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UC 81: "Reactors Power"  
TID-4500 (14th Ed.)  
Contract AT (10-1)-880

# ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

## MONTHLY PROGRESS REPORT APRIL 1959

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*Aerojet-General Nucleonics*

A SUBSIDIARY OF THE AEROJET-GENERAL CORPORATION

NEAR SAN FRANCISCO AT SAN RAMON, CALIFORNIA

**AGN**



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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

Monthly Progress Report  
April 1959

25 May 1959

by

The Military Projects Division

AEROJET-GENERAL NUCLEONICS  
San Ramon, California  
A Subsidiary of the Aerojet-General Corporation

Approved by *G. A. Linenberger*  
G. A. Linenberger  
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Contract AT(10-1)-880





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ARMY GAS-COOLED REACTOR SYSTEM PROGRAM

Monthly Progress Report for April 1959

ABSTRACT

This monthly progress report covers the activities of the Army Gas-Cooled Reactor System Program for April 1959. The program includes a water-moderated heterogeneous reactor (Gas-Cooled Reactor Experiment I), a graphite-moderated homogeneous reactor (Gas-Cooled Reactor Experiment II), a mobile gas-cooled reactor (ML-1), and the coordination of the Gas Turbine Test Facility. The pages that follow report the progress of each project, the associated tests and data evaluation, the applicable design criteria, and the fabrication of reactor components.

INTRODUCTION AND SUMMARY

GCRE-I is the first experimental reactor for the AGCRSP. The Idaho operations groups is now almost up to full strength, and continues to prepare and review procedures for the facility as well as conduct training sessions for technicians and supervisors.

The design and development effort is nearly complete. Production of the plates for the GCRE-I fuel element was 95% complete by the end of April. Part of the advanced fuel element program is developing an integral spacer: arc welding is the most promising technique at this time.

GCRE-II is an advanced back-up reactor system for GCRE-I. During April, preliminary core design continued. Fuel loaded graphite bodies have been obtained for tests. A thermal cycle test for fuel elements has been designed.

The ML-1 (Mobile, Low-Power Reactor) is directed toward constructing a field prototype power generating unit by January 1961. The program will incorporate design features generated by all previous programs. The preliminary hazards summary report was completed and submitted for approval in April. A specification was issued during April for the turbine-compressor set. Invitations were sent to 25 vendors for bids on the nuclear instrumentation. Fabrication started on test models of the control rod blades and tests will start in June. Preliminary design was completed on a reactor shielding design for an 18 ton package. Studies were made on possible methods for achieving a 15 ton package.

Parallel to the gas-cooled reactor development is coordination of the Gas Turbine Test Facility (GTTF) construction and operations. Information from this project will be combined with that from GCRE-I to make a complete power package. The government-furnished turbine-compressor for the GTTF is now supposed to be delivered on 12 June, although slippage in the delivery of the casing may change the date.



I. THE GAS-COOLED REACTOR EXPERIMENT-I

A. STATUS AND REVIEW

1. A/E Services and Idaho Operations

The A/E support of GCRE-I continued in the areas of field inspection, inspecting and expediting the work of the sub-contractors, and the preparation of change orders. By the end of April, the Idaho Operations Group was essentially completely staffed. The group continued to prepare and review procedures for the facility as well as conduct training sessions for technicians and supervisors.

2. Reactor Design and Development

The effort in this area nears completion. Core design is complete except for the design of handling tools for fuel elements and orifices. Core fabrication is nearly complete in many areas: the dummy fuel elements and source holders are 50% complete; the lead shield cooling system and moderator cooling manifold are complete except for inspection; the fuel element storage rack covers are complete except for anodizing and inspection, and the materials have all been ordered for the fission product monitor back-up and the gas duct stiffener. The control blades are ready for fabrication, a shutdown mechanism is being tested and two more are 90% complete. The shutdown mechanism is being cycled to study the long term characteristics of the system.

3. Reactor Instrumentation

The thermocouple harnesses are 95% complete, the detector head for the fission product monitor is being recalibrated, and a portion of the temperature loop is being re-wired before shipment to Idaho. All the equipment destined for Idaho is being inventoried and packed.

4. Fuel Element Development and Fabrication

The GCRE-I element plate production was 95% complete at M & C Nuclear at the end of April. The plates are processed and the elements assembled on a production basis. Four production core units (PCU) and one production instrumented unit (PIU) are complete. The slippage in clip production is delaying assembly.

Two I-3P fuel elements and draw bar assemblies were completed and shipped to the ETR (Experimental Test Reactor) site in Idaho during April.

The IA fuel element program has been directed toward the development of an integral spacer on the fuel plate edges. Forging, slitting, compression bending, puddle welding and arc welding are the new methods investigated. Arc welding is the most promising technique at this

time. This method produces a satisfactory joint, but it does distort the plate. A heat treating operation may overcome this distortion.

The inert components for the IA-1T are 90% complete. The completion of this element awaits the development of a satisfactory technique for attaching spacers to the plate.

A parametric study was undertaken of the IB element with 7, 19 and 37 pin clusters for the proposed ML-1 operating conditions. The study led to four systems, two designs with smooth pins, and two with finned pins. All satisfy the temperature criteria (1650°F hot spot) and have allowable pressure drop.

Gas corrosion tests, creep tests, and tube burst tests are continuing in the material evaluation study for the IB element.

Completion of the IB-1T awaits  $\text{UO}_2$  pellets. BMI is now preparing these pellets.

The capsule containing UN-stainless steel specimens (designated BMI 28-1) was inserted in the Materials Test Reactor in cycle 120. The specimens are being irradiated with the maximum specimen temperature of about 1750°F. Two capsules containing  $\text{UO}_2$  pellets (designated 27-1 and 27-2) are being irradiated, and are operating with a maximum specimen temperature of 1650°F. Modification of the Battelle Research Reactor loop has been completed and is undergoing a pre-operational check in the BRR pool. Installation of the BMI-16 components have progressed. Cold check-out will start during the first week in May: the major component yet to be installed is the in-reactor tube.

#### B. A/E SERVICES

The A/E support of the GCRE-I included field inspection, inspecting and expediting in the shops of Farnsworth and Chambers subcontractors, and preparation of change orders to the Increment I and II contracts.

#### C. IDAHO OPERATIONS

##### 1. Personnel

During the month, the GCRE-I operating organization was completed except for the Nuclear Engineer. The personnel status at month end is summarized on the following page.



<u>Title</u>	<u>At San Ramon</u>	<u>At Idaho</u>	<u>Total</u>
Operations Manager	1	0	1
Operating Superintendent	1	0	1
Shift Supervisors	3	1	4
Maintenance Supervisors	2	0	2
Staff	2	0	2
Technicians	6	7	13
Clerical	<u>2</u>	<u>2</u>	<u>4</u>
Total	17	10	27

## 2. Activity

The preparation and review of operating procedures for the GCRE-I facility continued throughout the month. As of the end of the month, the status of procedures is as follows:

<u>ANSOP SECTION</u>	<u>DRAFT COMPLETE</u>	<u>APPROVED LOCALLY</u>	<u>SUBMITTED FOR IDO APPROVAL</u>	<u>APPROVED IDO</u>
0 Office Procedure	17	8	2	2
1 Accountability	0	0	0	0
2 Radiological Safety	6	2	0	0
3 Non-radiological Safety	2	1	0	0
4 Security	0	0	0	0
5 Reactor Operation	133	73	37	0
6 Systems Operation	27	0	0	0
7 Mechanical Maintenance	10	0	0	0
8 Instrument Maintenance	12	0	0	0
9 Experimental Operation	20	0	0	0

Eleven informal training sessions conducted during the month gave the technicians and supervisors an opportunity to discuss phases of facility operations.

Administrative preparation for operations at the GCRE-I site included the following:

a. A purchase order was approved and issued to Phillips Petroleum Co. covering services and materials required at the site.

b. An interchange of correspondence completed the integration of the GCRE-I facility into the NRTS security system.

c. The establishment of an accountability station at the GCRE-I site was requested.

The 1/48th scale model of the GCRE-I facility was received from the vendor. A storage and shipping case (approximately 18 by 26 by 56 inches) was also supplied by the vendor. Figures 1 through 4 are photographs of the model.

D. REACTOR DESIGN AND DEVELOPMENT

1. Core Design and Fabrication (Task 32-80)

a. Design

1) Fuel Element Handling Tools

The types and numbers of tools needed to replace fuel elements and orifices have been determined. Preliminary sketches of concepts are complete, and designs are about 10% complete.

2) Other Designs

Designs have been completed for the fission product monitor back-up system, the duct connector, and the reactor external temperature monitor system. The only designs now needed to complete GCRE-I are the handling tools.

b. Fabrication

1) Dummy Fuel Elements and Source Holders

These items are 50% complete and no difficulty is anticipated in completing the job.

2) Lead Shield Cooling System and Moderator

These assemblies are complete except for inspection, and are awaiting the crates needed for shipment to Idaho.

3) Fuel Element Storage Rack Covers

These units are complete except for anodizing and inspection.

4) Fission Product Monitor Back-up and Gas Duct Stiffener

Materials have been ordered and some items are being fabricated.

2. Control Rod Actuators and Blades (Task 32-81)

a. Fabrication

1) Safety Mechanisms

A bid contract to fabricate two safety pistons was let on 23 April 1959. These pistons will be used as spares in Idaho.

2) Shutdown Mechanisms

A second shutdown assembly is being tested. Two more shutdown mechanisms are 90% assembled, and the two remaining assemblies are 50% complete.

3) Control Blades

The side stiffener modifications to the fine, shim and setback blades are complete and the blades are ready to be plated. The safety and shutdown blades have been completed and also are ready to be plated.

b. Testing

One shutdown mechanism is being continuously cycled to study endurance and reproducibility of scram time. Scram time is about 0.3 seconds instead of 0.5 seconds as reported earlier. There was some difficulty in cocking the scram spring, but the problem appears to have been solved.

Minor tests have been run to insure that the control rod leak detectors work properly. A rust-inhibitor-type electrolyte (sodium dichromate) will be used in the control rod housings to increase the conductivity of leakage water and assure proper operation of the detector. This step was necessary because of the low conductivity of the water in the GCRE-I pool ( $1\mu\text{mho}$ ).

Simple experiments are being conducted to determine the extent of corrosion of uncoated lead in peroxygenated water (0.03%) solution which simulates the radiolytic decomposition of the moderator during operation. A coating is being developed which can be applied in the field to repair any damage to the coating on the lead reflector.

Unusual drag appeared during the test of a shutdown blade in a Sanfordize-coated guide. The manufacturer explained that a 0.002 in. coating gives a softer coating than the optimum coating of 0.0013 in. An extra guide assembly has been stripped of the coating and will be tested with the optimum thickness. A back-up guide design has been developed incorporating rollers. This design will be tested early in May.

## E. REACTOR INSTRUMENTATION

### 1. Temperature Measurement (Task 32-70)

The thermocouple harnesses are 95% complete, and lack only the attachment of the 52-pin plugs. These plugs are being assembled to the Marman flanges, after which they will be leak checked. The completed harnesses and the instrumented fuel element will be tested as a system.

### 2. Fission Product Monitoring (Task 32-71)

The detector head for the fission product monitor is being recalibrated by extending the range of the energy defection efficiency curve with Cobalt-60 (1.17 mev and 1.33 mev) and Cesium-137 (0.662 mev).

### 3. Control Console and Panel (Task 32-76)

The period test circuit and the heater bypass amplifier used in the temperature loop are being re-wired before shipment to Idaho. All the equipment destined for shipment to Idaho is being inventoried and packed. This includes the health physics equipment, thermocouple scanner, spare parts, and special test equipment.

## F. FUEL ELEMENT DEVELOPMENT AND FABRICATION

### 1. Fuel Element I

#### a. Fuel Plate Manufacture at M&C Nuclear (task 34-34)

Production of GCRE fuel plates was about 95% complete by the end of April. All except one or two lots of size "C" plates were scheduled to be shipped on or before the end of April. The total number of enriched plates shipped when the job is complete will be 295 size "A", 306 size "B", 310 size "C" and 338 size "D". There may be some additional size "C" plates. These quantities include certain plates rejected by Azusa for surface defects and a number of plates with inert margins slightly below the 0.025 in. minimum.

Further studies of the surface defects show that minor changes or improvements would not prevent all such defects. Etched pits occurred primarily on size "B" and "C" plates and almost none were observed in the "A" and "D" plates. It was found that pits deeper than about 0.0005 in., and some types of hidden cavities below the surface, were clearly indicated by a dye penetrant test. Such surface inspection could be performed by Aerojet after the plates are formed.

M & C tested about 1200 plates and found only one definite case where the boiling  $\text{HNO}_3$  test showed that some  $\text{UO}_2$  was leached

from the matrix of the plate. A number of plates showed surface contamination, but this dropped off to background levels when the plates were tested individually. The single leak was attributed to surface pits visible to the naked eye.

The batches of plates with the best visible surface conditions were also the best according to the boiling  $\text{HNO}_3$  and the bend tests.

A few failures occurred in the bend tests when the center of the bend was near the side margin of the plate. In these cases the bond failed between the matrix and the inert margins. The failures were attributed to those isolated cases where fine  $\text{UO}_2$  particles (1 to 10 microns) were formed at the side bond, presumably due to fractures or abrasion of larger  $\text{UO}_2$  particles. The fine  $\text{UO}_2$  particles were not present in other plates in the lots which failed to pass the bend test.

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

Four PCU and one PIU were completed. The plates are being processed on a production basis. The surface quality of the plates has deteriorated in later shipments. Photographs of typical defects in surface cladding are shown in Figures 5 through 8. Some of these defects result in cladding failure during forming and show up in fluormetric analysis. The combined visual inspection and acid etch rejection rate is about 10%. The defects appear to be fine pits, surface porosity, cladding laps, and foreign matter rolled into the surface. (See Figure 7). In many cases the particles are more distinct after the acid etch. (The condition is also apparent in photograph 4, Figure 6). The original acid etch criteria was a maximum of  $24 \mu\text{g}$  uranium in a 4.000 liter batch of etch solution. The next acceptance criteria was  $5 \mu\text{g}/\text{plate}$ .

The assembly schedule is being delayed by slippage in the production of other components. Only 16 zirconium cans were available for fabrication by the end of April. Ten orifices have been received, and the vendor has been authorized to fabricate the 100 units remaining. Seventy-nine end casting plugs are available for fabrication. Clips are pacing the production program at this time. Clip production is behind schedule, and the quality does not always meet the specifications. In some cases the  $30^\circ$  angle on the arm has been missing, and in others it has been on the top side of the notch instead of the lower side. The arms frequently have been mis-aligned as much as 0.004 in. (Typical "B" and "C" clip cross-sections illustrating these defects are shown in Figure 9). Rework is required on such clips if the plate fit-up is to meet the required flow channel dimensions.

Inspection procedures have been established to insure that adequate inspection records are maintained on production items. The system was begun with the I-3P units. The pressure vessel has been completed for the leak test of the PIU pressure seal bushings and flange arrangement.



c. I-3P Inpile Test (Task 23-95)

The data recorded during the critical experiment for the BMI-16 loop was analyzed during April. Fabrication began on a silver neutron shield 0.092 in. thick. The purpose of the shield is to maintain the desired power level in the test element by reducing the neutron flux seen by the test element. The critical experiment also determined the axial neutron flux distribution expected in the I-3P fuel element. The experimentally-determined flux distribution was compared to the chopped cosine curve originally assumed for the IBM-650 heat transfer calculations, and it was found that the former is flatter than the assumed chopped cosine curve and has rising tails at each end of the fuel element. This tail shifts the element hot spot slightly towards the gas exit end of the fuel element.

Two I-3P fuel elements and drawbar assemblies were shipped from Azusa to the ETR site in Idaho on 17 April 1959. The assemblies were visually checked, and the thermocouples were resistance-checked for damage at the ETR site. The inspection showed damage to thermocouple 4 on unit 1, this being the only damage found.

d. Dummy Element (Task 23-62)

Fabrication of the dummy plugs is 50% complete.

2. Fuel Element IA

a. Engineering (Task 23-90)

Effort on the IA program during April was directed toward the development of an integral spacer. This is described under the following section, Task 23-94.

b. Fabrication Development (Task 23-94)

(This task continues the work previously reported under Task 23-45)

Additional methods for making a fuel plate with an attached spacer have been explored in an effort to find a process with more reliability and less cost than those processes developed to date.

Forging methods were investigated and demonstrated that plate edges can be forged to the widths needed for GCRE spacers. Scrap plate edges were set in a simple die placed in a vise. A block was laid on the edge of the plate protruding from the die, and the block struck with a hammer. The plate was heated to about 1500°F between blows. Some variations were made in plate protrusion, height and stress relieving temperature. Plate edges were forged to the widths needed, but forging made from greater widths had split edges and undesirable laps in the metal. The metal in the position occupied by  $UO_2$  in a located plate

did not show any flow or adverse conditions which might be encountered during forging of loaded plates. A microphotograph of a forged specimen is shown in Figure 10 and a photograph of a specimen after machining is shown in Figure 11. No further work is now planned on this process because other processes appear more promising. In addition, the production problems of forging would probably lead to higher costs.

The edge of a short piece of fuel plate scrap has been successfully slit with a slitting saw. The considerable wear shown by the tool, coupled with the problem of holding tolerance on a full length plate, has restricted exploration of this approach.

Two features have been added to the process of cold bending alternate flaps on a fuel plate. In the process now called compression bending, the flap is heated to about 1500°F halfway through the bend, and the final one-third of the bend made while forcing the flap back toward its root. Sixteen flaps have been made this way without a visible sign of crack failure. The plate will be further inspected for cracks and microphotographs made to determine the final position of the  $\text{UO}_2$  stainless steel area and the bend. This work has been done with readily obtainable but rather coarse dies. A more suitable die is being developed for reliable and consistent fabrication of flaps. The exact heat treatment requirements will also be studied.

Puddle welding is a heliarc process wherein a fuel plate is clamped in a fixture with the edge of the plate projecting into a channel: weld metal is then flowed into the channel to merge with the plate edge. This has worked with short sections of unloaded scrap fuel plate in a channel made of graphite. The metal can be placed around the plate edge so there is no weld penetration of what would be the  $\text{UO}_2$  region in a loaded plate. The weld metal was rough in these early attempts, but machining to final size undoubtedly will always be needed. A microphotograph of a puddle-welded and machined specimen is shown in Figure 12. This method produces considerable distortion of the fuel plate. Puddle welding shows fair promise, but no further action is planned at this time because two other promising approaches are entering the die and fixture development stage.

Arc-welding thin strips to the edge of a fuel plate now is the most promising technique. Figure 13 shows one of the first full length pieces of flat steel with one welded spacer. The distortion is 0.100 in. camber, but it might be less for a curved plate with two welded edges. Such distortion probably can be adequately corrected with the heat treat operation now used on fuel plates. A microphotograph of the weld area is shown in Figure 14. The fillet between the spacer and the plate is desirable. The penetration reaches into what would be the fuel zone. This may not be allowable if it weakens the structure, or lets  $\text{UO}_2$  particles get to the surface. However, further development of welding techniques may eliminate this problem. Welding development will continue, and close tolerance fixtures will be available for full length plates in May

## c. IA-1T Inpile Test (Task 23-96)

The inert components for the IA-1T elements are 90% complete. Completion of the IA-1T element awaits development of techniques for attaching spacers (cf Task 23-94).

3. Fuel Element IB

## a. Engineering (Task 23-91)

A brief survey of new techniques of loading uranium in pin-type elements was completed. It is believed that uranium carbide and nitride technologies are not firmly established although these materials are potentially better fuels. Uranium dioxide apparently is still the best fuel for use in this program at this time. Many loading techniques have been conceived and some have been worked on at other sites. Three techniques of note are: the extrusion of  $UO_2$  cylinders, isostatic pressing, and swaging of powder-filled tubes. The ultimate reduction in cost is the incentive for these processes. These processes now give lower density ceramic bodies than pelletizing and would be expected, therefore, to give shorter burst lifetimes as governed by the release of fission product gases. The IB pin, however, has the design leeway, within the accuracy of present theory, to achieve 10,000 hr lifetime with the use of low-cost loading techniques. Activity will be resumed later on the problem of new loading techniques.

The present reference IB design is being refined for insertion and long-term irradiation in the GCRE. The first element to test this design (denoted IB-R) is due to go into the GCRE in the middle of fiscal year 1960. The basic fuel element design incorporates features of the IB-1A<sub>T</sub>, including the use of 19 smooth pins. Instrumentation will be of the IB-1A<sub>T</sub> type using disc thermocouples. The principal design features, therefore, will have been tested in the BRR loop before the IB-R elements are assembled. The present analytical effort is devoted to improvement of temperature optimization among the 19 pins. This involves IBM machine computations of power distributions as well as flow and temperature distributions.

A parametric study of 7, 19, and 37 pin clusters was undertaken for the operating conditions currently proposed for the ML-1. These numbers of pins were chosen because they make well-spaced geometrical arrays, and cover a significant range in total number. Interrelations were established between maximum pin surface temperature, reactor pressure drop, size of pins, and amount of heat transfer surface. The heat transfer surface was varied by using fins and changing pin diameter (though simple correlations were used to predict fin temperature). The relationships led to the following systems: smooth surface, 0.150 in. OD tubes in clusters of 37; smooth surface, 0.23 in. OD tubes in cluster of 19; 0.19 in. OD tubes with fins in clusters of 19; and 0.38 in. OD tubes with fins in clusters of 7. These systems satisfy the temperature criteria (1650°F hot spot) and all have acceptable pressure drops, although the clusters with more pins should have lower pressure drops peak temperature being equal. On the

hand, increasing the number of pins increases the possibility of mechanical instability during operation, and increases fabrication cost. Further work is needed to establish comparative burst lifetimes of the systems, but present knowledge indicates that any one of these systems is capable of achieving the required lifetime.

b. Fabrication Development (Task 23-93)

Some of this task effort has contributed to the IB-1T element reported under Task 23-97. The fuel pins for this element now are to be brazed and welded at Aerojet. These procedures are being prepared and checked.

Plans have been made to braze the bearing spacers to pins, load pellets in pins, and weld the pins closed. Equipment is being procured and tools and fixtures for this work are being fabricated. The hood, to be used in handling pellets and loading them in pins, is being installed. A glove box is on order for use when welding the pins. Since the IB pins use nickel-base alloys instead of zirconium, the welds are not sensitive to oxygen and a helium glove box can be used in place of a vacuum chamber.

c. Material Evaluation (Task 23-65)

1) Gas Corrosion Testing

The first 2500 hr phase of a 10,000 hr gas corrosion test is due to be completed about mid-May. The primary interest is in Hastelloy X, Hastelloy R-235 and Inconel 702; but Carpenter 20 Cb and Hastelloy 8286 (Inor 8) have been included for screening purposes, and AISI type 318 as a comparison alloy. The test operates at 1750°F in a 200 psi pressure nitrogen plus 0.5% oxygen atmosphere, except for AISI type 318 which will be tested at 1650°F.

A 1000 hr screening test, in which these alloys will be exposed to an air atmosphere, will also be completed in mid-May. The alloys will be exposed to 300 psi at 1750°F, except for AISI type 318 which will be exposed to 1650°F. Data on the behavior of AISI type 318 in air will be compared to the data obtained in the nitrogen plus 0.5% oxygen test. Air corrosion data on these alloys are needed to determine the compatibility of air with the system in case air must be used as a working fluid.

A test is being performed at 200 psi and 900°F in the reference nitrogen plus 0.5% oxygen atmosphere on GTTF recuperator materials. Details of this test are described in a later section of this report.

## 2) Creep Testing

A 1000 hr creep test to evaluate Hastelloy X, Hastelloy R-235 and Inconel 702 at 1750°F should be completed about mid-May. The specimens in this test are stressed at 200 psi in air. Additional long-term creep tests are planned to evaluate these alloys at other stress levels.

## 3) Tube Burst Tests

These tests are made to evaluate the modes of tube failure and the tube strength when subjected to internal stress such as that exerted by the release of fission gas. It is hoped to extend this approach into long-term stress corrosion and creep.

Seven trials were performed using 0.225 in. OD Inconel and Inconel 702 tubing. The first five tests were stopped when the welds failed before the tubes were brought to temperature. Two successful tests were performed. In these, one tube contained an initial pressure of 1125 psi, and the other 980 psi. The pressures at 1750°F (1270 and 1130 psi respectively) correspond to stresses of 3514 and 2656 psi, respectively. The tubes failed by brittle fracture after three hours at temperature. Figure 15 shows the conditions observed after tube failure. The tube with initial pressure of 1125 psi increased 0.040 in. in diameter, and the other increased 0.005 in. in diameter.

The tests were run as follows: the tubes were evacuated, charged with various internal pressures of nitrogen gas, and placed in a tube furnace. The furnace was gradually brought to 1750°F while the pressure changes were plotted at 100°F intervals. Several tubes of reference alloys (each tube containing different gas pressures) will be exposed to reference GCRE atmospheres and temperatures for 1000 hr. It is planned to modify the existing equipment to allow use of greater pressures.

### d. IB-1T Inpile Test (Task 23-97)

The pellet and fuel pin purchase order to a commercial vendor was cancelled due to procurement problems. The UO<sub>2</sub> pellets will be made by BMI, and the pellets loaded in pins at Aerojet.

The IB-1 $\alpha$ T element is instrumented as far as possible (60%) before receipt of the fuel. After the receipt of the fuel pellets (early in May) the pins will be loaded and the assembly completed.

The status of the non-fuel components to be machined is: IB-1 $\alpha$ T - - complete; IB-1 $\beta$ T - - complete; and IB-1 $\theta$ T - - 75% complete.

## 4. Battelle Memorial Institute (Task 23-53)

(Note: the following tasks are delineated under the BMI designations).



a. Design and Fabrication of a Capsule for Irradiation of UN and  $\text{UO}_2$ -Stainless Steel Dispersion Fuels (Task 1-E)

This capsule (designated as BMI-28-1) was inserted in MTR cycle 120 on 6 April 1959. The specimens are being irradiated with the maximum specimen temperature of about 1750°F.

b. Design and Fabrication of a Capsule for Irradiation of Solid  $\text{UO}_2$  Specimens (Task 1-F)

Two more capsules, similar to the one in Task 1-E, have been fabricated. Each capsule (designated BMI-27-1 and 27-2) contains six Inconel clad, solid  $\text{UO}_2$  pellets (35 wt % enrichment) pins. BMI-27-2 began irradiation in reactor cycle 119 on 20 March in a thermal neutron flux said to be  $0.8 \times 10^{14}$  nv. This is below the design flux for the capsule. The capsule is to operate at a maximum specimen temperature of 1650°F.

Capsule 27-1 identical to 27-2 was inserted in MTR cycle 120 on 6 April, and is operating at a maximum specimen temperature of 1650°F.

The table below summarizes the initial capsule temperature performance as monitored by thermocouples mounted close to the surface of the specimen. The axial flux peak is at the capsule center, but the specimens at the ends of the capsule operate at a higher temperature than those in the center. Reasons for the behavior of the capsule are being investigated.

IN-PILE TEMPERATURE DATA FOR CAPSULE 27-2 DURING MTR CYCLE 119  
FULL REACTOR POWER

Specimen (a)	Initial Temp. (°F) After Startup, No Auxiliary Heat In- put (b)	Temp. (°F) After Startup with an Auxiliary Heat Input of 600 w	Temp. (°F) After 10 days Irradiation Auxiliary Heat In- put Varying between 100 and 500 w
1 (c)	1250	1400	1370
2	820	1020	1320
3	930	1160	1110
4 (c)	1175	1260	1255
5	1420	1625 (heater control)	1625 (heater control)
6	1420	1550	1550

(a) Specimens numbered consecutively from top to bottom of capsule.

(b) The specimen surface temperature is approximately 75°F higher in all cases than the thermocouple indication.

(c) Specimens 1 and 4 are SIR-type pins, and the others contain solid  $\text{UO}_2$  pellets.

c. Modification of the BRR Loop (Task 1-G)

The 10 hp blower was disassembled and reassembled twice during this period to reduce the leakage of oil through the labyrinth seal. This leakage was reduced before the start of additional loop assembly. The complete system was leak-tested and the assembly returned to the reactor pool after the leaks were sealed.

d. BMI-16 Loop Fabrication - ETR (Task 1-H)

Power and control wiring is being installed, and is completed in the basement and machinery room. Final connections have not been made. The machinery room door and shielding wall are 95% complete, but will not be finished until all wiring has been installed.

The damaged console has been repaired. All wiring and mechanical installation needed for performance of a cold checkout without the inpile tube is finished. Instrument calibration and inter-panel wiring have also been completed.

The parts needed for insertion and removal of a fuel specimen have been fabricated. These include the drawbar shield, insertion attachment, open loop shield, concrete shield, spool piece and spool piece adapter. Two fuel specimen casks were completed and turned over to a vendor to have the lead poured. Shipment to the site can be completed after checkout of the drawbar shear. The items needed for loop removal as well as one in-reactor tie bar, an in-reactor hanger and a particle filter shield were completed.

Reactor flux data obtained from the ETRC experiment and information obtained for Phillips Petroleum Co., since the ETRC experiment were used in building new attenuator blocks.

e. Post-Irradiation Examination of the I-1T (Task 1-J)

Radiochemical burnup analysis are being conducted on sections removed from the axial center of fuel plates A-1 and D-11.

## II. THE GAS-COOLED REACTOR EXPERIMENT II

### A. SUMMARY AND REVIEW

This project continued preliminary core design during April. Fuel loaded graphite bodies have been obtained for tests. A thermal cycle test for fuel elements has been designed.

### B. CAPSULE TESTS (Task 43-61)

Minnesota Mining and Manufacturing Co. currently is applying a thin layer of unfueled graphite on the surface of fueled graphite bodies prior to coating with silicon carbide so that the surface of the final coat will be free of uranium silicide. The process is being applied to 8 wt % natural  $\text{UO}_2$  graphite bodies in preparation for coating the enriched  $\text{UO}_2$  fuel graphite specimens for the II-1C in-pile capsule. The coated enriched specimens are scheduled to be delivered to BMI by 11 May for incorporation into the II-1C capsule. This capsule is scheduled to go into the MTR on 29 June.

Test and design specifications have been completed for the II-2C capsule. This capsule is intended to provide long-term corrosion and integrity data in a reference gas environment. It will be designed, fabricated, irradiated and examined by BMI.

### C. MATERIALS TESTING AND EVALUATION (Task 43-64)

#### 1. Fuel

Two other vendors, in addition to Great Lakes Carbon Co. are willing to prepare fueled graphite to GCRE-II specifications. Samples will be obtained from both Minnesota Mining and Manufacturing Co., and National Carbon Co. for comparison with the material prepared by Great Lakes Carbon Co.

Negotiations are under way with Great Lakes Carbon Co. to determine certain basic physical characteristics of 8 wt % fueled graphite. The material already has been prepared by Great Lakes and partial shipment received by AGN.

Three vendors -- National Carbon Co., Minnesota Mining and Manufacturing Co., and Brush Beryllium Corp. -- are willing to undertake programs to study the feasibility of incorporating beryllium oxide or beryllium carbide in fueled graphite. Great Lakes has not yet decided whether or not they wish to work with beryllium compounds.

2. Coatings on Fuel

National Carbon is a potential second source of silicon carbide coatings on fueled graphite.

Alloy'd Research Corp. and Liquid Metals, Inc. can produce barrier coatings (such as molybdenum, chromium, etc.) on graphite by vapor deposition. Specimens of barrier coatings on fueled graphite will be procured for compatibility tests.

3. Cladding Alloys

Corrosion tests in the reference atmosphere continue on Hastelloy X, Inconel 702, and Carpenter 20 Cb at 1750°F and 200 psi. The initial 2500 hr phase is scheduled to be completed by mid-May.

Initial (1000 hr) corrosion tests in air on the above alloys at 1750°F and 300 psi will be completed in mid-May. Initial (1000 hr) creep tests on Hastelloy X and Inconel 702 at 1750°F are also scheduled to be complete in mid-May. Details of the gas corrosion and creep tests are reported in Task 23-65.

4. Testing

The helium leak tester is scheduled to be completely installed in the first week of May. The initial tests will take place on graphite bodies canned at Aerojet.

5. Reflector Materials

Inquiries were sent to suppliers and fabricators of reflector materials. The Brush Beryllium Corp. can produce massive shapes of high density in beryllium and beryllium oxide. Metals Hydrides Corp. can produce high density shapes of zirconium hydride.

D. CAPSULE DESIGN (Task 43-62)

Work on the design of the II-2C capsule was terminated at Aerojet this month. The design will be continued at BMI. A meeting of cognizant BMI and Aerojet personnel was held to discuss design and fabrication of the capsule, and the test objectives and conditions. Work will begin at BMI early in May.

## E. FUEL ELEMENT PRELIMINARY DESIGN (Task 43-91)

An IBM-704 Code (CHOP), which is being developed primarily for ML-1 cycle analysis, will be modified to make it useful, also, to the GCRE-II system. The first application of CHOP to GCRE-II will be to resolve, on the basis of total system weight, the choice between zirconium hydride and beryllium (or beryllium oxide) as the reflector material. The detailed cycle analysis is needed because the loss in cycle efficiency resulting from the increased cooling requirements of zirconium hydride may more than offset the savings in shielding weight offered by the zirconium hydride reflector.

The geometrical considerations which determine the dead space formed between the outer ring of the fuel element hexagons and the cylindrical pressure vessel wall were investigated. The study determined that this dead space could be kept under 13% of the total core volume if the number of hexagons sized to fit the vessel was one of the following: 31, 55, 85, 102, or 121. Variations from these numbers, below 121, result in an increase in dead space. The decision on the number of hexagons will be based on many considerations, among which are metal fraction, fabrication feasibility, and stress characteristics. The exact hexagon number and size, however, will be based on the required core dimensions and one of the above numbers.

A welding test slug, similar to the Carpenter 20 Cb slug reported last month, was made of Inconel 702. The welding results were visibly inferior to the slug of Carpenter 20 Cb. The welding difficulty was attributed to the presence of graphite dust in the vicinity of the weld. Additional studies of Inconel 702 welding will be made under more controlled conditions.

A small scale thermal cycling test unit has been designed. Purchase orders for about 90% of the required materials have been placed for delivery before 1 June.

A schematic flow diagram of the system is shown in Figure 16. The main components 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 fabrication drawings for the test element are complete. This element will contain

a 3 in. section of graphite impregnated with natural  $\text{UO}_2$  and silicon coated; two graphite spacers 3.5 in. long; and two firebrick insulators 0.250 in. thick. The canning and tubes will be fabricated from Inconel 702. The hexagonal can will be 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 has been designed to remove the heat generated in the test section and by the re-circulating blower. Water for the heat exchanger will be cooled in the existing cooling tower.

### 4. Re-circulating Blower

A Roots type blower has been chosen for re-circulating the system gas. The blower will be housed in a pressure shell since the gas pressure is 400 psi. The designed capacity of the blower is 1600 lb/hr of nitrogen at inlet conditions. This capacity is sufficient to cool the test element from  $1400^\circ\text{F}$  to  $400^\circ\text{F}$  in 30 minutes.

### 5. Control Circuit

Two manual set, interlocking meter relays are used in the control circuit. One meter relay will control the cycle temperature range of the test specimen through remote operation of system bypass valves and the test section heater. The other meter relay is 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 to detect the presence of helium in the loop. 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.

## F. NEUTRONICS (Task 43-93)

PDQ calculations were performed in April to determine the advantages of using a fuel-moderator body composed of graphite containing beryllium carbide. The initial results show a saving in both diameter and length of about 5 in. for a 60 v/o graphite-40 v/o beryllium carbide material compared to plain graphite. A more exact evaluation will be made; however, it is certain that a substantial amount of shield weight would be



saved if such a material were used.

PDQ calculations show that the reactivity worth of the central island of the reference (graphite) reactor is essentially the same as for the equivalent amount of fuel. The calculations also showed that the use of the equivalent amount of fuel leads to a more favorable peak-to-average power ratio. The choice of whether or not to use a central moderating island must be based on questions of control worth.

G. DESIGN AND FABRICATION OF A CAPSULE FOR IRRADIATION OF  
FUELED GRAPHITE SPECIMEN - BMI Task 1-I

Capsule 29-1 is similar in design to capsules 27-1 and 27-2, but contains six clad  $\text{UO}_2$  (8 wt %, highly enriched) graphite specimens. The design surface temperature is 1750 F. It is expected that this capsule will start irradiation in June.



III. THE ML-1 (MOBILE, LOW-POWER REACTOR)

A. SUMMARY AND REVIEW

The Preliminary Hazards Summary Report, IDO-28537, was completed and submitted for approval during April. This report covers operation and maintenance of the ML-1 power plant at NRTS, but was not extended to include operation at other sites.

A specification for the turbine-compressor set (AGN TM-353) was issued during April. This specification will be used in procuring spare parts for GTTF under a separate contract.

Invitations to bid on the nuclear instrumentation were sent to 25 vendors.

Fabrication started on test models of the control rod blades. Tests will begin in June.

Preliminary design was completed on a reactor shielding design for an 18-ton package. Studies were made on possible methods of achieving a 15-ton package through the use of tungsten, and by relaxing the requirements for shutdown radiation shielding.

B. AUXILIARIES

1. Facility Design Criteria (Task 55-100)

The degree of ML-1 dependence on support facilities is being determined through a follow-up study. Certain maintenance requirements were assumed to provide a basis for such a study. It was also assumed that failures will occur in a critical component of either the reactor or the power conversion equipment. Consideration was given to mechanical design and radiation hazards in analyzing the methods and procedures of disassembly-assembly of such components. Tentative work locations and tool requirements have been established. Information on suitable "hot shop" facilities at NRTS has been reviewed.

2. Shock Mounts (Task 55-200)

The lack of statistics and the inconsistency of available data on dynamic environments made it necessary to formulate

arbitrary ground rules for the design criteria of the prototype shock and vibration isolation system. If the results from full scale tests so require, the design criteria will be modified.

The Aerojet test facilities in Sacramento were reviewed. A railroad track and switch engine for "humping" tests is available. The environmental test facilities, excluding a temperature chamber, will not be available until 1960. Technical assistance on test instrumentation is available.

3. Field Applications (Task 55-300)

Final design of the reactor and power conversion mockups have begun. The packages will be fabricated from reinforced concrete and will include tie-down points at the same locations as on the ML-1 packages. The tie-down plan provides supplementary shock mounting as well as distribution of the shock loads to the aircraft or vehicle structure. The mock-up skids will be provided with shock mountings for future shock and vibration tests.

4. Fluids Processing (Task 55-400)

Ammonium pentaborate tests were completed and a summary report is being written for publication in May. Excerpts from this test report will be included in the May monthly progress report.

The current reactor package design is being reviewed to provide a basis for proceeding with the water treatment equipment design. Tentative sizes have been established for the moderator system bypass demineralizers.

Feasibility studies are continuing on the nitrogen gas manufacturing unit.

C. POWER CONVERSION EQUIPMENT

1. Design and Specification (Task 56-100)

A specification for the turbine-compressor set was issued during April as AGN TM-353. This specification will be used to procure GTTF spare parts under a separate contract. The GTTF turbine-compressor set is being reviewed to determine the feasibility of using a similar unit on ML-1. The review, now 75% completed, will be finished in May.

Specifications for the recuperator are about 80% complete. The conceptual design of the recuperator is about 80% complete, and should result in reduction of the thermal stresses anticipated in the GTTF unit.

The conceptual design of the pre-cooler indicates it is possible to achieve an approach temperature of approximately 25°F at maximum ambient conditions with the new reference design. This new design fits within the lateral area restrictions of the ML-1 power conversion skid. The reference design incorporates a perforated plate fin instead of the plain plate fin used on GTTF, and uses four fans which require about 45 hp. It may be possible to further improve pre-cooler geometry. It may be necessary to raise the pre-cooler by means of jacks when the skid is in place in order to achieve proper inlet flow conditions.

Optimum arrangement of the major components on the power conversion skid is being studied. Increased generator speeds, in-line arrangement of the recuperator, modifications to the turbine, generator and gear box are being investigated.

## 2. Materials Research & Development (Task 56-200)

A development program was started late in March to determine the brazing cycle for type "A" nickel pin fin structure for use in the ML-1 recuperator. The effects of the four variables in the brazing cycle will be studied in a four-stage program.

Stage I. Establish proper thickness of braze materials.

Stage II. Establish effect of dwell time at braze temperature on metal to brazing material alloying.

Stage III. Establish the effect of heating and cooling rates from braze material melting temperature to brazing temperature and back to solidus temperature.

Stage IV. Establish effect of maximum braze temperature.

Samples have been ordered to the necessary specifications. Completed samples of one stage are visually examined before samples are ordered for the next stage. The samples are sections three pins wide by 2 in. long made with type "A" nickel sheet and type "A" nickel pins. (Typical samples are shown in Figure 17.) The size and shape of these pins are identical to the pins used in the GTTF recuperator. Copper braze was used as a reference, backed up with two silver alloys and one nickel alloy. The results of the first two stages of this program are:

Stage I. Five thicknesses of copper braze material were used. A minimum thickness of 0.0030 in. of copper was established. AMS 4772 silver braze alloy provided the best joint of the back-up materials, and a thickness of 0.0040 in. was the most satisfactory.

Stage II. Four dwell times were run, ranging from one-half hour to three hours at three different brazing alloy equivalent foil thicknesses. All of the samples were satisfactory. No erosion was noted for the long dwell cycles.

The above samples (and some electroless nickel-coated GTTF recuperator samples) are included in current gas corrosion tests. They are being tested in 300 psi reference ML-1 nitrogen atmosphere and 300 psi air at 900°F. Comparative strength tests will be made on reference and exposed samples to determine the effect of exposure on the braze joint.

Pin fin core samples made of the same materials as the GTTF recuperator have completed a 2000 hr gas corrosion test. This core has Armco iron pins between sheets of AISI type 430 stainless steel. The gas composition is reference ML-1 nitrogen at 200 psi and 900°F. One of the exposed samples is pictured with an unexposed sample in Figure 18. The exposed sample had an over-all dark blue color. An adhering scale was deposited on the stainless sheet and a soft, powdery scale was built up on the Armco iron pins. Microphotographs are being made to determine the corrosion penetration. The table below lists the weight gain recorded for the 2000 hr test.

#### GAS CORROSION TEST

##### WEIGHT CHANGE-GTTF TYPE PIN FIN RECUPERATOR CORE

ATMOSPHERE: ML-1 REFERENCE NITROGEN AT 200 PSI 900°F

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

Sample No. 3 1000-hour test duration. All other samples weighed at 2000 hrs.

#### 3. Systems Analysis and GTTF Experimental Planning (Task 56-300)

A program has been completed to use a digital computer to analyze ML-1 steady state performance. Accuracy of the program will be determined and initial calculations are scheduled to start in May.

#### 4. Electrical Equipment Design (Task 56-400)

The reference design for the electrical switch gear is as follows: 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. Power for the plant auxiliary equipment is tapped off at the generator side of a circuit breaker and is transformed down to 480 v three-phase. All of the auxiliary motors, heaters and lights operate at 480 v. A transfer switch on the secondary side of the 4160/480 v transformer allows the auxiliary equipment to be driven either by the main generator or by an auxiliary power supply.

Commercially available equipment meets the general shock and vibration requirements for the ML-1 plant. A preliminary design report which includes this equipment has been completed.

Tentative plans have been established for sequencing the electrical equipment for the different types of shutdown required during operations at Idaho. In general, the plan is to drop the ML-1 to minimum power compatible with the malfunction which has occurred within the power plant system. An auxiliary power supply will be used as required for the acceptance testing at Idaho until the plant has proven that it can operate without dependence on auxiliary power.

A preliminary investigation has been made of the effects of radiation damage to the electrical components. No trouble is expected with any of the electrical components on the power conversion skid except where lubrication is involved. Further investigation is being made of the damaging effects of radiation on bearing lubricants.

The use of 400 cycle equipment is being investigated to see if it would save weight and space on the power conversion skid. The addition of an auxiliary 400 cycle generator, operating off the same gear box with the main 60 cycle generator, to supply 400 cycle power to the auxiliary motors and equipment may offer substantial weight savings.

##### 5. Mechanical Design (Task 56-500)

Sketches have been completed for a 1/8th scale model of the ML-1 structure and equipment, and the purchase order has been sent out to fabricators. The model will be used in equipment layout work.

The analytical solution has been completed for the temperature distribution at the insulation joint on the pipe between the recuperator and the reactor, and a numerical example is being computed. In addition, a preliminary design was completed for the charcoal filter in the gas clean-up system following the transfer compressor. The design includes a vessel to operate at 3000 psig, and a free connection, gas pre-cooler for the seal vent compressor.

Inquiries have gone to vendors to determine if satisfactory pressure reducing valves, relief valves, solenoid-operated shut-off valves, and fast operation control valves are available; and to determine which of the available gas transfer compressor machines would be optimum in terms of weight, reliability and operatableness.



6. Process Design (Task 56-900)

Final mechanical flow sheet drawings have been completed for the power, gas transfer, drying and lubrication circuits.

D. INSTRUMENTATION AND CONTROLS

1. Neutron Monitoring (Task 57-200)

Specifications for the transistorized nuclear instrumentation were submitted to 25 manufacturers. They were given 45 days to complete their bid quotations. The specifications include one spare channel for each range of operation. (A channel of instrumentation includes the neutron detector, inter-connecting cable, amplifiers, and safety chassis.)

A capacitance bridge type position indicator is being investigated for use as a fast rod position indicator, one that has a small amount of inertia in the receiving unit. The regulating rod has a relatively fast speed, and indicators with large moments of inertia would over-run and be out of calibration. In the use of the capacitance bridge type, the rod carriage would position a variable air gap capacitor which would be compared to a fixed capacitor in the bridge circuits. The deviation of the current across the bridge would reflect the position of the rod.

2. Health Physics (Task 57-300)

A tentative specification has been written for the RAMS health physics equipment needed for the power conversion skid. The final specifications will reflect the radiation calculation at the ion chamber locations.

The design criteria has been written for the health physics equipment for the Personnel Change Building and the Auxiliary Building.

Tentative specifications for the gas particulate count monitor (mentioned in the March Progress Report - IDO-28538) will not be finalized until the configuration of the pre-cooler is determined.

3. Control Analysis Instrumentation (Task 57-400)

No effort was expended on this task during April.

4. Dynamic Analysis (Task 57-500)

The use of an axial flow compressor, instead of a radial flow type, is being considered because of an expected increase in compressor efficiency. The constant speed lines are almost vertical on the map of an axial flow compressor. Runs made on the analog simulator

show that the control plan, as now conceived, would work well with the axial compressor without modifications.

5. Process Instrumentation (Task 57-600)

The process instrumentation block diagram was reviewed in a series of meetings of the instrumentation group. Each instrument destined for the control cab was reviewed in detail.

The following instruments were added to the existing list of process instruments: wind speed and direction indicators, barometric pressure indicator, outside air temperature indicator, shield water conductivity indicator and alarm, shield water temperature indicator and alarm, and the moderator surge tank level indicator and alarm. These changes are being added to the process instrumentation block diagram.

6. Power Control (Task 57-800)

A load bank, necessary for the final test of the power system, will have the following characteristics:

- a. Varying power factor: 0.3 lead to 0.8 lag
- b. Step changes of load (manual operation)
- c. Power load requirement: 0 to 500 kw
- d. Temperature range: 30 to 110°F
- e. Separate controls (remote from load)
- f. Portable

Manufacturers have been asked to supply information and quotations for this equipment.

It is estimated that 225 wires will be required for the signal leads, motor controls, solenoid valve controls, temperature indicators, etc.

A motor controls schematic is now in rough draft form. This is a simplified schematic without provision for sequence of operation, interlocking of necessary equipment, or control of equipment during emergency shutdown. These areas are still tentative and will not be firm until late in May. This information will be integrated into the schematic as soon as possible.

The front panel configuration, for mounting processing equipment in the instrument cab, will be standard 8-3/4 in. high by 19 in. wide with Western Electric mounting dimensions. The use of mechanical slides with this equipment, for quick disconnect and ease of maintenance, will be studied.

## E. REACTOR ENGINEERING

1. Control Rods (Task 58-100)

No effort was expended on this task during April.

2. Reactor Shielding (Task 58-200)

Preliminary design is nearly complete on a reactor with a total package weight of 18 tons. The preliminary design is shown in Figure 19. The general nuclear and heat transfer calculations are complete, and ML-1 Notes Numbers 86 and 89 summarize the results. The design of individual components will be reviewed as the designs progress.

A corrosion and environmental testing program has been started to evaluate the design and indicate any possible problem areas. The detailed design drawings for the shielding are expected to be complete by 1 August. The final design package of individual components will be issued for bid as they are prepared.

The present ML-1 design criteria requires an 18-ton package weight, and a minimum exclusion area of 500 feet radius during full-power operation. The dose will not exceed 5 mr/hr at the perimeter of the exclusion radius during full-power operation. Twenty-four hours after shutdown, the dose will not exceed 5 mr/hr at the driver's compartment of the prime mover. (Note: There may be considerable error in the dosage values given here because of the uncertainties involved in the design of shielding for mobile systems. All dosage factors given here are expected to be correct within a factor of three.)

The conceptual design presented here is believed to meet the above requirements. A reduced package weight will require more costly materials and/or an increase in shutdown shielding dose over that prescribed above.

For a power reactor with completely self-contained shielding, both shutdown and full-power shielding must be considered. Use of a high density material, designed for gamma shielding of the shutdown reactor, will minimize shutdown weight at the expense of an excessively large exclusion area during full-power operation.

The shutdown shielding is preferential in direction to minimize weight (i.e., the dose expected in the cab of the prime mover will be lower than at other points equally distant from the reactor). The preferential nature of the shielding is true only after reactor shutdown, and the radiation during operation will be nearly isotropic.

The water in the shield tank will be drained before the reactor is transported. Filling the tank with water will reduce the dose by a factor of about 10. (Maintenance, as far as possible, should take place while the shield tank is full of water.) The dose rate for shutdown will be doubled if the moderator water is drained. After shutdown, the radiation dose will consist almost entirely of gamma photons.

The dose received during full-power operation will consist of both fast neutrons and gamma photons; thermal neutron fluxes are expected to be negligible. The dominant source of gamma during operation will be  $(n,\gamma)$  capture outside the high density shield. Escape of fast neutrons from the shielding will be by streaming down the gas ducts.

The first component of the radial shielding is the lead fast neutron reflector. The inner face of the lead is insulated to reduce heat transfer between the reflector and core. Water filled tubes imbedded in the reflector will remove the heat from the reflector. Calculations show that boiling can be tolerated in the reflector coolant passages. Lead thick enough to shield the shutdown radiation will make up the remainder of the high density shielding.

A sheet of boral is placed outside the reflector to reduce the thermal neutron flux in the outer portion of the shielding to the point where  $(n,\gamma)$  capture will no longer be a significant contribution to the gamma dosage.

A thick blanket of boron salts (probably ammonium pentaborate) will surround the lead shield. The high percentage of boron in the salt insures that  $(n,\gamma)$  captures in the outer portion of the shield will be suppressed enough to make a 500 ft exclusion radius adequate.

The necessity of locating the gas ducts at the ends of the cylindrical core means that more radiation will escape from the shielding axially than radially. In addition to the high density components, it is necessary to extend the boron salt blanket over both the upper and lower plenums and portions of the gas ducting to increase the axial shielding. Since boron salts form an excellent insulator, a means must be provided for removal of the heat conducted to the plenums and ducts from the coolant gas. Thin annuli will be provided between the ducting and the boron salt blanket to permit convective cooling of the hot surfaces by the shield water. The water in these annuli, however, will be the source of undesirable  $(n,\gamma)$  reactions so thin lead will be buried in the blanket (see Figure 19).

Some activation of the tungsten plates is expected because of the large number of resonance captures in tungsten of neutrons in the 25 ev range. Fortunately the energy of the emitted photons is less than 0.7 mev and the stainless steel in the plenums will be an adequate shield.

The reactor shield tank will be filled with highly borated water during operation and, when possible, during maintenance. This water will attenuate the fast neutron flux, and most of the thermal neutron capture in the tank water will be by boron nuclei. Any reactions, therefore, will produce little capture radiation, and an additional effect of the boron will be to reduce activation of the components outside the core. Ammonium pentaborate will be used to provide high boron concentration in the shield tank. (Tank corrosion problems will be investigated.)

### 3. Reactor Facilities (Task 58-300)

Investigation shows that it will be necessary to control the temperature of the shield water. Conduction losses by the plenums and gas ducts to the shield water make it necessary to cool the shield water to prevent boiling when the ambient temperature reaches 125°F. Furthermore, the shield water must be held at temperatures above 110°F to maintain 18 wt % ammonium pentaborate in solution.

A second heat exchanger, included in the moderator return system, will accomplish this. Since the temperature of the moderator will be controlled at about 170°F, as it passes through the secondary heat exchanger it will heat the shield water in cold weather and cool it in hot weather. This should maintain shield water temperature between 150°F and 190°F.

The reactor tank will have approximately 1/2 in. thickness of permanent insulation to reduce the heat loss during cold weather. This insulation will stabilize shield water temperatures and permit maintenance during cold weather without draining the shield water.

Detail stress analysis and design has started on the tank, and the final drawings of the tank are scheduled to be started in May. The preliminary design is complete for the heat exchangers. Final design awaits the detailed evaluation of the heat loads. The final bid requests for heat exchangers and for the tank are scheduled to be sent out in August and September, respectively.

### 4. Pressure Vessel (Task 58-400)

A layout drawing was started for the reactor pressure vessel and its internal insulation. Drawings and advance quotation requests were prepared for the large forged pieces of stainless steel required in the pressure vessel. These forged pieces include one-piece forgings for the tube sheet and side walls, and forged flanges for the upper and lower covers.

Specifications were completed and advance quotation requests sent to potential vendors for the fabrication of the pressure tubes. These tubes are 1.756 in. ID with 0.020 in. walls and will be made of AISI type 316 stainless steel.

Specifications and drawings were completed for the upper flanged connection of the pressure vessel. These connections are designed to be disconnected by remote controls, and a "V" band type coupling will be used for this joint to permit access to the fuel elements.

A model was fabricated of the fuel element support and "hold down" to determine its workability. Minor modifications are being incorporated on the mock-up for further evaluation.

Sample coupons are being fabricated to simulate the tube-to-tube-sheet joint. The three types of joints being considered are a rolled and welded joint similar to the GCRE-I pressure vessel joint; a brazed joint, not rolled; and a rolled and brazed joint. Brazing is considered because of the possibility that welding on the stainless steel tube-sheets will cause intolerable warpage.

The design criteria have been determined for final design of a 1/4 scale model to simulate the gas flow in the pressure vessel plenums. The model will be used to determine optimum shape of the plenum and the tungsten baffle for minimum pressure losses, and to determine the actual pressure drop in the pressure vessel.

The heat transfer characteristics of the stainless steel sheets used as radiation insulation was analyzed. The analysis showed that the minimum effective distance between barriers was 0.010 in. Furthermore, the efficiency of the insulation increases with the number of barriers, but beyond three barriers there was little gain in thermal efficiency compared to the increase in fabrication complexity. The calculated heat transfer through the insulation was 5200 BTU/hr/ft<sup>2</sup> for one radiation barrier; 2400 BTU/hr/ft<sup>2</sup> for two barriers, and 2000 BTU/hr/ft<sup>2</sup> for three barriers.

#### 5. Control Rod Blades (Task 58-500)

The three types of blade mounts mentioned in previous reports are being fabricated. The facility for testing these mounts was described in the March Progress Report (IDO-28538). Equipment is being fabricated to test the two gear-driven control blade mounts. The design of equipment to test the mount in which the blades are connected by links and actuated by a reciprocating shaft will be completed in May. These tests will begin in June.



#### IV. COORDINATION OF THE GAS TURBINE TEST FACILITY

Under a subcontract from AGC, AGN has assumed responsibility for overall direction of the GTTF work scope. Authorization for a \$275,000 spare parts contract is expected in May. Aerojet has requested bids for the spare turbine-compressor set to expedite schedule requirements.

ERDL and Aerojet personnel visited Stratos and its subcontractor, Textile Manufacturing Co., to establish a realistic delivery date for the turbine-compressor set. Delivery will be made on 12 June 1959 providing Textile delivers the turbine-compressor set casings by 29 April. Any change in this schedule requires the approval of the ERDL project engineer. (It has been learned since the conference that casing warping problems will delay the delivery date by a week).

The following problem areas were investigated during April:

1. The first harmonic (natural frequency) of the gear box quill shaft was found by the graphical static deflection method. Work continues, using changed end conditions, and includes exploring the feasibility of using a solid shaft.

2. The coupling between the turbine-compressor set and the gear box interferes when the turbine-compressor set face seal is removed. After study of the problem, it appears that redesign of the seal and selection of the proper coupling will solve the problem.

3. Aerojet is setting up a computer program to furnish rpm/time and system/pressure functions when an impingement nozzle is used on the first stage of the compressor.

4. Aerojet is studying the GTTF piping to determine the modifications necessary for cycle tests on the turbine-compressor set. ERDL will be asked to authorize procurement and installation of necessary materials for these modifications.





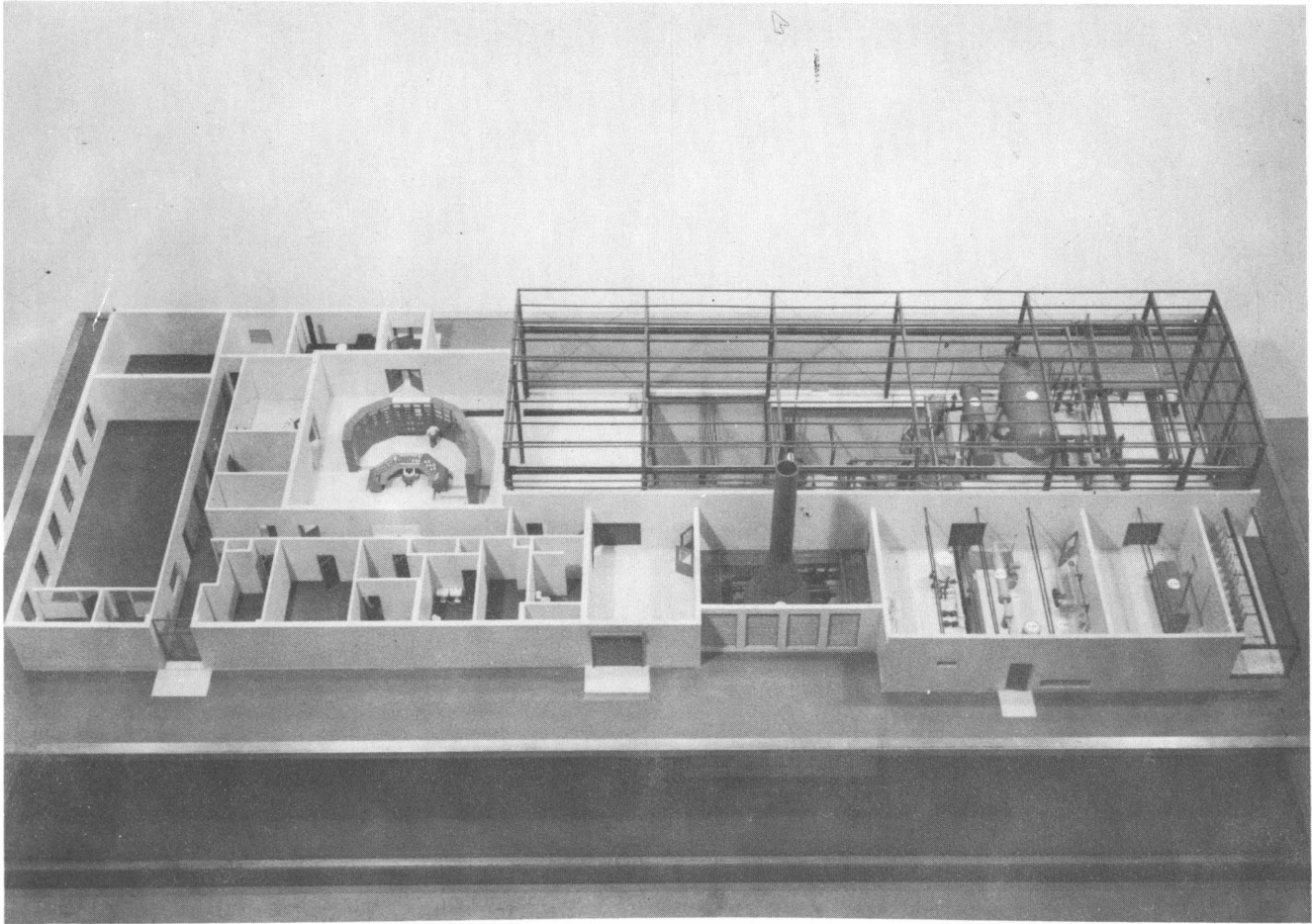


FIGURE 1. FRONT VIEW OF THE FACILITY

The model of the GCRE facility at the National Reactor Testing Station, Idaho Falls, Idaho, is shown as it would appear to an observer looking northward at the front of the building. The door on the left is the main personnel entrance. The 1/48th scale model has sufficient detail to give a realistic display of many of the mechanical aspects of the actual facility.



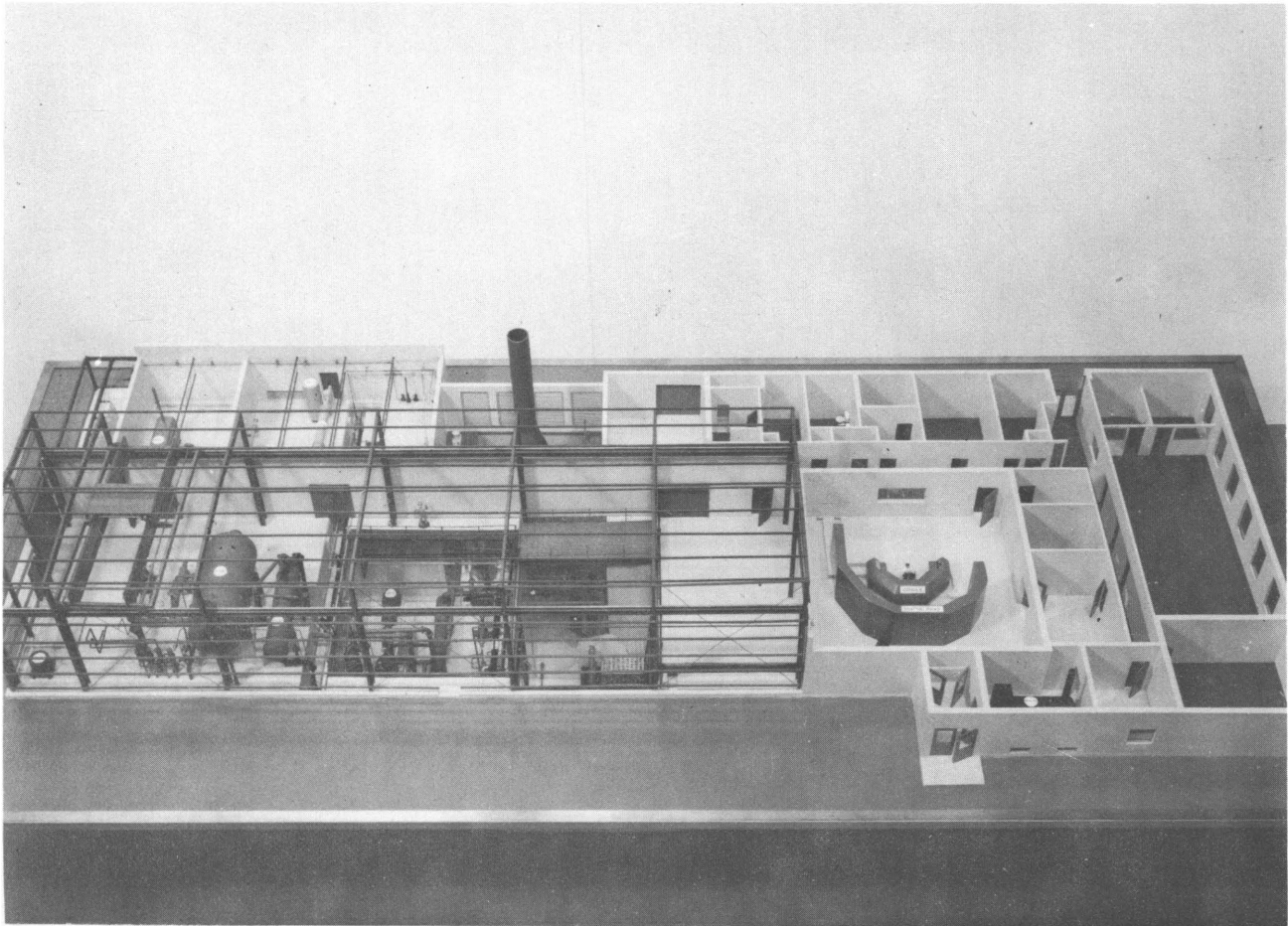


FIGURE 2. THE FACILITY FROM THE REAR

The north side of the building is shown here. Features of this model include a fitted packing case, and four legs that screw into the bottom of the model so it forms its own display table. The packing case is carefully fitted to the model to provide maximum protection in transit and in storage.



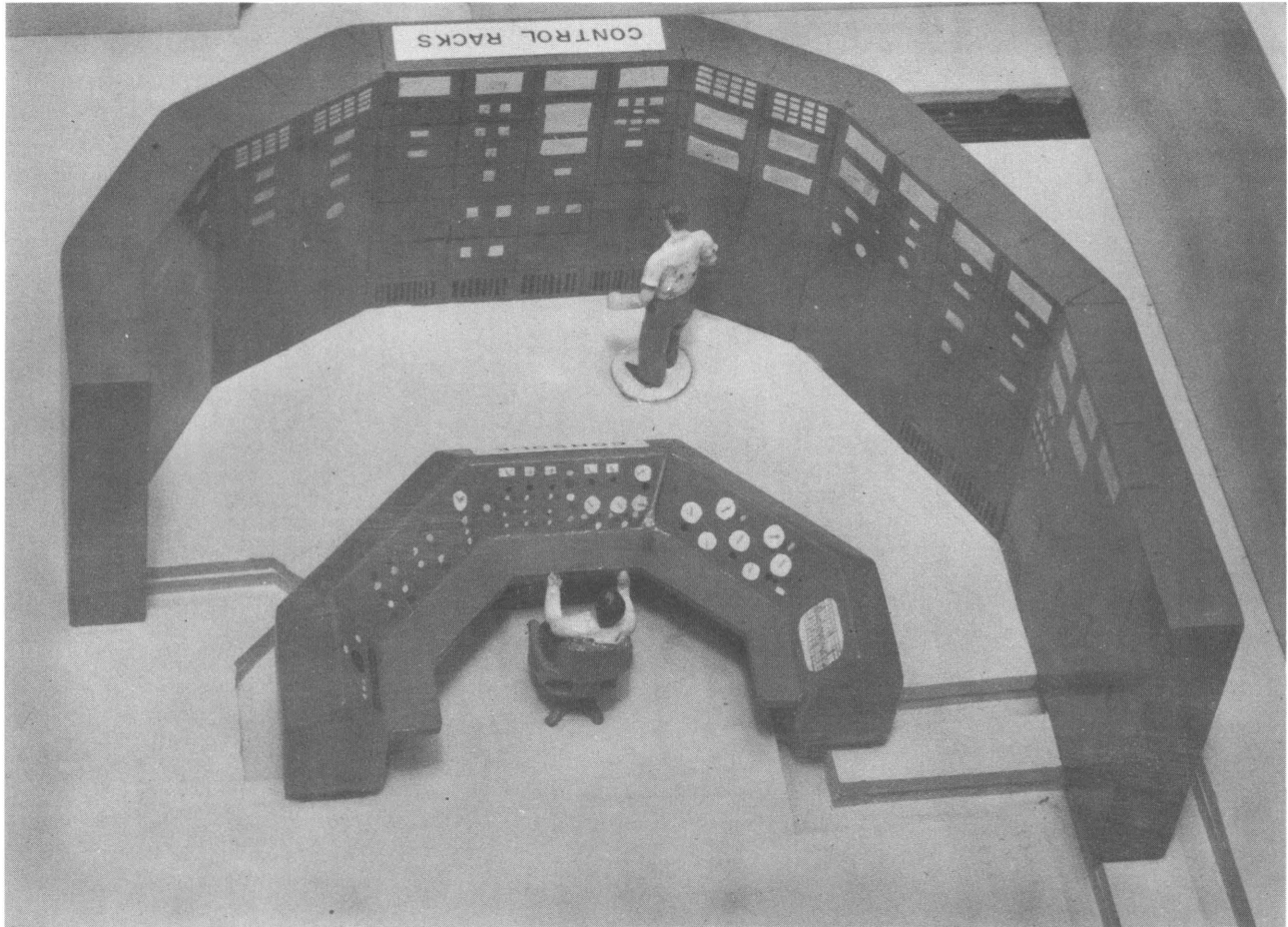
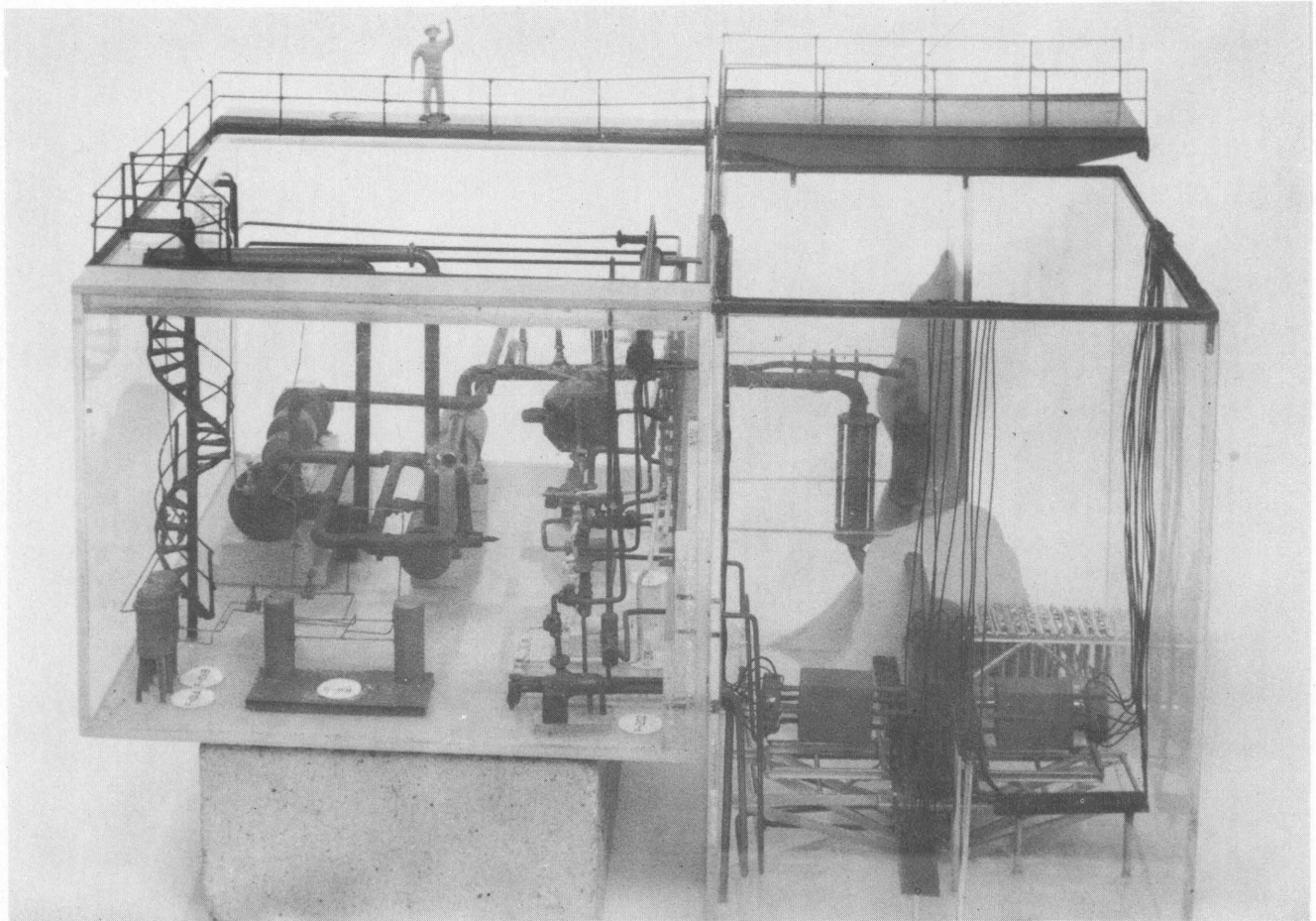


FIGURE 3. THE CONTROL ROOM

The control console and instrument racks are seen in more detail here. The figures of the operator and the technician provide a visual scale of the size of the facility.



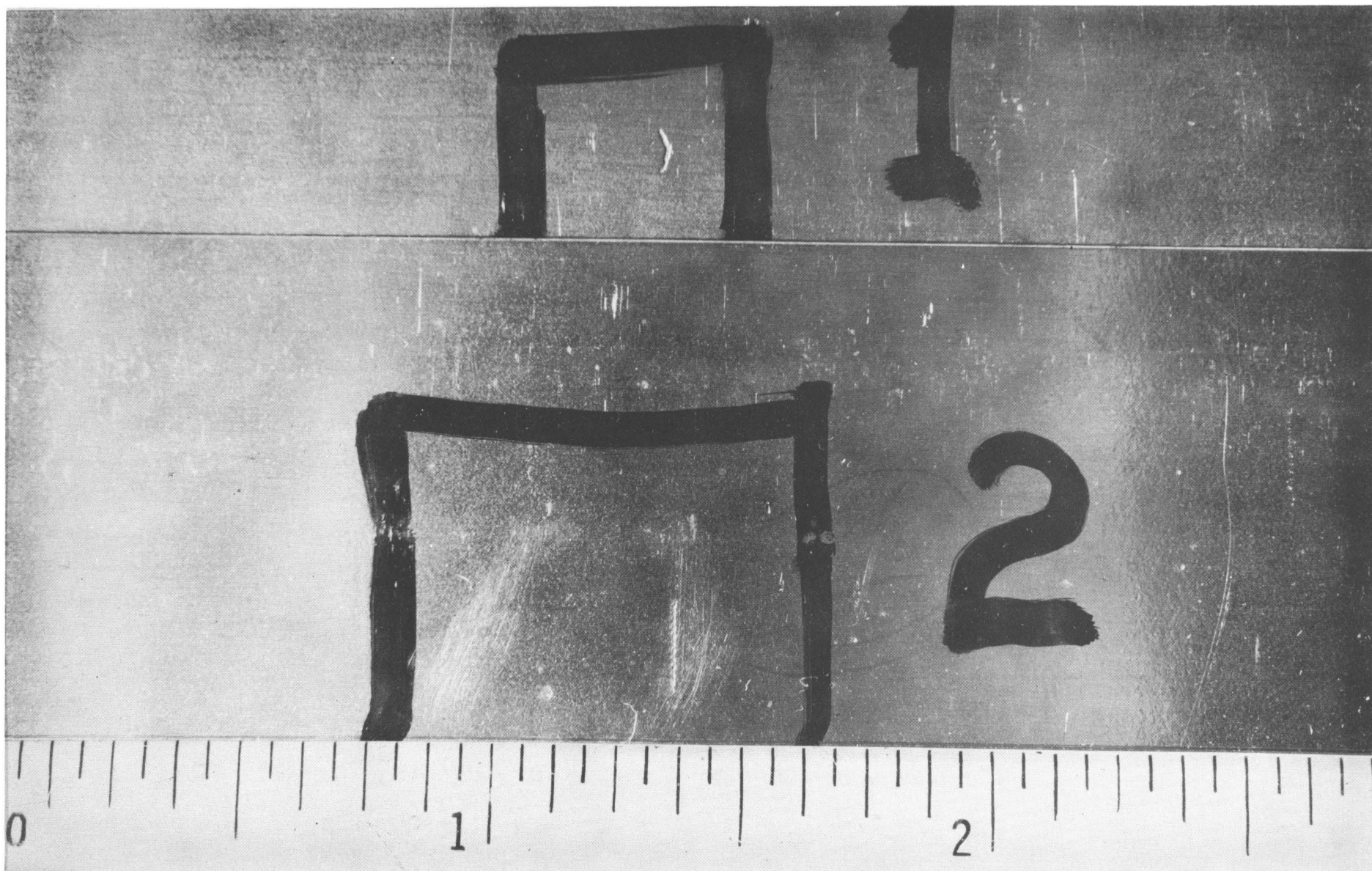


**FIGURE 4. THE EQUIPMENT PIT AND THE REACTOR PIT**

The reactor pit and mechanical equipment pit as viewed from above. The reactor pit, on the right, shows the reactor on its support structure and the depleted fuel element storage rack above it. The reactor pit will be flooded during operations.

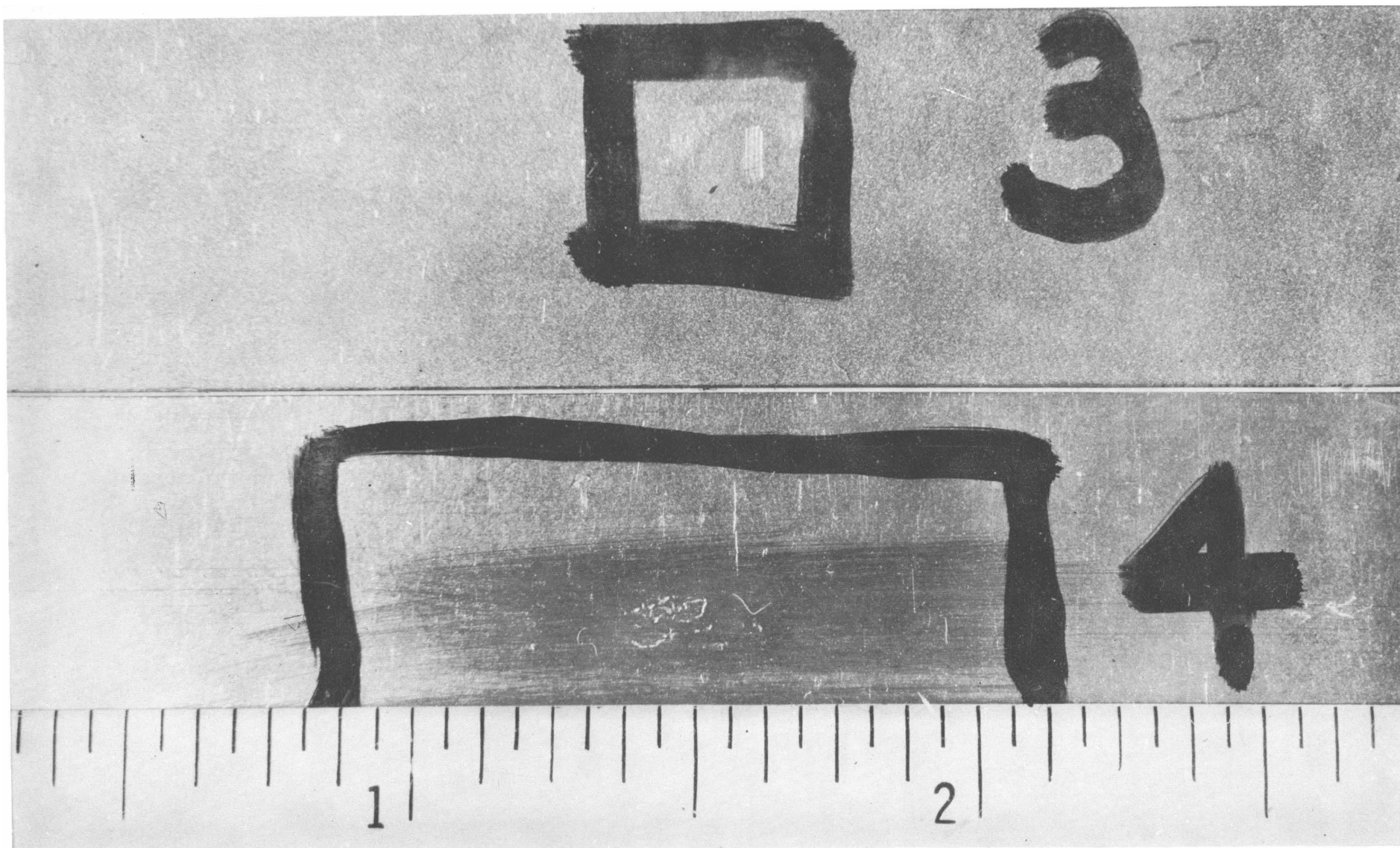






PHOTOS 1 AND 2 - TYPICAL NICKS AND DENTS IN AS  
RECEIVED PLATES.

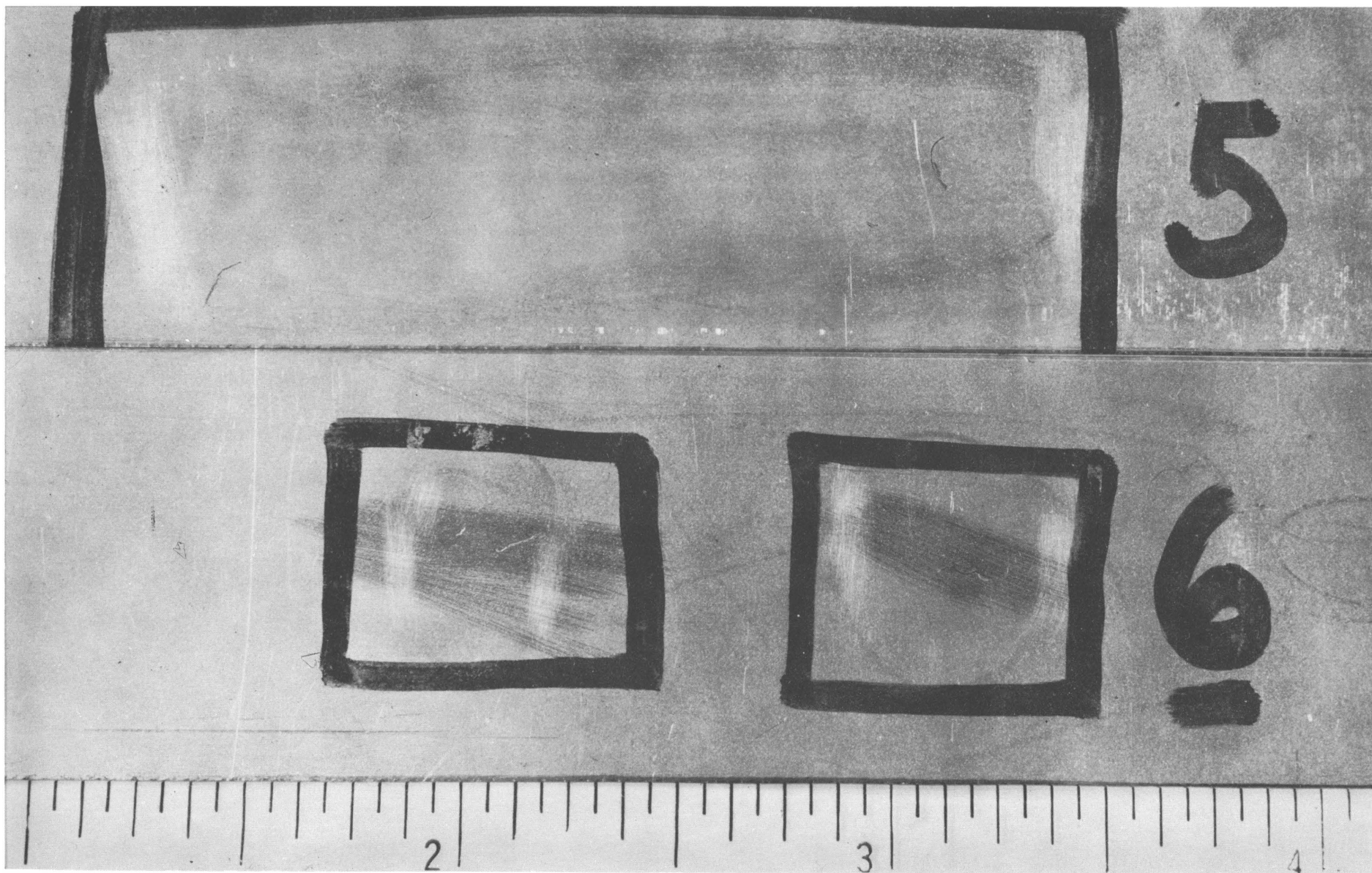




PHOTOS 3 AND 4 - SURFACE DEFECTS FROM IMPROPER HANDLING AND ROLLED IN MATERIAL.







PHOTOS 5 AND 6 - DEFECTS APPEAR TO BE CAUSED BY STEEL WOOL PARTICLES ROLLED INTO SURFACE.



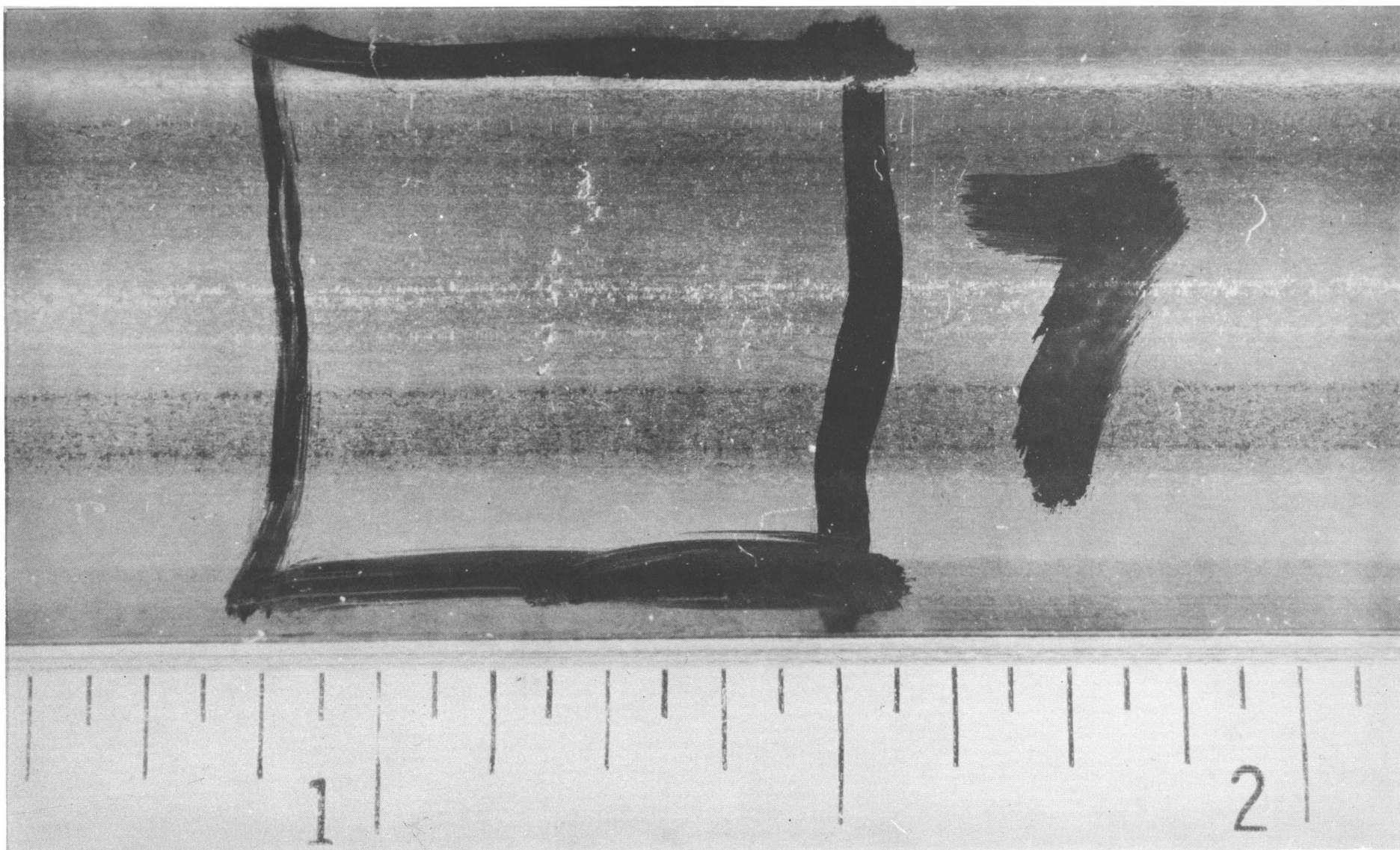
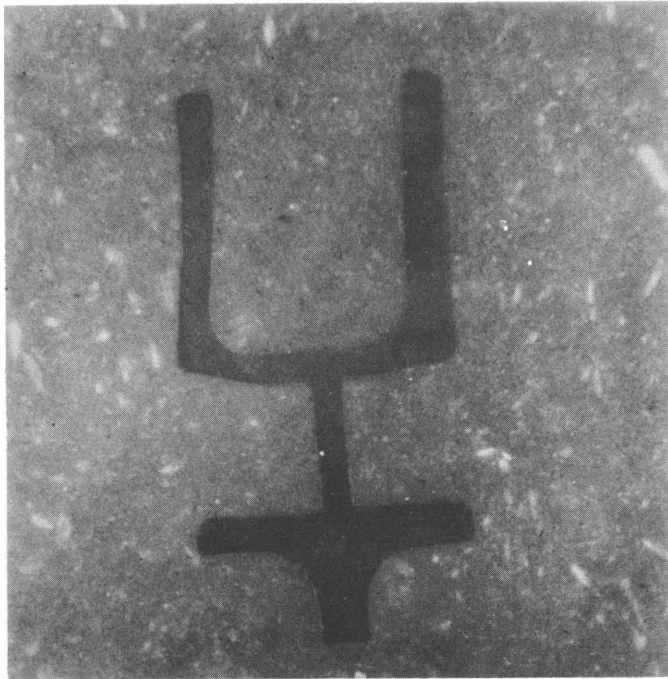


PHOTO 7- DEEP SURFACE DEFECTS ROLLED INTO SURFACE.  
THIS PLATE DID NOT PASS ACID ETCH.

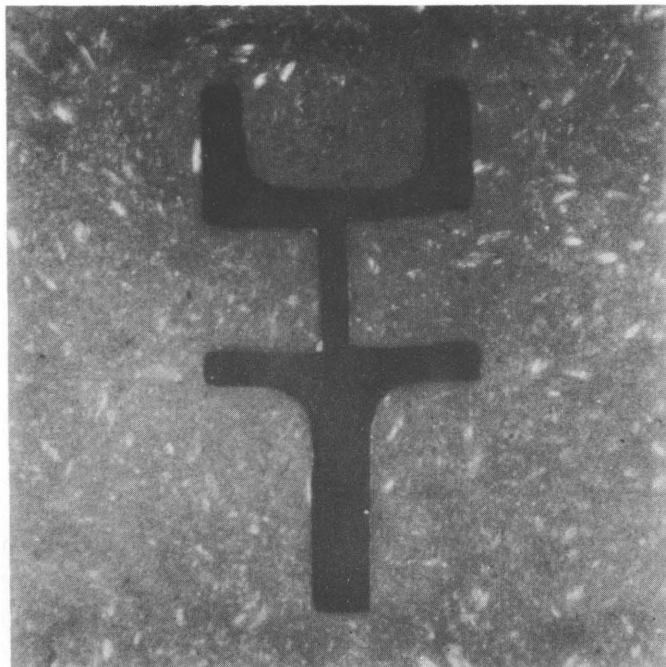




## *TYPICAL CLIP DEFECTS*

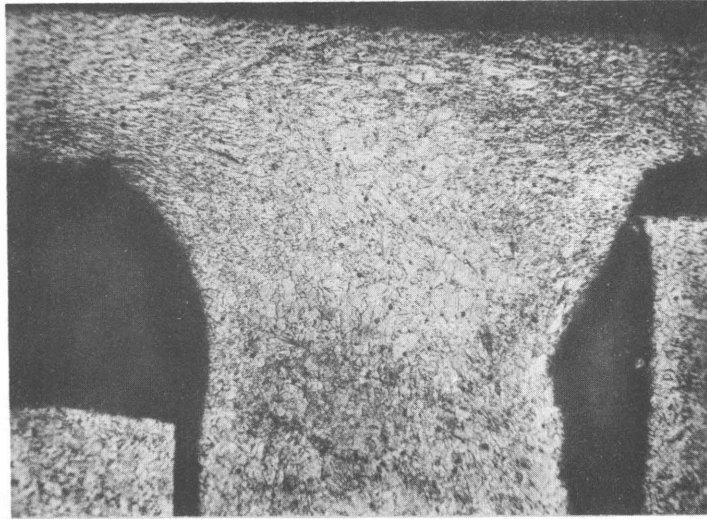


Typical section of B clip as received  
showing misplaced angles and mismatch  
of notches. (10X)

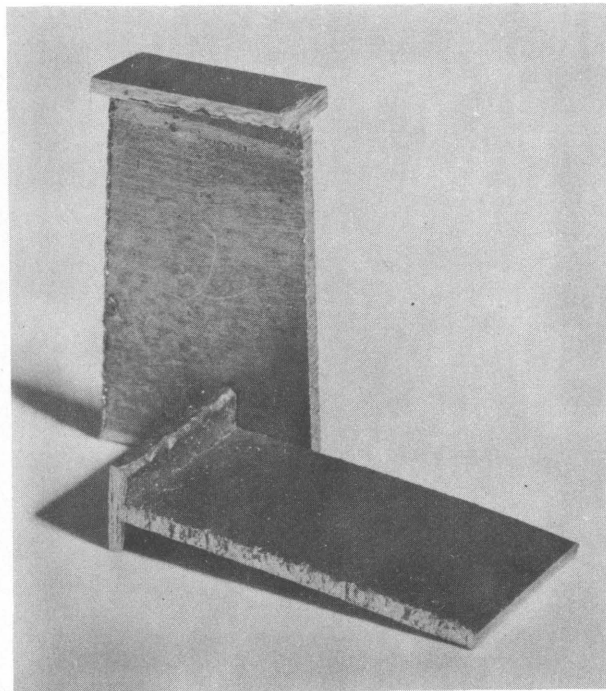


Typical section of C Clip as received  
showing misplaced angles and mismatch  
of notches. (10X)



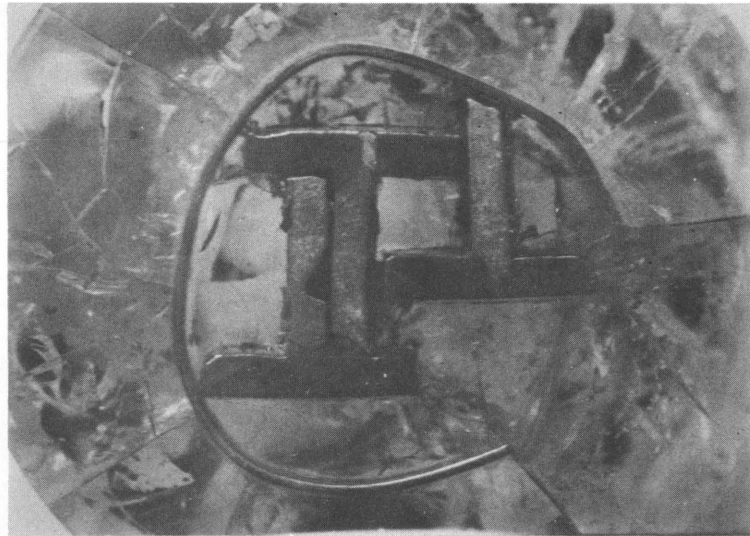


**FIGURE 10.** Microphotograph of a forged specimen of "T" section.  
Cupric sulfate etchant . About 40 x.

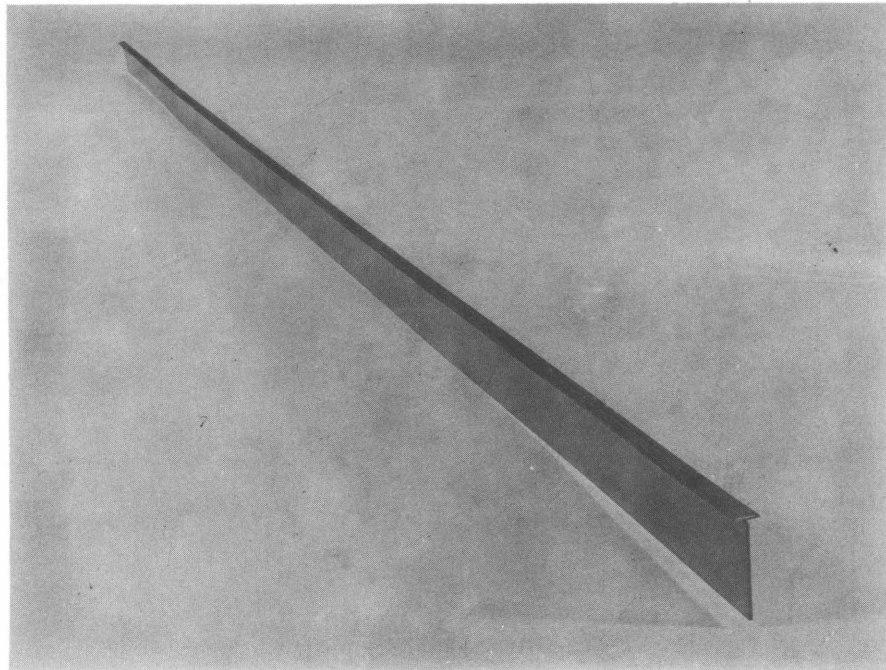


**FIGURE 11.** Specimen after machining.



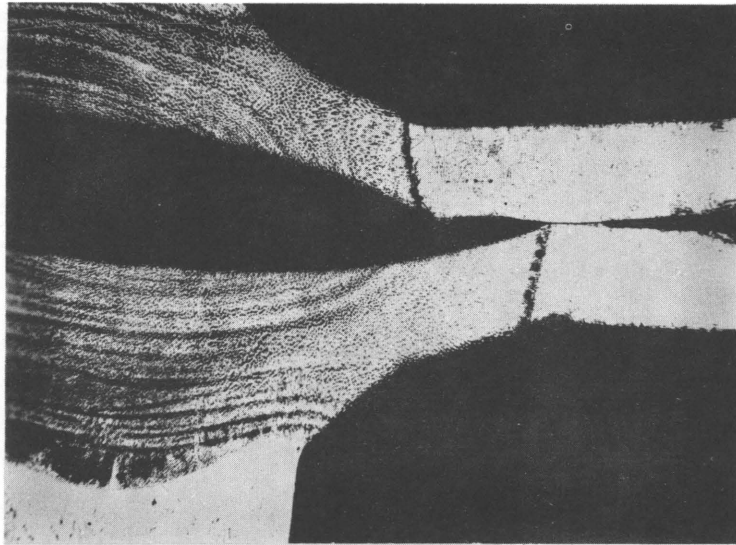


**FIGURE 12. Puddle welded and machined specimen – 4.5 X.**

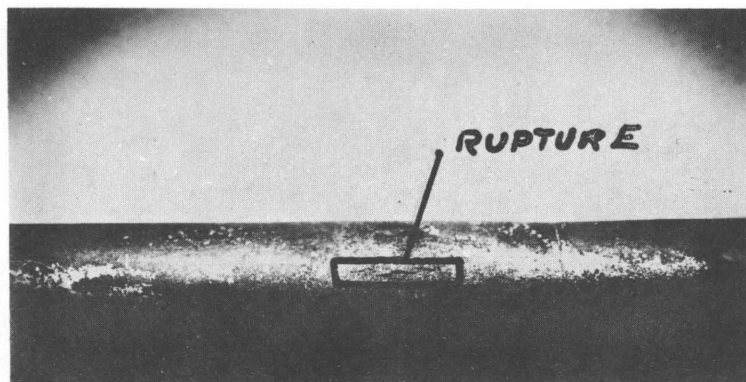


**FIGURE 13. Full length piece of flat steel with one welded spacer.**





**FIGURE 14.** Microphotograph of the weld area. About 60x.



**FIGURE 15.** Ruptured section of Inconel 702 tube — 3½ x





*SCHEMATIC FLOW DIAGRAM  
THERMAL CYCLING TEST SYSTEM FOR GCRE II  
FUEL ELEMENT DEVELOPMENT*

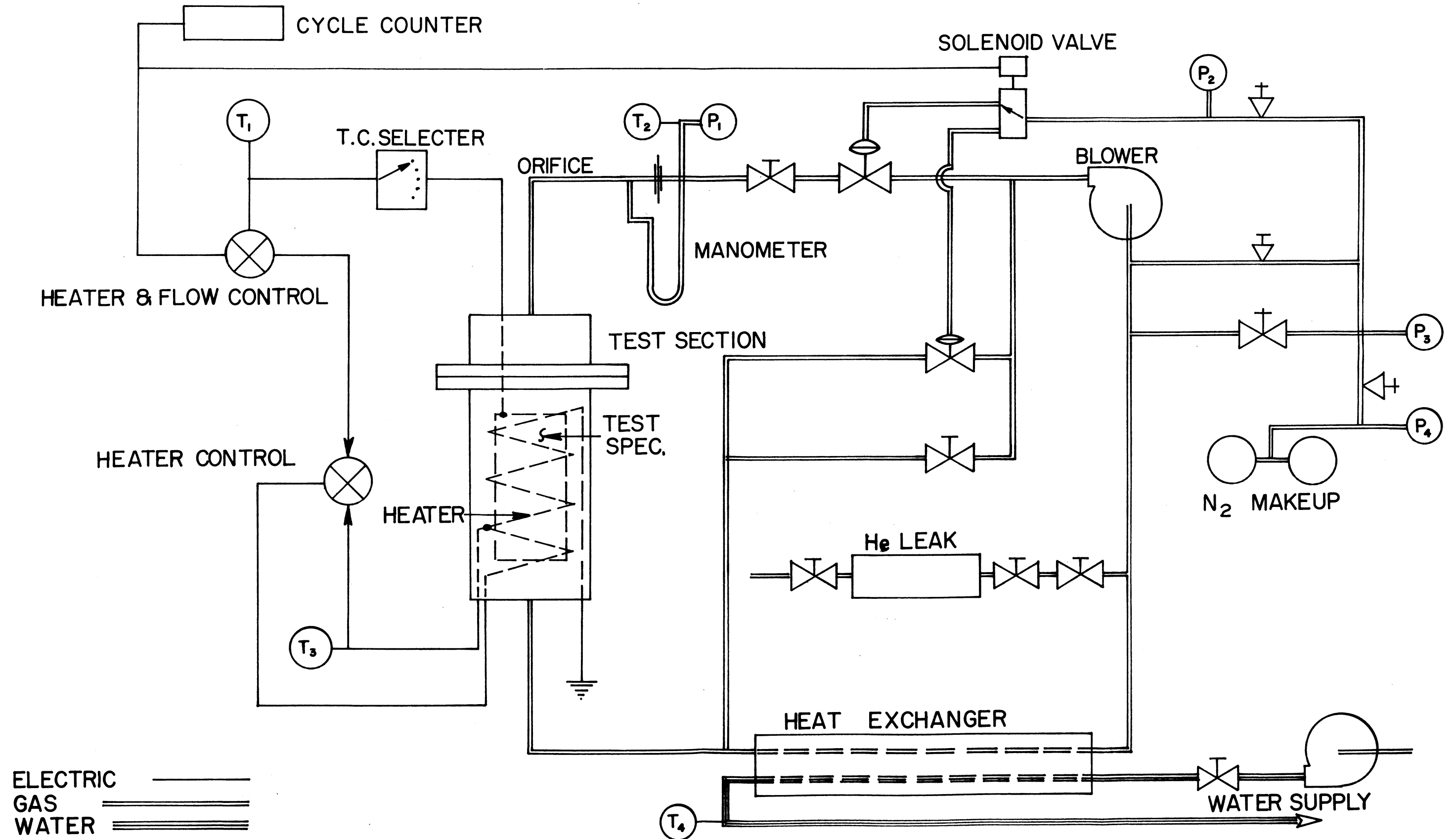
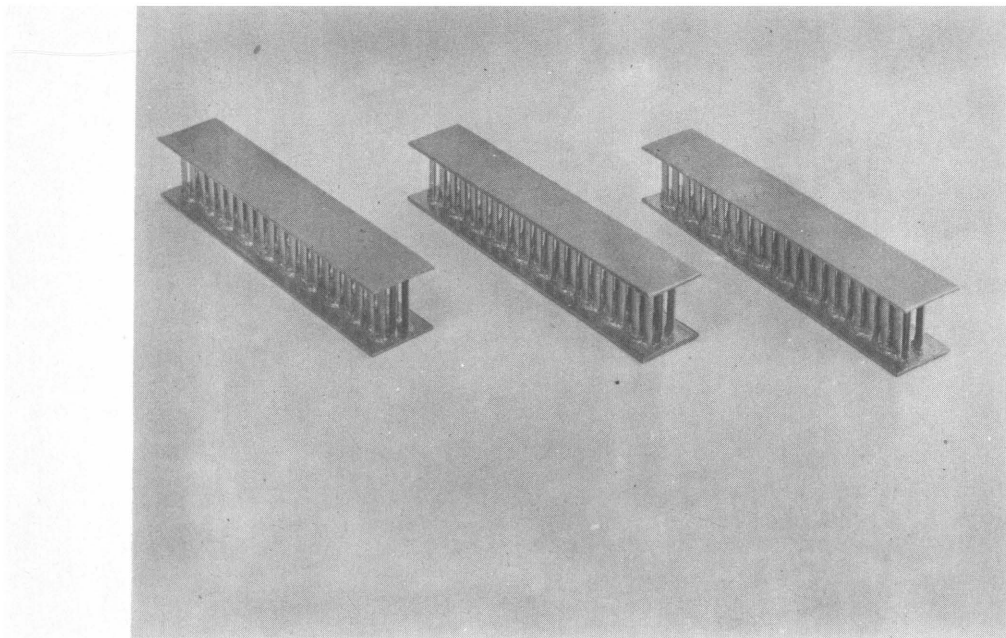
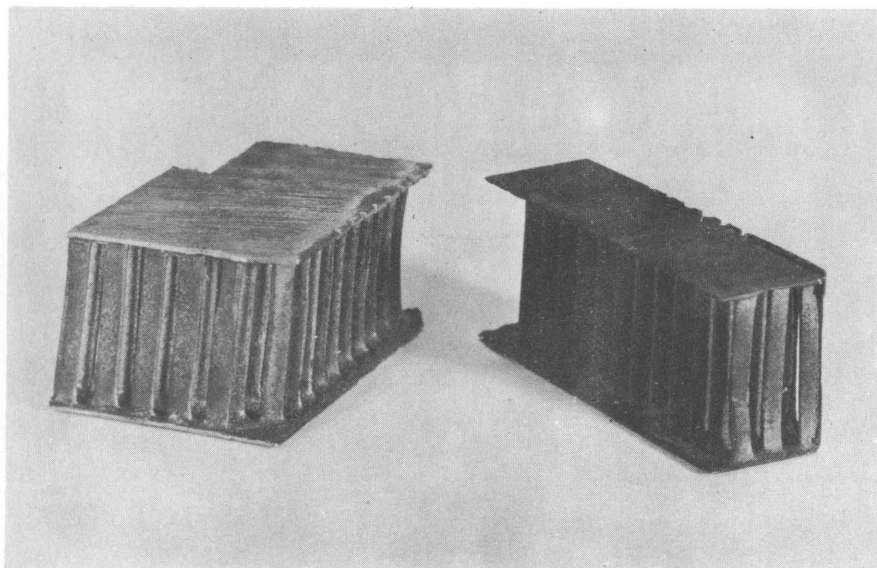


FIG. 16





**FIGURE 17. Typical sample of pin fin structures as received.**



**FIGURE 18. Exposed and unexposed pin fin samples showing results of 2000 hr gas corrosion test.**



# PRELIMINARY DRAWING OF ML-1 SHIELDING

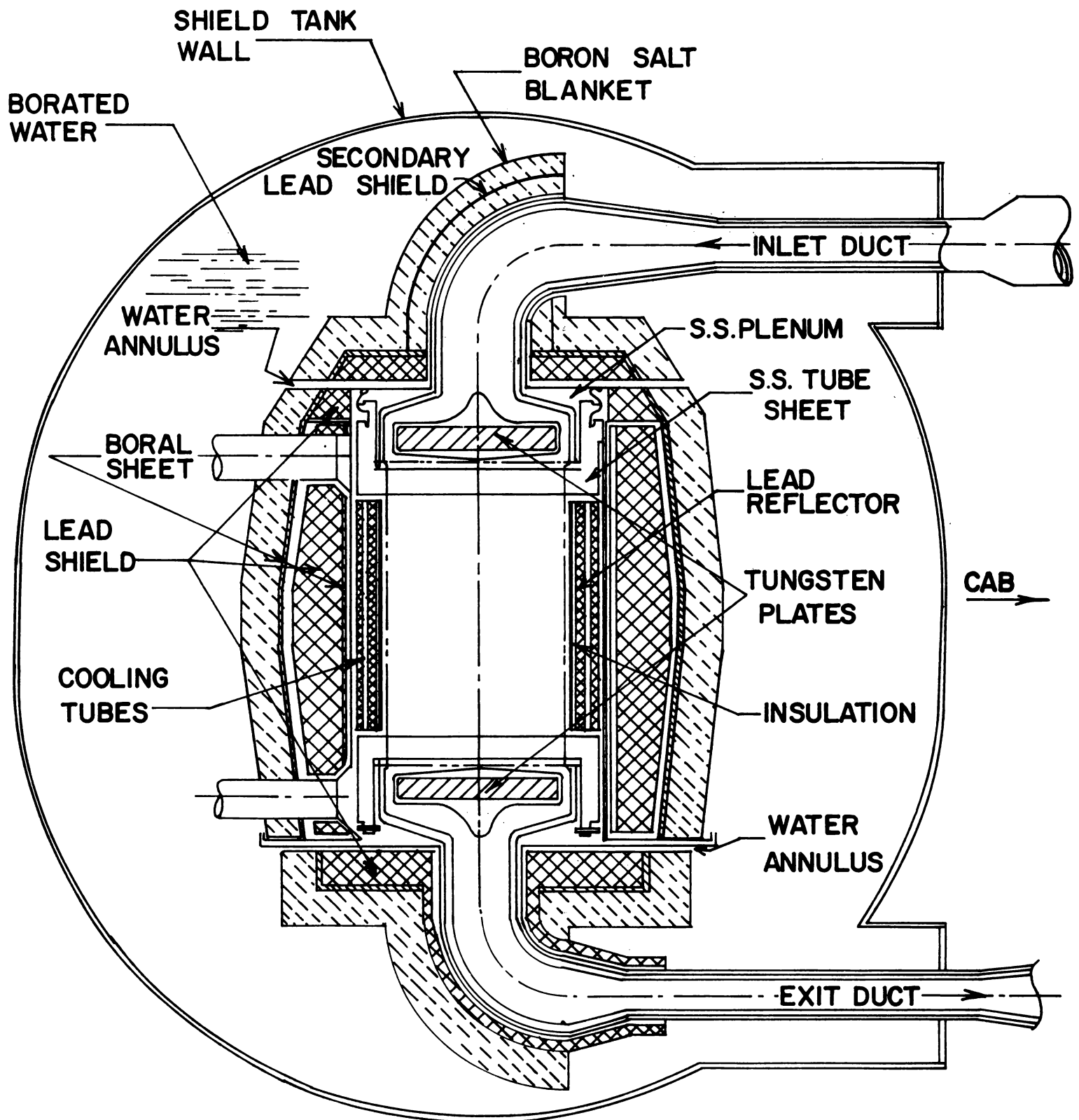


FIG. 19



TABLE 1 - PRODUCTION YIELD OF M & C PLATES

<u>Plate Size</u>	<u>Lot Nos.</u>	<u>Total No. Plates Started</u>	<u>Hot Area Rejects</u>	<u>Plate Finishing Rejects</u>	<u>Yield</u>	<u>Net Yield</u> (Corrected for Destructive Test)	<u>% Yield</u>	
B	B-2 thru B-9	186	15	44	127	117	66.5%	117/176
B <sup>(1)</sup>	B-10 thru B-17	192	6	32	154	145	79.3%	145/183
C <sup>(1)</sup>	C-2 thru C-12	329	17	60	252	242	75.5%	242/319
D <sup>(1) (2)</sup>	D-3 thru D-5	132	2	2	128	124	97.0%	124/128

Footnotes

(1) More plates are in process. Final yield data are unavailable as yet.

(2) Size "D" is normally assumed to be the most difficult to make!





TABLE 2 - DESCRIPTION OF HOT AREA REJECTS

<u>Lot Numbers</u>	<u>Total No. Rejects</u>	<u>Compact Defects</u>			<u>Rejects After Hot Rolling</u>			
		<u>Erroneous Weight</u>	<u>Cracks &amp; Defects</u>	<u>Non- Homogeneous</u>	<u>Cracks</u>	<u>Laps &amp; Seams</u>	<u>Blisters</u>	<u>Surface Pits</u>
B-2 thru B-9	15	3	2	1	1	2	6	-
B-10 thru B-17	6	2	-	-	-	-	1	3
C-2 thru C-12	17	5	6	1	1	-	4	-
D-3 thru D-5	2	1	-	1	-	-	-	-



TABLE 3 - DESCRIPTION OF PLATE FINISHING REJECTS

<u>Lot Numbers</u>	<u>Number of Rejects</u>	<u>Dimensional Defects</u>			<u>Layout and Machining Defects</u>			<u>Surface Defects</u>	<u>UO<sub>2</sub> Leak (in HNO<sub>3</sub>)</u>
		<u>Meat Width</u>	<u>Meat Length</u>	<u>Excessive Camber (After Decambering)</u>	<u>Inert (1) Margin Width</u>	<u>End (2) Margins</u>	<u>Outside Dimensions of Plate</u>		
B-2 thru B-9	44	2	-	-	5	6 <sup>(4)</sup>	2	29 <sup>(3)</sup>	-
B-10 thru B-17	32	12	3	-	3	-	3	11	-
C-2 thru C-12	60	2	-	1	44	-	4	8	1
D-3 thru D-5	2	1							

(1) also called "side closure"

(2) also called "end closure"

(3) 20 of the 29 were caused by a non-recurring type of human error

(4) most of the 6 may be recovered though not within tolerance



TABLE 4

PMU-1

GAP MEASUREMENTS AFTER RAPID THERMAL CYCLE

<u>Can to A Gap</u> (Specified gap $.054 \pm 0.004$ )			
<u>Top</u>	<u>As Assembled</u>		<u>Bottom</u>
.054	.054	.053	.052
.053	.053	.053	.052
.054	.054	.054	.054
.050	.050	.052	.052
.052	.052	.053	.053
.050	.050	.050	.050

After 1, 4 and 20 cycles unable to obtain  
measurements cannot use probe without disassembly

<u>A to B Gap</u> (Specified gap $.097 \pm 0.004$ )			
<u>Top</u>	<u>As Assembled</u>		<u>Bottom</u>
.099	.098	.098	.097
.096	.094	.095	.095
.098	.098	.098	.098
.097	.095	.095	.095
.096	.095	.095	.095
.099	.096	.096	.096

<u>After 1 Cycle</u>			
.098	.098	.098	.097
.097	.095	.096	.096
.096	.093	.095	.099
.098	.096	.096	.093
.100	.098	.098	.098

<u>After 4 Cycles</u>			
.096	.095	.095	.103
.097	.097	.098	.101
.103	.099	.100	.105
.100	.099	.099	.100

<u>After 20 Cycles</u>			
.098	.093	.094	.104
.102	.100	.098	.099
.099	.097	.096	.099



TABLE 4 (Cont'd.)

<u>B to C Gap</u> (Specified gap $.044 \pm 0.004$ )			
<u>Top</u>	<u>As Assembled</u>		<u>Bottom</u>
.045	.045	.045	.044
.044	.045	.045	.045
.045	.042	.043	.043
<u>After 1 Cycle</u>			
.046	.044	.045	.045
.042	.044	.048	.049
.044	.042	.043	.042
<u>After 4 Cycles</u>			
.046	.049	.051	.051
-	.046	.046	.046
	.044	.044	.045
<u>After 20 Cycles</u>			
.043	.044	.043	.044
.046	.043	.046	.046
.049	.045	.043	.047
<u>C to D Gap</u> (Specified gap $.088 \pm 0.004$ )			
<u>Top</u>	<u>As Assembled</u>		<u>Bottom</u>
.088	.088	.088	.090
.088	.088	.088	.088
.088	.088	.090	.090
<u>After 1 Cycle</u>			
.089	.091	.096	.092
.089	.092	.092	.087
<u>After 4 Cycles</u>			
.092	.093	.096	.093
.093	.094	.094	.090
<u>After 20 Cycles</u>			
.089	.091	.093	.091
.089	.093	.093	.094
.094	.093	.093	.096





TABLE 5

PMU-5

GAP MEASUREMENTS AFTER RAPID THERMAL CYCLE

<u>Can to A Gap</u>		(Specified gap $.054 \pm 0.004$ )	
<u>Top</u>	<u>As Assembled</u>		<u>Bottom</u>
.054	.052	.052	.053
.050	.047	.050	.047
.047	.050	.049	.049

After 1, 4 and 20 cycles unable to obtain  
measurements cannot use probe without disassembly

<u>A to B Gap</u>		(Specified gap $.097 \pm .004$ )	
<u>Top</u>	<u>As Assembled</u>		<u>Bottom</u>
.105	.096	.095	.098
.107	.100	.102	.098
.106	.095	.098	.099
.100	.093	.095	.094
.109	.099	.104	.103
.110	.097	.100	.100

<u>After 1 Cycle</u>			
.100	.102	.097	.098
.097	.094	.096	.097
.097	.098	.101	.102

<u>After 4 Cycles</u>			
.097	.097	.099	.100
.098	.096	.100	.099
.100	.105	.102	.105
.097	.102	.098	.099
.097	.097	.098	.099

<u>After 20 Cycles</u>			
.096	.096	.099	.092
.101	.102	.101	.101
.101	.103	.101	.102
.098	.097	.097	.098
.097	.097	.100	.098
.099	.103	.100	.099



TABLE 5 (Cont'd.)

<u>B to C Gap</u>			(Specified gap $.044 \pm .004$ )
<u>Top</u>	<u>As Assembled</u>		<u>Bottom</u>
.050	.044	.045	.043
.045	.042	.043	.044
.041	.041	.043	.043
<u>After 1 Cycle</u>			
-	-	.045	.043
.045	.048	.048	.045
x	x	.048	.044
<u>After 4 Cycles</u>			
.050	.049	.046	.049
x	x	.043	.044
.046	.047	.046	.046
<u>After 20 Cycles</u>			
.046	.049	.049	.049
.047	.043	.044	.044
.048	.049	.049	.049
<u>C to D Gap</u>			(Specified gap $.088 \pm .004$ )
<u>Top</u>	<u>As Assembled</u>		<u>Bottom</u>
.103	.099	.096	.091
.101	.096	.095	.094
.099	.096	.094	.092
.099	.093	.092	.096
.096	.099	.094	.090
.100	.092	.092	.089
<u>After 1 Cycle</u>			
.093	.094	.096	.090
.094	.093	.095	.094
<u>After 4 Cycles</u>			
.090	.097	.099	.090
.090	.093	.093	.092
<u>After 20 Cycles</u>			
.092	.099	.095	.091
.089	.092	.094	.092







