading of mockups of the reactor and power conversion skids (duplicating the weight, overall dimensions and center of gravity) on an M-172 trailer, a railroad flatcar and C-130, C-133 and C-124 aircraft (Figure 18).⁽¹⁾

A special tiedown system was developed for use during plant transport. In this system, short lengths of nylon rope were used with the standard Military tiedown equipment to absorb the shocks associated with railroad car "humping" or aircraft emergency landings.⁽²⁾ This system was demonstrated to be adequate in tests conducted with the skid mockups and a railroad flatcar.⁽¹⁾

A detailed manual for loading, transport and off-loading the ML-1 plant was prepared utilizing the data and experience accumulated during the mockup tests.⁽³⁾ This manual was prepared to Army operational manual format and includes procedures for all contemplated (trailer, rail or air) modes of transport.

The transport capability of the plant was partially demonstrated when the ML-1 fully-fueled reactor skid and control cab were relocated from the ML-1 test site to the GCRF. The transfer was made on an M-172A trailer and the entire operation was conducted without difficulty.⁽⁴⁾

FIGURE 18. ML-1 POWER CONVERSION SKID MOCK-UP BEING LOADED IN C-124 AIRCRAFT

PACKAGING BIBLIOGRAPHY

1. IDO-28555*, "ML-1 Transportability Studies," 23 March 1960

Describes the results of comprehensive tests of loading and tiedown techniques to demonstrate compliance of the design with the Military Characteristics. Mockups, which duplicated the size, weight and center of gravity of the major plant skids, were loaded and transported on the M-172 trailer and a standard railroad flatcar. In addition, the mockups were loaded onto and off-loaded from C-130, C-133 and C-124 aircraft. The component test programs for the shock mounts and the nylon rope used in the tiedown system are also described.

2. AGN TM-379, "Proposed Nylon Rope Tiedown," 15 July 1960

Describes the tiedown and shock isolation system for air transport of the ML-1 power plant to assure plant serviceability following an emergency transport aircraft landing. The system incorporated lengths of nylon rope in conjunction with standard Military tiedown hardware. The design analysis and the results of component and system tests are also included.

3. AGN TM-389, "ML-1 Transportation Manual," October 1961

Presents the detailed procedures for the transport of the ML-1, including loading, tiedown, transport, off-loading and site preparation procedures, for all contemplated modes of ML-1 transport.

4. AN-AGCR-866, "ML-1 Reactor Relocation," 8 April 1965

Describes the results of the transport of the ML-1 reactor skid from the ML-1 test facility to the GCRF, including time and manpower requirements and radiation dose level data.

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12. System Design

a. Power Plant Design

The design of the ML-1 power plant is discussed in detail in the ML-1 Design Report⁽¹⁾ and the ML-1 Design Data Report.⁽²⁾ The safety analysis of the operation of the ML-1 plant at the NRTS is discussed in the preliminary hazards summary report⁽³⁾ and the final hazards summary report.⁽⁴⁾

b. Systems Analysis

One of the unique characteristics of a closed Brayton cycle power system is the significant interaction which occurs between system components and the effect of these interactions on the total power output. Figure 19 shows the ML-1 cycle diagram with typical full power state point data. A computer program⁽⁵⁾ was developed to permit the rapid computation of system performance based on component performance input data. This program contributed significantly to the original design of the power plant as well as to the understanding of the results of test operation and the formulation of modifications to power plant components.

A reference energy balance⁽⁶⁾ for the ML-1 power plant (Figure 19) and reference system analysis⁽⁷⁾ were published to serve as a basis for comparison of subsequent design and test data. A table of coefficients⁽⁸⁾ was developed from the system analysis data to permit the rapid approximation of the effect of changes in component performance on system performance without the requirement for a CHOP code run (Table 14). The transient behavior of the power plant was predicted on the basis of analog computer runs which formulated several standard and non-standard operating conditions.⁽⁹⁾

A computer program was developed for the rapid analysis and evaluation of ML-1 power plant test data.⁽¹⁰⁾ This program incorporated a statistical adjustment routine which determined state point temperatures and pressures that satisfied system energy and pressure balances. This code was used to evaluate the major body of test data generated during the conduct of ANSOP 16650 from which updated component performance characteristics were determined⁽¹¹⁾ and new influence coefficients were developed.⁽¹²⁾ The data developed during the conduct of ANSOP 16650 was also used to evaluate the effect of the several variables associated with t-c set operation and to define the optimum match points for maximum system output.⁽¹³⁾⁽¹⁴⁾

c. Special Studies

A program was initiated to evaluate the adequacy of the ML-1 scram systems by comparing the measured responses of the system with the theoretically determined maximum permissible response time based on the results of analog computer simulations of system behavior.⁽¹⁵⁾ This work was not completed but a parallel study of the adequacy of the ML-1/GCRF scram circuits was completed.⁽¹⁶⁾

A program of analytical evaluation and laboratory testing was undertaken to determine if the ML-1 power plant would operate satisfactorily with air as a reactor coolant and power cycle working fluid.

TABLE 14 - ML-1 INFLUENCE COEFFICIENTS

<u>Parameter</u>	<u>Influence Coefficient⁽¹⁾</u>
Turbine inlet temperature	.8 kw/ ^o F
Ambient temperature	-2.0 kw/ ^o F
Pressure drop ⁽²⁾	-17.0 kw/(% $\Delta P/P$)
Turbine efficiency	17.0 kw/%
Compressor efficiency	17.2 kw/%
Precooler effectiveness	10.8 kw/%
Recuperator effectiveness	1.4 kw/%
Bypass fraction	-14.5 kw/%
(1) Influence coefficients determined by calculation based on data obtained at near design operating conditions but coefficients have not been experimentally verified.	
(2) Applicable for either high-pressure or low-pressure side of system.	

The obvious decrease in logistic requirements for field operation provided the incentive for the investigation. Areas to be explored included the effect on thermodynamic performance, the effect on materials performance, the redesign requirements and the possibility of an air/oil vapor explosion. The results of these investigations⁽¹⁷⁾⁽¹⁸⁾⁽¹⁹⁾ were as follows:

- The effect on thermodynamic performance would be a change in net plant output of $\pm 2-4\%$.
- No problems were evident in the area of materials performance although only limited data was available on the air corrosion resistance of low-cobalt Hastelloy X.
- Significant redesign of the t-c set seals and bearings, lubrication system and makeup gas supply system was required.
- The radioactivity of the plant during operation or when shut down would not be increased significantly.
- The possibility of an oil-vapor/air explosion could be eliminated by design and administrative controls.

A study was performed to evaluate the feasibility of the use of a battery powered direct current auxiliary power system for the ML-1. This study revealed that the battery system was too large and heavy to be acceptable under the Military Characteristics and that a significant development program would be required to produce a suitable d.c. starting motor.⁽²⁰⁾

Liaison was established with and a series of briefing discussions conducted for representatives of the U. S. Army Human Engineering Laboratories. This activity formed the basis for an HEL evaluation of the ML-1

NOTE: 1. ALL STATE POINT (e.g., Δ) DATA ARE IN $^{\circ}\text{F}$, psia, and Btu/lb
 2. VALUES BASED ON DATA POINT NO. 529, ANSOP-16650
 AT NEAR DESIGN CONDITIONS

FIGURE 19. ML-1 POWER PLANT SCHEMATIC ENERGY BALANCE

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control cab⁽²¹⁾ which produced much useful information incorporated in the re-design of the ML-1 system and the preliminary design of the ML-1A plant. A preliminary study of spare parts and operating supplies requirements for one year of field operation of the ML-1 was completed.⁽²²⁾

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SYSTEM DESIGN BIBLIOGRAPHY

1. IDO-28550*, "ML-1 Design Report," 16 May 1960

Presents the design criteria, design calculations and reference design for the ML-1 power plant.

2. AN-AGCR-385, "ML-1 Design Data Report," 7 December 1962, 19 April 1963, and 21 February 1964

Presents current design data for the ML-1 power plant primarily for use by Government agencies for the review of proposed modifications of maintenance and operating procedures and for general orientation.

3. IDO-28537*, "Preliminary Hazards Summary Report for the ML-1 Nuclear Power Plant," 22 April 1959

Presents an analysis of the safety of operation of the ML-1 power plant based on the preliminary design of the plant.

4. IDO-28560*, "ML-1 Final Hazards Summary Report for the ML-1 Nuclear Power Plant," November 1960

Presents an analysis of the safety of operation of the ML-1 power plant at the NRTS based on the final design of the plant.

5. AGN TM-388, "CHOP - A Computer Program for Calculating Steady State," May 1963

Describes the computer program developed for the calculation of steady state thermodynamic performance of closed Brayton cycle power plants.

6. AN-AGCR-401, "ML-1 Energy Balance," December 1962

The energy balance for the ML-1 power plant is presented and compared with the limited data generated during the conduct of ANSOP 16615.

7. AN-AGCR-610, "ML-1 Reference System Analysis," 8 November 1963

Updates earlier work to present the first meaningful systems analysis based on as-built configurations and includes provision for the consideration of off-design component performance.

8. AN-AGCR-448, "ML-1 Design and Performance Influence Coefficient Tables," 18 January 1963

A table of influence coefficients for use in the rapid estimation of the effect of component changes on system performance is presented along with the derivation of the mathematical expressions for the development of coefficients and an evaluation of the accuracy of the results.

9. AN-AGCR-359, "ML-1 Transient Behavior," 14 September 1962

Reports the results of an analog evaluation of the transient behavior of the ML-1 system following a reactor scram from full power, large electrical load changes, a t-c set seizure and a coolant system rupture. In general, the most severe transients occurred during seizure of the turbine-compressor set or gross leakage from the primary coolant system. The significant reduction of flow rates reduced the convective heat transfer coefficients. However, most of these transients were of short duration, resulting in relatively small temperature changes in component materials.

10. AN-AGCR-843, "The ML-1 Data Reduction Computer Code DRML-1," March 1965

Describes the computer program developed for the reduction and evaluation of ML-1 test data. The code determines state point temperatures and pressures based on statistical adjustment of test data to satisfy system energy and pressure balances.

11. AN-AGCR-844, "ML-1 System Performance Characteristics," 25 March 1965

A series of curves depicting the performance of the ML-1 system components based on data generated during the conduct of ANSOP 16650 is presented.

12. AN-AGCR-928, "ML-1 Systems Influence Coefficients," 20 August 1965

The mathematical expressions for the derivation of updated ML-1 system influence coefficients are presented along with a revised list of all coefficients based on data generated during the conduct of ANSOP 16650.

13. AN-AGCR-931, "ML-1 Cycle Optimization," August 1965

The results of a study of the effects of turbine-compressor set design variables on system performance are presented. Conclusions are drawn from the information developed in 480 computer runs exploring the full range of operating parameters. Turbine expansion ratio, compressor pressure ratio, mass flow rate and loop pressure losses are the major parameters investigated.

14. AN-AGCR-935, "ML-1 "CHOP" Hardware Description," 17 September 1965

Presents a description of the ML-1 component performance characteristics in the input format suitable for use with the CHOP computer program.

15. AN-AGCR-873, "ML-1 Scram Response Time Analysis," 22 April 1965

This interim report describes the program to be conducted and the results obtained as of the time of publication in the investigation of the adequacy of the ML-1 system scram circuits.

16. AN-AGCR-961, "ML-1/GCRF Scram Circuit Response Study," December 1965

The program of scram circuit response characteristics measurements and allowable response time determinations (based on analog computer simulations of system performance) is summarized. It is concluded that the characteristics of the ML-1/GCRF scram circuits are adequate to protect the system from damage in the event of any of several abnormal operating conditions.

17. AN-AGCR-519, "Operation of the ML-1 with Air as the Working Fluid," May 1963

Summarizes the results of an evaluation of the data available as of the time of publication concerning operation of the ML-1 power plant on air and outlines the program of additional investigations required to confirm the acceptability of the concept.

18. AN-AGCR-626, "ML-1 Air Cycle Operation Interim Report and Proposed Program," 16 December 1963

Presents a report of the status of the air cycle investigations and a revised projection for completion of the program.

19. AN-AGCR-444, "Preliminary Report on Lube System Study for ML-1 Air Cycle," December 1962

Summarizes the results of the study as well as the lube system modifications required for air operation of the ML-1. The study concluded the major disadvantage of air operation to be the explosive hazard that could exist if oil entered the primary gas loop. Two simplified, reliable lube system designs are presented and a preliminary discussion of the advantages and disadvantages of each is included.

20. AN-AGCR-541, "Feasibility Study of ML-1 DC Auxiliary Power System," June 1963

Reports the results of a study to evaluate the feasibility of the use of battery powered d.c. auxiliary power system for the ML-1. It is concluded that the weight and size of the battery system are not consistent with the Military Characteristics and that a significant program of development is required to produce an acceptable d.c. powered start motor.

21. U. S. Army Ordnance Technical Memorandum 8-62, "A Human Engineering Evaluation of the ML-1 Mobile Nuclear Power Plant," April 1962
O.C.M.S. Code 5010.21.81902

Presents the results of a Human Engineering evaluation of the ML-1 control cab and establishes Human factors design criteria for use in subsequent modifications and redesign.

22. AN-AGCR-384, "ML-1 Spare Parts List," 30 November 1962

Presents the results of a study to define the spare parts and operating supplies required for one year of field operation of the ML-1 power plant.

D. ML-1 FABRICATION

The fabrication of the ML-1 power plant was funded with construction funds administered under Contract AT(10-1)-880; the USAEC provided funds for the construction of the reactor skid, control cab and auxiliaries while the DOD funded the construction of the power conversion skid.

The reactor skid was fabricated at the AGC Downey Plant in accordance with the designs developed by AGN. This work began in the fall of 1959 and the skid was delivered to the NRTS in February 1961. The fuel for the reactor was fabricated at AGN, San Ramon.

The control cab was fabricated at AGN and delivered to the NRTS in February 1961. This schedule did not permit a complete checkout; this activity was accomplished during the low-power reactor test program conducted at the NRTS in 1961 and early 1962.

The fabrication of the p/c skid was delayed, initially, by developmental problems with the heat exchangers and, subsequently, by delays in the delivery of an acceptable t-c set. The construction was performed at the AGC Azusa Plant. A completed skid, with a partially tested t-c set but without an alternator, was delivered to the NRTS in June 1962.

FIGURE 20. ML-1 TEST SITE

E. ML-1 TEST FACILITY

The design of the ML-1 test facility at the NRTS, Idaho, was performed by Norman Engineering Co. based on design criteria developed by Aerojet.⁽¹⁾ The construction of the facility was completed under a contract awarded by USAEC-ID in late 1960. The facility was designed to provide a weather protected test area for the power plant and a control/analysis site for operating personnel. The control and test areas, each surrounded by a barbed wire perimeter fence, were located approximately 500 feet apart, with a 12-foot high earth berm between the areas to provide a direct radiation shield for the control site during power plant operation (Figure 20).

The major structure in the test area is the test building, a steel framed structure approximately 64 feet long, 40 feet wide, and 45 feet high. The extraordinary height of the test building was provided to permit air circulation above the power conversion skid precooler during power plant operation and sufficient clearance to erect the shield tank extension during reactor refueling operation. A 40 hp vaneaxial type blower is installed in the test building wall to minimize cooling air recirculation through the precooler during operation. Each end of the test building is equipped with motor-operated sliding doors which are normally open during power plant operation. Support facilities at the test area include a contaminated waste handling system, personnel change areas, heating and ventilating equipment, and shield solution mixing and handling equipment.

The major structure in the control area is the auxiliary control building, a cement block building approximately 36-feet wide, 52-feet long, and 10-feet high. The instrumentation used to provide data for the analysis of power plant operation is housed in this structure along with the heating and ventilating equipment, emergency power equipment and the utility water system.

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ML-1 TEST FACILITY BIBLIOGRAPHY

1. AGN TM-349, "ML-1 Facility Design Criteria," 30 March 1959

The design criteria for the facility to be used for performance and developmental testing of the ML-1 power plant are presented.

F. ML-1 TEST OPERATION

The experimental test program for the ML-1 was divided into two phases. Phase I experiments were designed to provide data for the evaluation of core neutronics, shielding characteristics, reactor skid component operation, and support equipment operation. Reactor power levels during this phase were restricted to 6 kw or less. The Phase II test program was initiated in mid-1962 when the power conversion skid was delivered to the NRTS. The experiments in Phase II were designed to demonstrate the adequacy of the concepts incorporated in the design of the plant and to provide plant and component performance evaluation data.

All test operations were performed under the provisions of a comprehensive set of controls and procedures. The controls (limits on plant operation) were specified in the Technical Standards (ANTS). The detailed procedures for each operation (ANSOP) were prepared in advance of the test and were reviewed and approved by both AGN and USAEC-ID personnel prior to use.

The Phase I experimental program was initiated in February 1961 with the arrival of the ML-1 reactor skid at the NRTS. Extensive checkout, calibration, and low power tests were performed to confirm theoretical predictions of reactor neutronics and shielding characteristics, and to permit a thorough evaluation of the reactor controls and instrumentation and the operation of skid components and support equipment. During this phase approximately 150 modifications were made to the reactor skid components, the control cab, and the control and analysis instrumentation to correct noted deficiencies or to improve reliability and safety. The Phase I program was terminated in June 1962; the most significant experiments performed in this phase are summarized in Table 15.

The Phase II experimental program, designed to determine the performance characteristics of the ML-1 as an integrated power plant, was initiated in June 1962 with the arrival of the power conversion skid at the NRTS and the initial coupling of the reactor and power conversion skids. Following extensive checkout and instrumentation calibration, the initial ascension to power (Initial Self-Sustained Operation, ANSOP 16615) was performed in September 1962. Unusually high temperatures in the reactor tube sheet during this experiment necessitated the incorporation of a fuel element bypass flow modification which was completed in January 1963.

The second power ascension, together with a 100-hour endurance run, (ML-1 Full Power and Limited Endurance Test, ANSOP 16625) was performed in February-March 1963. During this test, the plant delivered a maximum of 252 kw(e) and 3.17 Mw(t) reactor power. Subsequent to the successful 100-hour run, a reactor pressure vessel leak developed in the seal weld in one of 18 lower plenum cap shear pins. The reactor was unfueled, transported to the GCRE facility, and disassembled for repair. While the reactor was being repaired, the CSN-1 turbine-compressor set was removed from the power conversion skid for a complete teardown and inspection. The inspection revealed excessive bearing wear and the set was shipped to the vendor's shop for evaluation and correction of the problem. The modified t-c set, redesigned CSN-1A, was shipped to the NRTS in February 1964. Prior to refueling the repaired reactor,

TABLE 15 - ML-1 PHASE I TEST PROGRAM SUMMARY

ANSOP ⁽¹⁾ No.	Title	Purpose	Results	Comments	Reference
16200	Initial Critical Experiment	Determine cold, clean critical loading for reactor	Critical attained with five rods out, shim 2 at 65° and 47 fuel elements loaded.	No shield water or upper cap	AN-IDOP-20
16208	Operating Core Loading Experiment	Determine number and position of reactivity shims to give 1.5% $\Delta K/K$ excess in fully-loaded cold, clean core.	61 fuel elements with 28 reactivity shims had excess of $\sim 1.5\%$ $\Delta K/K$.	1.5% $\Delta K/K$ cold clean is equivalent to $\sim 3.7\%$ $\Delta K/K$ hot clean. No shield water or upper cap.	AN-IDOP-29
16210	Control Rod Calibration	Determine reactivity worth of control rods.	Shim/safety rods: 1.50% $\Delta K/K$; regulating Rod: 0.44% $\Delta K/K$	Performed with 61 fuel elements and 28 reactivity shims; both rod drop and period techniques used. No shield water or upper cap.	AN-IDOP-34
16211	Shield water and tungsten baffle worth determination.	Determine reactivity worth of shield water and upper cap/baffle assembly.	Cap/baffle: -0.20% $\Delta K/K$ Shield water: -0.066% $\Delta K/K$.	Excess of cold clean core in operating configuration: 1.52% $\Delta K/K$ Shutdown margin: -4.73% $\Delta K/K$.	AN-IDOP-34
16212	Wet critical experiment	Demonstrate fully loaded core was subcritical with one safety rod fully withdrawn and coolant passages flooded with water.	Fully loaded flooded core critical with the two safety rods and regulating rod out, two shim rods in and third shim at 38.5°.		AN-IDOP-36
16214	Reactor drying experiment	Demonstrate feasibility of drying concept and establish required drying time.	Reactor dried satisfactorily in ~ 40 hr	Drying air at 575°F inlet temperature and ~ 235 cfm	AN-IDOP-47
16216	ML-1-10 power and flux mapping experiment	Determine power and flux distribution in ML-1 core	Distributions essentially identical with design predictions.	Used special replacement elements with fuel capsules and gold wire sensors.	AN-IDOP-47 AN-IDOP-37
16234	Moderator water temperature reactivity determination	Determine reactivity coefficient of moderator water temperature.	Coefficient was 4×10^{-5} $\Delta K/K$ -°F from moderator water temperature of 100°F to 180°F.	Coolant (gas) pressure coefficient was -6.2×10^{-6} $\Delta K/K$ -psi from 60 to 300 psi.	AN-IDOP-46
16252	Reactivity worth of ML-1 flooding	Determine reactivity charge associated with flooding of the reactor coolant passages.	Flooding worth +3.5% $\Delta K/K$		AN-IDOP-47
16240 & 16241	Reactor shielding evaluation	Evaluate ML-1 shield design at low power.	No design deficiencies observed.		IDO-28609
16262 & 16263	Core physics experiments	Develop data on partially loaded core with variety of reactivity shim patterns in dry and flooded configuration.	Experiment successfully completed.		AN-IDOP-47 AN-IDOP-49

(1) Aerojet-General Nucleonics Standard Operating Procedure; detailed pre-approved procedures for the conduct of tests and other operations.

the fuel elements were thoroughly inspected and failure of five lower spiders was noted. The spider failure was determined to be the result of substandard material and unfavorable tolerance buildup during manufacture. Replacement of all fuel element lower spiders with an improved design was completed in January 1964. The reactor was refueled and the power plant reassembled in February 1964.

Checkout and calibration operations were completed in early April and the third power test (ML-1 Full Power and Limited Endurance Test, ANSOP 16625A) was performed during April-May 1964. During this test, the t-c set was operated for 828 hours, including 664.5 hours at self-sustained conditions, and the maximum shaft power output was 130 kw. The reactor power level was limited by high fuel element temperatures. Fuel element inspection following the test failed to reveal any abnormalities, however, additional instrumentation was installed to verify the indicated high temperatures. Inspection of the CSN-1A t-c set after this test revealed that the turbine nozzle spacer shim had moved axially and contacted the first-stage rotor blades. The rotor assembly and turbine inlet lines were repaired by the vendor and the CSN-1A was reassembled in July 1964.

The power plant was reassembled and checkout and calibration operations completed in August 1964. The ML-1 Thermodynamic Performance Test (ANSOP 16650) was initiated on 31 August and terminated on 30 November 1964. During this test the CSN-1A was rotated for 1877 hours including 1775 hours at self-sustained conditions. The reactor accumulated 4064 Mwh. The fuel element instrumentation confirmed that the central ring of six elements operated at the highest temperature in the core due to a non-optimum flow distribution orifice pattern.

Routine maintenance was completed in early December 1964 and the Power Plant Endurance Test (ANSOP 16635) and Sustained Full Power and Preliminary Afterheat Evaluation (ANSOP 16656) were performed during December 1964 and January 1965. During these tests the t-c set was operated for 682 hours, including 630 hours at self-sustained conditions. The reactor accumulated 1719 Mwh. Subsequent to the shutdown on 23 January, a leak was discovered in the lower region of the reactor pressure vessel. The reactor was uncoupled from the power conversion skid and transported, in the fully-fueled condition, to the GCRF for repair of the leak.

The reactor was operated in the GCRF (Section V-G-3) until early September 1965, when the ML-1 test program was terminated by a leak in the pressure vessel. The nature and location of this leak had not been determined by the time of the AGCRSP termination on 1 November 1965.* The total exposure of the ML-1-I core was 7675 Mwh.

The Phase II program included the following test series: 1) Pre-Operational Tests; 2) Initial Power Tests; 3) Steady State Performance and Endurance Test; and 4) Transient Tests. The test program is summarized in Table 16; a summary chronology of test operations is shown in Table 17.

*. The disassembly and inspection of the pressure vessel is continuing under USAEC cognizance with another contractor; it is understood that the findings of this work will be published under document No. IDO-17159.

TABLE 16 - ML-1 PHASE II TEST PROGRAM SUMMARY

<u>ANSOP⁽¹⁾ No.</u>	<u>Title</u>	<u>Purpose</u>	<u>Results</u>	<u>Comments</u>	<u>Reference</u>
	Pre-operational Tests	Checkout power plant instrumentation and controls.	Satisfactorily completed, with required modifications in 90 days.	Included following: ANSOP 16290-Package Coupling ANSOP 16405-Precooler Checkout ANSOP 16425-Speed Control System Checkout ANSOP 16423-Electrical System Checkout ANSOP 16500-Non-Nuclear Instrumentation Checkout ANSOP 16475-Gas Analysis System Checkout ANSOP 16450-Gas Supply System Checkout ANSOP 16605-Initial Pressurization ANSOP 16415-Initial Lubrication System Checkout ANSOP 16602-Cold Rotation & Instrumentation Compatibility Test ANSOP 16451-Emergency Cooling System Checkout Ultimate Shutdown System Checkout	AN-AGCR-90 AN-AGCR-303 AN-AGCR-305 AN-AGCR-321 AN-AGCR-408 AN-AGCR-304 AN-AGCR-308 AN-AGCR-310 AN-AGCR-316 AN-AGCR-307 AN-AGCR-306
16615	Initial Power Tests	Demonstrate plant design concept, provide data for gross evaluation of plant performance and demonstrate reactor stability.	Maximum output power 50kw at 1.6 Mw reactor power.	Test terminated by excessive reactor calandria tube sheet temperatures.	AN-AGCR-457
16625 ⁽²⁾	Full Power Test	Determine maximum output power within component design limits.	252 kw at reactor power of 2.94 Mw and 95 psia compressor inlet pressure. 195 kw at reactor power of 3.2 Mw and 120 psia compressor inlet pressure.	Test included 100 hr run at 2.2 Mw reactor power. Reactor stability evaluated with Milletron transfer function computer; stability excellent.	AGN TM-402 AGN TM-403
16625A ⁽²⁾	Full Power & Limited Endurance Test	Determine maximum output power, define t-c set operating limits, evaluate speed control system and evaluate shielding.	823 hrs of t-c set operation with 665 hrs of self-sustaining condition. Maximum reactor power: 3.03 Mw; maximum output power: 130 kw; evaluation data obtained.	Pressure vessel leak repaired and fuel elements modified prior to test. New bearings installed in t-c set.	IDO-28634
16650 ⁽²⁾	Thermodynamic Performance Test	Obtain detailed t-c set performance data. Determine loop state points. Accumulate operating time and experience.	More than 1000 data points collected. Maximum reactor power: 2.97 Mw; maximum output power: 101 kw. 1877 hrs of t-c set operation with 1775 hrs of self-sustaining conditions. 4064 Mwh accumulated on reactor.	T-C set turbine first stage rotor repaired prior to test.	AGN TM-410
16656 ⁽³⁾	Preliminary Afterheat Evaluation	Determine fuel cladding temperatures following scram from full power with normal and emergency post-scram cooling.	Maximum fuel element cladding temperature was 1750°F minutes after scram from 3.1 Mw (7460 Mwh on core) with normal t-c set coast-down (75 sec).	Leak at pressure vessel lower plenum seal weld repaired and reactor skid relocated to GCRF at completion of this test.	AN-AGCR-878

(1) Aerojet-General Nucleonics Standard Operating Procedure

(2) Steady state performance and endurance tests.

(3) Transient test.

TABLE 17 - ML-1 TEST PROGRAM CHRONOLOGY

<u>Operation</u>	<u>Date</u>	<u>Comment</u>
Reactor skid checkout	Feb-Mar 1961	Checkout of reactor skid components and controls prior to initial critical.
Initial critical and low power testing	Apr 61-June 62	Extensive low power operation to verify theoretical predictions in neutronics and shielding. Evaluation of reactor control and instrumentation.
Power plant assembly and checkout	June-Sept 1962	Initial reactor and power conversion skid coupling. Start motor installation and control and analysis instrumentation installation. Checkout and compatibility test of all required support systems. Calibration of all control and analysis instrumentation.
Initial self-sustained operation (ANSOP 16615)	Sept. 1962	Initial power ascension using ML-1 reactor as heat source. Power limited by high tube sheet temperature.
Fuel element bypass flow fix	Sept 62-Jan 63	Installation of silver plated "O" rings under each fuel element and spring loaded holddown devices to eliminate bypass flow which caused high tube sheet temperatures.
ML-1 full power and limited endurance test (ANSOP 16625)	Feb-Mar 1963	Second power ascension including 100-hour endurance run.
Pressure vessel repair, t-c set modification and fuel element modification.	Mar-Feb 1964	Reactor unfueled, transported to GCRE and disassembled for repair of shear pin leak, t-c set bearing changes and aerodynamic modifications completed, modified lower spiders installed in all fuel elements.
Power plant assembly and checkout	Feb-Mar 1964	Power plant reassembled with CSN-1A. Checkout of all systems. Calibration of all control and analysis instrumentation.
ML-1 full power and limited endurance (ANSOP 16625A)	Apr-May 1964	Third power ascension included 828 hours of t-c set rotation. Reactor power limited by high fuel element temperatures.
CSN-1A inspection and instrumented fuel element installation.	June-July 1964	T-C set rotor and turbine inlet duct repaired. Additional instrumented elements installed to verify high temperatures noted during ANSOP 16625A.
Power plant checkout	July-Aug 1964	Reassembly of power plant and checkout of all systems. Calibration of all control and analysis instrumentation.
ML-1 thermodynamic performance test (ANSOP 16650)	Sept-Nov 1964	Fuel element instrumentation confirmed central ring of elements had non-optimum orifices. Reactor accumulated 4064 Mwh. T-C set rotated 1877 hours. Shutdown for routine maintenance.
Power plant endurance test (ANSOP 16635) and preliminary afterheat evaluation (ANSOP 16656)	Dec 1964-Jan 1965	Included three scrams, with t-c set coastdown to stop, from high reactor power. Reactor accumulated 1719 Mwh. T-C set rotated 682 hours. Pressure vessel leak discovered subsequent to shutdown.
Pressure vessel repair	Feb-Apr 1965	Installed an omega-type seal between lower plenum cap and lower plenum to repair leak in trepan type seal which was the result of thermal cycling.
ML-1 reactor skid checkout	May-June 1965	Reactor skid installed in modified GCRF and checked out. Calibration of all control and analysis instrumentation.
ML-1/GCRF operation	July-Sept. 1965	ML-1 reactor operated in GCRF until September 1965 when program was terminated by a leak in the pressure vessel.

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ML-1 TEST OPERATION BIBLIOGRAPHY

1. AN-IDOP-20, "Initial Critical Experiment for ML-1," 31 March 1961

The conduct of, experimental data generated during and results of the ML-1 dry critical experiment are discussed.

2. AN-IDOP-29, "Operating Core Determination for ML-1," 9 May 1961

The conduct of the experiment to define the reactivity shim pattern in the 61-element ML-1 core is discussed and the results of the experiment are presented.

3. AN-IDOP-34, "Final Control Rod Calibration, Shield Water and Tungsten Baffle Reactivity Worths for ML-1," 11 May 1961

The conduct and results of the experiments performed to calibrate the ML-1 control rods and to determine the reactivity worth of the shield water and the upper cap/baffle assembly are discussed.

4. AN-IDOP-36, "Wet Critical Experiment for the ML-1," 13 June 1961

The performance and results of the critical experiment conducted with the ML-1 reactor in the flooded condition are reported.

- 5a. AN-IDOP-47, "Summary of ML-1 Phase I Experimental Test Program," 22 February 1962

- 5b. AN-AGCR-86, "Test Addendum I" to above, 3 July 1962

All Phase I experiments are discussed and the results summarized.

6. AN-IDOP-37, "Proof Test for Gold Wire Data, ML-1-10 Power Mapping Experiment," 14 August 1961

The data collected from counting of the gold wire detectors irradiated in the ML-1-10 power mapping experiment are analyzed to verify the validity of the conclusions.

7. AN-IDOP-46, "Moderator Water and Gas Pressure Reactivity Coefficients for ML-1," 11 January 1962

The results of the experiments performed to determine the ML-1 moderator water temperature coefficient of reactivity are summarized.

8. IDO-28609*, "ML-1 Shielding Design Report," October 1963

The analytical and experimental program contributing to the design of the ML-1 radiological shields is discussed along with a summary of the evaluation of the performance of the shield based on data generated during ML-1 test operation.

9. AGN-IDOP-49, "Flooded Critical and Associated Core Physics Experiment," 5 March 1962

Describes the conduct and results of a series of zero power experiments in the ML-1 reactor to explore the neutronic characteristics, especially in the flooded condition, of a variety of partial core loadings and reactivity shim patterns.

10. AN-AGCR-90, "Initial Precooler Checkout and Moderator/Lube Oil Cooler Checkout," 14 August 1962

Reports the results of operability tests of the precooler and moderator/lube oil cooler air cooling system. The only malfunction observed was in the louver position indicating circuit; this was corrected.

11. AN-AGCR-303, "Speed Control System Checkout," 17 August 1962

Reports the results of operability tests of the overspeed bypass and speed control valves and associated control circuitry. The overspeed system functioned satisfactorily but the speed sensing system for the speed control valve did not function properly. An alternative speed sensor was installed and the system checked out satisfactorily.

12. AN-AGCR-304, "Gas Supply System Checkout," 15 August 1962

The gas makeup control system was checked and, after some modification, operated satisfactorily.

13. AN-AGCR-305, "Electrical System Initial Checkout," 22 August 1962

Reports the results of the initial compatibility and functional tests of the electrical system. Several changes were required in the wiring between the p/c skid and control cab; these were completed and the system approved for operation.

14. AN-AGCR-306, "ML-1 Ultimate Shutdown Device Checkout," 5 September 1962

Operational tests of this system revealed that 10.75 gal of fluid were injected in a 9 minute and 20 second period during which the storage pressure dropped from 120 to 37 psi; this performance was essentially identical to design predictions.

15. AN-AGCR-307, "ML-1 Non-Prototype Emergency Cooling System Checkout," 3 September 1962

Reports the results of functional checks of the emergency cooling system under both automatic and manual control. The system rejected 3.1 to 1.3 lb/sec of gas over a 4 minute period and the gas flow was in the correct (through the reactor) direction.

16. AN-AGCR-308, "Initial Pressurization of the ML-1 System," 17 September 1962

Reports the results of the initial pressurization of the power plant reactor coolant/power conversion system working fluid system. Pressure was increased from 20 to 130 psia in 10 psi increments. No problems were encountered; at 117 psia the leak rate was 3.9 lb/hr.

17. AN-AGCR-310, "Initial Lube System Checkout," 16 October 1962

Reports the results of functional and leak tests of the rotating equipment lubrication system. Only minor, easily correctible problems were encountered.

18. AN-AGCR-316, "Initial Cold Rotation and Instrumentation Compatibility Tests," 23 August 1962

Reports results of first rotation of t-c set with the starting motor. The unit was rotated at quarter-speed for 81 minutes and at half-speed for 84 minutes. All support systems operated normally; starting motor power requirements were measured as a function of compressor inlet pressure.

19. AN-AGCR-321, "Initial Non-Nuclear Instrumentation Checkout," 6 November 1962

Reports the results of the compatibility tests and calibration of the non-nuclear power plant instrumentation.

20. AN-AGCR-408, "Evaluation of ML-1 Hydrocarbon Analysis System," 15 December 1962

Reports the results of tests which demonstrated that the Beckman 109 flame ionization hydrocarbon analyzer was superior to the Beckman Model 21 infrared analyzer although the former required shielding to prevent distortion of the output signals by the reactor radiation.

21. AN-AGCR-457, "Initial Self-Sustaining Operation of the ML-1 Nuclear Power Plant," February 1963

Reports the results of the initial operation of the ML-1 plant which demonstrated the feasibility of the closed-cycle gas-cooled nuclear power system. The maximum measured output power was approximately 50 kw at the reactor power of 1.6 Mw. The power output was limited by excessive tube sheet temperature.

22. AGN TM-402, "Final Test Report Initial Full Power and Limited Endurance Tests of the ML-1 Nuclear Power Plant," August 1963

Reports the results of the initial full power test of the ML-1 power plant. The plant was operated for 101 hours at self-sustained conditions. The maximum power output measured was 247 kw and the maximum reactor power attained was 3.44 Mw. The tube sheet heating problem observed in the earlier test was not apparent during this operation.

23. AGN TM-403*, "ML-1 Reactor Stability Tests," June 1963

Reports the results of the digital computer (Milletron) analysis of noise levels in the reactor flux records during operation of the ML-1. The analysis revealed no indications of unstable tendencies.

24. IDO-28634*, "Full Power and Limited Endurance Test of the ML-1 Nuclear Power Plant," September 1964

Reports the results of sustained full power test of the ML-1 during which 1493 Mwh were accumulated on the reactor and 823 hours of t-c set operation were accumulated. The maximum measured output power was 139 kw. The reactor and supporting systems operated satisfactorily but the performance of the t-c set was poorer than predicted.

25. AGN TM-410, "ML-1 Thermodynamic Performance Test," May 1965

Reports the results of extended operation (4064 Mwh and 1877 hours of t-c set operation) during which more than 1000 data points were collected. Maximum output power was 100 kw(e). Reactor and shielding operation were as predicted but the performance of the t-c set was poorer than anticipated.

26. AN-AGCR-878, "Preliminary Test Report - ML-1 Nuclear Power Plant Tests," 14 May 1965

Reports the results of extended testing (5057 Mwh with 2561 hours of t-c set operation) of the power plant during which a variety of experiments (afterheat determinations, effect of variations in shielding solution concentrations, etc.) were conducted. The maximum power output measured was 75 kw at a reactor power of 3.1 Mw.

G. ML-1 SKID OPTIMIZATION PROGRAM

1. Background

The test operation of the ML-1 power plant discussed above had, by June 1964, resulted in the following conclusions:

- The basic design principles and concepts incorporated in the ML-1 had been satisfactorily demonstrated.
- Further evaluation of power conversion components (especially the t-c set) in the compact skid configuration was not the optimum approach to completion of the development of this system.
- The remaining developmental and testing objectives for the reactor skid could be accomplished adequately, and probably more expeditiously, by operation of the skid with the GCRF test loop.

As a consequence, it was decided to discontinue power plant testing in January 1965 in favor of separate skid testing. The reactor skid and control cab were relocated to the GCRF for further testing. In the meantime, a program of providing and demonstrating (by test operation) an improved set of power conversion system components was undertaken.

2. GCRF Modification

A four-phase program for modification of the GCRF was proposed which would not only provide the capability for testing the ML-1 reactor skid at design conditions but also for testing advanced pressure vessels and gas-cooled reactors at more severe conditions (up to 1400°F and 500 psia).⁽¹⁾⁽²⁾ The first step of this program was accomplished; this permitted the testing of the ML-1 reactor skid at GCRF design conditions (1200°F and 220 psia).

The design work was initiated in the fall of 1963⁽³⁾⁽⁴⁾ and completed early in 1964 when Aerojet delivered design drawings and specifications⁽⁵⁾⁽⁶⁾ to the USAEC-ID. Construction work to implement the design was begun under the cognizance of USAEC-ID in April 1964 and completed a year later. The modification work consisted, essentially, of conversion of the GCRF pool to a dry pit facility with top shielding and the modification of the control room to accommodate the ML-1 control cab. The modified facility was reactivated and ready for operation near the end of April 1965.⁽⁷⁾

3. ML-1/GCRF Operation

The fully-fueled ML-1 reactor skid and the ML-1 control cab were transferred from the ML-1 test site to the GCRF⁽⁸⁾⁽⁹⁾ to prepare for reactor skid testing. A leak between the reactor coolant and shield water systems occurred when the seal weld between the lower plenum and the lower plenum cap failed from fatigue associated with thermal cycling. The leak was repaired by installing a modified omega seal over the defective trepanned seal weld. Following a series of pre-startup checks, the ML-1 reactor was operated

TABLE 18 - ML-1 SKID OPTIMIZATION TEST PROGRAM SUMMARY

ANSOP ⁽¹⁾ No.	Title	Purpose	Results	Comments	Reference
16295	ML-1 Reactor Skid Relocation	To transfer fully-fueled reactor skid from ML-1 test site to GCRF and, concurrently, perform a time study and accumulate radiation dose data.	Transfer successfully completed. Reactor skid can be uncoupled and loaded by 6 men in 1.5 hr; each man receives ~1000 mrem. Dose rate at truck cab with reactor only loaded ~9 mrem/hr. Full power plant loading by 6 men in 1.5 hr with ~150 mrem exposure. Dose rate at truck cab with full power plant loaded ~ 5 mrem/hr.		AN-AGCR-866
5900	Pre-Startup Check	Perform required test loop and reactor skid pre-startup checks.	All checks completed satisfactorily.		
15238	Reactivity Coefficient Measurements.	To remeasure major reactivity coefficients using in-line analog computer.	Moderator temperature coefficient at 70°F = +113 $\times 10^{-6}$ $\Delta K/K$; at 150°F = +41 $\times 10^{-6}$ $\Delta K/K$ Coolant pressure coefficient = -5.2 $\times 10^{-6}$ $\Delta K/K$ -psi from 40 to 250 psi at constant temperature. Coolant temperature coefficient + 0.37 $\times 10^{-6}$ $\Delta K/K$ -°F from 600 to 750°F at constant pressure.	In-line analog computer performed exceptionally well.	AN-AGCR-926
15302	Control Rod Calibration	To recalibrate ML-1 control rods using in-line analog computer.	Shim rod worth = 1.54% $\Delta K/K$ Regulating rod worth with shims 1 & 3 @ 60° = 0.46% $\Delta K/K$; with shims at 40° = 0.89% $\Delta K/K$. Cold shutdown margin = 5.4% $\Delta K/K$. Cold excess = 2.20% $\Delta K/K$		AN-AGCR-925
5091	Plant Startup	To perform initial startup of ML-1 reactor skid in GCRF.	Experiment successfully completed.		
15200	Initial Power Operation	To permit operational evaluation of startup, normal operation and shutdown procedures.	Operated 215 Mwh, made radiation surveys.	Operation terminated by malfunctions in emergency diesel-generator and subsequent detection of pressure vessel leak.	
16231	Silver Deposition in ML-1 Moderator System	To determine the amount of silver removed from the control rods.	Silver loss to date = 0.7% of original amount.		AN-AGCR-874

for a total of 215 Mwh at power levels up to 2 Mw. The results of this test program are summarized in Table 18.⁽¹⁰⁾⁽¹¹⁾⁽¹²⁾

During preparations for plant startup early in August, evidence of a leak between the reactor coolant and moderator system was observed. A program of diagnostic investigations⁽¹³⁾ was undertaken which led to the conclusion that a leak existed and that partial disassembly of the reactor skid and a destructive examination of the pressure vessel would be required to determine the exact nature and location of the leak.⁽¹⁴⁾⁽¹⁵⁾ At the time of termination of AGCRSP activities on 31 October, all preparations were complete for the removal of the pressure vessel from the reactor skid and subsequent inspection of the pressure vessel.⁽¹⁶⁾⁽¹⁷⁾ Cognizance for the completion of the disassembly and inspection program was assigned by USAEC-ID to PPCo. This work began early in February 1966; AGN provided a consultant and observer during the activity.

4. P/C Component Development

a. Turbine-Compressor Set

The results of ML-1 test operation up to the fall of 1964 had clearly indicated that, while the CSN-1A t-c set appeared to represent a sound mechanical design, a major aerothermodynamic modification would be required before the machine would deliver design power. Consequently, a program for modification of the unit was developed⁽¹⁸⁾ and proposals for the work were solicited from a number of qualified vendors. The evaluation of the proposals⁽¹⁹⁾ resulted in the award to the General Electric Co. (Evendale, Ohio) of a subcontract for the modification. Work began in April 1965 and, at the time of termination of the AGCRSP activity, the aerothermodynamic redesign had been completed⁽²⁰⁾ and the mechanical design of the unit and of the test facility were in progress.⁽²¹⁾ Basically, the modification contemplated: (1) alterations in the design of the compressor inlet guide vanes, the addition of one row of guide vanes in the compressor outlet and the conduct of model flow tests to verify the adequacy of these designs; (2) the modification of the configuration of the turbine inlet and diffuser and the conduct of model flow tests to check these designs; (3) the complete redesign of the turbine rotors and stator blading; and (4) the assembly of a test facility for evaluation of the modified unit and the conduct of an evaluation test program.

b. Precooler

The procurement of a replacement precooler for the ML-1 was initiated in 1964 under the ML-1 improvements program. A performance specification⁽²²⁾ was prepared which served as the basis for requests for proposals from qualified vendors. This specification, in addition to establishing size, weight, and coolant pressure and temperature conditions which were compatible with the ML-1 power plant, specified an effectiveness of 94%. After an evaluation of the proposals, a fixed-price contract was awarded to Stewart-Warner for the design⁽²³⁾ and fabrication of the precooler core. The concept of the Stewart-Warner heat exchanger was changed (Griscom-Russell had used a finned-tube concept; the improved precooler incorporated a plate-fin concept) to eliminate the leakage experienced in the original unit as well as to improve

performance. AGN undertook the design⁽²⁴⁾ and fabrication of the cooling air assembly. Metallurgical investigations were performed to augment the data presented in the literature on the high-temperature, long-term creep properties of the aluminum alloys proposed by Stewart-Warner for use in the pre-cooler.⁽²⁵⁾

A special test facility was constructed for performance evaluation tests of the precooler assembly⁽²⁶⁾ and testing of the unit was initiated. Early test data indicated that the performance of the cooling air assembly exceeded specifications and that the precooler properly performed at least as well as specified.⁽²⁷⁾ Testing was interrupted by termination of the AGCRS Program activity before the data necessary to evaluate fully the precooler performance could be compiled.

c. Recuperator

An evaluation of the performance of the Griscom-Russell recuperator installed in the ML-1 was difficult because of the compact configuration which resulted in flow stratification and significant temperature gradients at the sensor location. In the fall of 1964, however, after special instrumentation had been installed, sufficient data were obtained to indicate that the effectiveness of the unit was far less (70% as compared with a design specification of 79%) than that required to deliver rated plant power output without exceeding maximum allowable fuel element temperatures. A study⁽²⁸⁾ was performed which defined the characteristics of a high-performance (83% effective) compact recuperator and served as the basis for the solicitation of proposals from qualified vendors. Although acceptable, fixed-price proposals were received, the procurement of the improved recuperator had not been initiated by the time of termination of AGCRSP activity.

d. Alternator

As a result of investigations completed during the ML-1A preliminary design (Section VII), it appeared that the procurement of a commercial alternator to ML-1 specifications was feasible and the development of a brushless alternator was discontinued. A study was performed to define the performance requirements of such a unit⁽²⁹⁾ and, after reviewing proposals from qualified vendors, a subcontract was awarded to Fairbanks-Morse for the design and fabrication work.⁽³⁰⁾ The machine was delivered and a test facility was designed and fabricated⁽³¹⁾ for the detailed evaluation of the alternator performance but the AGCRSP was terminated before testing could begin.

e. Lubrication System

An improved lubrication system for the t-c set was designed, fabricated and partially tested⁽³²⁾ by the time of termination of the AGCRSP activity.

ML-1 SKID OPTIMIZATION PROGRAM BIBLIOGRAPHY

1. AN-AGCR-779, "Preliminary Analysis GCRE Modification Program," November 1964

Reviews the anticipated requirements for the use of GCRF in support of AGCRSP activities and outlines a plan for accomplishment of a four-step modification program to satisfy these requirements.

2. AN-AGCR-915, "GCRF High Performance Modification - Conceptual Design Report," 22 July 1965

Presents the conceptual design of a single step modification of the GCRF to provide the capability for testing at a reactor outlet temperature up to 1400⁰F and coolant system pressures up to 500 psia.

3. SR-63-15, "Title II Design of the Modification of the GCRE Facility," 27 November 1963

Presents the proposed scope of work and schedule for the accomplishment of the Title II design work.

4. AN-AGCR-629, "Shielding Calculations in Support of the GCRE Modification," 17 December 1963

Presents the results of an evaluation of the predicted performance of the shielding design for the GCRF.

5. Technical Specifications for the Modification of the GCRE Facility for ML-1 Reactor Skid Testing, Contract AT(10-1)-880, 6 March 1964

Presents the technical specifications for the subject modification.

6. AN-AGCR-772, "Operation of ML-1 Reactor Skid in GCRE: Safety Evaluations Report," October 1964

Presents an evaluation of the safety considerations associated with the operation of the ML-1 reactor skid in the modified GCRF. It is concluded that the consequences of the maximum credible accident are less severe than for the previously approved operation of the ML-1 reactor at the ML-1 test site or of the GCRE-I reactor in the GCRF.

7. AN-AGCR-684, "GCRE Activation," 13 April 1964

Summarizes the program of work to be undertaken in connection with reactivation of the GCRF.

8. AN-AGCR-838, "Safety Considerations in the Relocation of the ML-1 Reactor to the GCRF Pit," 22 March 1965

Reviews the hazards associated with the relocation of the ML-1 reactor skid to the GCRF pit and makes recommendations concerning this operation.

9. AN-AGCR-866, "ML-1 Reactor Relocation," 6 April 1965

Summarizes the results of the time studies and radiation level measurements made during the relocation of the ML-1 reactor skid from the ML-1 test site to the GCRF.

10. AN-AGCR-926, "ML-1 Reactor Preliminary Test Report Reactivity Coefficient Measure Experiment 15238," 6 August 1965

Presents the results of reactivity coefficient determinations made with the ML-1 reactor and GCRF using the in-line analog computer techniques.

11. AN-AGCR-925, "ML-1 Reactor Preliminary Test Report Reactivity Measuring Experiment - Control Rod Calibration Experiment 15302," 30 July 1965

Reports the results of experiments performed to calibrate the ML-1 control rods using an analog computer connected to neutron detector for direct evaluation of reactivity changes as well as the conventional rod drop and period techniques.

12. AN-AGCR-874, "Silver-Indium-Cadmium Control Rod Corrosion," 26 April 1965

Presents the results of an experiment performed with the ML-1 to assess the rate of silver disappearance from the control rod neutron absorbers. It is concluded that, after five years of operation, the loss of silver is so small that no effect on reactivity can be observed.

13. AN-AGCR-934, "Pressure Vessel Disassembly and Inspection Program," 14 September 1965

Summarizes the results and findings of the program of diagnostic investigation performed on the ML-1 reactor after defueling.

14. AN-AGCR-943, "Procedures for Partial ML-1 Reactor Skid Disassembly," 1 October 1965

Outlines the work scope required for the partial disassembly of the ML-1 reactor skid to permit removal of the pressure vessel for inspection.

15. AN-AGCR-949, "Destructive Disassembly of ML-1 Pressure Vessel," 18 October 1965

Outlines the program of destructive disassembly of the ML-1 pressure vessel in the GCRF pool and the associated inspection steps which will be performed to identify and characterize the pressure vessel leak.

16. AN-AGCR-951, "Procedures for Removal of the ML-1 Reflectors and Control Blades - ML-1 Disassembly-Inspection Program," 27 October 1965

Presents the detailed procedures for the removal of the reflectors and control blades from the ML-1 pressure vessel after transfer of this assembly to the GCRF pool.

17. AN-AGCR-960, "Interim Report - ML-1 Reactor Disassembly-Inspection Program," December 1965

Summarizes the status of work under the ML-1 reactor disassembly and inspection program as of the time of termination of the AGCRSP activities.

18. AN-AGCR-853, "CSN-1A Modification Program," 3 December 1964

Defines the scope of the modification program for the CSN-1A t-c set to improve the aerothermodynamic performance and power output of the machine.

19. AN-AGCR-852, "CSN-1A Modification Proposals, Evaluation and Recommendation," 2 April 1965

Summarizes the results of the vendor evaluation which led to the selection of the General Electric Company as the subcontractor for the CSN-1A modification.

20. AN-AGCR-912, "Trip Report, CSN-1A(M) Aerothermodynamic Design Review at G.E., 30 June 1965," 8 July 1965

Summarizes the results of the AGN review of the aerothermodynamic redesign of the CSN-1A t-c set as developed by G.E.

21. AN-AGCR-963, "CSN-1A Turbine-Compressor Set Modification (Interim Report)," 15 December 1965

Summarizes the status of the CSN-1A modification program as of the time of termination of AGCRSP activities.

22. Specification AGC-60252, Supplement 1A, "Heat Exchanger, Gas-to-Gas (Precooler ML-1A)," 20 May 1964

23. Precooler Design

- a. "Preliminary Design Report, Model 8453-A Precooler," South Wind Division, Stewart-Warner Corp., 17 July 1964
- b. AN-AGCR-743, "Improved ML-1 Precooler Preliminary Design Evaluation," 7 August 1964
- c. AN-AGCR-753, "Trip Report: Stewart-Warner, Indianapolis, Ind., August 17-18; Fairbanks-Morse, Beloit, Wis., August 19," 26 August 1964
- d. "Addendum A - Preliminary Design Report, Model 8453-A Precooler," South Wind Division, Stewart-Warner Corp., 18 December 1964
- e. AN-AGCR-782, "Addendum A, Preliminary Design Report, ML-1 Improved Precooler," 1 December 1964
- f. "Final Design Report, Model 8453-A Precooler," South Wind Division, Stewart-Warner Corp., 15 March 1965

These documents describe the evolution of the design of the improved precooler and the results of the evaluations of the design performed by AGN

24. AN-AGCR-745, "Improved ML-1 Precooler Air Side Cooling Assembly - Final Design Report," August 1964

Presents the details of the design and predicted performance of the cooling air assembly for the improved precooler.

25. AN-AGCR-965, "Creep Properties of Aluminum Alloys," January 1966

Presents the results of laboratory tests of the aluminum alloys used in the improved precooler and the extrapolation of the experimental data to predict the performance of these materials at high temperatures after 50,000 hr.

26. AN-AGCR-938, "ML-1 Precooler Test Loop Design Report," September 1965

Describes the design of the test facility and equipment fabricated for the performance evaluation test of the improved ML-1 precooler assembly.

27. AN-AGCR-966, "Final ML-1 Improved Precooler Test Report," January 1966

Reports the results of the performance tests of the cooling air assembly which indicates that the design specifications are exceeded by significant amount. Also reports the preliminary results of the thermodynamic evaluation of the precooler and the effect on precooler performance of the installation of p/c skid components in the compact configuration.

28. AN-AGCR-835, "Advanced Recuperator Study," March 1965

Reports the results of investigations of advanced compact recuperators for use in a Brayton cycle nuclear power plant including the results of analysis to define the deficiencies in the Griscom-Russell recuperator and the effect of these deficiencies on plant operation, the development of design criteria for the advanced recuperator and the results of contacts with potential vendors to evaluate the feasibility of the design and the cost and schedule for procurement of advanced components.

29. AN-AGCR-668, "Selection of ML-1A Alternator Size," 12 March 1964

Reports the results of the study which defined the 600 kw capacity alternator as that required for the ML-1 power plant.

30. AN-AGCR-700, "Recommendations for Purchase of ML-1 Alternator," 26 May 1964

Summarizes the results of the review of vendor proposals and recommends the award of the alternator subcontract to Fairbanks-Morse.

31. AN-AGCR-933, "Design Report, Alternator Test Facility," September 1965

Presents the detailed design of the alternator test facility fabricated for the performance evaluation of the Fairbanks-Morse alternator.

32. AN-AGCR-964, "CSN Lubrication System Design Report," 17 December 1965

Summarizes the design of the improved lubrication system as well as the component tests which had been performed at the time of the termination of AGCRSP activities.

VI. ML-1 TECHNOLOGY PROGRAM

A. CONCEPT AND MISSION

During the development program for the ML-1 power plant, many areas of basic technology were uncovered which were not investigated beyond the point of demonstrating satisfaction of specific program objectives. In addition, the performance characteristics of many of the developed components were not defined. Components were generally tested to demonstrate satisfaction of the requirements for application to the ML-1 only and no attempt was made to assess the ultimate capability of the device. As a consequence, the body of data describing the ML-1 technology was limited and contained many unexplored or not completely understood areas.

A relatively modest program directed toward a more thorough exploration of the ML-1 technology was undertaken in the spring of 1964. The basic objective was the precise definition of the state-of-the-art of the ML-1 technology to permit the use of this technology in other applications.

The areas which were investigated prior to the termination of the program in the fall of 1965 are discussed below.

B. PRESSURE VESSEL TECHNOLOGY

The objectives of the pressure vessel (calandria, plenums and gas ducts) technology program were limited to a refinement of the analytical techniques employed in the design of ML-1-type pressure vessels (to permit the definition of the probable performance limits of the ML-1 pressure vessel as well as to provide a firm basis for the design of higher performance assemblies) and the evaluation of high performance materials for potential use in an advanced pressure vessel. Work began in November 1964 and in December of that year the USAEC directed that the activity be brought to an orderly conclusion. During the limited lifetime of this program, a photoelastic experimental program was completed which resulted in the conclusion that the values recommended by Langer and O'Donnell⁽¹⁾ for the effective elastic constants employed in the analysis of ligament stresses represent a good approximation of the bending constants but result in values of the tensile elastic modulus which is about 40% too high.⁽²⁾ Several modifications were made to the TSA computer code for stress calculations which improved the precision and flexi-

bility of the code.⁽³⁾ No significant activity was undertaken in the area of evaluation of high performance materials for use in ML-1 type pressure vessels.

The conceptual design of an advanced pressure vessel correcting the known deficiencies in the ML-1 design at that time was completed⁽⁴⁾; this design did not knowingly include corrective measures to prevent a leak of the type existing in the pressure vessel at the time of program termination since the nature and location of that leak had not been determined.

C. HASTELLOY X TECHNOLOGY

A program was initiated to evaluate four aspects of the performance of Hastelloy X as a cladding material for fuel in an ML-1 type reactor:

- Corrosion in Air - A comprehensive program of laboratory exposure of sheet and tubing specimens was undertaken. Samples were exposed at 1900 and 2000°F for periods up to 7200 hr (of a planned 10,000 hr test program) and subjected to detailed physical properties and metallurgical evaluation. The results of this activity indicate that Hastelloy X is suitable for use at 1900°F but that the physical properties of the material do not warrant consideration for use at 2000°F.⁽⁵⁾ A limited amount of information was developed which indicates the possibility of altering the alloy composition to improve properties at 2000°F.
- Oxidation of Hastelloy X - A series of experiments were performed with a continuous weighing device to explore the extent and mechanism of oxidation of Hastelloy X in air at 1750 and 1850°F. Significant differences were found between the performance of low and commercial cobalt content alloys and the data indicated that the oxidation has an exponential temperature dependency and cubic time dependency.⁽⁶⁾
- Hastelloy X Creep Properties - Preparations for the centrifugal creep testing of specimens of Hastelloy X were completed and samples were tested at 1800°F for 1000 hrs and at 1900°F for 700 hrs. This program was terminated before any evaluation of the test data could be undertaken.⁽⁷⁾
- Hastelloy X Creep Rupture Strength - Samples were prepared for short term tests which would yield data from which the evaluation of 10,000 hr creep rupture strength of Hastelloy X tubing could be determined. The program was terminated prior to the initiation of experimental activities.

ML-1 TECHNOLOGY PROGRAM BIBLIOGRAPHY

1. Langer, B. and O'Donnell, W., "Design of Perforated Plates," ASME 61-WA-115, 1962

This paper presents a method for estimating the elastic constants of perforated plates suitable for the evaluation of the design of such assemblies.

2. AN-AGCR-910, "Summary Report - ML-1 Pressure Vessel Technology Evaluation," July 1965

Describes the photoelastic experimental program undertaken to develop values of the elastic constants to be used in the analysis of ML-1 type tube sheets for comparison with the results of the work by Langer and O'Donnell.

3. AN-AGCR-911, "Conceptual Design Report - Advanced ML-1 Pressure Vessel," July 1965

Presents the conceptual design of an ML-1 type pressure vessel which eliminates the known deficiencies in the ML-1 assembly.

4. AN-AGCR-890, "TSA, An IBM 7090/7094 Computer Program for Analysis of ML-1 Tube Bundle Stresses," June 1965

Describes the modifications made to the code to permit inputting Langer and O'Donnell data or data developed during the photoelastic experimental program.

5. AGN TM-414*, "Studies of Hastelloy X Corrosion in Air at 1900 and 2000°F," December 1965

Presents the results of laboratory studies of the corrosion properties of Hastelloy X for periods up to 10,000 hours.

6. AGN TM-413*, "Isothermal Oxidation of Hastelloy X at 1750 and 1850°F in Air," December 1965

Presents the results of the experimental program conducted with a continuous weighing apparatus to evaluate the oxidation properties of Hastelloy X.

7. AN-AGCR-956, "Centrifugal Creep Testing of Hastelloy X - Status Report," November 1965

Presents the status of this program at the time of termination of AGCRSP activities.

VII. ML-1A PROGRAM

A. ML-1A PERFORMANCE SPECIFICATION

As indicated earlier, one of the AGCRSP objectives established in June 1959 was the development of plans and specifications for a military prototype of a mobile gas-cooled nuclear power plant (ML-1A) which could serve as the basis for fixed-price procurement. As the work under the ML-1 program proceeded, it became apparent that this plant could not be considered as a prototype because of the requirement for auxiliary equipment and instrumentation to permit an assessment of the performance of selected components and the subsequent realization that additional development would be required in certain areas.

As a consequence, the original objective was modified to require the development of performance specifications for the ML-1A plant by 30 June 1963. In addition, the preparation of outlines of the operating manuals required for field service was authorized. The performance specification was to reflect the latest information available from the ML-1 development program and was to be specific in those areas where it was felt that development was complete but more general in the areas where additional developmental activity was indicated.

In satisfaction of this directive, an ML-1A "Bid Package" was transmitted to the USAEC on 28 June 1963.⁽¹⁾

B. ML-1A PRELIMINARY DESIGN

The U. S. Army authorized and funded the development of a preliminary design for the ML-1A plant in the fall of 1963. The performance criteria for the plant were specified in a draft QMR (Appendix E) which included several significant changes from the ML-1 Military Characteristics*.⁽²⁾ The preliminary design activity covered all aspects of the plant design, including the reactor and shielding⁽³⁾⁽⁴⁾⁽⁵⁾, fuel elements⁽⁶⁾, the precooler⁽⁶⁾, alternator⁽⁶⁾⁽⁷⁾, start motor⁽⁶⁾, instruments and controls⁽⁸⁾⁽⁹⁾⁽¹⁰⁾ and systems analysis.⁽¹¹⁾⁽¹²⁾⁽¹³⁾⁽¹⁴⁾⁽¹⁵⁾

*Major areas of difference included more stringent radiation dose rate criteria during plant operation, more stringent power quality specifications, the specification of a maximum allowable plant weight and relaxation of the allowable auxiliary power requirement from 45 to 65 kw.

Following a review with the Government in February 1965⁽¹⁶⁾ at which time several design decisions were agreed to, the ML-1A preliminary design was completed and published⁽¹⁷⁾ along with a preliminary safety analysis.⁽¹⁸⁾ The nuclear design activity included a recommended program for the determination of power distribution in the ML-1A core to permit the definition of the reactivity shim and orifice patterns.⁽¹⁹⁾

The ML-1A preliminary design activity reflected the significant changes determined to be required by the ML-1 development program. The concept for the improved precooler was altered from the finned-tube type to a plate-fin assembly to improve the effectiveness and reliability. A commercial brush-type alternator was determined to be acceptable because the equipment on the power conversion skid could be rearranged to accommodate the larger size machine. The preliminary designs for an improved starting system and a simplified and improved lubrication system were developed. The incorporation of these advances into the ML-1 program was discussed earlier (Section V-G).

The ML-1A preliminary design was reviewed with cognizant and potential-user Government agencies in September 1964. In November 1964, AGN was requested to perform a study to evaluate the feasibility of redesigning or supplementing the ML-1A shielding to reduce the radiation dose rate during operation by a factor of 10 and following shutdown by 67%. The results of this study⁽²⁰⁾ indicated that the desired radiation criteria could be achieved by modification of the internal shielding and the addition of an external shield (increasing the size and weight of the plant). Further reductions in radiation dose rates during reactor operation appeared possible by the use of expedient shielding.

ML-1A PROGRAM BIBLIOGRAPHY

1. ML-1A Bid Package

- a. AGC-60222, "Performance Specification - Nuclear Power Plant, Mobile, Low Power, Trailer Mounted, Gas-Cooled, 300-500 kw," January 1963
- b. AN-AGCR-266, "Outline of Operating Manual for ML-1 Mobile Nuclear Power Plant," June 1963
- c. AN-AGCR-267, "Outline of Pre-Operation and Post-Shutdown Procedures for ML-1 Mobile Nuclear Power Plant," June 1963
- d. AN-AGCR-268, "Outline of Maintenance Procedures for Reactor Package Assembly, ML-1 Mobile Nuclear Power Plant," June 1963
- e. AN-AGCR-269, "Outline of Maintenance Procedures for Power Conversion Assembly, ML-1 Mobile Nuclear Power Plant," June 1963
- f. AN-AGCR-270, "Pre-Startup and Maintenance Procedures for Gas Storage Equipment Assembly, ML-1 Mobile Nuclear Power Plant," June 1963
- g. AN-AGCR-271, "Outline of Maintenance Procedures for Control Cab and Instrumentation, ML-1 Mobile Nuclear Power Plant," June 1963
- h. AN-AGCR-272, "Outline of Maintenance Procedures for Cable Reel Assembly, ML-1 Mobile Nuclear Power Plant," June 1963
- i. AN-AGCR-273, "Outline of Maintenance Procedures for Reactor Drying Equipment Assembly, ML-1 Mobile Nuclear Power Plant," June 1963
- j. AN-AGCR-274, "Outline of Operating and Maintenance Procedures for Fuel Element Replacement, ML-1 Mobile Nuclear Power Plant," June 1963
- k. AN-AGCR-275, "Water Treatment Facilities Manual, ML-1 Mobile Nuclear Power Plant," June 1963
- l. AN-AGCR-276, "Health Physics Manual for the ML-1 Mobile Nuclear Power Plant," June 1963

NOTE: A compilation of ML-1 operating and maintenance ANSOP, selected portions of the ML-1 Design Data Report (AN-AGCR-385), a compilation of ML-1 as-built drawings, a compilation of ML-1 procurement specifications and copies of the ML-1 Transportation Manual (AGN TM-389) were also transmitted with the ML-1A Bid Package.

2. AN-AGCR-591, "Review of QMR for ML-1A," 24 September 1963

Defines and discusses the implications of the difference between the requirements of the QMR and the ML-1 Military Characteristics and concludes that a design change which will satisfy all requirements of the QMR is feasible.

3. AN-AGCR-644, "ML-1A Reactor Preliminary Design Special Progress Report," January 1964

Presents an interim summary of the design of the ML-1A reactor for comment and consideration by the Government.

4. AN-AGCR-831, "Interim Report - ML-1A Shielding Optimization Study," 9 April 1965

Outlines an experiment to be conducted at the ML-1 to determine the performance of the shield concepts selected for the ML-1A reactor.

5. AN-AGCR-904, "Test Report on the ML-1 Radial Tank Extension Experiment," 15 June 1965

Summarizes the results and findings of the ML-1A shielding experiment conducted at the ML-1.

6. AN-AGCR-635, "ML-1A Special Progress Reports," 31 December 1963

Presents an interim report of the status of the preliminary ML-1A fuel element, precooler, start motor and alternator designs.

7. AN-AGCR-668, "Selection of ML-1A Alternator Size," 12 March 1964

Discusses the considerations involved in the selection of the size of the alternator for the ML-1A and concludes that a 750 kva (600 kw) unit is required to take full advantage of the anticipated t-c set output and low ambient temperatures.

8. AN-AGCR-653, "ML-1A Instrumentation Study," 17 February 1964

Presents the results of a review of the ML-1 instruments to determine those required for the ML-1A plant as well as the ML-1A basic instrument plan and the listing of additional instrumentation required for diagnostic and analysis activities during the initial plant startup.

9. AN-AGCR-657, "ML-1A 50-Cycle Power Capability," 27 February 1964

The results of a study of alternate methods for providing 50 cycle capability in the ML-1A power plant are presented along with the conclusion that a separate converter skid represents the most logical approach to providing this desired capability.

10. AN-AGCR-661, "ML-1A Control Cab Layout," 2 March 1964

The preliminary layout of the ML-1A control cab is presented. This design represents the initial review by Human Engineering Laboratories personnel and was submitted for Government consideration and review.

11. AN-AGCR-573, "ML-1A Preliminary Energy Balance and Systems Analysis," 27 September 1963

The preliminary systems analysis and energy balance for the ML-1A power plant are presented.

12. AN-AGCR-636, "ML-1A Reference Systems Analysis," 6 January 1964

The reference systems analysis for the ML-1A power plant is presented along with component performance curves which permit an estimate of plant performance at conditions other than the reference design point. The system discussed in this document incorporates the CSN-2 t-c set.

13. AN-AGCR-659, "ML-1A Reference Systems Analysis Using the Stratos TCS-670-2 Turbine-Compressor Set," 12 March 1964

The reference systems analysis for the ML-1A power plant incorporating the Stratos TCS-670-2 t-c set is presented.

14. AN-AGCR-674, "ML-1A System Performance as a Function of Ambient Temperature," 12 March 1964

Reports the results of an analysis to define the performance of the ML-1A reference system as a function of ambient temperature.

15. AN-AGCR-672, "60 KW Startup for ML-1A with CSN-2," 12 March 1964

The results of an analysis to determine the startup power requirements for the ML-1A power plant are presented. It is concluded that the plant can be started up with less than 60 kw (58.1 kw) at an ambient temperature of 125°F and that the startup power requirements are reduced as the ambient temperature increases.

16. AN-AGCR-817, "ML-1A Preliminary Design Review," 12 February 1965

Reports the results of meetings with representatives of various Government agencies to review the interim ML-1A preliminary design and to discuss and resolve certain design questions.

17. AGN TM-408, "ML-1A Preliminary Design Report," 1 June 1964

The preliminary design of the ML-1A power plant is presented in five volumes as follows:

- I. Introduction, Power Plant Description and Systems Analysis

- II. Detailed Description of the Reactor and Fuel Element Design
- III. Detailed Description of the Power Conversion Skid, Instruments and Controls System and Auxiliaries
- IV. Appendixes
- V. Preliminary Design Drawings

18. IDO-28628 (Draft), "ML-1A Safety Analysis," September 1964

The analysis of the safety of the operation of the ML-1A power plant at the NRTS is presented. The maximum credible accident is defined and the results of this and less severe accidents are evaluated.

19. AN-AGCR-658, "ML-1A Initial Power Measurements," 27 February 1964

An experiment is outlined to determine the power distribution in the ML-1A reactor and to provide data required for the selection of the reactivity shim pattern and the fuel element orificing pattern.

20. AGN TM-412, "ML-1A Shield Optimization Study," 24 November 1965

Reports the results of the analytical evaluation of several modifications to the ML-1A internal shield as well as several concepts for external and expedient shielding. It is concluded that the radiation dose rate criteria established for the study can be met only by significantly increasing the size and weight of the power plant.

VIII. ML-2 PROGRAM

The conduct of a study to evaluate the feasibility of and to define the significant design criteria for an advanced mobile nuclear power plant was authorized by the USAEC in the fall of 1961 and completed early in 1962. The results of this activity indicated that significant improvements (with respect to the ML-1 characteristics) could be achieved in an advanced power plant.⁽¹⁾

Upon completion of the evaluation of the results of the preliminary study, the USAEC authorized a detailed evaluation and the development of the conceptual design for a second generation gas-cooled mobile nuclear power plant (ML-2). This study considered a wide variety of possible designs for a plant capable of producing 500 kw(e) including evaluation, optimization and selection of the coolant/working fluid, the thermodynamic cycle, the reactor concept, the critical materials and the shielding concept. The system which evolved as the optimum gas-cooled nuclear power plant incorporated a closed regenerative Brayton cycle with air as the working fluid and an yttrium hydride-moderated, UO_2 -fueled all-pin reactor core. The evaluation indicated that the selected concept represented significant improvements in terms of reliability, simplicity, mobility, and logistic support requirements over existing military power plants.⁽²⁾ Following the completion of the study outlined above, the AEC requested that a limited investigation of an unmoderated reactor concept be undertaken. This investigation revealed that, although the unmoderated concept was satisfactory for the power plant and had several attractive thermodynamic characteristics, in the overall evaluation the concept was not superior to that recommended as a result of the earlier work.⁽³⁾

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ML-2 PROGRAM BIBLIOGRAPHY

1. AGN TM-396, "The Feasibility Study of an Advanced Nuclear Power Plant," September 1962

Reports the results of feasibility studies of a 500 kw(e) mobile nuclear power plant and discusses design concepts, materials and fuel cycles for such a plant which represent significant improvement (when compared with the ML-1) in mobility, reliability, simplicity, core life-time, radiation levels and maintainability. The results of parametric and thermodynamic analyses are presented.

2. IDO-28584, "Study of Mobile Gas-Cooled Nuclear Power Plants," September 1962 (Two volumes)

Reports the results of a study of a wide variety of possible designs for a mobile gas-cooled nuclear power plant to produce 500 kw(e) for military applications. The considerations which led to the selection of the optimum reactor coolant/working fluid, thermodynamic cycle, reactor concept, and critical materials are discussed. The conceptual designs for several possible systems are presented. The High Density Moderated Reactor System (HDMR) is presented as the optimum concept for the specified application. This system incorporated a closed regenerative Brayton cycle operating on air with a UO_2 fuel yttrium-hydride moderated all-pin reactor.

3. IDO-28584, Addendum I, "Study of Mobile Gas-Cooled Nuclear Power Plants," June 1963

The results of a study to evaluate the feasibility of the use of an unmoderated europium-poisoned reactor in a mobile gas-cooled nuclear power plant are presented. A comparison is made between characteristics of such a reactor and those of the HDMR reactor and it is concluded that this concept is not superior to the HDMR.

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IX. RELATED AGCRSP PROGRAMS

1. CONTRACT DA-44-009-ENG-3252

This contract, between the U. S. Army, ERDL, and Aerojet, provided for the completion of the architect-engineering phase of the design of the Gas Turbine Test Facility (GTTF) at Ft. Belvoir, the procurement of components (except the t-c set) and the construction of the test facility and the initial checkout operation of the facility. The t-c set (TCS-560) was procured by ERDL under a contract with the Stratos Division of Fairchild Engine and Aircraft Co. The work began in early 1959 and was completed in July 1960.

2. CONTRACT DA-44-192-ENG-8

This contract, also between ERDL and Aerojet, provided for:

- The test operation, maintenance and modification of the GTTF.
- The procurement of two t-c sets for the ML-1 power plant (CSN-1 from Clark Bros. and TCS-670 from Stratos).
- The procurement of a nitrogen heater for p/c skid testing at AGC facilities.
- The procurement of an improved recuperator for the GTTF.

This contract was in force from July 1960 until May 1963; all work was completed except the recuperator procurement (Authorization for this activity was withheld by ERDL).

3. APCEF PROGRAM

This work was sponsored at the GTTF site by ERDL (APCD Branch) under contracts with Burns & Roe and, subsequently, Mechanical Technology, Inc. The goal of this activity was the completion of development of the Stratos-series t-c sets (responsibility for completion of the development of the Clark machines was assigned to USAEC-Aerojet). As of the date of termination of the AGCRSP activity, limited success had been obtained in the testing of the TCS-560 (the machine had operated for many hours and developed up to 200 kw(e)).

No testing of the TCS-670 beyond that conducted under the AGCRSP (see Section V-C-6) had been undertaken and the projected continuing program contemplated evaluation of the TCS-670-2, scheduled for delivery early in 1966.

4. BACKUP T-C SET PROCUREMENT

The ERDL (APCDB) initiated procurement of backup t-c sets (TCS-670-2 from Stratos and CSN-2 from Clark Bros.) in the fall of 1962 when it appeared that neither the CSN-1 or TCS-670 would operate for sufficient periods of time to provide data for further development. As of the date of termination of AGCRSP activity, the CSN-2 machine had been received by APCDB and assigned to the USAEC for use in the ML-1 program; the TCS-670-2, as indicated earlier, is scheduled for delivery early in the spring of 1966.

APPENDIX A

TECHNICAL PROGRESS REPORT BIBLIOGRAPHY

This appendix lists and provides typical abstracts of the technical progress reports published under the AGCRSP. Monthly progress reports, published until September 1962, and semiannual reports, published for each six month period until 31 December 1962, documented technical progress until those dates. Thereafter, the format of the monthly report was altered to provide a summary of program management activities and all technical accomplishment was documented in quarterly technical progress reports. In general, monthly reports were not published for the last month of the period covered by a semiannual report.

All reports listed in this appendix were published under IDO identification numbers; semiannual and quarterly reports and the monthly reports through February 1959 were distributed to agencies and individuals listed in the appropriate category of the then current edition of the TID-4500 distribution list. The other monthly progress reports were distributed only to a limited number of persons closely associated with the AGCRSP. All reports were published under Contract AT(10-1)-880 by Aerojet-General Nucleonics, San Ramon, California; this repetitive information is omitted from the listing in the interest of conserving space and making the list more useful.

A. MONTHLY PROGRESS REPORTS

These reports summarize the technical accomplishments during the report period but do not present detailed data, etc. Each significant area of current activity is discussed and a current tabulation of ML-1 power plant design characteristics is presented as an appendix in each of the AGCRSP reports.

<u>Period Covered</u>	<u>Type Report</u>	<u>Document No., IDO</u>	<u>Report Date</u>
21 Apr-20 May, 1957	GCRE Monthly	28501	23 Sept 1957
21 May-20 June, 1957	GCRE Monthly	28502	23 Sept 1957
21 June-20 July, 1957	GCRE Monthly	28504	7 Jan 1958
21 July-31 Aug, 1957	GCRE Monthly	28515	7 Jan 1958
Sept, 1957	GCRE Monthly	28516	15 Jan 1958

<u>Period Covered</u>	<u>Type Report</u>	<u>Document No., IDO</u>	<u>Report Date</u>
Oct, 1957	GCRE Monthly	28517	18 Feb 1958
Nov, 1957	GCRE Monthly	28518	8 Jan 1958
Jan, 1958	GCRE Monthly	28520	28 Feb 1958
Feb, 1958	GCRE Monthly	28521	26 Mar 1958
Mar, 1958	GCRE Monthly	28523	28 Apr 1958
Apr, 1958	GCRE Monthly	28524	28 May 1958
May, 1958	GCRE Monthly	28525	26 June 1958
July, 1958	GCRE Monthly	28527	29 Aug 1958
Aug, 1958	GCRE Monthly	28528	30 Nov 1958
Sept, 1958	GCRE Monthly	28529	31 Oct 1958
Oct, 1958	AGCRSP Monthly	28531	24 Nov 1958
Nov, 1958	AGCRSP Monthly	28532	30 Dec 1958
Jan, 1959	AGCRSP Monthly	28535	28 Feb 1958
Feb, 1959	AGCRSP Monthly	28536	31 Mar 1959
Mar, 1959	AGCRSP Monthly	28538	30 Apr 1959
Apr, 1959	AGCRSP Monthly	28540	25 May 1959
May, 1959	AGCRSP Monthly	28541	June 1959
July 1959	AGCRSP Monthly	28543	August 1959
August 1959	AGCRSP Monthly	28544	September 1959
Sept, 1959	AGCRSP Monthly	28545	October 1959
October 1959	AGCRSP Monthly	28546	November 1959
November 1959	AGCRSP Monthly	28548	December 1959
January 1960	AGCRSP Monthly	28551	February 1960
February 1960	AGCRSP Monthly	28553	March 1960
March 1960	AGCRSP Monthly	28554	April 1960
April 1960	AGCRSP Monthly	28556	May 1960
May 1960	AGCRSP Monthly	28557	June 1960
July 1960	AGCRSP Monthly	28559	August 1960
August 1960	AGCRSP Monthly	28562	September 1960
September 1960	AGCRSP Monthly	28563	October 1960
October 1960	AGCRSP Monthly	28565	November 1960
November 1960	AGCRSP Monthly	28566	December 1960
January 1961	AGCRSP Monthly	28568	February 1961
February 1961	AGCRSP Monthly	28569	March 1961
March 1961	AGCRSP Monthly	28570	April 1961

<u>Period Covered</u>	<u>Type Report</u>	<u>Document No., IDO</u>	<u>Report Date</u>
April 1961	AGCRSP Monthly	28571	May 1961
May 1961	AGCRSP Monthly	28572	June 1961
July 1961	AGCRSP Monthly	28574	August 1961
August 1961	AGCRSP Monthly	28575	September 1961
September 1961	AGCRSP Monthly	28576	October 1961
October 1961	AGCRSP Monthly	28577	November 1961
November 1961	AGCRSP Monthly	28579	December 1961
December 1961	AGCRSP Monthly	28580	January 1962
January 1962	AGCRSP Monthly	28583	February 1962
February 1962	AGCRSP Monthly	28585	March 1962
March 1962	AGCRSP Monthly	28586	April 1962
April 1962	AGCRSP Monthly	28587	May 1962
May 1962	AGCRSP Monthly	28588	June 1962
June 1962	AGCRSP Monthly	28589	July 1962
July 1962	AGCRSP Monthly	28592	August 1962
August 1962	AGCRSP Monthly	28593	September 1962
September 1962	AGCRSP Monthly	28594	October 1962

B. SEMIANNUAL PROGRESS REPORTS

This series of reports documents the details of technical accomplishment during the report period. The information is grouped under discussions of the major projects and data, curves, photographs, etc., are included to support the technical conclusions presented. A typical semiannual progress report contains an abstract, a discussion of test operations and the engineering support for same, a presentation of information concerning developmental activities (reactor, fuel elements, power conversion equipment, instruments and controls, etc.), a discussion of major design activities (GCRE-II, ML-1A, etc) and a summary of activities under AGCRSP-related contracts other than AT(10-1)-880 (see Section IX).

<u>Period Covered</u>	<u>Type Report</u>	<u>Document No., IDO</u>	<u>Report Date</u>
1 Nov 1956-30 June 1957	GCRE Semiannual	28505	February 1958
1 July-31 Dec 1957	GCRE Semiannual	28519	June 1958
1 Jan-30 June 1958	AGCRSP Semiannual	28526	October 1958
1 July-31 Dec 1958	AGCRSP Semiannual	28533	February 1959
1 Jan-30 June 1959	AGCRSP Semiannual	28542	July 1959
1 July-31 Dec 1959	AGCRSP Semiannual	28549	January 1960

<u>Period Covered</u>	<u>Type Report</u>	<u>Document No., IDO</u>	<u>Report Date</u>
1 Jan-30 June 1960	AGCRSP Semiannual	28558	July 1960
1 July-31 Dec 1960	AGCRSP Semiannual	28567	January 1961
1 Jan-30 June 1961	AGCRSP Semiannual	28573	July 1961
1 July-31 Dec 1961	AGCRSP Semiannual	28581	February 1962
1 Jan-30 June 1962	AGCRSP Semiannual	28590	August 1962
1 July-31 Dec 1962	AGCRSP Semiannual	28602	February 1963

C. QUARTERLY PROGRESS REPORTS

These reports are, essentially, an extension of the detailed technical documentation presented in the semiannual progress reports (see B above). The format and content are similar to those of the semiannual; an introductory section summarizing progress under the AGCRSP up to the beginning of the report period is included in each quarterly report.

<u>Period Covered</u>	<u>Type Report</u>	<u>Document No., IDO</u>	<u>Report Date</u>
1 Jan-31 Mar 1963	AGCRSP Quarterly	28607	May 1963
1 Apr-30 June 1963	AGCRSP Quarterly	28612	August 1963
1 July-30 Sept 1963	AGCRSP Quarterly	28617	November 1963
1 Oct-31 Dec 1963	AGCRSP Quarterly	28621	February 1964
1 Jan-31 Mar 1964	AGCRSP Quarterly	28626	May 1964
1 Apr-30 June 1964	AGCRSP Quarterly	28632	August 1964
1 July-30 Sept 1964	AGCRSP Quarterly	28637	November 1964
1 Oct-31 Dec 1964	AGCRSP Quarterly	28641	February 1965
1 Jan-31 Mar 1965	AGCRSP Quarterly	28645	May 1965
1 Apr-30 June 1965	AGCRSP Quarterly	28649	August 1965
1 July-31 Oct 1965	AGCRSP Quarterly	28653	January 1966

APPENDIX B

ML-1 MILITARY CHARACTERISTICS

This appendix presents the subject characteristics as of 30 November 1960. No revisions have been made since that date.

- I. Title: Mobile Nuclear Power Plant, 300-500 kw, ML-1
- II. Security Classification:
 - A. Existence of Project: Unclassified
 - B. Military Characteristics: Unclassified
 - C. Plant Design and Performance: Unclassified
- II. Proposed Use: There are requirements within the military services for general purpose portable nuclear power plants with electric power outputs in the range from 300 to 500 kw. The ML-1 is the prototype of field units which will be procured to meet these requirements. Examples of military operations requiring a power source without dependence on continuous fuel or cooling water supply are: support of tactical missile systems, air head operations including field hospitals, remote locations, etc.
- IV. Environmental Characteristics: The plant shall have the inherent capability for acceptable performance under the Basic Operating Conditions, Extreme Cold Weather Operating Conditions, and Extreme Hot Weather Operating Conditions as established in Paragraphs 7a, 7b, 7c of AR 705-15; and shall be capable of safe storage and transportation under conditions as established in Paragraph 7d of AR 705-15. A cold weather kit may be utilized for operation under extreme cold weather conditions.
- V. Power Characteristics:
 - A. Nominal Output at "Design Conditions" (Ambient temperature -65°F to 100°F).

- (1) Electrical power output in the range of 300-500 kw, at 0.8 power factor.
- (2) 2400/4160 volts, 3 phase, 60 cycles per second

B. Output at "Off Design Conditions":

- (1) Net output of the system in the event of hot weather ($>100^{\circ}\text{F}$) or high altitude operation shall be limited only by the plant heat rejection capacity.
- (2) Operation shall be possible up to 500 kw net output except as limited by plant heat rejection capacity.

C. Power Quality:

- (1) Objective: Steady state and transient frequency and voltage control characteristics as specified by MIL-G-14609 (CE).
- (2) Minimum Requirement: Steady state and transient frequency and voltage control characteristics as specified by MIL-G-10328A (CE).

D. Other Requirements:

- (1) Automatic power level control is required
- (2) The power plant shall operate satisfactorily in parallel with other units of similar rating.
- (3) The power plant shall be capable of 50 cycle operation with corresponding reduced power output.
- (4) The power plant shall be treated for the elimination of interference with radio communication in accordance with applicable Signal Corps specifications.

VI. Field Operating Characteristics:

- A. The plant unit (connected reactor and power conversion package) shall be assembled on a standard military semi-trailer (M-172 or M-172A1) and all preparations completed for relocation to an operating site within six (6) hours after unloading from any aircraft.
- B. The plant shall be capable of being installed and delivering rated power within twelve (12) hours after arrival at an operating site.
- C. The plant shall be capable of relocation to a new site beginning twenty-four (24) hours after reactor shutdown following operation for extended periods at full power. Completion of disassembly and loading shall be within thirty-six (36) hours after reactor shutdown.

D. Nuclear Radiation Allowed:

- (1) Allowable radiation, twenty-four (24) hours after shutdown at twenty-five (25) feet from the reactor in the direction of the cab (without water shield), is fifteen (15) mr/hr with the power conversion package in place.
- (2) No personnel shall receive greater than 3 rem/quarter.

E. Equipment may be operated in any plane within five degrees (5°) from level.

F. Control of the power plant will be from a separate shelter which shall be transportable as a unit on a standard $2\frac{1}{2}$ ton truck (M-35) which, during plant operation, shall be connected to the power plant by quick disconnect electrical cables of sufficient length (up to 500 ft) to permit flexibility in the relative locations of the power plant and control shelter.

G. Plant design shall include necessary safety features for protection of personnel in the immediate area from results of reasonably conceivable mechanical, electrical, or nuclear malfunctions.

H. During plant operation, radiation shielding integral to the plant may be supplemented by field expedient materials (earth, gravel, wood, water) sufficient to reduce radiation levels outside the combined shield to within dosage levels prescribed by the Surgeon General D/A. The integral plant shield (excluding the supplemental expedient shielding) shall be adequate to reduce the residual radiation level following full power operation for extended periods of safe levels in time to permit relocation twenty-four (24) hours after reactor shutdown.

I. The operating crew for three-shift operation of the field unit shall consist of no more than seven (7) men (one operator per shift plus mechanic-electrician, radiological safety technician and officer or NCO in charge). The crew will be specially trained for operation, maintenance, and plant installation at the site.

VII. Transportability Characteristics:

A. Objective: Capability of being transported overland (highway and cross-country), by rail, by water, and by aircraft in accordance with AR 705-8, Transportability.

B. Requirements: The ML-1 plant shall be consistent with the following transportation requirements:

- (1) General: The "power plant unit" is defined for transportability as the complete integral power plant including reactor, shutdown shielding, power conversion equipment and skid mounting. Control unit, auxiliary power unit, auxiliary equipment packages and trailer are not included in this definition. Required modes of transport are indicated below.

(2) Primary Mode of Transportation:

- (a) The primary mode of transport of the power plant unit shall be overland on standard military semi-trailer. Power plant unit integrity is required.
- (b) The control unit shall be one package which can be transported intact and which is suitable for both operation and transport on a standard military 2½ ton truck.

(3) Secondary Modes of Transportation: (Power plant unit integrity required.)

- (a) Secondary modes of transport for the power plant unit where unit integrity is required include:
 - 1. Normal railway freight service U. S. and foreign.
 - 2. Water freight by ship or barge.
- (b) In addition, when necessary and where satisfactory clearances can be arranged, the power plant unit shall be transportable trailer-mounted on the above transit types.
- (c) Similarly, when truck-mounted, the truck and control unit shall be transportable as one package when necessary and where satisfactory clearances can be arranged for the above transit types.

(4) Secondary Modes of Transportation: (Power plant unit integrity not required.)

- (a) The power plant unit may be separated into two packages for transport by:
 - 1. USAF C-130.
 - 2. USAF C-124C.
 - 3. Sled.
- (b) Trailer mounting is not required for (a) above.
- (c) The control unit shall be transportable intact. Truck mounting is not required for (a) above.
- (d) The complete power plant including the control unit and auxiliary equipment shall be transportable in one USAF C-133.

(5) Other Requirements:

- (a) The power plant shall be transportable in Phase III of airborne operations.
- (b) Design for air transport shall conform to requirements of USAF Specification MIL-A-8421A for transportation of material in military cargo aircraft.
- (c) Provisions shall be made to permit rapid loading and unloading of each unit by standard weight handling techniques.
- (d) Shock and vibration protection shall be provided integral with the plant consistent with each transport type as indicated below:

- 1. Maximum shock transmitted to cargo bed of transport vehicle without loss of serviceability, utilizing a shock isolation tiedown system:

<u>Direction</u>	<u>Acceleration (g)</u>	<u>Duration of versed sine pulse, (sec)</u>
Fore and aft	15	0.030
Forward only	8	0.1
Fore and aft	3	Static
Vertical up	4	0.030
Vertical down	4.5	0.1
Lateral	2	0.030

- 2. Maximum shock transmitted to cargo bed of transport vehicle without loss of plant serviceability, utilizing rigid tiedown:

<u>Direction</u>	<u>Acceleration (g)</u>	<u>Duration of versed sine pulse, (sec)</u>
Fore and aft	5	0.1
Fore and aft	3	Static
Vertical	4.5	0.1
Lateral	1.5	Static

- 3. Steady vibrations in transit without loss of serviceability.

<u>Source</u>	<u>Peak Amplitude (inch)</u>	<u>Frequency (cps)</u>
Railroad flatcar	0.12	20-30
Semi-trailer	0.05	>20
Ship	0.05	15-20
Aircraft	0.05	>20

(e) Weight:

1. Heaviest package for air transport is 15T.
2. Trailer load (M-172 or M-182A1) is 30T.
3. Control shelter is 2.5T.

(f) Size:

1. Largest package shall be transportable on C-124, C-130, and C-133 aircraft; and on railroad with limits as prescribed by the Berne International Clearance Diagrams.
2. Reactor and power conversion skids shall be transportable as a unit on the M-172 or M-172A1 semi-trailer.
3. Maximum height when trailer-mounted is 150 inches. Power plant height when trailer-mounted may be reduced to 132 inches for the purpose of passing low clearance obstacles outside CONUS.
4. Length and width shall be compatible with a standard military semi-trailer (M-172 or M-172A1).

- (g) When trailer-mounted and in transit, the plant shall have the capability of shallow fording in fresh or salt water, as defined in AR 705-2300-8. Extreme caution must be taken to avoid a nuclear accident due to core flooding.

VIII. Logistical Characteristics:

- A. The plant shall operate with no more than periodic field maintenance service for 10,000 hr between major overhaul.
- B. Routine plant maintenance shall be performed every 30 days or at greater intervals, as determined by operational testing.

- C. The plant shall have 50,000 hr total life.
- D. The plant shall be capable of full power operation for 10,000 hours between refueling operations. Refueling may be accomplished in the field.
- E. The plant shall be capable of operation without a continuous water supply. Any initial quantity of water (for shielding or reactor moderator) will be kept to a minimum and of a quality which can be supplied by standard military purification equipment capable of delivering 600 gph.
- F. Radioactive waste from the plant will be kept to a minimum. Disposal of radioactive waste will be in accordance with field procedures approved by the Chief Chemical Officer. (NOTE: Disposal will be subject to AEC approved procedures at NRTS.)
- G. The plant will be completely self-sufficient to permit steady state operation without the need for auxiliary equipment, except for process fluid storage and makeup. For startup and shutdown operations, the plant should require no more than a 45 kw diesel generator and necessary construction equipment for erection of the supplemental expedient shield. Equipment for expedient shield construction shall not be part of the overall plant equipment.
- H. The power plant shall be equipped with all special purpose tools required for normal maintenance and startup, but not for refueling. Special equipment and procedures required for field refueling shall be specified. The prototype plant shall be equipped with a Maintenance Package as referenced in AR 705-5 and AR-750-6. The field plant shall be equipped with complete operating manuals defining detailed operating, maintenance, installation and relocation procedures. The manuals shall conform to the Publication Standards and Style Guide for Corps of Engineers Equipment Publications.

APPENDIX C

TURBINE-COMPRESSOR SET DESIGN CHARACTERISTICS

	<u>Stratos</u>	<u>Clark</u>
Speed, rpm	18,338	22,000
Turbine		
Type	Axial	Axial
Stages	2	2
Rotor Material	Incoloy 901	A-286 (first stage) AISI-422 (second stage)
Rotor Blade Material	Inco 713	M-252
Stator Blade Material	Inconel	N-155 or 19-9DL
Expansion Ratio	2.38	2.42
Compressor		
Type	Axial	Centrifugal
Stages	2	11
Rotor Material	AL355T-71	Type 403 stainless steel
Rotor Shift Material	SAE 4340	SAE 4340
Case Material	Type 304 stainless steel	Type 304 stainless steel
Compression Ratio	2.72	2.765

	<u>Stratos</u>	<u>Clark</u>
Seals		
Journal	Buffered labyrinth	Buffered labyrinth
Interstage	Labyrinth	Labyrinth
Shaft	Buffered labyrinth	Oil-Buffered double "L" ring
Bearings		
Journal	Tilting pad	Babbitt
Thrust	Kingsbury type	Kingsbury type (in low pressure area)
Support	Overhung turbine	Turbine and compressor supported between bearings

APPENDIX D

ML-1 DRAWING LIST

This appendix presents a partial list of the ML-1 drawings. The drawings included on this list show the plant assembly and the major subassemblies; copies are on file with both USAEC-ID (GCB) and USAEC-HQ (DRDT-ARM).

1-423600 (Sheets 1 thru 4)	ML-1 Mobile Nuclear Power Plant
1-405710 (Sheets 1 thru 6)	P&I Diagram, ML-1
1-408030	Single Line Diagram, Electrical Power, ML-1
1-423601 (Sheets 1 thru 8)	Reactor Package, ML-1
1-424178	Demineralizer Assembly, IE-501
1-410174	Fuel Element Assembly, ML-1
1-423445 (Sheets 1 and 2)	Reactor Primary Assembly, ML-1
1-423154	Drying Skid Assembly, ML-1
1-423571	Control Cab Assembly
1-403611	Control Cab Subassembly
1-405704	Pressure Vessel Assembly
090720	Power Conversion Assembly, ML-1
13203E2200 (Sheets 1 thru 5)	Heat Exchangers General Assembly
13203E2110 (Sheets 1 thru 8)	Recuperator Assembly
13203E2210 (Sheets 1 thru 3)	Precooler Assembly
13203E2250 (Sheets 1 and 2)	Oil Cooler Assembly
13203E2270 (Sheets 1 thru 3)	Moderator Cooler Assembly
13203E2710 (Sheets 1 and 2)	Turbine Compressor Set Assembly (Clark)
13203E2680 (Sheets 1 and 2)	Turbine Compressor Assembly (Stratos)

APPENDIX E

QUALITATIVE MATERIEL REQUIREMENT
FOR A MOBILE, LOW POWER, NUCLEAR
POWER PLANT, 300 KW (U)

Section 1 - Statement of Requirement

1. Statement of Requirement

- a. The security classification of technical and operational aspects of this item is UNCLASSIFIED.
- b. This mobile, low power nuclear power plant will fulfill the requirement for 300 kw of mobile, minimum logistics, electrical power for use by the Army in the field. It should operate for 10,000 full power hours on a single nuclear fuel loading. The combined reactor and power conversion unit should be mountable on a standard 25 ton low-bed trailer and should be transportable in standard cargo aircraft. The power plant will be capable of furnishing full power within twelve hours after arrival at an operating site and be capable of displacement within thirty-six hours after shutdown following extended full power operation.
- c. This plant is a new item. Although it will not completely replace conventional power generation equipment, it will provide the Army with a new capability for supplying electrical power in areas and under conditions where the supply of POL represents a difficult logistic problem.
- d. This plant will be assigned on a priority basis to installations requiring a minimum logistics power source, such as tactical operations centers, field army missile defense systems, remote camps of stations and airhead command and control centers. Its seven man operating crew can provide for continuous 24-hour power generation.
- e. The primary mode of transportation of the combined reactor and power conversion equipment shall be overland on a standard military semi-trailer. The control unit shall be transported

on a standard military 2½ ton truck. Secondary modes of transportation shall include rail freight, water freight and standard military cargo aircraft. For air transportation, reactor-power conversion unit integrity is desired but not essential. The complete power plant including the control unit and auxiliary equipment shall be transportable in one USAF C-133. The power plant shall be transportable in Phase III of airborne operations.

Section II - Operational, Organizational, and Logistics Concepts

2. Operational Concepts

- a. This mobile nuclear power plant will serve as a primary power source for installations such as tactical operations centers, field army missile defense systems, airhead command and control centers, remote camps or stations and Strike Force Command Posts. It will be capable of operation under the Basic Operating Conditions, Extreme Cold Weather Operating Conditions (with cold weather kit) and Extreme Hot Operating Conditions as established in paragraphs 7a, 7b, and 7c of AR 705-15, and will meet storage and transportation criteria established by paragraph 7d of AR 705-15.
- b. This plant will be capable of 24 hour operation by its seven man crew. Essential maintenance will be performed on a 30 day cycle; the plant, however, will operate for 10,000 full power hours on a single loading of nuclear fuel.
- c. This forty (40) ton mobile nuclear power plant will replace, on a priority basis, diesel generating equipment weighing approximately twenty (20) tons and consuming two and one-half (2.5) tons of fuel oil per 24 hour period. A single loading of nuclear fuel will therefore eliminate the logistics effort required to supply approximately 298,000 gallons (1040 tons) of fuel oil for consumption.

3. Organizational and Logistics Concepts

- a. This plant will fulfill mission-type assignments to units or installations having a priority requirement for mobile, minimum logistics power generation.
- b. Crew requirements for continuous 24 hour operation will consist of one officer (or noncommissioned officer) in charge plus six enlisted men. Special training will be the responsibility of the Army Nuclear Power Program, and crews will be trained concurrently with procurement of field plants.
- c. This plant will require minimum logistics support when assigned an operational mission. Initially all maintenance required will be performed by the crew, supplemented as necessary by a maintenance team organized in accordance with the TOE 5-500 concept.

Section III - Justification, Feasibility, and Priority

4. Reasons for Requirement

- a. There is an urgent operational requirement to reduce POL consumption in the field army. Existing conventional power generating equipment cannot be significantly improved with respect to fuel consumption. Achievement of a major reduction in local POL requirements is possible only through the use of mobile nuclear power plants.
- b. This requirement is supported by subparagraphs 1110b(4), 1112i, and 1139b(1) of the CDOG.

5. Technical Feasibility: An experimental prototype mobile, low power, nuclear power plant (ML-1), which is owned by the U. S. Atomic Energy Commission, is currently undergoing an extensive test program at the National Reactor Testing Station in Idaho. The program to date has successfully demonstrated the technical feasibility of this item.

6. Priority: In accordance with subparagraph 16d, AR 705-5, the recommended Department of the Army priority is I.

Section IV - Characteristics

7. Performance Characteristics

- a. (essential). The plant output at design conditions (ambient temperature - 100°F) shall be a minimum of 300 kw net electrical, 2400/4160 volts, 3 phase, 60 cycles per second, 0.8 power factor at sea level.
- b. (essential). The plant shall perform with steady and transient frequency and voltage control characteristics as specified in Specification MIL-G-10328A (CE) except that voltage waveform deviation factor shall comply with Specification MIL-G-14609A (CE).
- c. (desirable). The plant shall perform with steady and transient frequency and voltage control characteristics as specified in MIL-G-14609A (CE).
- d. (essential). The plant shall be capable of self-sustained operation without the need for an external power supply. No more than 60 kw of external power shall be required for startup and shutdown purposes.
- e. (desirable). An internal means for starting the turbine-compressor set should be provided to reduce external electric power required for startup and shutdown to less than 45 kw.
- f. (essential). The plant shall meet the requirements for radio interference suppression in accordance with AR 105-68.

- g. (essential). The plant shall operate satisfactorily in parallel with other units built in compliance with Specification MIL-G-10328A (CE).
- h. (desirable). The plant shall operate satisfactorily in parallel with other units built in compliance with Specification MIL-B-14609A (CE).
- i. (essential). The plant shall be capable of being installed and delivering rated power within eighteen hours after arrival at an operating site. Twelve hours is desirable.
- j. (essential). The plant shall be capable of being prepared for relocation to a new operating site within forty-eight hours after shutdown following full power operation. Thirty-six hours is desirable.
- k. (essential). The power plant shall be capable of safe operation when sited in any location within 5% of level.
- l. (essential). The plant shall be capable of operation under the Basic Operating Conditions, Extreme Cold Weather Operating Conditions (with cold weather kit), and Extreme Hot Weather Operating Conditions (at a reduced power level) as established in paragraphs 7a, 7b, and 7c of AR 705-15, and will meet storage and transportation criteria established in paragraph 7d of AR 705-15.
- m. (desirable). The plant shall be capable of conversion to produce 50 cycles per second power at 83% of design power level.

8. Physical Characteristics

- a. (essential). The reactor unit and power conversion unit shall not weigh more than 15 tons each. Coupled, the weight shall not exceed 30 tons. The control cab weight shall not exceed 2.5 tons. Total systems weight shall not exceed 40 tons. The maximum number of packages comprising the total system shall not exceed 7.
- b. (essential). The reactor and power conversion unit, coupled, shall be capable of transportation by:
 - (1) Standard military semi-trailer.
 - (2) Railroad, with limits as prescribed by the Berne International Clearance Diagrams.
 - (3) Water freight by ship or barge.
 - (4) The control cab shall be transportable in a standard military 2½ ton cargo truck. Remaining packages shall be capable of transport on cargo vehicles with capacities no greater than 5 tons.

- c. (essential). The plant shall be designed for transportation in standard military cargo aircraft in conformance with USAF Specification MIL-A-8421B. The plant shall be capable of transportation, with reactor and power conversion unit uncoupled, by:

- (1) USAF C-130 (three aircraft).
- (2) USAF C-124 (three aircraft).
- (3) USAF C-133 (one aircraft).

Three aircraft loads include reactor unit, power conversion unit, and control cab, auxiliary skids and operating crew.

- d. (essential). Integral shock and vibration protection shall be provided with the plant, consistent with each of the above modes of transportation.
- e. (essential). When operated in flat, open terrain the control cab shall not be located more than 500 feet from the reactor and the resulting radiation exposure shall not exceed 5 mrem/hr.

9. Maintenance Characteristics

- a. (essential). The plant shall be capable of operation without a continuous water supply. The initial quantity of water required will be minimized and of a quality which can be supplied by standard 600 gph military water purification equipment.
- b. (essential). The plant shall be equipped with all special purpose tools required for installation at a prepared site, startup, shutdown, and normal maintenance.
- c. (desirable). The plant shall be capable of being refueled in the field, when the crew is supplied with special equipment not normally issued with the plant.
- d. (desirable). Routine scheduled plant maintenance shall be required not more frequently than every 30 days. Maintenance should be accomplished within 48 hours, including a 24 hour waiting period to permit radiation to decay to within acceptable limits.
- e. (desirable). The power plant shall operate for 10,000 full power hours between major overhauls and between fuel changes. It should require no longer than 30 days to accomplish both overhaul and fuel change.
- f. (desirable). The power plant shall have a useful service life of 50,000 full power hours.
- g. (desirable). The system shall be designed for minimum maintenance.

- h. (essential). Whenever practical, repair by replacement of modules shall be utilized. Modules are defined as reactor skid, power conversion skid, control cab, and control cable.
- i. (essential). The design shall permit ease of accessibility to frequently checked components such as batteries, filters and lubrication points.

10. Human Engineering Characteristics

- a. (essential). The plant shall be capable of safe operation at the design power output.
- b. (essential). Allowable nuclear radiation during full power operation shall be 5 millirem per hour or less at the control cab when the plant is operated in flat, open terrain and the control cab is located 500 feet from the reactor. Radiation shielding integral to the plant may be supplemented as required by field expedient materials (earth, gravel, wood) sufficient to achieve the above radiation level where operational requirements do not permit a 500 foot separation between the control cab and the reactor.
- c. (essential). Allowable nuclear radiation twenty-four hours after shutdown and twenty-five feet from the reactor (without water shield, with power conversion unit in place) in the direction of the tractor-trailer cab shall not exceed 15 millirem per hour.
- d. (essential). The power plant shall be provided with both automatic and manual power level control.
- e. (essential). Radioactive waste resulting from operation of the plant shall be kept to a minimum.
- f. (essential). Human factors engineering characteristics of the plant will include, but not be limited to, consideration of each of the following, in terms of the intellectual, physical and psychomotor capabilities of the intended user.
 - (1) Physiological requirements; e.g., visual and auditory perceptual requirements.
 - (2) Human space requirements for ease of operations and maintenance. All items must accommodate 90% (5th percentile to 95th percentile) of the intended user personnel in terms of height, width and other relevant physical dimensions. This must include cold weather personal clothing and personal or protective equipment (e.g., CBR individual protection) worn or carried by user personnel.
 - (3) Control display relationship.
 - (4) Work space layout.

- (5) Communications.
- (6) Environmental factors such as temperature, humidity, dust, noise, vibration, blast, noxious fumes, chemical and biological agents.
- (7) Safety in operation and maintenance.
- (8) Readability of dials, meters, counters, etc.
- (9) Operability of switches, knobs, controls, etc.
- (10) Comfort only as required to ensure crew efficiency.
- (11) All practical operations shall be automated for ease of operator control.

11. Priority of Characteristics

Characteristics are listed in relative order of priority in paragraphs 7, 8, 9, and 10 above.

Section V - Personnel and Training Considerations

12. Quantitative and Qualitative Personnel Considerations

- a. The crew requirement for continuous 24 hour operation will consist of one officer (or non-commissioned officer) in charge and six enlisted men organized in accordance with the TOE-5-500 concept.
- b. The operating crew will all be graduates of the Nuclear Power Plant Operators Course (AR 350-224) and in addition will be trained on the engineering test model (ML-1) and/or the service test model (ML-1A). Satisfactory completion of the Nuclear Power Plant Operators Course will result in award of MOS 358.

13. Training Considerations

- a. Training requirements are as outlined in paragraph 12b above.
- b. To facilitate crew training, a control cab simulator will be required to provide training in certain procedures that cannot be safely practiced using an actual power plant. This simulator will be procured with O&M funds as a training item.

Section VI - Associated Considerations

14. Training Devices: As specified in paragraph 13b above, a control cab simulator will be required for crew training. There is no developmental effort required prior to procurement of this device.

15. Related Materiel: Accountability and responsibility for nuclear power plants will be in accordance with standard supply procedures, except for the special nuclear material. Procedures for procuring and accounting for special nuclear material are defined in AR 700-25.
16. Cover and Deception: The power plant should resemble a standard non-signature item of rolling equipment normally seen in field army areas.
17. Probable Interest: This power plant may fulfill mobile power requirements of the Air Force, Navy and Marine Corps. Although British and Canadian forces may require a special policy for control of the fissionable material, interest is probable.
18. Existence of Developmental Items of Other Services, Armies, or Countries:

None.
19. Communication Security:

Not applicable.
20. Additional Comments:
 - a. The noise level of the operating plant should not exceed 80 db (75 cps) at the exterior of the control cab when the plant is operated in flat, open terrain and is separated from the control cab by 500 feet.
 - b. The use of this nuclear power plant should not increase the vulnerability of an installation through enemy action over the vulnerability associated with a similar installation which uses a conventional power plant.
 - c. For certain applications, the use of this nuclear power plant may require construction of a distribution network to provide power to several users simultaneously.
 - d. Except in time of war or national emergency, the nuclear core will be removed from this plant when it is transported beyond the boundaries of an approved operating area. The movement of the nuclear core and any other radioactive materials or equipment beyond these boundaries will be in accordance with procedures and criteria established by the AEC and other regulatory agencies for transportation of hazardous materials. In time of war or national emergency this plant and associated radioactive materials will be moved as deemed necessary by the responsible military commander with due regard for the public safety.

APPENDIX F

ML-1 PLANT LAYOUT AND P&I DIAGRAMS

Selected drawings showing the ML-1 power plant layout and P&I diagrams are included in this Appendix.

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- NOTES** SEE DRAWING 423242 FOR NUCLEAR CABLE ROUTING ON REACTOR AND POWER CONVERSION SKID.

423600

COPY_____

1	423154	DRYING SKID	43
1	423847	SKID ASSEMBLY, GAS STORAGE	43
1	423571-1	CONTROL CAB ASSEMBLY	43
1	090720-1	POWER CONVERSION SKID	43
1	423601	REACTOR PACKAGE	43

PART NO.		DISPOSITOR		NOTES		BY 80464-11221		DATE	
LIST OF MATERIALS									
DRAWN BY C. W. TOLL		TITLE ML-1 MOBILE NUCLEAR POWER PLANT
CHECKED BY DESIGN				NUCLEONICS SAN RAMON, CALIFORNIA					
PROJECT									
STRESS									
MATERIAL		RELEASE TO	SCALE	CALC BY	DWG. SIZE	DWG. TYPE	DWG. NO.		
PRODUCTION			1/16			E	423600		
CUSTOMER		RELEASE DATE		ACT. BY			COOK NO.	09336	SHEET 7 OF 8

F-6

- P&I LIST
- SHEET
- 1 LEGEND
 - 2 PROCESS GAS SYSTEM
 - 3 MODERATOR & SHIELD WATER SYSTEM
 - 4 DEOXYGENATION SYSTEM
 - 5 LUBE OIL & SEAL GAS SYSTEM
 - 6 AUXILIARY SYSTEMS
 - 7 HYDROCARBON & EFFLUENT MONITORING SYSTEM; GAS SAMPLING SYSTEM
 - 8 REACTOR FLOODING SYSTEM

	MAIN GAS LINE		LUBE OIL LINE		LUBE OIL WITH ENTRAINED GAS LINE		CHECK VALVE		SPEC CATEGORY
	SECONDARY GAS LINE		SHIELD WATER LINE		MODERATOR WATER LINE		FLANGE		PIPE LINE SIZE
	FREON REFRIGERATION LINE		INSTRUMENT SENSOR OUTPUT		THERMOCOUPLE LEAD		FOOT VALVE		STRAINER
	SENSING LINE		CONTROL LINE		DYNAMOMETER COOLING WATER		PIPE PLUG		FILL
	DIFFERENTIAL PRESSURE SENSOR		DIFFERENTIAL PRESSURE SWITCH		RESISTANCE ELEMENT TEMPERATURE SENSOR		RUPTURE DISC		CAP
	CONDUCTIVITY SENSOR		RADIATION DETECTOR		PRESSURE SENSOR		LOUVRE		RO-METER
	CONTROL VALVE		SOLENOID VALVE		EXPLOSIVE VALVE		VENT		FILTER TRAP
	PRESSURE CONTROL VALVE		FLOW REGULATING VALVE		TEMPERATURE CONTROL VALVE		QUICK DISCONNECT		HEATER
	RELIEF VALVE		PRESSURE DIFFERENTIAL CONTROL VALVE		DIST. WATER		D-XXX DEOXYGENATION COLUMN		IE-XXX ION EXCHANGER
	BLOWOUT DISC		SERVICE WATER		HOSE CONNECTION		HX-XXX HEAT EXCHANGER		T-XXX TANK, TRAP OR VESSEL
	THERMISTER OR RESISTANCE BULB SENSING LINE		PUMP		COMPRESSOR		MOTOR		SEPARATOR

INSTRUMENT READOUT LOCATION

EQUIPMENT MOUNTED	CONTROL CAB MOUNTED	ANALYSIS BUILDING MOUNTED	GAUGE BOARD #1 MOUNTED	GAUGE BOARD #2 MOUNTED

INSTRUMENT DESIGNATION			
LTR	1st LETTER	2nd LETTER	3rd LETTER
A	CONDUCTIVITY	ALARM	AMPLIFIER
C	FLOW	CONTROL	CONTROL
F	HAND ACTUATED	INDICATING	HIGH
I	CURRENT (AMPS)	INDICATING	INDICATING
L	LEVEL	LOG	LOW
Lg	MOISTURE	MONITOR	
M	NEUTRON COUNT RATE		
N	NUCLEAR PERIOD		
Np	OVERLOAD		
O	PRESSURE	PERIOD	
P	POSITION	POSITION	
Pn	VIBRATION		
Q	RADIATION	RECORDING	RECORDING
R	SPEED	SAFETY or SWITCH	
S	TEMPERATURE	TRANSMITTER	
T	TORQUE	VALVE	
Tq	VALVE	WIND	
V	WATT (POWER)	PATCH PANEL	PATCH PANEL
W	SCRAM	SCRAM	SCRAM
X	HYDRO CARBON		
Y			
Ch			
d	RATE OF PRESS CHANGE-NEG		
d	SUBSCRIPT REFERRING TO-DIFFERENTIAL		
e	SUBSCRIPT REFERRING TO-ERROR		
B	BLOCK	JACK	
J	CALIBRATION	SPARE	
K	FREQUENCY		
Fr	NUCLEAR		
N	OXYGEN		
O ₂	PRESSURE DIFFERENTIAL		
Pd	COUNT-RATE		
Cr			

PIPING SPECIFICATION CATEGORY					
CODE	MATL	SPEC CATEGORY	PRESS PSIG	TEMP OF	REMARKS
A	STAINLESS STEEL	A1			
		A2	3000	500	
		A3	3000	200	
		A4	300	200	
		A5	300	200	FLEX HOSE
		A6	10	600	INSUL FLEX HOSE
		A7	1000	250	TUBE
		A8	2500	150	
B	ALUMINUM	B1	100	200	
		B2	30	200	FLEX HOSE
C	NON METALLIC	C1	50	150	RUBBER HOSE
		C2			COPPER TUBE

A1 PIPING RATINGS			
LINE NO	PRESS PSIG	TEMP OF	REMARKS
101-10-A1	300	600	INTERNAL INS
102-14-A1	150	500	FURNISH WITH HX
103-10-A1	150	250	
104-8-A1	300	500	FURNISH WITH TC
105-10-A1	300	600	INTERNAL INS

DRAWN: HAMIL, DATE: 1-23-64
CHECKED: [Signature], DATE: 2-10-64
DESIGN: [Signature]
STRESS: [Signature]
DESIGN ACTIVITY APPD: [Signature]
CUSTOMER: [Signature]

09336 J 405710

SCALE: RELEASE DATE: 1-15-68 SHEET: 1

SCHOLIN JR. 0-10-63		Nucleonics		SAN RAMON, CALIF.	
CHECKED	DESIGNED	TITLE			
STRESS	0-10-63	P&I DIAGRAM ML-1			
PRODUCT	2-13-63	AUXILIARY SYSTEMS			
DESIGN ACTIVITY APP'D	09336	CODE	09336	OWS NO.	405710
CUSTOMER	J	RELEASE DATE	SHEET 6

DRAWN/KNOWLES/IO-9-64	AGN AEROSPACE-GENERAL ELECTRONICS
CHECK	
DESIGN	
SYNOPSIS	
DESIGN ACTIVITY APPD	
CUSTOMER	
P & I DIAGRAM ML-1 HYDROCARBON & EFFLUENT MONITORING SYSTEMS	
CODE/REV NO.	09336
DATE	405710
SCALE	
RELEASE DATE	
SHEET	7

DRAWN	KNOWLES	4-9-64	AGN	AEROJET-GENERAL NUCLEONICS	
CHECK			SAN RAMON, CALIFORNIA		
DESIGN			TITLE		
STRESS			P & I DIAGRAM ML-1		
			REACTOR FLOODING AND		
			PURITY CONTROL SYSTEM		
DESIGN ACTIVITY APPD			CODE IDENT NO.	DWG SIZE	DWG NO.
CUSTOMER			09336	C	405710
			SCALE	RELEASE DATE	SHEET 8

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NOTE: This is the 111th 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, 28646, 28647, 28648, 28649 and 28653. The information herein is regarded as preliminary and subject to further checking, verification and analysis.

AEROJET-GENERAL NUCLEONICS