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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

SEMIANNUAL PROGRESS REPORT

1 January through 30 June 1959

31 JULY 1959

Aerojet-General

NUCLEONICS

A SUBSIDIARY OF THE AEROJET-GENERAL CORPORATION

SAN RAMON, CALIFORNIA

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Published 31 July 1959

by

The Military Projects Division
Aerojet-General Nucleonics
San Ramon, California
A Subsidiary of the Aerojet General Corporation

Approved by

G. A. Linenberger
Supervising Representative
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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

SEMIANNUAL PROGRESS REPORT

for

1 January through 30 June 1959

A B S T R A C T

This report covers the work progress of the Army Gas-Cooled Reactor Systems Program. The program includes the Gas-Cooled Reactor Experiments I and II; the ML-1, a portable gas-cooled reactor; and the Gas Turbine Test Facility. This semiannual progress report covers the design criteria, testing programs, data evaluation, and fabrication programs of these projects. The scope of the report includes architect-engineering services, fuel element development and fabrication, materials development, experimental heat transfer, reactor instrumentation, reactor design and development, and nuclear experiments.

PREFACE

The Army Gas-Cooled Reactor Systems Program (AGCRSP) at Aerojet-General Nucleonics includes four projects: The Gas-Cooled Reactor Experiment I (GCRE-I); the Gas-Cooled Reactor Experiment II; the ML-1 (a military low-power reactor); and the Gas Turbine Test Facility (GTTF).

The Gas-Cooled Reactor Experiment I will be a test facility designed to provide data on fuel element lifetime, dynamic reactor behavior and control characteristics. The Gas Turbine Test

Facility will provide operating data on power conversion equipment, specifically the turbine-compressor set. The experience and data gained from these two projects will be used in the design of the ML-1. Thus the attainment of the Army's goal of a portable reactor power plant is being undertaken in a series of logical development stages.

The Gas-Cooled Reactor II will serve as an advanced back-up for the GCRE-I. The major difference between GCRE-I and GCRE-II is that the former is a water-moderated heterogenous reactor, while the latter is graphite moderated and homogenous. GCRE-II is designed to provide more power per pound of reactor, to have longer fuel element life-time and to operate at higher temperatures and pressures than GCRE-I.

The ML-1 is directed toward the construction of a prototype field power generating unit by April 1961. This project includes not only the design of the reactor, shielding, control rods and turbo-machinery set, but also the instrumentation, the shock and vibration mounts, the skids, and the test facilities.

Responsibility for the contract work scope on GTTF was transferred from the Architect-Engineer Division of the Aerojet-General Corporation (AGC) to the Military Projects Division of Aerojet-General Nucleonics (AGN) during the report period. Direct participation by AGN started on the project in January and included direction of all work except that portion of facility construction yet to be completed. The A-E Division, AGC, retained responsibility for this work. (Although the major portion of the GTTF work is not funded under AEC Contract AT-(10-1)-880, a discussion of the activities of the program is included here to lend continuity to this report.)

The organization of this report reflects the four areas of responsibility outlined above. The first section covers GCRE-I, including building and construction, reactor design and development, reactor instrumentation, and fuel element development and fabrication. (The latter section covers only the GCRE-I portion of fuel element development and fabrication - the GCRE-II section has its own program in this area.) The figures and tables for this portion of the report will be found at the end of the section, and for convenience, the numbers are preceded by "I".

The second portion of this report covers GCRE-II, including capsule tests, materials testing and evaluation, capsule design, core preliminary design, in-pile loop tests, and neutronics. The figures and tables for this portion of the report will be found at the end of this section. The numbers are preceded by "II".

The third portion of this report is devoted to ML-1. This includes the auxiliaries, power conversion equipment, instrumentation and controls, and reactor engineering. The figures, tables and appendixes for this section are located at the end of the section and bear the

prefix "III". The final portion of the report is on the Gas Turbine Test Facility, and discusses the status of the work, the San Ramon support operations, delivery of the turbine-compressor set, the open cycle test program, contract modifications, and the spare parts contract.

THE GAS-COOLED REACTOR EXPERIMENT-I

I. INTRODUCTION AND SUMMARY

Construction of the GCRE-I facility at the National Reactor Testing Station (NRTS) was about 92% complete by the end of June. It is expected that the facility will be accepted from the Increment II construction contractor by early September. The operating organization is completely staffed and is engaged in training and the preparation of procedures.

Design and development of the GCRE-I reactor continued throughout the report period. The activity during the latter half of the period was largely confined to liaison with the fabrication vendors and supervision of test activities.

All design, development, procurement, and fabrication were complete at the end of June. Control rod actuators were still being tested. All equipment required for installation by the Increment II contractor had been delivered. Almost all of the equipment to be installed by AGC personnel after completion of construction was packed for shipment to the site.

Thermocouple harnesses were fabricated for the reactor instrumentation during the report period. The fission product monitor system was fabricated; and the instrumentation, stainless steel sampling tubes, gas sampling rack, and the control cable were shipped to Idaho. Equipment for the reactor neutron monitoring (including the pressurized housings, the detector stands, the pressurized hoses, the ion chamber coaxial cables and the control rod extension cables) were assembled, tested and shipped to Idaho. Instrumentation for radiation monitoring of the GCRE-I facility was completed at San Ramon in January. The equipment for monitoring the temperature within the reactor vessel was procured, and the wiring was completed for the console and control racks.

Fuel plate production at M & C Nuclear was completed in June. Before completion, the contract was modified to include one experimental lot of plates containing Mallinckrodt "Spherical UO_2 ". The modification was made to determine the effect substituting this commercial-grade UO_2 for the Geneva-type hydrothermal UO_2 would have on fabrication and forming properties.

Fuel element fabrication and inspection techniques were established during the report period. Fifty-five production elements and four instrumented elements have been completed. An additional 25 production elements are scheduled under this task. Delivery of several key components delayed the schedule at times, but these problems have been overcome.

Two I-3P test elements were fabricated and shipped to the Engineering Test Reactor (ETR) site in April for irradiation in the BMI-16 loop. Flux measurements were conducted to determine the flux distribution and the neutron shield thickness required to adjust the neutron flux level to the test element. Installation of the BMI-16 loop was delayed by the installation of other loops at the ETR.

Fourteen dummy elements with voids were fabricated for use in the GCRE-I core. These void plugs will block extra pressure tubes to restrict flow and heat losses.

A major change in the fuel element program was made in mid-June due to a reduction in the ML-1 system weight. The change shifted the emphasis to a modified IB fuel element and in effect cancelled the IA element program. Reducing the permissible weight for the ML-1 reactor package involved a change to a 24-in. long core, and the change in length excluded the use of an IA dispersion-type element. Before the IA program was phased out in June, the effort on that program was placed on developing a method for attaching spacers to the edges of fuel plates. Arc welding was the most successful method.

This program change shifted the IB element design emphasis to a shorter element. The IB element will be designed for the second core loading of GCRE-I and the first core of the ML-1. The reference element for the second core loading of GCRE-I then, will be a 19-pin element with 22 in. of fuel length (allowing 2 in. for a reflector in the core).

Fabrication of the IB-1T in-pile elements was the major effort in the IB fabrication development program during the report period. Facilities to handle pellets were established; a method was found to differentiate between various pellet enrichments; the brazing problems were defined for the attachment of bearing spacers, and other problems associated with the assembly of the pins were solved. Instrumentation techniques were developed to measure the temperature of the inner wall of the IB-1T element. Design and fabrication of a spacer with lower flow friction continues. Nickel-based commercial alloys are being studied through gas corrosion, creep rupture and tube burst tests. Hastelloy-X was chosen as a reference material for the IB-1L core after studying the preliminary data from these tests. The IB-1 α T test irradiation began late in June, and the IB-1 ϕ T and IB-1 β T tests are planned to follow this initial test.

UN-stainless steel plates were fabricated in the program at Battelle Memorial Institute. The capsule containing UN-stainless steel coupons began irradiation in April, and completed irradiation on 24 June 1959. Two capsules containing solid UO₂ pin-type specimens were fabricated, and started irradiation in March and April. One capsule was removed 24 June to provide early data and the other capsule is continuing irradiation to a higher burn-up.

II. FACILITIES CONSTRUCTION AND OPERATION

A. ARCHITECT-ENGINEER SERVICES

The A/E Division of Aerojet-General Corporation (AGC) at Covina completed evaluation and approval of all equipment specifications submitted by the construction contractors. This group also revised the A/E drawings to reflect the "as built" status of the facility and, at the request of IDO, performed expediting and quality control inspections in the shops of the construction contractor's vendors.

A field inspection staff of three engineers and a clerk followed the construction work at the site throughout the period. This group insured compliance with plans and specifications, handled correspondence between the USAEC and the construction contractor, and maintained information on the status of the project for the Commission. An instrument inspector was added to the staff for one month to assist during the peak load.

B. OPERATIONS SECTION

1. Organization and Personnel

The operating organization for the GCRE-I facility (consisting of 11 professional or technical employees and 17 non-professional personnel) was completely staffed during the report period. As of 30 June 1959, 26 members of the group were assigned at the site.

2. Operating Procedures

The operating group concentrated on preparing operating procedures during the report period. At the end of the period, the status of the procedures was as follows:

<u>TITLE</u>	<u>ESTIMATED % COMPLETED</u>	<u>NUMBER WRITTEN</u>	<u>NUMBER APPROVED</u>
Technical Manual (ANTM)	65	--	--
Technical Standards (ANTS)	95	11	10
Standard Operating Procedures (ANSOP)			
Section 0 - Office Procedure	20	19	2
Section 1 - Accountability	10	1	1
Section 2 - Radiological Safety	40	7	4
Section 3 - Non-Radiological Safety	20	4	0
Section 4 - Security	0	0	0
Section 5 - Reactor Operation	85	140	68
Section 6 - Auxiliary Operation	40	31	0
Section 7 - Mechanical Maintenance	80	10	2
Section 8 - Instrument Maintenance	70	12	0
Section 9 - Experimental Operation	60	42	2
Standard Operating Logs (ANSOL)	60	24	--

3. Training

The personnel assigned at San Ramon participated in a formal training program during May. The program consisted of courses in nine subjects and included fifty-eight hours of classroom lectures and discussions. The program is being repeated at NRTS for the personnel who joined the organization at the site.

III. REACTOR DESIGN AND DEVELOPMENT

A. CORE DESIGN AND FABRICATION (Task 32-80)

1. Reactor Pressure Vessel

The reactor pressure vessel was delivered to San Ramon in January. The vessel was built to conform with Section VIII of the ASME Boiler and Pressure Vessel Code and was stamped accordingly. Tests were made in February to check the vessel fit-up with the upper and lower plenums, plenum insulation, fuel element support assembly, fast neutron reflector and the reactor stand assembly. All assemblies fit properly. The fuel element storage rack was also completed and inspected. The above units, with the exception of the reflector, were all shipped to Idaho for installation.

The vessel was pressure-tested at San Ramon and two deficiencies were noted: The stainless steel bolts at the main flanged joints in the reactor galled excessively. New bolts were procured that eliminate this problem and insure adequate gasket-seating forces. Leaks were detected in the gas sampling tubes near the exit point from the instrumentation flange, and a pressurized back-up system was designed to correct this situation. The new bolts and the backup system will be installed by Aerojet operating personnel after construction is complete.

2. Inlet and Exhaust Ducts

Study of the differential expansion of the inlet and exhaust ducts revealed that a significant lateral shearing force was induced on the reactor pressure tubes. A duct stiffener was designed and fabricated to reduce these forces to a negligible value. The duct stiffener will be installed by the operating crew.

3. Fast Neutron Reflector and Moderator Cooling Systems

The reflector and cooling systems were fabricated and inspected, and were packed for shipment to Idaho for installation by the operating crews.

No satisfactory substitute has been found for the steel-aluminum silicone coating now used to seal the lead reflector. Any cuts or tears resulting from field handling of the lead will be repaired by the "Metallizing" process used in original fabrication.

4. Fuel Element Handling Tools

The handling tools were conceived, designed and about 50% fabricated by the end of June. No difficulties are anticipated in completing this task.

5. Miscellaneous Units

The dummy fuel elements, source holders, instrumentation cable trays, and covers for the fuel element storage rack were completed and are ready for shipment to Idaho.

(Note: This is the final report to be submitted on Task 32-80. All work is completed).

B. CONTROL ROD DESIGN AND FABRICATION (Task 32-81)

All control rod assemblies were completed during the report period and all component tests initiated. The shim, setback and fine rod mechanisms were shipped to Idaho for installation. Tests on the remaining units will continue until the units are required for installation at Idaho.

1. Control Blades

These units were fabricated and tested by the end of June. As a result of tests, the safety and shutdown blades were modified by grooving the bottom edge to insure better guiding action. The shim, fine and setback blades were reinforced by the addition of longitudinal, riveted, stainless steel strips.

2. Latches

Tests revealed that the action of a safety blade could cause the latch to open, separating the blade from the mechanism. This was corrected by grinding a 5° rack angle on the mating portions of the latch. Tests are continuing.

3. Guide Assemblies

Tests revealed that friction drag of the safety and shutdown control blades in the guides was excessive. The guide was modified and seven "V" wheels were set in the bottom track and four flat rollers in the top track. The modified guide assembly is now performing satisfactorily.

4. Safety Mechanisms

More than 400 test cycles have been logged on the safety mechanisms. The latch release mechanism and the fluid dashpot perform properly. The "V" packing became sticky after prolonged exposure to water; saturating the packing with silicone oil "D" before installation reduced the stickiness. Tests of alternate packings are scheduled for early July. A higher force scram spring was being installed at the end of the period, and will be tested.

5. Shutdown Mechanism

All shutdown mechanisms were being tested at the end of the report period. Over four hundred test cycles had been completed on the five mechanisms remaining at San Ramon.

A new scram spring was designed and installed which is strong enough to provide a scram in less than 0.5 seconds without being too strong for the magnet to hold in the compressed position.

IV. REACTOR INSTRUMENTATION

A. REACTOR TEMPERATURE MEASUREMENTS (Task 32-70)

Thermocouple harnesses for both the upper and lower plenums were fabricated during the report period. The lower plenum thermocouple harnesses were attached to the reactor before it was shipped to Idaho. The thermocouple extension leads were shipped to Idaho in March. The leak at the 52 pin plug on the lower thermocouple harness was rectified by attaching the plug to the Marman flange.

The thermocouple extension harnesses for the instrumented fuel element were fabricated and will be shipped to Idaho for installation.

(Note: All work on Task 32-70 has been completed and no further report will be made).

B. FISSION PRODUCT MONITOR (Task 32-71)

The fission product monitor system was fabricated during the report period. The instrumentation, stainless steel sampling tubes, gas sampling rack and the control cable were shipped to Idaho.

Cobalt 60 (1.17 mev and 1.33 mev) and cesium 137 (0.662 mev) were used to calibrate the fission product detector head.

(Note: All work on Task 32-71 has been completed and this is the final report).

C. REACTOR NEUTRON MONITORING (Task 32-72)

The neutron detector stands and housing were fabricated during the report period. The pressurized housings, the detector stands, the pressurized hoses, the ion chamber coaxial cables and the control rod extension cables were assembled, tested and shipped to Idaho.

(Note: Work on Task 32-72 is complete and this is the final report).

D. FACILITY MONITOR SYSTEM (Task 32-72)

This task covered instrumentation for radiation monitoring for the GCRE-I facility at IDO. All work was completed at San Ramon in January. The equipment will be shipped to Idaho for installation.

E. DATA REDUCTION AND TRANSIENT INSTRUMENTATION (Task 32-74)

This task provided equipment for temperature monitoring within the reactor vessel. All instrumentation was procured and the wiring completed for the console and control racks. The equipment was delivered to Idaho for installation by the construction contractor.

F. CONTROL CONSOLE AND PANEL (Task 32-76)

All major work on the control console and racks was completed before shipment to Idaho in January for installation by the construction contractor.

A period test circuit was designed and fabricated for use in testing the period circuits in the neutron detection system. The circuit was tested and packaged for shipment to Idaho for use by the operating crew. The mixed mean gas temperature recorder (R9) was packaged for shipment to Idaho for installation.

(Note: Work has been completed on Task 32-76 and this is the final report).

V. FUEL ELEMENT DEVELOPMENT AND FABRICATION

A. FUEL ELEMENT I

1. Fuel Plate Manufacture at M & C Nuclear (Task 34-34)

The final shipment of two additional lots of type "C" plates completed production. Before completion, however, the contract was modified to include one experimental lot of plates containing Mallinckrodt "Spherical UO_2 ", designated Phase IV of the fuel plate program. Phase IV is scheduled to be completed 21 July.

The total number of plates delivered in Phase III was 300 - "A"; 308 - "B"; 331 - "C"; and 338 - "D". Production of plates was accelerated in March and April and the program was more than 90% complete by the end of April. Production of final lots was slower because several small batches were initiated after the yield was established on the majority of plates. The production yields obtained on the four sizes differed from 70 to 93%. In detail, the Type "A" yield was 76%; Type "B", 74%; Type "C", 70%; and Type "D", 93%. (These figures represent only the production of flat plates at M & C, and do not take into account rejection of plates after further processing).

The yields were lower than assumed from several causes: the errors in transferring the X-ray images and inscribing the plates for machining resulted in less-than-specified side closure (about 0.010 to 0.015 in. less inert margin); a number of plates were rejected for surface defects (scratches and dents); and a few plates were pitted to the point where uranium leakage was evident in the boiling nitric acid test.

The serious camber problem inherent in the GCRE design was solved by M & C with their de-cambering technique, a technique that undoubtedly saved hundreds of plates.

A later problem developed after forming and caused the rejection of more than 20 Type "C" plates. This condition was first noted as a localized wrinkled area on the cladding. In many cases, the boiling HNO_3 test revealed exposed UO_2 . Preliminary metallography revealed large areas of what appeared to be carbon pickup or some other contamination originating at the bond between the cladding and the core.

There were no adequate data at the close of the report period on the cause of this problem. M & C will further investigate the defects. About 40 extra Type "C" plates were made up during production to avoid a shortage of this size.

2. Fuel Element Manufacture at Azusa (Task 34-41)

Fuel element fabrication and inspection methods were established during this report period, and the methods were scrutinized during fabrication of five Production Mock-up Units (PMU) and one Production Mock-up Unit-Instrumented (PMUI) before production of the final elements. The mock-up units were further used for thermal cycling, stress relieving, and core insertion tests. After fabrication of the mock-up units, four instrumented elements and 55 production elements were completely assembled by the end of June. Assembly of 25 additional production elements will complete this task. The production rate of these elements was paced by delivery of several key components, including the fuel plates, zirconium tube, and spacer clips.

Clip production was delayed by the vendor's difficulty in meeting the quality specification. Two steps were taken to solve this problem: the vendor was given technical assistance, and two back-up approaches were explored. By the latter part of May, the quality level of the clips was improved so that the back-up programs were cancelled. (The vendor used an intermediate heat treating operation and expanded production through a second shift).

All the plates had been processed through forming and heat treating by the end of June. About 25% of the Type "C" plates had below-standard edge margins (0.020 to 0.025 instead of 0.030 in.). These plates were held for the last portion of the assembly. M & C Nuclear's examination of the edge closure showed that radiographic definition of the edge is accurate within 0.005 in., insuring that where the margin is defined by radiograph as being 0.020 in., the plate will have a minimum of 0.015 in. of margin. These plates were used for assembly of production elements, and are identified by special production serial numbers.

Previous reports have mentioned the decline in plate quality in later shipments. At present, 8 to 10% of the "A", "B" and "D" plates, and about 20% of the "C" plates shipped to Azusa are rejected. Chief causes of rejection are minute cracks in wrinkled cladding areas, deep pits, holes, nicks and scratches. The cracks and holes are readily picked up by the acid etch inspection. Some laps in cladding were found, but the incidence was small. These laps open up or chip off when the plate is formed.

One explanation offered by M & C for the high incidence of "C" plate rejection (due to clad wrinkling and cracking) was that the clad was put on with its direction of rolling 90° to the final plate rolling direction. Stringers in the clad material also contributed to the wrinkling. Figures I-1 through I-3 illustrate this defect. Figure I-1 shows a typical wrinkled area. This was cut parallel to the wrinkle, and a microphotograph was made of the region under the clad in the un-

wrinkled area. Figure I-2 shows that a lens-shaped area of different microstructure exists under the clad. Figure I-3 is an area showing a crack detected in a similar defect by acid etch inspection. Comparative microhardness tests show that the defect area is 10% harder than the matrix. It appears to be a carburized area produced by contamination of the original compact.

Fifty-five Production Core Units (PCU) were completed by the end of June. Twelve of these have stainless steel cans, the others have zirconium cans. The 80 units planned will all have AISI Type-318 stainless steel clips and burnable poison foils.

The hand fit-up required for each unit makes assembly tedious. Eighty-two zirconium cans have been received - only 54 meet the original specifications. Five cans are one to two inches too short and cannot be used, and the other 23 are below specifications as much as 0.1 in., but can be used.

All of the orifice blanks have been machined to the proper orifice size, and the exact orifice diameter has been etched on the lip to assist in assembly at the reactor site. The orifices will be shipped separately from the units for later installation.

The four Production Instrumented Units (PIU) have been completed and each unit given a leak and pressure test to assure its integrity. An actual reactor seal flange and Weatherhead seal was used. Twenty-psi Freon-12 pressure was applied and a GE Type H-1 Halide Leak Detector used. The leakage at all points was less than 1.0×10^{-8} cc/sec at 20 psi. Each thermocouple was checked, and the actual resistance was recorded for the two wires and each wire to ground after each unit was placed in its shipping crate. About 5% of the thermocouples had erratic resistance readings compared to the original as-received readings. Two percent of the thermocouples were lost in fabrication.

3. I-3P In-pile Test (Task 23-95)

The in-pile loop, designated BMI-16, was designed to evaluate the performance of the GCRE production fuel elements. The first element to be tested, designated I-3P by Aerojet, is a GCRE production element modified for installation in the loop. The loop will be placed in the ETR and experiments will begin during the first half of FY 1960. BMI personnel designed and built the BMI-16 loop; however, initial operation of the loop will be a joint venture between BMI and Aerojet.

a. Objective

The objective of the I-3P experiment is to obtain engineering data on the GCRE first-core element with the test element operating at the maximum power and coolant flow rate found in the GCRE. The main interests in this test are corrosion, heat transfer, instrumentation, mechanical distortion, gaseous diffusion, vibration and radiation damage.

b. Test Conditions

The test conditions were specified to simulate the maximum conditions found in the GCRE. The normal thermal neutron flux in the loop position in the ETR was high by a factor of about three. Tests made in the ETR critical facility indicated that a silver neutron shield 0.092 in. thick would reduce the thermal flux in the test element to the desired value. Power generation in the test element increases by about 70% during the fuel cycle of the ETR due to a neutron flux increase. The initial and final anticipated power generation has been specified, so the average fission-plus-gamma energy produced over the 21-day cycle is 45 kw.

Test parameters will be controlled initially throughout the test element by varying the coolant inlet temperature to maintain constant fuel plate temperature. This method of control will be used for the first cycle of irradiation, and then the method of control will be changed so that the element outlet gas temperature is held constant by varying the coolant flow rate through the element while holding the gas inlet temperature constant.

The test conditions are shown in Tables I-1, I-2, and I-3. Table I-1 shows the nuclear conditions; Table I-2, the thermodynamic conditions; and Table I-3, the anticipated power generation in the test element.

c. I-3P Element

The loaded portion of the I-3P element is identical to the GCRE element. The assembly drawing of the I-3P element is shown in Figure IV-17 IDO-28533. The main difference in the two elements is in the end sections where the element was modified for insertion into the BMI-16 loop. This is the re-entry-type loop where the hot gases pass down the outside of the element and enter the element from the bottom; thus the gas flow through the element is reversed from that of the GCRE. The tail section of the I-3P element was designed so that the gaps between the plates can be measured before disassembly. This assures that any mechanical distortion measured after irradiation will be due to reactor environment rather than to displacement during hot-cell handling.

d. Instrumentation

The locations of the 25 thermocouples used in the I-3P element are in about the same relation to element temperatures as they occur in the GCRE elements. Thermocouples 9 and 15 measure the mixed-mean inlet gas temperature; thermocouples 1 and 16 measure the mixed-mean exit gas temperature; thermocouples 10, 11, 12, 13, 14 and 17 measure the exit gas temperature from each channel; and thermocouples 3 and 18 through 25 measure material temperatures at the calculated element hot spot. Axial temperature variation is measured by thermocouples 2 through 8 located in the center support spine. These thermocouples, with the exception of 3, directly face the reactor core. Thermocouple 3 is rotated 30° clockwise from the other axial thermocouples.

The in-pile loop is instrumented to measure coolant flow and pressure. Activity of the gas will be recorded to determine if fission product diffusion or leakage will be problems in the GCRE.

e. Status

Two fuel elements were manufactured at Aerojet and shipped to the ETR 15 April 1959. The coolant channel gaps were measured and the thermocouples checked for continuity after their arrival at NRTS. Flux measurements were conducted in the ETRC to determine flux distribution and the thickness of neutron shield needed to adjust the neutron flux to the test element.

A trial insertion of one unit into the in-pile leg was completed without incident in June.

The loop equipment installation at the ETR has been completed except for in-reactor piping. The loop is now undergoing a cold check-out during which time the limiting parameters will be set and all safety circuits and components checked.

The loop hazard report was prepared by BMI and approved by the ETR Safeguards Committee. The I-3P element hazard report prepared by Aerojet was submitted to the ETR Safeguards Committee.

4. Dummy Element (Task 23-62)

The GCRE-I Reactor assembly is made to accommodate up to 73 fuel elements. Since the plan is to operate with 61 elements, the extra pressure tubes will be blocked with dummy elements to restrict the flow and heat losses. Two types were designed, one with an inner void, and the other with a graphite center. Only the void plugs (dummy elements) were fabricated because critical assembly tests indicated that graphite plugs would not be needed. Fabrication of these void plugs was completed in May 1959.

B. FUEL ELEMENT IA

1. IA Engineering (Task 23-90)

Because it was decided to use the IB (pin-type) fuel element for the first ML-1 core, the IA program is being phased out. During the report period, the following methods of integral flanging were pursued: shaper flange, resistance welding, forging, compression bending, puddle welding, and tungsten-inert gas welding. The last method proved to be the most promising, and all others except resistance welding were dropped. Resistance welding was investigated on a limited basis.

A capsule containing three dispersion plate coupons was inserted in the Materials Test Reactor (MTR). The capsule was removed after four cycles of irradiation (about 1500 hr) and shipped to BMI for post-irradiation examination. (See Task 23-53).

2. IA Fabrication Development (Task 23-94)

a. Shaper Flange

This method used a shaper to split the edge of a fuel plate to form a continuous spacer projecting from each edge along the full length of both sides of the plate. Two "C"-size inert stainless steel plates were formed oversize and flanged with the shaper. Mechanical difficulties placed emphasis on operator skill, but none of the stainless steel pieces were satisfactorily flanged.

Similar flanging tests were made on natural uranium nitride and oxide plates from material supplied by EML. Metallurgical examination revealed excessive fine carbide precipitation and sigma phase formation in the plates. This contributed to tool wear and failure through non-uniformity and hardness. Harder tools, such as the carbides, did not help because of their brittleness. Other machinery difficulties slowed the development of shaper technique.

Only three out of seven plates flanged by this method were usable for the full length, and the method took forty hours a plate with the available equipment. Special tooling probably would improve the quality and reduce production time, but the time required for preparing each plate might be reduced only to about twenty hours. Even under these conditions it is expected that the cost of the shaper method would be high. The shaper flange tests were discontinued in February 1959 because of the relative inefficiency of the method.

b. Resistance Welding

This method seam-welded a flange to the edges of plates. Preliminary welding parameters were established by one vendor on twenty-five strip samples 8 in. long and on four pieces 28 in. long, using holding fixtures developed by Aerojet. The study showed that only low-strength stick welds could be produced.

c. Forging

In this method, the edge of the fuel plate is locally heated to 1500°F and then forged in a die to the shape of a spacer. Longitudinal cracks 5/16 in. wide developed in forming flanges, and the edges were irregular. The method was dropped.

d. Compression Bending

This technique used tabs bent alternately from the edges of the plate. A small die was made for forming alternate tabs, and samples were produced with 1000-psi hydraulic pressure. The disadvantage of this method is the cost of dies.

e. Puddle Welding

This process involves placing a plate so that the edges project up into a fixture where a Heliarc "stringer" is used to puddle metal into a trough, casting a flange on the fuel plates. The problem with this technique is in building the fixtures out of non-metallics (such as ceramics) to prevent contamination.

f. Tungsten-inert Gas Welding

The technique fuses together a vertical section of a "T" 0.045 in. thick and a horizontal flange 0.015 in. thick by arc-welding over the top of the flange. The initial efforts in this direction using Heliarc welding head with automatic arc control were successful.

Fixtures for welding, milling and heat treating "C" size plates were designed and manufactured. Successful welds were made on flat, inert stainless steel as well as on flat, natural UO_2 -loaded plates. Acid etch tests and photomicrographs showed that this method would be satisfactory. One size "C" ring was completed by welding curved AISI Type-318 stainless steel plates. The ring was annealed at 1700°F in an argon atmosphere after welding.

Inspection after heat treating showed maximum camber of 0.062 in. which could readily be removed. Tear resistance tests showed that the strength of the flange was adequate.

The following work will be done to bring this program to a logical conclusion: Design of welding, milling, and heat treating fixtures already under way, for all sizes of rings, will be completed and preliminary thermal cycling tests will be performed on a welded specimen.

3. IA-1T In-pile Test (Task 23-96)

This test was intended to evaluate the in-pile performance of a GCRE-I element using integral flanges and UN core material. A GCRE element design was modified to fit the Battelle Research Reactor (BRR) gas loop. Inert parts for two test elements were manufactured by Aerojet and were stored until a successful integral flange was developed for the fuel plates. No further work will be devoted to this task in view of the decision to drop the IA element.

C. FUEL ELEMENT IB

1. Engineering (Task 23-91)

The recent decision to shorten the length of the ML-1 core resulted in a significant re-orientation of the IB program. The second core loading for GCRE-I has been designated IB-1L element. This is a 19-pin fuel element, 22 in. long, with ML-1 axial power distribution. The fabrication schedule requires that the basic design characteristics of the element be defined in July 1959.

Finned heat-transfer surfaces cannot be used for this element because of schedule limitations. Smooth pins will be used, and excessive hotspot temperatures will be nullified by placing the burnable poison properly. The heat-transfer data from the IB-1dT in-pile test will be used to predict heat transfer coefficients for gaseous flow parallel to clusters of pins. The problem of optimum temperatures for the IB-1L is being attacked by using a heat transfer machine code in conjunction with flux information from the machine-coded P-3 approximation to transport theory.

If optimum temperatures are to be achieved with pins containing partially enriched pellets, the large flux depression in the fuel predicted by P-3 runs requires that the middle ring and central pin of each fuel element be loaded with highly enriched fuel, and the outer ring be kept at a lower enrichment. Large loadings of U-235 in pellet form are required for the IB-1L for the following reasons:

- a. The increased leakage from the shortened core increases the required k_{∞} .
- b. The resonance escape probability decreases as the amount of U-238 increases.
- c. The U-238 is a significant parasitic absorber of thermal neutrons.
- d. Loading highly enriched fuel into the inner fuel rings, where the flux is low, is basically an inefficient use of fissionable material.

Pellets of UO_2 have been the reference fuel to date, but an alternate approach is to use an annulus of highly enriched UO_2 in each pin. Possible fabrication procedures to accomplish this are currently being evaluated.

Hastelloy X has been selected as the reference pin material because of its superior creep strength and oxidation resistance in the vicinity of 1700°F compared to Inconel-X or other nickel base alloys. These, in turn, are superior to stainless steels. The fabrication schedule may dictate the acceptance of the standard cobalt impurity in the Hastelloy X for IB-1L. However, the adverse effects of cobalt impurity on radioactive contamination of the coolant, and on reactivity, make it desirable to use special alloy heats for future core loadings.

The shortened core for the IB-1L increases power density and thus accentuates the heat removal problem. This problem is solved by using larger diameter pins and higher flow rates to increase heat-transfer area and coefficients.

The prediction of fuel element lifetime is intimately associated with determining allowable stress in the fuel pin. The temperature, materials, and sizes being considered from pin fuel elements make it impossible to follow standard engineering practice. The ASME code criteria

have been used to determine the limits of the allowable stress (at the temperatures of interest, the allowable stress is always limited by creep) and then the Larson-Miller technique has been used to extrapolate creep test data to the times and temperatures of interest. A safety factor is then calculated based on the ratio of allowable stress to actual stress. Stresses resulting from thermal gradients have not been considered and are believed to relieve themselves by minor plastic flow. The time dependence of the actual stress has not been considered because of the complexity of the problem; but rather it is assumed that the final stress has existed throughout the period of exposure. Such a procedure is conservative, since the actual stress continually increases with time because of the production and release of gaseous fission products during operation.

A limited literature study was made of methods of loading fuel material into pins during the development of the pin-type fuel element. The more highly developed technology of UO_2 makes it the favored fuel. Uranium carbide and nitride are potentially better fuels because of their greater thermal conductivities and densities; but they are essentially still in the development phase. The lifetime of UO_2 pellets can be improved by adding metal slivers to the powders before compacting; this improves their thermal conductivity.

The production of fuel pellets is commercially established, but the large number of steps in the process and the close tolerances required result in higher costs than for other loading processes. An alternative method to give an annular fuel ring is known. This process has the advantages of low fuel temperatures and fewer fabrication steps, but does allow the release of more of the fission product gases from the ceramic.

Three other processes are currently in the development stage at other sites and show promise of reduced fabrication costs. These are isostatic pressing, the swaging of powder-filled tubes, and the extrusion of fuel cylinders. The main problem is to obtain high densities with these fabrication methods since low densities generally are indicative of high gas release and, thus, short lifetimes.

A parametric study was undertaken for the 28-in. core under ML-1 operating conditions. Interrelations were established between maximum pin surface temperature, element pressure drop, pin size, and the amount of heat-transfer surface. These relationships lead to the following systems, all of which satisfy the pressure drop and temperature criteria: Cluster of 37 smooth tubes, 0.15 in. OD; cluster of 19 smooth tubes, 0.23 in. OD; cluster of 19 finned tubes, 0.19 in. OD; and cluster of 7 finned tubes, 0.38 in. OD. Further refinements will be required to determine lifetime as a function of tube stress resulting from fission gas pressure buildup.

The parametric study indicated that a finned, seven-pin element would meet the design criteria and could decrease fabrication costs compared to an unfinned, 19-pin cluster. If pellets are loaded in these pins, they will require a large central hole to keep the UO_2 at acceptable temperature.

2. Fabrication Development (Task 23-93)

This task, during the report period, was chiefly concerned with methods for fabricating in-pile instrumented fuel elements, such as the IB-1T series. The IB-1ØT in-pile element was made with pellets, and all the instrumentation was placed outside the fuel pins. The IB-1βT element is similar, but used finned tubing. The β element has thermocouples entering the pins at their tops and threading down through holes in the pellets. The thermocouples terminate with a junction in a metal disc press-fitted to the metal tube of the pin, thereby giving an average reading of tube temperature.

Facilities were established for handling pellets: vented hood boxes for inspecting, weighing and calibrating; a weld box for welding in a helium atmosphere; and test vessels for leak checking fuel pins with a helium leak detector.

A test was run to ascertain if it is possible to differentiate between various enrichments by measuring gamma emission. The test proved that relatively standard equipment could differentiate enrichments. A method for differentiating between pellets with 25% and 50% enrichment has been established.

The completed pins are supported on a wire slipped through two holes in the head of the pin and bearing on the tube sheet. (A single pin is shown in Figure I-6, IDO-28538.) One end of the wire is bent to retain it, and the pin head has a shoulder to keep the pin from rising. Thermal cycle tests showed that this arrangement was strong enough.

The study of problems of assembly of instrumented elements was carried out with models. Assembly procedures, routing and joining of thermocouples, and mating of parts were techniques established by trial assemblies of the instrumented fuel element; and skills of personnel also were established for this work. The techniques developed to join the 5-mil thermocouple wires to the lead wires are illustrated in Figure I-4. A small "C" shaped tool supports one end of the thermocouple and one end of the lead wire to prevent relative movement at the joint. The 5-mil wire is spot welded to the lead wire, and then both are encased in plastic.

The method of obtaining temperature readings in the element (mentioned above) is desirable because the thermocouples are located inside the pins and do not interfere with gas flow. The disc, to which the thermocouple is attached, is insulated on both sides by 0.020 in. zirconia discs. According to test data, the mean wall temperatures of the pin would be 8°F below true mean temperature in an in-pile test if the same disc-to-pin contact has been made as in the laboratory. Even if the contact resistance were increased by 400% the disc would only read 11°F high. A tool is being developed to swage out the disc to insure positive contact with the pin wall.

The IB-1ØT element, a nuclear mock-up for the IB-1T series, will be used for flux and power measurements in a critical experiment and in the Battelle Research Reactor in-pile loop. Five pins in this element have removable short central sections which can be used to count

fission product activity after an irradiation. Methods were developed to seal fuel pellets in these central sections. A capsule sealed by filling one end with Epon resin and cementing a 0.003-in. aluminum foil to the other end is shown in Figure I-5. The tube attached to the capsule was used to fill it with helium when checking for leaks in the diaphragm with the helium leak detector. Methods were also developed for placing monitor wires inside the pins by using a thimble of aluminum tubing 0.040 in. dia. with 0.005 in. thick walls. One end was sealed and threaded through the pellets and the other end flared over and set in a seal in the head of the fuel pin. This arrangement forms a well in which a manganese-copper monitor wire can be inserted and withdrawn from within the fuel pins without breaking the metal seals.

Three elements of the IB-1R type will be built and inserted in the GCRE-I reactor at Idaho. The outer shell is the same as that used on the plate-type element except that the inner liner becomes the main support member in place of the central spine. There are 19 fuel pins of 0.225 in. OD with a fuel length of 28 in. The fuel is UO_2 pellets of 25 wt % enrichment in the outer ring of pins and 52.4 wt % enrichment in all other pins. The pins are supported from the top by an open, two-level spider and are spaced by strut airfoils along their length. These airfoils are cast integrally with a sleeve which is later machined to slip over the pins. The sleeve is then welded or brazed to the pins. The pins are free to expand downward on the lower spider spacer pins. There is an expansion chamber of 5 in. above the fuel to permit build-up of pressure from gas released from the pellets.

Deflection in fuel pins is being studied to see if it is possible to reduce the number of spacers. The movement of the fuel pins and lowering of pressure drop are the main problems. The proposed integral airfoil is shown in Figure I-6. It is first investment-cast as a rod with airfoils, then the rod is drilled, and the foils are machined to size.

3. Materials Development (Task 23-65)

This task is concerned with the selection of a suitable cladding material for the GCRE-IB pin-type fuel element. Little is known about the behavior of materials under the conditions of high operating temperatures, hot spot temperatures of 1750°F and the desired 10,000-hr operating life. This task has the assignment of developing gas corrosion and creep-strength data on commercial nickel base alloys selected as the most readily available materials for this application. Alloys selected for initial investigation were Inconel, Inconel X, Inconel 702, Hastelloy X, Hastelloy R-235 and Inor 8. Carpenter 20 Cb and AISI Type-318 were tested for comparison purposes.

a. Gas Corrosion Tests

A preliminary screening test was completed in January on the above alloys exposed to reference GCRE conditions for 1000 hr. The test conditions included 1750°F at 200 psi in an atmosphere of nitrogen plus 0.5% oxygen with a -30°F dewpoint. The weight changes observed on these specimens after a 1000-hr exposure were:

The reference alloys selected, based on these corrosion and creep tests, were Hastelloy X and Inconel 702, in that order.

After this inspection, the alloys were exposed to reference test conditions for 2500 hr. At the end of this period, specimens of Inconel 702, Hastelloy X and Hastelloy R-235 were removed from the frames, weighed and examined. Weight changes are tabulated below:

WEIGHT CHANGES ON NICKEL ALLOYS AFTER 2500 HOURS

<u>Alloy</u>	<u>Weight Change (grams)</u>	<u>Weight Change g/cm²</u>
Hastelloy X 1	+0.0535	+0.0046
2	+0.0153	+0.0013
Inconel 702 1	+0.0853	+0.0073
2	+0.1010	+0.0087
Hastelloy R-235 1	+0.1787	+0.0154

Metallographic examination of these specimens yielded the microphotographs shown in Figures I-3, 9 and 10. A zone in which precipitation occurred can be seen on all three specimens. Intergranular corrosion proceeded 0.003 in. into the Hastelloy X specimens, while precipitation progressed 0.006 in. Inconel 702 showed a smaller precipitation zone (0.003 in.) and no intergranular corrosion had started.

Hastelloy R-235 showed a zone of precipitation about 0.009 in. wide with intergranular corrosion sites.

The 0.006 in. zone on Hastelloy X was not particularly brittle, nor did it spall off even when subjected to severe bending. A test performed on a Hastelloy X specimen compared favorably with tensile tests on a reference, unexposed specimen.

MECHANICAL PROPERTIES OF HASTELLOY X (ROOM TEMPERATURE)

	<u>Reference</u>	<u>Exposed Specimen</u>
Yield Strength (psi)*	66,060	75,980
Tensile Strength (psi)	110,550	107,350
Elongation, %	45.0	22.0

Note: *0.2% Offset

Exposure apparently reduced the elongation of this material significantly with a consequent increase in yield strength. The elongation, while reduced, is still considerable and the material remains useful.

After removing these specimens, new ones were added and tests started for a duration of 5,000 hours.

Concurrent with the reference gas corrosion test, a 1000-hr screening test started in which air at 300 psi was the test atmosphere; test temperatures and dewpoints were the same as for the nitrogen test. The same alloys were exposed in this test. Weight changes on these alloys are tabulated below:

WEIGHT CHANGES ON NICKEL-BASE ALLOYS AFTER
1000-HR EXPOSURE IN AIR AT 300 PSI, 1750°F

Alloy	Weight Change (grams)	Weight Change g/cm ²
Inconel	+0.1003	+0.0087
Inconel X	+0.1578	+0.0136
Inconel 702	+0.1292	+0.0148
Hastelloy X	+0.0809	+0.0069
Hastelloy R-235	+0.1336	+0.0115
Carpenter 20 Cb	(+0.2829	+0.0246
	(-0.3075	-0.0267
Inor 8	(-0.7674	-0.0661
	(-0.7594	-0.0654

Metallographic analyses are being performed on all specimens.

b. Creep Rupture Tests

Creep screening tests were performed on several nickel-base alloys at 1750°F at various stress levels. Three alloys were selected for further tests: Hastelloy-X and R-235 were selected for strength, and Inconel 702 for ductility and corrosion resistance.

Commercial creep machines were purchased and tests started on Hastelloy X and Inconel 702. The tests on Hastelloy X specimens were run at 4800 psi and 3200 psi stress levels at 1750°F. Both ruptured after about 160 hr. at the gauge marks. Another series of tests were started in which Hastelloy X specimens were stressed at 3000 psi and 2000 psi, and Inconel 702 was stressed at 3000 psi; the gauge marks on these specimens were placed in less critical areas. All tests were operating satisfactorily.

Four representative alloys were evaluated in a

1000 hr. test at 1750°F at 2000 psi stress level. The results are:

1000-HR CREEP RESULTS

1750°F - 2000 psi - Stress

<u>Alloy</u>	<u>Elongation (%)</u>
Hastelloy R-235 (1)	1.2
(2)	1.6
Hastelloy X (1)	2.1
(2)	2.2
Inconel 702	26.1
Inconel X	35.0

Longer term tests are proceeding.

c. Tube Burst Tests

The objective of this phase of tests is to subject tubing to dynamic load conditions similar to those caused by fission product buildup within the pin-type fuel elements. A quantity of helium (measured by pressure) was introduced into the tubing specimens, the ends welded shut, and the specimens placed in a furnace. Changes in the tubing as a result of this exposure were measured or observed.

Eleven trials have been run with 0.225-in. OD seamless Inconel and welded Inconel 702 tubing. Poor seal weld performance hampered initial tests. Two Inconel tubes were charged with initial pressures at room temperature of 980 and 1125 psi, respectively, which correspond to wall stresses of 2656 and 3514 psi at 1750°F. Both tubes failed after three hours at temperature: the failure occurred as a small crack in the tube wall. The tube stressed to 3500 psi increased 0.040 in. in diameter; and the tube stressed to 2656 psi increased 0.005 in. (Figure I-10).

Four tubes, three Inconel 702 and one Inconel, were later stressed to 1500 and 4500 psi at 1750°F. The Inconel tube was stressed to 1500 psi level. The Inconel 702 tubes stressed to 4500 psi burst in the weld zone, and both 1500 psi tests failed due to poor seal welds. These will be re-run with modified end welds. Lower stresses to permit longer duration tests and tests using the reference Hastelloy X tubing are planned.

4. IB-1T In-pile Test (Task 23-97)

The in-pile test is intended to evaluate the overall performance of a GCRE-IB advanced fuel element for both allowable and optimum temperatures. There are three elements in this set: IB-1ØT,

IB-1 α T, and IB-1 β T. The IB-1 α T test began 22 June 1959, and will run for about three cycles; one cycle for engineering runs, one cycle for thermal cycling, and the remaining cycle for engineering runs. Hot-cell work will be done to determine the effect of thermal cycling. The neutron flux distribution will be measured after this test with a nuclear mockup designated IB-1 θ T which will be in the reactor for about one day. The IB-1 β T test will begin immediately after the flux run.

The objectives of this experiment are to obtain heat transfer, mechanical stability, neutronics, thermocouple installation and material characteristics of the IB fuel element.

a. Description of the Elements

1) The IB-1 α T

This element (shown in Figure 1, IDO-28541) has an active (uranium bearing) length of 22.75 in. and is 1.75 in. OD at the center. The basic structural members are the bell, the outer liner and the nose. The outer liner is welded to the nose and is attached to the bell by three socket-head screws. The screws are wired, and the wires spot-welded to the outer liner. The bell is interconnected to the column support and locked in place by a welded lock ring. The column support is attached to a plug that seals the fuel element into the loop thimble.

The fuel pins are hung from a tube sheet. The tube sheet rests on a shoulder in the bell, and is locked in place with a lock-ring, spot-welded to the bell. The fuel pins have a shoulder plug so they can be accurately located in the tube sheet and locked in place with a key of 0.032 in. Inconel wire passed through a hole in the pin head. A key is passed through a second hole in the pin head and bent over to retain the wire. There is a 45° chamfer in the tube sheet to help lock this wire into the head of the fuel elements. The fuel pins are of Inconel and have an OD of 0.225 in. with 0.030 in. thick wall. They are separated down the length of the assembly by spacers called "bearings".

The fuel pins are assembled by first brazing the bearings to the pins. The UO₂ fuel pellets are stacked in the pins in a helium atmosphere to a height of 22.75 in., and a disc placed on top of the last pellet to prevent motion. A spacer (6 in. long, 1/8 in. OD tube, with a 5-mil wall) is then used to provide an expansion chamber and to prevent the pellets from shifting. A plug is Heliarc welded to the ends of the fuel pins. This completes the fuel pin assembly and the pin is leak tested. The outer ring of fuel pins are 25% enriched UO₂ and all other pins are 50% enriched UO₂.

The bearings (Figure I-11) are made from Inconel rods machined in the form of a top with an airfoil cross-

section. These bearings are located at 6-in. (approx.) intervals along all pins except three of the inner pins. A total of 48 bearings are brazed to the fuel pins in positions chosen to give maximum support with minimum pressure drop.

The insulation sub-assembly is made separately and slipped over the pin sub-assembly, tube sheets, pins and cone. The insulation sub-assembly includes an inner liner, the insulation annulus, the outer liner, the bell and the nose. The inner liner is AISI-Type 316 stainless steel tube with a 0.010 in. thick wall. "Doughnuts" of Thermoflex are compressed to form the insulation annulus. The outer liner is stainless steel for the in-pile element, although the GCRE-I element will be subjected to lower temperatures at this position and will use zirconium. The bell-to-thimble fit forms the seal between the inlet and outlet gas, and makes a seal which will leak at about 8 to 18 lb/hr. The bell supports this sub-assembly and is connected to the column support by a lock ring.

2) The IB-1~~BT~~T

The difference between the IB-1~~BT~~T and the IB-1~~BT~~T elements are:

	<u>IB-1BTT</u>	<u>IB-1BTT</u>
Pin OD	0.225 in.	0.205 in., stepped to 0.245 in.
Pin ID	0.163 in.	.163 in.
Pellets	Solid	Holes provided for the instrumented pins
Pin Closure	Heliarc-Welded	Instrumented pins brazed
Heat-Transfer Surface	Smooth	Stepped and finned

The step-up from a 0.205 in. dia. smooth pin to a 0.245 in. dia. finned pin is shown in Fig. I-12. A short length of smooth surface in the finned area is left to provide an attachment point for brazing on a bearing spacer.

b. Instrumentation

All thermocouples are chromel-alumel, magnesium oxide - insulated and Inconel-sheathed (Fig I-13 and I-14). The IB-1~~BT~~T element will utilize both bare-wire and enclosed-end-junction thermocouples. The bare-wire types are for measuring surface temperatures and are simply spot-welded to metal at appropriate locations. In the enclosed-end type, the ends are sealed with a porcelain head to prevent erosion of the MgO insulation by the gas stream. The unit is supported in place by clips.

The disc-type thermocouple concept will be tried in the IB-1¹T element. In this type, the sheath is inserted through holes provided in the center of the pellets, and the AISI-Type 316 stainless steel disc is press-fitted to the inside of the fuel pin. Thus, pin-metal temperature will be measured rather than fuel pellet temperature. Conventional base-junction thermocouples will be used on the outside of the pins as a reference. The IB-1¹T element is instrumented to gather data on: 1) longitudinal profile of liner; 2) longitudinal profile of the fuel pin; 3) radial pin-to-pin temperatures at hot spot; 4) peripheral temperature variation around a single pin; 5) inlet and outlet gas temperatures to measure coolant enthalpy increase; and 6) temperature of gas at four radial positions to indicate the amount of coolant mixing.

The IB-1³T element is instrumented to gather data on: 1) longitudinal profile of liner; 2) longitudinal profile of "A" (outer circle) and "B" (inner circle) pins along both diameters of the pins; 3) radial pin-to-pin temperatures at hot spot; 4) inlet and outlet gas temperatures to measure coolant enthalpy increase; 5) peripheral temperatures around an "A" and "B" pin; 6) temperature at center of UO₂ pellets; and 7) accuracy of the thermocouple-disc attachment within the pin calibrated against an open-ended thermocouple.

c. Test Conditions

1. Nuclear Conditions, IB-1T

Thermal neutron flux (vol avg.)	$5.6 \times 10^{12} \text{ n/cm}^2\text{-sec}$
Peak-to-average vertical flux distribution	1.26
Peak-to-average peripheral flux distribution	1.10
U ²³⁵ per element	421.5 g
Pin loading, weight percent	25* & 50**
Length of meat (inches)	22.75

*25 wt % loading in outer row of pins

**50 wt % loading in all other pins

2. Thermodynamic Conditions

Element Power, kw	35
Gas Inlet Temperature, °F	800
Gas Outlet Temperature, °F	1240
Calculated nominal maximum surface temperature, °F	1530
Gas flow rate, lb/hr	1000
Operating pressure, psig	200
Gas condition:	
Nitrogen, minimum	99.5%
Oxygen, maximum	0.5%
minimum	0.1%

D. BATTELLE MEMORIAL INSTITUTE (Task 23-53)

(Note: the following material is detailed under BMI task nomenclature)

1. UO₂ - Stainless Steel Properties (Task I-B)

Work on this task (i.e., measurement of mechanical and physical properties of UO₂-stainless steel dispersions) was completed during the report period. A topical report - "The Effect of Fabrication Variables on the Structure and Properties of UO₂-Stainless Steel Dispersion Fuel Plate", BMI-1322 - was issued.

2. Fabrication of UN-Stainless Steel Plates (Task I-C)

A total of 44 enriched and 36 natural UN-stainless steel plates were fabricated. A topical report discussing plate fabrication and tabulating the properties of these plates is being prepared. The plates were to be formed and assembled into an in-pile test element having the GCRE-I dimensions and configurations in connection with the IA fuel element program.

3. Irradiation of UO₂-Stainless Steel Dispersions (Task I-D)

Capsule 11-7, containing four UO₂-stainless steel matrix (14 Cr, 18 Ni, and 3 Mo) specimens, was irradiated in the MTR to an estimated burnup of 17 atom % of the originally contained U-235 at temperatures in the vicinity of 1600°F. In two of the specimens 25 wt% highly enriched UO₂ was used; in the other two, 30 wt%. All the samples were clad in AISI Type-318 stainless steel.

The capsule was examined in the BMI hot-cell in

January. All four specimens were badly blistered and had swelled considerably. The 25 wt% specimens were in slightly better condition than were the 30 wt% specimens. One 30 wt% specimen was cracked across its width. Radiochemical burnup analyses are being performed. These results shows that the limit of operation for 25 and 30 wt% UO_2 -stainless steel dispersions at these temperatures had been greatly exceeded.

4. Design and Fabrication of a Capsule for Irradiation of UN-Stainless Steel and UO_2 -Stainless Steel Specimens (Task I-F)

The capsule design was complete and fabrication well advanced at the beginning of the report period. The capsule is double-walled and has an electrical heater; the specimens are suspended in a NaK bath inside the inner capsule. Auxiliary heat to achieve the desired temperatures in the event of low fluxes is provided by four 1-kw heaters. The specimens were to reach operating temperatures of 1650°F and burn-up between 9 to 12 atom % U-235. An unperturbed flux of 0.7×10^{14} nv (th) for six MTR cycles was considered necessary to achieve this burn-up. The capsule actually received $0.8 - 0.9 \times 10^{14}$ nv (th) when inserted in MTR Cycle 120 on 10 April. The auxiliary heaters were not used. Representative capsule operating data is shown below:

Representative Capsule Operating Data

<u>Thermocouple</u>	<u>Thermocouple Temperature °F</u>	<u>Calculated Specimen Surface Temperature °F</u>
1	1600	1620
2	out	--
3	1605	1625
4	1730	1750
5	1700	1720
6	1570	1590

Thermocouples are located as follows: 1 and 3 opposite top specimen, 4 opposite specimen 2, 5 opposite specimen 3 and 6 opposite specimen 4.

It was estimated that the desired burnup would be achieved in five cycles. Later data and calculations indicated that irradiation to 12 atom % would probably be too severe, especially as little spread in burnups would be achieved. Irradiation time was reduced by one additional cycle and a burn-up of 10.5 atom % U-235 accepted. The capsule was removed after four cycles on 24 June.

5. Design and Fabrication of Capsule for Irradiation of Solid UO_2 Pin-Type Specimens (Task I-F)

Design of another double-walled capsule with auxiliary heaters was completed in December. The specimens are suspended in a bath of NaK within the inner capsule. Six Inconel-clad specimens were placed in the capsule. Four of the specimens are UO_2 solid pellets (35% enriched), two are highly enriched annular oxide pellets. Two identical capsules were fabricated, one for irradiation to 5.8 atom % U-235 burnup and one for 9.7 atom % U-235 burnup. Each of these capsules contain three 1-kw heaters wrapped around the inner capsule. A flux of 1.09×10^{14} nv (th) was required to achieve the desired burnups in four and seven cycles, respectively.

The 5.8 atom % burnup capsule (27-2) started irradiation 17 March; the 9.7 atom % burnup capsule (27-1) started 6 April. Although the design neutron flux for both capsules was 1.09×10^{14} nv (th), it is estimated that capsule 27-2 is operating in a flux of 0.8×10^{14} nv, and that capsule 27-1 is in a flux of 1.1×10^{14} nv. Representative thermocouple values are tabulated below.

Capsule 27-1 appears to be operating in a severe flux gradient. This, combined with a design error which added about 30% more thermal resistance between inner and outer capsule walls, caused specimen temperatures to be higher than expected with the fluxes encountered. As a result, where design fluxes are encountered, excessive specimen temperatures are generated. Capsul 27-1 is operating on the edge of the thermal flux gradient, with two specimens operating at close to design temperatures but at less than design fluxes.

Capsule 27-2 has been operating with the profile shown on the table below. Apparently, this operation profile is the result of the error in calculations (mentioned above) and the inclusion of capsule biasing (to compensate for flux gradients).

REPRESENTATIVE THERMOCOUPLE TEMPERATURES
OF CAPSULES 27-1 AND 27-2 (BEFORE MOVEMENT)

<u>Thermocouple Number</u>	<u>Temperatures $^{\circ}\text{F}$ 27-1</u>	<u>Temperatures $^{\circ}\text{F}$ 27-2</u>
1	1000	Control 1650
2	1270	out
3	out	1150
4	out	1330
5	1480-1510	1460
6	1675	out

REPRESENTATIVE THERMOCOUPLE TEMPERATURES
OF CAPSULES 27-2 AFTER MOVEMENT)

<u>Thermocouple Number</u>	<u>Thermocouple Temperature °F</u>
1	Out
2	Out
3	1080
4	Out
5	1335
6	Control 1650

Capsule 27-2 was recently moved to equalize specimen burnup. The new position was not successful, and it was moved again to a safer position with lower fluxes. In this position an additional cycle would have been required to attain the desired burnup. However, the capsule was removed 24 June to obtain early data.

6. Modification of the BRR Loop (Task I-G)

The BRR loop was modified to increase specimen power capacity and to increase system temperatures and flow rates. The modifications include: 1) construction and installation of a larger blower; 2) design and fabrication of a new test section; 3) installation of a pre-heater; and, 4) installation of additional recording and controlling instrumentation. The loop was installed in the back pool, away from the reactor core, early in April for check-out and leak repairing. The loop was operated to obtain information on loop operating characteristics.

Loop operation started 23 June with the irradiation of the IB-1A/T element, but instrumentation problems caused shutdown later that week. Operation should start again on 1 July with the next BRR cycle.

7. Design and Fabrication of the BMI-16 (ETR) Loop (Task I-H)

The purpose of this task was to design, build and install an in-pile loop for testing prototype gas-cooled reactor fuel elements in the ETR. All loop components as well as fuel element handling equipment, attenuators and casks were fabricated during the report period. The components have all arrived at NRTS. One control panel was severely damaged in transit, but has been repaired and installed.

Inserting the in-pile tube is the only step to be completed before loop operation. This is now scheduled for 24 July. When this is completed, hot check-outs, loop operation, and specimen irradiation can start, in that order. The I-3P element (a GCRE-I production element) was shipped to Idaho 15 April, and will be irradiated in this loop.

A hazards report on the BMI-16 loop was prepared by BMI and forwarded to PPCo. in February. After modifications, the Hazards

Committee approved the loop for operation. The operating manual was forwarded to PPCo. in March for comment, and has since been published in final form. The I-3P elements hazards summary, prepared by Aerojet, was forwarded to PPCo. and was approved.

A critical experiment was performed in the ETR critical assembly on 24 February to obtain flux information for use in fabricating neutron attenuators. These attenuators are necessary to reduce the ETR flux to about that of the GCRE-I.

Operator training is underway and should be completed early in July. When this training is complete, only insertion of the in-pile-tube remains to complete the loop. Some check-outs will be necessary after in-pile-tube installation, but loop operation can begin when these are complete.

8. IB-1T Operation (Task I-L)

The IB-1 α T started irradiation 23 June but was interrupted. Irradiation is scheduled to start again in July and continue for two BRR cycles. During irradiation, the element will undergo a series of thermal cycles to determine its resistance to such conditions. After irradiation, the element will be examined in the Battelle hot-cell.

The IB-1 β T irradiation program will include both steady-state operation and shock testing, and will last until about 1 December 1959.

9. IB Fuel Fabrication

The object of this task was to provide fuel pellets for the IB fuel elements. The fuel pellets were fabricated by BMI in order to meet the desired delivery schedule.

The pellets were of two designs: one solid, with an outside diameter of 0.160 ± 0.002 in; and one hollow, with the same outside diameter, but with an inside diameter of 0.0480 ± 0.004 in. Both pellets were designed to be about twice as long as their diameter. In addition, each type of pellet was to be supplied in two UO_2 enrichments, 25 and 50 wt% U-235. The pellets were designed to have a density of 92%, but production attained 94%.

About 2000 solid pellets were completed for the IB-1 α T on 5 June. The 4000 hollow pellets for the IB-1 β T are scheduled to be finished early in July. This task will be terminated when pellet production is complete.

FIG.I-1 PLATE C18-21 M&C 34 10X
M NO. L-6940 - CIRCUMFERENTIAL WRINKLES
IN PLATE AFTER FORMING - CONVEX SIDE UP.

FIG.I-3 PLATE C18-26 75X
NO. 5087 - SECTION OF LONGITUDINAL CRACK
IN CIRCUMFERENTIAL WRINKLE SIMILAR TO
FIGURE I-1. PLATE FAILED ACID ETCH TEST.

FIG. I-2 PLATE C18-21 M&C 34 NO. 5088 - 1-2-3-4 75X
SECTION THROUGH WRINKLED AREA SHOWN IN FIGURE I-1 - THIS PLATE DID NOT PASS ACID ETCH TEST EVEN THOUGH NO
CRACKS COULD BE FOUND VISUALLY - LENS SHAPED AREA OF UNDETERMINED MICRO STRUCTURE BELOW
WRINKLED CLAD IS SLIGHTLY HARDER THAN SURROUNDING MATRIX.

FIG. I-4 TECHNIQUE FOR JOINING 5-MIL THERMOCOUPLE WIRES TO LEAD WIRES

FIG. I-5 CAPSULE SEALED BY FILLING ONE END WITH EPON 828-20% ACCELERATORY, AND CEMENTING A 0.003-MIL ALUMINUM FOIL AT THE OTHER END.

LETTER	DATE	CHANGE	BY	CHK'D

 ACTUAL SIZE

0-403433

COPY ☐

FIG. I-6

MAKE FROM CASTING 0-403297

MATERIAL

NO. REQD.	TEST AMT.	MODEL	DRAFTSMAN	DATE
			G. PENFIL	6-22-58
			CHECKED	
			STRESS	
			ENGINEER	
			APPROVED	
			PRODUCTION	
			MATERIAL	

PROPOSED
 INTEGRAL AIRFOIL
 FOR
 IB-IR ELEMENT

AEROJET-GENERAL
 NUCLEONICS
 SAN RAMON
 CALIFORNIA

0-403433

DRAWING NUMBER

1. THESE SURFACES MUST BE PARALLEL WITHIN .001 PER INCH
 2. DIA. MARKED C MUST BE CONCENTRIC WITHIN .001 TIR

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES.

FIG.I-7 HASTELLOY R-235 AFTER 2500-HR. EXPOSURE.

FIG.I-8 HASTELLOY X AFTER 2500-HR. EXPOSURE.

FIG.I-9 INCONEL 702 AFTER 2500-HR. EXPOSURE.

FIG.I-10 SMALL CRACK IN TUBE AFTER 3 HRS AT 1750 °F

ACTUAL SIZE

BEARING LOCATION (REF O-403553-*)						% ENRICHMENT
PIN NO.	*	A	*	B	*	
1	(-2)	6.5	(-2)	14.0	(-2)	50
2	-	-	-	-	-	50
3	(-2)	8.5	(-2)	16.5	-	50
4	-	-	-	-	-	50
5	(-2)	3.5	(-2)	11.5	-	50
6	-	-	-	-	-	50
7	(-2)	9.5	(-2)	18.5	-	50
8	(-2)	5.0	(-2)	21.0	-	25
9	(-2)	5.5	(-2)	12.5	(-2)	25
10	(-2)	7.0	(-2)	15.0	-	25
11	(-2)	10.0	(-2)	17.5	-	25
12	(-2)	4.5	(-2)	13.0	(-2)	25
13	(-2)	7.5	(-2)	15.5	(-2)	25
14	(-2)	3.0	(-2)	11.0	(-10)	25
15	(-2)	5.0	(-2)	13.5	(-2)	25
16	(-2)	6.0	(-2)	14.5	(-2)	25
17	(-2)	9.0	(-2)	17.0	-	25
18	(-2)	4.0	(-2)	12.0	(-2)	25
19	(-2)	8.0	(-2)	16.0	-	25
SPARE	-	-	-	-	-	50
SPARE	-	-	-	-	-	25
SPARE	-	-	-	-	-	25

AR	-	WELDED - INCO NO. 62 PER MIL-R-3031
AR	-	OPTIONAL UO ₂ PELLETS (SEE CHART) REF SPEC

O-403554

COPY A

NO.	PART NO.	DESCRIPTION
1	O-403553/10	BEARING
34	O-403553/2	BEARING
22	O-403557	TUBE
24	O-403551	POINT - PIN ELEMENT
22	O-403550	SPACER - PIN ELEMENT
24	O-403549	DISC - PIN ELEMENT
22	O-403548/PLUG	PIN ELEMENT

6. TO BE STRAIGHT WITHIN .005 PER 12 IN.

4. REFER TO CHART.

5. HELIARC WELD IN AN INERT ATMOSPHERE USING INCO NO. 62 WELD ROD.

2. TAG WITH PIN NO.
1. REMOVE ALL BURRS & SHARP EDGES.

NOTES:

FIG. I-II

DATE	1-2-59
BY	EXB
CHKD	12/16/58
APP'D	12-16-58

Q-403539	1B-10-T
HEAT TREAT	FINISH
TOLERANCE UNLESS OTHERWISE NOTED	
LINEAR	ANGULAR
XXX	XXX
INDICATES SURFACE ROUGHNESS	
12-18-58	

TUBE ASSY - PIN ELEMENT	
DATE	12-18-58
SCALE	4/1
RELEASE DATE	12-18-58
APPROVED	

AEROJET-GENERAL NUCLEONICS
SAN RAMON, CALIFORNIA
O-403554
PART NUMBER

FIG.I-12 THE IB-1 β T FINNED PIN.

THERMOCOUPLE LOCATIONS OF IB-1αT

- OPEN-ENDED THERMOCOUPLES
- CLOSED-END THERMOCOUPLES

FIG. I-13

THERMOCOUPLE LOCATIONS OF IB-1BT ELEMENT

$X/L = 0.050$

$X/L = 0.200$

$X/L = 0.350$

$X/L = 0.500$

$X/L = 0.730$

$X/L = 0.900$

- OPEN-ENDED THERMOCOUPLES
- CLOSED-END THERMOCOUPLES
- ▲ THERMOCOUPLES WITH DISC ATTACHMENT

FIG. I-14

TABLE I-1

NUCLEAR CONDITIONS I-3P

Thermal Neutron Flux	=	$6.2 \times 10^{13} \text{ n/cm}^2 \text{ sec}$
(Unperturbed, midplane, ETR full power, 0 MWD, integrated average over fuel cycle in BMI-16 position, cycle-8 measurements)		
Gamma Heating Rate	=	1.0 watt/gm
Midplane-to-Axial Average Thermal Neutron Flux (in Test Element)	=	1.2
Unperturbed Flux/Flux at Element Surface	=	4.0
Flux at Element Surface/Volume Average Flux in Element	=	1.5
Unperturbed Flux/Volume Average Flux in Element	=	6.0
U^{235} per Element	=	316 g
Fission Cross-Section	=	550 b
Plate Loading	=	30 wt %
Plate Thickness	=	.045 in. (.006 in. clad; .033 in. meat)
Length of Fuel Plate	=	28 in.
Factor for Rise in Neutron Flux During ETR Fuel Cycle	=	1.7

TABLE I-2

THERMODYNAMIC CONDITIONS

Average Element Power (Fission plus Gamma)	=	45 kw
Gas Inlet Temperature	=	1000 to 710 ⁰ F
Gas Outlet Temperature (Mixed-Mean)	=	1320 to 1260 ⁰ F
Measured Surface Temperature	=	1425 ⁰ F
Gas Flow Rate	=	1376 lb/hr
Operating Pressure	=	200 psig
Pressure Drop (Across Element)	=	13.6 to 12.0 psi
Gas Condition		
Nitrogen, Minimum	=	99.5%
Oxygen, Maximum	=	0.5%
Minimum	=	0.1%
Humidity, Maximum	=	0 ⁰ F dew point

TABLE I-3

ANTICIPATED POWER GENERATION IN TEST ELEMENT - ETR FULL POWER

	<u>Initial</u>	<u>Average</u>	<u>Final</u>
Fission Energy Plate A (Outer)	15.33 kw	19.87 kw	26.04 kw
Fission Energy Plate B	8.60	11.10	14.60
Fission Energy Plate C	5.76	7.43	9.78
Fission Energy Plate D	3.41	4.40	4.78
Total Fission Energy =	33.1	42.7	56.2
Gamma Heating Plate A 523 g	.523 kw		
B 523 g	.523 kw		
C 442 g	.442		
D 292 g	.292		
Center Spine 111 g	.111		
Clips 216 g	.216		
Insulation Liner 165 g	.165		
Total Gamma Heating =	2.3 kw	2.3 kw	2.3 kw
Total Energy Produced	35.4 kw	45.0 kw	58.5 kw
Desired Thermal Neutron Flux in Element (Volume and Axial Average)	2.23×10^{12}	2.88×10^{12}	3.79×10^{12}

GAS COOLED REACTOR EXPERIMENT - II

I. INTRODUCTION AND SUMMARY

The GCRE II is a homogeneous solid moderated reactor. It is an advanced backup to the heterogeneous water-moderated reactor which is reference for the ML-1a Army field power plant. The project is in the preliminary design and development stage. Previous work is reported in a feasibility study, IDO-28530.

The GCRE II preliminary design effort envisages a fuel element formed of a dispersion of a suitable compound of uranium in high density graphite, which is coated and then canned in a high temperature alloy. Coolant passages in the reference concept consist of a large number of tubes (nominal 0.25 inch diameter) passing axially through the cylindrical core. The core will be divided into hexagonal or segmental fuel elements which include cans, tubes and end reflectors, as well as coated fuel-moderator sections. Surrounding the core is a moderating reflector of beryllium, beryllia or zirconium hydride; the core and reflector are surrounded by a pressure vessel and shielding. Control rods are placed in the reflector and may also be placed in a moderating island in the center of the core.

During this report period, a low level of analytical work was maintained on heat transfer and neutronics. The principal effort was devoted to the procurement of samples of fuel, coating and canning alloys, and to testing these materials. Leak tests of coatings and can welds and corrosion tests of several alloys, have been performed. An irradiation capsule test (II-1C) has been prepared and is ready to go in-pile. Another capsule (II-2C) for long term radiation and corrosion testing is now in the design stage. Work has been initiated on an in-pile loop test element for insertion in the BMI-16 loop in December 1959.

II. CAPSULE TESTS (Task 43-61)

A. II-1C CAPSULE (BMI Task II-D)

The II-1C Capsule is complete and will start irradiation in July. The capsule contains six pins of graphite coated with siliconized silicon carbide and canned in Hastelloy X, Inconel 702 and Carpenter 20 Cb alloys, containing 8 wt % highly enriched UO_2 .

The capsule design was completed in March and resulted in a capsule similar to the solid UO_2 pin capsules described in Task 23-53. It is a double-wall design with auxiliary heater in which six specimens are suspended in an NaK bath within the inner wall. Three 1 kw heaters provide additional heat to compensate for low flux levels. Specimen surface temperature will be $1700 \pm 50^\circ F$, with burn-up of 15 a/o U-235. To achieve this, an unperturbed flux of 5×10^{13} nv (th) will be required.

The specimens are 0.25 in. OD fueled SiC-coated graphite clad with 0.025 in. Hastelloy X, Inconel 702 and Carpenter 20 Cb. They were placed end to end in the inner capsule. The capsule was assembled and checked and is due to start irradiation in MTR cycle 125 20 July as BMI 29.

B. II-2C CAPSULE (BMI Task II-E)

Design started in January on a capsule to provide long term burnup, fission product retention and in-pile corrosion data on GCRE-II reference and back-up materials (the II-2C capsules). The following data are to be obtained from the hot cell examination of the II-2C capsule:

1. Corrosion of the area near the tube-to-tube sheet and tube sheet-to-can welds. These areas may be susceptible to corrosion as the result of volatilization of some alloying elements and carbide precipitation during welding.
2. Stability of the coating as well as its ability to retain fission product gases.

3. Effect of fuel burn-up at reference loading and irradiation time.
4. Ability of the coating to prevent metal carburization.
5. Radiation damage to reference materials.

This capsule will allow evaluation of many fueled and non-fueled components under reactor operating conditions and in configurations approximating reactor design. Fueled graphite, (8 wt % UO_2) coated with SiC and clad with one of three nickel-based alloys, will undergo long term (5000 hr) irradiation in the BRR. The reference coolant gas will be continuously circulated in this capsule under reference pressures to provide information for the evaluation of metal corrosion.

A unique temperature control mechanism will be utilized to reduce the bulk inherent in capsules containing NaK and auxiliary heaters. Temperature control will be maintained by varying the thermal conductivity of helium in an annulus surrounding the fuel specimen. The thermal conductivity is controlled by the pressure at which the helium is maintained.

The three specimens to be irradiated will be stacked one on top of the other. Each specimen will consist of two fueled graphite blocks separated by an unfueled block. Honeycomb spacers will be installed at each end between the fueled graphite blocks and the header plates. The whole assembly will be totally enclosed in a nickel-based alloy can joined to the capsule headers. The specimen will be centered and enclosed in a heavy outer can so that a pre-determined gas annulus exists for control of gas flow.

A thermal mock-up of the II-2C capsule has been fabricated and is being tested at BMI. A nuclear mock-up for the II-2C capsule was completed and shipped to BMI. It contains fueled graphite prepared by Great Lakes Carbon Co. Components are being fabricated for the II-2C capsule which is scheduled to go in-pile 1 October. Graphite (from three vendors), two coating materials, and two can materials are to be tested in the II-2C in the following combinations:

<u>Subcapsule No.</u>	<u>Graphite Vendor</u>	<u>Coating Material</u>	<u>Can Material</u>
II-2aC	Great Lakes Carbon	SiC	Inconel 702
II-2bC	Minnesota Mining & Mfg.	C + SiC	Hastelloy X
II-2cC	American Metal Products	C + ZrC	Hastelloy X

CONDITIONS

Control point temperature	$1700^{\circ}\text{F} \pm 50^{\circ}\text{F}$
Perturbed flux (2200 m/sec)	$4.8 \times 10^{12} \text{ n/cm}^2\text{-sec}$
Volumetric heat generation rate	$3.5 \times 10^6 \text{ Btu/hr-ft}^3$
Irradiation time	5500 hours
Burnup	6.6%
Atmosphere	GCRE reference
Atmosphere flow-rate	$0.1 \text{ lb/hr} \begin{matrix} +100\% \\ - 50\% \end{matrix}$
Atmosphere pressure	$400 \pm 5 \text{ psia}$

III. MATERIALS TESTING AND EVALUATION (Task 43-64)

A. FUEL

Discussions and negotiations are underway with four vendors who have technical and facility capabilities for producing fueled graphite: Great Lakes Carbon Co., Minnesota Mining and Manufacturing Co., American Metal Products Co., and National Carbon Co. The Great Lakes Carbon Co. produced fueled graphite for the II-1C and the II-2C in-pile test capsules. AGC, Sacramento, is determining thermal expansion and thermal conductivity characteristics of samples of this material up to 2000°F. Negotiations are in process to procure fuel from the other vendors and to make basic studies on this fueled graphite, as well as to procure fueled graphite for the II-2C capsule from 3M and American Metal Products Co.

B. COATINGS

The silicon carbide coatings produced by 3M and by AGC Sacramento, are being evaluated. These evaluations include the corrosion characteristics of silicon carbide coatings in reference atmosphere and in air at 1750°F; compatibility studies of the coating on fueled graphite with various cladding alloys; thermal cycling stability of SiC coatings on graphites; and permeability studies on SiC coated graphites.

Other types of coatings being investigated are refractory metal coatings of tungsten, chromium, molybdenum and zirconium carbide. These coatings will be produced by vapor phase deposition.

C. CLADDING ALLOYS

The cladding alloys of interest are Hastelloy X, Inconel 702, (both nickel-base, high temperature alloys) and Carpenter 20 Cb, an iron-base alloy. Corrosion tests are being run in the reference atmosphere and in air at 1750°F and 200 psi. Creep data are being obtained on the cladding alloys at 1750°F at stress levels of 1000, 2000, and 3000 psi. The 250 hr corrosion tests in nitrogen and the

1000 hr corrosion tests in air have been completed for these alloys. The preliminary data indicate that Hastelloy X has superior oxidation resistance. The 1000 hr creep tests on Hastelloy X and Inconel 702 have been completed. These tests (at 1750°F and 1000 and 2000 psi) show that Hastelloy X has better high temperature creep resistance. (Also see Task 23-65).

D. EQUIPMENT

A helium leak tester was installed and procedures established for leak testing welded cans for fueled graphite. This unit will be used for checking porosity or cracks in graphite coatings, and for testing canned fuel element assemblies.

A hot silicone oil bath leak tester was set up and tested for checking porosity and cracks in silicon carbide-coated graphite. It is also being used for locating leaks in canned graphite assemblies.

IV. CAPSULE DESIGN (Task 43-62)

This task was undertaken to develop capsule designs that would maintain reference peak temperatures for long periods of time. This implies the development of methods of temperature control which do not rely on electric heaters.

Several means of temperature control were investigated: variable heat conduction area across a thin gas gap; variable gas composition (and, therefore, thermal conductivity) in a thin annulus surrounding the specimen; and variable natural convection heat transfer by variations in coolant gas pressure. Analysis showed that either of the latter two methods would give adequate temperature control over the range of flux uncertainty. A brief test was performed to verify the analysis of the last-named method.

The II-2C design was turned over to BMI in April for final design, fabrication and irradiation. AGN is to supply the specimens to final dimensions for the capsule test (Figure II-1). Progress on this task is presented under Task 43-61.

V. CORE PRELIMINARY DESIGN (Task 43-91)

A. DESIGN OBJECTIVES AND REFERENCE SYSTEM

Adherence to current thermodynamic design specifications should produce a reactor system of relatively high cycle efficiency and satisfy the weight requirements of a portable system. (The thermodynamic design specifications are listed in Table II-1.) The reference coolant is the same as for ML-1, and the maximum gas and metal temperatures represent the feasible limit of available materials.

Reference design criteria are given in the second part of Table II-1. This system is believed to satisfy the thermodynamic design specifications and to be mechanically feasible with presently available materials and techniques. Departures will be made from this design as advanced fabrication techniques are demonstrated. Ultimately it is expected that ceramic coating techniques will be developed to a point where canning metal becomes unnecessary. It is not anticipated that this development will occur in time for use in the first GCRE-II core. Development programs initiated during the past six months are scheduled to yield results by the first quarter of calendar 1960 on which a firm reactor design can be established.

The reference fuel element is a hexagonal can containing a stack of graphite blocks 3 in. long, loaded to about 8 wt % of highly enriched uranium. (Figure II-2 is a conceptual sketch of the reference fuel element.) The coolant flows axially through 19 tubes 1/4 in. OD in the graphite. The fuel element contains a 3 1/2 in. reflector section of BeO at the outlet end. The fuel element stands on a supported and canned 4 1/2 in. thick inlet reflector of zirconium hydride. Coolant flows upwards in tubes through the inlet reflector to the fuel element. The fuel elements are positioned between the lower support plate and the upper alignment plate by a pin arrangement (See Figure II-2).

The 4 in. thick side reflectors will preferably be made from canned zirconium hydride. Compartmentalization of the ZrH may be necessary to prevent hydrogen migration at the high operating temperatures of the reflector. The high temperatures of the reflector

allow the heat generation in the reflector to contribute efficiently to the power cycle. The principal advantage of a ZrH reflector over BeO or Be is that the higher density of ZrH saves about 1.2 tons of shielding weight due to the higher gamma absorption close to the core. Final choice of side reflector material will depend on further study of fabrication techniques for ZrH, Be and BeO, and on analysis of the effect of various methods of reflector cooling on cycle efficiency and reactor design.

The use of an island reflector in the center of the core is not yet established. It has been shown that the island does not aid criticality and complicates removal of core heat. Since it may be desirable from the standpoint of reactor control to retain the island concept, no decision will be made on the island until preliminary control rod designs and analyses are complete.

B. DESIGN ANALYSIS

Design analyses have, in general, been in support of the reference fuel elements. The results of these analyses are summarized below:

1. Stress Analysis

Relative expansion of canning material and graphite may cause stress concentrations in the coolant tube, hexagonal can corners, and end caps. For this reason a thermal cycling test program described in a following section has been initiated.

Stress considerations require a carefully controlled fit between coolant tubes and holes in graphite. Since it is likely that fuel elements will be formed from a stack of hexagonal, drilled pieces of graphite, an intolerably high stress will exist in the tubes, at the junction of graphite blocks (due to relative thermal expansion of the metal and graphite) if the necessary clearance to assemble the element (0.0001 in.) occurs on opposite side of adjacent stacked blocks. To eliminate this effect it is proposed to gun drill the assembled fuel element stacks prior to insertion of coolant tubes.

2. Thermal Analysis

For a coolant channel design employing simple tubes, parametric analysis resulted in a system which used 1800, 3/16 in. OD tubes (0.025 in. wall thickness) to remove 4 Mw of heat from the core without exceeding a 1750°F maximum metal surface temperature. Friction pressure drop through the tubes, including allowance for orificing, is 18 psi. To reduce orificing requirements, fuel elements adjacent to the side reflector, where heat generation is the highest, have 1/4 in. OD coolant tubes. The analysis was based on a power distribution supplied by PDQ calculations and hot channel factors of 1.2 and 1.3 on the bulk temperature rise and film temperature drop, respectively. Computations

were made by an IBM-704 program, COP; a heat transfer and fluid flow code for reactor cooling analysis.

Future thermal analyses will continue parametric heat transfer calculations with attention given to extended surface methods of heat removal. Future studies will also include re-determination of hot channel factors, internal temperature distribution in the graphite, reflector cooling, and analysis of flow between hexagonal cans.

3. System Analysis

Net plant efficiency is about 14% for the GCRE-II reference system as a heat source in the ML-1 power plant. This calculation was made on the assumptions of 100°F ambient temperature, 1400°F turbine inlet temperature, and 100 kw auxiliary power requirements. Computations were made with CHOP, an IBM-704 code developed to calculate system operating characteristics on the basis of the capabilities of specific pieces of hardware: the reactor, the regenerator, precooler, turbine, compressor and generator. Continued use will be made of this code for parametric analysis to determine the optimum cycle state points and component size of the system. This analysis will be combined with a parametric analysis of reactor characteristics, using COP to determine a power plant system of minimum weight and reasonable reactor design requirements. Included in this system analysis will be the effect on total system weight of various choices of side reflector materials.

4. Weight Analysis

The weight estimates of several conceptual systems are compared in Table II-2. Shielding and core weights were calculated for various conceptual systems, but auxiliary weights such as control rods, skids, shock mounts, etc., were estimated from ML-1 studies. (The relative increase in shielding weight due to streaming and activation are also based on ML-1 calculations.) Emphasis on weight analysis will increase as work progresses towards the optimum system.

C. FABRICATION DEVELOPMENT

1. Welding Studies

Welding of Inconel 702 has proven difficult. Sample welds have been made on a test unit to simulate the joining of coolant tubes to the fuel element end cap. One unit was made helium leak tight, as welded, but developed leaks when thermal cycled to 1800°F. Satisfactory seams welded to form hexagonal cans have not been produced by either manual or microwelding. It is believed that some of the welding difficulties can be attributed to the presence of aluminum which volatilizes during welding.

Samples of Hastelloy X have been received. No welding studies have been made as yet. It is predicted that welding should be less difficult with Hastelloy X than with Inconel 702 because the former does not contain any highly volatile constituents.

Advanced welding techniques are being investigated. Test specimens of Hastelloy X and Inconel 702 have been shipped to a vendor for experiments with electron beam welding techniques. Resistance welding is being considered as a method for joining tubes to the fuel element end caps. Welding studies in a controlled atmosphere will start soon at San Ramon.

2. Tool Development

Two special tools have been developed for fabrication studies. The first tool is a jig for precision lathe drilling small sections of graphite. Precision techniques for drilling graphite will be needed for thermal cycling test elements, and the first in-pile test element. A milling and welding fixture for the sheet metal sections from which hexagonal fuel element cans are formed was the second tool developed.

3. Instrumentation Development Unit

The purpose of the instrumentation development unit was to demonstrate the concept of measuring coolant tube surface temperature by means of 0.060 in. duplex sheathed thermocouples attached to the outer tube wall. (Fig. II-3 is a photograph of such a unit before and after assembly.) The thermocouples were inserted through holes drilled in the end cap, and were inserted into broached grooves in the graphite. The thermocouples were brazed with low melting Microbraz (1800°F) after first plating the area with 0.0005 in. of nickel. All other joints were closed by welding. Silicone oil tests showed that the brazed thermocouple joints were leak tight.

4. Manufacture of Hexagonal Cans

Two methods of manufacturing hexagonal fuel element cans are being developed. The first method uses precision sheet metal techniques. The can is formed by seam-welding two half-hexagon cans. Thus far the half-hexagons have been formed by hand braking; but it is obvious from the preliminary work that special fixtures and a power brake will be needed to form the cans with reasonable tolerances. The other method of hexagonal can manufacture is explosive forming. The feasibility of this technique is being evaluated. Both programs will be continued until a satisfactory method of hexagon manufacture can be established.

D. THERMAL CYCLING UNIT

A thermal cycling test loop is being built to investigate possible modes of fuel element failure from combinations of thermal and external pressure stresses. Specifically, the test will investigate possible failure in the region of the fuel element end cap caused by coolant tube bending stress, corner stress induced in the hexagonal can, and the cracking or chipping of the graphite coating at the load bearing point. The stresses of principal concern exist at the end cap region, therefore, the need for a full length thermal cycling unit is not justified at this stage of fuel element development. The current test unit is large enough to accommodate a specimen 11 in. long and 5 in. diameter. The first test will be 100 cycles on an 11 in. length of a 3 in. reference hexagonal fuel element. The unit will then be used either to screen fuel elements of different hexagonal dimensions or geometries, or to continue tests of the reference element.

The main components of the test loop (Fig. II-4) are:

1. Test Section

The test section is a pressure vessel 14 in OD by 3 ft. high. It is designed to test elements up to 5 in. in diameter by 11 in. long. It contains a tubular heater that can heat the element to 1400° F.

2. Test Element

The first test element will be a right hexagonal cylinder 3.125 in. across the flats and 10.875 in. long. The test element contains 19 coolant tubes 0.3125 in. OD and 8 thermocouples. The element contains a 3 in. section of graphite loaded with natural UO_2 and silicon coated; two graphite spacers 3.5 in. long; and two firebrick insulators 0.250 in. thick. The canning and tubes are fabricated from Inconel 702. The hexagonal can is fabricated from two half sections welded on the can flats. A special fixture has been designed for milling the two half sections, and for holding them in position for welding.

3. Heat Exchanger

A water-cooled single tube counter-flow heat exchanger removes the heat generated in the test section and by the re-circulating blower. Water for the heat exchanger is cooled in the existing cooling tower.

4. Re-circulating Blower

A Roots-type blower has been chosen for recirculating the system gas. The blower will be housed in a pressure

shell since the gas pressure is 400 psi. The design capacity of the blower is 1600 lb/hr of nitrogen at inlet conditions. This capacity is sufficient to cool the test element from 1400 to 400°F in 30 minutes.

5. Control Circuit

Two manual set, interlocking meter relays will be used in the control circuit. One meter relay will control the cycle temperature range of the test specimen through remote operation of the system bypass valves and the test section heater. The other meter relay will be used to prevent over-heating of the heater. The entire circuit is designed to fail safe.

6. Nitrogen Make-up System

A nitrogen make-up system has been included to maintain system pressure and to operate control valves.

7. Helium Leak Detection

Samples of the primary loop gas will be withdrawn at intervals and bled into the helium leak detector. The presence of helium in the loop will indicate the failure of a test element; each test element will contain a small pressure of helium to signal a failure.

Orders have been placed for all materials needed to fabricate the test loop. Fabrication is in progress on the test section, heat exchanger and the first test element. The blower will be shipped by the vendor about 15 August. The loop is scheduled to go on line in September.

VI. IN-PILE LOOP TESTS (Task 43-92)

Work was started late in May on the in-pile loop test program. The objectives of the first test will be to verify calculations and design methods and to check the heat transfer and fluid flow characteristics of the reference GCRE-II geometry. The conceptual design of the first element is complete, and a layout drawing is being made. The element is scheduled to go in-pile in December.

The flux depression caused by the loop piping and test element was investigated. The calculations, based on the BMI-16 loop and the test element dimensions and materials, were run by using the P_3 code on an IBM-650 computer. The peak-to-average flux ratio in the test element was calculated to be 1.2. The unperturbed-to-element average flux ratio is 1.86.

Calculations were made using COP (Core Optimization Program Code developed for IBM-704) to determine the fluid flow and thermal characteristics of the first in-pile test element. The calculations were made to evaluate the effects on the nominal peak surface temperature of gas inlet temperature, gas temperature rise, heat generation rate, and coolant channel size. The element will be designed to operate at the GCRE-II peak temperature. The results of the calculations are being analyzed.

VII. NEUTRONICS (Task 43-93)

The major portion of the task effort was devoted to parameter optimization studies, using PDQ and SNG codes on the IBM-704 computer.

The reactivity worth of the central island of the reactor was shown to be essentially the same as that of an equivalent amount of fuel.

Neutron energy spectra were obtained for such design conditions as adding beryllium carbide, flooding with boron and water, and placing control elements in the reflector. A number of problems evaluated the effects of variations in fuel loading, radius, void fraction, and metal fraction.

Calculations indicated that flooding the reactor with a saturated, aqueous solution of ammonium pentaborate produced a negative reactivity sufficient to permit considerable confidence in the shut-down technique.

An evaluation of the use of zirconium hydride (in place of beryllium) as the reflector indicated that little change in reactivity resulted from this substitution. Less lead is required in the shield if zirconium hydride is used. However, the final choice of reflector must be based on cooling and fabrication considerations.

Neutron cross-sections for various fission products were prepared for accurate predictions of fission-product poisoning.

Studies demonstrated that a reduction in core size does not entail an increased fuel inventory, because the critical mass is essentially independent of radius in the range of moderator-to-fuel atom ratios under consideration.

VIII. DESIGN AND FABRICATION OF A CAPSULE FOR IRRADIATION OF FUELED
GRAPHITE SPECIMENT - BMI TASK I-I (Task 43-53)

Work on this task is detailed under Section II, Capsule Tests,
Task 43-61.

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LETTER	DATE	CHANGE	BY	CHK'D
A	7/2/59	ONE V-GROOVE WELD TO SEAM WELD PER E.O. #N-00300	Fiscus	UPHOLM 7/1/59

0-422050

COPY A

FIG. II-1

AR.	—	WELD ROD	INCO #702 PER MIL-R-5031 CLASS
1	0-422055-4	BLOCK - YELLOW GRAPHITE	
2	0-422055-2	BLOCK - RED GRAPHITE	
1	0-422054	WELDING CUP	
1	0-422053	CAP - END	
1	0-422052	BODY - CAPSULE	
2	0-422051	SANDWICH - HONEYCOMB	
2	-4	.3125 DIA x .026 WALL TUBE	INCONEL 702
6	-2	.026 SHEET - INCONEL 702	
NO REQ'D	PART NO	DESCRIPTION	MATERIAL

APS-M2.35
APPLICABLE
AEROJET SPECIFICATIONS

NO. REQ'D	NEXT ASST.	MODEL	CHECKED	DATE	9/18/59
HEAT TREAT	FINISH	STRESS	6/26/59		
TOLERANCE UNLESS OTHERWISE NOTED					
LINEAR	.XX	± .03			
TOL.	.XXX	± .010			
ANGULAR TOL.					
INDICATES SURFACE ROUGHNESS					
FINISH PER MIL STD 10 UNLESS OTHERWISE NOTED					
SUB-CAPSULE ASSY. - II-2C			AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA		
SIMILAR TO			SCALE	2	DWG. SIZE
RELEASE DATE			CAL. WEIGHT		
6/26/59			ACT. WEIGHT		
MATERIAL			0-422050		

3. TIG. WELD PER SPEC. APS-M2.35 USING INCO #702
NO EQUIV.

2. WELD SYMBOLS PER JAN-STD-19

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

FIG. II-3
EXPLODED AND ASSEMBLED VIEWS OF INSTRUMENTATION DEVELOPMENT UNIT.

GCRE-II THERMAL CYCLING LOOP

FIG. II-4

TABLE II-1

PRINCIPLE CHARACTERISTICS OF THE GCRE-II REFERENCE CORE

I. Thermodynamic Design Specifications		
A.	Outlet Gas Temperature	1400°F
B.	Maximum Metal Temperature	1750°F
C.	Coolant	Nitrogen + 0.5% Oxygen
II. Reference Design Criteria		
A. Thermodynamic Criteria		
1.	Reactor Power	4 MW
2.	Coolant Pressure	400 psia
3.	Inlet Gas Temperature	840°F
B. Fuel Element		
1.	Overall Length, Including Outlet End Reflector	34½ inches
2.	Fuel Bearing Length	31 inches
3.	Shape	Hexagonal, 3 in. across flats
4.	Number of Coolant Tubes/Element	7 or 19
5.	O. D. of Coolant Tubes	3/16 to 1/4 in.
6.	Canning Metal Thickness	0.025 in.
7.	Approximate Number of Elements	75
8.	Approximate Diameter of Core	25 in.
C. Reflector		
	Side	4½ inches, Be or Zr H _x
	Inlet End	4½ inches, Zr H _x
	Outlet End	3½ inches (included in fuel element), BeO
D. Fuel Element Materials		
1.	Moderator and Fuel - Graphite containing about 7 w/o highly enriched uranium	
2.	Graphite Coating - Silicon Carbide	
3.	Carbi Canning - Inconel 702 or Hastelloy X, 0.025 in. thickness	

TABLE II-2

GCRE II - WEIGHT SUMMARY

<u>Case</u>	<u>Tolerance</u>	<u>Reactor Package Weight (Tons)</u>
Reference reactor	5 mr/hr	25.4
Reference reactor	ML-1	20.6
ZrH Reflector replaces Be	ML-1	19.4
Power reduced to 400 kw (e) at 25% efficiency	ML-1	16.3

ML-1

I. INTRODUCTION AND SUMMARY

The nomenclature of the Portable Gas-Cooled Reactor (PGCR) as reported in the previous semi-annual report, IDO-28533, has been changed to ML-1. This indicates that this is the first of the Army's mobile (M), low power (L) nuclear plants for electrical generation.

During the period from January to July 1959 final design was started on the ML-1 power plant. Bid packages were prepared for several of the long-lead components, including turbine-compressor set, recuperator, tungsten radiation shields, pressure vessel, and nuclear instrumentation. Criticality of the ML-1 critical experiment at BMI was achieved in June with a loading of 59 fuel elements of the IA design.

In June, the ML-1 program was given a major redirection. The present military characteristics for this power plant require that the maximum package should not exceed 15 tons and that the height of the power plant mounted on each M-172 trailer should not exceed 132 in. In analyzing these requirements, it appeared that they could not be met utilizing the existing core size. A major reactor redesign was started, based upon the use of a shortened IB fuel element. In addition to the core changes, many modifications were required in other components and in the structural design.

The ML-1 Preliminary Hazards Summary Report, IDO-28537, was issued in April, covering operation of the ML-1 power plant at NRTS. This report received preliminary approval from the Reactor Hazard Evaluation Staff, (RHES).

II. AUXILIARIES

A. FACILITY DESIGN CRITERIA (Task 55-100)

The information needed to determine the ML-1 facility requirements was compiled from a number of sources including such external organizations as ARM, IDO and ERDL. These requirements dictated the facility concept and are the basis for the design criteria. (Layout of the facility is shown in Figures III-1 and III-2).

The ROM costs of facility design and construction were estimated; and schedules were prepared, based on the required starting date for installing the power plant at NRTS. The facility requirements, description, cost estimates, schedules, and drawings were incorporated in "ML-1 Facility Design Criteria", AGN TM-349.

The degree of ML-1 dependence on support facilities was studied. To provide a basis for this study, certain maintenance requirements were assumed. It was also assumed that failures would occur in a critical component of either the reactor or the power conversion equipment. Mechanical design and radiation hazards were considered in analyzing disassembly-assembly of such components. The results of this study were compiled in "ML-1 Maintenance and Repair Problems", AGN-TM 356.

B. SHOCK MOUNTS (Task 55-200)

1. Scope

The shock isolation design covers four phases: Analysis, System Design, Procurement and Testing.

2. Analysis

The available data on dynamic environment cover a wide range of shock spectra. (Appendix A) Statistics on occurrence of shock loads in the literature ("Shock and Vibration Bulletins" and ASTIA AD-204095) on various modes of transit are insufficient in scope and load conditions to serve as adequate guides for the ML-1. However, the preliminary design criteria were established, based on various military specifications

and available data. The versed-sine pulse was selected for transit shocks, based on railroad "humping" (standard friction-type draft-gear) and aircraft crash landing conditions and records. (The equations for versed-sine pulse are included in Appendix B). The shock durations were specified with sufficient conservatism, except as otherwise specified in MIL-A-8421A.

The simplified version of a full-scale mock-up for aircraft loading tests, scheduled for completion in September-October 1959, will be used to determine the dynamic environment and the preliminary shock isolation system requirements. Preliminary test procedures have been prepared, and detailed test procedures and requirements are in preparation. The analysis of load conditions for shock mounts during ramp loading operations is 90% complete.

3. System Design

The current design concept of the shock isolation system consists of 32 elastomer isolators and an elastic tie-down system. The maximum load transmitted to the power plant will not be over 5 G for the specified shock loads.

a. Elastomer Isolators

The isolators will be mounted under the base of the structure primarily for vertical shock isolation, and secondarily for isolation of steady and random vibrations during transit or operation. (The isolator design is shown in Figure III-3).

The non-linear characteristics of the elastomer design are based on a tangent formula, as follows:

$$F = \text{Force} = k_0 x + ax^3$$

Where k_0 = initial spring constant, lb/in.

x = deflection, in.

a = design constant, lb/in.³

The load vs. deflection, natural frequency and spring constant of the isolators are plotted in Figures III-4 and III-5.

An alternative design with non-linear stainless steel springs (MET-L-FLEX) is also being considered. Either design will be less than 6 in. high and weigh less than 1400 lbs.

b. Elastic Tie-Down System

Horizontal shock and vibration in transit are mainly isolated by the built-in elasticity of the tie-down system for use in aircraft, railroad flat car, ship or barge. The details of the design of this system are described in the pertinent sections. The load-deflection, natural frequency and spring constant of the tie-down system are plotted in Figures III-6 and III-7.

4. Procurement and Acceptance Tests

A specification (AES-F-1) based on the preliminary design of the shock isolators, was prepared and issued for advanced quotations at the end of May 1959. Delivery will require 6 to 10 weeks.

The acceptance test fixtures for shock isolators are being designed, and test procedures are being prepared.

5. Tests

The shock isolation system will be tested on the ML-1 mock-up. Data will be obtained on shock transmission, static and dynamic deflections, natural frequencies, system stability, and safety devices. No attempt was made in the design of the mock-up test to simulate the component vibration, radii of gyration and mounting details for the ML-1 plant.

C. FIELD APPLICATIONS (Task 55-300)

1. Transportation

Extensive studies were conducted on the applicable transport media as well as tiedown criteria for transport. Loading and transport manuals are being prepared for each of the vehicles under consideration. These manuals are based on the results of studies by Aerojet as well as information compiled from governmental and commercial sources. The ML-1 is being designed for land, sea, and air transport. The transport aircraft are the C-130, C-124 and C-133.

The tie-down criteria for each of these aircraft are based on MIL-A-8321A specification for aircraft emergency landing and flight conditions. The specification outlines the following shock loads:

FORWARD (Along horizontal longitudinal axis) 8 G

AFT (Along horizontal longitudinal axis) 3 G

LATERAL 1.5 G

VERTICALLY (Upward) 3 G in flight

VERTICALLY (Downward) 4 G (emergency landing)

Maximum tie-down point loadings, deck bearing loads, compartment loads, center of gravity locations, etc., were calculated in accordance with the applicable Air Force Technical Orders (Figure III-8).

In view of the grid pattern of aircraft tie-down fittings and the large number of tie-downs used for restraint at the 8 G forward load, a load distribution system was designed to insure that the rated capacity of any single tie-down fitting would not be exceeded. As a result, a nylon rope/pulley system will be placed between the package and the standard aircraft tie-down devices. The nylon rope will be a single, continuous

member for each package, threaded fore and aft through the pulleys to provide 16 restraining strands per package. (Standard tie-down straps will loosely span the pulley system in parallel with the nylon rope to provide back-up restraint in case of rope failure) This system will uniformly distribute the tension per strand. For cases involving the use of many low-capacity aircraft fittings, it will be necessary to divide the loads equally between several tie-down fittings. This will be accomplished by using a secondary load distribution system.

In addition to providing the primary load distributing system, the nylon rope will serve as a shock absorber. A package displacement of about 8 in. is required to reduce an 8-G forward load to 5 G's. This will be accomplished by making use of the inherent elasticity of the nylon rope.

A detailed presentation of the proposed tie-down system was made to the Base Air Freight Officer and Loadmaster at Travis AFB. There was general agreement that the proposed system was adequate.

The primary means of land transportation will be the M-172 trailer. The ML-1 can be loaded as a single, coupled unit. A height limitation of 132 in. is imposed upon the package by foreign transport requirements. Available trailer tie-down devices will be used to secure the package. The nylon tie-down system will be used for aircraft and rail flatcar tie-down. Short duration vertical ("humping") shocks in rail transport will be absorbed by the elastomer shock mounts.

2. Mock-Up Packages

Full-scale steel and concrete mock-ups of the ML-1 reactor and power conversion packages have been designed, and construction drawings are completed. These mock-ups simulate the external dimensions and configurations, overall weight, center of gravity location, the tie-down locations, and shock mounting of the ML-1 (Figures III-9 and III-10).

The mock-ups will be used for aircraft, trailer, and railroad loading studies, as well as for dynamic shock tests to evaluate the tie-down shock mount system.

3. Field Installation

Preliminary sketches have been prepared for a lightweight, cold weather shelter. The shelter will be an insulated nylon tent about 1700 ft² in. surface area. The tent will be large enough to accommodate the ML-1, a 45-kw generator, a mixing tank, and water treatment equipment. A partial floor of wood will be provided to expedite maintenance and reduce heat losses. The calculated heat loss at a -65°F ambient temperature is expected to be about 1.2×10^5 Btu/hr. Portable heaters will be provided to maintain the shelter temperature at + 50°F. The heaters weigh about 400 lbs. each.

D. FLUIDS PROCESSING (Task 55-400)

The preliminary design of the moderator clean-up system is complete. The moderator system initially will be filled with high-purity water from the make-up water system. Low corrosion concentrations of 1 ppm, or less, will be maintained by continuously by-passing 3 gpm through one of two parallel demineralizers.

Laboratory studies are being made to determine the feasibility of manufacturing nitrogen directly from the air. Such a system would supply nitrogen for the system without weight and complexity of commercially available, packaged nitrogen producers.

Preliminary design of the make-up water system is underway. All water used for the neutron shield tank and moderator system will be demineralized to minimize corrosion and neutron activation of impurities. The neutron shield water must contain about 1 wt % boron concentration before it enters the tank in order to meet nuclear safety requirements. Accordingly, filtered, demineralized water will be mixed with sufficient ammonium pentaborate in 200-gallon batches in a small mixing tank, and transferred to the neutron shield tank.

Preliminary design of the reactor core drying equipment is 90% complete. The reactor interior surface will be dried in two stages after reactor core flooding and draining. The first stage consists of passing pre-heated air at 650°F through the reactor and exhausting the humid air. When the gross water is removed, the pipe system will be closed, and the final traces of moisture will be removed by passing the gas through a molecular sieve desiccator.

A test program was conducted to determine the effects of flooding the reactor with a 3% ammonium pentaborate solution. Principal conclusions were: 1) Gross amounts of pentaborate solution would be retained in the fuel element insulation after the reactor gas passages were flooded; and 2) the boron components will not volatilize under elevated temperature. Based upon these tests, fresh water will be used for reactor flooding.

III. POWER CONVERSION EQUIPMENT

A. DESIGN AND SPECIFICATIONS (Task 56-100)

This task is responsible for design, procurement, test and installation of the major components for the power conversion package. As a result of the 15-ton weight limit, the GTTF machinery arrangement has been modified to take advantage of the weight and efficiency improvement provided by an in-line turbine-gearbox-generator configuration. General component design has been investigated; and firsthand inquiries were made to find the problem areas of GTTF equipment, from both the vendors and from examination of the equipment at Ft. Belvoir. Suggested solutions to known problems have been provided in specifications for various components. The components included in this task have delivery dates compatible with the schedule which specifies an April 1961 criticality date for the ML-1.

1. Recuperator

Recuperator design problem areas have been investigated. The vendor for the GTTF recuperator was visited to acquire a firsthand understanding of the design and manufacturing problems encountered in producing the first (GTTF) unit. Several conceptual designs were produced to investigate methods for reducing transient-induced thermal stresses, for improving flow distribution, and for reducing losses in the headers. The final recuperator design will include expansion-compensating supports and a tie-down feature. A specification including three suggested conceptual designs was completed. The bid package for the recuperator was sent to several manufacturers of heat exchangers for quotation.

2. Pre-Cooler

Preliminary investigation of design problem areas is complete. The technical specification is in the final draft stage and includes general configuration, suggested type of extended surface, and limiting dimensions. Raising the pre-cooler above the skid roof on jacks has been considered as a means of improving performance. The additional plenum space created would produce better flow distribution, and permit more freedom in arranging components on the skid; but this complication may be unnecessary.

The pre-cooler design probably will incorporate aluminum structure, perforated plate fins, six independent fans, and some means of controlling the minimum nitrogen exit temperature. The nitrogen inlet temperature rises rapidly during a reactor scram for full-load and must be controlled at a safe level to protect the aluminum headers and heat-transfer surface. Current information shows that such control is possible; otherwise it may be necessary to design for higher temperatures and heavier materials.

3. Turbine

Bids have been received for a turbine-compressor set, based on "Turbine-Compressor Set Specifications for GTTF Spare Parts Contract", AGN TM-353. An analysis of these bids eliminated all but three potential suppliers. The plants of these three bidders were visited and evaluated, and their financial and contractual positions were examined. A vendor will be selected on the basis of this information, and the purchase order for the spare turbine-compressor set for GTTF will be presented to the Army at Fort Belvoir for approval during July 1959.

The Stratos 560 (GTTF) turbine design has been evaluated by AGN, by AGC's Turbo-Machinery Division, and by two independent consultants. Among the conclusions were:

- a. Performance probably will be below that required by GTTF.
- b. Internal leakage is high, particularly from the thrust balancing piston.
- c. The external shaft seal will probably require replacement more often than every 10,000 hr.
- d. Maintaining clearances for bearings and labyrinths may be difficult.

In addition, the applicability of Stratos' 560 unit to ML-1 was examined. It was generally concluded that ML-1 requirements necessitated, among other things, re-design of the casing and substantial development to obtain the efficiencies stipulated in the purchase specification.

4. Gearbox

Preliminary investigations of design problem areas are complete. Discussions with the GTTF vendor indicate that the use of single reduction and light casing design will make possible substantial weight savings and an increased efficiency. Increasing generator speed to 3600 rpm makes single-stage speed reduction appear feasible. Specifications will be released for bid during August 1959.

5. Generator

Preliminary investigations of generator design problems are proceeding. A 3600-rpm unit, using standard electrical insulation, statically excited, and incorporating a starting motor, appears both desirable and feasible. Specifications will be released for bid during July 1959.

B. MATERIALS RESEARCH AND DEVELOPMENT (Task 56-200)

This task was initiated in February 1959 to determine the materials and fabrication techniques needed to build a power conversion unit capable of a 50,000-hr operational lifetime when exposed to the highly-oxidizing ML-1 reference coolant gas. It was planned to use much of the experience obtained in fabricating the GTTF equipment, recognizing that the GTTF equipment was designed for use with a coolant gas of low oxidizing characteristics. (The GTTF gas is essentially pure nitrogen). The oxidizing characteristics of the ML-1 coolant gas would shorten the lifetime of the GTTF recuperator. Work was started to establish fabrication techniques for oxidation-resistant materials, an approach justified by later corrosion tests.

In attempting to substitute an oxidation-resistant material it was difficult to find compatible brazing materials. A development program was started to determine the proper combination of materials and brazing techniques needed to fabricate the recuperator pin fin core sections. The fabrication parameters needed to produce the best joint were established after making several sets of samples. The effects of gas corrosion, and the comparative strengths provided by various fabrication techniques are being tested on these samples.

The use of copper-cored nickel wire for the pin material is being studied. If a pin can be made from this material, it will increase recuperator effectiveness by about 10%.

1. Brazing Program

The Armco-iron pins used in the GTTF recuperator core will not give a satisfactory lifetime in the ML-1 coolant gas at maximum recuperator temperatures. Type "A" nickel, which has thermal conductivity similar to Armco-iron, was selected as the reference material for the pins.

The purpose of the brazing program was to determine if Type "A" nickel could be brazed successfully with a high-temperature alloy and the time-temperature cycle required for a 500-pound heat exchanger core. Copper was selected as the most desirable brazing material. Two silver-base alloys and one nickel-base alloy were selected as back-up materials. The copper braze was applied in powder form, using a Ferrotherm Co. proprietary technique. Typical brazed specimens are three pins wide by two inches long (Figure III-11). The following conclusions were drawn from the tests:

a. The nickel-base metal has a high affinity for liquid copper, and the copper braze material dissolves into the parent metal, forming a completely unbrazed joint when using 0.001 to 0.002 in. of braze material. A good joint is obtained with about 0.004 in. of braze material (Figure III-12).

b. Because of the high affinity of nickel for copper, the braze material flows only a short distance. Adequate wetting was observed in the samples containing 0.004 in. of copper braze (Figure III-13). It may be difficult to get a good back braze on the heavier closure bars.

c. The surface of the braze is slightly roughened, but this finish should be acceptable with the present passage size.

d. High temperatures and long dwell time produced no noticeable erosion.

e. Successful brazes were achieved with the following time-temperature cycle in a brazing atmosphere of dry hydrogen:

- 1) Heating rate - 60°F/hr
- 2) Brazing temperature - 2050 to 2150°F
- 3) Dwell time range - 1/2 to 3 hr
- 4) Cooling rate - 100°F/hr

The back-up silver braze alloys and nickel-base braze alloy were carried through only one of the three test phases. Only silver alloy AMS NO. 4772 was carried through the entire program. This silver alloy has the following characteristics:

- a. Composition - 54% silver; 40% copper; 5% zinc; 1% nickel; and 0.2% lithium
- b. Melting range - 1325 to 1575°F
- c. Cost - about 10 times that of copper
- d. Used in foil form

Silver alloy brazing produced acceptable results and would be practical for brazing copper-cored pins. The results of silver brazing are summarized below:

a. The nickel-base material has a lower affinity for silver than for copper, and since the alloy is 40% copper, a greater amount of alloy is needed than would be necessary with 100% silver. A good braze for all joints is obtained with about 0.003 to 0.004 in. of alloy (Figure III-14).

b. The silver alloy wets better and flows better than copper.

c. The brazed surface is relatively smooth.

d. Successful brazing was achieved with the following time-temperature cycle using dry hydrogen as a brazing atmosphere:

- 1) Heating rate - 60°F/hr
- 2) Brazing temperature - 1680°F
- 3) Dwell time - 30 minutes
- 4) Cooling rate - 100°F/hr

2. Corrosion Tests

Gas corrosion tests were conducted to determine the effect of environment on materials and braze joints produced by various time-temperature cycles (Figure III-15). The first gas corrosion tests were conducted in 200-psi reference GCRE atmosphere at 900°F on pin fin core samples of the GTTF recuperator materials. The samples (Figure III-16) were continually exposed to fresh gas. The table below shows the weight increase of these samples after a 2000-hr exposure:

GAS CORROSION TEST

(Weight change in grams for GTTF-type pin fin recuperator core in an atmosphere of nitrogen plus 0.5% oxygen at 200 psi and 900°F)

<u>Sample No.</u>	<u>Original Wgt.</u>	<u>Final Wgt.</u>	<u>Net Change</u>
1	4.4935	4.5555	+0.0620
2	3.8702	3.9135	+0.0433
3*	3.9482	3.9587	+0.0105
4	3.9694	4.0380	+0.0686
5	3.9705	4.0090	+0.0245
6	4.2637	4.2952	+0.0315

*Exposed for only 1000 hours

New test sections were then set up, and core samples were exposed at 900°F in 300 psi of reference nitrogen atmosphere and air. (The air tests are being run to establish the feasibility of using air as a coolant). The air corrosion tests have been on-stream for about 1250 hr. Most of the samples showed small weight gains at the 1000-hr check. The weight gains may be due to excess copper braze materials rather than the parent metal. Weight gains were negligible on the silver and nickel alloy brazed samples.

Three samples of the aluminum pre-cooler material, exposed in a 500°F zone of the 300-psi air pressure tube, showed no weight gain nor evidence of attack.

3. Copper-Cored Fin

A copper-cored nickel fin, now being formed, is expected to increase recuperator effectiveness by 10 to 15% with a corresponding decrease in the size of the equipment. A sample fin

showed no cracking in the foot when checked with dye penetrant. (Figures III-17 and 18).

4. Future Work

Work will continue on development and brazing of the copper-cored pin. Pull tests will be made on small pin fin samples to determine the comparative strengths produced by variations in methods. Gas corrosion tests will determine the effect of exposure on the strength of joints.

C. SYSTEMS ANALYSIS AND GTTF EXPERIMENTAL PLANNING (Task 56-300)

Hand calculations were used to perform steady-state cycle analyses during preliminary design of ML-1, and then an IBM-704 program was developed for detailed cycle analysis. Current work includes analysis of the gas injection system during plant load increase.

Before the completion of preliminary design, an IBM-704 program was set up to investigate system performance at various loads and ambient temperatures, the effect of component changes on system performance, and the start-up requirements of the system. Initial computer runs were based on a reactor design with a pressure drop of 13 psi at full load; and runs were made at several loads and ambient temperatures. Results are shown in Figures III-19 through III-22. The design changes required to meet the weight limitation of the 15-ton package were considered in investigating the effects of higher reactor pressure drops. Results of these investigations, on a preliminary basis, are shown in Figure III-23. The self-sustaining temperature at various fractional speeds will be determined to define the start-up requirements of the system.

An approximate analysis showed that an average gas injection flow rate of 9 lb/sec for three seconds would be required under load increase. A more detailed compressible flow analysis is being conducted to determine the correct size of valves and pipes needed to assure free-flowing operation.

The experimental planning phase of this task reviewed the GTTF open-cycle procedure to insure early acquisition of the data considered vital to ML-1.

D. ELECTRICAL EQUIPMENT DESIGN (Task 56-400)

1. Switchgear Center Lay-out and Single Line Diagram

A preliminary lay-out has been made of all electrical switchgear in the power conversion skid switchgear center. The lay-out is based on standard electrical switchgear dimensions and is complete except for mounting and wiring details. A specification has been written, and a preliminary electrical power single line diagram has been completed.

2. Auxiliary Equipment Demand

A preliminary tabulation of the load demand by the auxiliary equipment was prepared for different stages of operation. The rated load was assumed to be drawn from all equipment as soon as it went into operation. The figures stated in this tabulation will require adjustments because of having assumed rated load, and because auxiliary motors and heaters may be added, deleted, or changed in rating during the final design.

3. Electrical Switchgear Manufacturers

Available components that would meet specifications were discussed with major electrical switchgear manufacturers. Since the current rating of the high-voltage switchgear needed for ML-1 is lower than standard for the industry, custom built high-voltage, low-current components are being investigated. Current and voltage ratings for the low-voltage switchgear needed for motor control centers conform to standard industrial ratings. Custom built high-voltage switchgear and the adaptation of aircraft 400-cycle, low-voltage switchgear to ML-1 60 cycle requirements are being investigated in an effort to save weight and volume on the power conversion skid.

4. Operation of Motors on Low-Frequency Power Supply

Calculations were made to determine the minimum frequency at which motors in the auxiliary equipment could be switched on and used during startup of the turbine-compressor set. The calculations show that motors driving a fan load can be switched on when the frequency is at 1/10 rated value. Motors driving machine loads cannot be turned on, until 1/3 rated frequency is reached. The generator will require little or no voltage regulation during startup.

5. Electrical Switchgear Reference Design

In the reference design for electrical switchgear, the output of the three-phase, 2400/4160-volt, wye-connected, 60-cycle, 400-kw generator is fed through a circuit breaker to the external load and parallel stations. The power for the plant auxiliary equipment is tapped off at the generator side of the oil-filled circuit breaker and is transformed to 480 volts, three-phase. Nearly all of the auxiliary motors and heaters will operate at 480 volts. A few small single-phase motors and all lights and outlets will be taken from a small, 480-volt:120/240-volt transformer. A transfer switch located on the secondary side of the 4160/480-volt transformer allows auxiliary equipment to be driven either by the main generator or by an auxiliary power supply. Commercially available equipment appears to meet the general shock and vibration requirements for the ML-1 plant. Tentative plans have been established for sequencing electrical equipment for different types of shut-downs during operations at Idaho. In general, the plan is to drop the ML-1 to the minimum power compatible with the malfunction in the power plant system. An auxiliary power supply will be used when necessary during acceptance testing at Idaho and until it is proven that the plant can operate independently of auxiliary power.

6. Generator Circuit Breaker

An evaluation is being made of a three-phase, 4160-volt, 400-kw vacuum switch circuit-breaker in place of the oil-filled circuit breaker, to save weight on the power conversion skid. The vacuum circuit breaker weighs 300 lb less than the oil-filled circuit breaker. The vacuum breaker can carry 75 amp rms continuously at a power factor of 0.8 lagging, and interrupt fault currents up to 335 amp rms with 425 amp inrush.

7. Surge Calculations

Calculations were made of the amount of capacitance and inductance to be added to the system output to protect the generator from lightning or other electrical surges. A $1/2 \mu$ f three-phase surge capacitor must be added across the output lines, and a three-phase inductor, with 170 μ henries per phase, must be added in series with the output line.

8. Instrument Transformer Hook-up

The wiring installation has been designed for the potential transformers and current transformers. The potential transformers will be hooked up open delta primary and secondary with the "V" point of the open delta grounded on the secondary side. The transformer voltage ratio will be 4160/120 volts. The above arrangement is compatible with all meters and relays provided that the power factor meters are supplied with a wye box and are not internally grounded to the case. The current transformers will be wired with a grounded wye configuration. The current ratio will be 100:5. This arrangement is compatible with all meters and relays.

9. Motor Control Center Elementary Diagram

The elementary diagram has been completed showing all power drawn from the 480-volt bus on the power conversion skid. The diagram includes all circuits and control elements of the equipment. It emphasizes the control elements of the circuit and their functions, as distinguished from the physical arrangement of the conductors, or other components.

E. MECHANICAL DESIGN (Task 56-500)

Layout drawings have been prepared for other equipment on the power conversion skid, based on outlines for representative designs. One set of drawings was based on a "U" arrangement of the power circuit equipment similar to that used for GTTF. Drawings were also prepared for an "in-line" arrangement, with the generator on the opposite side of the gear box from the turbine-compressor set (Figures III-24 and 25). This set of drawings met the new requirement of an overall package height of 132 in. The "in-line" arrangement offers several advantages, particularly with regard to gear-box simplification, and has been adopted for all future work.

Studies were made of temperature distribution in the pipe between the recuperator and the reactor. Weight estimates were made, and centers of gravity were computed for various arrangements. This information was needed for the design of shock absorber system and transportation tie-down system. A 1/8-scale, wood model was made of the power conversion equipment.

Two design concepts are being analyzed for the power conversion skid structure. The first scheme uses a truss extending from the floor to the base of the pre-cooler on each side of the skid, with cross trusses at the panel fronts to support the heavy power circuit equipment. The other scheme uses a shallow truss immediately adjacent to the heavy equipment, supporting it without cross-trusses. In this scheme, the pre-cooler is supported on four posts extending from the two lower trusses. The two schemes will be compared from the standpoints of weight and simplicity.

A conceptual design was prepared for the end fittings of the removable members for the main structural truss. Preliminary design has been completed for the folding hood for the pre-cooler air inlet.

F. PROCESS DESIGN (Task 900)

Mechanical flow sheets are complete for all power conversion equipment. These include one overall flow sheet plus a more detailed flow sheet for each of the four circuits; power, gas transfer, drying, and lubrication. The flow sheets show temperatures, pressures, flow rates, and vessel capacities. Each piece of equipment, valve, instrument, and pipe section is identified by a code symbol.

Preparation of the power circuit flow sheet (Figure III-26) was largely a matter of recording requirements of other tasks. It will be subject to change when the major pieces of equipment are procured and final temperatures, pressures and flow rates are available.

The lubrication circuit (Figure III-27) was prepared only for reference, inasmuch as the supplier of the turbine-compressor set will design the system.

Most of the effort on this task was concentrated on the drying circuit (Figure III-28), and the gas transfer circuit (Figure III-29). Process descriptions and preliminary equipment specifications were written for these circuits. Preliminary designs were prepared for specialized equipment such as the admission gas heater, charcoal filter, accumulator and receiver tanks, drying gas heater, and desiccator. Inquiries have gone to vendors for information on such commercial items as the two gas compressors, mechanical filters, expansion joints, drying gas blower, and valves. A detailed cost estimate was made.

Final purchase specifications are being written for the gas transfer compressor, the solenoid valves, blocking valves and check valves.

IV. INSTRUMENTATION AND CONTROLS

A. NEUTRON MONITORING (Task 57-200)

This task is responsible for providing instrumentation and controls for safe, reliable nuclear reactor operation. A block diagram of the nuclear control system illustrates the 7 safety and monitoring channels (Figure III-30). The amplifiers are transistorized to minimize weight, volume, and power requirements. The reliability of transistorized instruments will also prevent spurious scram signals from shutting down the reactor.

The reactor startup procedure is designed to prevent operator error. All rods are interlocked to prevent withdrawal before the following conditions are met:

- the personnel safety switches are energized;
- a source count-rate is detected;
- all the rods are down and seated in the reactor; and
- the shielding tank water is at the proper level.

Mechanical and electrical interlocks prevent the withdrawal of more than one control rod at a time. A control schematic is about 80% complete.

Specifications were written for the nuclear instruments and were submitted to vendors. Their quotations are being evaluated.

A preliminary control console layout was made to illustrate the location of all operating nuclear controls. All controls and calibrating circuits are located within reach of the operator. Meters for monitoring startup count-rate and period, intermediate log power and period, high level power, and ion chamber voltage will be mounted directly in front of the operator on the control console.

The following abnormal conditions will cause the reactor to scram:

high power level;
periods faster than one second;
power failure; and
abnormally high reactor outlet temperature.

A manual scram also will be provided.

B. HEALTH PHYSICS (Task 57-300)

This task includes the health physics instrumentation and equipment needed for the IDO ML-1 test site as well as the field instrumentation and equipment which will normally accompany an operational ML-1. Only the latter will be required to meet MIL specifications on shock, vibration, temperature, and other field conditions. These specifications are covered in other sections of this report.

1. Instruments and Equipment for IDO Test Site

Certain health physics requirements are peculiar to the test site. Each of the major areas of the site will require health physics equipment. During this report period health physics design criteria have been written.

General specifications have been written for the equipment needed for the personnel change building. This equipment, to be procured "off the shelf", includes portable survey equipment, special clothing, and fixed radiation monitors. The portable equipment will be used by personnel entering the test building.

The gas and particulate monitor for the test building will be located near the power conversion skid to sample gas through flexible hoses from each of the six pre-cooler fans. The gas will enter an isokinetic sampling probe and will be directed to a shielded scintillation counter measuring the integrated radiation from a moving filter paper, and then into a graphite trap where it will be monitored by a second shielded scintillation detector. A lightweight boom will support the sampling probes above the pre-cooler. The controls and read-out equipment will be mounted in racks in the auxiliary building.

The sensitivities of the gas and particulate counters are as follows:

Gas Counter: This counter will give a minimum net increase in count-rate of 600 cpm in 20 minutes when the I-131 concentration is 10^{-9} $\mu\text{c/cc}$, and the gas flow-rate is 1 cfm.

Particulate Detector: This counter will give a minimum net increase in count-rate of 350 cpm in 3 minutes, when the Y-90 concentration is 10^{-10} $\mu\text{c/cc}$.

and the gas flow-rate is 10 cfm. The equipment will record a maximum count rate equivalent to 3×10^3 Maximum Permissible Concentration for this isotope.

Detailed specifications have been written for the pre-cooler monitor.

The health physics control center normally will be in the auxiliary building. Specifications are being written for the following equipment: one beta-gamma line-operated ratemeter for monitoring the air conditioning filters, one moving-filter type air monitor, two beta-gamma survey meters, and 24 personnel self-reading dosimeters and charges.

2. Field Instruments and Equipment for ML-1 Control Cab

The cab will normally be the health physics control center when the ML-1 is in the field. Ten channels of detectors will be used in the remote area monitoring system. (An instrument block diagram is shown in Figure III-31.) The chambers will be attached to the power conversion skid frame in such a manner that they can be easily removed from the outside. Only one detector will be monitored at a time. The signal and power leads from each detector will enter a junction box on this skid. Specifications are being written for this equipment. In addition, the following equipment will be required:

- 2 each "Cutie Pie"
- 2 each Fast-Slow Neutron Counters
- 4 each Geiger Counters
- 1 each Line-Operated Ratemeter
This will be located in the control cab
and will be used to monitor personnel.
- 2 each Self-Reading Dosimeters and Charger

C. CONTROL ANALYSIS INSTRUMENTATION (Task 57-400)

This task has two purposes: To provide suitable instrumentation and equipment to show that the ML-1 meets design specifications, and to provide sufficient instrumentation for trouble shooting. The equipment will be commercially available items and will not be required to meet military specifications. The complete analysis instrumentation block diagram is shown in Figure III-32.

1. Nuclear information is indicated in the control cab and recorded in the auxiliary building. Standard, null-balance type strip chart recorders will be used to record:

- Log count-rate
- Period from the startup and intermediate channels
- Log-n
- High level safety

Specifications have been written for these recorders.

2. Instrumentation for Power Measurement

The following will be used in determining the characteristics and amounts of ML-1 electrical power output:

Recording wattmeter
Watt-hour meter
VAR-meter
Generator output voltage recorder
Generator frequency recorder

Specifications have been written for the above recorders.

3. Instrumentation for Special Process Measurements

The following information will be recorded:

Error in by-pass valve position
Turbine speed error
Reactor outlet temperature

4. Pressure Measurements

The following pressure measurements will be made:

Input and output turbine and compressor pressures
Pressure drops across the cold and hot side of the
recuperator
Pressure drops across the cold and hot side of the
pre-cooler
Pressure drops across the reactor

Pressure data will be recorded on an oscillograph and indicated on a digital voltmeter.

5. Temperature Measurements

The following temperature measurements will be made:

Coolant temperature rise through the reactor
Inlet and outlet turbine and compressor temperatures
Inlet and outlet temperatures of the pre-cooler gas
Ambient air temperature
Recuperator hot inlet and outlet temperatures
Recuperator cold inlet temperature

6. Special Instrumentation

Pressures, temperatures, flow-rates, and power will be recorded during dynamic tests on an 8-channel direct-writing oscillograph.

A strain gauge indicator and two 20-point switching and balancing units will be available to obtain information from strain gauges located on the pre-cooler, the recuperator and the reactor.

D. DYNAMIC ANALYSIS (Task 57-500)

The analog circuit simulating the ML-1 power plant has been continuously modified and up-dated during the report period. Many runs have been made to test different control schemes or various major loop component designs. Transient temperature data for specifications for the two heat exchangers were derived for various shutdown conditions. The following is a chronological record of computer operations and results for the six-month period.

1. Multiplying Circuits

Most of the multiplying circuits were changed from the mechanical servo-type to the newer thyrite resistor-type, because the latter type is faster and hence more accurate in a complex system. The thyrite circuits respond in the order of microseconds because they are all electronic; but the mechanical circuits have a larger lag time, and hence the computer circuits may introduce instabilities when dealing with a complex system such as the ML-1 simulation. These instabilities may be quite independent of the system dynamics. The simulator (IBM computer) performance improved after this change, and there is increased confidence in the results.

2. Over-Power Runs

Earlier calculations had indicated that the system may be over-pressured after a load increase. It was demonstrated that the drop in reactor temperature is not responsible for the introduction of more than the necessary amount of gas. It was demonstrated that thermal inertia was unnecessary in the ML-1. The lags inherent in the control equipment, valves, sensors, etc., as well as the system time lag between the addition of gas and the change in shaft speed, are the factors that cause over-pressure. A solenoid valve (on-off) was found to be more satisfactory than a proportional feed valve.

3. New Compressor Equations

Two new compressor equations were wired into the computer so that valid results for speed variations from zero to rated value were provided. These modifications were necessary for the projected shutdown and startup tests.

4. Axial Compressor Map

An axial compressor map was installed and compared as an alternate to the radial compressor. The analog showed that the proposed control scheme worked equally well for the two types of compressors.

5. Temperature Transients

Recuperator and pre-cooler temperature transients were

plotted for full load changes, for a scram with the pre-cooler fans powered by an external source, and for a scram with the pre-cooler air flow dropping to free convection.

6. Control Scheme Modification

A modification to the control scheme was proposed to prevent driving the compressor into a surge during a load increase. (The compressor outlet pressure increases, and speed drops during a load transient). The proportional addition of gas to both the low- and high-pressure sides of the compressor does not solve the problem.

7. Time Scale Changes

The pressure and flow circuits were re-scaled so that computer time was 1/20 real time. These runs were made to determine that time constants associated with the computer components, and hence independent of the physical system, were not influencing the results. Valve time constants, the turbine-compressor set shaft dynamics, and pipe volumes were changed; but temperatures were held constant. The high-speed pressure transients were the only interest in these runs. The conclusion was that both added inertia in the turbine-compressor shaft and a small amount of by-pass are needed for stability.

8. Pre-Cooler Temperatures

The pre-cooler temperatures rise to dangerous levels after scrams. The following solutions are being studied: an emergency power supply for the pre-cooler fans, use of a flash evaporator, and control of heat transfer from reactor to pre-cooler. Perhaps this heat transfer can be throttled to protect each component without additional equipment. This study is continuing.

9. Change to IB Fuel Elements

The change to type IB elements in the reactor core will be incorporated into the simulator. The new core has more heat capacity; but much of this capacity is contributed by the ceramic. The dynamics of the plant should not be affected by the change in type of fuel elements because it is not expected that the heat storage in the ceramic can be drawn from on a short-time basis.

E. PROCESS INSTRUMENTATION (Task 57-600)

Four main factors were considered in setting up process instrumentation: the minimum number of instruments consistent with good operational practice; the ruggedness, weight, accuracy, size and reliability of instruments; the parameters to be controlled and how to control them; and availability, especially with respect to special requirements.

It was determined that good operational control could be obtained with the 23 instruments shown in the list on the following page:

INSTRUMENTS TO BE MOUNTED IN CAB

<u>IDENTIFICATION</u>	<u>NAME</u>
1. PI-12	Compressor Out Pressure
2. TIC-12	Turbine Inlet Temperature
3. MIA-11	Pre-cooler Outlet Moisture Content
4. PI-11	Compressor Inlet Pressure
5. TIA-11	Compressor Inlet Temperature
6. SIC-12	Compressor Speed
7. TI-52	Bearing Temperatures
8. PI-52	Lubricating Oil Filter Outlet Pressure
9. FILA-51	Lubricating Oil to Compressor Rear Bearing, Flow
10. FILA-52	Lubricating Oil to Compressor Front Bearing, Flow
11. FILA-53	Lubricating Oil to Gearbox, Flow
12. TIC-51	Lubricating Oil Supply, Temperature
13. PIA-31	Accumulator Pressure
14. TI-31	Accumulator Temperature
15. TI-42	Nitrogen After-Cooler Outlet Temperature
16. PIA-32	Nitrogen Make-Up Pressure
17. MI-12	Nitrogen Drying Moisture Content
18. TIC-71	Moderator Water Temperature
19. TIA-72	Moderator Shield Water Temperature
20. PIA-71	Moderator Water Pressure
21. CIA-71	Moderator Shield Water Conductivity
22. CIA-72	Moderator Water Conductivity
23. FIA-71	Moderator Water Flow

The following parameters are annunciated:

1. VAL-11	Compressor Vibration
2. SHA-11	Compressor Shaft Speed
3. LS-51	Gear Box Oil Level
4. PHLA-51	Lubricating Oil Filter Outlet Pressure
5. TA-53	Lubricating Oil Sump Temperature
6. LS-31	Transfer Compressor Oil Level
7. LLA-71	Moderator Surge Tank Liquid Level

Amplifiers and transmitters will use solid-state components where possible. Transistors and magnetic amplifiers, or a combination of the two, appear to be acceptable. A survey of suppliers of this type of equipment shows that availability will not be a serious problem.

Dynamic analysis of the system indicated that two parameters should be controlled: the reactor outlet temperature, and the compressor speed.

The reactor outlet temperature will be controlled by positioning the regulating rods with a servo motor. This requirement presents no problem since standard components can be used. Compressor speed control, however, is more complicated and requires a special amplifier, not yet available on the market. Preliminary layout and design have started on the speed control system. Compressor speed will

be controlled by regulating the quantity of coolant gas to the reactor by means of a servo-operated by-pass valve between the compressor and the recuperator. In addition to an indicator showing the absolute speed, a high-error and high-speed signal is supplied.

Preliminary layout and design of the graphic panel have started. The main considerations are: space requirements, human engineering aspects, maintenance considerations, physical construction, and graphic design (See Figure III-33).

Specifications have been prepared for about 80% of the instrumentation.

F. POWER CONTROL (Task 57-800)

The power control instrumentation will monitor the output and provide switching for load shifts. Meters will monitor the generator voltage, current, frequency, and power factor. A synchroscope will be used to allow phasing-in of the generator to existing power bus networks. Each of the three phases of generated power will have its own voltage and current measuring systems so phase loads can be balanced. The plant output power as well as the generator output power will be metered. In subsequent units, only the plant output power to the load will be measured, and the instrumentation will be similar to that used in conventional power plants.

1. Location of Relays and Regulators

The optimum location of the protective relays and voltage regulator was studied with reference to weight problems, equipment, compatibility, mechanical vibration and shock problems, cabling problems, and operational desirability. From these considerations, it was concluded that:

a. The protective relays will be mounted on the power conversion skid, with provision for isolating mechanical vibration and shock to minimize cabling and weight problems.

b. From the operational standpoint, the voltage regulator should be mounted on the instrument cab. However, it appears now that only the rheostat control of the voltage regulator will be mounted in the cab.

2. Voltage Regulation Requirements

A small voltage regulator (about 3 kw) is being considered for use with nuclear and process instrumentation. (This will be in addition to the overall, plant voltage regulation). Inquiries have been sent to several manufacturers, and a specification will be written.

3. Load Bank and Load Bank Control

A load bank will be needed for the final test of the power system. A specification was written and is ready for release.

4. Power Instrumentation

Power instruments have been selected. Appropriate potential transformers and current transformers will be used to supply the correct voltage for the instruments since the output of the generator is 4160 volts. The potential transformers will be used in an open delta configuration with a transformer ratio of 4160/120 v. The current transformers will have a ratio of 20/1. Navy high-shock type instruments will be used for this instrumentation due to the environmental conditions. A wattmeter using the Hall generator principle will be used for the DC signal proportional to power output. The instruments will use flush-mounted dials with 250° scales and have an accuracy of 1% or better. The specifications for these instruments are in rough draft form.

Instrumentation requirements are shown on the schematic drawing (Figure III-34).

6. Power Control Panel

A preliminary layout is being drawn of the power control panel for the auxiliary equipment. This panel will include controls and indicators for the motor contractors, heaters and solenoid valves. Integrated into this panel will be sequence of operation and the interlock of necessary equipment, or control of equipment during emergency shutdown. The power control panel circuit has been designed, and components are being selected.

7. Control Cab

The control cab will house the control equipment needed in the operation of the ML-1 facility, including the following: process panel, nuclear panel, power instrumentation panel, power control panel, DC power, lighting, air conditioning, meteorological equipment, etc. The general characteristics of the cab are:

Empty Weight	1200 lb
Rated Shelter Payload	6300 lb
Inside Dimensions	134 in. long
	76 in. wide
	73½ in. high
Outside Dimensions	140-5/8 in. long (overall)
	82-5/8 in. wide
	80-13/16 in. high
Inside Volume	433 cu ft

The preliminary layout and design of the cab are in process. A half-size paper model of the control console has been used to determine the best configuration of the panels. A specification will be written for the control cab when requirements are defined for the operation of the ML-1 facility.

8. Cables and Connectors

Cables are being designed to transmit signals between

the power conversion skid and the control cab. Two cables will be used for this application, each 500 feet long with about 125 pairs of conductors, and fitted with AN-type quick-disconnect connectors. These cables will be designed to meet the environmental conditions set forth in the ML-1 design criteria, and specifications will be written to cover the cable requirements.

V. REACTOR ENGINEERING

Stages of the development of the preliminary design of the reactor were reported in "PGCR Preliminary Design Report" AGN TM-341, dated 15 January 1959, and "ML-1 Preliminary Hazard Report" IDO-28537, dated 22 April 1959. The preliminary designs were revised after the publication of these reports to reduce the overall weight and height of the reactor package to conform to military requirements for the ML-1 power plant. Work now is being carried out on a reactor package weighing 15 tons that will comply with the foreign transport requirements of 132 inches total height when mounted on a standard M-172 trailer.

No effort has been made in the following to describe the intermediate steps in arriving at the present design, since the history of the development of the preliminary designs was covered in detail in the above reports. The present design is described in detail with particular emphasis on progress made during June 1959.

A. CONTROL ROD ACTUATORS (Task 58-100)

This task was begun 1 June 1959 and is responsible for the two types of actuators required for the ML-1 control rods. The reactor requires the use of five shim-scam and one regulating-type actuator. The possible designs of the actuators are dictated by the design of the control rod blades (Task 58-500).

The actuators will be mounted in the wall of the neutron shield tank. Preliminary designs indicate that all the required mechanisms and seals can be housed in a package about 12 x 12 x 6 inches. The 6-inch dimension will be in a horizontal direction, radial to the core, so that a minimum of shielding water will be displaced.

Specifications are being written for the actuator packages and for individual components. If potential vendors do not respond adequately, the information on individual components will permit proceeding with detailed design of the control rod actuators.

B. REACTOR SHIELDING (Task 58-200)

This task is responsible for design, fabrication and testing of the nuclear radiation shielding for the ML-1 reactor. The shielding must provide both operational and shutdown shielding. The operational shielding requirements include attenuating both neutron and gamma radiation to acceptable levels at reasonable exclusion radii, and to prevent radiation damage to ancillary reactor equipment on both the reactor and power conversion package skids. The shutdown requirements are to reduce the nuclear radiation 24 hours after shutdown to levels which will permit shipment of the reactor package, and maintenance of the power plant equipment.

Lead was used as the primary shielding in the initial designs because of its relative low cost and ease of fabrication. The initial designs were based on GCRE-I core size. The radial shielding was lead, and the axial shield was a lead and tungsten composite. Subsequent designs used additional amounts of tungsten to provide better low-weight shielding. As the design progressed, certain military specifications were found to be unattainable in terms of the original core design without seriously compromising the maximum permissible radiation dose level. In particular, the original core design required excessive shielding weight and vertical package height; therefore, the core's axial dimension was shortened to achieve a more compact system.

1. Shielding Specifications

The specifications under which the ML-1 Reactor package is being designed, insofar as they affect shielding, are as follows:

- a. Maximum package weight, including all non-removable components associated with the reactor: $15 \pm \frac{1}{2}$ ton. (The range representing the accuracy of weight calculations.)
- b. Maximum overall height of the reactor package - measured from road surface to highest point on trailer-mounted package on the Army M-172 trailer: 132 inches.
- c. Radiation dosage at the cab of the prime mover of the above semi-trailer to be 15 mr/hr one day after reactor shutdown following extended full-power operation. (The estimated accuracy of current shielding calculations is a factor of three)
- d. Radiation dosages at points around the package which do not contribute directly to the dosage received at the cab to be in the following ratios after shutdown:

Back and sides of reactor = 10 x cabside dose

Below reactor = 10 x cabside dose

Above reactor = 100 x cabside dose

- e. Minimum exclusion area during operation compatible with weight limitations imposed on system. Expedient shielding is acceptable if necessary to limit exclusion area to 500 feet or less. Radiation dosage at perimeter of exclusion area is not to exceed 5 mr/hr (gammas and neutrons).

- f. Shielding and associated components to withstand the following shock loadings:

Fore and Aft	8	G
Lateral	2	G
Vertical up	4½	G
Vertical down	5½	G

2. Basic Design

a. Nuclear

The usual approach to lightweight systems is to design a shield to attenuate neutrons and gamma dose at the same rate. This method was rejected in this design because the basic requirement is to shield a shutdown reactor where high neutron fluxes are not present. The basic shielding concept has been to surround the core with high-density material to provide shielding against fission product radiation after shutdown. During full-power operation, the high-density material will provide shielding against gamma activity from the core. Neutron shielding will be provided by including a hydrogenous material (heavily loaded with boron) outside of the high-density shield. Since little neutron attenuation is required in the shutdown condition, most of the neutron shield will be removed to save weight in transit. Boron will be used to suppress secondary radiation produced outside the core.

b. Materials

The shielding must be as compact as possible. For this reason, tungsten is one of the important high-density materials in the system; its excellent high-temperature properties allow it to be used in the hot gas stream provided it is canned or otherwise protected from an oxidizing atmosphere. In addition, tungsten has the highest density of any readily available material, is economical, fabricable, and only slightly activated by neutrons. This high density results in a significant weight saving over lead. When alloyed with small amounts of copper and nickel, tungsten is easier to fabricate. The fabricated density is at least 16.5 gm/cm³, considerably more than lead.

Tungsten is activated in neutron flux. Resonance capture of epithermal neutrons accounts for a considerable portion of this activation. The resultant gamma photons are relatively weak (0.7 Mev); and covering the activated surfaces with layers of other shielding materials reduces the radiation to levels comparable to the remainder of the system. Tungsten is used as part of the fast neutron reflector to save weight.

Tantalum has been specified for some applications in this system because it has the high density of tungsten alloy, but is less brittle and more readily fabricated. A typical application is in the control blade support structure where it is used to reduce streaming effects.

The remaining applications of high-density materials use lead and stainless steel. The activation of lead is negligible, and stainless steel is used only in regions where activation is unimportant or easily shielded.

There are two parts to the hydrogenous neutron shield. The first is a layer of canned ammonium pentaborate salts attached directly to the outer surfaces of the reactor shielding and gas plenums. This salt layer blankets the entire reactor. Ammonium pentaborate, $(\text{NH}_4)_2\text{B}_{10}\text{O}_{16} \cdot 8\text{H}_2\text{O}$, is a crystalline substance that looks like ordinary table salt. Its hydrogen content, in the form of water of crystallization, plus a significant amount of boron, give it good neutron thermalizing power. Most of the neutron capture occurs in boron because of the high boron-hydrogen ratio. The subsequent secondary radiation is composed of readily-stopped alpha particles and low-energy gamma radiation. These properties make ammonium pentaborate a valuable shielding material for use in metal-hydrogen systems.

The rest of the neutron shield consists of heavily borated water contained in an outer shield tank. The boron will be held in suspension by dissolving 18 wt % ammonium pentaborate. This amount of salt is completely soluble in water at 118°F.

The use of boral to suppress thermal neutron fluxes in most of the outer high-density material will minimize activation radiation during shutdown.

3. Description of Components

A preliminary design of the ML-1 reactor is complete and is described below (Figure III-35). The shielding of this system may be classified under two headings, radial and axial.

The radial shielding components (permanently attached to the core) are the lead-tungsten fast neutron reflector, the lead radial shielding, and the radial boron-salt blanket.

The axial shielding components consist of tungsten baffles, canned in stainless steel and located in the gas stream, stainless steel in the gas plenums, lead shielding placed outside the plenums, and the axial salt blanket. In addition, the stainless steel tube sheets contribute substantially to the axial shielding.

4. Radial Shielding Components

a. Reflector

The reflector determines the preferential nature of the radial shielding. One-half of the reflector is composed of tungsten

and lead, and the other half is constructed solely of lead. This increases the effective density of the shielding on the cab side. The tungsten is located far enough from the core to have a negligible reactivity effect. The inner (or core) face of the reflector is insulated to prevent boiling in the core from nuclear heating in the reflector. The insulation is canned in aluminum.

The control mechanisms are located between the reflector segments and are supported in a stainless steel and tantalum structure which provides good shielding.

A water annulus separates the reflector from the remainder of the shield. Moderator water flows through this annulus to remove nuclear heat generated in the shield. Calculations show that boiling in this annulus will not affect stable operation of the reactor.

b. Radial Shield

Lead is the basic material of the radial shield. A sheet of boral will cover the inner face of the lead to reduce thermal neutron fluxes and the subsequent secondary radiation from the lead during operation. The lead is canned in stainless steel.

c. Radial Salt Blanket

The radial salt blanket is crystalline ammonium pentaborate packed to a density of about 1 gm/cm^3 . It is canned in stainless steel and surrounded by a boral sheet to reduce activation of the stainless steel.

5. Axial Shielding Components

a. Tungsten End Shields

The earlier vane-type baffle was discarded because of fabrication difficulties and the problem of tungsten activation. The end shields now are solid tungsten discs contained in stainless steel housings. These housings protect the tungsten from oxidation in the hot gas stream, attenuate the radiation coming from the activated tungsten, and provide additional shielding from fission product radiation. In addition, the housings are shaped to minimize resistance to gas flow.

b. Gas Plenums and Ducts

The main purpose of the plenums is to contain the coolant gas. They are thick enough, however, to contribute substantially to end shielding. The gas inlet and outlet ducts have been flattened to reduce overall height, and have been offset by using 90° elbows to reduce neutron streaming during operation.

c. Lead Axial Shielding

Lead shielding is placed outside the plenums to further attenuate radiation. The lower gas exit duct is also shielded with

lead to provide lower radiation levels at the bottom than exist at the top of the reactor.

Axial Salt Blankets

The axial salt blankets are composed of the same material as the radial blanket, and are canned in aluminum.

6. Shielding Tank

The entire reactor and shield assembly is contained in a water tank. The tank is kept full of heavily borated water during operation; the boron is in the form of dissolved ammonium pentaborate. The shield tank is insulated, and a portion of the moderator heat is dumped into it through a water-to-water heat exchanger to maintain the minimum temperature (118°F) needed to hold the ammonium pentaborate in solution. To reduce the overall height, a separate, removable tank (kept on top of the main tank during operations) is removed before transporting the reactor.

7. Experimental Work

Extensive tests are being conducted to determine the characteristics of many of the materials associated with this system in the predicted environment. The use of solid ammonium pentaborate as a neutron shield is unprecedented in the field of reactor design. Tests are underway to determine its behavior under prolonged exposure to high temperatures. Irradiation capsules, prepared for insertion in the MTR to determine radiation damage properties, currently await reactor space. Tests conducted to determine the attainable packed crystal density have achieved a density of 1 gm/cm³ in the laboratory.

Solubility of the ammonium pentaborate has been verified in the laboratory. The time involved in attaining various concentrations under varying temperatures is being estimated.

Corrosion of various materials, such as aluminum and stainless steel, in these solutions is being studied.

The prospects of performing an extensive shielding mock-up experiment using a low-power reactor are being studied.

C. REACTOR FACILITIES (Task 58-300)

This task is responsible for the design and fabrication of the reactor shield tank and for devising a method of removing heat from the reactor moderator. In addition, this task is responsible for the secondary support facilities associated with reactor maintenance, such as fuel element servicing equipment.

The reactor tank contains the borated shield water during reactor operation and provides a personnel exclusion area around the reactor during periods of inactivity. The size of the reactor tank is dictated by clearance requirements of various modes of transportation. The reactor tank is about 9 feet in diameter by 8 feet tall. The tank is made of aluminum to reduce its weight.

The tank consists of a cylindrical section with a hemispherical bottom and a relatively flat cover. A small dome on the cover provides adequate shielding of the upper gas duct during operation. The reactor is supported in the reactor tank by a circular plate between the cylindrical and hemispherical portions of the tank. This plate is gusseted to the skirt of the tank for structural strength. Attachment points for transportation tie-downs are provided on the support ring periphery. The entire package is shock-mounted on an elastomer base.

Two methods will be used to combat the effects of cold weather on the borated shield water. The effectiveness of the shield during operations will be reduced if the temperature of the shield water falls low enough to precipitate the ammonium pentaborate. In addition, the shield water could freeze and rupture the tank during extended cold weather. One method of combating these possibilities will use thermal insulation on the exterior of the tank to minimize heat loss. The other method is to incorporate a moderator-water to shield-water heat exchanger in the moderator cooling loop. The heat exchanger will serve three purposes: 1) heat the shield water to prevent freezing; 2) act as a hot weather heat sink; and 3) provide a heat sink during shutdown if the main water-to-air heat exchanger fans are inoperative. (Without forced convection through the latter heat exchanger, the air-side film coefficient would be too low to derive much benefit from the heat exchanger, and an auxiliary heat sink is necessary.)

The following mechanisms produce moderator heat: 1) slowing down of fission neutrons; 2) absorption of gamma radiation; 3) conduction of heat from the reactor gas coolant through the reactor pressure vessel; and 4) gamma ray absorption in the fast neutron reflector which then is conducted into the moderator. The aggregate heat load from these sources is calculated to be 1.4×10^6 Btu/hr.

The moderator cooling system is designed to prevent boiling of the moderator water in the vicinity of the cooling system and thus avoiding possible nuclear instability. (A flow diagram for the system is shown in Figure III-36.) Present indications are that water temperatures of 180°F at the inlet and 210°F at the outlet combined with a flow rate of 100 gpm will be satisfactory. Mounting limitations make it necessary to use two cores for the heat exchanger.

Demineralizers are required to maintain a high degree of water purity in order to minimize induced activity from neutron activation. In view of the low operating temperatures, water purity requirements (to reduce corrosion, deposition and fouling problems) are of secondary importance.

Fresh water flooding would increase reactivity if there was a leak into the reactor core. The moderator loop will therefore be designed to have a water volume sufficient to cover no more than half the vertical height of the fuel elements.

Loop components, wherever possible, will be of aluminum to minimize weight. The limitations in total reactor package weight dictate that some parts of the moderator cooling loop must be located on the power conversion skid. The transfer of heat exchangers, fans and controls (about 1000 lb) is being investigated.

Water demineralizer columns are mounted on the side of the reactor package farthest from the power conversion skid to minimize radiation problems to personnel assembling or dis-assembling service connections.

Methods of removing fuel elements are being studied and will be more fully investigated before freezing the design. However, a transfer coffin and lightweight hoist (Figure III-37) are basic to all current plans. A 10-ft (vertical) tank extension is placed over the reactor tanks and filled with fresh water to protect personnel during fuel element removal.

The present preliminary design provides a submerged hoist near the lower end of the tank extension to handle the upper plenum. This hoist will be rail-mounted to permit moving the plenum to provide unrestricted access to the fuel elements. Separate hoists are desirable for the plenum and casket to minimize handling and to reduce the probability of accidents when loading or removing the heavy weight of the upper plenum.

D. PRESSURE VESSEL (Task 58-400)

This task is responsible for design, fabrication and testing the pressure vessel, internal insulation, gas ducts, fuel element support, neutron source, and mounting for the internal nuclear shielding.

The reference pressure vessel design for ML-1 uses a stainless steel tube bundle of 61 pressure tubes attached to a tube sheet at each end of the vessel. The pressure tubes enclose the fuel elements and separate high-pressure nitrogen coolant from moderator water. The tubes have 20-mil walls, and the dimensions are fixed at 24 inches long by 1.756 inches ID. The tube sheets are attached to the upper and lower plenums which act as collectors for inlet and outlet gas coolant.

The inlet plenum cover is removable for access to the fuel elements. Nuclear shielding, in the form of tungsten discs, is provided inside the plenums. The core is encased in lead and tungsten shielding which also acts as a fast-neutron reflector. Six pairs of control rods are arranged on the circumference of the vessel. The blades attached to the control rods extend into the tube bundle between the pressure tubes. The moderator water enters and leaves the core through radial pipes in the circumferential shield.

The pressure vessel will be built to the following criteria: overall height on trailer, 132 inches; weight, 15 tons; lifetime, 5 years; and loading, 5 G vertically and 8 G horizontally. The design adheres to military specifications as well as the allowable stresses defined by the ASME pressure vessel code. The following cycle state-points were used in the design: inlet gas, 800°F and 300 psi; exhaust gas, 1200°F and 300 psi. Bulk metal to provide additional shielding has been added wherever possible.

Status of the design of the pressure vessel is:

1. The final layout drawings are being completed. These drawings are being revised to incorporate design changes dictated by the recent change from an 18-ton package to a 15-ton package.

2. Bids have been requested from vendors.
3. Bids for fabricating long lead-time items have been requested from vendors.
4. Final detail design has begun on pressure vessel components.

The individual components of the pressure vessel are discussed in the following paragraphs.

1. Tube Bundle

Stress analysis is complete for the tube bundle. It includes the effect of shortening the core from 30 to 24 inches. The tube sheet will be made from AISI Type-304 stainless steel; and the pressure tubes, from AISI Type-304L. The calculations show that a 3-inch-thick tube sheet is needed when 0.020-inch wall tubes are used.

Quotations received for fabrication of these pressure tubes did not meet tolerance specifications. Use of an explosive forming process is being investigated.

Three methods of making the joint between the tube and tube sheets have been investigated because of the importance of this joint to structural integrity and pressure tightness. Mock-ups of these methods were built and tested; a rolled and welded joint, a brazed joint, and a rolled and brazed joint were all subjected to an increasing tensile load until failure. The joints are similar in strength and satisfactory for their intended application. A mock-up of the tube sheet was built and is being tested to determine the amount of warpage which results from a welded joint. The method of fabrication will be chosen on the basis of these tests.

2. Plenums

The plenums will each be machined from a single forging of AISI Type-304 stainless steel. The head for the upper plenum will be attached with either a bolted or a V-bank flange. The lower plenum cover will be pinned and welded in place. Bids were received for the plenum forging. Tooling drawings are being made for the machining of the plenums.

3. Flow Model

A flow model ($\frac{1}{4}$ -scale) of the pressure vessel was designed and is being built to provide data for use in minimizing the pressure drop in the plenums and around the tungsten baffles. The model will be used for three purposes: to determine the actual pressure drop in the plenum; to show how the shape of the plenums may be changed to minimize the pressure drop; and to determine the shape needed for equal flow distribution through each pressure tube.

High-pressure nitrogen gas will be blown through the scale mock-up of the upper plenums. A removable disc of aluminum, simulating the tungsten shield, will permit the shape to be changed. Flow through the model will be controlled by an orifice.

4. Gas Ducts

Two 90° elbows in each gas duct are used to provide more effective shielding of the core. (Figure III-35) The transition from round pipe at the plenums to flat ducts was needed to reduce the overall package height to 132 inches. The ducts are attached to the reactor tank. Bellows are used between the ducts and the reactor tank to provide for thermal expansion.

A half-inch layer of aluminum will be cast around each duct in the areas with nuclear shielding. The aluminum will provide a heat path to the tank water, and is needed to maintain at acceptable levels the temperatures of the adjacent salt blanket and lead shielding.

5. Insulation

Radiation barrier insulation will be used in the plenums and gas ducts. This is in the form of concentric sheets of thin-gauge stainless steel spaced about 0.10 inches apart. Calculations show that 3 or 4 stainless steel sheet barriers will be required. Laboratory tests are underway to determine the long-term emissivity of possible barrier materials. Three types of chromalloy steel and an aluminum-aluminum oxide alloy (ALCOA M-257) are being tested.

6. Shield to Pressure Vessel Joint

The pressure vessel is joined to the circumferential shield at the upper and lower plenums. The upper joint, a mechanical joint, has been designed. The lower joint will provide a bellows action to accommodate expansion of the pressure tubes during operation.

7. Fuel Element Support and Hold-Down

The fuel element support and hold-down is a one-piece unit with no small moving parts that could break off and enter the gas stream. The upper portion of the fuel element is a cylinder split into six segments. A circumferential ledge on the upper portion of the cylinder matches a similar groove in the support. The element is installed by compressing the segmented cylinder so that the ledge snaps into the groove. A mock-up of this unit was built and tested. A handling tool was designed to operate the hold-down mechanism, and a mock-up of this tool is being built.

E. CONTROL ROD BLADES (Task 58-500)

Control of the reactor will be achieved by neutron-absorbing blades located in the moderator. This method, of all the types possible, appears to offer the best means of control with the fewest unknowns, and needs the least development. Reliability is one of the prime considerations

in the design of those portions of the control system which fall within the core. Since parts within the core can only be replaced in a hot-shop, these parts are being designed and developed for maximum simplicity and reliability. Weight is another primary consideration for this reactor. The mounting for the blades has therefore been designed so as not to appreciably increase the weight of the shielding required.

Control will be provided by six pairs of blades, one regulating and five shim-scam pairs. Each blade will extend about 10 inches into the core and be 3.3 inches wide. They will be mounted at the midpoint between the tube sheets within the reflector. The blades will move in vertical planes, one blade swinging upward when retracting, the other downward. They will be rotated toward the center of the core to insert poison.

Two designs are being studied. In one the blades are connected by gears driven by a rotary shaft. The other uses blades which move as in the gear-driven design, but are connected by links and driven by a reciprocating shaft (Figure III-38). Models have been fabricated of these two preliminary control blade mounts. Two types of gear-driven blade mounts have been built, one with stainless steel ball bearings, and one with taper-sleeve bearings with electrolized bearing surfaces. These models were built chiefly of AISI Type-304 stainless steel except for the bearings and gears (these were AISI Type-431 hardened stainless steel) and the blade tip guides (aluminum). The gear teeth and other bearing surfaces were also electrolized. These models are expected to provide test data which will indicate any weak areas in the design so that the prototype mounts can correct any deficiencies.

The test equipment for use with these models has been fabricated and is being assembled. Two test units were fabricated, one for use with the gear-driven mount, and the other for use with the reciprocating shaft mount. Both units are capable of actuating the blade through 90° shim-scam motion, 45° shim-scam, full stroke regulating, and regulating-hunting motions. The blade assemblies will be mounted on mock-up pressure tube bundles and suspended in water for cycling tests. Dash-pot and drive shaft forces will be determined by strain-gauges. Immersion heaters and ice will be used during cycling to simulate moderator water temperature extremes. Additional cycling will be completed with ammonium pentaborate and silica flour in the test water to determine the corrosion and abrasion resistance of the blade mount mechanism under conditions far worse than expected during operations.

The blade assemblies, mounted on the pressure tube mock-ups, will comprise a package that will be tested for 8 G shock loadings. A simple method has been designed that makes use of a calibrated spring to attain the shock loadings. Vibration tests are also planned to simulate transportation and operation-induced frequencies.

Blade tests are scheduled to begin in July. The recent changes of the reactor core dictated some modest changes in the control blade design which are already under way. However, it is expected that the test data will enable the new design to meet the requirements of the prototype reactor.

F. NEUTRONICS (Task 58-600)

The assembly of the ML-1 critical experiment for IA fuel elements was started during May at BMI. The fuel elements were of the design used in the GCRE-I critical experiment, but the foils were re-wrapped for greater uniformity. The lattice spacing was based on experimental results with the GCRE-I critical experiments to achieve uniform radial power distribution across the core. The ML-1 stainless steel tubes and tube sheets were mocked-up.

Criticality was achieved in June with a loading of 59 elements and an excess reactivity of 0.4%. This was close to the calculated criticality at 58 elements.

Flux and power mapping were completed. The radial peak-to-average power ratio across the core was 1.13.

Work will be completed on the IA critical experiment. This includes IB substitution and ML-1 control rod measurements. This information will be useful in designing the IB core.

Planning was started on the fabrication of a new ML-1 critical experiment using the short IB fuel elements. This work should be completed before freezing the design on the ML-1 fuel elements, but the schedule is very tight.

LETTER	DATE	CHANGE	BY	CHK'D

COPY

FIG. III-1

TO GCRE
APPROX. 1/2 MI.

No. REGD.			Next Assy.	MODEL	DRAFTSMAN	DATE	MATERIAL	
HEAT TREAT			FINISH		CHECKED		PLOT PLAN	
TOLERANCE UNLESS OTHERWISE NOTED					STRESS		AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA	
LINCAR	.XX	± .03			ENGINEER			
TOL.	.XXX	± .010			APPROVED		SIMILAR TO	SCALE
ANGULAR TOL.					PRODUCTION		RELEASE DATE	CAL. WEIGHT
INDICATES SURFACE ROUGHNESS					MATERIAL		ACT. WEIGHT	DWG. SIZE
FINISH PER MIL-STD-10 UNLESS OTHERWISE NOTED								C
								DRAWING NUMBER

LETTER	DATE	CHANGE	BY	CHK'D

FIG. III-2

MATERIAL			ML-1 FACILITY TEST SITE PLOT PLAN		AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA	
NO. RECD	NEXT ASSY.	MODEL	DRAFTSMAN	DATE		
HEAT TREAT			CHECKED			
TOLERANCE UNLESS OTHERWISE NOTED			STRESS			
LINEAR	XX	OS	ENGINEER			
TOL.	XXX	DIO	APPROVED			
ANGULAR TOL.			PRODUCTION			
INDICATES SURFACE ROUGHNESS						
SIMILAR TO				SCALE	DWG. SIZE	
RELEASE DATE				CAL. WEIGHT		

LETTER	DATE	CHANGE	BY	CHK'D

SIDE VIEW
LONGITUDINAL LOADING DUE TO
FRICTIONAL DRAG & IRREGULARITIES
ON DECK
DIRECTION REVERSIELS

END VIEW
TRANSVERSE LOADING
SAME AS LONGITUDINAL
DIRECTION REVERSIELS

SHOCK MOUNT LOAD DIAGRAM
NO SCALE

SPRING RATE - SHEAR - 825 LBS./IN.

O-408/13

COPY

ISOLATOR INSTALLATION

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

FIG. III-3

NO. REQ.		NEXT ASBY.		MODEL		DRAFTSMAN		DATE	
HEAT TREAT		FINISH		TOLERANCE UNLESS OTHERWISE NOTED		STRESS		ENGINEER	
LINEAR		XX		± .03		APPROVED		PRODUCTION	
TOL		XXX		= .010		MATERIAL		MATERIAL	
INDICATES SURFACE ROUGHNESS		FINISH PER MIL STD. 10 UNLESS OTHERWISE NOTED		SIMILAR TO		SCALE		DWG. SIZE	
RELEASE DATE		CAL. WEIGHT		ACT. WEIGHT		O-408/13		D	

ISOLATORS
SHOCK & VIBRATION
(PORTABLE POWER
PLANT)

AEROJET-GENERAL
NUCLEONICS
SAN RAMON
CALIFORNIA

O-408/13
DRAWING NUMBER

LOAD-DEFLECTION & FREQUENCY OF ELASTOMERS

FIG.III-4

SPRING CONSTANT VS. DEFLECTION OF ELASTOMER

FIG.III- 5

LOAD-DEFLECTION CURVE
 1 1/8" DIA. "BROKEN-IN" NYLON ROPE
 BREAK STRENGTH=32,000 LBS.

FIG. III-6

SPRING CONSTANT VS. DEFLECTION
1 1/8" DIA. "BROKEN-IN" NYLON ROPE
BREAK STRENGTH=32,000 LBS.

FIG.III- 7

NO. REQD.			NEXT ASBY			MODEL			DATE JAN 1971			DRAFTSMAN E. J. ...			MATERIAL			TIEDOWN SYSTEM ML-1 DOUGLAS C-133 A			AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA								
HEAT TREAT			FINISH			TOLERANCE UNLESS OTHERWISE NOTED			CHECKED			STRESS			ENGINEER			SIMILAR TO			SCALE 1" = 8"			DWG. SIZE D					
LINEAR			XX			± .03			TOL.			XXX			± .010			APPROVED			RELEASE DATE			CAL. WEIGHT			ACT. WEIGHT		
ANGULAR TOL.			INDICATES SURFACE ROUGHNESS			FINISH PER MIL STD. 10 UNLESS OTHERWISE NOTED			PRODUCTION			MATERIAL			DRAWING NUMBER 0-408116														

LETTER	DATE	CHANGE	BY	CHK'D
A	5-12-59	MODIFIED TIEDOWN & COUPLING STRUCTURE	JSA	
B	5-20-59	RELOCATED ELASTOMERS	JSA	
C	6-29-59	REVISED SUBSTRUCTURE; OVERALL DIMENSIONS TO CONFORM WITH O-408103	JSA

0-408073

COPY C

FIG. III-9

NO. REQ.		NEXT ASSY.		MODEL		J. ALCORN DRAFTSMAN		DATE 5-12-59	
HEAT TREAT		FINISH		TOLERANCE UNLESS OTHERWISE NOTED		CHECKED		STRESS	
LINEAR		.XX		± .03		ENGINEER JWB		5/24/59	
TOL.		.XX		± .010		APPROVED		MATERIAL	
INDICATES SURFACE ROUGHNESS		FINISH PER MIL STD. 10 UNLESS OTHERWISE NOTED		PRODUCTION		MATERIAL		MATERIAL	

MATERIAL		MOCKUP ASS'Y. - REACTOR PACKAGE ML-1		AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA	
SCALE 1" = 1'	DWG. SIZE D	RELEASE DATE	CAL. WEIGHT	0-408073	
ACT. WEIGHT	DRAWING NUMBER

POWER CONVERSION MOCK-UP ML-1

**FIG.III-11, TYPICAL BRAZE SAMPLES.
SPECIMEN ARE 3 PINS WIDE BY TWO
INCHES LONG.**

**FIG.III-12, UNBRAZED PIN JOINT,
250 X, UNETCHED.
COPPER BRAZE, NICKEL PIN,
NICKEL PLATE. TYPICAL OF
BRAZED JOINTS ON SAMPLES
USING 0.001 IN. COPPER BRAZE
MATERIAL. NOTE INSUFFICIENT
ALLOY TO FLOW INTO JOINT.**

**FIG.III-13, BRAZED PIN JOINT
250 X, UNETCHED
COPPER BRAZE, NICKLE PIN,
NICKEL PLATE. TYPICAL OF
BRAZE JOINTS ON SAMPLES
USING 0.004 IN. OF COPPER
BRAZE. BUILD UP OF EXCESS
COPPER SEEN IN THIS
PHOTOGRAPH WAS LESS AT
OTHER JOINTS, BUT AREA
BETWEEN PIN AND PLATE WAS
ADEQUATELY FILLED.**

FIG. III-14 , BRAZED PIN JOINT , 500 X.
SILVER BRAZE , NICKEL PIN , NICKEL PLATE .
TYPICAL OF BRAZED JOINTS ON SAMPLES USING
0.004-IN. AMS 4772 SILVER BRAZE MATERIAL .

FIG.III-15 GAS CORROSION APPARATUS SHOWN HERE: 1) LINDBERG FURNACE; 2) NITROGEN TUBE IN FRONT (300 PSI); 3) AIR TUBE IN REAR (300 PSI); 4) NITROGEN SAFETY BLEED VALVE; 5) AIR SAFETY BLEED VALVE; 6) GAS SUPPLY; 7) TEMPERATURE CONTROL; 8) TEMPERATURE RECORDER.

FIG. III-16 INTERGRANNULAR CORROSION-ARMCO IRON, 800X

FIG. III-17 COPPER CORED PIN, CROSS SECTION, 250X

FIG. III-18 COPPER CORED PIN, FOOT AREA, 250 X

TEMPERATURE ENTROPY DIAGRAM: FULL LOAD, 100°F AMBIENT
(For a reactor with a low pressure drop)

AMBIENT TEMPERATURE	100 °F	NET ELECTRICAL POWER	330 kw
BYPASS	1 %	AUXILIARY POWER	70 kw
COMPRESSOR FLOW	82820 lb/hr	REACTOR POWER	2593 kw
TURBINE FLOW	82000 lb/hr	CYCLE EFFICIENCY	12.7 %

NOTE: All temperatures are in °F, to nearest degree. All pressures are in PSIA, rounded to nearest unit except as noted.

FIG. III-19

TEMPERATURE ENTROPY DIAGRAM: NO LOAD, 100°F AMBIENT
(For a reactor with a low pressure drop)

AMBIENT TEMPERATURE	100	°F	NET ELECTRICAL POWER	0	kw
BYPASS	1	%	AUXILIARY POWER	70	kw
COMPRESSOR FLOW	41300	lb/hr	REACTOR POWER	1008	kw
TURBINE FLOW	35100	lb/hr	CYCLE EFFICIENCY	0	%

NOTE: All temperatures are in °F, to nearest degree. All pressures are in PSIA, rounded to nearest unit except as noted.

FIG. III-20

TEMPERATURE ENTROPY DIAGRAM: FULL LOAD, 0°F AMBIENT
(For a reactor with a low pressure drop)

NOTE: All temperatures are in °F, to nearest degree. All pressures are in PSIA, rounded to nearest unit except as noted.

FIG. III-21

TEMPERATURE ENTROPY DIAGRAM: NO LOAD, 0°F AMBIENT
(For a reactor with a low pressure drop)

NOTE: All temperatures are in °F, to nearest degree. All pressures are in PSIA, rounded to nearest unit except as noted.

FIG. III- 22

TEMPERATURE ENTROPY DIAGRAM: FULL LOAD, 100°F AMBIENT
(For a reactor with a high pressure drop)

AMBIENT TEMPERATURE	100 °F	NET ELECTRICAL POWER	330 kw
BYPASS	1 %	AUXILIARY POWER	70 kw
COMPRESSOR FLOW	101300 lb/hr	REACTOR POWER	3114 kw
TURBINE FLOW	100300 lb/hr	CYCLE EFFICIENCY	10.6 %

NOTE: All temperatures are in °F, to nearest degree. All pressures are in PSIA, rounded to nearest unit except as noted.

FIG. III- 23

LETTER	DATE	CHANGE	BY	CHK'D

COPY

FIG. III-24

NO. REQ.		HEAT TREAT	MODEL	DRAFTSMAN	DATE	MATERIAL	
					6/5/59		
HEAT TREAT		FINISH		CHECKED		PLAN VIEW-IN LINE	
TOLERANCE UNLESS OTHERWISE NOTED		TOLERANCE UNLESS OTHERWISE NOTED		STRESS		SCHEME-POWER	
LINEAR	XX	± .05		ENGINEER		CONVERSION	
TOL.	XXX	± .010		APPROVED		UNIT - ML-1	
INDICATES SURFACE ROUGHNESS		PRODUCTION		SIMILAR TO		SCALE	DWG. SIZE
FINISH PER MIL STD. 10 UNLESS OTHERWISE NOTED		MATERIAL		RELEASE DATE		1/2"	D
				CAL. WEIGHT			
				ACT. WEIGHT			

AEROJET-GENERAL
NUCLEONICS
SAN RAMON
CALIFORNIA

0-408725

DRAWING NUMBER

LETTER	DATE	CHANGE	BY	CHK'D

COPY

FIG. III-25

NO. RECD.		RECD. BY	DATE	MATERIAL	
NEXT ASSY.		MODEL	6/9/69	ELEVATION VIEW, IN-LINE SCHEME, POWER CONVERSION UNIT - ML-1	
HEAT TREAT		FINISH	CHECKED	AEROJET-GENERAL NUCLEONICS	
TOLERANCE UNLESS OTHERWISE NOTED		STRESS	ENGINEER	SAN RAMON CALIFORNIA	
LINEAR	.XX	± .08	APPROVED	SIMILAR TO	SCALE 1"=1'
TOL.	.XX	± .010	PRODUCTION	RELEASE DATE	CAL WEIGHT
INDICATES SURFACE ROUGHNESS		MATERIAL		ACT. WEIGHT	DWG. NO. D
FINISH PER MIL-STD-10 UNLESS OTHERWISE NOTED					0-408124
					DRAWING NUMBER

LETTER	DATE	CHANGE	BY	CHKD

STATE POINTS TEMP. & PRESS.
ML-1 NOTE NG

AMBIENT TEMP		AMBIENT TEMP		
-65°F		125°F		
PRECOOLER FAN	6	6		
FLOW RATE 100/SEC				
REACTOR OUTPUT KW				
CYCLE EFF. %				
NET PLANT EFF. %				
STATE POINT	TEMP. °F	PRESS. PSIA	TEMP. °F	PRESS. PSIA
①				
②				
③				
④				
⑤				
⑥				

AMBIENT TEMP		AMBIENT TEMP		
-65°F		125°F		
PRECOOLER FAN	6	6		
FLOW RATE 100/SEC				
REACTOR OUTPUT KW				
CYCLE EFF. %				
NET PLANT EFF. %				
STATE POINT	TEMP. °F	PRESS. PSIA	TEMP. °F	PRESS. PSIA
①				
②				
③				
④				
⑤				
⑥

- 1ST LETTER
- DEFINITION
- A - ATMOSPHERIC COOLER
 - B - BLOWER OR FAN
 - C - COMPRESSOR
 - D - DRYER
 - E - ELECTRICAL GENERATOR
 - F - FILTER
 - G - GEAR BOX OR BELT DRIVE
 - H - HEATER
 - J - EXPANSION JOINT
 - K - SHOCK ABSORBER
 - L - PLE B LINE
 - M - MOTOR
 - N - OVER RUNNING CLUTCH
 - P - LIQUID PUMP
 - S - SPECTACLE OR BLIND FLANGE
 - R - STRUCTURE
 - T - TURBINE
 - U - SHAFT COUPLING
 - V - VALVE
 - W - FLYWHEEL OR CABLE WHEEL
 - X - FLUID HEAT EXCHANGER
 - Y - SET POINT
 - Z -
- 3RD LETTER:
- H -
 - L -
 - C -
 - D - DRYING CIRCUIT
 - E -
 - G - GAS MAKE UP CIRCUIT
 - H - SPACE HEATING CIRCUIT
 - J -
 - K -
 - L - OIL CIRCUIT
 - MIP STARTING MOTOR
 - POWER GEAR BOX
 - POWER COMPRESSOR
 - M - MOLENTON CIRCUIT
 - 60 CYCLES
 - 480 VOLTS
 - P - FORDEN CIRCUIT
- ALTERNATOR:
- 3400 RPM
 - 33 PHASE
 - 43 KW
 - 60 CYCLES
 - 4160/2400 VOLTS
- GEAR BOX
- G.I.P
- DESIGN HP

COPY

FIG. III-26

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

NO. READ		NEXT ASSY.		MODEL		DRAFTSMAN		DATE	
HEAT TREAT		FINISH		TOLERANCE UNLESS OTHERWISE NOTED		STRESS		ENGINEER	
LINEAR		XX		1.03		APPROVED		PRODUCTION	
TOL.		XX		1.010		APPROVED		MATERIAL	
ANGULAR TOL.		XX		1.010		APPROVED		MATERIAL	
INDICATES SURFACE ROUGHNESS		XX		1.010		APPROVED		MATERIAL	
FINISH PER MIL STD. 10 UNLESS OTHERWISE NOTED		XX		1.010		APPROVED		MATERIAL	

MATERIAL		AEROJET-GENERAL NUCLEONICS	
MECHANICAL FLOW SHEET, SYSTEM "P"		SAN RAMON CALIFORNIA	
SIMILAR TO		SCALE	
RELEASE DATE		CAL. WEIGHT	
ACT. WEIGHT		D	
DRAWING NUMBER		0-408062	

LETTER	DATE	CHANGE	BY	CHKD

- 1ST LETTER DEFINITION
- A-ATMOSPHERIC COOLER
 - B-BLOWER OR FAN
 - C-COMPRESSOR
 - D-DRYER
 - E-ELECTRICAL GENERATOR
 - F-FILTER
 - G-GEAR BOX OR BELT DRIVE
 - H-HEATER
 - J-EXPANSION JOINT
 - K-SHOCK ABSORBER
 - L-PIPE LINE
 - M-MOTOR
 - N-
 - O-OVER RUNNING CLUTCH
 - P-LIQUID PUMP
 - Q-SPECTACLE OR BLIND FLANGE
 - R-STRUCTURE
 - Equip. SUPPORT FRAME
 - HOUSING, TIE DOWN SYSTEM & RECUPERATOR SUPPORT
 - S-STORAGE TANK
 - T-TURBINE
 - U-SHAFT COUPLING
 - V-VALVE
 - W-FLYWHEEL OR CABLE WHEEL
 - X-FLUID HEAT EXCHANGER
 - Y-
 - Z-

- 3RD LETTER
- A-
 - B-
 - C-
 - D-DRYING CIRCUIT
 - E-
 - F-
 - G-GAS MAKE UP CIRCUIT
 - H-SPACE HEATING CIRCUIT
 - J-
 - K-
 - L-OIL CIRCUIT
 - POWER GEAR BOX
 - POWER COMPRESSOR
 - M-MODERATOR CIRCUIT
 - N-

- COPY
- P- POWER CIRCUIT
 - Q-
 - R- REFRIGERATION CIRCUIT
 - S-
 - T- TRANSFER CIRCUIT
 - U-
 - V-
 - W-
 - X-
 - Y-
 - Z-

FIG. III-28

NO. REQD.		NEXT ASSY.		MODEL		DRAFTSMAN		DATE	
HEAT TREAT		FINISH		STRESS		ENGINEER		APPROVED	
TOLERANCE UNLESS OTHERWISE NOTED		LINEAR		TOL.		ANGULAR TOL.		INDICATES SURFACE ROUGHNESS	
SIMILAR TO		SCALE		DWG. SIZE		RELEASE DATE		CAL. WEIGHT	
ACT. WEIGHT		DRAWING NUMBER		D		0-408059		DRAWING NUMBER	

- LEGEND
- PHASE I REACTOR ATMOSPHERIC DRYING
 - PHASE II REACTOR DESICCATOR DRYING
 - PHASE III DESICCATOR REGENERATION
 - PHASE IV GAS DESICCATOR DRYING UNDER NORMAL OPERATION

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

LETTER	DATE	CHANGE	BY	CHKD

- 1ST LETTER DEFINITION
- A—ATMOSPHERIC COOLER
 - B—BLOWER OR FAN
 - C—COMPRESSOR
 - D—DRYER
 - E—ELECTRICAL GENERATOR
 - F—FILTER
 - G—GEAR BOX OR BELT DRIVE
 - H—HEATER
 - J—EXPANSION JOINT
 - K—SHOCK ABSORBER
 - L—PIPE LINE
 - M—MOTOR
 - N—
 - O—OVER RUNNING CLUTCH
 - P—LIQUID PUMP
 - Q—SPECTACLE OR BLIND FLANGE
 - R—STRUCTURE
 - EQUIP. SUPPORT FRAME
 - HOUSING TIE DOWN SYSTEM

- 3RD LETTER
- A—
 - B—
 - C—
 - D—DRIVING CIRCUIT
 - E—
 - F—
 - G—GAS MAKE UP CIRCUIT
 - H—SHOCK HEATING CIRCUIT
 - J—
 - K—
 - L—OIL CIRCUIT
 - M—MODERATOR CIRCUIT
 - N—
 - P—POWER CIRCUIT
 - Q—
 - R—REFRIGERATION CIRCUIT

- 5—STORAGE TANK
- T—TURBINE
- U—SHAFT COUPLING
- V—VALVE
- W—FLYWHEEL OR CABLE REEL
- X—FLUID HEAT EXCHANGER
- Y—
- Z—

COPY

FIG. III-29

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

NO. REQD.		NEXT ASSY.		MODEL		DRAFTSMAN		DATE	
HEAT TREAT		FINISH		STRESS		ENGINEER		SCALE	
TOLERANCE UNLESS OTHERWISE NOTED		STRESS		ENGINEER		SCALE		DWG. NO.	
LINEAR		XXX		APPROVED		PRODUCTION		RELEASE DATE	
TOL.		ANGULAR TOL.		INDICATES SURFACE ROUGHNESS		FINISH PER MIL STD-10 UNLESS OTHERWISE NOTED		ACT. WEIGHT	

MECHANICAL FLOW SHEET, SYSTEM "T" GAS TRANSFER CIRCUIT		AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA	
0-408060		DRAWING NUMBER	

LETTER	DATE	CHANGE

NOTE: THE BLOCKS AS SHOWN ARE FUNCTIONAL AND NOT PHYSICAL CHASSIS EXCEPT THE METER PANEL

CABLE LENGTH BETWEEN CONTROL CAB & REACTOR AREA IS 600'

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

FIG. III-30

NO. READ		TEST ASBY	MODEL	DEL. VALLE PREF. TAYLOR	DATE	4-10-59
TEST TREAT		TESTED	CHECKED	STRESS		
TOLERANCE UNLESS OTHERWISE NOTED		ENGINEER	APPROVED			
LINEAR	SE	C 03				
TOL	SEE	C 010				
INDICATES SURFACE ROUGHNESS		PRODUCTION				
FINISH PER MIL STD 10 UNLESS OTHERWISE NOTED		MATERIAL				
NUCLEAR INSTRUMENT BLOCK DIAGRAM-ML-1				AEROJET-GENERAL NUCLEONICS SAN RAMON, CALIFORNIA		
SIMILAR TO		SCALE	NONE	RELEASE DATE CAL. WEIGHT ACT. WEIGHT		
DRAWING NUMBER		0-408503				

LETTER	DATE	CHANGE	BY	CHK'D

0-408119

COPY

NOTE: MOUNT ALARM INDICATOR, RELAY, SET POINTS, SWITCHES, INDICATOR FOR R/HR. AND AMPLIFIER ON SAME PANEL.

FIG. III-31

No. REQD.		NEXT ASSY.		MODEL		E. B. SANTOS DRAFTSMAN		DATE 6/22/59	
HEAT TREAT		FINISH		CHECKED		STRESS		ENGINEER	
TOLERANCE UNLESS OTHERWISE NOTED									
LINEAR	.XX	± .05		APPROVED		6/23/59		RELEASE DATE	
TOL.	.XXX	± .010		ANGULAR TOL.		INDICATES SURFACE ROUGHNESS		SCALE NONE	
FINISH PER MIL STD. 10 UNLESS OTHERWISE NOTED				PRODUCTION		MATERIAL		DWG. NO. D	
BLOCK DIAGRAM- RAM SYSTEM, ML-1								AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA	
0-408119								DRAWING NUMBER	

PROCESS CONTROL PANEL

LETTER	DATE	CHANGE	BY	CHK'D

0-408083
COPY

FIG. III-34

SW-1				
CONTACT	POSITION	1	2	3
1-2	OFF	X		X
3-4		X		
5-6			X	
7-8			X	X

SW-2				
CONTACT	POSITION	OFF	ON	
1-2		X		
3-4		X		
5-6		X		
7-8		X		

L. DEL VALLE DRAFTSMAN		DATE 5-21-59	MATERIAL SCHEMATIC- POWER INSTRUMENTATION, CONTROL CAB ML-1	AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA
CHECKED 5/26/59		5-26-59		
HEAT TREAT TOLERANCE UNLESS OTHERWISE NOTED			STRESS TENSILE CAL. V. 5/26/59	
LINEAR TOL. XXX ± .001			APPROVED	
INDICATES SURFACE ROUGHNESS XXX PER MIL STD 10 UNLESS OTHERWISE NOTED			PRODUCTION	
MATERIAL			D	
DRAWING TO RELEASE DATE			SCALE NONE CAL. WEIGHT ACT. WEIGHT	
0-408083			DRAWING NUMBER	

ML-1 REACTOR ELEVATION

FIG. III-35

The diagram illustrates the TRIGA reactor system, showing the flow of coolant and the associated instrumentation. Key components include:

- REACTOR:** The central component, with a removable section for natural convective air flow.
- HE₃:** A heat exchanger located between the reactor and the steam generator.
- ST (Steam Tank):** A vertical tank connected to the steam generator.
- DM₁ and DM₂:** Differential manometers used for pressure measurement.
- IE₁ and IE₂:** Intermediate heat exchangers.
- P₁ and P₂:** Pumps used for circulating the coolant.
- Valves:** Various valves (SV₁ through SV₉, HV₁ through HV₁₀, DV, FCV) control the flow of coolant and steam.
- Sensors:** Temperature (TC), pressure (PC), and current (CI) sensors are used for monitoring the system.

The system is designed to allow for natural convective air flow, which is a key feature of the TRIGA reactor.

ML-1 FUEL ELEMENT REMOVAL

FIGURE 37

CONTROL BLADE MOUNT,
LINKAGE-TYPE, ACTUATED
BY A RECIPROCATING SHAFT

FIG. III - 38

APPENDIX A

TRANSIT SHOCK & VIBRATION SPECTRA

		<u>Load Factor G</u>			<u>Duration</u> (Sec)	<u>Freq., Cps</u>
	<u>Spec.</u> <u>Source</u>	<u>Long¹</u>	<u>Vertical</u>	<u>Lateral</u>		
I. Gen'l	(1)	<u>15</u>	<u>4</u>	<u>2</u>	--	--
II. R.R.	(2)	14	9	3.5	--	--
	(3)	5.6-14.6	3.2-4.8			
	(4)	30 (humping)	--	--	.40-.080	6-12 (20)
	(6)	--	--	--	--	Δ v=100 in/sec.
	(7)	2.6	2.9	2	--	2-15cps (vib) 70-230 cps (Transient)
	(8)	4-5.9	4.3-9.5	1-4	--	--
	(9)	20-22	--	--	.03	17
	(10)	--	--	--	--	3-70
III. Truck Trailer	(4)	--	8	--	.04-.08	6-12
	(6)	--	--	--	--	4-80 (>20)
	(9)	--	-4) +2)	--	.03	17
	(10)	--	--	--	--	3-150
IV. Aircraft	(4)	0.8	5.5	1.5	.010-.030	17-50
	(5)	8	4.5	1.5	0.1	5
		3	3	1.5	0.1	5
	(6)	--	--	--	--	0-100 (> 60)
	(9)	8	--	--	.03	17
	(10)	--	--	--	--	5-300
	(11)	8-16	4-18	--	.06-.17	6-17
Ship	(6)	--	--	--	--	0-20
	(10)	--	--	--	--	3-100

SOURCES OF DATA

- (1) Major Ednie to Gordon, 7-4-58, MR-RLL-T-75
- (2) Col. Lamback to Chang, 2-27-59, TCTMA FA/T (RCS1 #19)
- (3) Shock & Vibration in Rail Shipments, 2-20-59, TELECON
- (4) Air Force Spec. 106-A for Missiles
- (5) MIL-A-8421 A (USAF) & Ednie to Biehl, 2-10-59
- (6) Merrill = Missile Engineering Handbook, 1959
- (7) S. P. Railroad - Garin to Chang, 3-26-59
- (8) New York Central = Acceleration Studies Rail Freight Equipment
- (9) Weapon (bomb) Handling Specifications
- (10) ASTIA AD-204095, Dec. 1958
- (11) NACA Conference on Airplane Crash-Impact Loads, Crush Injuries and Principles of Seat Design for Crush Worthiness, April 17, 1956.

APPENDIX B

Inverted Sine Pulse

$$f(t) = 1 - \cos \frac{2\pi t}{\tau} \quad \text{for } 0 \leq t \leq \tau$$

= dimensionless excitation function

$$\frac{x}{x_s} = \frac{1}{2(1 - \frac{\tau^2}{T^2})} \left[1 - \cos \omega t - \left(\frac{\tau}{t} \right)^2 \left(1 - \cos \frac{2\pi t}{\tau} \right) \right], 0 \leq t \leq \tau$$

$$= \frac{1}{(1 - \frac{\tau^2}{T^2})} \left[\sin \left(\frac{\omega \tau}{2} \right) \sin \omega \left(t - \frac{\tau}{2} \right) \right], \tau \leq t \leq \infty$$

T = natural period of vibrating system

τ = duration of pulse

X = absolute displacement of mass

$$x_s = \frac{F}{k}$$

$$\omega = \sqrt{\frac{k}{m}} = \frac{2\pi}{T}, \quad \text{circular natural frequency}$$

F = Force

k = Spring constant

THE GAS-TURBINE TEST FACILITY

I. INTRODUCTION

Responsibility for the contract work scope on the GTTF was transferred from the Architect-Engineer Division of Aerojet-General Corporation to the Military Projects Division of Aerojet-General Nucleonics during the report period. Direct participation by AGN started on the project in January and included the direction of all work except that portion of facility construction yet to be completed. The A-E Division retained responsibility for completion of this work.

Responsibility of Aerojet on the GTTF contract includes the entire facility except procurement of the turbine-compressor (T-C) set. This item was contracted for separately by the Army Corps of Engineers. The delivery date for the T-C set has been set back by the supplier to August 1959 as the result of problems encountered in manufacture and assembly.

Major GTTF activities during the period 1 January to 30 June 1959 are reported in the following sections.

II. AGN SUBCONTRACT

A subcontract for overall direction of the contract work scope was reassigned, internally, during the report period. This internal shift of cognizance was made to simplify and expedite coordination of GTTF with other aspects of the Army Gas-Cooled Reactor Systems Program conducted by Aerojet.

III. STATUS OF GTTF WORK

The work accomplished at Fort Belvoir during the report period consisted primarily of facility construction and inspection, and acceptance of components purchased from various vendors by Aerojet. The machinery set was completely assembled with a dummy T-C set. The instrumentation was assembled and checked out. All other components are in place and construction work is complete except for minor wiring and mechanical work. The construction subcontractor, under Aerojet

direction, is cleaning the components and piping, and leak testing the system.

IV. SAN RAMON SUPPORT OPERATIONS

Technical support has been provided to the Fort Belvoir group by Aerojet personnel at San Ramon as required. Examples of this support are as follows:

Assistance in preparing test plans and procedures;

Calculating the first critical speed of the gear box quill shaft to determine whether the shaft design is safe at rated speed; and,

Investigating the coupling between the T-C set and gear box to determine removability and compliance with specifications for power transmission, speed and speed sensing capabilities.

Impingement start-up and dynamic torsional vibration modes of the T-C set are now being studied.

V. TURBINE-COMPRESSOR SET DELIVERY

Delivery of the T-C set to GTTF has been re-scheduled several times during the report period, and it is now estimated to arrive at Fort Belvoir in August 1959. Fabrication and assembly problems were the causes of the delay.

The slippage of the T-C set delivery compromises the ML-1 schedule, inasmuch as GTTF operational test data cannot be fully utilized in the design and procurement of ML-1 components, particularly the T-C set.

VI. OPEN CYCLE TEST PROGRAM

It has been determined that open cycle tests of the T-C set are for the best interest of AGCRSP. Working drawings were prepared for facility modifications required at the GTTF and procedures were prepared for the test. Approval has been requested of ERDL by Aerojet for the procurement and installation of the materials required for the facility modifications.

VII. CONTRACT MODIFICATIONS

Several contract modifications have been requested by Aerojet during the report period. These modifications have included, besides formalizing authorized changes, extension of contract completion date and justification of additional funds. In addition, the Contracting Officer has been informed that the committed contract

funds have reached the 85% point.

VIII. SPARE PARTS CONTRACT

A recommended GTTF spare parts program and a procurement proposal was submitted by Aerojet at the request of ERDL. Subsequently Aerojet was awarded a contract for procurement of the spare parts. Major items on this contract include a T-C set and recuperator to be furnished to ML-1 specifications.