BAW 1033

LIQUID METAL FUEL REACTOR EXPERIMENT

QUARTERLY TECHNICAL REPORT

MAY 1957 — SEPTEMBER 1957

THE BABCOCK & WILCOX COMPANY

ATOMIC ENERGY DIVISION

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MAY 1957 - SEPTEMBER 1957

AEC CONTRACT NO. AT(30-1) - 1940
B&W CONTRACT NO. AEJ-46

SUBMITTED TO THE
UNITED STATES ATOMIC ENERGY COMMISSION
BY
THE BABCOCK & WILCOX COMPANY

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I. ABSTRACT

The Test Program Planning Group was reorganized and renamed the Research and Development Coordination Group. The group will continue to handle all materials problems, materials inspections, and specification reviews; they will also assume the functions of the Test Program Planning Group until the need arises for the latter.

The Project Materials Handbook was revised, and sections on beryllium and helium were added.

A list of areas in which research is lagging or lacking was compiled and necessary research requests initiated.

Graphite was ordered from three vendors for outgassing, sump penetration, graphite-to-metal seal, alternate impregnating, and carbide formation tests.

Discussions with valve manufacturers indicate that fabrication of Croloy 2-1/4 bellows is the only real problem associated with production of the valves.

The uncertainty of $\pm$ 5 inches for the reference design critical diameter is expected to be reduced greatly as additional information on physical properties of actual reactor materials becomes available, and as the results of the LMFRE critical experiment and allied work are analyzed.

For the reference design, the contribution of inelastic scattering in bismuth to the total k_{eff} was calculated to be about 2 percent.

The positive xenon temperature coefficient of reactivity was recomputed, for the case of no xenon removal, and found to be in good agreement with previous calculations.

The U-235 addition necessary to override fission product poisoning was checked by raising thermal cut-off energy from 0.07 ev to 0.2 ev, increasing the required U-235 additions by only 10 ppm.

The relaxation length of thermal neutrons for the concrete surrounding the reactor was calculated to be approximately 8 cm.

A number of inadequacies were noted in the generalized set of core transient equations reported in the last quarterly report. These equations have been improved

to permit a final steady state solution, to improve treatment of the delayed neutron precurson concentration, and to introduce a radial temperature dependence in the graphite.

Based on a number of four-group two-region calculations to determine the range of sizes of LMFRE critical experiments, the experiment is being designed to accomodate cores up to 56 inches in diameter and reflector thicknesses up to 28 inches.

Graphite vendors' comments on specifications and preliminary working drawings for core-end reflector assembly and side reflector indicate that (1) core OD and side reflector ID tolerances should be relaxed slightly, and (2) the design does not provide access for cleaning loose scale from core fuel passages. To overcome the latter, an alternate top reflector arrangement, which includes a quadrant-type graphite tank and drilled block insert, was completed.

The core Bi/C volume ratio was revised to 0.5 (on a cell basis) and top and bottom reflectors were redesigned to give a Bi/C volume ratio of 0.2 (avg.) measured at the horizontal midplane.

Other modifications include:

1. Change in top and bottom reflector flow-channel diameter from 2 to 1.5 inches.
2. A saw-tooth type cemented joint recommended by graphite vendors.
3. A larger central-test-hole diameter to make the ΔT identical to an average flow channel ΔT.

A 1 1/2-in. core flow channel and a 2.3-in. square pitch were specified for the reference core.

Detailed engineering work on the reactor vessel and associated internals was completed.

Control rod thermal calculations have been initiated to determine the practicality of solid rods and required cooling rates.

Brookhaven National Laboratory was requested to determine weepage rates of bismuth through graphite and whether a threshold pressure actually exists below which certain graphites absorb no bismuth.

A report on thermal calculations for the reference design was issued.

Recalculated fission product poison buildup, assuming various amounts of U-Bi absorbed in graphite, indicates this effect to be unimportant in the LMFRE (for a 20 MW-year core life).

A new high-density (2.2) graphite, developed at the University of Michigan, does not appear advanced enough for LMFRE use, but may be useful in future LMFR's.

The required initial U-235 concentration for the reference design was estimated to be about 800 ppm by weight. The Physics Section reported this concentration will require an active core diameter between 45 and 47 inches.

A. D. Little, Inc. began a study to develop a continuous uranium monitor for the LMFRE.

Major preliminary design modifications affecting Systems Engineering are:

1. The standby pumps were removed from core and reflector primary cooling systems to simplify piping.
2. A side-stream degasser handling 10 percent of the primary system flow replaced the full-stream degasser.
3. More accurate information on xenon solubility in bismuth justified a drastic reduction in the off-gas system carrier-gas flow rate.
4. The number of cells required to contain the plant was reduced.
5. Concrete between cells was eliminated.
6. The water-cooled steam condenser was replaced by an air-cooled type to provide more flexibility in site selection.

Investigation of primary system dump time requirements indicate that a rapid dump is not required for nuclear reasons.

A preliminary investigation indicates overpressure in the reactor plant is 7 psi following the maximum credible accident.

A request was initiated for research on the corrosion of lead-magnesium alloy, an alternate intermediate fluid, which might replace sodium if current research indicates the latter cannot be used.

Since component arrangement depends largely on the remote maintenance system used, plant arrangement will fluctuate until a basic remote maintenance design is adapted.

Major plant changes affecting Mechanisms Engineering include:

1. Substitution of steel posts for concrete walls between most cylindrical cells.
2. Rearrangement of hot cells below the canyon floor.
3. Revision and elimination of some plant auxiliary equipment and services with subsequent reductions in cost estimates for remote maintenance equipment and facilities.

An alternate maintenance scheme which utilizes an overhead-bridge manipulator operating in a sealed component trench, is being investigated and shows promise of becoming technically and economically more attractive than the rotating plug-cylindrical cell concept.

Comparison of an Aqueous Homogeneous Reactor Plant with a Liquid-Metal Fuel Reactor Plant indicates the latter appears to have many important technological and economical advantages from a maintenance standpoint.

A liquid-metal column control rod is being developed to operate by displacing a confined fluid up into an annulus around a plunger tube when the tube is driven down into the fluid.[P]

Research and development specifications stating the problems which need experimental investigation before LMFRE construction were prepared.

A study to determine what data can be obtained from the ETR loop's (No. 2) degasser and how this data can be used in the LMFRE degasser design has been started.

Since it has not been determined whether chloride or fluoride fuel processing is better, research on each process has been recommended.

Uranium solubility studies to supplement BNL work have been started at the Babcock and Wilcox Research Center.

The maximum tolerable sodium concentration in the fuel is not now apparent. Problems which might arise from sodium leaks into the fuel have been considered.

Uranium oxidation from the fuel could introduce varying amounts of uranium to the core, seriously damaging the reactor. The nature of uranium oxides and nitrides which might form in the loop will be studied.

The simplest method of adding uranium, zirconium, and magnesium to the LMFRE fuel stream will be direct solution of the metals in the fuel.

Schemes for handling fission gases from ETR loop (No. 2) have been studied. Recommendations for the loop gas handling design are:

1. The U-Bi solution should be degassed continuously to maintain a low concentration of the longer-lived gaseous isotopes.
2. Off-gas will be held on charcoal beds before disposal to permit sufficient radioactive decay.
3. A large by-pass charcoal bed with auxiliary cooling and pumping may be advisable for use during the shut-down of contaminated circulating air systems.

Previous quarterly reports on work at the Research Center covered primary design and construction of test apparatus. In the most recent report period, initial operations of most projects began and some results are reported.

Several dynamic test loops are in operation as part of the materials testing program. The capsule program is well underway with statistical methods being used to plan the sequence of test specimens. This apparatus has been so unusual that the number of stations is being doubled.

The prototype work which depends upon general design work in the project is becoming more clear as system requirements are specified. As a result certain phases of the test work are also underway.

The in-pile work is progressing satisfactorily. The "scoping report" has been discussed with representatives of Arco. Preparations are being made for insertion of the in-pile portion of the test loop in a critical facility for the ETR.

The number of cells for reactor and auxiliary equipment containment was reduced from 14 to 7.

Any component removed from the cells can be safely stored in a 31-ft-deep storage pool adjacent to the cells.

An 80-ton-capacity overhead crane, operated from a windowed control room, will be located in the reactor canyon above the cells.(P)

Remote maintenance equipment will be mounted on a heavy cast iron plug stored in the canyon floor above a maintenance hot cell.

There will be a steel door in one end of the canyon (for component removal) and two steel doors leading into the master control room and hot change room.

The office and laboratory area will be L-shaped and the basement area beneath will house the auxiliary systems not requiring heavy shielding.

The building will be ventilated so the air flow is directed from the least radioactive areas to the most radioactive areas and then to the hot filter room.

The anticipated total electrical load will be 3000 kw.

It has been assumed that any site selected will provide buildings, which will be remodeled to accomodate the required facilities for administrative offices, cafeteria, rest rooms, etc.; 10,000 sq. ft. will be needed.

Six months have been estimated for completion of a final building design and twelve months for reactor building erection.

II. RESEARCH AND DEVELOPMENT COORDINATION

(J. P. Holliday)

A. GENERAL

1. Test Program Planning Group Reorganization

The Test Program Planning Group has been redesignated the Research and Development Coordination Group which will coordinate all R&D activities within the project. The group will review each request for new R&D work for compatibility with project objectives, technical justification, scheduling and adaptability to existing programs. The group will continue to handle all materials problems, specification reviews and materials inspections. Functions of the Test Program Planning Group will be assumed by the R&D Group until the need for a separate Test Program Group arises.

2. Procedure for Initiating Research and Development

A procedure for initiating LMFRE R&D was established. It outlines the steps necessary from the initiation of an R&D request to either the awarding of a sub-contract or issuance of an E number at the B&W Research Center.

3. Research and Development Requests

The following have been received and forwarded for approval.

a. Rates of dissolution of U, Zr and Mg in Bi.

b. Selection of gas sparger plate material.

c. Degasser in-pile tests.

d. Corrosion of characteristics of a Pb-Mg eutectic.

e. Solubility of U in Bi.

f. Gas permeability of coated graphite.

g. Na-Bi reactions.

h. Rate of U-C formation.

i. Testing of valve bellows.

j. Mechanical properties, penetration and absorption testing of a new graphite.

k. Diffusion of U in Bi-impregnated graphite

l. Precipitation characteristics of U-Bi in heat exchanger tubes.

m. Electrolytic addition of U, Zr and Mg in Bi.

4. The Project Materials Handbook

The Project Materials Handbook underwent several revisions, and sections on beryllium and helium were added. Many of the original graphs were redrawn and issued for greater legibility. Approximately thirty-five additional handbooks were issued.

5. Research and Development Program

The overall R&D Program was reviewed to ascertain areas in which research is lagging or lacking. A list of these areas was compiled and requests for this research have been initiated.

B. MATERIALS

1. Graphite

a. A representative of the Project visited BNL from May 18, 1957 to June 7, 1957 to obtain all available information concerning experimental work on impregnated graphite. Data were obtained on Bi penetration, outgassing, growth under irradiation, helium permeability, radiation effects, and the stored energy phenomenon. Subsequently, meetings were held in Lynchburg with several graphite vendors who may supply graphite for testing purposes. In these meetings B&W presented the problems associated with the use of a graphite core, and the vendors presented fabrication difficulties which will be encountered. The extent of assistance the vendors will give B&W in this phase of the R&D program was also decided.

b. Graphite was ordered from three vendors for outgassing, sump penetration, graphite-to-metal seal, alternate impregnating, and carbide formation tests. A letter justifying the ordering of this graphite from three different vendors was approved by the AEC. Preliminary screening runs for graphite-to-metal seal tests were completed at the B&W Research Center, successful seals having been developed from small graphite samples.

c. Material specifications for the LMFRE graphite core were distributed for comments. Preliminary test specifications were prepared for outgassing and sump penetration.

d. The final grade of graphite for the LMFRE will be chosen from the three grades selected for testing, based on results of the following tests:

(1) Neutron absorption cross section.

(2) Bismuth absorption.

(3) Bismuth weepage.

(4) Homogeneity.

Pieces of each graphite will be held in reserve for final testing.

2. Croloy 2-1/4 and 1-1/4

a. Croloy 2-1/4 was specified for use in a sodium environment provided the allowable stress valves for Croloy 2-1/4 (Table UCS-23 of the Unfired Pressure Vessel Code) is reduced 20 percent for service at 900 F and 20 percent for service at 1000 F.

b. A meeting was held in Lynchburg to select cleaning procedures and uranium and additive concentrations for the first four dynamic corrosion loops (exclusive of the utility test loop) at the B&W Research Center. Table I lists the selected conditions.

TABLE I

ALLIANCE CORROSION LOOP START-UP CONDITIONS

Loop No.	Cleaning	Pre-conditioning	U	Zr	Mg	Heat Treatment	Time (hrs.)
2	Acid	350 ppm Mg 0 ppm Zr 1000 F	1150ppm ± 50	175 ± 25	350	Normalized & Tempered	1500
3	Alcohol	350 ppm Mg 175 ppm Zr 1000 F	1150ppm ± 50	175 ± 25	350	Normalized & Tempered	1500
4	Alcohol	350 ppm Mg 175 ppm Zr	1150ppm ± 50	0	350	Annealed	1500
5	Alcohol	350 ppm Mg 175 ppm Zr 1000 F	1150ppm ± 50	175 ± 25	350	Annealed	1500

c. Samples of Croloy 2-1/4 containing high N to Al ratios were requested from the Jones and Laughlin Steel Corporation. These samples will be delivered to the B&W Research Center for use in the ZrN film theory program.

C. ENGINEERING

1. Shielding

A test specification for shielding experiments, written in cooperation with the Reactor Design Group, was issued on June 28, 1957. Later the Brookhaven in-pile loop was found to be unsuitable for the shielding experiments because of a hold-up tank in the in-pile section. The 31-second hold-up allows all but one group of delayed neutrons to decay. It was then determined that construction schedules for the ETR and MTR in-pile loops precluded their use for shielding experiments. At this point the LMFRE Physics Group was given the responsibility for this experiment. It is now considered likely that the experiment will utilize a Van De Graaff accelerator as a source of neutrons.

2. Bellows and Valves

Finding manufacturers willing to work toward the development of Croloy 2-1/4 bellows for valves and instruments constituted a major effort. Welded and formed bellows are the two types considered. Vendors interested in both types were found; however, no bellows have been fabricated to date. Discussions with valve manufacturers indicate that fabrication of Croloy 2-1/4 bellows is the only real problem to be solved in production of the valves. Arrangements have been made to furnish prospective bellows manufacturers with Croloy 2-1/4 strip of proper thickness since it seems impossible to obtain this material commercially.

3. Dump Valve Loop

A loop was designed by the Alliance Research Center to test the dump valve, liquid-level indicators, an EM liquid-metal pump, and a flanged joint. The design stage was completed and prints of the loop submitted to the project S&C Group for action on September 9, 1957. Fabrication of the loop should begin during the next quarter.

III. LMFRE REFERENCE DESIGN STUDIES

A. PHYSICS AND MATHEMATICS (T. C. Engelder)

1. Reactor Statics

a. Accuracy of Criticality Calculations

An attempt has been made to determine the uncertainty in the estimated critical diameter of 41.3 inches for the reference design ($N_{25}/N_{Bi} = 1000 \times 10^{-6}$) because of limitations in the size of commercially available reactor graphite. This analysis is given in the Reference Design Report (BAW-1019)[1] and indicates an uncertainty of $\pm$ 5 inches in critical diameter. It is expected that this uncertainty will be greatly reduced as more information becomes available on the physical properties of actual reactor materials, and the results of the LMFRE critical experiment and allied work have been analyzed.

b. Comparison of Multigroup Reference Design Calculations

The criticality calculation reported above was made using a modified 20-group Spectral Code.[(P)] Recently the complete 40-group Spectral Code[(P)] has become available. To check the consistency of various calculational methods, the reference design case has been run using four different codes in use at The Babcock and Wilcox Company. The results for the more elaborate calculational methods are in quite good agreement, as can be seen in the summary of Table II. The conditions of the calculations are given below.

Geometry = spherical
Inelastic scattering = neglected
Temperature = 437.5 C
$V_{Bi}/V_G = 0.5$
$N_{25}/N_{Bi} = 1000 \times 10^{-6}$
$\rho_G = 1.80\ g/cm^3$
$\rho_{Bi} = 9.88\ g/cm^3$
σ_a (G) = 0.0046 b (2200 m/s)
Reflector thickness = 60 cm
Equivalent core cylindrical diameter = 103.6 cm

TABLE II

COMPARISON OF REFERENCE DESIGN CASE FOR VARIOUS CALCULATIONAL METHODS

Computer Code(P)	k_{eff}
2-group, multi-region	1.00
20-group multi-group, multi-region	1.035
20-group Spectral + 4-group multi-region	1.039
40-group Spectral + 4-group multi-region	1.038

c. Two-Dimensional Calculations

Since the LMFRE reference design is a fully reflected cylinder, the one-dimensional calculations summarized above are subject to error when converted to an "equivalent" fully reflected cylinder. To study the effect of geometry, a limited number of two-dimensional problems have been prepared for solution using the CURE two-dimensional code and the IBM-701 computer:

(a) Solid graphite reflectors, Phase I parameters

(b) Solid graphite reflectors, latest reference design parameters

(c) Same as (b), but including fuel annuli and fuel in end reflectors

(d) Tantalum control rods, test holes, and bismuth annuli

(e) Same as (d), but with no control rods

(f) Same as (d), but with no test holes

Cases (a) and (b) have been completed. For case (a), in three groups, $k_{eff} = 1.061$. This compares with a value of 1.036 using the one-dimensional Spectral Code and converting to cylindrical geometry by increasing the spherical core volume by an arbitrary 10 percent. It is apparent that this artifice is overly conservative.

In order to compare the one and two-dimensional results more directly, the CURE Code has been rewritten in four groups. Case (b) has been run in four groups and the results used in an attempt to correlate the one and two-dimensional calculations to obtain relationships that might be used in the future to improve the one-dimensional calculations. This case is for a core H = D = 100 cm and a reflector thickness of 78 cm of pure graphite on the top and sides. The procedure is as follows:

(a) Using one-dimensional code, plot k_{eff} vs B_z^2 for side reflected cylinder. Choose B_z^2 for k_{eff} equal to that of two-dimensional calculation and obtain $\phi(r)$.

(b) Using one-dimensional code, plot k_{eff} vs B_z^2 for end reflected cylindrical slab. Choose B_r^2 for k_{eff} equal to that of two-dimensional calculation and obtain $\phi(z)$.

(c) Normalize thermal fluxes and plot vs r and z on the same graph.

(d) Plot diagonal flux, $\phi(D)$, assumed to be the product $\phi(r) \cdot \phi(z)$ and compare to diagonal flux obtained from the two-dimensional calculation.

The resulting flux plots are shown in Figure 1. It is apparent that the one-dimensional model over-emphasizes the thermal flux in the reflector and that the space flux variables are not separable.

d. Inelastic Scattering

All the calculations referred to thus far have ignored the effect of inelastic scattering in bismuth. Inelastic scattering can be included in the Spectral Code by allowing the inelastically scattered neutrons to distribute themselves uniformly in energy down to any lower cut-off energy. Using this model for the reference design with a k_{eff}=1.00 before inelastic scattering, the k_{eff} with inelastic scattering included is about 1.06 if the lower cut-off is thermal, and about 1.00 if the lower cut-off is about 0.85 Mev, the threshold for inelastic scattering in bismuth.

Since this uniform scattering model is unrealistic, attempts have been made to improve the representation of inelastic scattering. The Spectral Code has been modified to use the evaporation model to represent the inelastic scattering. On this basis, k_{eff} for the reference design is about 1.02, an inelastic scattering contribution of about 2 percent.

Because of the widely spaced discrete levels in bismuth up to about 2 Mev, the evaporation model is not entirely realistic either. To check the applicability of the evaporation model, a series of hand calculations were performed in which the evaporation model was used for neutron energies down to 2.6 Mev. Below 2.6 Mev, where the excitation levels of bismuth and corresponding cross sections are known, the scattering into a group was taken equal to the removal from a group higher in energy by an amount equal to the excitation energy. These results checked the evaporation model very closely and justified its incorporation in the Spectral Code to represent inelastic scattering.

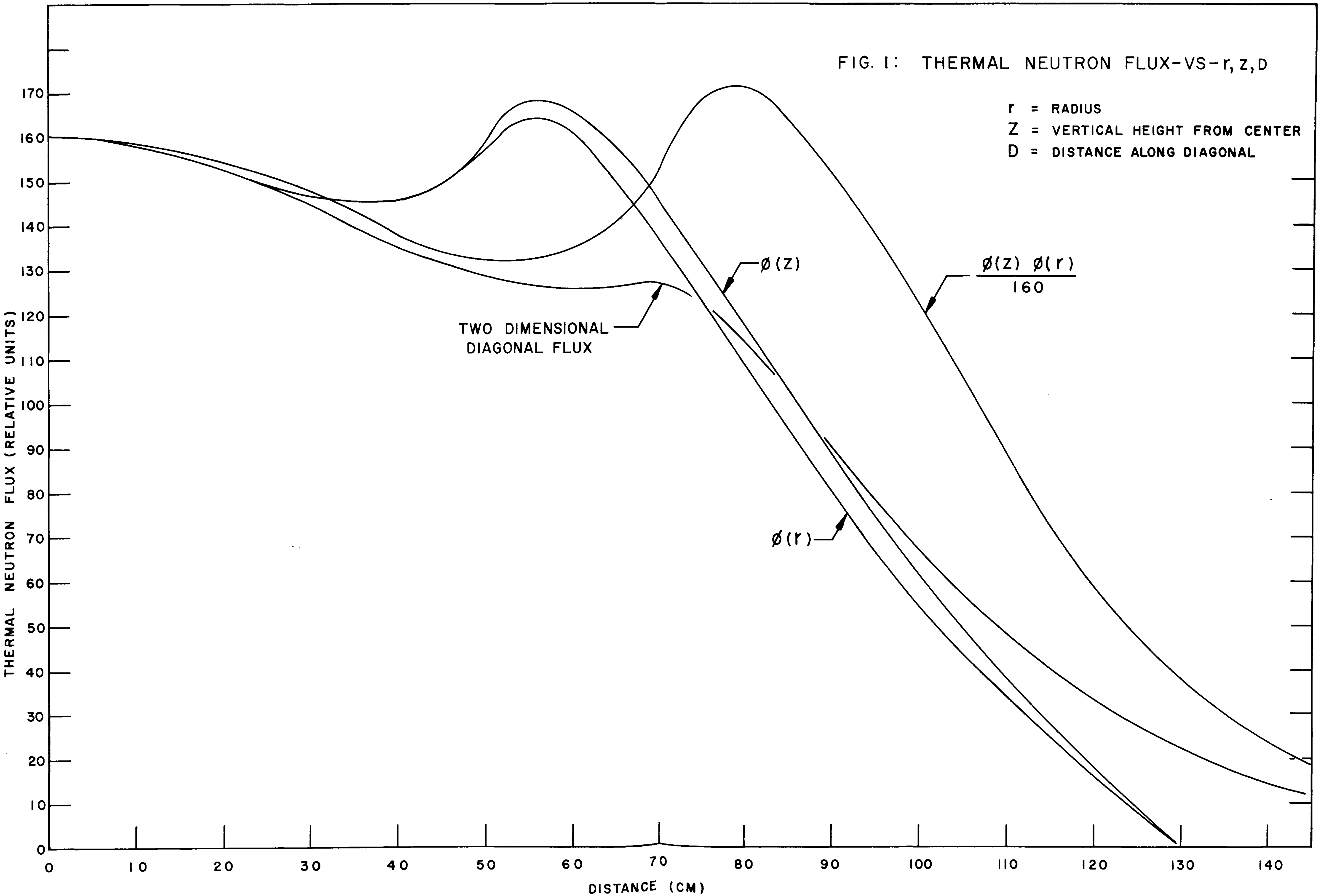


FIG. 1: THERMAL NEUTRON FLUX-VS-r, Z, D
r = RADIUS
Z = VERTICAL HEIGHT FROM CENTER
D = DISTANCE ALONG DIAGONAL
Ø(Z)
Ø(Z) Ø(r) / 160
TWO DIMENSIONAL DIAGONAL FLUX
Ø(r)
THERMAL NEUTRON FLUX (RELATIVE UNITS)
DISTANCE (CM)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
160
170

e. Effect of Fuel Penetration in Graphite

A criticality calculation and flux plot have been obtained for the case of fuel penetrating the graphite in both the core and reflector to the extent of 0.3 g/cm^3 of graphite. The Phase I case of an initial $N_{25}/N_{Bi} = 1000 \times 10^{-6}$ was chosen along with nuclear parameters from the corresponding two-dimensional CURE calculation. Under the conditions of fuel penetration, the critical concentration of U-235 in bismuth is reduced to $N_{25}/N_{Bi} = 826 \times 10^{-6}$. The corresponding flux plots are being used by the Reactor Design Section to compute reflector cooling requirements.

f. Effect of Reducing Allowable U-235 Concentration in Bismuth

In view of recent experimental data on the solubility of uranium in bismuth in the presence of additives and fission products, a preliminary calculation has been made of the increase in core size if the initial critical concentration is reduced from $N_{25}/N_{Bi} = 1000 \times 10^{-6}$ to 710×10^{-6}.

Calculations were made using the four-group, multi-region code[P] and a vertical buckling obtained from a previous two-dimensional calculation. The results indicate a critical core diameter of 45.3 inches and height of 47.2 inches for a critical concentration of $N_{Bi}/N_G = 710 \times 10^{-6}$ and a 30-in. solid graphite reflector (ends and side).

g. Neutron Source for Start-up

The flux distribution and leakage flux from a central, point neutron source have been calculated for the hot, unloaded (with U-235) LMFRE. For an Sb-Be source emitting 10^8 n/sec, the central thermal flux is about 3.5×10^5 n/cm^2 and the thermal flux at the surface of the reflector is about 5×10^2 n/cm^2 sec.

2. Reactivity Effects

a. Xenon and Samarium Temperature Coefficients

The positive xenon temperature coefficient of reactivity has been recomputed, for the case of no xenon removal, using the mean of the steady state values of the Xe-135 concentration at the core inlet and outlet. The resulting temperature coefficient is $+2.2 \times 10^{-5}/C$, in good agreement with previous calculations. In the worst case, if all of the xenon is assumed to adsorb on the core graphite, the xenon temperature coefficient becomes about $+4 \times 10^{-5}/C$.

Recent calculations have shown that the absorption cross section of

Sm-149 changes quite rapidly with temperature, for neutron temperatures above thermal. At 0.07 ev, $\overline{\sigma} = 4.6 \times 10^{-20}$ cm^2, and $\partial\sigma / \partial T = 42.08 \times 10^{-24}$ cm^2/C. These values lead to a samarium temperature coefficient of reactivity for the LMFRE of $+0.82 \times 10^{-5}/C$.

b. Effect of Poison on Fuel Concentration

The U-235 addition necessary to override fission product poisoning has been checked by raising the thermal cut-off energy from 0.07 ev to 0.2 ev. This change increased the required U-235 additions by only 10 ppm. Detailed information regarding the resonance structure of the fission products would be necessary to refine this approach.

3. Shielding Calculations

a. Reactor Cell

The problem of neutron activation of the concrete surrounding the reactor is important because one maintenance scheme requires access to the cell after the reactor vessel has been removed. In order to obtain the neutron absorptions as a function of distance in the concrete, a multi-group multi-region calculation of the neutron flux has been made and normalized to 20 MW of power. The relaxation length of thermal neutrons, a quantity sensitive to the hydrogen content of the concrete, was found to be approximately 8 cm. This can be compared to an experimental value of 11 cm reported in ORNL-2081 for concrete with a lower hydrogen content.

b. Primary Loop Shielding

Calculations of the shielding for the primary loop require a plane source of neutrons of delayed neutron energies. This problem has been run in twenty-group and slab geometry and fast and slow fluxes obtained in barytes, barytes + 1 percent boron, and BNL (iron loaded) concretes. In BNL concrete, the thermal and fast flux come into equilibrium after about one foot penetration, after which the fluxes fall off with a relaxation length of about 5 cm.

The case for barytes concrete is less clear. No equilibrium between the fast and thermal fluxes is observed up to a penetration at least three feet, and thereafter the thermal flux is influenced by the finite thickness of the slab (five feet). The fast flux appears to fall off exponentially with a relaxation length of about 7 cm after a penetration of about two feet. In this region the thermal flux decreases more slowly, with a relaxation length varying between 10 and 12 cm.

To attenuate the thermal flux faster, one case has been run with barytes loaded with 1 percent boron. In this case the fast flux was little affected, but the thermal flux showed indications of coming into equilibrium with the fast flux after about three feet of penetration.

The analysis of the attenuation properties of delayed neutrons in concretes is continuing. Experimental data will be examined as they become available.

4. Reactor Dynamics

a. Core Transient Analysis

A number of inadequacies have been noted in the generalized set of core transient equations reported in BAW-1019[1] and BAW-1024[2]. These equations fail to yield a final steady state solution. This casts doubt on the validity of the large pressure transients reported earlier, because they occurred at times when the lack of a steady state solution was important.

The equations have been improved to permit a final steady state solution, to improve the treatment of the delayed neutron precursor concentration, and to introduce a radial temperature dependence in the graphite. The reformulated code has been completed and checked out.

Attempts have been made to permit quicker machine solution of these equations by adopting a modification of Euler's method of integration instead of the Milne method previously used. Initial results indicated a savings in time of about 50 percent was possible, but on running several test cases wild oscillations in pressure were observed. This difficulty has not been surmounted to date; therefore, the next set of transient problems will be run using the original Milne integration routine.

Two additional refinements in the core transient analysis are under consideration: the introduction of change of phase to permit an analysis of bismuth boiling, and the division of the reactor into several space regions. The incorporation of these features in the generalized set of core transient equations will overload the Electrodata computer, so methods of simplification are being investigated.

b. Mathematic Stability of Core Transient Equations

Work is continuing on a mathematical investigation of the stability of the set of core transient equations. The procedure being followed is to linearize the equations, obtain the matrix elements from the coefficients of the variable terms, and compute the eigenvalues of the matrix. The stability of the system is then inferred from the signs of the eigenvalues. A computer program is now being written to obtain these eigenvalues.

c. System Analysis

The application of digital methods to problems of system analysis is being investigated as an adjunct to the general systems analysis being planned for the analog computer. A preliminary analysis of the problem is nearly complete.

d. In-Hour Relation

The in-hour relation previously derived for circulating fuel reactors has been reprogrammed to obtain solutions for shorter periods.

5. Critical Experiment

a. Size of Experiment

A number of four-group two-region calculations have been made to determine the range of sizes of LMFRE critical experiments. On the basis of these calculations, the experiment is being designed to accommodate cores up to 56 inches in diameter and reflector thicknesses up to 28 inches. Materials are being ordered to permit experiments with the following range of parameters:

$$V(Bi)/V(G) = 0.5$$
$$N(25)/N(Bi) = 0 \text{ to } 1200 \times 10^{-6}$$
$$\text{Reflectors} = 0 \text{ to } 28 \text{ inches}$$

Other V(Bi)/V(G) from 0.2 to 2.0 can be accommodated should circumstances require.

b. Materials

Preliminary specifications have been prepared for the bismuth, core and reflector graphite, fuel channel supports (either graphite or aluminum), and U-Al foils. Preliminary cost estimates and delivery dates have been obtained.

A number of foil disadvantage factor calculations have been made which have led to the choice of 0.010-in. 18 weight percent U-Al foils to minimize flux depression and self-shielding.

c. Tables

To accommodate the range of sizes and weights mentioned above, the moving and stationary tables have been designed to support a weight of about 22 tons each with a deflection less than 0.003 inches. Each table is 5 feet by 10 feet. A preliminary table design has been prepared and sent to manufacturers for estimates and comments. On the basis of information received, the final table design has been completed and sent out for sealed bids.

d. Table Drive

A variable speed DC motor is planned for the table drive. On the basis of preliminary gap-reactivity and reactivity-period calculations, the maximum approach speed will be 2 in./min. These calculations are being refined.

e. Neutron Source Requirements

Estimates have been made of the leakage flux from various neutron sources for start-up. It appears that a source strength of 5 curies of Po-Be will be adequate.

6. Other Experimental Physics

a. Delayed Neutron Shielding

Because of the importance of delayed neutrons in the shielding of the primary loop, an experiment is being planned to measure the relaxation length of delayed neutrons in concretes. Preliminary calculations show that the Li (p,n) reaction will produce neutrons of the proper energy for protons of 2.3 Mev. Using a beam current of fifty microamperes, a thermal neutron flux sufficient for indium foil activation can be obtained to a depth of two to three feet in barytes concrete.

Arrangements have been made with Brookhaven National Laboratory to use their Van de Graaff accelerator as neutron source. A 100 kev lithium target is being designed and other experimental details are being investigated.

b. Graphite Cross Sections

In the LMFRE, the critical concentration is strongly dependent on the absorption cross section of graphite. The developmental nature of the impregnated graphite to be used in the LMFRE makes it even more important to determine its nuclear properties. Arrangements are being made with Hanford to measure the cross section and its variation in the large 40-in. diameter pieces of developmental graphite.

Calculations have been made of the accuracy of diffusion length measurements in the 40-in. pieces as an auxiliary experiment.

B. REACTOR ENGINEERING (J. J. Happell)

1. Core and Reflector Design

Specifications and associated preliminary working drawings were completed early in the quarter for the core-end reflector assembly and side reflector. The specifications were submitted to graphite vendors, requesting fabrication time estimates and comments on manufacturing problems. Replies were received in August. Significant comments indicated (1) tolerances on the core OD and side reflector ID would have to be relaxed slightly, and (2) the design did not provide access for cleaning loose scale from core fuel passages. All vendors estimated 12 months fabrication time except one who required additional time to get base stock.

An alternate top-reflector arrangement was completed to provide access for cleaning core fuel passages. The arrangement includes a quadrant-type graphite tank and a drilled block insert secured by key blocks and graphite screws. The specified gasket material is a natural graphite filler.

The degree of scaling expected after core impregnation will be determined in the graphite R&D program.

The reference reactor core design was modified based on vendor recommendations and efforts to simplify the critical experiment design. The core Bi/C volume ratio was revised to 0.5 (on a cell basis) and top and bottom reflectors were redesigned to give a Bi/C volume ratio of 0.2 (avg.) measured at the horizontal midplane. The flow channel diameter of top and bottom reflectors was changed from 2 to 1.5 in. to achieve the required end reflector Bi/C volume ratio. Other modifications included a saw-tooth type cemented joint recommended by graphite vendors and enlargement of the central test hole diameter to make the ΔT identical to an average flow channel ΔT. Special fittings, to be placed at the entrance of each core fuel passage, were designed to reduce the probability of pluggage.

Based on thermal calculations to date, a 1 1/2-in. core flow channel and a 2.3-in. square pitch were specified for the reference core.

Additional detailed engineering work on the reactor vessel and associated internals was completed; specifications and drawings were distributed.

Preliminary layouts of metal core tank and single-fluid reactors have been completed utilizing the reference design vessel and internal components wherever possible. This approach provides for flexible design changes.

2. Graphite R&D Program

A preliminary program was organized and presented to graphite manufacturers at meetings in Lynchburg. Based on these meetings and subsequent work, large blocks of graphite will be ordered from each manufacturer for testing purposes. These test results will be the bases for selecting the most suitable LMFRE graphite. The program requires each vendor to submit part of a large block for U-Bi absorption, U-Bi weepage, nuclear cross section, density, and certain mechanical tests. After the LMFRE graphite is selected, the vendor will submit the remainder of the large test block for an extensive test program to obtain mechanical and thermal properties. Other tests will determine radiation damage effects, with emphasis on reducing thermal conductivity and dimensional stability. Further details on the R&D program may be found in section II (R&D Coordination).

A report from the B&W Alliance Research Center covered small-scale tests on graphite-to-metal compression seals specified for the reference design. The report is encouraging and indicates that small diameter seals can be made utilizing reasonable contact stresses.

A development proposal for beryllium thimbles required for reference design test ports and control rods has been received from Brush Beryllium Company. Sketches of the latest reference design thimbles were submitted to the R&D Coordination Group for distribution to Brush Beryllium Company and B&W Alliance Research Center.

A summary of BNL's research and development on graphite was received in June. Two items of interest indicate that uranium appears to diffuse into bismuth-impregnated graphite and that a threshold pressure seems to exist below which certain graphites do not pick up bismuth. The latter may be due to bismuth surface tension, and presents interesting possibilities in LMFR reactor design. BNL was requested to determine weepage rates of bismuth through graphite and whether a threshold pressure actually exists below which certain graphites absorb no bismuth.

3. Nuclear Engineering

a. Core Thermal Analysis

A report on thermal calculations for the reference design core was issued. Conclusions are:

(1) Normal steady-state operation presents no problems, assuming a factor of 1/3 reduction in the thermal conductivity of graphite due to radiation damage.

(2) Excessively high U-Bi temperatures may result if a peripheral channel becomes completely plugged.

(3) Partially plugged channels do not create excessive temperatures.

(4) Radiation effects (from absorbed U-Bi) on graphite properties can result in excessive thermal stresses in thick core sections.

Based on the last conclusion, a more detailed calculation has been initiated to determine the effects of reduced thermal conductivity in the reference core design.

The following reports also were completed:

1. "Some Steady-State Thermal Calculations Performed on the LMFRE Reactor and Some Conclusions Based on the Results of Those Calculations"
2. "Temperature Distribution Around a Blocked Channel of LMFRE Core"
3. "Hot Channel Factor Report"

b. Core Poison Buildup

Fission product buildup, assuming various amounts of U-Bi absorbed in graphite, was recalculated to obtain an estimate of expected poisoning. The conclusion, assuming a 20 MW-year core life, was that fission product poisoning is not important in the LMFRE reactor. The most conservative assumption regarding fuel concentration selection is that core graphite will not pick up U-Bi.

c. Control Rods

Control rod thermal calculations to determine the practicality of solid rods and required cooling rates were initiated in September; results will be reported in the next quarterly report.

d. Graphite

A meeting was held with Mr. H. A. Ohlgren at the University of Michigan to get information about a new high-density (2.2) graphite developed there. This material is not sufficiently advanced for LMFRE use, but may be useful in future LMFR's. Mr. Ohlgren indicated limited samples of coated and uncoated high-density graphite could be made available for the LMFRE graphite R&D program.

e. Shielding

A shield thickness was specified for the barytes shield around the ETR test loop. Conferences were held with the Physics Section to set parameters and agree on a procedure for obtaining R&D information necessary to calculate LMFRE shield thicknesses. Shield calculations were prepared to size Phase I and IA wall and canyon floor thicknesses. Expected steel liner activation in the reactor cell was calculated. Boral sheets will be required on both sides of the liner to permit access after removal of all hot components.

Vendors were contacted to obtain information on their high-temperature neutron detector proposals which are being revaluated with a view to starting contract negotiations during the last quarter of 1957.

The systems analysis (analogue) studies contract was completed and vendor negotiations should start soon. The initial studies, as outlined by the systems analysis committee, will include analysis of the primary loop and associated components.

On August 19 A. D. Little, Inc. began a study to develop a continuous uranium monitor for the LMFRE. This study should be completed by December 16, when a report will be issued and preliminary prototype design begun.

f. Fuel Concentration

The required initial U-235 concentration for the reference design is estimated to be about 800 ppm by weight based on reduced uranium solubility reported in BNL metallurgy memo No. 696 and a revaluation of the solubility safety margin. The Physics Section reported this concentration will require an active core diameter of 45 to 47 inches.

g. Instrumentation

Preliminary specifications covering reactor nuclear instrumentation have been completed and will soon be sent to vendors for proposals. This specification covers only the conventional, i.e., standard temperature detectors and instrumentation; high-temperature neutron detector development will be handled under a separate subcontract.

The coming quarter's work will be directed toward:

(1) Establishing necessary specifications and ordering samples for the graphite testing program

(2) Completing the control rod thermal study and specifying arrangement and materials

(3) Completing containment vessel specifications

(4) Undertaking a side reflector thermal analysis and specifying a channel drilling pattern

(5) Revising core and side reflector specifications and forwarding them to graphite manufacturers.

C. SYSTEMS ENGINEERING (S. S. Waldron)

1. Summary

The material prepared for the Reference Design Report[1] underwent certain modifications and is known now as the "Phase IA" design.

Study of individual system design problems continued. Particular emphasis was placed on start-up heating, dump-system parameters, intermediate-system design, offgas-system design, and pressure rise in the reactor plant area following an accident.

2. General

a. The Reference Design Report[1] included preliminary system schematics, system descriptions, and plant arrangement information. To back up this information and provide a basis for cost estimates, preliminary specifications were prepared for components.

b. The preliminary design was modified to reduce cost and take advantage of improvements in engineering design; the major modifications were:

(1) The standby pumps were removed from the core and reflector primary cooling systems to simplify piping. The interruption of service due to a component failure in an experimental reactor is considered tolerable. Spare pumps will be available to replace any that fail. Loss of pumping power does not create a hazardous condition in the LMFRE.

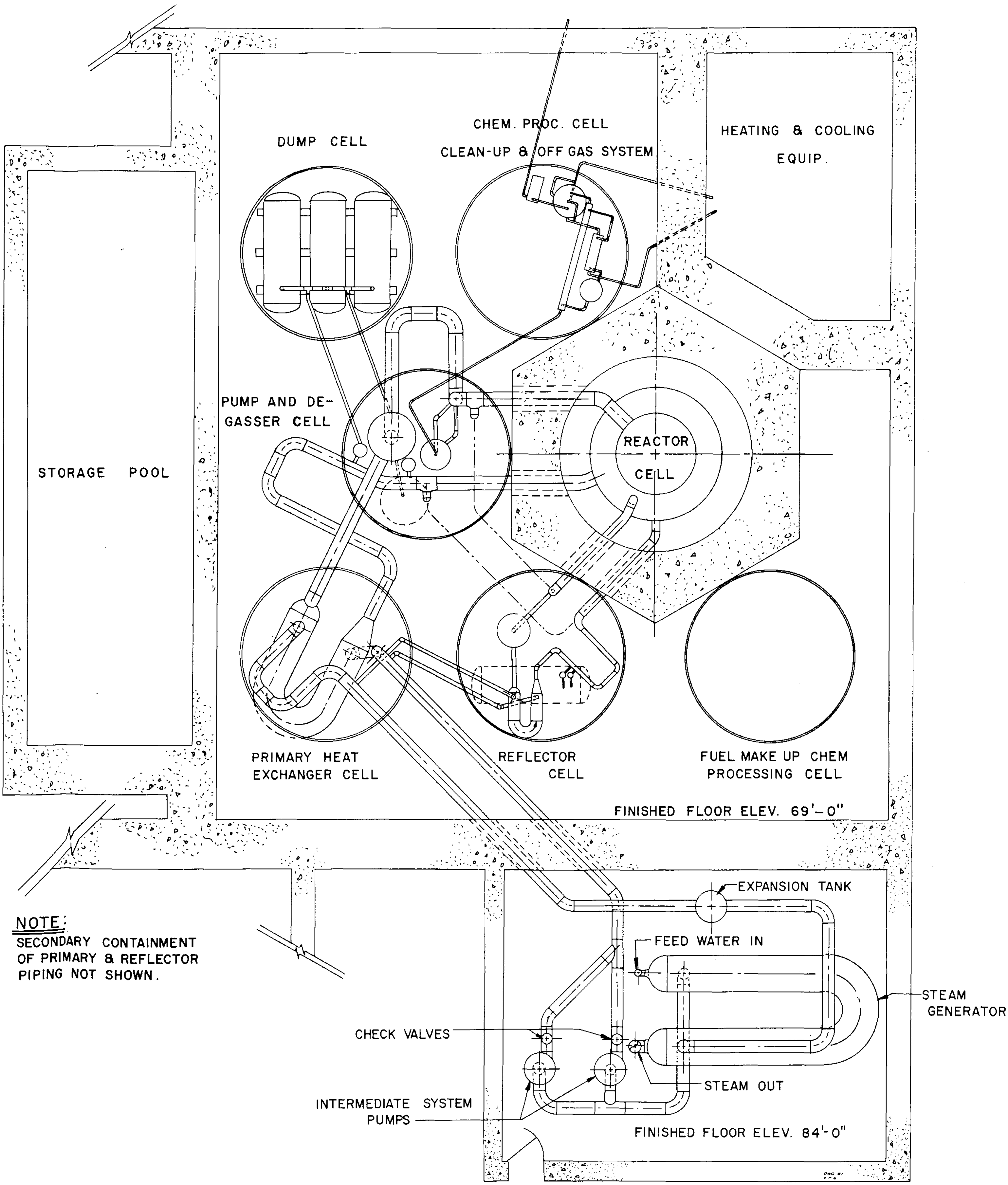
(2) The full-stream degasser was replaced by a side-stream type handling 10 percent of the primary system flow, since the latter appears sufficient for LMFRE function.

(3) More accurate information on xenon solubility in bismuth justified a drastic reduction in the offgas system carrier-gas flow rate. This reduction minimizes the importance of inert gas conservation. A throw-away system was adopted, eliminating recirculation of gases and the attendant charcoal beds.

(4) The number of plant-containment cells was reduced through the above changes and improved arrangements. The latest arrangement is shown in Figure 2.

(5) The concrete between cells was eliminated. It was thought originally that new equipment could be installed manually in any cell following removal

FIG. 2: EQUIPMENT ARRG'T. PHASE I A



of all radioactive equipment and decontamination. Shielding then would be necessary to attenuate radiation from neighboring cells. This ability would be important if the cell components were to be replaced by basically different equipment; however, it is thought now that all components will be replaced remotely with similar equipment.

(6) The water-cooled steam condenser was replaced by an air-cooled type to provide more flexibility in site selection.

c. After preparation of the Phase IA design, it was decided to recommend elimination of the steam generator and the entire steam system. The sodium intermediate system would then be cooled directly by air. At present, a steam generator for the LMFRE necessarily would include a third fluid to separate sodium and steam. It is hoped that knowledge of the Na-water reaction will have progressed to the point where this feature can be eliminated by the time an LMFR is built. Thus, a steam generator installed in LMFRE would not be representative of one for an LMFR. The steam generator also appears unnecessary for obtaining plant kinetics information; reactor kinetics can be investigated without the steam system, and steam generator kinetics can be determined elsewhere.

d. Study of containment requirements continued. The preliminary Reference Design Report required tight-fitting double containment for all systems containing fission products.[P] In the hazards analysis, the double containment was assumed breeched in the maximum credible accident, requiring complete reliance on the reactor plant containment. Since the reactor plant containment must be designed for this condition, elimination of the tight-fitting containment appeared justified. Alternate means for accomplishing the auxiliary functions of the tight fitting containment (e.g., system heating and cooling) are under study.

3. Primary and Auxiliary Systems

a. The primary system dump time requirements were investigated. It appears that a rapid dump is not required for nuclear reasons. Dump time is determined by the desire to reduce bismuth spill to a minimum practical value in the event of a leak. Based on this philosophy, two 4-in. dump lines currently are specified.

b. A preliminary investigation indicated that overpressure in the reactor plant is 7 psi following the maximum credible accident. This figure cannot be accepted as final due to uncertainties in the behavior of bismuth-air mixtures at high temperatures. Research to resolve these uncertainties was initiated.

c. Alternate bismuth melt and fill system and reflector cooling system designs were investigated.

d. Investigation of primary system heatup continued, including electrical and fluid systems; inductance heaters[(P)] and resistance systems appear most promising.

4. Intermediate and Auxiliary Systems

Redesign of the inert-gas and sodium systems was initiated, and a sodium-to-air heat exchanger design was investigated.

An alternate design for the heat-removal system, using air as the intermediate fluid, was investigated and rejected. The design proved large and unwieldy and presented a number of difficult problems (e.g., shielding and maintenance) which were not solved in the study time provided. A preliminary analysis also indicated the system would be difficult to control because it lacked stored energy.

A request was initiated for research on the corrosion of lead-magnesium alloy, an alternate intermediate fluid, which might replace sodium if current research indicates the latter cannot be used.

5. Arrangement of Equipment

Two conceptual equipment arrangements were prepared (Phase I and Phase IA). In addition to these formally presented arrangements, numerous other conceptions were considered and carried to various stages of representation in drawings.

Variations considered were:

a. Design of a minimum plant

b. Vertical heat exchanger

c. Reverse flow in reactor

d. No close fitting secondary containment

e. 14-, 16-, and 18-foot circular cells

f. Rectangular trench in place of circular cells

Since component arrangement depends primarily upon the type of remote maintenance employed, plant arrangement will fluctuate until a basic remote maintenance design is adapted. This is scheduled for December 1957 finalization.

6. Chemical Processing Equipment

Equipment efforts have been concentrated largely on volatile fission products removal and treatment of nongaseous radioactive waste products.

Major changes (Phase I to IA) in the conceptual design of equipment for volatile fission products removal, resulted from:

a. New information on xenon solubility in bismuth from ORNL, and acknowledged by BNL, indicates a much lower solubility than previously assumed.

b. Degassing efficiency was revised from 100 to 10 percent in light of additional studies on xenon poisoning and the effect of xenon on reactor negative temperature coefficient. These revised data and objectives led to the acceptance of side-stream in place of full-stream degassing and a drastic volume reduction in inert sweep or sparge gas.

Economics then dictated that once-through inert gas with decay storage containment replace removal of volatile fission products in charcoal beds by inert gas recycling.

The most significant conceptual design change in the radioactive liquid waste disposal facilities is addition of a means for precipitating bismuth and its removal by filtration.

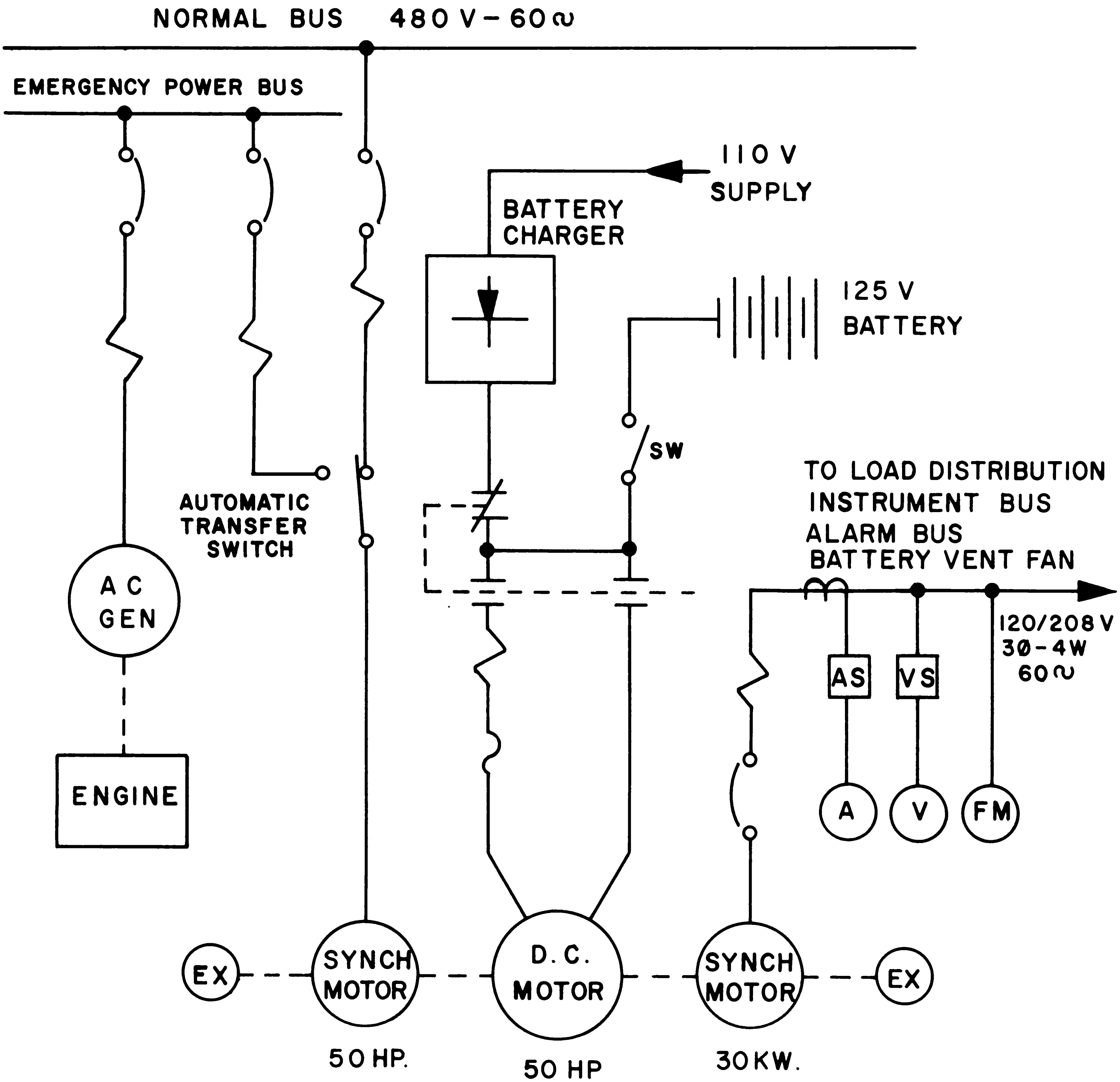
Little work was done on the revision of equipment design concepts for the other chemical processing facilities. Intensive work in this direction will begin when results of current R&D work become available.

7. Electrical Systems

a. A constant power supply system for instrumentation and other emergency loads was developed (Figure 3).

b. Study of electrical methods for primary system heatup continued.

FIG. 3: CONSTANT POWER SUPPLY SYSTEM



D. MECHANISMS ENGINEERING (G. R. Winders)

1. Major Plant Changes

The Phase I maintenance concept was revised for submission to the AEC. Major plant changes affecting Mechanisms Engineering included:

a. Substitution of steel posts for concrete walls between most cylindrical cells.

b. Rearrangement of hot cells below the canyon floor.

c. Revision and elimination of some plant auxiliary equipment and services with subsequent reductions in cost estimates for remote maintenance equipment and facilities.

2. Alternate Maintenance Scheme

a. Description

An alternate maintenance scheme is being developed which shows some promise of becoming technically and economically more attractive than the rotating plug-cylindrical cell (P) concept presented in the Reference Design Report.[1] This scheme would utilize an overhead-bridge manipulator operating in a sealed component trench, the roof of which would be the canyon floor. One end of the trench would be a horizontally-sliding door of sufficient height to allow passage of the bridge manipulator with the largest component attached to one of the bridge hoists. An inspection hot cell with a storage pool would be located beyond the door. The bridge manipulator would be used for remote operations within the trench and the inspection hot cell this manipulator would be supplemented in both areas by a radio controlled mobile truck mounting electronic manipulators and TV cameras. Remote viewing would be provided by portable periscopes in the canyon floor; supplementary TV equipment would be mounted on the bridge manipulator and spotted around the work area.

b. Modifications

Modifications of the overhead-bridge manipulator scheme are being made to explore its feasibility for a full-scale LMFR central station power plant. One interesting modification would eliminate a massive door at one end of the trench. Instead, the monorail mounted hoists and manipulator could travel off the bridge (moving at right angles to bridge travel) into a labyrinth alley leading to the inspection cell. This labyrinth arrangement could be used with either a lineal trench and an overhead traveling crane bridge or a circular trench and an overhead polar crane bridge.

3. Liquid-Metal Column Control Rod

a. Description

A liquid-metal column control rod is being developed to operate by displacing a confined fluid up into an annulus around a plunger tube when the tube is driven down into the fluid.[P] The fluid will be a liquid-metal alloy with sufficient neutron capture cross section to offer effective reactor control by the precise mechanical fluctuation of the column height in four core rod assemblies. This height will be determined by the plunger's location in the thimble tube, plunger movement coinciding with movement of an air cylinder piston to which the plunger is rod-connected. Downward or upward piston movement will produce column movement in the opposite direction.

b. Force For Piston Movement

(1) Routine Shimming

For routine shimming, a fractional horsepower electric motor and reduction gear will drive the end-block of a ball-nut assembly. Although the end-block will push the piston shaft, it will not be connected, so scramming of the piston will drop the piston shaft away from the end-block. Therefore, the piston will be scrammed independently of the shim drive equipment which offers no complications to a scramming operation.

(2) Scramming

For scramming, air pressure and spring action within the air cylinder will supply the downward thrust of the piston and plunger. During normal operation, air pressure will be held on both sides of the piston, and slight over-pressure will normally be carried on the underside to compensate for the thrust of a cylinder spring acting on top of the piston. In the shimming operation described above, the combined downward thrust exerted by the end-block on top of the piston shaft and the air and spring pressures on top of the piston, will be enough to drive the piston down against the thrust of the counter-acting air pressure on the piston's underside. However, for a scram operation, the air pressure on the piston's underside will be quick-exhausted and the combined force of the air pressure (accumulator fed) and spring action pushing on the piston's top side will drive the piston and plunger down through their stroke.

A pressure switch will be provided on the bottom of the shaft end-block so the switch will be depressed when the end-block is in contact with the piston shaft. Scramming the piston will drop the piston shaft and release the pressure switch which will break the holding current to the quick-exhaust valve on the bottom of the piston. Thereafter, to unscram the rod, the operator must drive the ball-nut assembly end-block down to depress the pressure switch and then close the quick-exhaust valve. This will permit air flow to enter the bottom of the cylinder to again drive up the piston and lower the liquid-metal column.

c. Relative Advantages

(1) The overall height of the reactor would be much less with liquid-metal column control rods because the maximum stroke length for the piston and plunger would be about 5 inches, whereas the corresponding length for a solid rod would be about 4 feet.

(2) The overall size and complexity of the rod system would be less for the liquid-metal rod, resulting in easier and less frequent maintenance and more reliable operation.

(3) Because of shorter strokes and lighter weights, the liquid-metal column rod would have a lower inertia system at high scram rates than the solid rod.

(4) Remote handling and replacement would be simpler with a liquid metal-rod system.

d. Problems Inherent To Development

(1) Development of reliable molybdenum bellows-type expansion joints and seal welds between molybdenum and Croloy. These joints and welds must operate under thermal cycling and thermal shock conditions and might have to endure dynamic stresses during rod movement.

(2) The development of a reliable metal alloy for a liquid-metal column. The best hyporthetical alloy would include the following:

(a) Fairly high neutron capture cross section
(b) Low melting point
(c) High boiling point
(d) Good irradiation stability
(e) Chemical compatability with containment and rod materials.
(f) Will not wet containment or rod materials.

The alloys being investigated generally contain various proportions of Bi, Cd, Pb, In, and Sn. One alloy showing promise is "Cerrolow", which has a melting point of about 117 F and the following composition by weight percent: Bi=44.7, Pb=22.6, In=19.1, Sn=8.3, Cd=5.3.

(3) Theoretical design studies and prototype testing must demonstrate effectiveness, reliability, and relative simplicity before final rod specifications can be determined.

4. Review of Two Reactor Concepts

Aqueous Homogeneous Reactor Plant and Liquid Metal Fuel Reactor Plant concepts were reviewed in regard to the broad effects of various features upon comparative costs of plant maintenance.

Conclusions

(1) Although the systems have many general similarities from a maintenance standpoint, the LMFRE concept appears to have many important technological and economical advantages over the Aqueous Homogeneous Reactor concept. The successful development of these advantages is dependent upon the engineering design, laboratory research, and plant-testing programs being conducted.

(2) The relative position of either system could be improved by developments in one plant not equaled in the other; i.e., the development of long-life components for one plant could have good effects on that plant's relative economic advantages. Conversely, insurmountable problems inherent to either plant could adversely affect that plant's relative economical and technological attractiveness.

(3) There is little sound basis now for formulating detailed maintenance cost estimates for either plant; however, a comparison of cost estimates and alternates for both plants might be valuable as a basis for preliminary design and administrative decisions.

5. Remote Maintenance Consultants

Walter Kidde Nuclear Laboratories (WKNL) has been selected by B&W and approved by the AEC as consultants on remote maintenance problems inherent to the LMFR concept.

WKNL's basic approach to remote maintenance and plant layout problems in the LMFRE is to start conceptual design work with the systems of a 550 MW central station LMFR plant. After various schemes have been compared, several will be extrapolated to LMFRE layouts for further development and comparison. Another work phase would then be required for final recommendations and development of maintenance equipment specifications and cost data for the scheme selected.

One plant arrangement conceived by WKNL would utilize a capsule arrangement (all components within a loop would be close-coupled and fit into a portable metal containment) which could be taken out as a unit and transported to a hot-shop for overhaul while the plant operated with a replacement capsule unit.

6. Facilities Visited

The following facilities were visited to inspect existing equipment and discuss proposed LMFRE facilities:

a. Argonne National Laboratory
b. Ames Laboratory
c. Oak Ridge National Laboratory
d. Savannah River Project
e. Sodium Reactor Experiment
f. Materials Test Reactor
g. Engineering Test Reactor
h. Experimental Breeder Reactor
i. Submarine Thermal Reactor
j. Aircraft Nuclear Propulsion Project, Remote Handling Facilities
k. National Reactor Testing Station, Chemical Processing Facilities

Product application conferences were held with some prospective vendors of remote handling and maintenance equipment.

E. CHEMICAL PROCESSING (R. D. Pierce)

1. Introduction

This quarter was primarily devoted to preparing research and development specifications to state problems needing experimental investigation before LMFRE construction, to suggest possible means of experimental approach, and to provide preliminary cost estimates for this work. The following research programs have been detailed:

a. Rapid bismuth clean-up.
b. Uranium solubility.
c. Effects of sodium (secondary coolant) and fuel mixing.
d. Uranium oxidation from U-Bi fuel.
e. Solution rates of additives in bismuth.

In addition, progress was made on fuel degassing problems. A study to determine the amounts of gaseous fission products generated in the ETR loop (No. 2) and safe methods of handling these gases was completed. A study to determine what data can be obtained from this loop's degasser, and the use of this data in the LMFRE degasser design has been initiated and should be completed during the next quarter.

2. Research and Development Programs

a. Rapid Bismuth Clean-up

Chloride and fluoride fuel processing research is being studied at BNL and ANL. LMFRE minimum plant processing requirements have been determined and sent to these laboratories. Since the more satisfactory method for processing LMFRE fuel is uncertain, research on each process has been recommended.

b. Uranium Solubility

Uranium solubility studies have been started at the B&W Research Center to supplement work at BNL. Because this data is important and effects of the many fuel constituents complex, a third independent study of this system is under consideration.

BNL has shown that zirconium suppresses uranium solubility in bismuth. It has been proposed to study the liquidus curve in the Bi-U-Zr ternary and to observe the effects of dissolved magnesium, iron, chromium, sodium, and representative fission products on this liquidus.

c. Effects of Sodium and Fuel Mixing

The determining factor in setting a maximum tolerable sodium concentration in the fuel is not now apparent. The following problems which might arise from sodium leaks into the fuel have been considered:

(1) Reactor Poisoning

A sodium concentration of about one percent will contribute only one percent poisoning in the reactor, so poisoning is expected to be unimportant in comparison to chemical effects.

(2) Sodium-Graphite Interaction

Bench scale tests will be required to study Na-Bi reactions with graphite. Graphite specimens will be immersed in bismuth containing normal fuel constituents and various amounts of sodium. The solution and graphite will be studied to detect any effects, particularly on physical properties of the exposed graphite.

(3) Effect on Solubility of Fuel Constituents

Dissolved sodium effects on solubilities of fuel constituents will be examined in the uranium solubility program.

(4) Heat Generation on Mixing of the Two Fluids

The adiabatic temperature rise for sodium bismuth mixing, which was prepared from BNL estimates, is shown in Figure 4. Possible heat generation effects at small leaks on local corrosion rates will be examined.

(5) Effect of Sodium on Corrosion of Croloy by the Fuel

Sodium's effects on Croloy corrosion by LMFRE fuel will initially be studied in capsule tests. The standard rocking capsule test will be used.

(6) Formation of Solids at Sodium Leaks

Sodium and bismuth form two intermetallic compounds, NaBi and Na_3Bi. These compounds may cause plugs in the heat exchanger tubes or might make leaks self-healing; these possibilities will be studied in bench scale experiments.

(7) Detection of Leaks

Methods to detect sodium leaks into the primary loop will be studied.

(8) Removal of Sodium from the Fuel

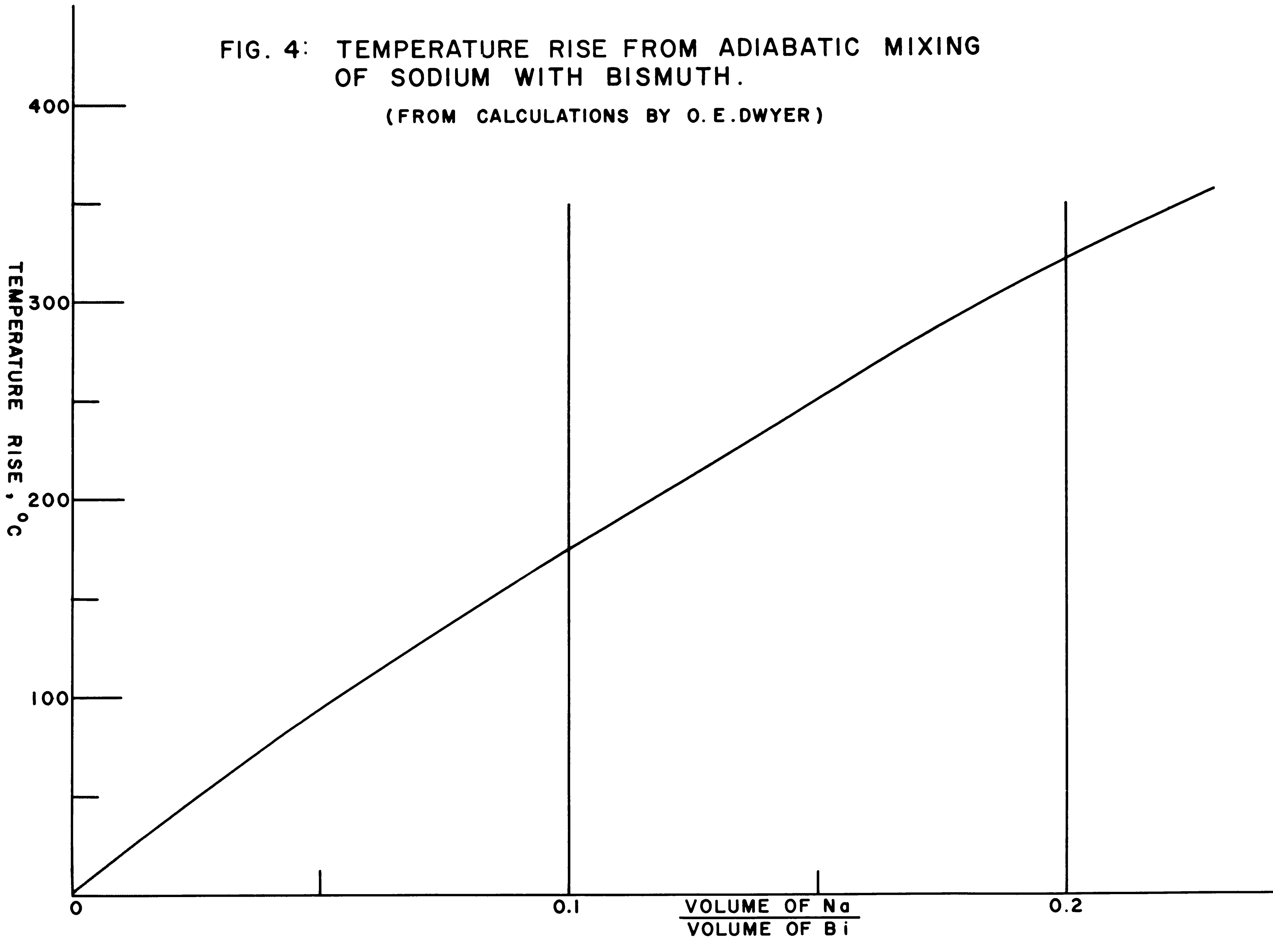
The bismuth clean-up process will remove sodium from bismuth; if some dissolved sodium is found tolerable in the fuel, operation with small leaks may be possible since continuous sodium removal appears feasible.

d. Oxidation of Uranium from U-Bi Fuel

Uranium oxidation from the fuel not only poses the problem of uranium losses, but also provides the possibility for introducing varying amounts of uranium to the core. Since the sudden presence of excess core uranium, as might result from a slug of uranium oxide, could seriously damage the reactor, the nature of uranium oxides and nitrides which might form in the loop will be studied. Normally, large amounts of uranium oxide or nitride will not form in the fuel because of the magnesium and zirconium gettering action.

FIG. 4: TEMPERATURE RISE FROM ADIABATIC MIXING OF SODIUM WITH BISMUTH.

(FROM CALCULATIONS BY O. E. DWYER)



(1) Reaction at a Leak

Since the primary loop will be at a slightly elevated pressure, fuel will flow outward from a leak. If air is present outside the leak, uranium and fuel additives will react. If a plug should be produced in a small leak or if a leak is very small, gas may diffuse into the fuel and react in the primary loop.

Small scale equipment will be constructed to observe fuel and air reactions at small leaks.

(2) Reaction at a Large Interface

The gas space in the reactor and degasser may be operated under slightly negative pressure which makes the introduction of air to the fuel a possibility at these points. The effects of such air-fuel contact will be examined.

e. Rates of Solution of Additives in Bismuth

Uranium, zirconium, and magnesium may be added to the LMFRE fuel stream by physical, chemical, or electrochemical methods. Assuming no uranium recovery by chemical processing, the simplest addition method will be direct solution of the metals in the fuel.

Adding uranium to the fuel during reactor operation will require that uranium be dissolved under conditions near saturation. These conditions will produce the lowest uranium dissolution rate and will be the first examined experimentally.

Magnesium and zirconium solution rates are also required for the design of fuel preparation equipment.

3. Fission Gas Handling in Radiation Test Loops

Schemes for handling fission gases from ETR loop (No. 2) have been studied. This loop is cooled with an air heat exchanger and is contained in an air envelope. Under certain emergency conditions, air will be circulated through the envelope to provide cooling. Safe disposal of these air streams must be provided under any leakage conditions.

As soon as a leak is detected the loop will be dumped and the air through the cooler turned off. If the highly improbable accident involving a plug and double leak occurs simultaneously in the in-pile section, contaminated air flow through the envelope cannot be discontinued, necessitating quick reactor shut-down.

The following are recommendations for the loop gas handling design:

a. The U-Bi solution should be degassed continuously to maintain a low concentration of the longer-lived gaseous isotopes. The efficiency of such degassing is uncertain; however, the efficiency will certainly be much greater than would occur at a leak.

b. Off-gas will be held on charcoal beds before disposal to permit sufficient radioactive decay. This will require about 20 cubic feet of charcoal at room temperature or about 1 cubic foot at -50 C.

c. A large bypass charcoal bed with auxiliary cooling and pumping may be advisable for use during the shut-down of contaminated circulating air systems.

IV. RESEARCH AND DEVELOPMENT

A. MATERIALS TESTING

1. E-1316, 1317 - Dynamic Test Loops (W. Markert, Jr.)

The four Croloy 2-1/4 test loops reported under construction in the last quarterly report have been completed except for minor modification to two of the pumps. This modification consists of installing an oil collector cup under the bottom pump bearing to prevent the bearing lubricating oil from leaking into the bismuth. (For a more complete explanation of this alteration, see section B,1.)

Two of these loops have been normalized and tempered and the other two are annealed. The test conditions for these four loops (Table III) were agreed upon at a meeting in the Lynchburg AED office on September 18, 1957.

TABLE III

TEST CONDITIONS - DYNAMIC TEST LOOPS

Loop	Heat Treat.	Clean Proc.	Precondi-tioning	U ppm	Mg ppm	Zr ppm	Temp F	T F	Vel. FPS
2	N & T	Acid	Zr 175 ppm Mg 350 ppm Temp 1000 F	1150 50	350	175 25	885	135	8
3	N & T	Alcohol	Zr 175 ppm Mg 350 ppm Temp 1000 F	1150 50	350	175 25	885	135	8
4	Anneal	Alcohol	Zr 0 ppm Mg 350 ppm Temp 1000 F	1150 50	350	0	885	135	8
5	Anneal	Alcohol	Zr 175 ppm Mg 350 ppm Temp 1000 F	1150	350	175	885	135	8

Loop 3 has been started on its preconditioning run. Loops 2, 4, and 5 will be started at approximately one week intervals.

The actual test run will begin after the loop has been preconditioned. The definition of "preconditioning" for these four loops is "to circulate bismuth with the prescribed additives except uranium until an equilibrium concentration has been reached or, if zirconium is not one of the preconditioning additives, until an equilibrium magnesium concentration has been reached. At this time, the bismuth should be dumped from the loop, new bismuth charged into the system, and a test run started with additives, including uranium."

Fourteen bismuth pumps were returned to the Deming Pump Company. These will be brought up to the latest development stage as determined by testing in the utility loop. The bulk of this work consists of installing the oil collector cups. The first of these pumps are due back at the rate of one per week, starting about the first week of October.

The zirconium concentrations reported in Table III have been reduced considerably from the original specification. This is based on Brookhaven's recent data on the solubilities of zirconium, magnesium, and uranium.

A change in the heat treatment procedures from that reported earlier is now in effect and concerns only the normalized and tempered tests. The original method was to fabricate the entire test loop and then place the entire unit (without pump) in the furnace for normalizing and tempering. Loops 2 and 3 were fabricated in this manner. All other normalized and tempered tests (original specifications) in both Croloy 1-1/4 and Croloy 2-1/4 are being made up of pipe and tubing that has been previously normalized and tempered. The entire test loop (except the pump) is then placed in the furnace for stress relieving all of the welds. This is more in line with standard procedures.

Construction and wiring of the remainder of the new loops is continuing. A total of 16 loops (piping only) have been constructed and heat treated. Two more normalized and tempered loops are under construction. No new loop fabrication will be started until a loop cleaning method is resolved. (If the loops are sandblasted, this must be done before manufacturing is started.) It is planned to continue rolling, heat treating, and machining the plate for heads, tube sheets, etc.

2. E-1318 - Static and Capsule Tests (W. Markert, Jr.)

All necessary test material and equipment are on hand and assembled for the tilting capsule program, including the drybox for inert welding that was delivered this quarter. Work has progressed on different cleaning procedures for both additives and capsules; present results indicate that variations in weight change of high magnitude are obtained when capsules are cycled under similar conditions of surface preparation and additive additions. To further evaluate this variation of results and achieve consistent changes, seven screening capsules were prepared with special additive cleaning, capsule preparation and loading procedures all conducted in an identical manner. This comparative testing was done on Croloy 2-1/4 material (normalized at 1700 F, tempered at 975 F, followed by sandblasting all components).

Each unit was filled with 450 grams of bismuth, 1000 ppm uranium, 350 ppm zirconium, and 350 ppm magnesium. These additives had been chemically cleaned and dried before weighing and loading in the drybox. The findings (Table IV, test 5) show that weight changes of specimens were closely comparable and of an order to permit further work under similar operating procedures.

Other methods of comparative work were performed using similar loading and cleaning procedures, but different surface preparation and capsule evacuation procedures. The results are shown in Table V, test 6. The differences in corrosion rates between tests 5 and 6 are due to varying the surface and metallurgical preparations from those used in test 5. The two tests of this group with a surface preparation and operation identical to test 5 resulted in comparable weight changes. It is recognized that the surface preparation under study will affect the corrosion rate of Croloy 2-1/4 specimens to some degree, and that identical preparations and procedures result in comparable weight changes.

Two more capsule units were cycled to determine if corrosion shown by specimen weight changes occurs during the solubility period of additives or after additives are in solution. Test 10 was conducted to determine this condition; capsule 89 was cycled in the normal manner, and capsule 90 was run isothermally at 975 F for 48 hours with cold end down before being placed on the test table for cycling. The findings (Table VI, test 10) indicate no appreciable differences, and the conclusion is that the solubility period does not affect the corrosion rate of Croloy specimens.

Metallurgical examination of these and other completed tests shows little oxide present on tube wall surfaces, but specimens with different weight losses appear to have similar surface irregularities and corrosion areas. It is rather difficult to determine accurately, even when comparing with similarily cleaned blanks, whether these irregularities are due totally to cleaning methods or solution attack. Tests are being set up to study capsules with honed internal surfaces and electrolytically polished specimens. These tests should establish basic standards from which the Metallurgy Section can evaluate the attack of liquid fuel solutions.

Work started on the statistical program as shown in Table VII, those units marked being considered first.

TABLE IV

SUMMARY OF TILTING CAPSULE TEST 5

CAPSULE NUMBER	TOTAL CYCLES & HOURS	MAT'L CROLOY	PREPARATION	GRAMS WT. CHANGE AT 975 F END	Mg/dm^2 975 F END	GRAMS WT. CHANGE AT 750F END	Mg/dm^2 750 F END	GRAMS SOLUTION 450 Bi	DESCRIPTION OF SPECIMEN AND INSIDE CAPSULE SURFACES *	REMARKS Chemical Analyses of bismuth solid ppm Mg	ppm Zr	ppm N
33	3109	$2\frac{1}{4}$	N-1700 T-975	-.0130	87.6	-.0128	86.5	1000ppmU 350ppmMg 350ppmZr	All hot end specimens appear coated and wetted.	405	215	975
	259		Sandblasted	-.0124	83.5	-.0176	118.2					
34	3109	$2\frac{1}{4}$	"	-.0129	87.0	-.0131	88.5	"	Cold end specimens were coated not adhering.	240	265	663
	259		"	-.0102	68.8	-.0119	80.5					
35	3109	$2\frac{1}{4}$	"	-.0156	105.0	-.0127	85.7	"	Capsule walls were wetted on hot & cold ends.	240	240	751
	259		"	-.0181	121.0	-.0036	Specimen broke loose					
36	3109	$2\frac{1}{4}$	"	-.0140	94.4	-.0109	73.6	"		335	280	988
	259		"	-.0122	82.3	-.0132	89.0					
37	3109	$2\frac{1}{4}$	"	-.0133	89.8	-.0103	69.5	"		370	245	975
	259		"	-.0148	99.8	-.0155	104.4					
38	3109	$2\frac{1}{4}$	"	-.0137	92.5	-.0139	93.7	"		335	235	933
	259		"	-.0156	105.0	-.0097	65.4					
39	3109	$2\frac{1}{4}$	"	-.0136	91.7	-.0126	85.0	"		160	230	663
	259		"	-.0127	85.6	-.0118	79.6					

* THESE DESCRIPTIONS ARE FROM VISUAL EXAMINATION

TABLE V

SUMMARY OF TILTING CAPSULE TEST 6

CAPSULE NUMBER	TOTAL CYCLES & HOURS	MAT'L CROLOY	PREPARATION	GRAMS WT. CHANGE AT 975 F END	Mg dm² 975 F END	GRAMS WT. CHANGE AT 750F END	Mg dm² 750 F END	GRAMS SOLUTION 450 Bi	DESCRIPTION OF SPECIMEN AND INSIDE CAPSULE SURFACES *	REMARKS
40	3719	$2\frac{1}{4}$	Sandblasted Evacuated	-.0335	226.0	-.0144	75.0	1000ppmU 350ppmMg 350ppmZr	Wetting-with bismuth adhering to capsule & specimens	Sandblasted-filled-sealed heated to 550 F
	310		N-1700 T-975	Saw	--	-.0065	43.7			
41	3719	$2\frac{1}{4}$	"	-.0548	369.0	-.0031	20.8	"	"	evacuated while under vacuum normalized &
	310			-.0511	344.4	-.0021	14.2			
42	3444	$2\frac{1}{4}$	N-1700 T-975	-.3055	2059.	-.0359	241.1	"	Adhering bismuth, no apparent wetting	tempered and cycled. Normalized, tem-
	286		Bi Plated	-.4327	2918.	-.0091	61.3			
43	3444	$2\frac{1}{4}$	"	-.0340	228.1	-.0072	48.5	"	"	pered bismuth plated-filled-sealed, cycles.
	286			-.0253	170.5	-.0282	190.0			
44	3116	$2\frac{1}{4}$	N-1700 T-975	-.1069	715.2	-.1025	690.8	"	No wetting No adherence	Normalized, tempered, acid cleaned, filled
	260		Acidcleaned	-.1043	703.98	-.1110	748			
45	3116	$2\frac{1}{4}$	"	-.1013	682.7	-.1000	674.	"	"	& sealed, cycled
	260			-.1057	712.4	-.1040	703.9			
46	3115	$2\frac{1}{4}$	Sandblasted Hydrogen fir	-.0042	28.	-.0024	16.2	"	No apparent wetting capsule clean sand blasted appearing.	Sandblasted, hydrogen fired thru normalize & tem
	263		N-1700 T-975	-.0022	14.8	-.0018	12.1			
47	3115	$2\frac{1}{4}$	"	-.0031	20.8	-.0019	12.8	"	"	per range filled sealed-cycled.
	263			-.0023	15.4	-.0022	14.8			
48	3088	$2\frac{1}{4}$	N-1700 T-975	-.0106	71.4	-.0085	57.2	"	Partial wetting adherence on cold end	Normalized & tempered, sand blasted filled
	257		sandblasted	-.0135	90.8	-.0093	62.5			
49	3088	$2\frac{1}{4}$	"	-.0073	51.8	-.0073	49.3	"	"	sealed, heated to 550 F, evacuated, cycled.
	257		"	drill.	-	-.0068	45.7			

* THESE DESCRIPTIONS ARE FROM VISUAL EXAMINATION

TABLE VI

SUMMARY OF TILTING CAPSULE TEST 10

CAPSULE NUMBER	TOTAL CYCLES & HOURS	MAT'L CROLOY	PREPARATION	GRAMS WT. CHANGE AT 975 F END	Mg dm^2 975 F END	GRAMS WT. CHANGE AT 750 F END	Mg dm^2 750 F END	GRAMS SOLUTION 450 Bi	DESCRIPTION OF SPECIMEN AND INSIDE CAPSULE SURFACES *	REMARKS
89	3100	2¼	As Rec'd	-.0081	54.5	-.0061	41	1500ppm uranium	Hot and cold end still sand blasted appear-	Cycled normal.
	259		Sandblasted	-.0045	30.3	-.0051	34.2	350 ppm Mg	ing slight adherences no wetting.	
90	3100	2¼	"	-.0050	33.6	-.0041	27.6	250ppm Zr	"	Cold end down isothermal at 975 F for 48 hours.
	259		"	-.0046	30.8	-.0048	32.2			

* THESE DESCRIPTIONS ARE FROM VISUAL EXAMINATION

TABLE VII

TILTING CAPSULE PROGRAM

			AS RECEIVED			NORMALIZE 1700 F TEMPER 975 F			NORMALIZE 1700 F TEMPER 1300 F		
		Fission Levels	Sand blast	Acid Clean	Bi Plate	Sand blast	Acid Clean	Bi Plate	Sand blast	Acid Clean	Bi Plate
BISMUTH 1500PPM URANIUM	350PPM Mg 250PPM Zr	0	C	C	C	C	C	C	C	C	C
		1	S	S	S	S	S	S	S	S	S
		2	S	S	S	S	S	S	S	S	S
	350PPM Mg	0	C	S	S	S	C	S	S	S	C
		1									
		2									
	1000PPM Mg	0	S	S	C	C	S	S	S	C	S
		1									
		2									
	350PPM Mg 1250PPM Zr	0	S	C	S	S	S	C	C	S	S
		1									
		2									
	1750PPM Mg 250PPM Zr	0	C	S	S	S	C	S	S	S	C
		1									
		2									
	1750PPM Mg 1250PPM Zr	0	S	S	C	C	S	S	S	C	S
		1									
		2									
	250PPM Zr	0	S	C	S	S	S	C	C	S	S
		1									
		2									
	1000PPM Zr	0	C	S	S	S	C	S	S	S	C
		1									
		2									
		0									
		1									
		2									
		0									
		1									
		2									
		0									
		1									
		2									

S - Scheduled C - Completed

Table VIII, test 8, which summarizes findings on the first group of capsules, shows that the weight differences vary on the capsule tests but are closely comparable within the individual capsules. It is felt that these conditions will level off after a period of time and that longer term tests should be made to evaluate these weight changes.

Since there is no definite indication which cleaning method or additive tested gives the best results, and in order to investigate all phases more rapidly, twenty more capsule stations are being planned.

Table IX outlines the longer term test periods and the proposed program, of which portions will be investigated early next quarter.

3. E-1281 - Chemical Analyses (W. A. Keilbaugh)

a. Wet Chemical Methods

The objectives of the analytical program for LMFRE are two fold.

(1) The development and testing of chemical procedures for the following requirements:

(a) The chemical control of additives

(b) Analyses of samples received from capsule testing and solubility studies.

(c) Determination of the extent of corrosion by analyses of corrosion products.

(d) Miscellaneous testing of materials employed in the LMFRE program.

(e) Analyses of fission products.

(2) The training of technical personnel in the use of chemical methods as they are developed for routine analyses.

The following procedures have been tested.

(a) Uranium - An additional and more rapid method for uranium, other than the previously reported dibenzoylmethane procedure, has been tested and proven satisfactory. This method employs a series of extractions using the final 8-quinolinol-chloroform extraction for color formation and spectrophotometric determination. While this method does not have the sensitivity of the dibenzoylmethane procedures for very low concentrations, it does not require the dilutions necessary for the more sensitive procedure. On the higher concentration level (1000 ppm), this more rapid procedure appears to have equivalent or better accuracy than the dibenzoylmethane method.

TABLE VIII

SUMMARY OF TILTING CAPSULE TEST 8

CAPSULE NUMBER	TOTAL CYCLES & HOURS	MAT'L CROLOY	PREPARATION	GRAMS WT. CHANGE AT 975 F END	Mg/dm^2 975 F END	GRAMS WT. CHANGE AT 750F END	Mg/dm^2 750 F END	GRAMS SOLUTION 450 Bi	DESCRIPTION OF SPECIMEN AND INSIDE CAPSULE SURFACES *	REMARKS
64	3100	2¼	As Rec'd	-.0031	20.9	-.0023	15.5	1500ppmU 350ppmMg 250ppmZr	Partial wetting on capsule and specimen	
	259		Sandblasted	-.0025	16.8	-.0040	27.0			
65	"	"	"	-.0179	120.5	-.0339	22.8	"	Bismuth adhering, no wetting, cold end appears deposited shut	
	"		"	-.0167	112.5	-.0326	22.0			
66	"	"	As Rec'd	-.0076	51.1	-.0069	46.5	"	No wetting - still dark appearing as when started. Cold and appears heavily deposited	
	"		Acid clean	-.0026	17.5	-.0062	41.8			
67	"	"	"	Saw		-.0138	93.0	"		
	"		"	Saw		-.0475	32.0			
68	"	"	As Rec'd	-.0109	73.5	-.0163	110	"	No wetting, still dark appearing as when ~~started. Cold end~~ lightly deposited	
	"		Bi plated	-.0117	78.5	-.0109	73.5			
69	"	"	"	-.0117	78.5	-.0103	69.3	"	good wetting on cap- ~~sule & specimens~~ coated.	
	"		"	-.0152	102.0	-.0098	66.0			
70	"	"	N-1700 T-975	-.0040	27.0	-.0019	12.8	"		
	"		Sandblast	-.0031	20.9	-.0024	16.2			
71	"	"	"	-.0019	12.8	-.0024	16.2	"		
	"		"	-.0030	20.2	-.0031	20.9			
72	"	"	N-1700 T-975	-.0028	18.8	-.0026	24.3	"	No wetting No change	
	"		Acid clean	-.0017	12.4	-.0033	22.3			
73	"	"	"	-.0011	7.4	-.0056	37.7	"		
	"		"	-.0018	12.1	-.0052	35.0			
74	"	"	N-1700 T-975	-.0053	35.6	-.0088	59.2	"	No wetting No change	
	"		Bi Plate	-.0044	29.6	-.0061	41.0			
75	"	"	"	-.0058	39.0	-.0022	14.8	"	good wetting capsule & specimen	
	"		"	-.0037	25.0	-.0025	16.8			
76	"	"	N-1700 T-1300	-.0062	41.7	-.0023	15.3	"	"	
	"		Sand blast	-.0062	41.7	-.0019	12.8			

* THESE DESCRIPTIONS ARE FROM VISUAL EXAMINATION

TABLE VIII (CONT'D.)

SUMMARY OF TILTING CAPSULE TEST 8 Cont'd.

CAPSULE NUMBER	TOTAL CYCLES & HOURS	MAT'L CROLOY	PREPARATION	GRAMS WT. CHANGE AT 975 F END	Mg/dm² 975 F END	GRAMS WT. CHANGE AT 750 F END	Mg/dm² 750 F END	GRAMS SOLUTION 450 Bi	DESCRIPTION OF SPECIMEN AND INSIDE CAPSULE SURFACES *	REMARKS
77	3100	2¼	N-1700 T-1300	-.0067	45.2	-.0038	25.6	1500ppmU 350ppmMg 250ppmZr		
	259		Sandblast	-.0087	58.5	-.0033	22.3			
78	"		N-1700 T-1300	X		X		"	Failure due to leak	
	"		Acid clean	X		X				
79	"		"	-.0058	39.1	-.0060	40.5	"	No wetting no change	
	"		"	-.0028	18.2	-.0028	18.9			
80	"		N-1700 T-1300	-.0063	42.5	-.0054	36.4	"	No wetting	
	"		Bi Plate	-.0051	34.4	-.0033	22.2			
81	"		"	-.0087	58.5	-.0080	54.0	"	No change	
	"		"	-.0093	62.8	-.0070	47.2			

* THESE DESCRIPTIONS ARE FROM VISUAL EXAMINATION

TABLE IX

TILTING CAPSULE PROGRAM

			Time Hrs.	As Rec'd. SB	As Rec'd. AC	Normalize 1700 Temper 975 SB	Normalize 1700 Temper 975 AC	Normalize 1700 Temper 1300 SB	Normalize 1700 Temper 1300 AC	As Rec'd Hard	Sandblast Hydrogen Fire N1700 T975	Sandblast Hydrogen Fire N1700 T1300	Sandblast Hydrogen Fire 975	As Received Hydrogen Fire N1700 T975	As Received Hydrogen Fire N1700 T1300	As Received Hydrogen Fire 975
Bi 1150 ppm uranium	350 ppm Mg 175 ppm Zr	0	250	C	C	C	C	C	S	C	S	S	S	S	S	S
			500	C	C	C	C	C	S	C	S	S	S	S	S	S
			1000	C	C	C	C	C	S	C	S	S	S	S	S	S
			2000	C	C	C	C	C	S	S	S	S	S	S	S	S
	0	0	250													
			500													
			1000													
			2000													
Bi 1150 ppm uranium	350 ppm Mg	100 ppm Zr	250	S	S	S	S	S	S		S	S	S	S	S	S
			500	S	S	S	S	S	S		S	S	S	S	S	S
			1000	S	S	S	S	S	S		S	S	S	S	S	S
			2000	S	S	S	S	S	S		S	S	S	S	S	S
		50 ppm Zr	250													
			500													
			1000													
			2000													
Bi 1000 ppm uranium	350 ppm Mg 175 ppm Zr	0	250	S	S	S	S	S	S							
			500	S	S	S	S	S	S							
			1000	S	S	S	S	S	S							
			2000	S	S	S	S	S	S							
	0	0	250													
			500													
			1000													
			2000													
Bi 1000 ppm uranium	350 ppm	100 ppm Zr	250	S	S	S	S	S	S							
			500	S	S	S	S	S	S							
			1000	S	S	S	S	S	S							
			2000	S	S	S	S	S	S							
		50 ppm Zr	250													
			500													
			1000													
			2000													

S - Scheduled C - Completed O - Operating

(b) Magnesium - Satisfactory reproducibility was not obtained on the colorimetric procedure and at present all magnesium results are obtained by spectrographic means. Further work is planned, possibly using flame photometric procedures.

(c) Zirconium - The determination of zirconium, using a metal reductant for bismuth removal and quercetin for color formation, has proven satisfactory and is being used primarily as a check on the more rapid spectrographic analyses.

(d) Iron - The testing of this procedure has been completed. Initial work was hampered by high blanks, but this problem has been resolved. The method has been used by technical personnel with good reproducibility and sensitivity.

(e) Manganese and Chromium - The testing of methods for these corrosion products is still in progress. The desired reproducibility has not been achieved in the chromium procedure.

b. Spectrographic Methods

(1) Zr and Mg in Bismuth

Due to erratic behavior of the spectrographic determination of Mg and Zr in bismuth from the bismuth pump loops and the tilting capsule tests, an investigation was made to improve the method.

The Mg/Bi and Zr/Bi intensity ratios were found to be dependent on the bismuth-HNO_3 solution concentrations. Increasing dilution gave rise to increased Zr/Bi intensity ratios, but a decrease in over-all line intensities. New standard solutions containing 10 percent Bi in HNO_3 replaced the 20 percent solution previously used, and new working curves were developed. Other aspects of the analytical procedure are still being investigated.

(2) Determination of Rare Earths as Fission Products

The standard samples prepared by adding excessive yttrium as a carrier and internal standard and high cross-sectional rare earths were analyzed and the R.E./Y line intensity ratios correlated with concentration. Tentative analytical working curves were prepared. Based on original sample size, the range covered 0.02-1 ppm of the individual rare earth.

4. E-1292 - Zirconium Nitride Theory (F. Eberle)

The zirconium-nitride film theory postulates that corrosion of the container material by liquid bismuth is inhibited through the formation of a protective film of zirconium nitride.

The nitrogen available for this film formation is present as acid-soluble nitrogen in the container material; the zirconium is added to the molten bismuth.

It is important to know the amount of nitrogen normally present in steel for this film formation. It is also important to know in what form nitrogen is present since only that nitrogen which is not tied up or "fixed" as stable nitrides is available for diffusion to the metal surface in contact with zirconium-containing liquid bismuth and ZrN film formation.

The "available" nitrogen is the so-called "acid soluble" nitrogen. Nitrogen generally is not determined when steel is analyzed, and there was consequently little information available on this constituent.

Aluminum as "acid soluble" aluminum is also important in this investigation as this element, when contained in solid solution, will combine with nitrogen to form a stable nitride. Like nitrogen, aluminum generally is not determined when steel is analyzed so little information was available on aluminum concentration.

Samples of 21 random commercial heats of Croloy 2-1/4 were obtained. These samples were quenched from 1800 F to retain all nitrogen in solution and thereby determine the maximum amount which could be made available by suitable heat treatment. Exposure to lower temperatures, such as pre- and post-weld treatments or the proposed operating temperatures, results in partial precipitation of nitrogen in the steel in form of nitrides, thereby reducing the amount available for film formation. The effect of such exposures to lower temperatures on the acid-soluble "available" nitrogen was studied with samples of three representative heats which were held for 24 hours at 750 F, 850 F, 950 F, 1050 F, 1300 F, and 1350 F to precipitate nitrogen in the form of nitrides. Chemical analyses for acid-soluble nitrogen will indicate the nitrogen available for protective film formation.

Table X gives the analyses of the 21 heats in the quenched condition for both soluble and insoluble nitrogen and aluminum. The determination for nitrogen and aluminum in the three representative heats after exposure to various temperatures is in progress.

5. E-1343 - Metallurgical Studies (F. Eberle)

a. Mechanical Properties of Croloy 1-1/4 and 2-1/4 as Affected by Heat Treatment.

The availability of nitrogen for forming a protective ZrN film, on whose presence the corrosion resistance of Croloys 1-1/4 and 2-1/4 in liquid bismuth is assumed to rest, depends to a significant extent on the heat treatment of the material. The latter must be chosen so that the nitrogen in the steel will not be tied up as stable nitride compounds and thereby become unavailable for diffusion and film formation. The commercially employed heat treatment for these materials does not promote nitrogen availability and must be modified for this purpose. In this respect normalizing treatments are more effective than the heat treatments commercially in use. Since heat treatment modifications change the mechanical properties, the latter must be determined to insure their adequacy for construction purposes.

Furthermore, any additional thermal treatments imposed upon the materials in fabrication, such as pre-heating and post-weld treatments, may likewise change the mechanical properties significantly. Two heat-treatment processes have been chosen for the LMFRE.

(1) An isothermal anneal, consisting of heating the material to 1610 F, holding it at 1610 F for 45 minutes, and then transferring the material to a furnace zone held at 1200 F, where it remains for 20 minutes before it is cooled in air. The effect of this heat treatment on the mechanical properties of the material at the elevated temperatures involved in the construction of an LMFR is not known.

(2) A normalizing treatment, consisting of heating the material to above the A_3 transformation temperature, followed by cooling in air. The thus air-hardened material must then be subjected to a tempering treatment which, for obtaining best nitrogen availability, should be as low as possible. The choice of a suitable tempering treatment is a matter of experimentally exploring its effect on the mechanical properties. It is thus obvious that the mechanical properties of these alloys as affected by non-conventional heat treatments, must be known. Their determination forms the object of some of our studies. The material used for this purpose consists of tubing from the two special commercial heats of Croloy 1-1/4 and 2-1/4 made for LMFRE.

The studies which have been carried out so far and the results obtained are shown in Table XI indicating that no objectionable properties are apparent.

b. Stress Rupture Tests in Bismuth

The procedure for testing Croloy 1-1/4 and Croloy 2-1/4 in a liquid metals atmosphere was discussed briefly in the previous quarterly report.

TABLE X

CROLOY 2-1/4 TUBING

Heat No.	Total Al	Acid-Sol. Al	Acid-Insol. Al by Difference	Total N by Addition	Acid-Sol. N	Acid-Insol. N
31319	0.019	0.020	-0.001	0.012	0.012	0.000
23407	0.038	0.027	0.011	0.012	0.012	0.000
11664	0.028	0.019	0.009	0.011	0.011	0.000
922	0.021	0.019	0.002	0.011	0.010	0.001
42835	0.017	0.010	0.007	0.012	0.012	0.000
31769	0.029	0.020	0.009	0.011	0.011	0.000
32478	0.029	0.020	0.009	0.014	0.014	0.000
11666	0.014	0.012	0.002	0.015	0.014	0.001
23100	0.028	0.020	0.008	0.015	0.014	0.001
43032	0.012	0.008	0.004	0.013	0.013	0.000
23317	0.036	0.038	-0.002	0.016	0.016	0.000
33134	0.021	0.020	0.001	0.012	0.012	0.000
23270	0.026	0.029	-0.003	0.012	0.012	0.000
11846	0.019	0.007	0.012	0.016	0.016	0.000
33240	0.027	0.017	0.010	0.015	0.015	0.000
42858	0.011	0.007	0.004	0.014	0.014	0.000
44212	0.021	0.020	0.001	0.013	0.012	0.001
33693	0.015	0.014	0.001	0.013	0.013	0.000
33694	0.008	0.004	0.004	0.010	0.010	0.000
33692	0.016	0.012	0.004	0.012	0.012	0.000
23405	0.028	0.028	0.000	0.013	0.013	0.000

TABLE XI

MECHANICAL PROPERTIES

CROLOY 2-1/4 TUBING

Heat Treatment: 1610 F for 45Min. - 1200 F for 20 Min./Air-Cool

Test Temp. F	Yield Strength	Ultimate Strength	% Elongation in 2"	Reduction of Area, %
R.T.	53,800	76,400	18.5	77.2
400	48,200	70,600	14.0	74.8
500	46,900	70,800	14.0	73.7
600	45,700	76,000	12.0	69.8
700	44,200	78,600	14.0	68.4
800	41,500	73,400	16.0	72.8
885	41,000	69,200	14.5	74.5
900	41,200	68,300	14.5	74.8
975	39,700	62,200	16.0	77.2

TABLE XI (Con't.)

MECHANICAL PROPERTIES

CROLOY 1-1/4 TUBING

Heat Treatment: 1610 F for 45 Min. - 1200 F for 20 Min./Air-Cool

Test Temp. F	Yield Strength	Ultimate Strength	% Elongation in 2"	Reduction of Area, %
R. T.	50,200	73,400	27.5	75.6
400	44,000	68,600	21.0	74.4
500	40,800	70,400	20.0	73.0
600	38,100	73,400	18.0	70.8
700	35,200	74,400	23.0	72.0
800	34,800	70,200	24.5	76.0
885	31,000	65,000	23.0	76.4
900	31,800	63,400	23.0	77.8
975	30,400	57,000	25.0	78.5

The test circuit is shown as a block diagram in Figure 5.

(1) General Test Procedure

Test procedure consists of continually evacuating Circuit 'A' (Figure 5) throughout the heating cycle, but interrupting the cycle to pull a maximum vacuum at 200 F, 400 F, 600 F, and 885 F (test temperature) or 800 F and 975 F. After a maximum vacuum has been obtained the vacuum portion of Circuit 'A' is cut out and the gas circuit 'B' is cut in and maintained for the duration of the test.

Circuit 'A', the vacuum side, consists of a thermocouple vacuum gauge, a cold trap to cool hot gases, a B-1 diffusion pump, and a Press-O-Vac mechanical pump. This circuit is designed to obtain a vacuum of five microns or less.

Circuit 'B', the gas side, consists of a gas source (tank of helium gas), dual gauge system to regulate pressure, a drying trap consisting of silica gel, a 1500 F furnace housing a trap containing zirconium chips, and a cooling coil to cool the gas prior to entering the test chamber. This system is designed to provide dry, clean helium gas to the test chamber under 2 pounds pressure.

(2) Progress on Auxiliary Equipment

All of the auxiliary equipment has been fabricated and is shown in Figure 6. The lower portion of the picture contains the vacuum circuit and the upper portion contains the gas circuit.

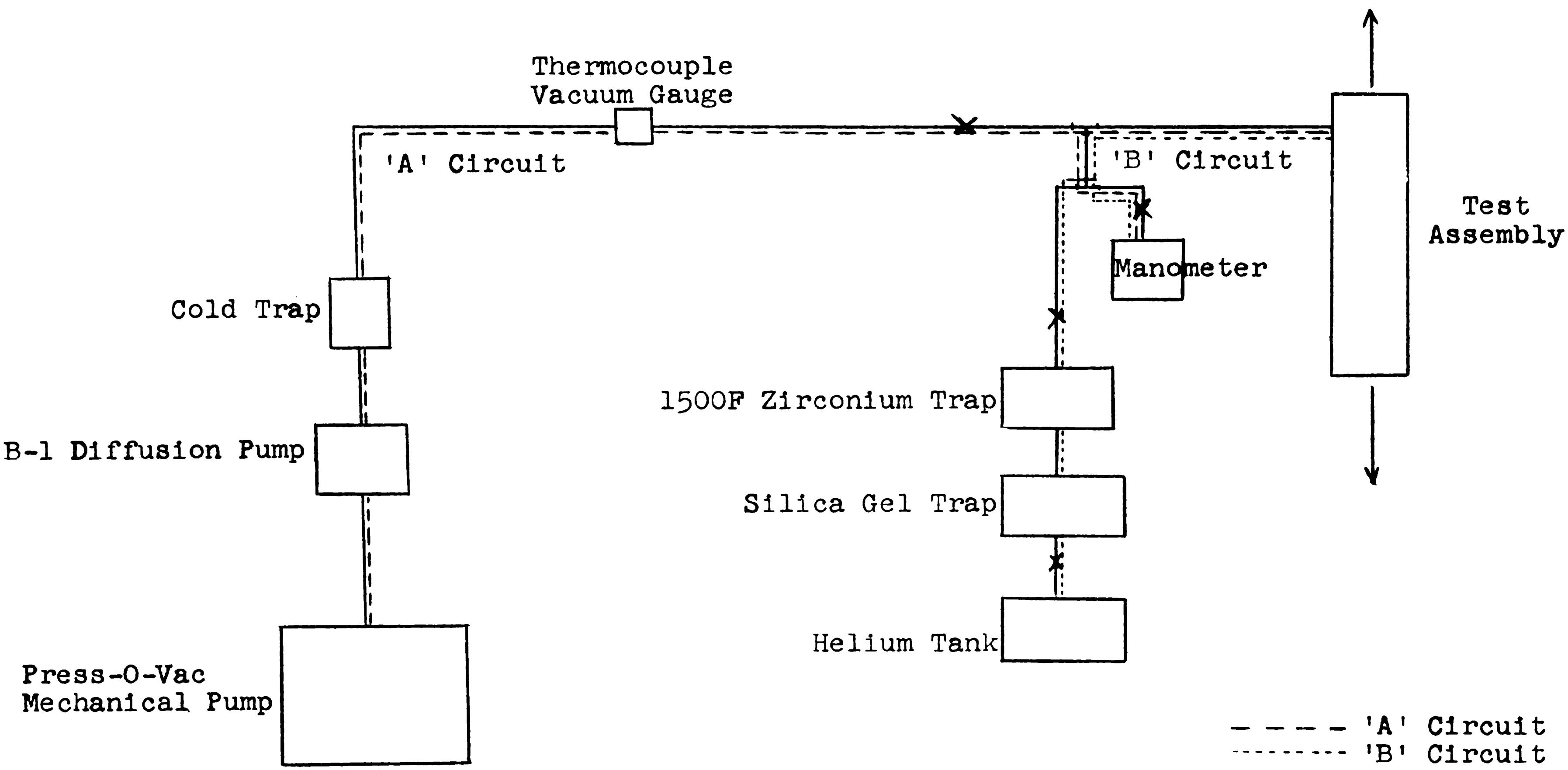
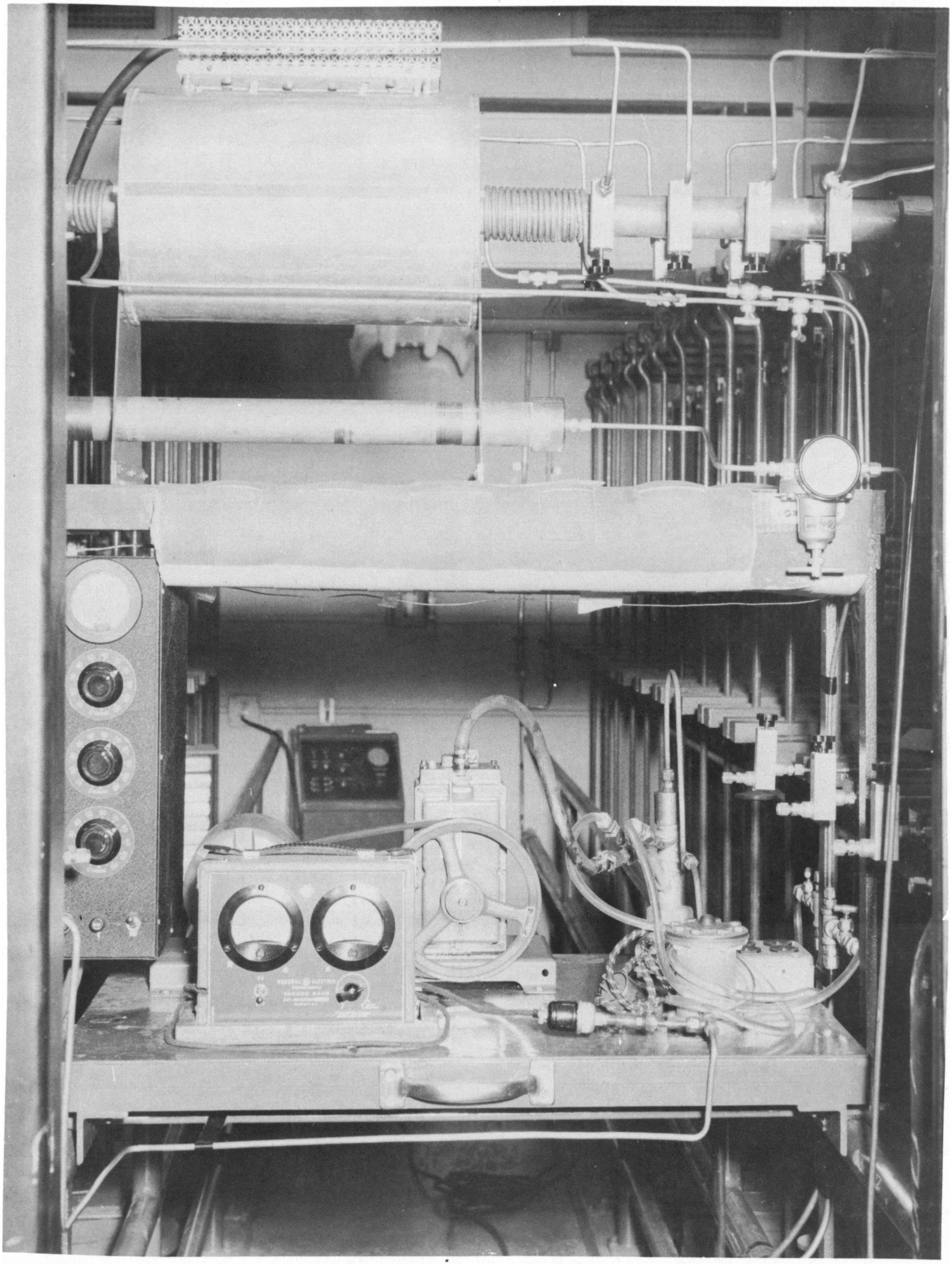


FIG. 5: BLOCK DIAGRAM OF LIQUID METALS STRESS-RUPTURE TEST

FIG. 6: AUXILLARY EQUIPMENT - STRESS RUPTURE TEST



(3) Test Assembly Procedure

All component parts of the test assembly shown in Figure 7 are made of the same material except the bellows which is 18Cr-8Ni in all cases. The bellows is located several inches above the liquid metal and outside the heating furnace and should have no influence on the test.

All welding on the test assemblies is done in a drybox to prevent oxide formation.

Prior to attaching the bellows, the metallic ingredients will be added and melted down in the drybox. The ratio of additives to be used is 1150 ppm uranium, 350 ppm magnesium, and 175 ppm zirconium.

(4) Progress on Test Assemblies

The first four assemblies are being welded and will be filled early in October. As soon as these four are assembled, a second set of four will be started through the assemblying process.

All material has been received except the high-speed multipoint recorder which will be shipped in October. However, this will not interfere with starting the test program.

B. PROTOTYPE TESTING

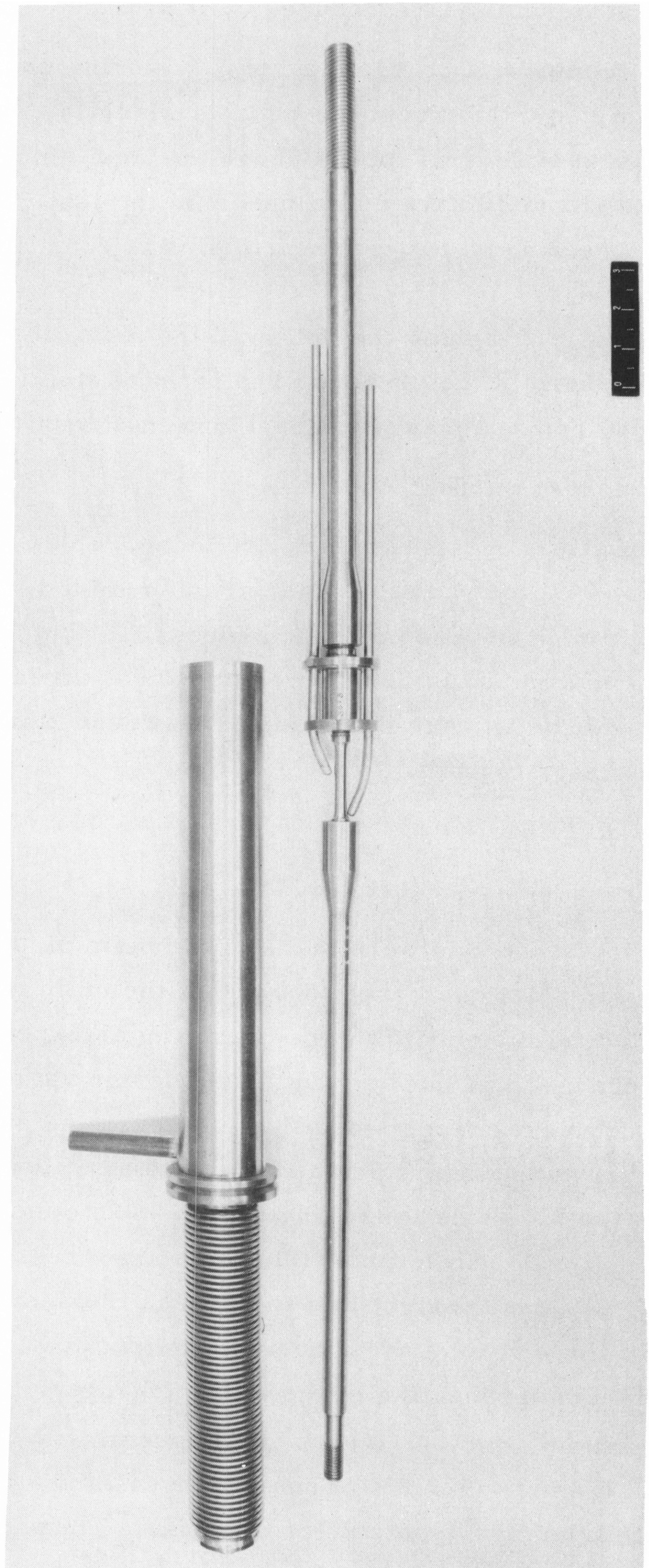
1. Utility Loop (W. Markert, Jr.)

Several modifications were made to the bismuth pump and test loop. It was stated in the last quarterly report that the main resistance-type heater and the NaK filled fin tube cooler were operating as expected. Only the regenerative heat exchanger had not been up to the design values.

The first heat exchanger modification consisted of reducing the clearance around the floating tube sheet since a large portion of the flowing bismuth leaked internally from the tube side to the shell side. The clearance reduction helped considerably but it was still not possible to obtain a reasonable heat balance on the exchanger. The next thought was to recheck the design calculations to determine if the floating tube sheet could be welded to the shell and not over stress the tubes upon preheating of the heat exchanger.

Upon start-up, the entire heat exchanger must be heated up beyond the melting point of bismuth (520 F). The exchanger is heated by Chromalox sheath type heaters strapped to the outside of the shell. Thus, in heating up, the shell is always hotter than the tubes inside, which places the tubes under a tensile stress. It was decided that this stress would not be prohibitive if the exchanger

FIG. 7: COMPONENT PARTS OF TEST ASSEMBLY - STRESS RUPTURE TEST



were heated up slowly enough. This condition would be the most severe that the heat exchanger would be subjected to.

The all-welded type exchanger has been installed in the utility loop and is undergoing tests now. At this writing, the overall heat transfer coefficient calculated from the test data is around 500 Btu/hr-ft^2 F/ft. A coefficient of 1000 was used in the original design. If this is the best transfer that can be obtained, it will be necessary to include more surface area in the loops that operate at 8-10 fps with a ΔT of 225 F. The present exchanger will satisfy the other test conditions providing the coefficient does not drop off with time. The heat transfer may increase with circulating time as wetting is promoted by the magnesium and zirconium additives in the bismuth.

A new pump was also installed in the utility loop during the report period. This pump utilized the U. S. Varidyne system for motor speed control to obtain the specified flow rate. This eliminated the Reeves Vari-Speed Drive at the pump which was subject to belt slipping. The new pump motor speed is controlled by a remote variable frequency supply which is a very satisfactory arrangement.

During operation of this pump, it was discovered that oil from the ball bearing lubricating system was leaking into the bismuth. The pump was removed and another unit installed. This pump has an oil collector cup below the bottom oil seal to contain any oil leakage. The oil collected in the container can be blown out, externally to the pump, by use of the 5 psi helium gas pressure kept inside the pump.

The modified pump has operated for a total of 825 hours with no trace of oil in the bismuth. Two hundred and fifty of those hours were at ΔT of 135 F. The flow is maintained at 8-10 fps.

Magnesium and zirconium have been added to the bismuth, and one or more bismuth samples are taken each work day and analyzed. These values seem to vary considerably. It is uncertain whether this variance is due to sampling techniques or sample analyses. Additional techniques are being tried to isolate and eliminate the variables. No uranium has been added to the loop.

An EM flowmeter was attached to this test loop. The millivoltage output varied with changes of bismuth flow which indicates that this type flowmeter would function satisfactorily if it was calibrated. In fact, the meter was more accurate in the reproduction of flow than the orifice and gas seal pots used on this test.

Future plans for the utility loop included completion of the heat exchanger test. After this, two one-inch stainless steel bellows valves will be installed in parallel in the piping for a small valve test for the in-pile test loops. This test will determine satisfactory operation of the valves from the mechanical standpoint only. It is also planned to install another shaft in the pump with an "Oilite" journal bearing in place of the standard ball bearing. This bearing may better eliminate the oil leakage problem since it is a bronze oil-impregnated bearing, requiring no additional lubrication. Other testing for this apparatus will consist of level indicators and additional flowmeter work.

2. <u>E-1370 Graphite-to-Metal Seal</u> (W. Markert, Jr.)

Ordinary commercial grades of graphite were screened in the testing apparatus and found to have no similarity to pile grade material. No seal could be made, since the bismuth absorption was as high as 47 percent and the leakage flow through the graphite was high.

Testing began on the impervious graphite and a seal was effected with an interface contact stress as low as 600 psi and a bismuth pressure of 100 psi. With an interface stress of 1600 psi, no trouble was experienced at any time allowing some latitude in surface finish of the graphite.

The impervious graphite, manufactured by Great Lakes Carbon Company and designated as MH4LM was found to have a thick impervious shell, about one-inch thick on all exposed surfaces. The casing or shell appears impervious to bismuth absorption or weepage. The relatively porous core absorbed two percent bismuth but allowed no weepage.

The samples tested are small, the exposure area to bismuth being only 2.8 square inches. Further work on larger specimens is necessary to obtain data for final design work.

Later tests, using argon gas to purge the graphite prior to pressure loading with bismuth, showed an increase in the permeability of the graphite. Work is in progress to enclose the functional parts of the screening apparatus for evacuation and degassing tests. The refurbished equipment should furnish pertinent data on permeation and weepage under controlled conditions.

Purchase orders for impervious graphite were sent from Lynchburg to three major graphite corporations. Delivery dates promised range from December 4, 1957 to February 1, 1958.

3. E-1371 - Dump Valves

E-1492 - Instruments & Controls (W. Markert, Jr.)

An apparatus has been designed to test the dump valve (2 1/2-in. size prototype) and other components which can be tested advantageously in this type of apparatus. Some of these are:

a. A check valve (2 1/2-in. size)
b. An EM pump
c. Various level controls
d. A 2 1/2-in. flange connection
e. Temperature and pressure indicators
f. A flowmeter (2 1/2-in. size)

The test arrangement consists of two tanks, with the upper tank located to maintain a 16.5-foot bismuth head on the dump valve. A pump will be used to return the bismuth to the upper tank. An EM pump has been suggested for this purpose but present plans are to use a Deming pump, similar to those used on the "Utility Loop", until delivery of the EM pump can be ascertained.

The entire unit will be heated by electrical resistance elements. Because of the fluctuating heating load of the different tanks and lines, four temperature controllers are required. The lower tank will be used as a melting tank since the quantity of bismuth to be used (12 cu ft) is much more than the capacity of existing melting facilities.

Helium will be used as a cover gas and will be maintained at a pressure just greater than atmospheric to prevent inward leakage. Leakage will be maintained at a minimum to reduce the amount of helium used and thus minimize contamination problems resulting from impurities in the helium.

Controls have been designed to provide automatic operation on cycling tests and interlocks for safety purposes.

4. E-1373 - Small Isolation Valves (W. Markert, Jr.)

One of the main problems concerning valve procurement for this program is that thin Croloy 2-1/4 strip is unavailable for bellows fabrication. A roll of Croloy 2-1/4 (4.850 in. wide x 0.140 in. thick) was obtained and half of it (400 lbs) cold rolled to 0.014 inch thickness. This material will be used to make the welded type bellows for small valves.

The "in-pile loops", (E-1369) which are in the design stage, will require a quantity of small one-in. valves. This will be the only test of valving in a radioactive loop before the LMFRE is put into operation. The tight schedule assigned this program prohibits developmental testing prior to installation of these loops. Therefore, it was decided to try to get the best commercial valves available to insure the successful operation of the tests which are highly dependent on the operation of these valves. Negotiations were made with the Wm. Powell Company to obtain one-in. valves of a "Y" pattern design. These valves would be made completely of Croloy 2-1/4 except for a Stellite 92 facing on the plug and seat. The valves are designed with a double bellows (Croloy 2-1/4) in a cascade arrangement as a safety feature in the event that the lower bellows (in contact with bismuth) fails. To prevent bismuth impingement, a can is provided around the lower bellows unit.

It is hoped that information from these tests will demonstrate whether the presently available small bellows valve (of good quality) is usable for isolation service on the LMFRE.

5. E-1368 - Beryllium "Thimbles" (M. Christensen)

We have explored the availability of beryllium in the tubular and rolled form desired for the LMFRE reactor and have studied existing information on the joining of beryllium. Apparently very little has been done on beryllium-to-beryllium fused joints and the information available is not encouraging. We have ordered a seven-pound lot of extruded beryllium pipe and plate for welding experiments. Arrangements are being made to insure proper safety in handling the toxicity problems.

C. INSTRUMENTATION

1. E-1335 - Continuous Uranium Monitoring System (W. A. Keilbaugh)

A three-day conference on the continuous monitoring problem was held between A.D. Little Company representatives, Lynchburg AED personnel, and Research Center representatives. Since the A.D. Little Company holds a contract to develop a method for solving this problem, the Research Center agreed to furnish them samples of uranium in bismuth, and steps were taken to prepare such samples.

We also agreed to furnish the A.D. Little Company with copies of our bibliography prepared from an extensive literature search.

2. E-1492 - Non-Nuclear Instrumentation (W. Markert, Jr.)

Various loops (Utility Loop and E-1371) will be utilized in carrying out work on engineering measuring and indicating instruments. Most of this quarter's work was directed toward the completion of test apparatus.

D. REMOTE MAINTENANCE

1. E-1280 - Remote Welding (M. Christensen)

The status of the three welding processes being investigated for remote maintenance application follows:

a. Gas Shielded Metallic-Arc Process

Investigation of the gas-shielded metallic-arc process continues for the automatic butt-welding of pipe.

Butt-welds in 6 1/2-in. OD x 5/8-in. wall carbon steel pipe have been made using a 1/16-in. diameter electrode and carbon dioxide as a shielding gas. Cross sections examined show some non-fused areas and fine porosity. Troubles encountered in welding were related to arc voltage control and difficulty in deposition of satisfactorily shaped weld beads for a full 360 degree travel. A constant-voltage type power source has been purchased for better control of the arc voltage, and the difficulties in bead shape have been partially eliminated by welding only in the vertical down position.

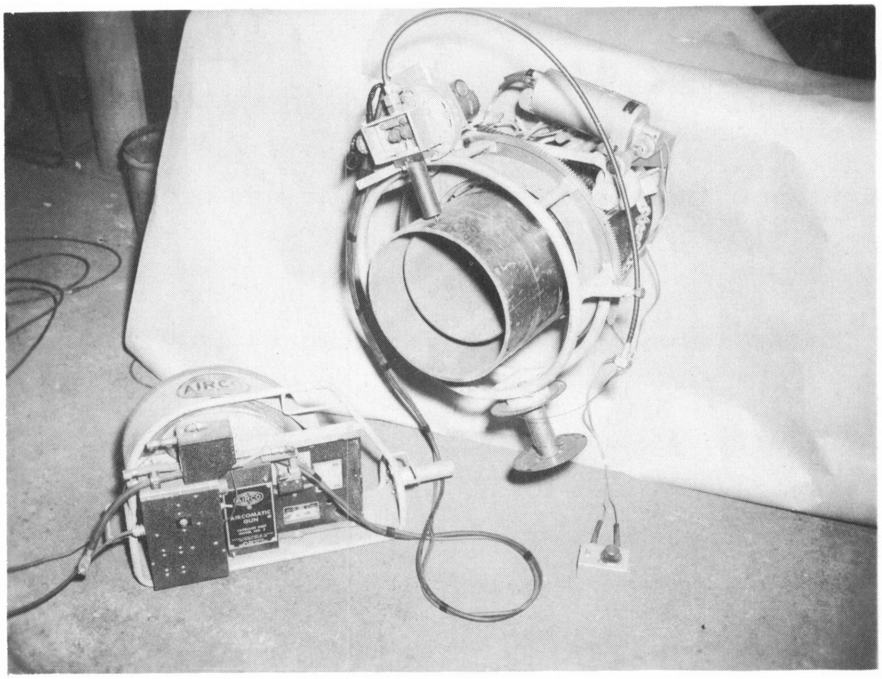
The present equipment for rotating the welding machine about the pipe consists of a turntable to which the wire feed motor is attached. To more closely study the weld problems related to remote control, a pipe beveling and cutting machine has been purchased from H&M Company. This apparatus is built to straddle the pipe to be welded. The welding head mounted on a split gear travels about the circumference of the pipe at desired speeds. The unit including motorized welding head, mounts, and wire reel is being assembled and is shown in Figure 8.

To prevent delaying the weld investigation while the new equipment is being designed, the old pipe traversing equipment with the constant voltage type power unit is being used.

Studies are being made on Croloy 2-1/4 pipe of 4 1/2-in. OD x 3/8-in. wall using 1/16-in. diameter wire of equivalent analysis. Figure 9 and 10 show the weld surface and face bend tests of a butt-welded joint made in pipe using a mixture of helium and argon gas for protection. These specimens, used for face bend tests, are in the as-welded condition and show limited ductility. Additional tests are being made to determine the effect of stress relieving.

Macro-sections examined from this weld continue to show non-fused areas and porosity.

FIG. 8: PIPE BEVELING AND CUTTING MACHINE



b. Tungsten Arc with Filler Wire

A machine designed to do position welding by this method can be purchased from Mr. L. C. McNutt of Wilmington, Delaware. A recently issued report describes this machine and the results of test welds made with it. The machine does a satisfactory job of contact welding and shows promise of remote operation.

c. Induction Forge Weld

A 300 KW high-frequency generator has been installed and is shown in Figure 11. This equipment is being used to obtain data on induction-forge butt welds. Inductor coils of four different designs have been built and are being tested for welding the butted ends of 3-in. schedule 80 pipe. These inductors differ in section and in diameter. They are being used in combination with different joint designs, heat cycles, and pressures to obtain data pertinent to induction welding. Welds made by this method are usually capable of satisfactory face bends. Satisfactory root bends have been obtained only in cases where there has been obvious excess heat causing a ragged upset on the inside of the pipe which would be objectionable in a fluid-carrying tube.

FIG. 9: WELD SURFACE OF SECTIONS FROM BUTT-WELDED JOINT

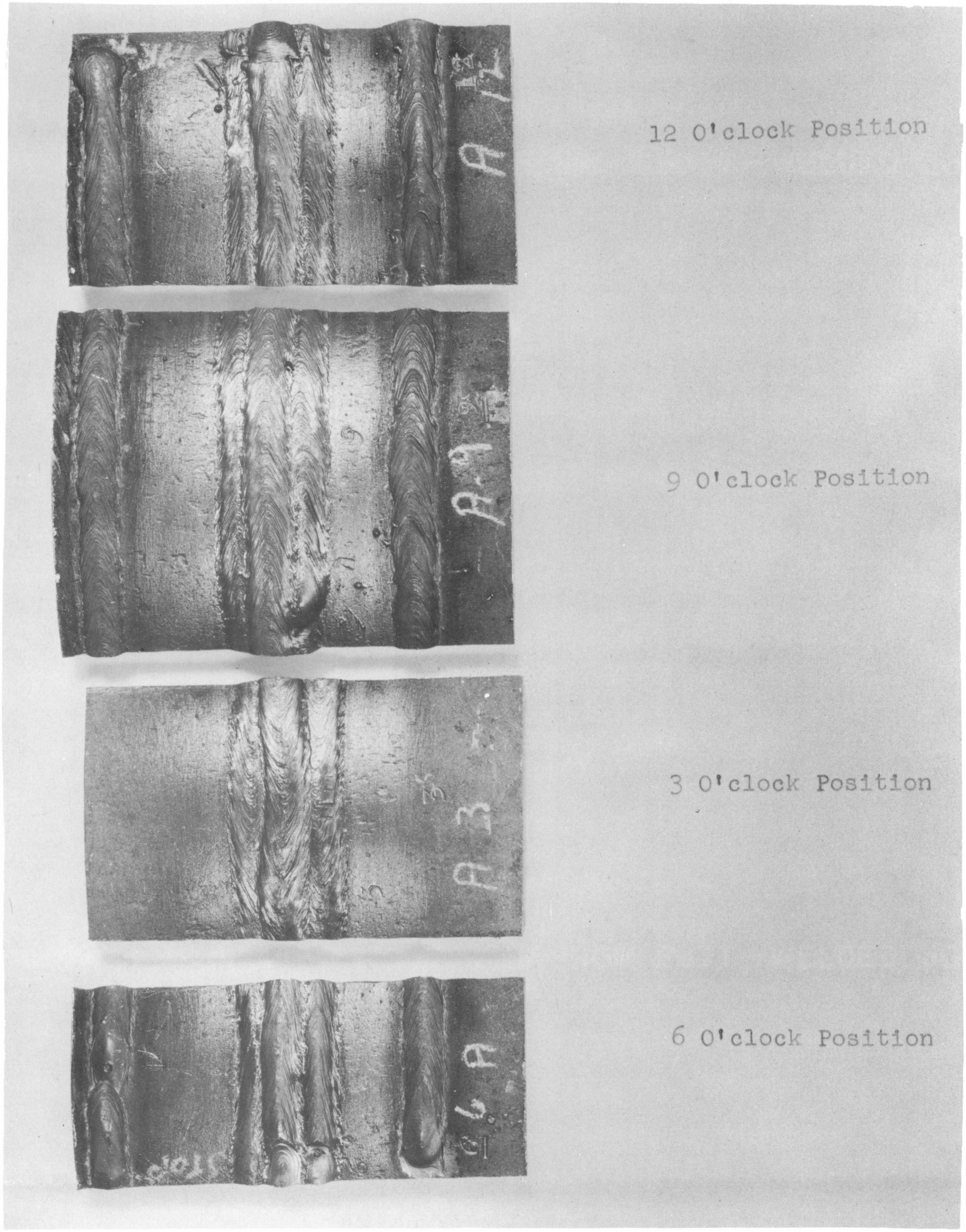


FIG. 10: FACE BENDS OF SECTIONS FROM BUTT-WELDED JOIN

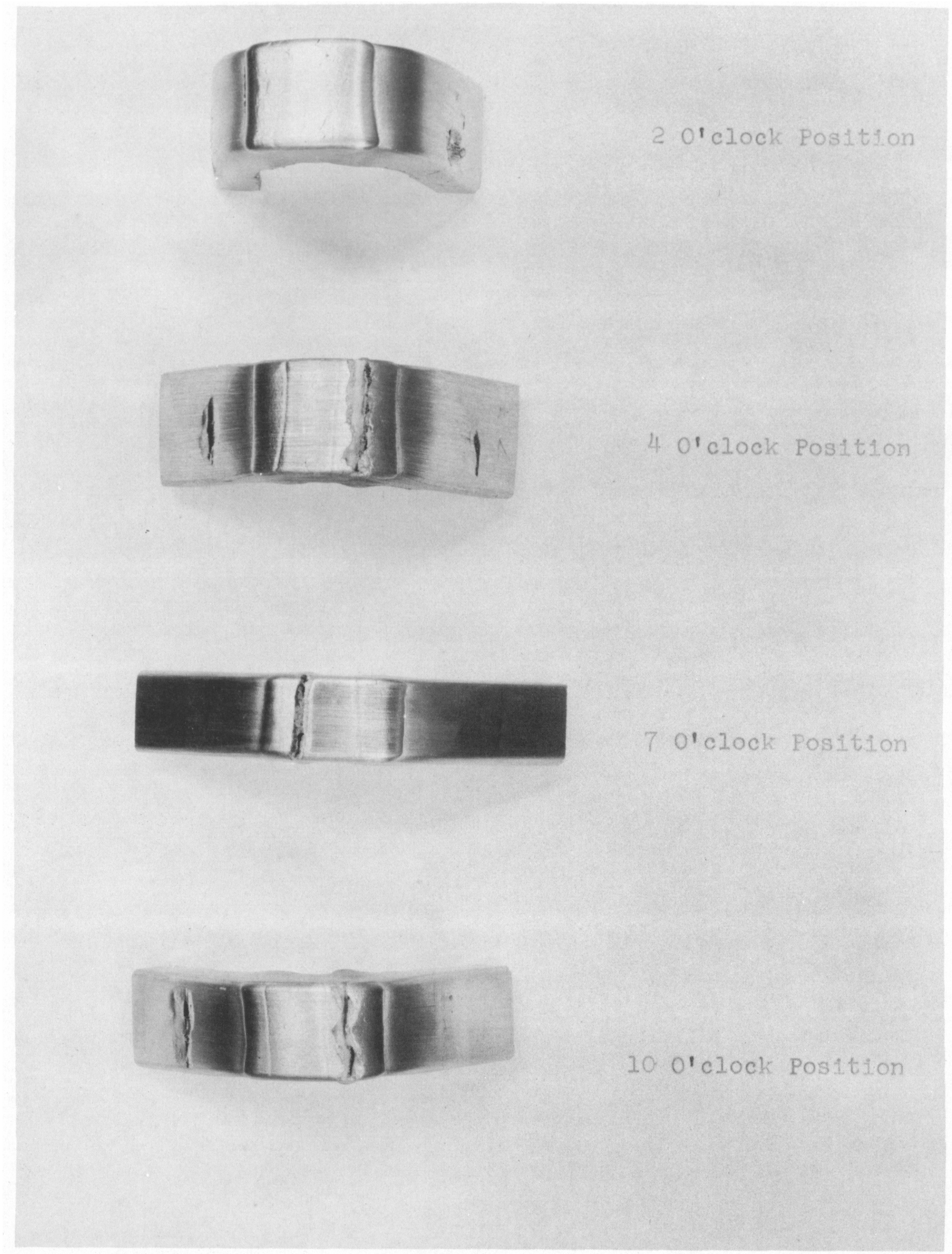
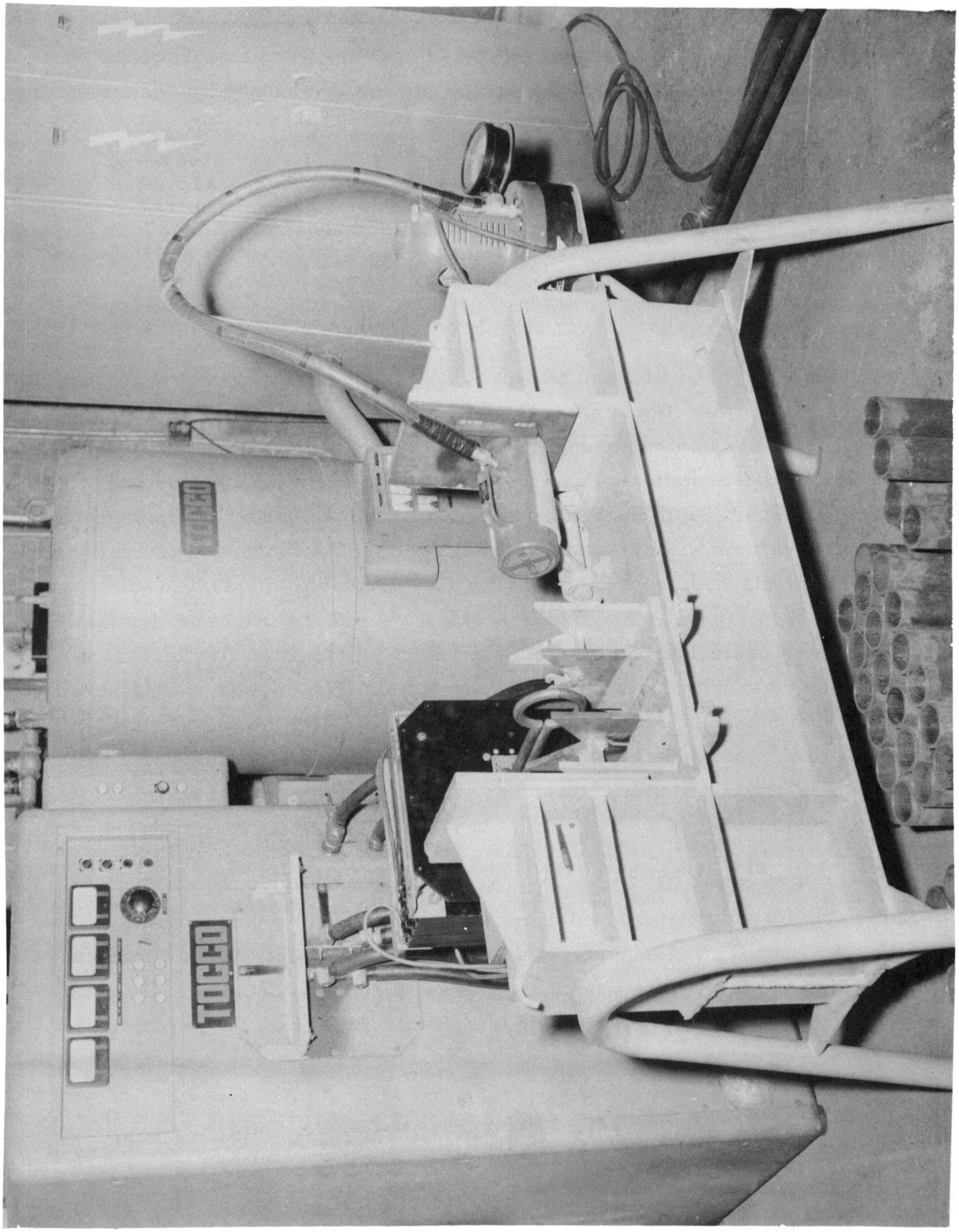


FIG. 11: 300 KW HIGH-FREQUENCY GENERATOR



The bulk of this data has been obtained on carbon steel, but Croloy 2-1/4 has been obtained in 3-in. schedule 80 and 6.625-in. OD with a .313-in. wall for trial by this method. Preliminary work with Croloy 2-1/4 indicates that it will not be more difficult to join than the carbon steel.

E. RADIATION LOOP TESTING

1. Radiation Test Loop No. 1 (BNL)

Design of the induction furnace and the cooler was approved. Pipe and Croloy 2-1/4 billets for the induction furnace are being processed in preparation for assembly. Pipe for the loop cooler has been finned and sandblasted. Other parts are still being fabricated.

Design of the melt and dump tank was completed. Final approval was transmitted after minor revisions were made.

Canning of the main circulating pump was completed, and the pump has been placed in a test loop.

Based on information developed during mock-up testing of the sampler, a final design has been completed. This component is being fabricated.

The in-pile section is ready for assembly, except for the beryllium test specimens.

The late delivery date of valves (February 1958) is still a problem. However, on-site fabrication of a test valve has been completed. Testing of this valve coincides with the main circulating pump test.

Fabrication of the regenerative heat exchanger is in process. A completed mock-up of the tube-to-tube sheet attachment is under study.

It has been agreed that the system can be assembled in the area assigned at the reactor, eliminating potential delay and expenditures involved in disassembling and moving the system after testing elsewhere.

The power wiring circuits were designed and the required wiring estimated.

Heat transfer and pressure drop calculations involved in the functional design of the system were compiled and forwarded to BNL for information purposes.

2. Radiation Test Loop No. 2 (ETR)

A scoping report was submitted to ETR. The technical aspects of the system and its suitability for installation were discussed in a meeting held at ETR. No major design problems were encountered. Several design modifications are in progress due to lack of space, shielding requirements, and post-irradiation removal problems.

Design of the system and the test program will soon be presented to the ETR Program Committee for engineering approval.

Mock-up design of the in-pile section is progressing. This mock-up will be used to obtain reactivity measurements at the ETR critical facility.

Procurement of instrumentation and other system components is under way.

Evaluation of the degasser and a fission gas escape problem associated with the system is continuing.

Mock-up of a liquid metal sampler is under construction.

3. Radiation Test Loop No. 3 (MTR)

The flux measurement necessary to evaluate the suitability of MTR facility HG-5 for an in-pile U-Bi loop, was made in an adjoining vertical hole due to the current inaccessibility of HG-5. The flux value obtained was 1.36×10^{13} n/cm^2/sec. The influence of this flux on the design of the system and the value of an experiment at this flux level are under consideration.

V. BUILDINGS, GENERAL SERVICE AND SITE DEVELOPMENT

(J. D. Kenney)

A. REACTOR BUILDING

1. Revisions

Revisions to the Phase I reference design include the following significant changes:

a. The number of cells was reduced from 14 to 7.

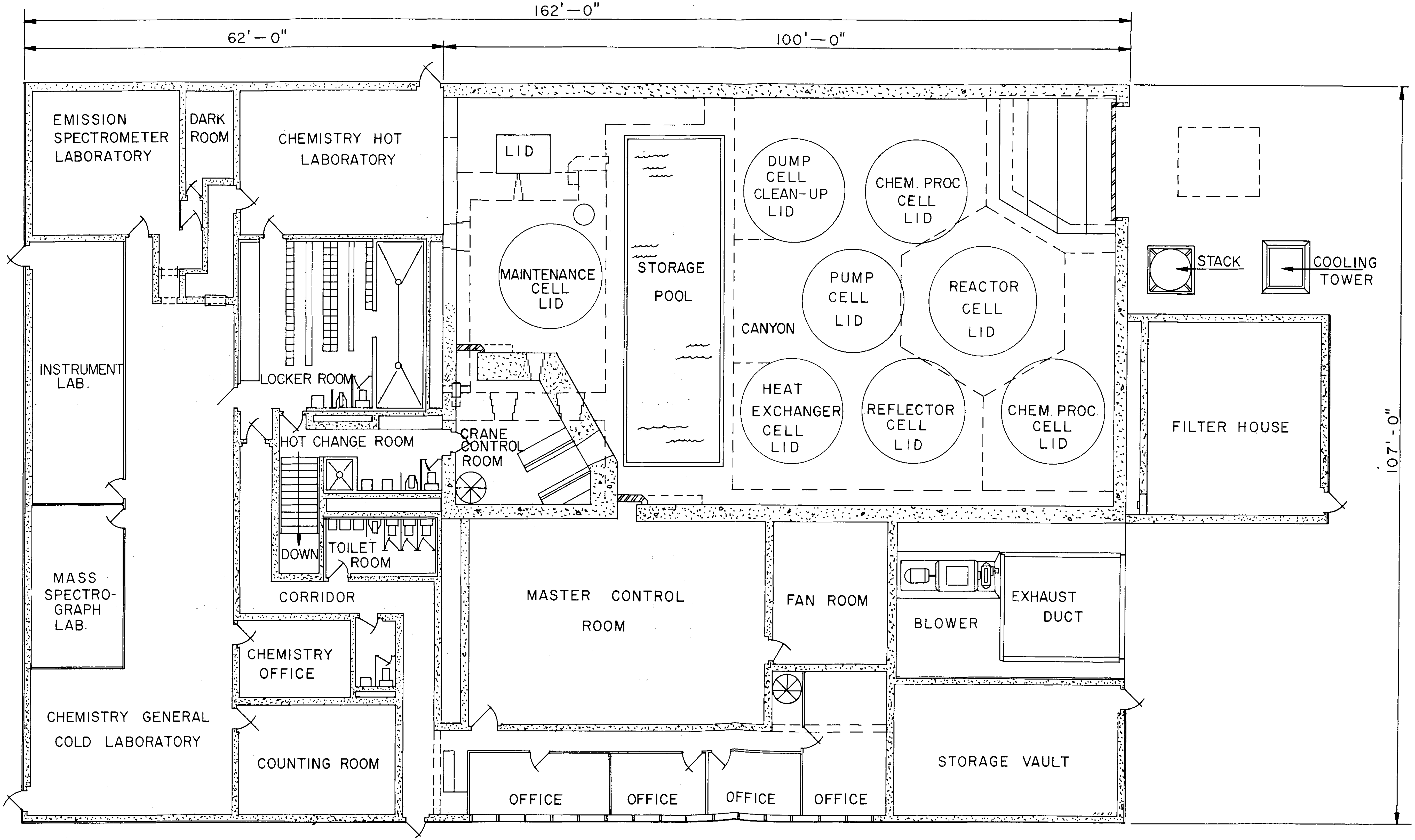
b. Space requirements were reestimated based upon more recent information on equipment sizes.

2. Reactor Bay

The reactor bay and office-laboratory area make up the two distinct areas of the reactor building whose principal dimensions are 162ft by 107ft. (See Figure 12.) The reactor canyon, containing the reactor and auxiliary equipment, will be constructed of concrete for radiation shielding. It is 100ft long, 63ft wide and extends approximately 46ft above ground level. The reactor and auxiliary equipment will be contained in seven cylindrical steel cells 31ft completely below grade. The canyon above these cells will house an 80-ton-capacity overhead crane whose rails are 26ft above the canyon floor. Any component removed from the cells can be safely stored in a 31-ft-deep storage pool alongside the cells. A heavily shielded crane control room with viewing windows for remote crane operation will be located in one corner of the canyon, and removable concrete plugs in the canyon floor above each cell will allow access of remote maintenance equipment.

This equipment will be mounted on a heavy cast iron plug with eccentric rotating discs.[(P)] The maintenance plug will be stored in a hole in the canyon floor alongside the storage pool. The hole will be directly above a maintenance hot cell in which the equipment on the plug will be used. Adjacent to the maintenance hot cell will be a robot operating room above which a removable concrete lid permits access of the robot to the canyon floor.

FIG. 12: GROUND FLOOR PLAN



The canyon walls, above and below ground floor, and the canyon roof will be of standard concrete. Canyon floor, shielding around the reactor cell, and crane control room will be of barytes concrete. The canyon roof will be supported on steel trusses from which a metal pan ceiling will be hung to seal off the truss area from radioactive dust. Lights will be recessed into this pan ceiling for ease of maintenance.

There will be a steel door (15ft by 20ft by 6in) in one end of the canyon for component removal, and steel doors (3ft by 7ft) will lead into the master control room and hot change room.

The seven cells in the canyon basement will contain the reactor primary pump, intermediate heat exchanger, dump tank, reflector system, and two chemical processing systems. A separate shielded area in one corner will contain contact maintainable heating and cooling equipment.

3. Office and Laboratory Area

This L-shaped area's longer leg will be 162ft by 44ft, the shorter leg 62ft by 63ft, and the whole area 11ft - 6in. high. The ground floor will contain offices for operating personnel and health physics, chemical laboratories, locker room and hot change room, master control room, ventilating fan room, and fissionable materials vault. The exterior walls will be concrete block with metal facing. In the basement beneath the laboratories will be a metallurgical hot cell, chemical hot cell, junior chemical cells, equipment and personnel decontamination areas, and manipulator operating rooms. (See Figure 13.)

The heavy shielding walls encompassing the hot cells will be of barytes concrete, the rest of the area of standard concrete construction. Stairs from the basement will lead directly to the hot change room above. Each hot cell will have a 4-in. steel access door, and special viewing windows of high-density glass will be installed in the walls.

The basement area beneath the office area will house the auxiliary systems which do not require heavy shielding. (See Figure 14.) These include two levels of switch gear, the steam generator, sodium storage room, inert-gas storage room, sodium reflector system, and a sub-level sodium dump tank room. This area will be of standard concrete for fireproofing and low-level shielding. There also will be an outdoor air-blast steam condenser with an exhaust stack extending above the canyon roof line and a 150-ft stack for discharging gases to atmosphere. A hot filter room (30ft by 30ft by 18ft) housing exhaust fans

FIG. 13: SECTION ELEVATION

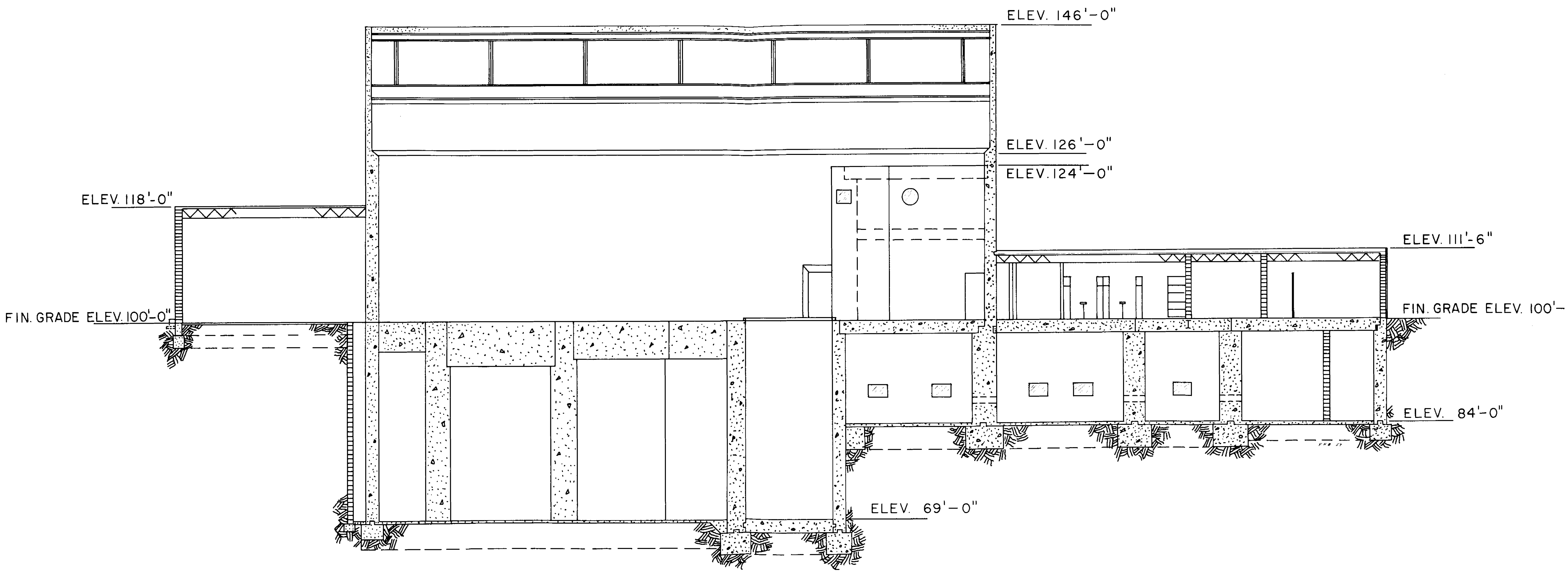
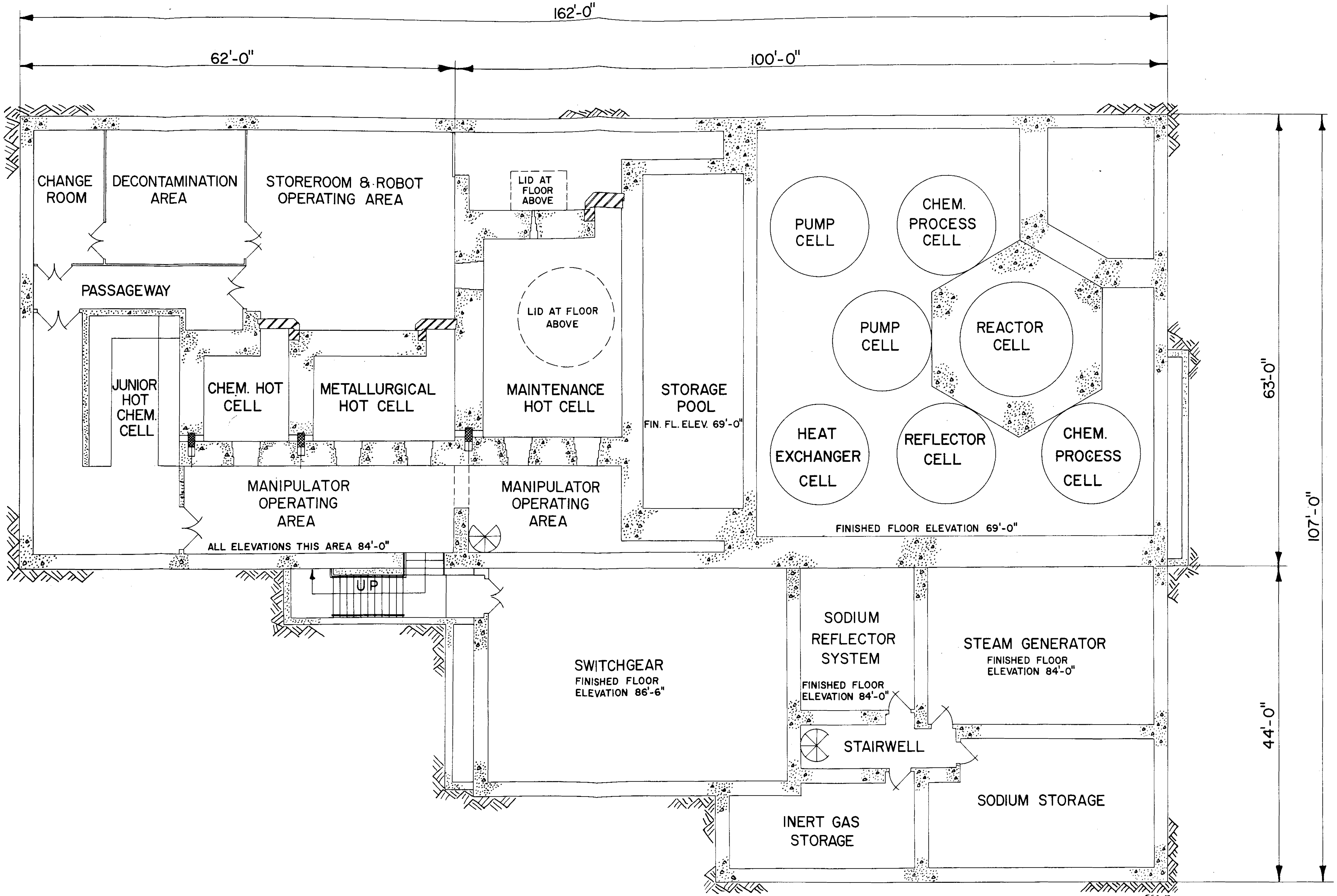


FIG. 14: FIRST LEVEL BASEMENT FLOOR PLAN



and special filters for removing radioactive gases from the reactor canyon, will be located alongside the stack at the end of the canyon.

The building will be ventilated so the air flow is directed from the least radioactive areas to the most radioactive areas and then to the hot filter room. Some laboratory areas will be air conditioned for humidity control, essential for instrumentation.

The electrical distribution system will be 480/277 for lighting, instrumentation, and heating circuits. Mercury vapor lights will be used in the reactor canyon and fluorescent lighting in other areas. The anticipated total will be 3000 KW.

B. REMODELING EXISTING BUILDINGS

It has been assumed that any site selected will provide buildings for the LMFRE. The buildings will be remodeled to accommodate the following:

1. A security and communications area for clearing all personnel, mail and telephone calls.
2. A first aid station for emergency treatment.
3. Offices for administrative, accounting, and associated personnel.
4. Offices for AEC representatives if required.
5. Cafeteria, rest rooms and locker rooms.
6. General storage area.
7. Machine shop for miscellaneous equipment fabrication and repairs.
8. Motive equipment storage and repair facilities.
9. Laundry for contaminated clothing.

These facilities will require a floor area of approximately 10,000 sq. ft. Normal services will be required for water, sewage, heat, and electric light and power.

C. SITE DEVELOPMENT AND OTHER SERVICES

Site development includes land clearing, Reactor Building excavation, and minimum grading and landscaping. Other services will encompass yard lighting, railroad siding, sanitary and storm sewers, water lines, steam lines, electric power lines, and access road. A hypothetical site has been assumed which provides these services at a distance of about 1000ft.

D. Schedules

A number of schedules have been prepared; six months are estimated for final building design completion and about twelve months for Reactor Building erection.