AECU-3559

REACTORS—POWER

UNITED STATES ATOMIC ENERGY COMMISSION

GAS COOLED NUCLEAR REACTOR STUDY

Final Report

By
A. S. Thompson

July 31, 1956

Nuclear Power Department
Studebaker-Packard Corporation
Detroit, Michigan

Technical Information Service Extension, Oak Ridge, Tenn.

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Final Report

By A. S. Thompson

Approved by O. E. Rodgers

July 31, 1956

Work performed under Contract No. AT(30-3)-214

Nuclear Power Department
STUDEBAKER-PACKARD CORPORATION
Detroit, Michigan

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INTRODUCTION

This is the final report submitted to the Atomic Energy Commission by the Studebaker-Packard Corporation in fulfillment of Contract AT(30-3)-214.

DESIGN SPECIFICATIONS

Contract AT(30-3)-214 calls for the study of a gas cooled reactor having the following design specifications and desired characteristics:

(a) Net capability of 15,000 KW of electricity at standard generating conditions and 0.60, 0.80 load factors.

(b) System will be operated in populated areas.

(c) Cost of SNM and source materials to be consistent with current classified pricing schedules.

(d) Conduct preliminary parametric studies of the system to determine the optimum gas turbine power plant cycle.

(e) Utilize a heterogeneous gas cooled reactor; liquid metal coolants are not acceptable.

(f) Conceive and utilize the best reactor which could be available in 3 years.

(g) Economy of the system is important, therefore physical size of the plant is not governing.

(h) Cycle efficiency of the system to be compatible with the most efficient conventional plant in this power range.

(i) Core life to be consistent with optimized plant operating costs.

(j) The plant must be capable of meeting its design output in an ambient temperature range of -40°F to +110°F.

(k) Radiation contamination to be considered under the condition of negligible vertical dispersion associated with a temperature inversion.

We understand that there is some interest, within the Atomic Energy Commission, in the potentialities for a nuclear power plant which is independent of the requirement of a large supply of cooling water. We have chosen to include this requirement for independence from a supply of cooling water as a limitation on our design since, in our opinion, this greatly increases the flexibility of use of such a power plant.

GENERAL

The investigation covered by this report contains two main parts. An investigation was made of the performance of a gas-cooled reactor, designed to provide a source of high temperature heat to a stream of helium. This reactor, in turn, is used as a source of heat for the air stream in a gas-turbine power plant. The reactor design was predicated primarily on the requirement for transferring a large amount of heat to the helium stream with a pressure drop low enough that it will not represent a major loss of power in the power plant. The mass of uranium required for criticality under various circumstances has been investigated by multigroup calculations, both on desk calculators and on an IBM-704 machine. The gas turbine power plant performance was studied based on a Studebaker-Packard-designed gas-turbine power plant for the propulsion of destroyer-escort vessels. A small experimental program was carried out to study some effects of helium on graphite and on structural steels.

CONCLUSIONS

This study has been based on the use of a nuclear reactor to supply heat to an open-cycle gas turbine. The gas turbine, and its heat exchangers, uses conventional gas turbine materials operating under conditions (turbine inlet temperature, 1400°F) which are presently attainable. The reactor, by use of the inert gas helium and the structural and moderating material, graphite, is enabled to operate at temperatures above those attainable with most engineering materials (2420°F). This open-cycle nuclear, gas turbine is able to operate with relatively good efficiency (maximum 30 percent, minimum 28 percent) over the operating range from one-fourth to full power, and requires no supply of water for cooling purposes. The reflector-moderated reactor, can be made critical, in sizes of interest for the nominal 60 megawatts heat output required for the operation of the 15 megawatt shaft output power plant, over a range of fuel enrichments above 10 percent using either graphite or beryllium as moderating material. The use of graphite requires a higher mass of uranium-235, or a higher enrichment, or a larger reactor than beryllium, in order not to exceed a reasonable concentration (assumed 25 percent by volume) of uranium carbide in the uranium carbide-graphite fuel matrix. The reactor (shown in Figure 1), having the minimum size determined by heat transfer considerations, can operate on 10 percent enriched fuel with beryllium moderator or with 20 percent enriched fuel with graphite moderator. An increase in the linear dimensions of the graphite moderated reactor of 50 percent with the same volume for fuel enables it to operate with 10 percent enriched fuel.

In order to achieve a higher thermal efficiency from the gas-turbine power plant it is necessary to find means to operate with a higher temperature at the inlet to the power turbine. This can be accomplished only with the development of new or better structural materials. These new or better materials may be better oxidation resistant materials for use in the air stream, such as are now being developed by gas turbine builders and others. The new materials may also be non-oxidation resistant, so that a high temperature turbine may be operated directly in the inert reactor cooling stream (helium or neon), this turbine then discharging its exhaust heat into the intermediate heat exchanger for use in the air cycle. For instance, turbines of graphite or alloys of molybdenum may be developed. To achieve an overall thermal efficiency of 40 percent would probably require a turbine inlet temperature above 1920°F, with presently available component efficiencies. With the emphasis which has been placed on the development of high efficiency gas turbine components by workers in the field it does not seem likely that large increases in thermal efficiency will be likely to come from large increases in component efficiency.

The open-cycle, nuclear-powered, gas turbine, power plant studied for this application has some unique advantages.

1. It can operate without a supply of cooling water.
2. It uses conventional gas-turbine components under conventional operating conditions to obtain a relatively good operating efficiency.
3. It uses a reactor based on graphite as a structural material and cooled with an inert gas (helium).

RECOMMENDATIONS

On the basis of our studies we conclude that the open-cycle, nuclear-powered, gas-turbine, power plant considered here merits further study as a potential competitor for use in smaller, stationary, power plants, particularly for regions in which there is a shortage of water for cooling purposes. We therefore recommend that a further analytical and experimental program be carried out to investigate a detailed design of such a power plant.

The experimental program should include particularly a study of the problems of uranium carbide-graphite fuel elements and inert gas coolants, as well as structural materials for high temperature gas-turbine components to operate in both oxidizing and non-oxidizing atmospheres. The analytical program should be aimed at determining the effects of the experimental program on the design and performance of the reactor and power plant.

In view of the complexity of heat exchange equipment and the serious consequences of a failure in this equipment a series of tests on the fabrication by welding and brazing and on high temperature operation of heat transfer units should be made. Designs which reduce thermal stresses on sudden changes in operating conditions should be proved.

There is considerable evidence that reflector-moderated reactors of the type considered here can operate under reasonable nuclear conditions. Still tests of reactor criticality should be run under both hot and cold conditions, and these should be used to predict the transient response of the reactor system.

Very flat neutron flux (power density) distribution curves are found for reflector-moderated reactors in which all moderating materials have been removed from the reactor core (see Table 2, Reactor 38). For this reason, as well as for mechanical reasons, high temperature metallic fuel elements for non-oxidizing atmosphere should be developed. In this respect columbium-uranium seems to show particular promise.

A program of experimental work, and of the analytical work which supports it, is shown below.

Experimental Program

Fuel Elements (uranium carbide-graphite)

Fabrication method for fuel elements

Structural properties of fuel elements

Radiation damage to fuel elements

Coolants (helium and neon)

Contaminant problems

Reprocessing problems

Reactor

Fabrication problems

Criticality tests

Heat Exchanger

Fabrication problems

Power Plant

Tests of heat exchanger materials

Tests of materials for gas turbines in oxidizing atmospheres

Tests of materials for gas turbines in non-oxidizing atmospheres

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Nuclear Calculations

Criticality

Conversion of fertile materials

Reactor control

Shielding

Safety

Heat Transfer Calculations

Reactor design

Intermediate heat exchanger design

Power Plant Studies

Materials Studies

Survey of available data on materials

Cost Analysis

First cost

Operating cost

DESIGN CONSIDERATIONS

This section gives a general description of the gas cooled reactor and associated gas turbine system which we have investigated in connection with the design specifications for our contract outlined above.

The reactor system which we have chosen to investigate consists of a beryllium or graphite-moderated reactor, helium cooled, with ceramic fuel elements supplying thermal energy to an intermediate heat exchanger. The intermediate heat exchanger in turn acts as the source of heat for three separate gas turbine units operating in parallel. The reactor is of the reflector-moderated type described in Studebaker-Packard Report NPD-1, "Reflector-Moderated Reactors for Power Purposes", dated March 21, 1956. The ceramic fuel elements are composed of a compacted and sintered mixture of uranium carbide and graphite arranged in the reactor in a cylindrical annulus through which the coolant helium flows. At the maximum power output from the power plant the helium coolant in the reactor is maintained at a pressure of 20 atmospheres. Part power conditions are satisfied by lowering the pressure on the helium system below 20 atmospheres. Full power operation requires the simultaneous operation of all three submultiple gas turbine power units. For successive levels of part power operations one or two of the three submultiple gas turbine power units are shut down enabling the remaining units to operate near their designed peak performance. Part-load performance requirements can therefore be satisifed efficiently by an open thermodynamic cycle which is independent of a need for cooling water. To render the efficiency of these open cycle gas turbines acceptable and to decrease the required pressure ratio in the compressors, corresponding to peak system efficiency, exhaust recovery regenerators are provided in the gas turbine cycle.

Certain features of this nuclear power system are designed to accomplish special purposes which seem to us desirable as part of our overall objective. Some description of some of these features follows.

Reflector-Moderated Reactor

The name reflector-moderated reactor is derived from the fact that neutrons are moderated in a region of the reactor which is physically separated from the nuclear fuel bearing reactor core. This type of reactor represents a departure from the usual practice of the design of thermal reactors in which the reactor core consists of both moderating material and nuclear fuel, either mixed or interspersed. It is also a departure from the practice of the design of fast reactors in which moderated materials are excluded from the reactor. This type of reactor is designed to achieve certain advantages both nuclear and mechanical. It is possible with the reflector-moderated design under some circumstances to achieve a relatively flat distribution of neutron flux and, therefore, reactor power using a uniform distribution of nuclear fuel in the reactor core. This results in advantages in the efficiency of burn-up of nuclear fuel and in the over-all thermodynamic efficiency of the power plant since it is not so limited by the existence of hot spots in the reactor core. A somewhat independent control of the life cycle of the neutron is achieved in this design. It is possible to some extent to control the neutron energies at which such phenomena as fission and the conversion of fertile materials occur by changes in reactor geometry. Some advantage in safety may result from a longer neutron life cycle time with the reflector-moderator. Some mechanical advantages are achieved since it is not necessary with the reflector-moderated reactor to mix moderator material more or less uniformly with fuel in the reactor core. The reactor core can hence be more concentrated giving the possibility of better designs from the standpoint of heat transfer, corrosion, mechanical design, and so forth. Some shielding advantage is achieved due to this concentration of the reactor core. Graphite was chosen as a moderator material for this reactor because of its nuclear and high temperature mechanical properties. It has the unique property that its short-time tensile strength at 4500°F is approximately double the short-time tensile strength at room temperature. It has creep properties similar to those of high temperature gas turbine materials at the temperatures at which these are used (1600°F). Its strength-weight ratio at 3600°F is roughly the same as that for gas turbine materials at their operating temperatures. Graphite is used for crucibles for handling many molten materials at high temperatures. The strength of graphite, as well as several other properties, apparently improves as its density is increased. Graphite has three limiting properties which must be taken into account in its use. It cannot be used in an oxidizing atmosphere at high temperatures, it is brittle at room temperature, and it is porous at normal (1.5 to 1.7 gm per cm^3) densities.

Since the use of beryllium as a moderating material gives lower critical mass its use also was investigated. Beryllium probably requires cooling at temperatures above 1200°F.

Reactor Core

The reactor core is composed of fuel elements formed from compacted and sintered uranium carbide and graphite. These fuel elements are cooled by helium at up to 20 atmospheres pressure. This combination was chosen because of the compatible nuclear and mechanical properties of the system. Helium is an inert gas chemically and does not become radioactive under neutron bombardment (except for a negligibly small fraction of helium-3). Uranium carbide fuel elements, particularly for relatively low concentrations of U-235, should exhibit acceptable radiation damage properties. *"Uranium carbide, UC_2, is chosen as the nuclear fuel because it can be heated to temperatures above 3600°F in contact with graphite and helium without appreciable reaction, vaporization, or formation of gaseous products. After removal from the reactor, it oxidizes easily in 2000°F air to give uranium oxides, which dissolve readily in nitric acid to give uranyl nitrate without evolving gases. The uranyl nitrate is a convenient material from which to separate the fission products and to regenerate the carbide."

Fuel Element

The primary key problem, in our opinion, for any high temperature, high-performance reactor for a stationary power plant is the design of a fuel element which can be manufactured and assembled into the reactor, which can be removed in a radioactive state from the reactor and reprocessed to recover the useful residue, and which is compatible with the requirements for cooling and structural needs. A very strong determining factor in the usefulness of the fuel element which satisifes these requirements is its

*"Small Gas-Cycle Reactor Offers Economic Promise", by Farrington Daniels, Nucleonics, Vol. 14, No. 3, March 1956, p 35.

relationship to the burn-up of fissionable material, in terms of operating life, and the possibility of operation, with a low fuel enrichment, with a significant conversion of the fertile diluent material to fissionable material. The operating life of a given fuel element may well be set by the percentage burn-up of fissionable material, particularly for reactors having low critical masses of fissionable material. The effect of burn-up of fissionable material may be offset by conversion of fertile to fissile material, although radiation damage to the fuel element may then set a prior limitation on the useful life of the fuel element. The fuel element chosen for this design, as has been stated, is a mixture of uranium carbide with graphite. There are two main methods of manufacture for these fuel elements which seem to us worthy of consideration. The first of these involves the impregnation of graphite with uranium oxide which is converted by heating to uranium carbide. This method has the inherent limitation that it applies only to relatively small content of uranium. Since the porosity of graphite is of the order of 25% this sets an upper limit on the amount of uranium carbide which can be forced into the porous graphite structure. In actual practice this limit is greatly reduced by the impregnation properties of uranium oxide in graphite which cause a gradient density from the surface inward and further reduced by the escape of gaseous products (carbon dioxide and carbon monoxide) during the heating process. An upper limit to the amount of uranium carbide included by this method, determined somewhat by size of fuel element, may be of the order of 5 percent by volume. This constitutes a serious limitation on the total amount of uranium, including both fissionable and fertile material, which may be included in the lattice and probably sets a lower limit on the fuel enrichment which can be used.

A second method for making uranium carbide-graphite fuel elements is to compact and graphitize a mixture of uranium carbide and carbon powders, in which case the limits on included uranium carbide are set by the properties of the resultant material. Graphite is considerably superior to uranium carbide in several of its known properties including primarily those which have to do with resistance to cracking under thermal gradients. It is probable also that the resistance to radiation damage may become markedly less as the uranium content is increased past some certain point. Experiments are needed to determine the deterioration of certain of these material properties as the concentration of uranium carbide is increased. We have assumed a limit for the purpose of our studies of 25% by volume of included uranium carbide and have used this limit to determine the fuel enrichment which would enable the reactor to achieve criticality.

Stainless steel, sandwich-plate, fuel elements were also considered in a preliminary way for this design. However, it is probably not possible to operate these fuel elements at surface temperatures over 1700°F at a maximum. Allowing appropriate temperature drops for a reasonable size of reactor core and for reasonable sizes of intermediate heat exchanger gives too low a turbine inlet temperature for the gas turbine power plant to achieve competitive performance. This problem would be somewhat eased in the case of the closed cycle gas turbine in which the same gas which cooled the reactor was also the power medium for the gas turbine cycle. In this case there would be only the one temperature drop required from the fuel element surface temperature to the gas turbine inlet temperature because of the absence of the intermediate heat exchanger. This, however, gets back to the closed cycle gas turbine which requires a large supply of cooling water and hence was not considered further here. Stainless steel, sandwich-plate, fuel elements suffer from the inherent limitation that the fission product poisons cannot be removed.

Helium Coolant

Helium was chosen for the reactor coolant for three primary reasons. First, it is an inert gas chemically. Second, it does not become radioactive. Third, it is a good heat transfer medium relative to other available gases. Since helium is chemically inert, a reactor such as this, having ceramic fuel elements, can be operated at very high temperatures (probably higher than $4000^{\circ}F$). Since helium does not become radioactive the only contamination in the helium system would be due to the leakage of radioactive materials from the reactor into the helium stream. This has both good and bad features. If the fuel elements are sufficiently porous to allow leakage of volatile fission products into the helium stream this can be used to rid the reactor of most of these fission products which are bad neutron poisons, particularly xenon-135 and samarium-149.

There are, however, some problems associated with the use of helium. The first is that helium is considered to be a difficult material to contain in a system in which essentially no leakage is allowed. Special precautions will be needed to prevent helium leakage. Also, since the circulating helium will undoubtedly contain a certain amount of gaseous fission products and possibly some fissionable materials, it will be necessary to reprocess a small percentage of the helium continuously to remove these materials.

This problem would exist, however, with any other gas unless the fuel element were sealed against leakage as they would be, for instance, with stainless steel fuel elements. In this case, however, the advantage of continuous removal of neutron poisons is lost.

The prevention of an excessive build-up of poisons and radioactivity of the helium stream requires some type of filtration and processing system. Various methods have been proposed, among which are cooling and filtration, diffusion cascades, storage for eventual decay of short-lived products, and absorption into fats. Among the above mentioned, cooling and filtration is possibly the most effective for a minimum amount of equipment. Cooling equipment for partial condensation separation techniques has been developed. The circulation of large quantities of gas is not economical and a low bleed-off rate would have to be used to be consistent with a good over-all power plant efficiency.

Because of the high acoustic velocity in helium (roughly three times that in air at the same temperature) the amount of work per stage which can be done by a helium compressor limited to reasonable rotative speeds, as determined by the properties of the structural materials of the compressor, is much lower than with air. The fan pressure drop for which this system as designed is, however, so low that this is not a significant harmful factor. Other inert gases, for instance neon, could be used and would have an acoustic velocity in a better range. However, none of the other inert gases have the advantage of helium of not becoming radioactive under neutron bombardment. Nitrogen has thermodynamic properties very close to those of air but it is not chemically inert and under neutron bombardment forms radioactive carbon-14 which must be removed.

Conversion of Fertile Materials

Under present restrictions on shipping fissionable materials to foreign countries it is desirable to maintain the enrichment of uranium below 20 percent of contained uranium-235, so that the reactor can be made available for locations away from the continental United States. For the maximum utilization of this enriched reactor fuel, and for eased burn-up conditions it is desirable to convert as much of the fertile uranium-238, contained in the reactor fuel, as possible to fissionable plutonium-239 during the operation of the reactor, and within the reactor core. To accomplish this it is necessary to minimize the leakage of neutrons, which requires controlling the neutron energies at which neutrons are absorbed in both fertile and fissionable materials.

Reactor Safety and Containment

Reactors are susceptible to two main kinds of catastrophic accident. The power in the reactor may increase rapidly beyond the capacity of the cooling system due to a change in the nuclear characteristics of the reactor. The cooling system may breakdown so that the power normally generated in the reactor is not removed. In either case the temperature rises in the reactor until the power generation is checked by a control instrument or by a compensating change in the nuclear characteristics of the reactor caused by the temperature rise. Various other accidents, which are not catastrophic, may require shutdown of the reactor, such as leakage of radioactive materials into the atmosphere or into unshielded parts of the power system, gradual loss of cooling capacity, mechanical breakdowns, etc.

In a gas cooled reactor, there are two main mechanisms which may limit a sudden rise in temperature. A negative temperature coefficient, if one exists, may be due to a reversible change in reactor geometry or neutron cross sections with changing temperature. If this mechanism does not succeed in shutting down the reactor its reactivity will eventually be shut off by a breakdown of some part of the reactor structure.

The temperature coefficient in a reactor is formed from a detailed and rather delicate balance among several compensating effects from expansions of different parts of the reactor and changes in cross sections with changes in temperature. These changes occur at different rates, depending on how closely linked they are to the fission process. Detailed calculations and experimental checks are needed for a reliable evaluation of the temperature coefficient.

The core of the reactor considered here is composed of uranium carbide in a graphite matrix. At about $4350^{\circ}F$ uranium carbide melts. Above about $4500^{\circ}F$ uranium carbide exists as a liquid in the graphite matrix and can be expected to diffuse to the surface and probably leak into the coolant stream. Uranium carbide volatilizes at about $7900^{\circ}F$. Graphite sublimes at about $6600^{\circ}F$. Above this temperature then the graphite structure breaks down very rapidly. Depending on the rate of power increase, the nuclear characteristics of the reactor can be expected to deteriorate quite rapidly above say $5000^{\circ}F$ as the contents of the reactor core are carried by the cooling stream into other parts of the helium system. Unless the rate of power increase is so high that the pressure of graphite vapor builds up explosively, it would be expected that the radioactivity scattered from the reactor could be contained within the helium system.

An additional component might be added to the reactor core (a fuse), so designed as to disintegrate ahead of structural damage to the core itself, spreading a neutron poison through the core to shut down the reactor.

Controls

There has not been time on this project for an adequate evaluation of the particular design problems of a reactor control system for this reactor. A schematic arrangement of the control system for the power plant is shown in Figure 20.

The reactor control system for a reflector-moderated reactor with a gas coolant might be expected to be different from some other reactors because of a change in the neutron lifetime, which depends primarily on two effects. First, a large fraction of the fissions are caused by epithermal neutrons. Second, as pointed out in Appendix I, the main lifetime of thermal neutrons in the reflector-moderated reactor is determined primarily by the scattering properties of the moderator, and therefore may be quite long. The first effect causes the neutron lifetime to be shorter, the second larger, than for a homogeneous thermal reactor. The longer the total lifetime of neutrons, the easier is the control problem.

Our calculations of reactor criticality have, so far, been one-dimensional. In order to measure the effectiveness of control rods it is necessary to do two-dimensional calculations.

Detailed calculations and tests are needed to determine the control requirements of the reactor.

Intermediate Heat Exchanger

An intermediate heat exchanger in the nuclear gas turbine system serves the purpose of isolating the reactor and its coolant mechanically from the atmosphere. Direct air cooling of the reactor would create several problems of which the main ones are the generation of radioactive argon-41 in such large quantities that it could not be discharged to the atmosphere, and the problems of handling oxidation in the reactor at high temperatures.

The helium-air heat exchanger is of the shell and tube counterflow type. A counterflow arrangment is used due to a more efficient heat exchange than that possible in other flow arrangements. A one-pass unit rather than a baffled multipass cross-counterflow arrangement was chosen to eliminate any large temperature variations along the tube and shell, and to keep the pressure losses to a minimum. The shell and tube type, with the helium flowing within the tubes, is desirable due to the high helium pressures involved.

Gas Turbine Power Units

A simple open cycle gas turbine has inherently a rather poor thermal efficiency at practical operating temperatures. A considerable increase in thermal efficiency for the open cycle gas turbine can be achieved by the addition of a properly designed regenerator which removes waste heat from the exhaust system and, by means of a heat exchanger, adds it to the air stream between the compressor and the primary heat exchanger. The increase in efficiency results in a decreased heat power output required from the reactor for a given power output from the plant, which in turn results in a reduced heat transfer problem in the reactor. Likewise the burn-up of fuel in the reactor is decreased giving a longer operating life for otherwise similar conditions.

The off-design (part load) operation of the gas turbine results in poor performance because of the need to operate the cycle at reduced turbine temperatures and under off-design operating conditions for the various components. There are two standard ways of handling this problem. One is by the use of a closed cycle in which the output is varied by varying the density of the gaseous power medium in the cycle. We chose not to use this method because of the requirement which it imposes for a large supply of cooling water into which to dump waste heat from the cycle. We have chosen instead to solve this problem of part load operation by having a multiplicity (three) of parallel gas turbine power plants, any number of which can be operated simultaneously to satisfy given conditions of loading. This enables the power plant operator to satisfy part load operating conditions with differing numbers of gas turbine units always operating reasonably close to an optimum design condition.

Alternative Power Plants

An alternative power plant, actually an adaptation of the one considered here, would allow a higher temperature of operation, and hence a higher efficiency, if certain component developments can be carried out. In this power plant a high temperature turbine is placed directly in the inert-gas, reactor-cooling, stream. This turbine is built from graphite or other high temperature material, such as a high strength alloy of molybdenum. The exhaust heat from this high temperature turbine is used by the intermediate heat exchanger to heat air in the air-turbine cycle. The materials for such a high-temperature cycle are not now available.

Thermodynamic Efficiency and High Temperature

The thermal efficiency of a gas turbine power plant of a given type can be improved either by raising the turbine inlet temperature or by increasing the efficiencies of the power plant components. With presently available components and materials a regenerative, gas-turbine, power plant such as that considered in this investigation can apparently achieve an over-all thermal efficiency of about 30 percent under standard-day conditions of operation. Considering the large effort which has been expended on the development of efficient gas-turbine components it does not seem reasonable to base the hope for a large increase in over-all thermal efficiency on the hope for greatly improved components. An increase in thermal efficiency to 40 percent will probably require a turbine inlet temperature of about 1900°F.

A higher temperature of operation can be achieved by the development of better structural materials for turbines and heat exchangers, and by improved cooling methods for turbines. Methods of turbine blade cooling are being investigated now by manufacturers of turbines and others. One of the more hopeful methods seems to be the use of internal cooling of the turbine blades from the evaporation of water, the water being circulated by natural convection in the centrifugal field of the rotating turbine wheel. It is possible with this method to operate the turbine blades as much as 1000°F cooler than the gas stream in which they operate.

There are two apparent directions for increasing the effective operating temperature in nuclear, gas-turbine, power plants. These are: (1) the development of higher-temperature, oxidation-resistant, structural materials for operation of the turbine in the high-temperature air stream; and (2) the development of high-temperature, non-oxidation-resistant, structural materials for operation of the turbine in a closed-cycle, non-oxidizing, working medium. The first of these alternatives is already being performed actively by groups interested in increasing the performance of present gas turbines. The most promising avenue at present, being developed by a metallurgical group at Massachusetts Institute of Technology under Nicholas Grant, appears to be the use of small, isolated, non-soluble particles, distributed through a metal matrix, to inhibit transcrystalline plastic flow. For instance, such particles of aluminum oxide in a pure aluminum matrix, obtained by sintering and extrusion, give appreciable strength properties at 900°F. Similarly there seems to be promise for the use of stainless steels, with aluminum-oxide particles, up to 2300°F. The second alternative, using an inert working medium, such as helium or neon, may involve the development of a graphite turbine, or perhaps a molybdenum alloy using the strengthening due to insoluble hard particles in a fashion similar to that mentioned above for aluminum and stainless steel.

For the purposes of the present study we have limited ourselves to the properties of presently available materials.

DESCRIPTION OF CALCULATIONS

Reactor

Criticality calculations of a preliminary nature have been done for spherical reactors on desk calculators using seven lethargy groups and about thirty-five space points. Since these calculations are slow and tedious, as well as being subject to an undue amount of human error, a larger program of multi-group calculations was carried out on subcontract basis for us by the Research Division of the Curtiss-Wright Corporation using their multigroup reactor code and the IBM-704 machine at the World Center Offices of the International Business Machine Corporation in New York City. These calculations used thirty lethargy groups and ninety space points. The results shown in this report are those from the IBM-704 calculations. The results from the desk calculators were used only to orient the machine calculations.

The calculations cover two groups of reactors, one with graphite reflector-moderators, the other with beryllium reflector-moderators, all reactors having uranium carbide-graphite fuel elements. Each group of reactors was done for three degrees of enrichment; 10, 20, and 100 percent. In each group of reactors calculations were made for a standard reactor configuration and for variations on this standard configuration which were designed to measure certain characteristics of gas-cooled reactors of the reflector-moderated type. This standard reactor configuration, a reflector-moderated reactor with a central island of moderator, is shown schematically in Figure 2 . In all, 48 reactors were calculated, each calculation being iterated until the reactor was approximately critical.

The over-all size of the standard reactor configuration was picked, primarily on the basis of heat transfer, to give a sufficient cross section for helium flow and sufficient heat transfer surface to remove 60 megawatts of heat with a 1000 HP circulating fan.

Power Plant

The gas-turbine power plant used for this study was scaled as directly as possible from a Packard-designed marine-propulsion power plant designated as MGT-10. This power plant has the same general requirement of high thermodynamic efficiency over a wide range of performance as the stationary power plant which we have investigated.

Heat Exchangers

The analyses used for heat exchangers were conventional.

DESCRIPTION OF POWER PLANT AND COMPONENTS

Discussion of Power Plant

The main components of the nuclear, gas-turbine, power plant are shown in Layout 1. A profile and an end view show relative spacing of the units. In Layout 2 is shown a cross section of a regenerative gas turbine unit and in Layout 3 is shown a layout of the reactor heat exchanger unit. An artist's conception of the plant, Layout 4, supplements these layouts. The flow diagram for the power plant system is shown in Figure 8.

All the high pressure equipment, such as the reactor-heat exchanger, helium storage tanks and helium processing equipment are placed at the basement level as a safety precaution. The reactor container is located to one side of the first floor power plant equipment for accessibility to the reactor and to the control room.

The power plant building, which is required for colder climates, is approximately 70 feet wide by 75 feet long by 30 feet high. The gas turbine power plant with the generators, starting motors, and various accessories takes up a floor space of about 2000 square feet on a flat deck.

A summary of power plant performance figures is shown in Table 4. The over-all thermal efficiency for the power plant is in the vicinity of 30 percent. Figure 9 shows the variation in over-all efficiency with part load conditions. As can be seen the variation over the operating range is of the order of one percent. Variations from standard-day conditions (80^oF) will change the over-all efficiency, this being higher for lower temperatures (about 39 percent at -40^oF). Figure 10 shows power plant efficiency and Figure 11 shows power output as functions of compressor pressure ratio and ambient temperature for the regenerated gas-turbine power plant.

Reactor

The reactor used as a heat source for the power plant is shown schematically in Figure 1. The schematic arrangement of the helium loop in which the reactor is set is shown in Figure 12. The fuel element on which the reactor is based is shown in Figure 13. The reactor container is shown in Figure 14 and the complete helium loop layout is shown in Layout 3. The reactor core is a cylindrical annulus which is split into truncated conical sections for easier handling in installation and removal. The core is surrounded, where possible, by a layer of boron carbide, which acts as an absorber of thermal neutrons to limit activation of the reactor shell.

The reactor is cooled by the flow of helium. At inlet to the reactor the helium flow is separated into two paths. The major portion of the flow, 86#/sec., cools the core containing the heat generating fissionable material and also cools the outside reflector-moderator material before entering the hot side duct to the heat exchanger. The second path with a flow of 5#/sec., cools the central moderator island.

The reactor is encased in a double walled, 10 foot diameter, spherical shell. The pressure shell enclosing the reactor and passages to the heat exchanger is 2 inches thick. A portion of the cooler helium stream is directed on the inside of the shell to maintain it at a temperature lower than 1400°F. The thermal stress is minimized across the thickness of the shell by insulating with 5 inches of diatomaceous earth. The insulation performs a twofold function of minimizing the temperature differential across the shell thickness and also limiting the heat losses. The insulation around the reactor is also essential to provide a reasonable concrete shielding temperature. There are essentially no longitudinal thermal stresses in the shell because it is free to move axially.

The outer container is a welded assembly of four pieces; a cap (4) which permits access for refueling, a top (2) and bottom (1) section and a control rod enclosure section (3). These are shown in Figure 14 with their respective weld lines.

The standard spherical reactor used for purposes of the calculations of nuclear characteristics is shown in the sketch of Figure 2. This reflector-moderated reactor is composed of a spherical, annular, reactor core surrounded, inside and out, by reflector-moderator of either graphite or beryllium. The reactor core has fuel elements composed of uranium carbide in a graphite matrix, with void spaces for circulation of helium coolant. The nuclear characteristics reported cover the standard reactor shown in Figure 2 and several variations from the standard configuration.

Table 1 shows the nuclear characteristics, as well as the over-all performance, of a graphite, reflector-moderated reactor, using 20 percent enriched uranium (20 percent U-235, 80 percent U-238) and cooled with helium. The nuclear characteristics are based on the standard reactor configuration shown in Figure 2. The total mass of uranium (U-235 and U-238) required to make this reactor critical is 58 kg. With the limitation of 25 percent by volume of uranium carbide contained in the graphite matrix this standard reactor configuration is capable of carrying 1860 kg of uranium. The part of the temperature coefficient for this reactor which is due to a change in the base for thermal neutrons is negative, and is about (at this temperature)

$$\Delta k/k = -2.8 \times 10^{-5} \text{ per } ^{\circ}C$$

In this reactor 37 percent of the fissions are caused by thermal neutrons, 63 percent by epithermal neutrons. The ratio of maximum to average power in the reactor core is about 1.22. The change in the mass of U-235 required to cause a given change in multiplication factor is given by

$$\Delta M/M = 160 \ \Delta k/k$$

Figure 3 shows the neutron fluxes at several lethargies, and the thermal flux, for the standard graphite reactor. Figure 4 shows the distribution of power density in the reactor core of the same reactor and Figure 5 shows the distribution of neutrons causing fissions among the lethargy groups as well as thermal fissions. Table 2 shows the dimensions and the nuclear characteristics of graphite, reflector-moderated reactors for several variations in geometry. Table 3 shows essentially the same variations for beryllium moderated reactors.

Figure 6 shows the flux distribution for a standard beryllium, reflector-moderated reactor having 10 percent enriched uranium for fuel. Figure 7 shows the fission distribution for the same reactor. In this reactor about 47 percent of the fissions are caused by thermal neutrons. The ratio of maximum to average power in the core is about 1.13. The part of the temperature coefficient due to a change in the thermal base is positive, at this temperature, and is given by

$$\Delta k/k = +3.7 \times 10^{-5} \text{ per } ^{\circ}C$$

The change in the mass of U-235 required to cause a given change in multiplication factor is given by

$$\Delta M/M = 326 \Delta k/k$$

The following nuclear effects of changes from the standard configuration of Figure 1 are worth noting. An increase in the density of the graphite moderator from 1.7 to 2.0 results in an increase of about 0.04 in multiplication factor. Increasing the graphite reflector thickness by a factor of 1.5 increases the multiplication factor by about 0.05. Increasing the over-all size of the graphite-moderated reactor by a factor of 1.5 (keeping dimensional similarity) with the same amount of fuel increases the multiplication factor by 0.07. Increasing the over-all size of the graphite-moderated reactor by a factor of 1.5, with the same amount of fuel (and the same volume of fuel space as in the standard reactor distributed about the same mean diameter) increases the multiplication by about 0.12. Adding 10 atomic parts per million of boron -10 to the graphite moderator resulted in a decrease of about 0.02 in multiplication factor. Removing the graphite from the fuel region (no moderator) flattens the power distribution curve (maximum to average power ratio = 1.19)

Intermediate Heat Exchanger

The performance figures for the intermediate heat exchanger (helium to air) are shown in Table 5. The heat exchanger is a one-pass counterflow shell and finned tube type.

The entire reactor, heat exchanger, and helium fan are housed in a gas tight casing. The helium passes through the center and around the reactor and enters the heat exchanger tubes at 1740 F. These tubes are spaced in an annulus around the center return duct. The cooled helium, 1240 F, leaving the exchanger then enters an eliptical plenum which houses the fan. In this plenum the helium rotates and accelerates through a 180° turn, through the fan and back to the reactor core

Helium Fan

The fan used to circulate the helium is a single-stage, axial-flow compressor designed to provide a weight flow of 91 lb. per sec. at a pressure ratio of 1.0065. The pressure losses in the helium system which must be overcome by the fan are shown in Table 7. The performance characteristics of the helium fan are shown in Table 8. The fan is driven by a 4800 volt, 1000 HP, variable-speed induction motor, capable of a 50 percent reduction in speed from its maximum speed of 3,600 rpm. For shutdown cooling the fan will be driven by a small variable speed motor, coupled directly on one side to the fan shaft, and, on the other, by a clutch to the 1000 HP motor. Figure 15 shows the efficiency and pressure ratio of the fan as a function of weight flow of helium, for various fan speeds.

Helium Storage Tank

The power plant requires a maximum of 100 pounds of helium to deliver its rated output. Storage drums are provided with an excess capacity of helium. The helium is stored in tanks 3 feet in diameter and 8 feet high (at a pressure of 1800 psi). These tanks are of welded construction with dished heads. They are made of a low carbon steel with a wall thickness of about $1\frac{1}{2}$ inches. The storage tanks are indicated in Layout 1.

The pressure required in the reactor-heat exchanger vessel is 6 atmospheres at room temperature. This pressure is simply obtained by throttling from the 1800 psi storage tanks. At 6 atmospheres the pressure vessel contains the necessary 100 pounds of helium. When the unit is brought up to its operating temperature the pressure will reach 20 atmospheres.

Gas-Turbine Unit

The gas-turbine power plant used for this study, and shown in Layout 2, was scaled as directly as possible from a Packard-designed 10,000 HP marine-propulsion power plant designated at MGT-10. This power plant has the same general requirement of high thermodynamic efficiency over a wide range of performance as is required for the stationary power plant for this investigation. The helium fan and helium-air heat exchanger required new designs. The performance of the power plant is shown in Table 4.

The power plant is composed of three independent units (compressor, turbine, regenerator) operating in parallel. The compressor is an eight-stage, axial-flow machine based on an NACA transonic compressor. The turbine consists of two parts, one coupled to the compressor, the other to the external power plant load. The compressor turbine has two axial stages, the power turbine three.

The gas-turbine regenerator (recuperator), shown in Figure 16, is a plate and fin heat exchanger having a two-pass cross-counterflow arrangement. Performance characteristics of the recuperator are shown in Table 6. The unit is made up of 28 separate core modules, 14 in each pass. The modules are shown schematically in Figure 17. The plate and fin arrangement is shown in Figure 18. In the figures and tables the nomenclature, air side and gas side, refer to the air from the compressor and the air from the turbine, respectively.

Startup and Shutdown Procedure

Special equipment and special procedures are required to start and stop a nuclear, gas-turbine, power plant. Because the 1,000 HP drive motor is not capable of stable operation below one-half speed it is necessary to provide a smaller motor on the same shaft to drive the fan at lower speeds. Because it is not feasible to operate a complete gas-turbine unit for minor cooling purposes a separate cooling fan is needed for circulation of air through the intermediate heat exchanger. In addition, a starting motor is needed for each of the three parallel, gas-turbine units.

The starting procedure is probably about as follows. The helium loop is filled with the desired amount of helium, and purged of contaminants. With the helium fan turning slowly to circulate a small amount of helium, the reactor is made critical and brought to approximately operating temperature. During this time the cooling fan in the air stream also is operated to prevent overheating of the intermediate heat exchanger. Next the starting motor on one of the gas-turbine units is engaged to bring it up to idle speed, the helium fan being kept under proper operating conditions in the process. From this point one gas-turbine unit can be brought to operating speed and power, or additional units can be started. A control system is needed which will prevent excessive temperatures in the reactor and intermediate heat exchanger, as well as control the speeds of the helium fan and gas turbine units.

After shutdown of the reactor a significant amount of power is generated by gamma and beta radiation. After the reactor has been in operation for a year or so at full power, about six percent of full power is generated immediately after shutdown decreasing to about one-half of one percent at the end of a day. This production of power after shutdown requires that the reactor core be provided with cooling to prevent overheating. This cooling can be provided in the present helium circuit by running the helium fan and the air cooling fan at low speeds. The rate of cooling is controlled by the amount of helium and air circulated which are functions of the density of the helium and the speeds of the helium and air fans.

If the reactor core is to be removed it is necessary that the level of power generation be low enough at the time of removal to allow time for removal without an excessive temperature rise. Figure 19 shows the rise in temperature in a fuel element, originally at 400°F and cooled only by radiation, as a function of time after cooling stops for different times after reactor shutdown. Since graphite oxidizes rapidly above 660°F the fuel element cannot be allowed to exceed this temperature. At the end of one week, Figure 19 indicates that about $2\frac{1}{2}$ hours would be available for removal of the core to some external, cooled, storage unit.

A schematic arrangement of a control system for the power plant is shown in Figure 20.

Fuel Burnup

The reactor for this power plant, operating at 60 megawatts of heat output, consumes about 22 kg per year of fissionable material. If 10 percent burnup of fissionable material is allowable it is then necessary to have 220 kg of fissionable material in the new reactor to provide for continuous operation of the reactor for one year at full power, assuming a uniform rate of burnup of fuel throughout the reactor core. In general, the power rate of burnup is not uniform (see Figure 4) and the lifetime of the core is primarily determined by the maximum power density. Removing the moderator from the core region greatly reduces the maximum average power density (Table 2, Reactor 38).

Estimate of Costs

The cost estimate for the gas cooled reactor power plant is shown in Table 9. It is broken down into prototype, engineering and design, and testing and development costs. The source of the cost data is as follows. In evaluating the prototype costs, full advantage was taken of all existing data for the MGT-10, and where it was possible this data was extrapolated to approximate the costs of the gas turbines and their associated equipment. The cost of the generators and some of the miscellaneous equipment is based on quotations from vendors as are the costs of the reactor-heat exchanger container and components. The reactor fuel cost is based on an extrapolated unclassified cost figure for U-235 and U-238. The outlay for the structure housing the power plant is based on an approximate dollars per square foot figure. Miscellaneous components and equipment were based on an estimated dollars per pound basis. The engineering design and testing and development costs were estimated from the MGT-10 program where possible. Other numbers were estimated on a man-hour basis with current Studebaker-Packard cost information.

The total cost for the first power plant, including development, is estimated at $11,710,000, or $781 per KW. The engineering design and testing and development costs would not appear in the cost of a second unit. Also, the special equipment necessary for building the first unit would be available for succeeding ones. Utilizing the 83% rule, which is customary in the airframe industry to relate unit cost to quantity produced, the cost of a second unit would be substantially under $5,000,000, or under $325 per KW.

EXPERIMENTAL RESULTS

The experimental work carried out under this contract was done at the Santa Barbara facility of Aerophysics Development Corporation, a subsidiary of Studebaker-Packard Corporation.

Preliminary Effort

The experimental work described herein was of a preliminary nature, the intent being to point the way for further exploratory effort. Consequently, the experimental program was subject to a limitation of budget, in time and money, which precluded an intensely quantitative attack. This limitation also required certain compromises in order to utilize materials and equipment that were most readily available.

Scope

Two items were investigated that are considered basic in the proposed power-plant design:

1. The rate of diffusion of helium through stainless steel at high temperature.
2. The rate of erosion of graphite at high temperature by a high velocity helium stream.

Helium Diffusion

Details of the procedure and equipment used in the helium diffusion tests are discussed in Appendix 2. Briefly, the diffusion tests were carried out as follows. One-half inch O.D. thin wall tubes of types 304 and 347 stainless were pinched off and welded at room temperature and pressure, (thereby containing air at one atmosphere pressure, absolute, at the start of the tests). Surrounding the one-half inch O.D. tube was another stainless steel tube spun down and welded to the one-half inch O.D. tube, thus forming an annular helium manifold. Two such test assemblies of each material were prepared - one for diffusion testing at room temperature, the other at high temperature. Lengths of tubing used and placement of welds was such that all welds would be outside the heated zone in the high temperature tests.

Initial operation of the high temperature tests was attempted at 1800°F and 10 atmospheres helium pressure. However, the outer tube soon failed, so to be on the safe side, the temperature was reduced to 1200°F and helium pressure to 5 atmospheres, gauge. While both the temperature and pressure are below that of the design condition for full load operation of the power plant, a comparison of 1200°F versus room temperature was believed to be a good compromise in view of the materials available and the short time remaining. Also, 1200°F is in the critical range for intergranular carbide precipitation, whereas 1800°F is some 200°F above the precipitation range. The results described in Appendix 2 indicate that the diffusion rate of helium through stainless steels will not be significant.

Graphite Erosion

Details of the procedure and equipment used in the graphite erosion tests are discussed in Appendix 2. Briefly, these tests were carried out as follows. Samples of HPC-7, HP3LM, and HLM graphite, furnished by Great Lakes Carbon Company, were subjected to the impact of a high velocity helium stream for a matter of several hours. Tests were conducted at room temperature and at a graphite temperature of approximately 2500°F, (helium total temperature about 700°F in these hot tests).

To conserve helium, a small nozzle was used (#58 drill), in both the room temperature tests and the hot tests. This nozzle was mounted with its axis normal to the graphite sample, and the nozzle exit plane was approximately 0.1 inch from the graphite surface. Greater than critical pressure was maintained at all times so that the impact velocity was MACH 1 or greater (due to free-jet expansion).

The results of the erosion tests were:

1. In the room temperature tests, no observable erosion after in excess of eight hours of testing.
2. In the high temperature tests, no observable erosion after some five hours of testing.

It is believed that these tests were quite severe and that it is, therefore, logical to conclude that erosion of the graphite in the design reactor will not offer a problem, unless due to secondary effects such as radiation damage or thermal cycling.

TABLE 1

GAS COOLED REACTOR

STANDARD GRAPHITE MODERATED REACTOR

Operating Conditions	
Heat Power Output	60,000 kw
Average Power Density in Core	46 watts/cm^3
Specific Power	32 kw/Kg of U-235
Helium Flow (Hot Stream)	86 lb/sec.
Helium Flow (Cooling Stream)	5 lb/sec.
Average Helium Inlet Temperature	1240 F
Average Helium Outlet Temperature	1740 F
Helium Outlet Temperature (Hot Stream)	1760 F
Helium Outlet Temperature (Cooling Stream)	1370 F
Maximum Temperature in Reactor Core	2420 F
Helium Pressure in Reactor (Maximum Power)	20 atmos.
Helium Pressure Drop in Reactor	0.404%
Dimensions	
Island Moderator (Graphite)	40 inch O.D.
Reflector Moderator (Graphite)	60 inch I.D.; 100 inch O.D.
Fueled Core (Uranium Carbide, Graphite)	40 inch I.D.; 60 inch O.D.
Pressure Vessel (Inconel)	10 feet diameter
Cooling Passages in Core	1260, one inch dia. holes spaced 1.39 inch on a triangular grid.
Free Flow Ratio in Core	0.47

TABLE 1 (CONTINUED)

Materials

Graphite in Island	2058 lbs.
Graphite in Reflector	25,216 lbs.
Graphite in Core	2570 lbs.
Uranium Carbide in Core	141.7 lbs.
Uranium-235 in Reactor	25.7 lbs.
Uranium-238 in Reactor	102.8 lbs.
Fuel Enrichment	20 percent
Consumption of U-235	62.8 gm/day
Conversion Ratio (to PU-239)	0.0645
Operating Time for 15 Percent Burnup	28 days

TABLE 2

NUCLEAR CHARACTERISTICS OF REACTOR VARIATIONS

Non-Critical Reactors
Moderator - Graphite (Average Density 1.7 gm/cm^3)

REACTOR NUMBER	COMMENTS	RADIUS (INCHES) TO OUTSIDE OF: ISLAND	FUEL	MODERATOR	% FUEL ENRICH.	MASS U-235 M (kg)	MULTIPLICATION FACTOR (k) FOR MASS M @ THERMAL TEMP. T_1=3425F	T_2=2723F	MASS U-235 $M_{5\%}$ (kg)	MULTIPLICATION FACTOR $k_{5\%}$ (k = 1.0 ± 5%) FOR MASS $M_{5\%}$ @ THERMAL TEMP. T_1=3425F	T_2=2723F	POWER RATIO IN CORE MAX. POWER / AVG. POWER	CON-VERSION RATIO	FISSION RATIO U-238 / U-235
1	Standard Graphite Reactor	20	30	50	100	126.7	1.3353	1.3523	9.142	0.9598	0.9910	1.1940	-	-
2					20	253.5	1.1136	1.1245	11.682	0.9857	1.0148	1.2176	0.0645	0.0014
3					10	507.1	0.97062	0.97765	558.69	0.9821	0.9889	3.2484	0.9133	0.1525
4	No Central Island - Same Fuel Volume	0	26.6	50	100	164.3	1.3414	1.3523	6.843	0.9314	0.9607	1.0955	-	-
5					20	328.6	1.0736	1.0807	91.240	1.0404	1.0519	2.1946	0.2997	0.0135
6					10	659.3	1.0009	1.0050	633.60	0.9919	0.9962	3.2594	0.7800	0.3320
7	Reflector 1.5 x Larger	20	30	60	100	128.3	1.4237	1.4418	7.212	0.9926	1.0240	1.1404	-	-
8					20	256.55	1.1857	1.1975	8.758	1.0190	1.0489	1.1518	0.0547	0.0010
9					10	513.1	1.0228	1.0305	443.03	1.0114	1.0196	2.7093	0.8383	0.1123
10	All Dimensions 1.5 x Larger	30	44.8	74.8	100	126.	1.4864	1.5124	10.212	0.9536	0.9842	1.1684	-	-
11					20	251.9	1.2845	1.3011	10.212	0.9341	0.9635	1.1697	0.0339	0.0048
12					10	486.8	1.0287	1.0390	413.59	1.0390	1.0500	2.6011	0.7937	0.0128
13	Boron Carbide Absorber in Center 2.22" O.D.	20	30	50	100	126.8	1.3296	1.3464	10.158	0.9820	1.0129	1.1960	-	-
14					20	253.5	1.1089	1.1196	11.682	0.9786	1.0073	1.2109	0.2753	0.0177
15					10	507.1	0.96670	0.97358	558.69	0.9787	0.9853	3.1827	0.9398	0.1507
16	Thermal Temperatures T_1= 1672 F T_2= 1283 F	20	30	50	100	126.8	1.3476	1.3432	9.650	1.0316	1.0465	1.2251	—	—
17					20	253.5	1.1201	1.1167	10.666	1.0168	1.0297	1.2394	0.0599	0.0014
18					10	507.1	0.97394	0.97164	558.69	0.9852	0.9830	3.4905	0.9517	0.1508
19	1.5 x Larger With Same Fuel Volume	35	40	75	100	141.6	1.4751	1.4985	10.805	1.0011	1.0299	1.0599	-	-
20					20	283.2	1.2945	1.3105	10.350	0.9644	0.9916	1.0596	0.0393	0.0007
21					10	566.5	1.1061	1.1169						
22	High Density Graphite in Reflector-Moderator ρ= 2.0 gm/cm^3	20	30	50	20	253.5	1.1719	1.1837	7.619	0.9714	1.0018	1.1828	0.04989	0.0009
23					100	126.8	1.4147	1.4328	6.095	0.9276	0.9590	1.1672	-	-
24					10	507.1	1.0041	1.0117	482.49	1.0004	1.0081	3.2594	1.0154	0.1200
38	Fuel Region Has No Moderator	20	30	50	20	253.	1.0573	1.0531	203.64	1.0455	1.0414	1.1877	0.3085	0.03428
48	Boron Impurity Added to Graphite 10 PPM	20	30	50	100	126.7	1.3012	1.3144	12.219	0.9763	1.0007	1.2142	-	-
49	Central Island Removed	0	30	50	100	90.	1.4012	1.4185	5.774	0.9647	0.9991	1.3369	-	-

TABLE 3

NUCLEAR CHARACTERISTICS OF REACTOR VARIATIONS

Non-Critical Reactors
Moderator - Beryllium (Average Density 1.6 gm/cm^3)

REACTOR NUMBER	COMMENTS	RADIUS (INCHES) TO OUTSIDE OF ISLAND	FUEL	MODERATOR	% FUEL ENRICH.	MASS U-235 M (kg)	MULTIPLICATION FACTOR (k) FOR MASS M @ THERMAL TEMP.		MASS U-235 $M_{5\%}$ (kg)	MULTIPLICATION FACTOR $k_{5\%}$ ($k = 1.0 \pm 5\%$) FOR MASS $M_{5\%}$ @ THERMAL TEMP.		POWER RATIO IN CORE MAX. POWER / AVG. POWER	CON-VERSION RATIO	FISSION RATIO U-238 / U-235
							T_1=1672F	T_2=1283F		T_1=1672F	T_2=1283F			
26	Standard Beryl-	20	30	50	100	75	1.4869	1.4748	5.091	0.9915	1.0073	1.1294	-	-
27	lium Reactor				20	125	1.2904	1.2800	6.262	1.0369	1.0492	1.1436	0.0834	0.00075
28					10	229	1.0725	1.0644	5.142	0.9462	0.9610	1.1330	0.0879	0.00262
29	No Central	0	26.6	50	100	74.5	1.4532	1.4443	3.802	0.9871	1.0029	1.1017	-	-
30	Island - Same				20	123.5	1.1918	1.1841	4.562	1.0138	1.0265	1.1389	0.0540	0.00079
31	Fuel Volume				10	228	0.97689	0.97083						
32	Reflector	20	30	60	100	74.2	1.5120	1.4974	4.555	0.9832	0.9971	1.1149	-	-
33	1.5 x Larger				20	123.5	1.3120	1.2995	4.555	0.9624	0.9761	1.1160	0.0383	0.00059
34					10	228	1.0895	1.0798	5.567	1.0009	1.0114	1.1283	0.0778	0.00157
35	All Dimensions	30	44.8	75	100	250	1.5787	1.5643	8.510	-	0.9783	-	-	-
36	1.5 x Larger				20	416	1.3118	1.3002	11.063	1.0426	1.0538	1.1462	0.0381	0.00039
37					10	765	1.0639	1.0551	9.021	0.9445	0.9582	1.1341	0.0822	0.00099
39	Boron Carbide	20	30	50	100	74.8	1.4841	1.4722	5.091	0.9881	1.0040	1.1274	-	-
40	Absorber In				20	125	1.2880	1.2777	5.091	0.9659	0.9815	1.1287	0.0403	0.00065
41	Center 2.22" O.D.				10	229	1.0705	1.0625	5.142	0.9429	0.9963	1.1309	0.0905	0.00150
42	Thermal Temperatures	20	30	50	100	75	1.5073	1.4871	5.600	1.0372	1.0256	1.1278	-	-
43	T_1 = 2140 F				20	125	1.3082	1.2905	5.091	0.9802	0.9693	1.1242	0.0403	0.00066
44	T_2 = 1672 F				10	229	1.0867	1.0726	7.637	1.0713	1.0575	1.1513	0.1057	0.00202
45	1.5 x Larger	35	40	75	100	84.5	1.4857	1.4682	8.531	0.9619	0.9725	1.0475	-	-
46	With Same Fuel				20	141	1.3661	1.3493	9.953	0.9983	1.0065	1.0525	0.0379	0.00176
47	Volume				10	259	1.1733	1.1592	9.669	0.9629	0.9714	1.0525	0.0849	0.00152

TABLE 4

POWER PLANT PERFORMANCE

Performance Data	(One Unit)	(Three Units)
Power - Generated Output (kw)	5,719	17,158
Over-all Thermal Efficiency	30%	30%
Air Flow - lb./sec.	104	312
Inlet Duct Total Pressure Recovery	98.1%	98.1%
Compressor to Turbine Total Pressure Recovery	94.3%	94.3%
Turbine Discharge Total Pressure Recovery	96%	96%
Compressor Pressure Ratio	5.45	5.45
Compressor Efficiency	86%	86%
Gas Generator Turbine Efficiency	87%	87%
Power Turbine Efficiency	87%	87%
Recuperator Effectiveness	72%	72%

Station	Pressure Atmos.	Temperature -°F
Ambient	1.0	80
Compressor Inlet	0.981	80
Compressor Outlet	5.35	465
Regenerative Inlet	5.29	465
Regenerative Outlet	5.15	744
Heat Exchanger Inlet	5.14	744
Heat Exchanger Outlet	5.06	1400
Gas Turbine Inlet	5.05	1400
Gas Turbine Outlet	1.96	1053
Power Turbine Inlet	1.95	1053
Power Turbine Outlet	1.04	852
Recuperator Inlet	1.021	852
Recuperator Outlet	1.004	573
Exhaust Exit	1.0	573

TABLE 5

INTERMEDIATE HEAT EXCHANGER

Type - One-pass counterflow, shell and tube type unit with longitudinal finned tubes.

Material	Inconel
Effectiveness	68.7 percent
Air Flow Rate	312 lb./sec.
Helium Flow Rate	91 lb./sec.
Air Inlet Temperature	744°F
Helium Inlet Temperature	1740°F
Air Outlet Temperature	1400°F
Helium Outlet Temperature	1240°F
Air Inlet Pressure	5.13 atm
Helium Inlet Pressure	20 atm
Air Outlet Pressure	5.05 atm
Helium Outlet Pressure	19.98 atm
Number of Tubes	4260
Length of Finned Tube Section	120 inches
Outside Diameter	1.050 inches
Inside Diameter	0.824 inches
Tube Wall Thickness	0.113 inches
Number of Fins Per Tube	12
Fin Thickness and Height	.060 inches x .25 inches
Tube Arrangement	Triangular
Tube Spacing	1.70 inches
Shell Diameter	120 inches
Heat Transfer Area	
Air Side	26,600 ft.^{2}
Helium Side	9,200 ft.^{2}
Free Flow Area	
Air Side	21.80 ft.^{2}
Helium Side	15.75 ft.^{2}

TABLE 6

RECUPERATOR

Type - Two-pass cross-counterflow, plate and fin heat exchanger

Material	AISI 316-L Stainless Steel
Effectiveness	72 percent
Number of Cores	28
Number of Cores Per Pass	14
Weight Per Core	855 lbs.
Fin Spacing, Air Side	8 per inch
Fin Spacing, Gas Side	8 per inch
Plate Spacing, Air Side	0.5 inch
Plate Spacing, Gas Side	0.5 inch
Flow Length, Air Side	96 inches
Flow Length, Gas Side	62 inches
Flow Length Per Core, Gas Side	30 inches
No-Flow Width	144 inches
No-Flow Width Per Core	8.5 inches
Number of Rows Per Core, Air Side	8
Number of Rows Per Core, Gas Side	9
Heat Transfer Area Per Core, Air Side	1505 $ft.^2$
Heat Transfer Area Per Core, Gas Side	1505 $ft.^2$
Free Flow Area Per Core, Air Side	0.728 $ft.^2$
Free Flow Area Per Core, Gas Side	2.620 $ft.^2$
Fin Thickness	0.010 inches
Plate Thickness	0.015 inches

TABLE 7

HELIUM LOOP PRESSURE LOSSES

Helium Flow	91#/sec.
Hot Side	1740 F
Cold Side	1240 F
Pressure	20 atmos.

1 - 4* Reactor Loss - Split into two streams:

		Cooling Stream 5#/sec.	Hot Stream 86#/sec.
1	Inlet Loss	0.001	0.001
1 - 2	Core and End Piece	0.265	0.289
2	Outlet	0.135	0.034
2 - 3	Turning	Negligible	0.075
3 - 4	Outer Shell	0.002	0.002
4	Outlet Turn	0.001	0.003
4 - 5	Diffusion	0.024	0.024
	TOTAL	0.428	0.428
5 - 6	Heat Exchanger		0.100
6 - 7	Rotation and Acceleration		0.063
8 - 9	Diffusion from Fan		0.024
9 - 10	Return Duct		0.015
10 - 1	Diffusion to Reactor		0.021
1 - 10 - 1	TOTAL $\frac{\Delta P}{P}$ %		0.651
7 - 8	Pressure Rise Across Fan $\frac{\Delta P}{P}$ %		0.651

* Refer to Figure 3 for identification of points.

TABLE 8

HELIUM STAGE BLADE DESIGN DATA

	RADIUS RATIO	NUMBER OF BLADES	BLADE CHORD	SOLIDITY	DESIGN CHAMBER	THICKNESS % CHORD	ANGLE SETTING
ROTOR	1.000	29	2.92	.75	.65	.06	51.48
	.876			.856	.790	.08	46.45
	.752			.998	.975	.10	40.08
STATOR	1.000	34	3.33	1.000	1.25	.10	9.39
	.876			1.140	1.325	.10	10.72
	.752			1.330	1.392	.10	12.30

TABLE 9

COST DATA

	PROTOTYPE	ENGINEERING & DESIGN	TESTING & DEVELOPMENT	TOTAL COST	TOTAL COST $/kw
Reactor & Heat Exchanger (includes all helium circuit components)	2,485,000	1,500,000	3,380,000	7,365,000	491
Gas Turbine Power Unit (includes all ducting, recuperator, and controls)	1,200,000	750,000	900,000	2,850,000	190
Generators & Miscellaneous Equipment	515,000	-	-	515,000	34.3
Power Plant Building	180,000	-	-	180,000	12
Reactor Fuel Cost (approximated)	800,000	-	-	800,000	53.3
TOTAL	5,180,000	2,250,000	4,280,000	11,710,000	780.6

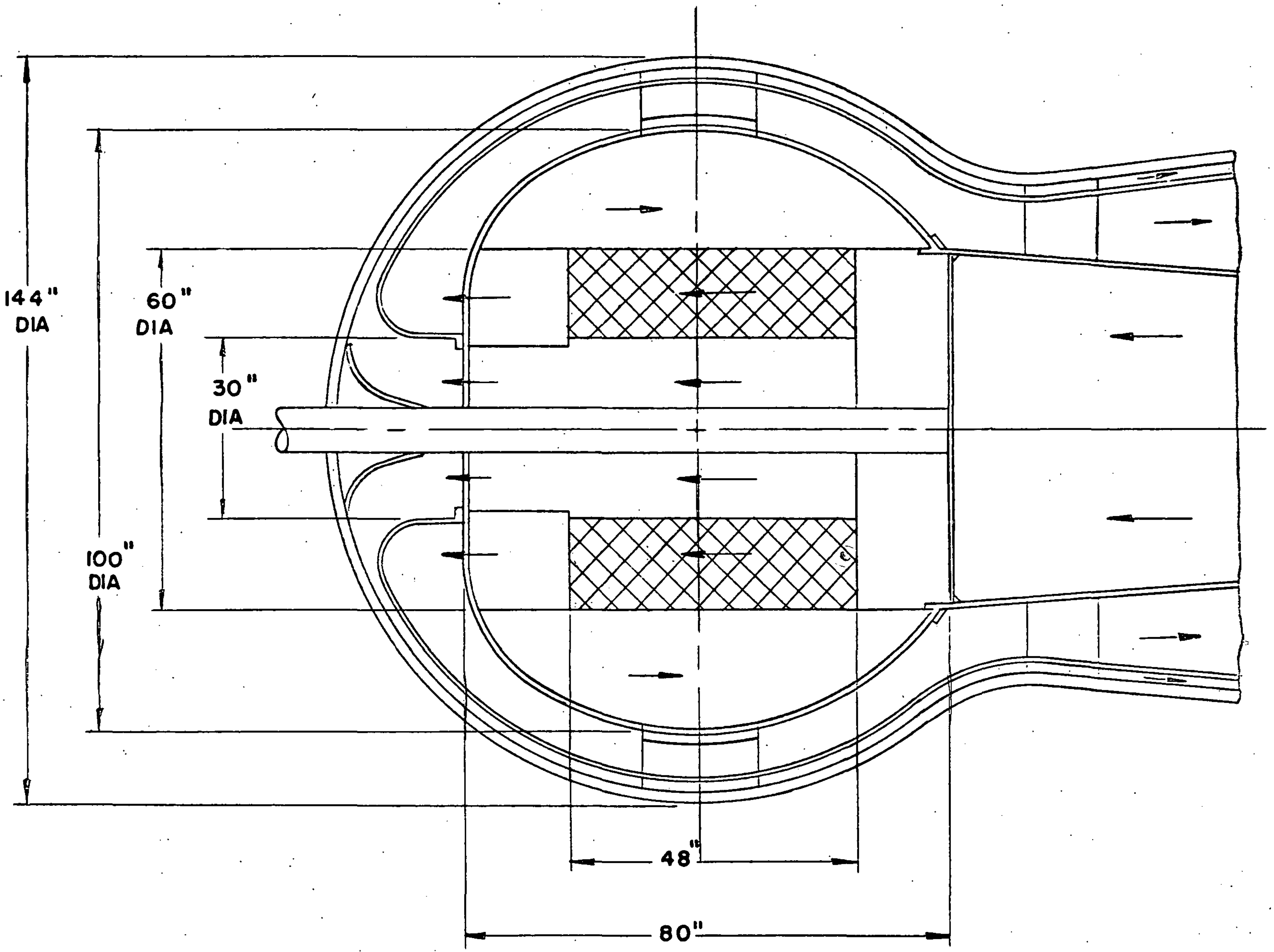


FIGURE 1

REFLECTOR MODERATED REACTOR

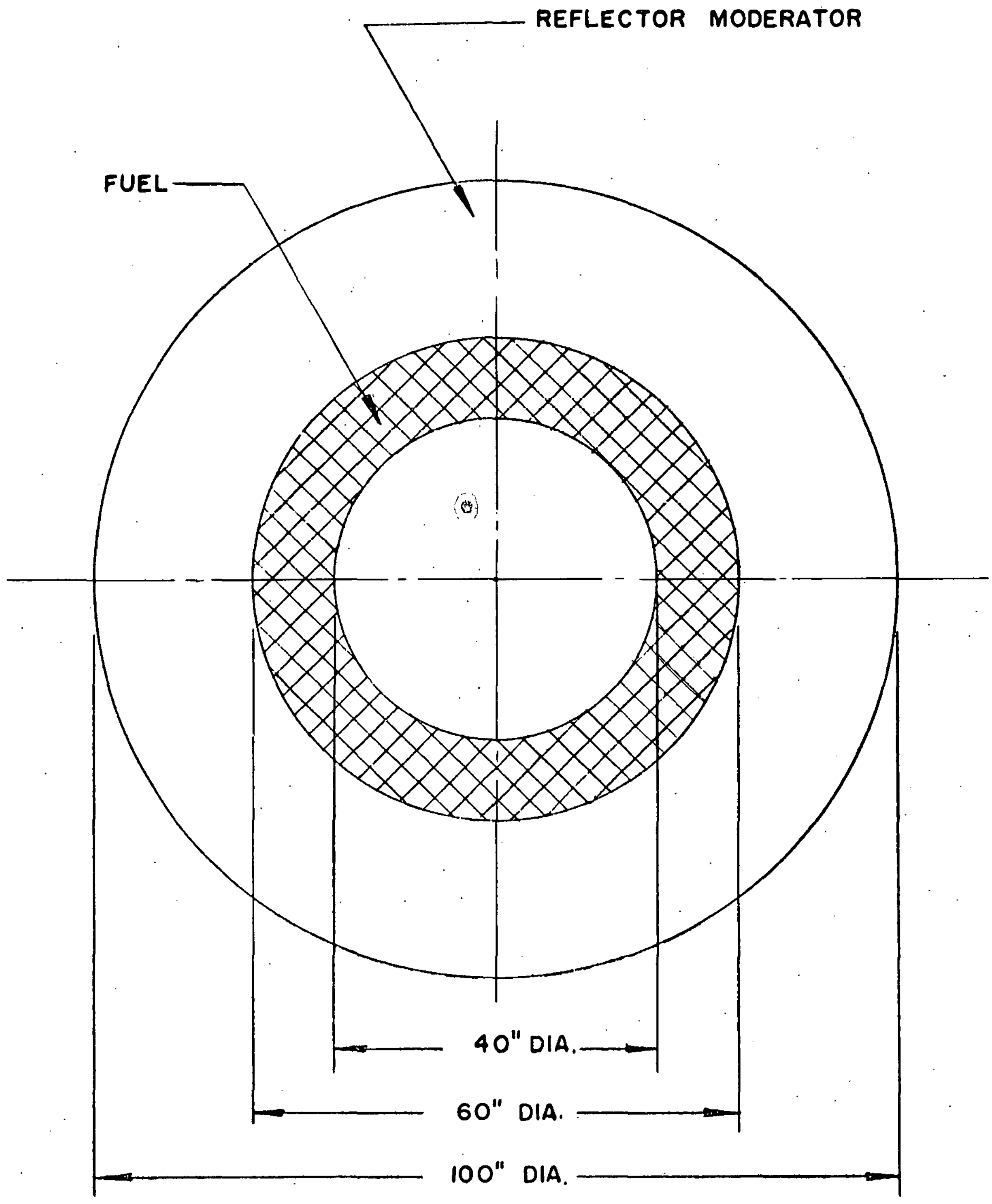


FIGURE 2

STANDARD SPHERICAL REACTOR

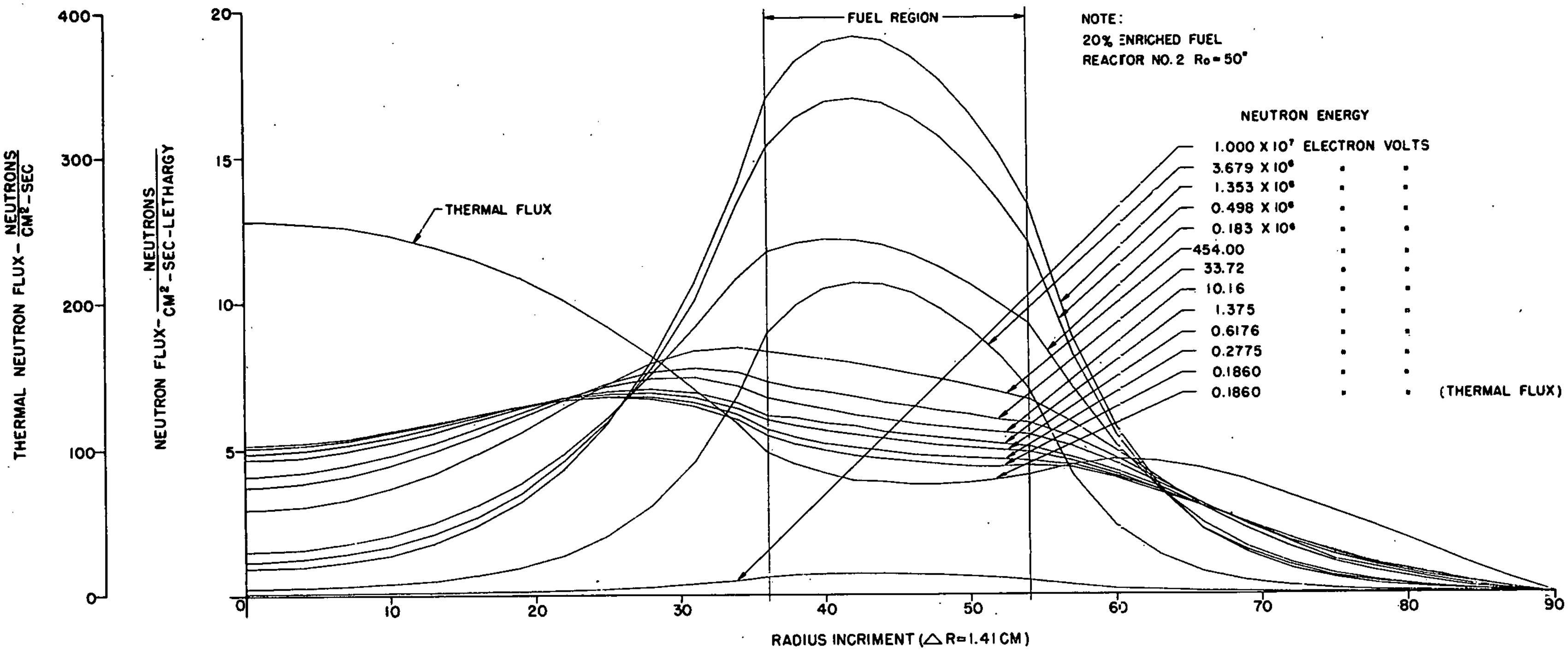


FIGURE 3

NEUTRON FLUX DISTRIBUTION
IN STANDARD GRAPHITE REACTOR

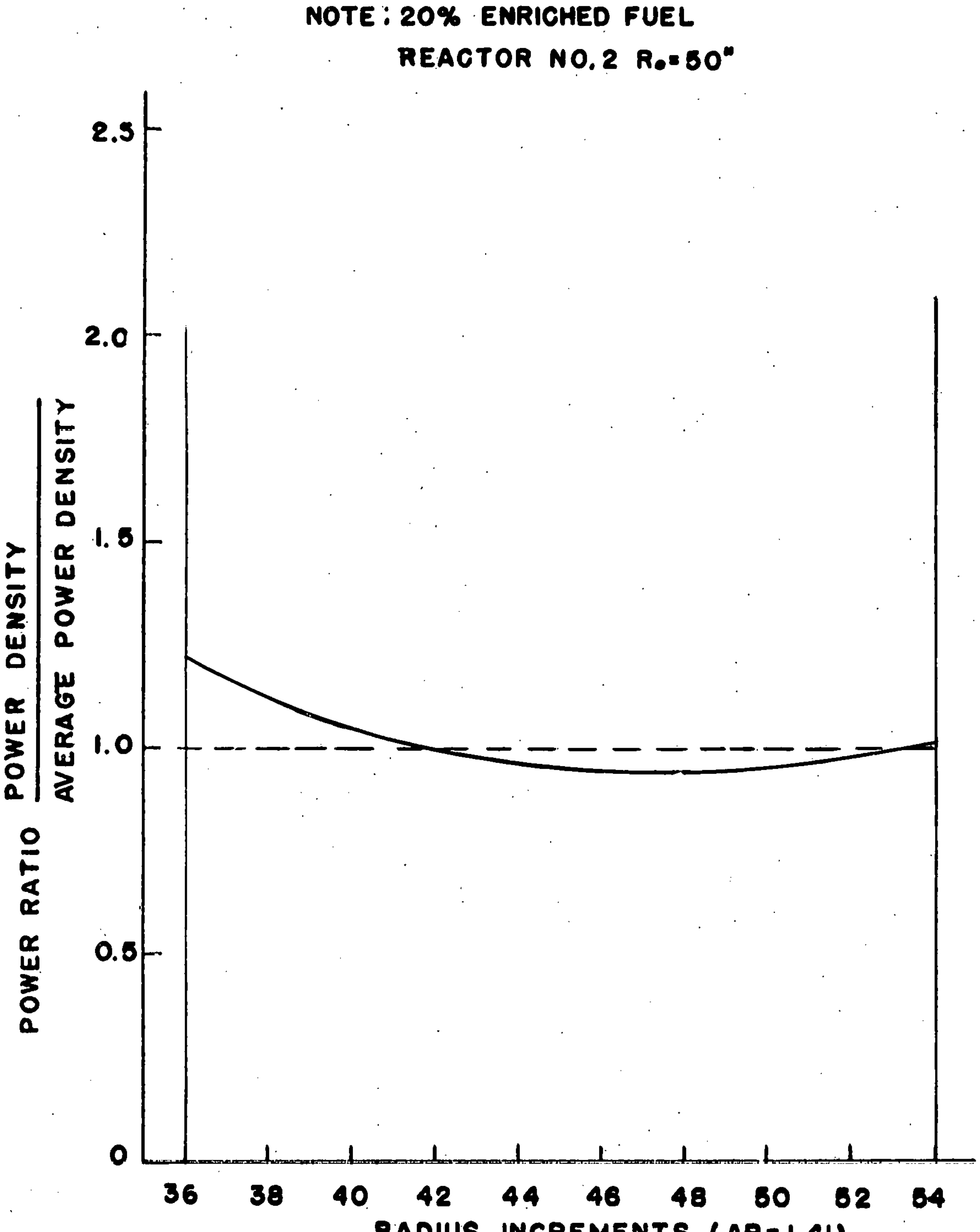


FIGURE 4

POWER DISTRIBUTION IN REACTOR CORE
FOR STANDARD GRAPHITE REACTOR

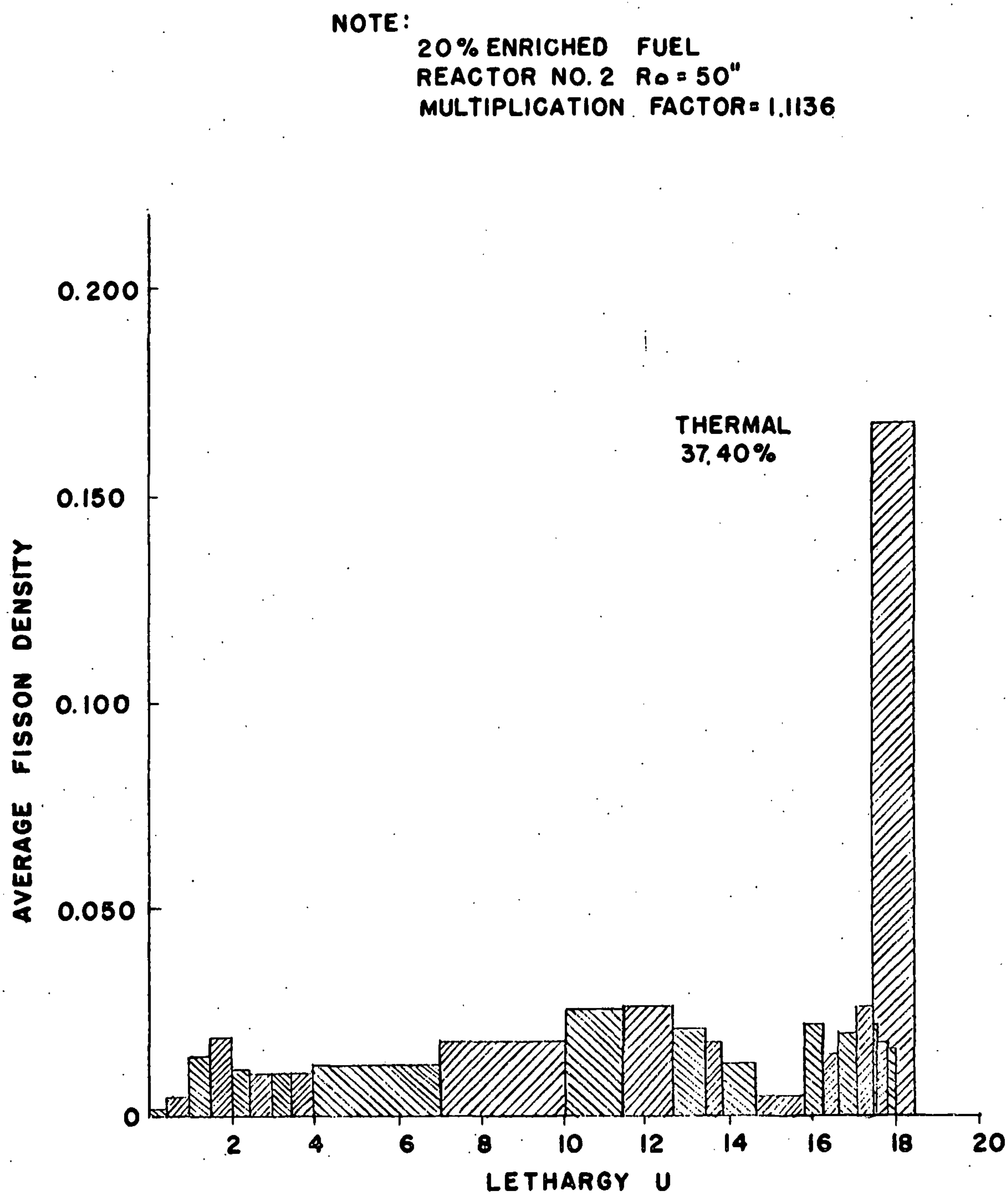


FIGURE 5

FISSION DISTRIBUTION IN LETHARGY FOR STANDARD GRAPHITE REACTOR

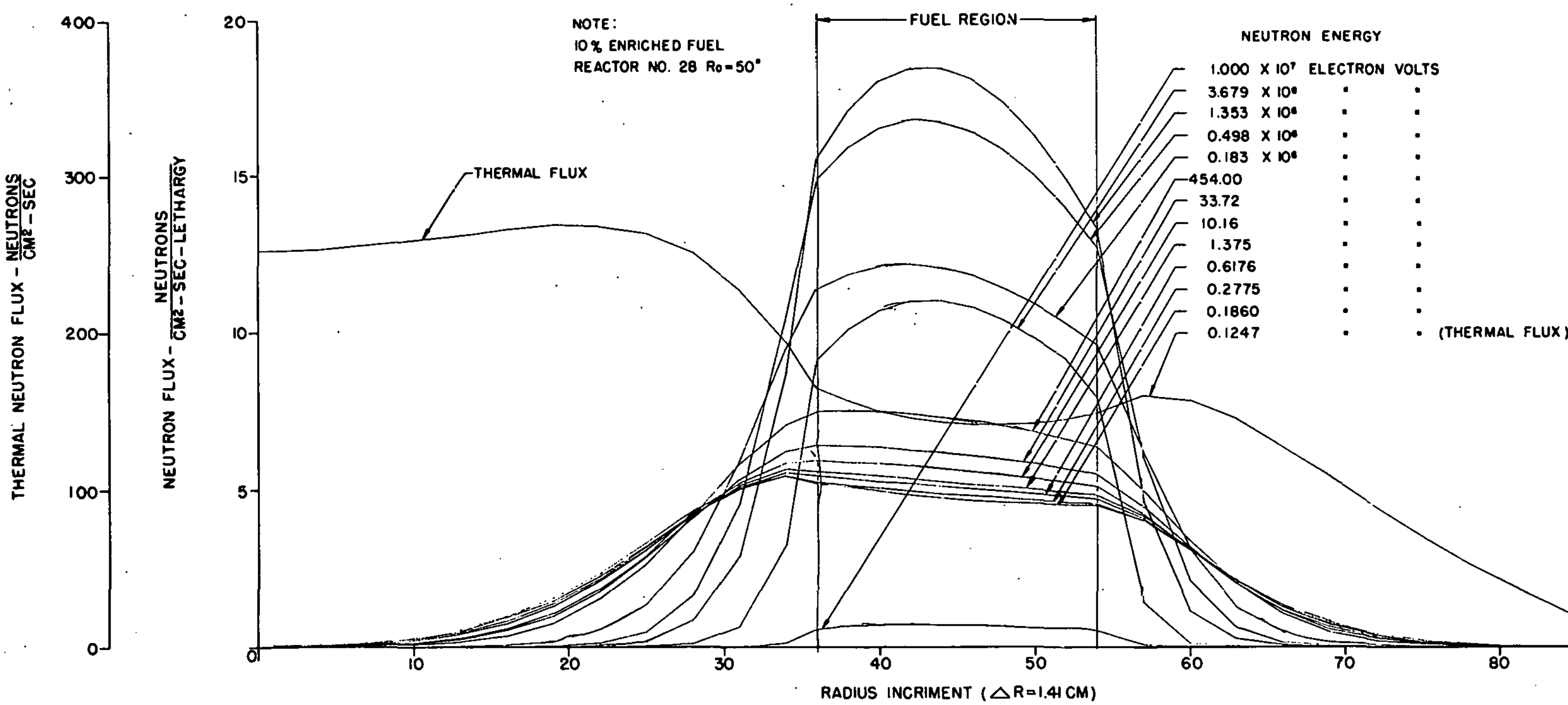


FIGURE 6

NEUTRON FLUX DISTRIBUTION IN STANDARD BERYLLIUM REACTOR

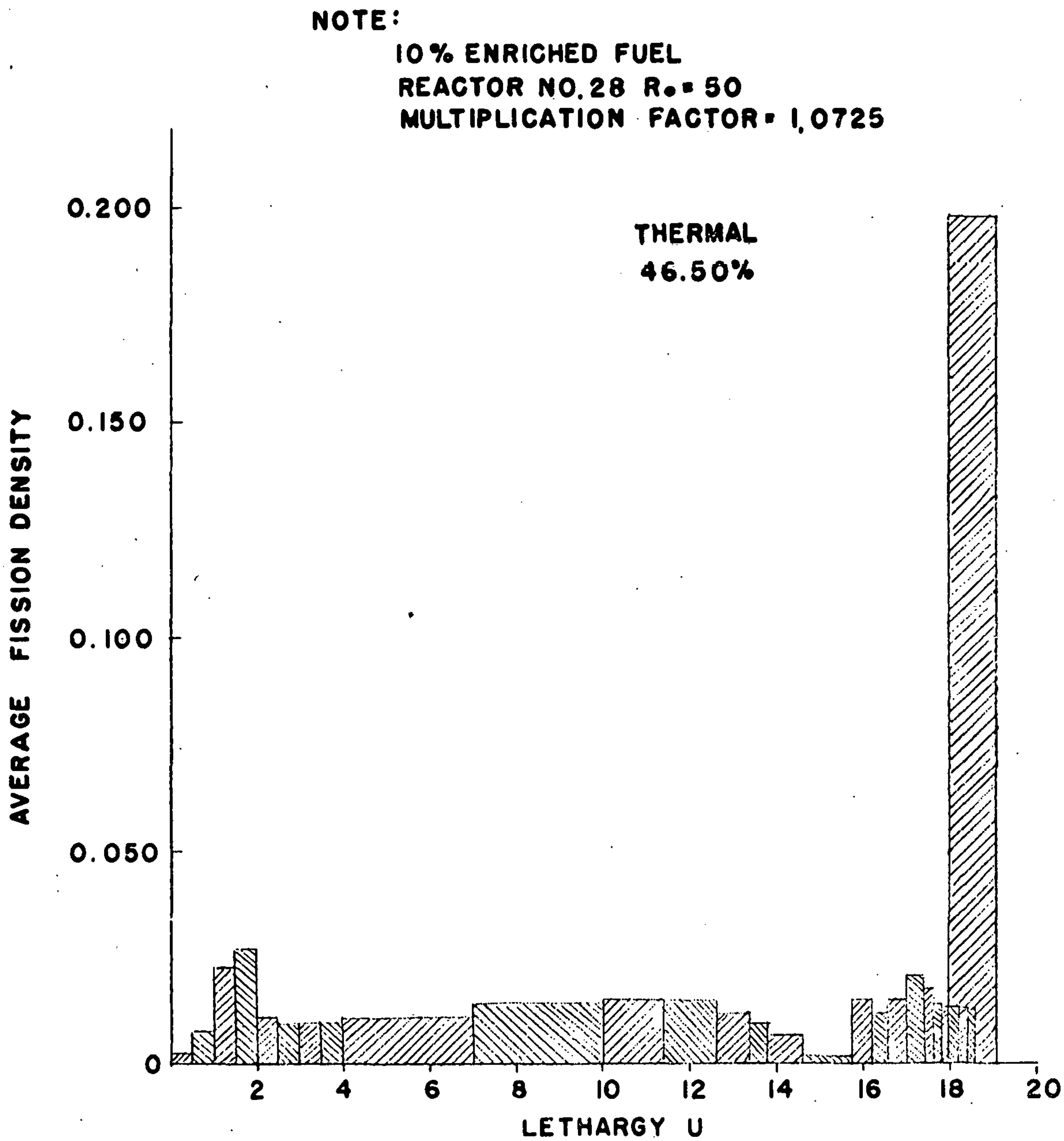


FIGURE 7

FISSION DISTRIBUTION IN LETHARGY
FOR STANDARD BERYLLIUM REACTOR

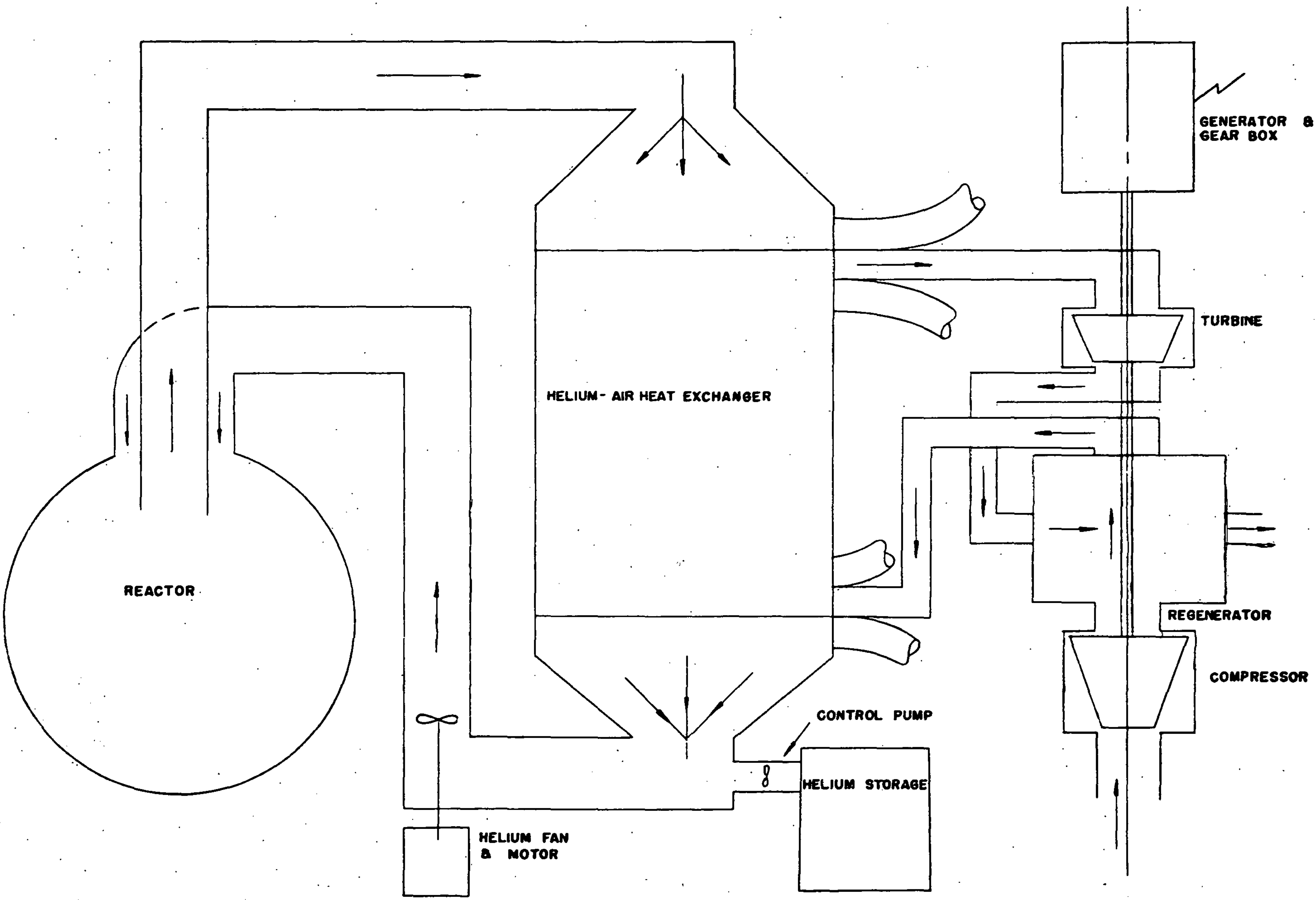


FIGURE 8
FLOW DIAGRAM. OF POWER SYSTEM

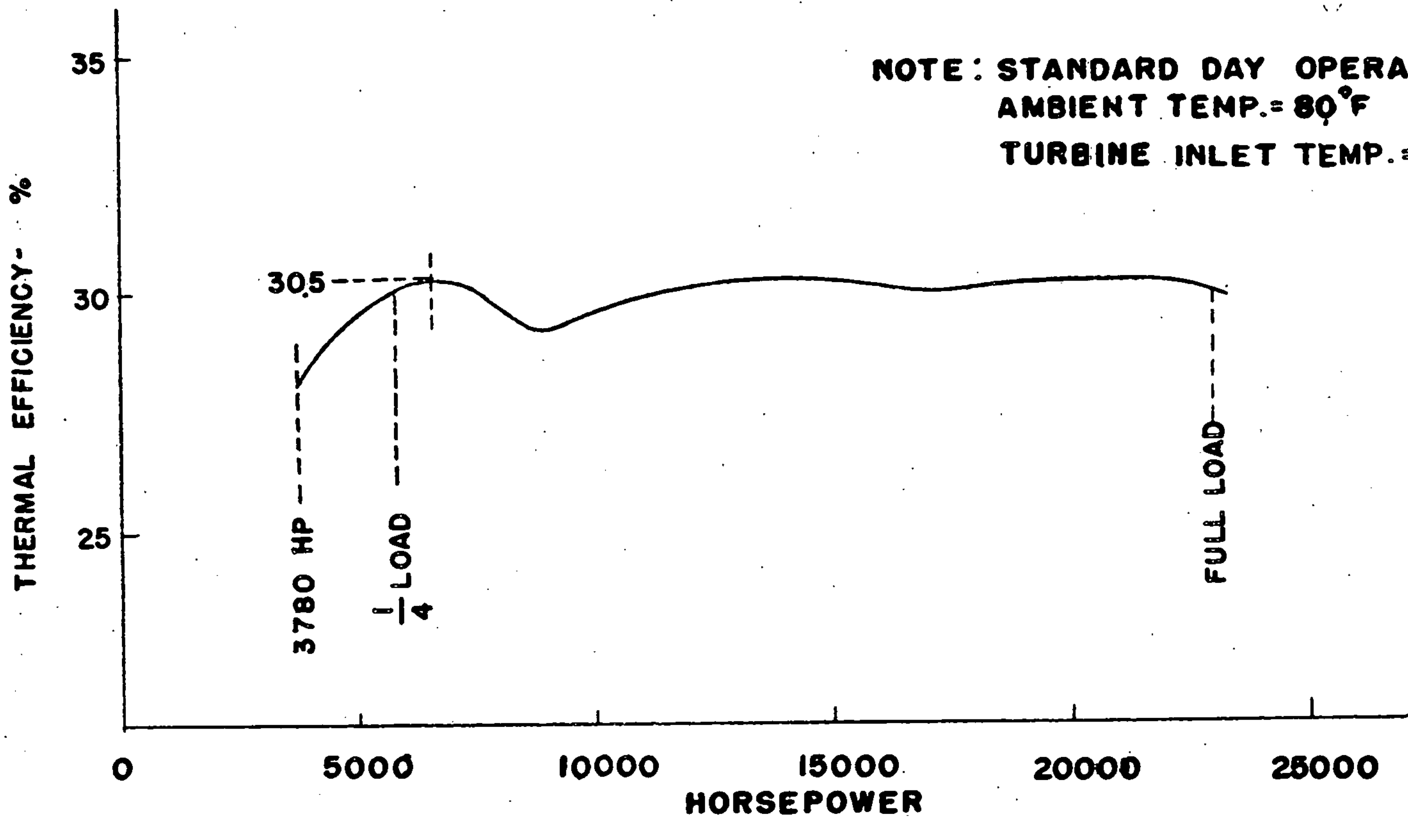


FIGURE 9

PART-LOAD PERFORMANCE OF NUCLEAR POWER PLANT

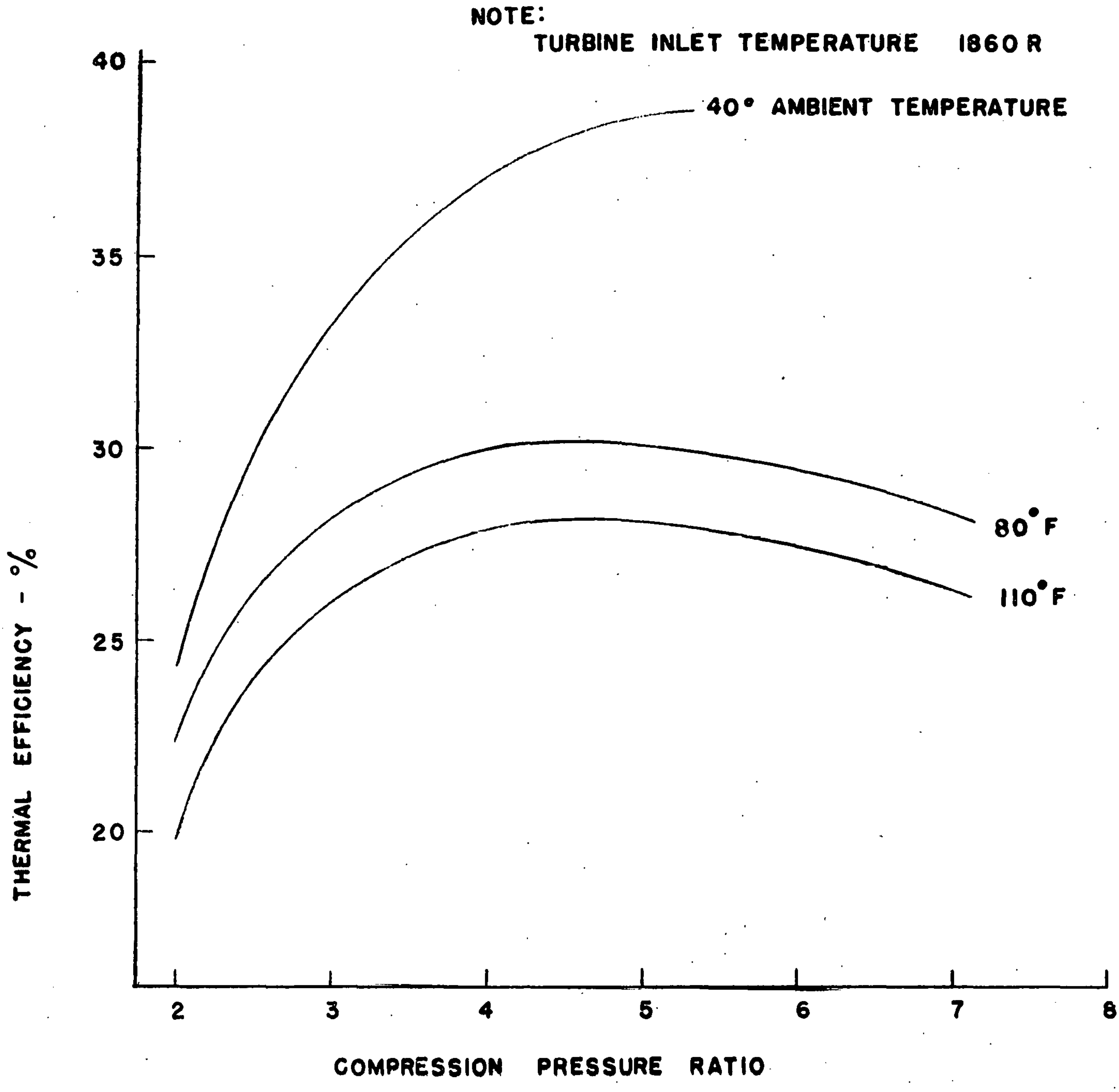


FIGURE 10

GAS TURBINE REGENERATIVE CYCLE EFFICIENCY

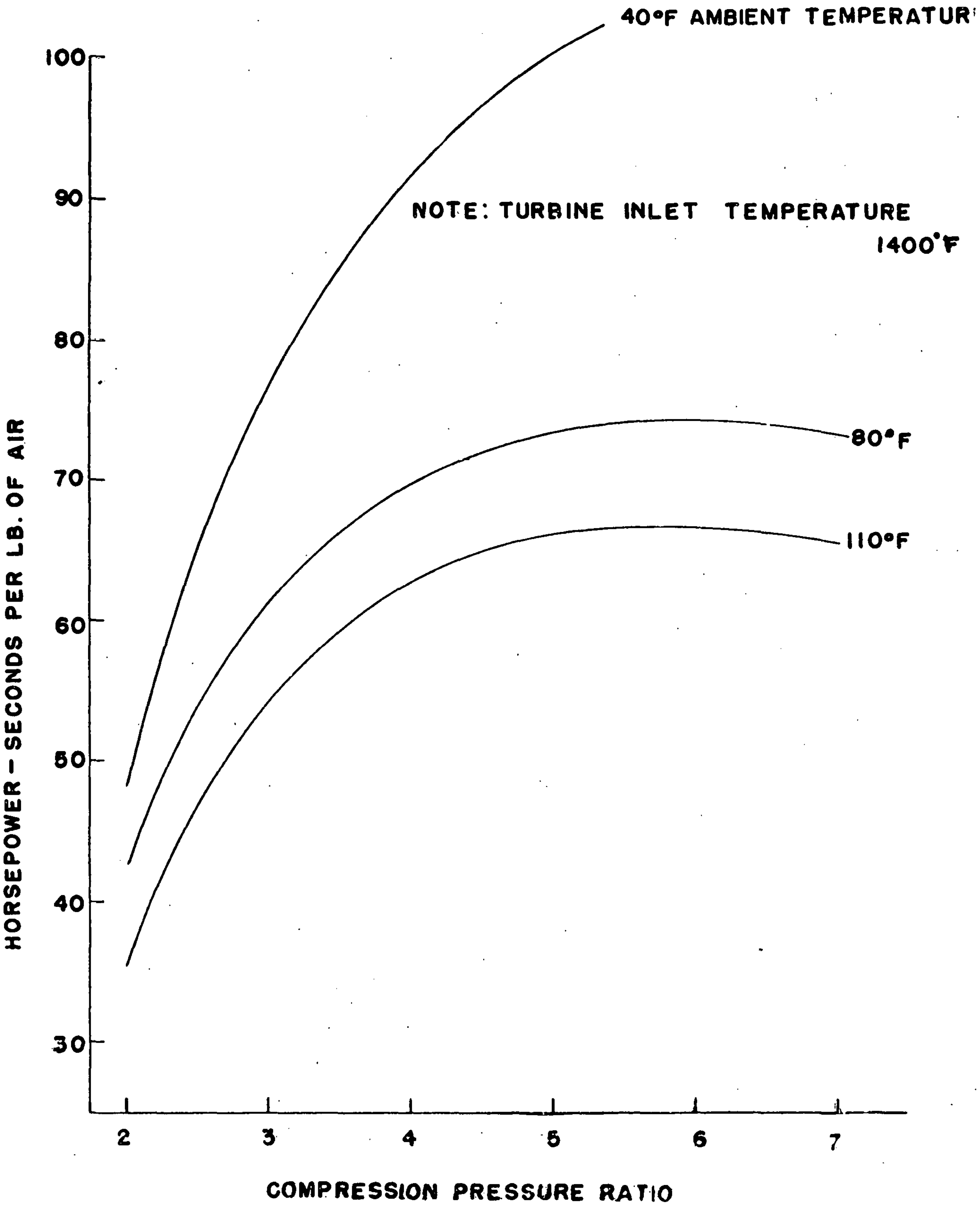


FIGURE 11

GAS TURBINE REGENERATIVE CYCLE POWER OUTPUT

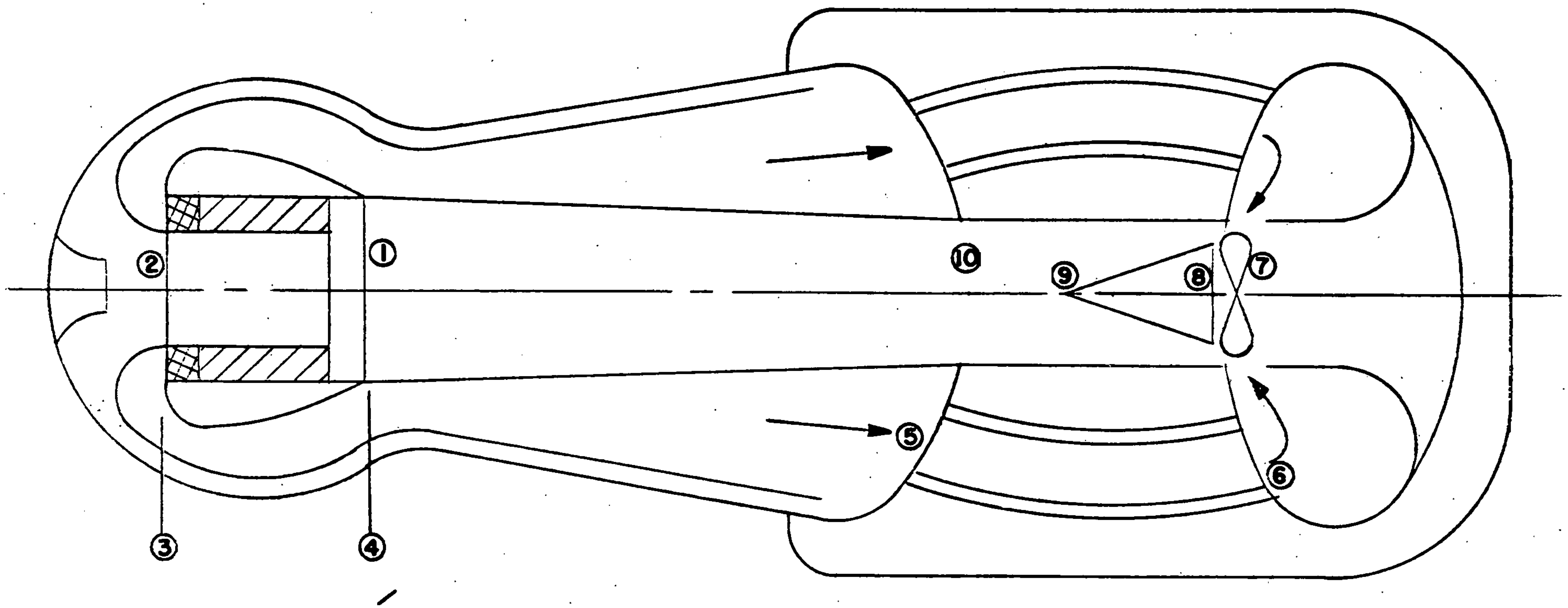


FIGURE 12

SCHEMATIC OF HELIUM LOOP

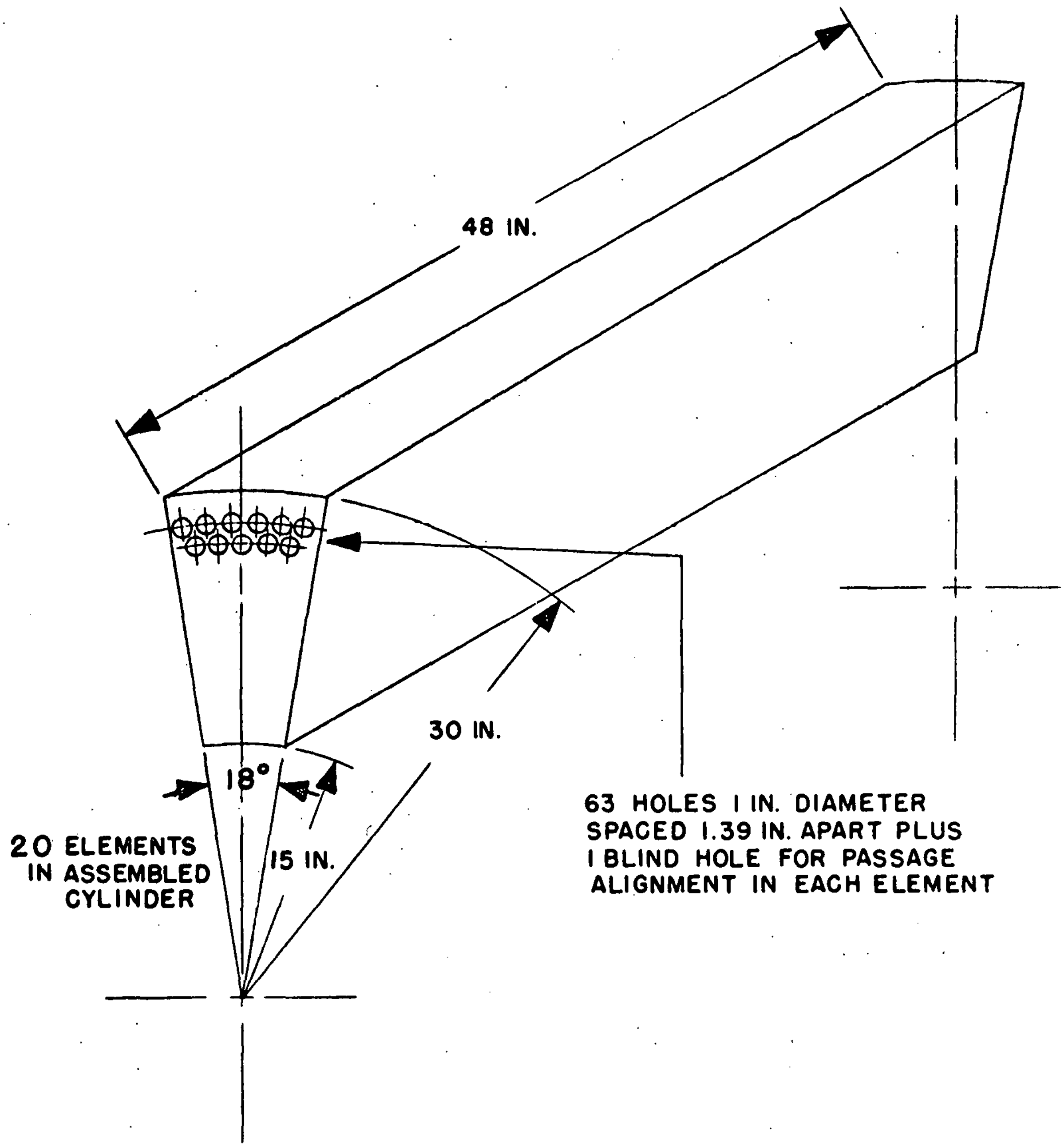


FIGURE 13
REACTOR FUEL ELEMENT

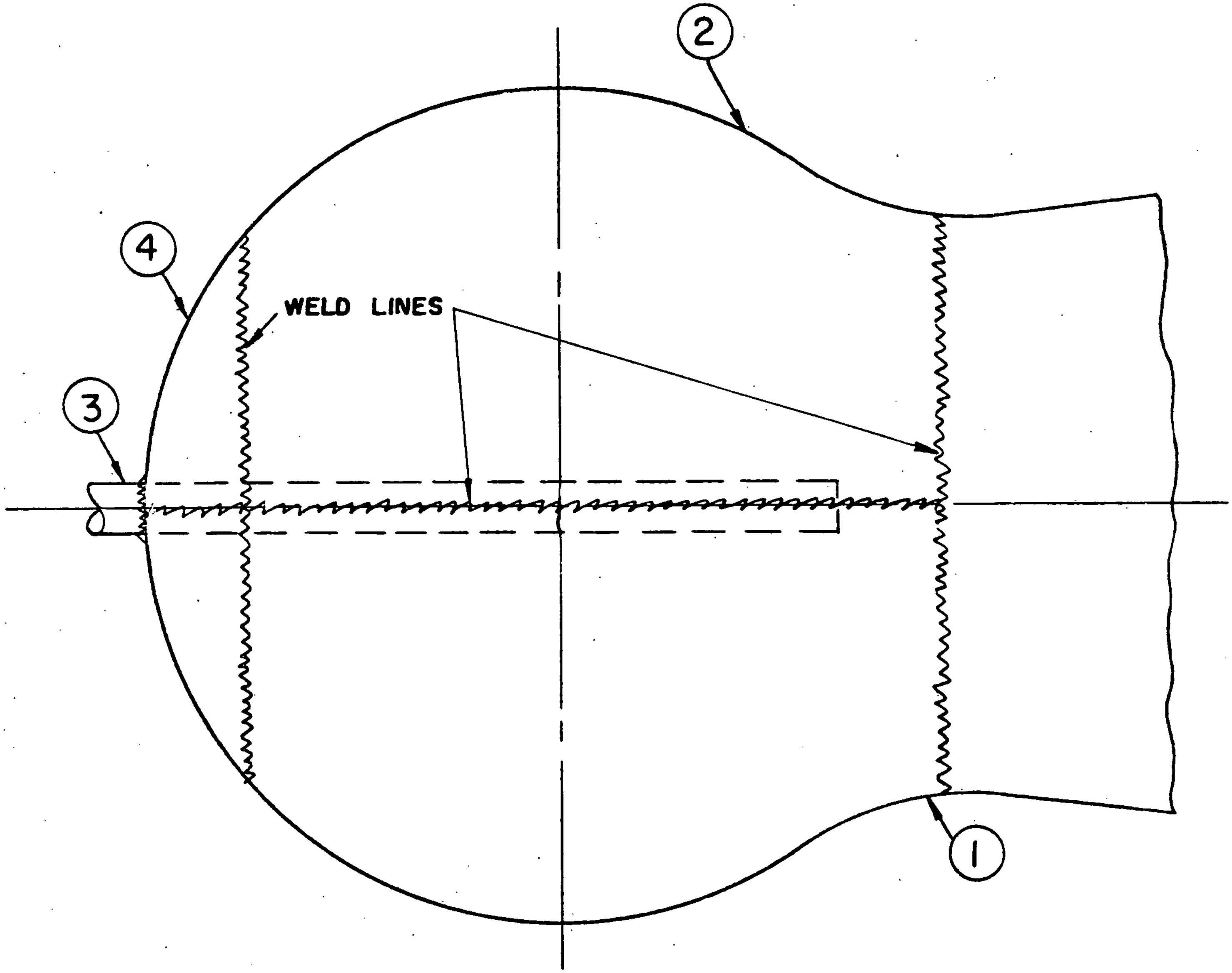


FIGURE 14

REACTOR CONTAINER

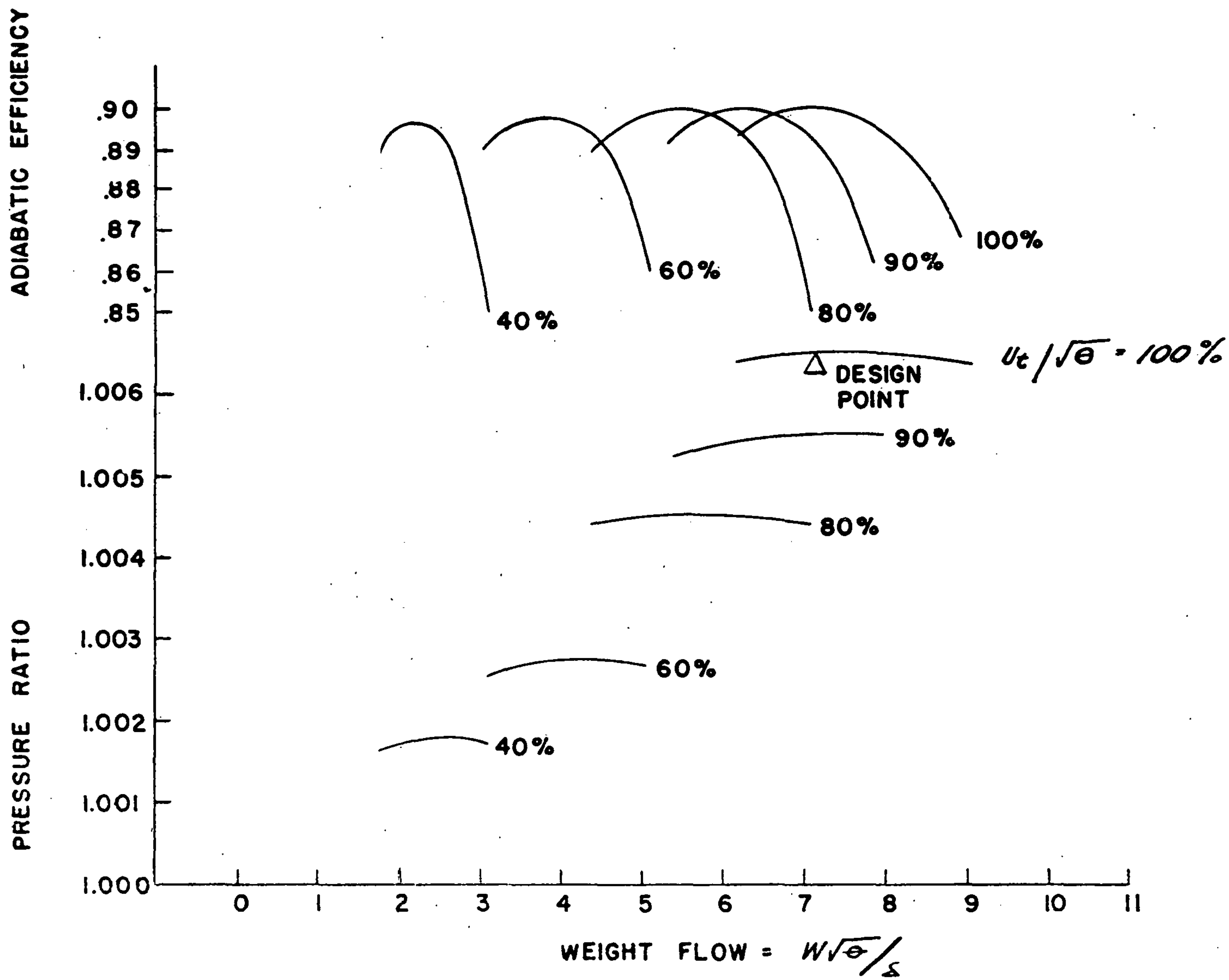


FIGURE 15

HELIUM COMPRESSOR PERFORMANCE

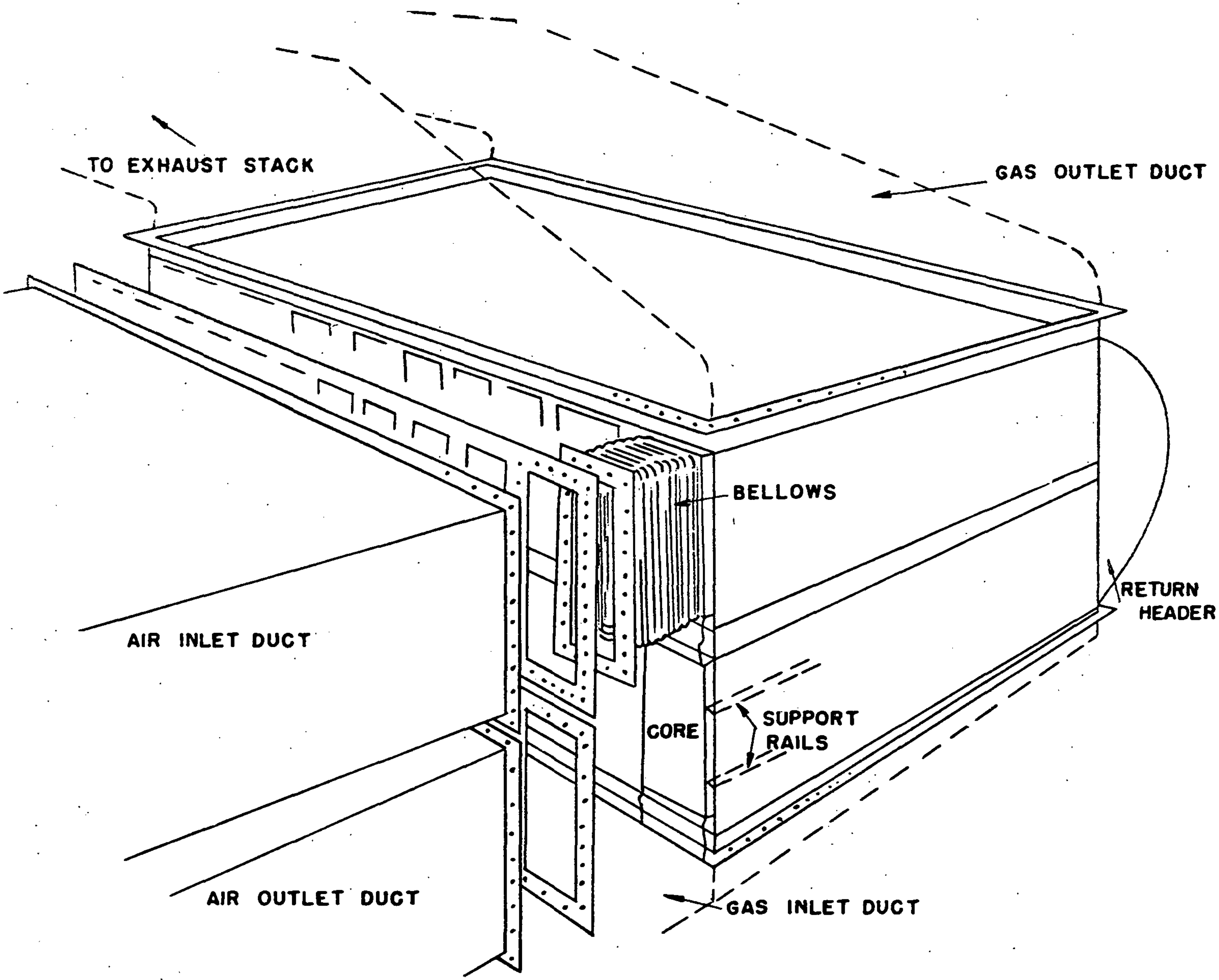


FIGURE 16
GAS TURBINE RECUPERATOR UNIT

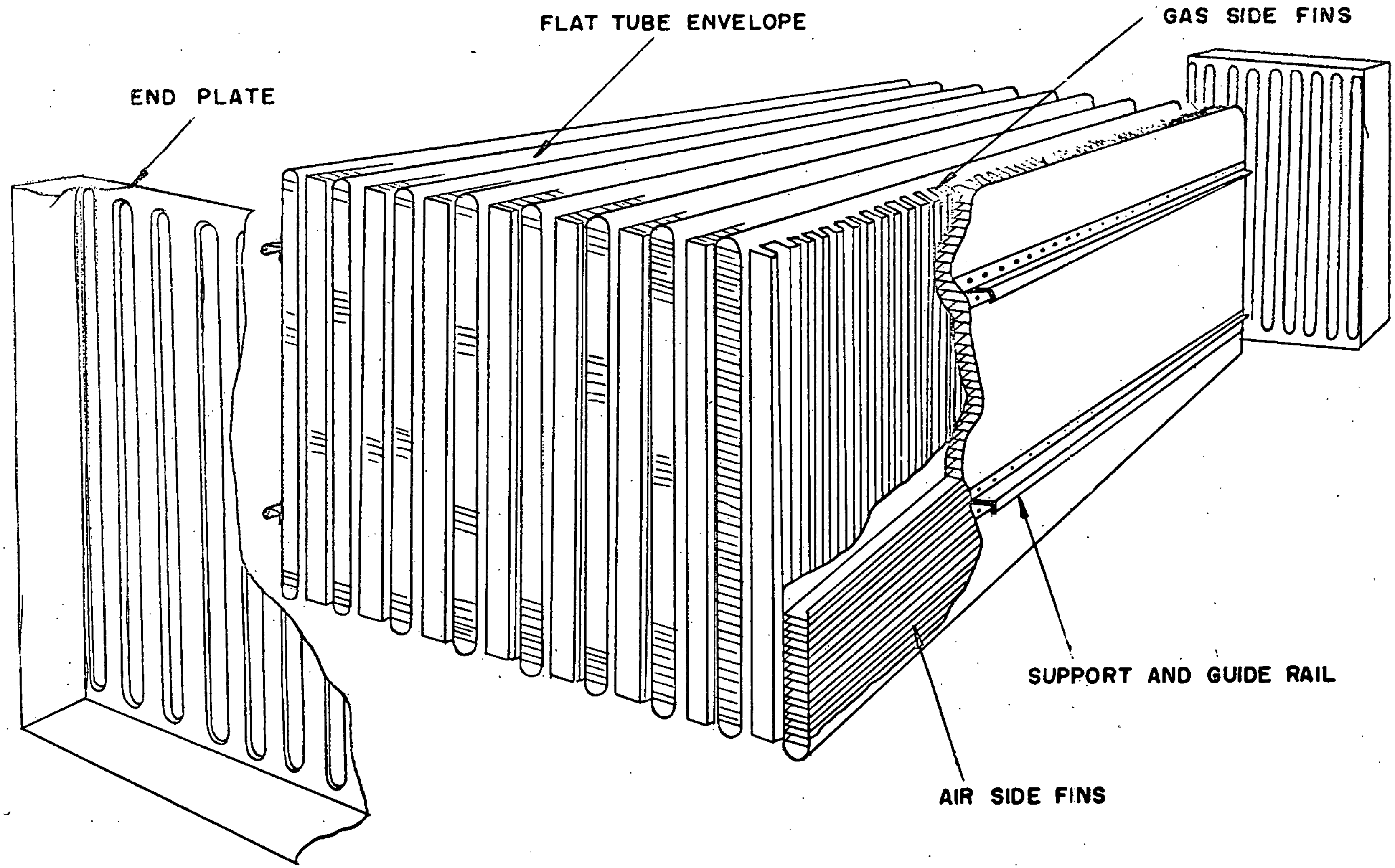


FIGURE 17

RECUPERATOR CORE MODULE ASSEMBLY

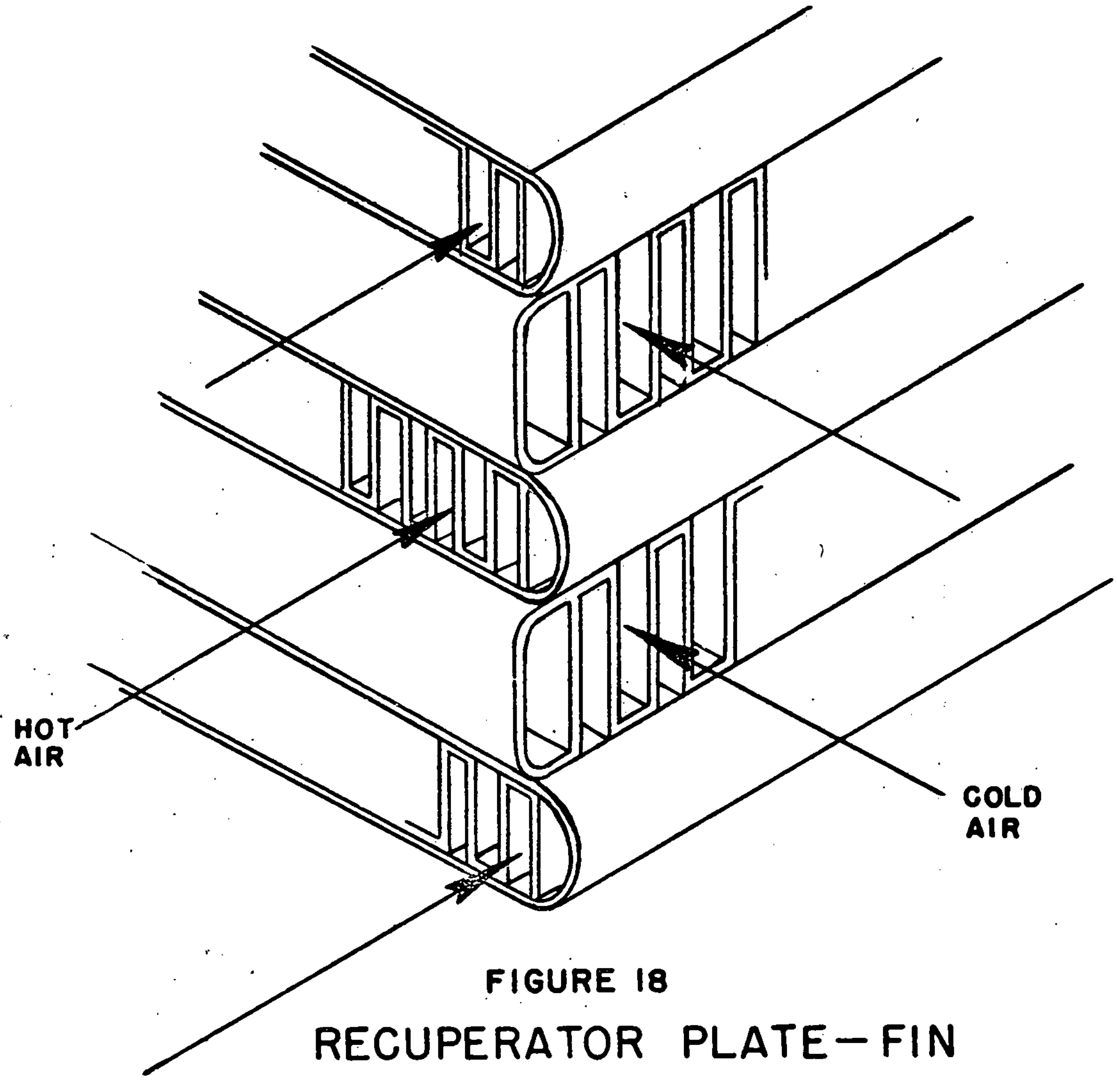


FIGURE 18

RECUPERATOR PLATE—FIN
ARRANGEMENT

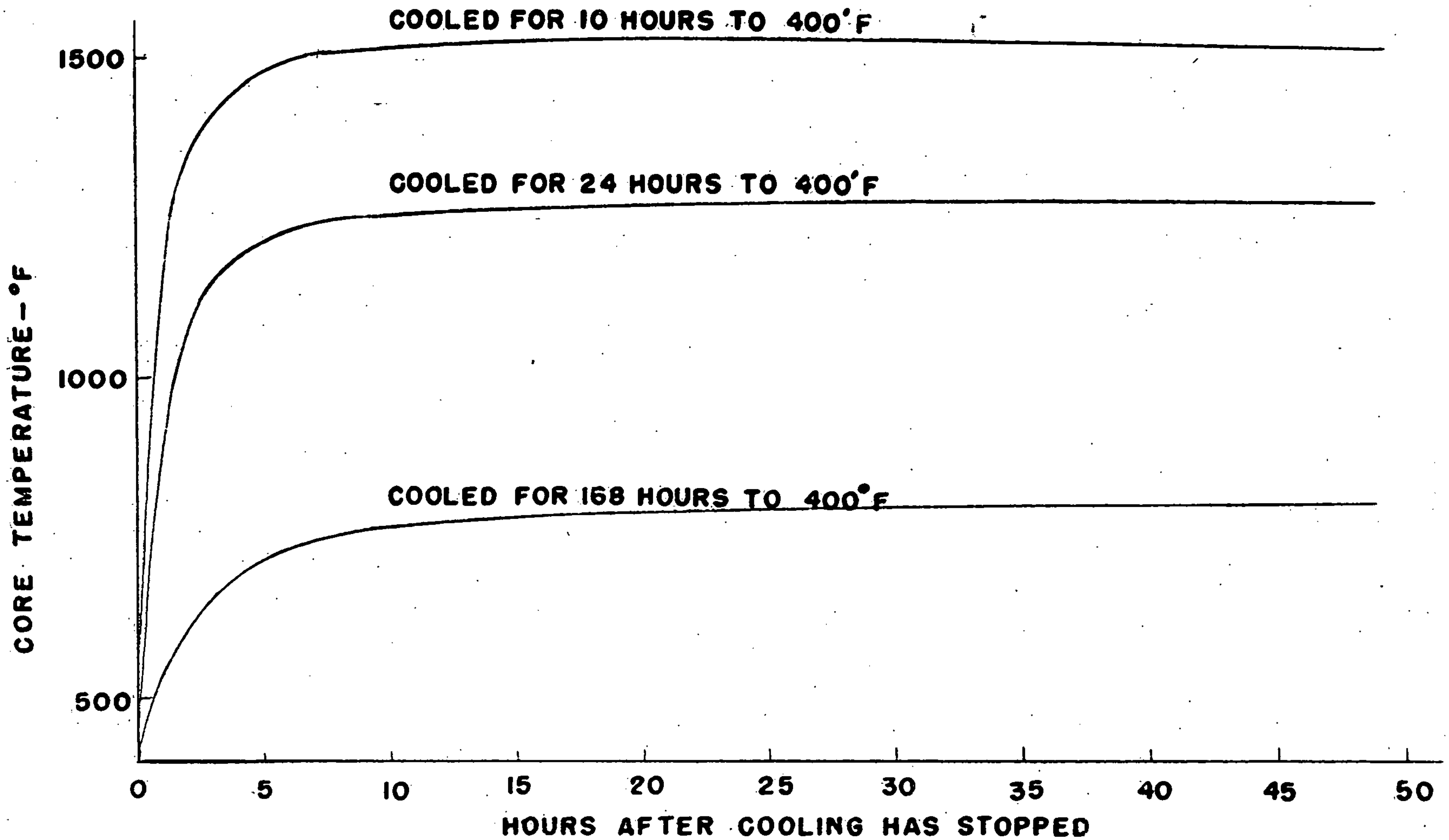


FIGURE 19

RADIATION COOLING AFTER SHUT DOWN

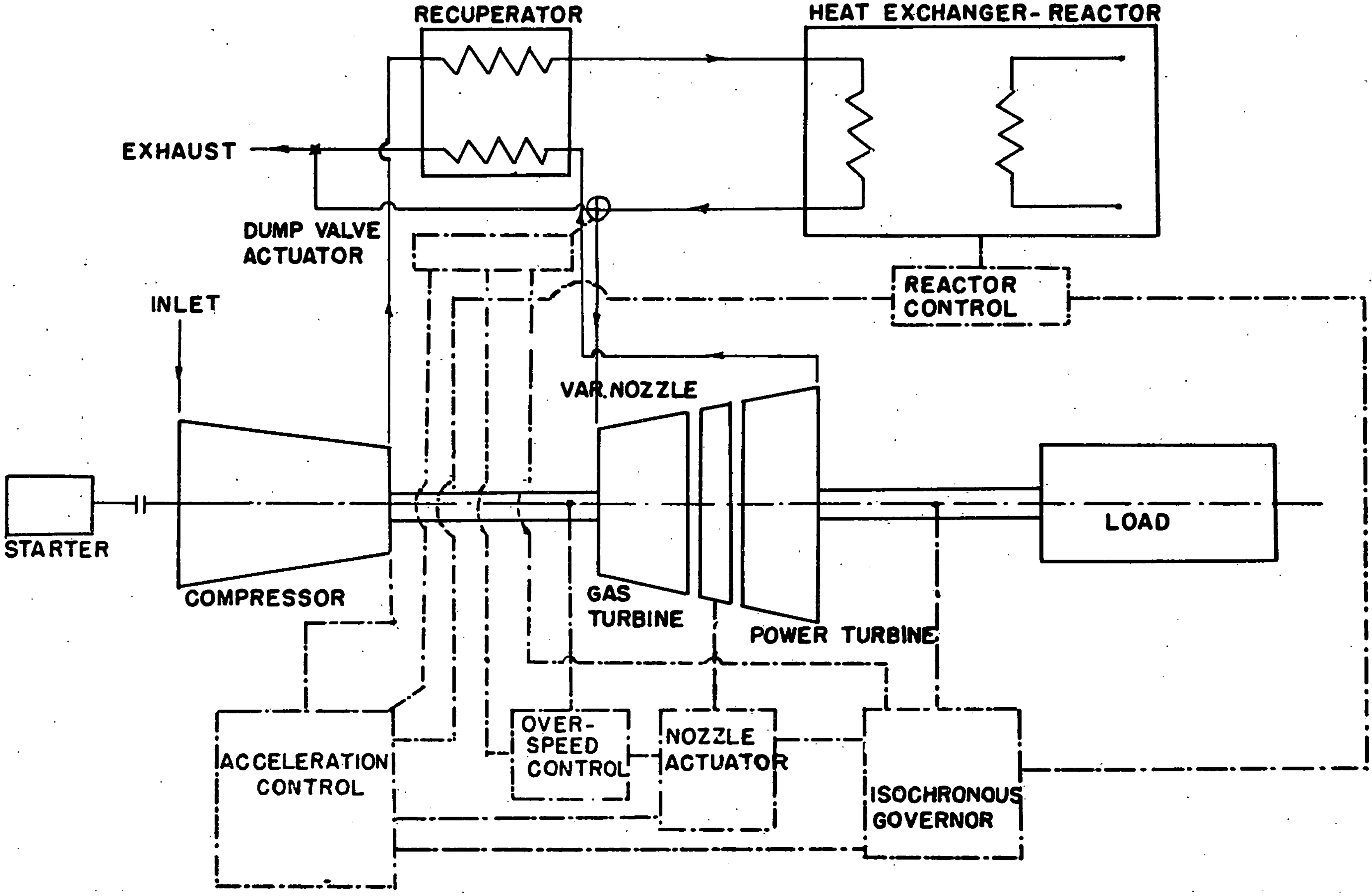
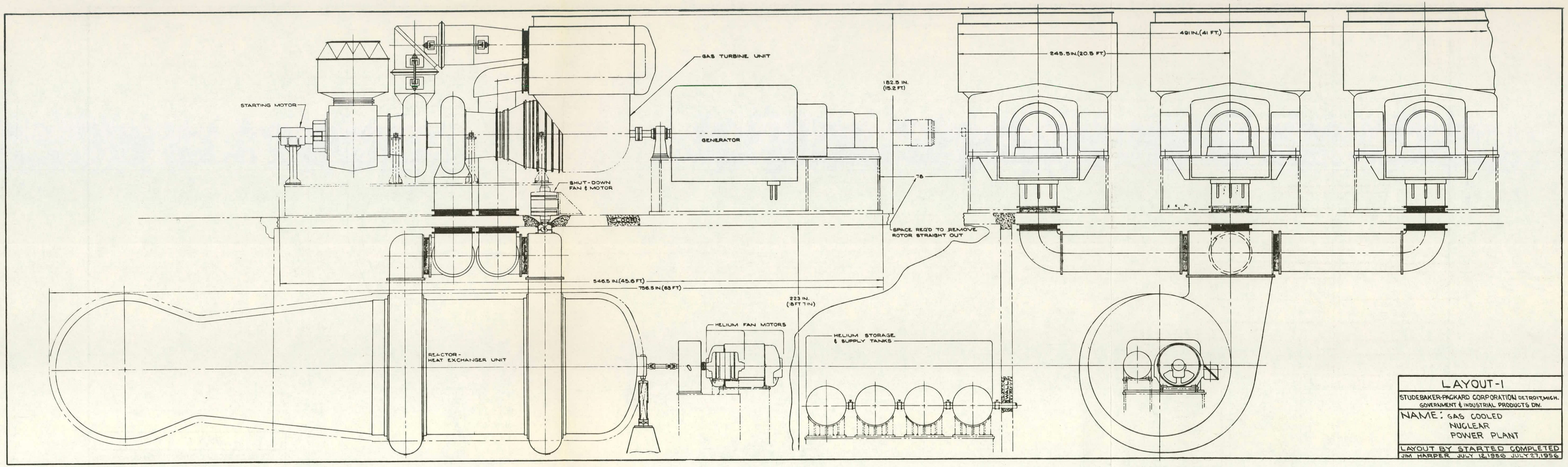
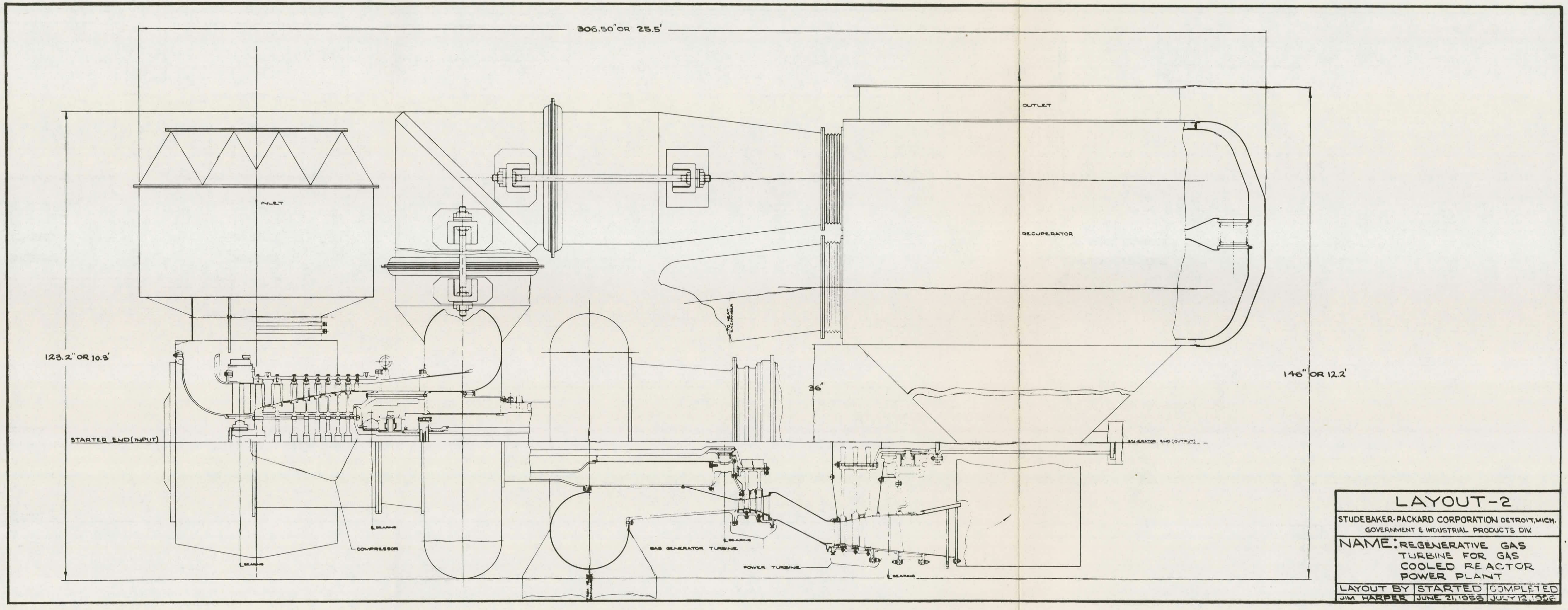
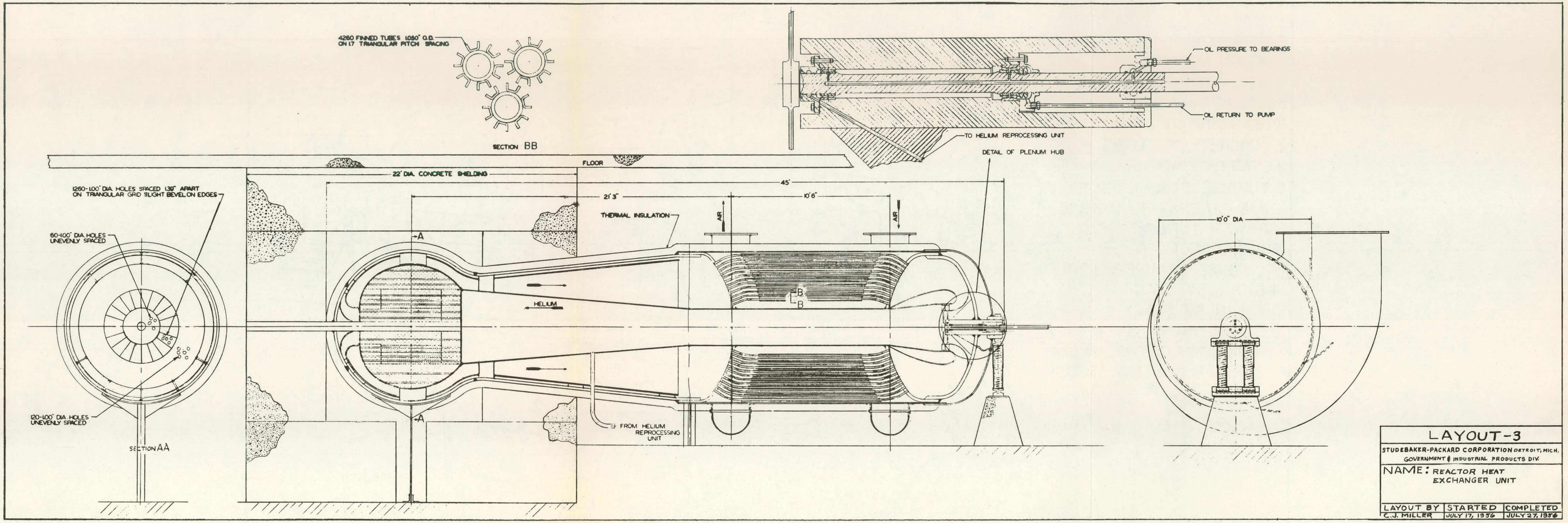


FIGURE 20

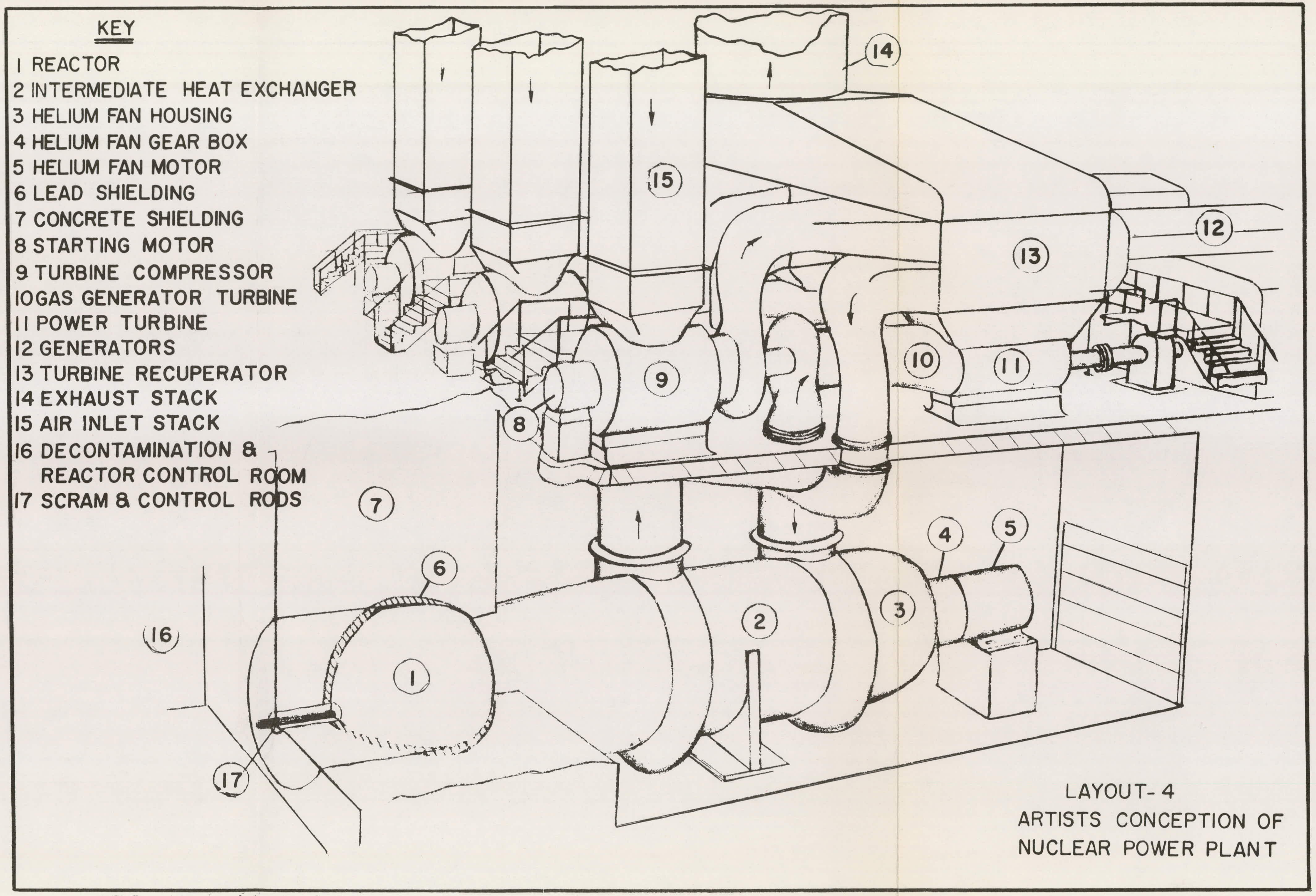
SCHEMATIC CONTROL DIAGRAM
OF NUCLEAR POWER PLANT


STARTING MOTOR
GAS TURBINE UNIT
GENERATOR
SHUT-DOWN FAN & MOTOR
182.5 IN. (15.2 FT)
491 IN. (41 FT.)
245.5 IN. (20.5 FT)
78
SPACE REQ'D TO REMOVE ROTOR STRAIGHT OUT
546.5 IN. (45.6 FT)
756.5 IN (63 FT)
223 IN. (18 FT 7 IN)
HELIUM FAN MOTORS
HELIUM STORAGE & SUPPLY TANKS
REACTOR-HEAT EXCHANGER UNIT
LAYOUT-1
STUDEBAKER-PACKARD CORPORATION DETROIT, MICH.
GOVERNMENT & INDUSTRIAL PRODUCTS DIV.
NAME: GAS COOLED NUCLEAR POWER PLANT
LAYOUT BY STARTED COMPLETED
JIM HARPER JULY 12, 1956 JULY 27, 1956


306.50" OR 25.5'
INLET
OUTLET
RECUPERATOR
123.2" OR 10.3'
146" OR 12.2'
36"
STARTER END (INPUT)
GENERATOR END (OUTPUT)
COMPRESSOR
GAS GENERATOR TURBINE
POWER TURBINE
℄ BEARING
LAYOUT-2
STUDEBAKER-PACKARD CORPORATION DETROIT, MICH.
GOVERNMENT & INDUSTRIAL PRODUCTS DIV.
NAME: REGENERATIVE GAS TURBINE FOR GAS COOLED REACTOR POWER PLANT
LAYOUT BY | STARTED | COMPLETED
JIM HARPER | JUNE 21, 1956 | JULY 12, 1956



4260 FINNED TUBES 1.050" O.D. ON 1.7 TRIANGULAR PITCH SPACING
SECTION BB
OIL PRESSURE TO BEARINGS
OIL RETURN TO PUMP
TO HELIUM REPROCESSING UNIT
DETAIL OF PLENUM HUB
FLOOR
22' DIA. CONCRETE SHIELDING
45'
21' 3"
10' 6"
1260-1.00" DIA HOLES SPACED 1.39" APART ON TRIANGULAR GRID SLIGHT BEVEL ON EDGES
60-1.00" DIA HOLES UNEVENLY SPACED
THERMAL INSULATION
AIR
AIR
HELIUM
10' 0" DIA
120-1.00" DIA HOLES UNEVENLY SPACED
SECTION AA
FROM HELIUM REPROCESSING UNIT
LAYOUT-3
STUDEBAKER-PACKARD CORPORATION DETROIT, MICH.
GOVERNMENT & INDUSTRIAL PRODUCTS DIV.
NAME: REACTOR HEAT EXCHANGER UNIT
LAYOUT BY
C.J. MILLER
STARTED
JULY 17, 1956
COMPLETED
JULY 27, 1956



LAYOUT- 4
ARTISTS CONCEPTION OF
NUCLEAR POWER PLANT

APPENDIX 1

Lifetime of Thermal Neutrons

In a homogeneous thermal reactor the neutron lifetime is determined primarily by neutron capture properites of the reactor materials and is given by the ratio

$$\Delta T_1 = \frac{1}{\Sigma_{ath} \; v_{th}} \qquad (1)$$

where Σ_{ath} is macroscopic capture cross-section for thermal neutrons and v_{th} is the velocity of a neutron in thermal equilibrium with its surroundings. In a reflector-moderated reactor the lifetime of thermal neutrons is determined primarily by the time they spend in the reflector-moderator before entering the fuel region, in which they are quickly absorbed because of the high local density of fuel. It is the purpose of this Appendix to estimate the mean lifetime of thermal neutrons in the reflector-moderated reactor and to compare this with the mean lifetime for the homogeneous reactor from Equation (1).

The general equation for the slowing down of neutrons, in the steady state, is

$$\nabla \cdot D \nabla \varphi - \Sigma_a \varphi = \frac{\partial (\xi \Sigma_s \varphi)}{\partial u} - S(u) f \qquad (2)$$

where D is the diffusion coefficient, Σ_a is the macroscopic absorption coefficient, ξ is the logarithmic decrement for the moderating material, Σ_s is the macroscopic scattering cross section, u is the neutron lethargy, S(u) is the lethargy spectrum of fission neutrons, f is the source strength of neutrons from fission, and φ is the neutron flux. For the thermal region the slowing down term becomes indeterminate since both ξ and du become zero. We can then write

$$\left.\frac{\partial\,(\xi\,\Sigma_s\,\varphi)}{\partial\,u}\right]_{th} = \Sigma_{s\ th}\ \frac{\partial\,\varphi_{th}}{\partial\ N_{th}} \tag{3}$$

where

$$\partial\,N_{th} = \frac{\partial\,u}{\xi} \tag{4}$$

is the number of thermal collisions. Consider a region of reflector-moderator in which the coefficients are constant, and in which there are negligible absorptions and sources in the region, the absorptions occurring in reactor fuel outside the reflector-moderator. Then Equation (2) becomes

$$\nabla^2\,\varphi_{th} = \frac{\Sigma_{s\ th}}{D_{th}}\ \frac{\partial\,\varphi_{th}}{\partial\ N_{th}} \tag{5}$$

The time between successive thermal collisions is

$$\frac{1}{v_{th}\ \Sigma_{t\ th}} \tag{6}$$

The time interval for $\partial\,N_{th}$ thermal collisions is then

$$\partial_t = \frac{\partial\,N}{v\ \Sigma_t} \tag{7}$$

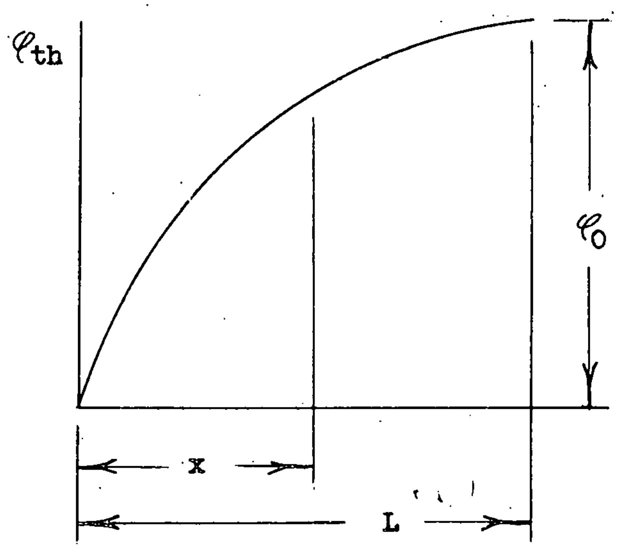
Equation (5) becomes, with $D_{th} = \dfrac{1}{3\ \Sigma_{t\ th}}$

$$\nabla^2\,\varphi_{th} = \frac{3\ \Sigma_{s\ th}}{v_{th}}\ \frac{\partial\,\varphi_{th}}{\partial\,t} \tag{8}$$

For a slab geometry, by separating variables

$$\varphi_{th} = \varphi_0 \sin\left(\frac{\pi x}{L}\right) e^{-\frac{\pi^2 v_{th} t}{3 \Sigma_{s\ th} L^2}} \tag{9}$$

where the geometry is as shown below



Leakage from the surface, at x = 0, is given by

$$D_{th} \ \varphi_0 \ \frac{\pi}{L} \ e^{-\frac{\pi^2 v_{th} t}{3 \Sigma_{s\ th} L^2}} \tag{10}$$

The mean leakage time for all neutrons is

$$\Delta T_2 = \frac{D_{th} \ \varphi_0 \ \frac{\pi}{L} \int_0^{\infty} t e^{-\frac{\pi^2 v_{th} t}{3 \Sigma_{s\ th} L^2}} dt}{D_{th} \ \varphi_0 \ \frac{\pi}{L} \int_0^{\infty} e^{-\frac{\pi^2 v_{th} t}{3 \Sigma_{s\ th} L^2}} dt} \tag{11}$$

Performing the integrations

$$\Delta T_2 = \frac{3 \; \Sigma_{s\ th} \; L^2}{\pi^2 \; v_{th}} \tag{12}$$

Since it has been specified that all the neutrons leaking from the reflector-moderator are immediately absorbed (by fuel), Equation (12) represents the mean lifetime of thermal neutrons in the reflector-moderated reactor to the extent that Equation (9) is a good representation of the neutron distribution in the moderator. This must be checked empirically for reactors under consideration. From Equations (1) and (12) the ratio of the mean lifetime of thermal neutrons in the reflector-moderated reactor to that in the homogeneous reactor is given by

$$\frac{\Delta T_2}{\Delta T_1} = \frac{3 \; \Sigma_{s\ th} \; \Sigma_{a\ th} \; L^2}{\pi^2} \tag{13}$$

Relatively long lifetimes of thermal neutrons in the reflector-moderated reactor hence result from a high value of the scattering cross-section ($\Sigma_{s\ th}$), a high critical mass ($\Sigma_{a\ th}$), and a relatively wide distribution of thermal neutrons in the reflector moderator (L^2).

APPENDIX 2

Experimental Program

The experimental work on the erosion of graphite by helium and the diffusion of helium through stainless steel at high temperatures was carried out by the Aerophysics Development Corporation at Santa Monica, California. The report in its entirety was reproduced as received. Details of procedure and equipment as well as results are presented here.

TO: Bill Parrish DATE: 31 July 1956

FROM: Structures Test

SUBJECT: Final Report, Studebaker-Packard Contract
AT(303)-214
Inter-Co. Work Order No. 1

PURPOSE OF TEST:

Experimental investigations to determine the feasibility of proposed design schemes for gas cooled reactors. In particular,

1) The rate of diffusion of helium through stainless steel at high temperature.

2) The rate of erosion of graphite at high temperature by a high velocity helium stream.

1) DIFFUSION TEST

PROCEDURE:

The design of the identical hot and cold diffusion test samples was determined by furnace dimensions and the availability of steel tubing. The number of welds on the samples was kept at a minimum, and the welds of the hot samples were all located outside the furnace. Figures 1 and 2 show the 1/2" diameter test tube which was located inside a 1 1/2" diameter tube. A helium atmosphere was established between these two tubes, the pressure of 80 psi being supplied by a connection to a helium bottle. The hot samples were, furthermore, placed inside a LECO 2600 furnace which maintained a temperature of 1200°F by means of a controller connected to a thermocouple inside the furnace, Figure 3. To detect any leaks, the samples were built the following way:

A) The two end plates were welded to the 1/2" diameter tube. Any weld burn-through was detected by visual inspection.

B) The ends of the test tube were pinched off and welded. The tube was heated in the center with ends submerged in water to check for leaks of test tube.

C) The outside, 1-1/12" diameter, tube was slid over the small tube and welded to the end plates.

D) The helium supply lead was welded to the outside tube.

All the welding was done in an argon atmosphere.

The sample tubes were of welded and redrawn construction, and were of the following materials:

Type 304 Stainless Steel, .035" Wall
Type 347 Stainless Steel, .020" Wall

A total of four test samples, a cold and a hot one of each material, were assembled. In addition to these four test specimens, a fifth small tube was closed and sealed the same way and at the same time to determine the standard content of helium in the atmosphere at the beginning of the test.

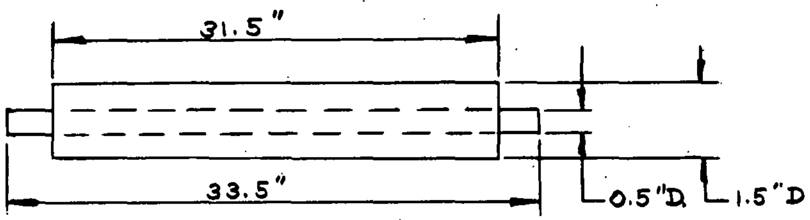
The amount of helium in the test tubes after the diffusion test was determined by mass-spectrography at the Consolidated Engineering Corporation, Pasadena.

RESULTS:

TEST TUBE	WALL in	MANIFOLD PRESSURE psig	FURNACE TEMP. °F	t TIME DAYS	d_0 HELIUM CONTENT * Vol Pts. In Million
(0) 347 O	.020	-	-	-	0
(1) 304 C	.035	80	80°	20	9
(2) 304 H	.035	80	1200°	20	15
(3) 347 C	.020	80	80°	20	12
(4) 347 H	.020	80	1200°	20	23

* Analysis by Consolidated Engineering Corporation, Pasadena.

ANALYSIS:



Exposed surface: $S = \frac{\pi}{2} \cdot 31.5 = 49.5 \text{ in}^2$

Volume of Tube: $V = \frac{\pi}{4} (\frac{1}{2})^2 \; 33.5 = 6.60 \text{ in}^3$

Instead of calculating an actual diffusion constant a simpler value f = FLOW RATE OF HELIUM per in^2 exposed surface per day will be used.

The volume of helium which diffused though the walls of our test tube (at room temperature, atmospheric pressure) is

$$V \times do = 6.60 \times do$$

$$f = \frac{V \times do}{S \times t} = \frac{6.60}{49.5} \frac{do}{20} = 6.66 \times 10^{-3} \times do \quad \frac{\text{in}^3 \text{ helium}}{\text{in}^2 \text{ day}}$$

These values reduced to a wall thickness .020" are listed as f_1.

TEST TUBE		WALL in	$f \left(\frac{\text{in}^3 \text{ helium}^*}{\text{day in}^2} \right)$	f_1
(1)	304C	.035	60×10^{-9}	105×10^{-9}
(2)	304H	.035	100×10^{-9}	175×10^{-9}
(3)	347C	.020	80×10^{-9}	80×10^{-9}
(4)	347H	.020	154×10^{-9}	154×10^{-9}

*Helium at room temp., atmospheric pressure

In both cold and hot test Type 347 Stainless Steel has a lower flow rate than Type 304.

It is proposed to take photomicrographs of sections through the cold and hot tubes to determine any changes in grain structure.

2) EROSION TEST

PROCEDURE:

In these tests helium was blasted at graphite at high velocity in order to find any erosive action helium might have. This was done at room temperature and at a high temperature - close to what is found in reactor operations.

All the graphite used in this test was supplied by Great Lakes Carbon Company.

The type of samples used for cold and hot tests is shown in Figures 4, 5, and 8.

Micrographs were taken before and after the test. Figure 12 to Figure 19.

TYPE OF TEST	HPC-7	HP3LM	HLM
COLD	1 Sample	1 Sample	1 Sample
HOT	1 Sample	None	1 Sample
PROPERTIES OF MATERIAL			
Apparent Density g/cc	1.81	1.70	1.90
Electr. Resistivity Ohm/in	.00032	.00034	.00026

COLD TEST

The polished side of small graphite cubes was set 1/10" away from a welding tip whose orifice had been reduced to .042" diameter by silver brazing and drilling with a No. 58 drill. This tip was connected to a helium bottle, the flow being regulated by a pressure reducer and a flow meter. The graphite was placed in a small plexiglass box for protection. This box was adequately vented to prevent build-up of back pressure. Figures, 4, 6, and 7.

HOT TEST

The current of a 400 Amp DC-Welder was used for the resistance heating of the graphite pieces up to the desired temperature. For this purpose the center section of the test-piece was reduced to a small cross-sectional area (1/4 x 1/4 in) to obtain the proper electric resistance. The test piece itself was placed in a box made up of stainless steel and asbestos and was supported at both ends. Two copper end pieces screwed into the graphite provided a connection to the welder leads. A calibrated Chromel-Alumel Thermocouple inserted into the graphite close to the blast point and connected to a Millivoltmeter was used for adjusting the welder current and consequently maintaining a constant temperature. (Figures 9, 10, 11)

The helium was controlled by a pressure reducer and a flow meter and the helium bottles were connected to allow continuous operations. Before blasting at the graphite through a .042" diameter stainless steel nozzle, the helium passed a heating coil inside a furnace where it was heated up to about 700°F. Special precaution had been taken to provide an oxygen free atmosphere inside the blast box. This was successfully achieved by building the box as tight as possible and using the helium itself for automatic sealing. The starting sequence was:

1. Turn on helium
2. Start furnace
3. Start welder

Shut down was done in the opposite way.

RESULTS:

COLD TESTS

The results can be summarized in the following way:

SAMPLE	TIME Hrs./min.	p_1* psig	VOLUME** ft^3/hour	REMARKS
HPC-7	8/10	40	100	No erosion Fig. 13
HP3LM	8/10	40	100	No erosion Fig. 16
HLM	11/10	40	100	No erosion Fig. 18

* = Pressure set at regulator
** - Flow rate measured at Flowmeter

HOT TESTS

SAMPLE	TIME Hours	p_1 psig	VOLUME ft^3/Hour	TEMP. °F	WELDER CURRENT Amp	REMARKS
HPC-7	1	40	100	60°	-	No erosion Fig. 14
	4	40	45	2500°	280	
HLM	1	40	100	60°	-	No erosion Fig. 19
	4	40	45	2500°	280	

ANALYSIS

In order to establish at least sonic flow at the nozzle. the flow meter was opened completely in both cold and hot tests. Furthermore, the pressure was increased above the critical pressure, i.e., above the point where a further increase in pressure does increase the flow rate.

The critical pressure was 30 psia, whereas, the pressure was actually set at 40 psig. The acoustic velocity of helium at 510°R is 2700 ft/sec.

Flow of helium was measured by a flow meter. This measurement and the known flow area of the nozzle were used to confirm the acoustic velocity at the nozzle exit. The relationship between velocity and flow volume for helium and the 0.042 in^2 orifice is

$$U \text{ (speed ft/sec)} = 29 \times \text{volume } (ft^3/hr)$$

On the cold samples we had

Volume = 100 ft^3/hour

U = 2900 ft/sec

or, within the accuracy of our calculations, the speed of sound.

Fred Heller

Fred Heller

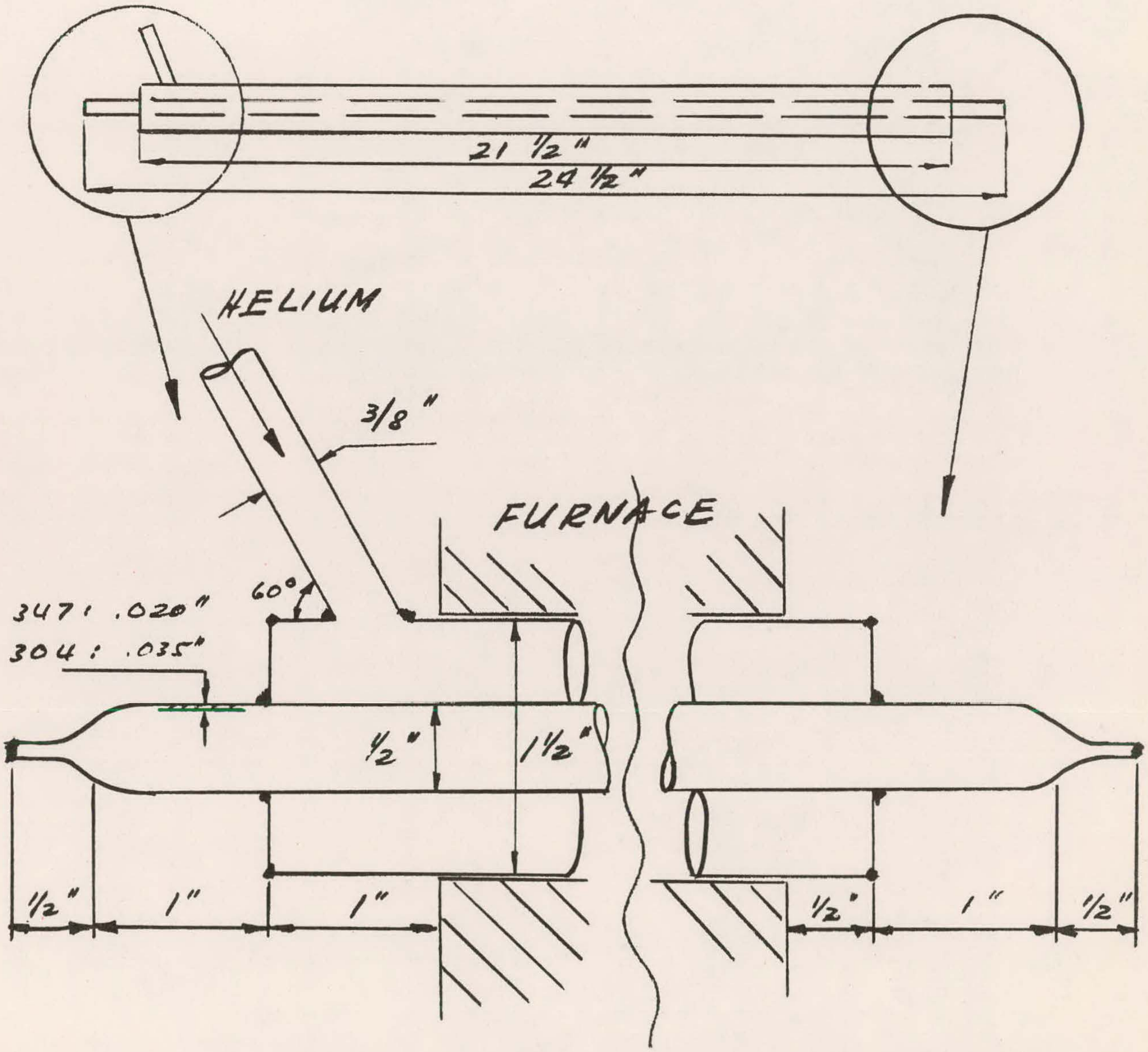


Figure 1 Diffusion Tube

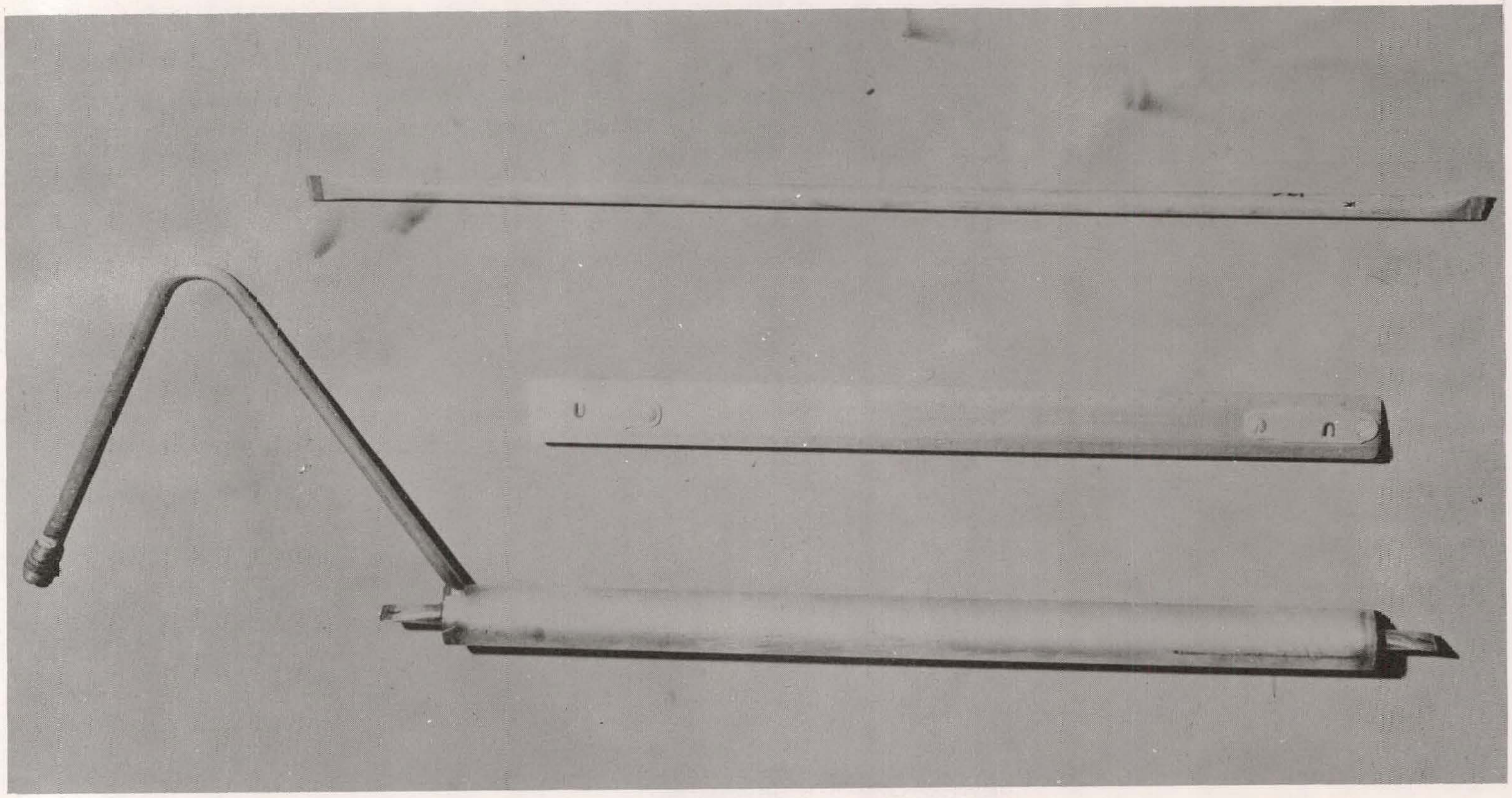


Figure 2 Helium Diffusion Test
Top: Test Tube
Bottom: Test Tube Installed In Outer Tube
With Helium Supply Tube

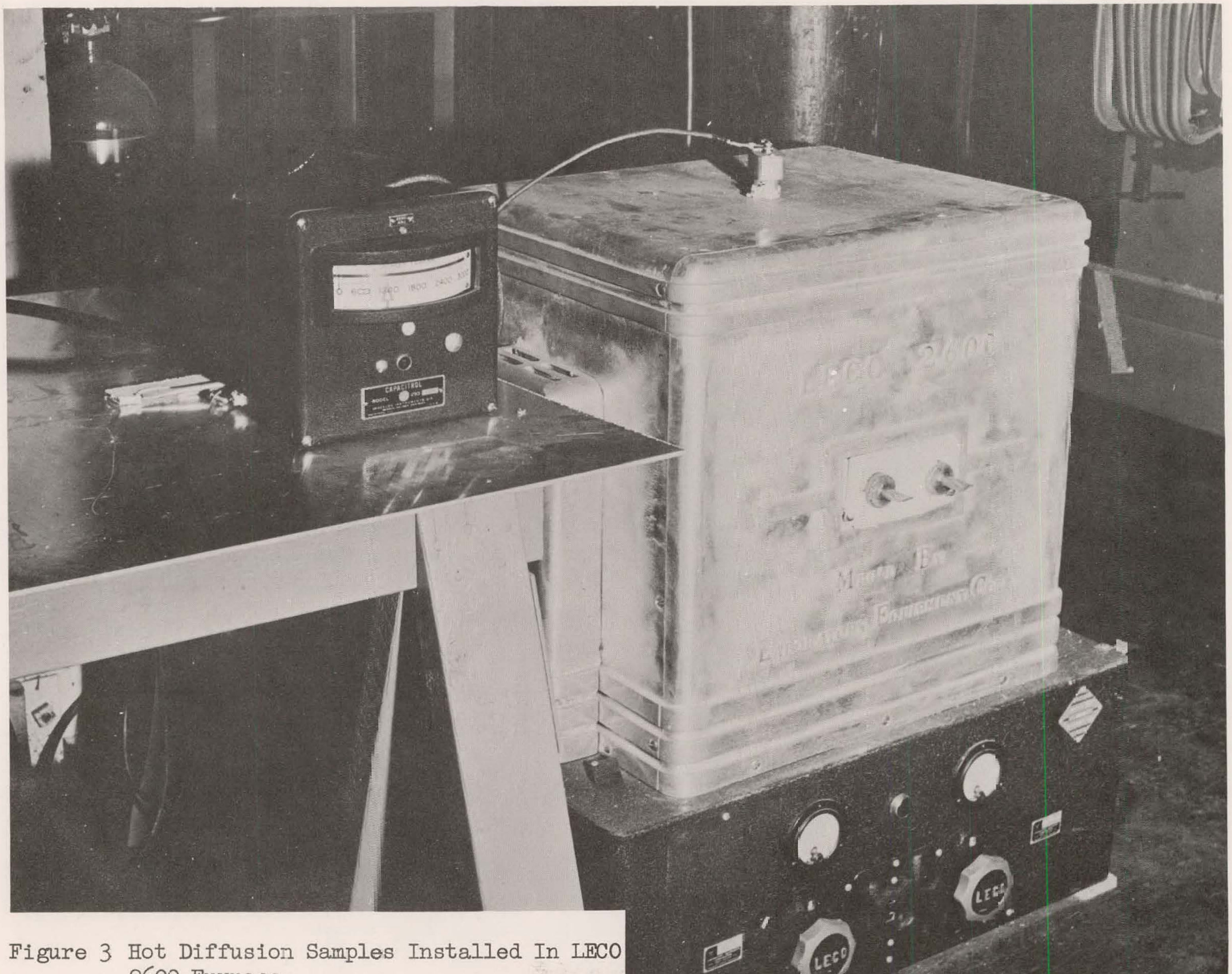
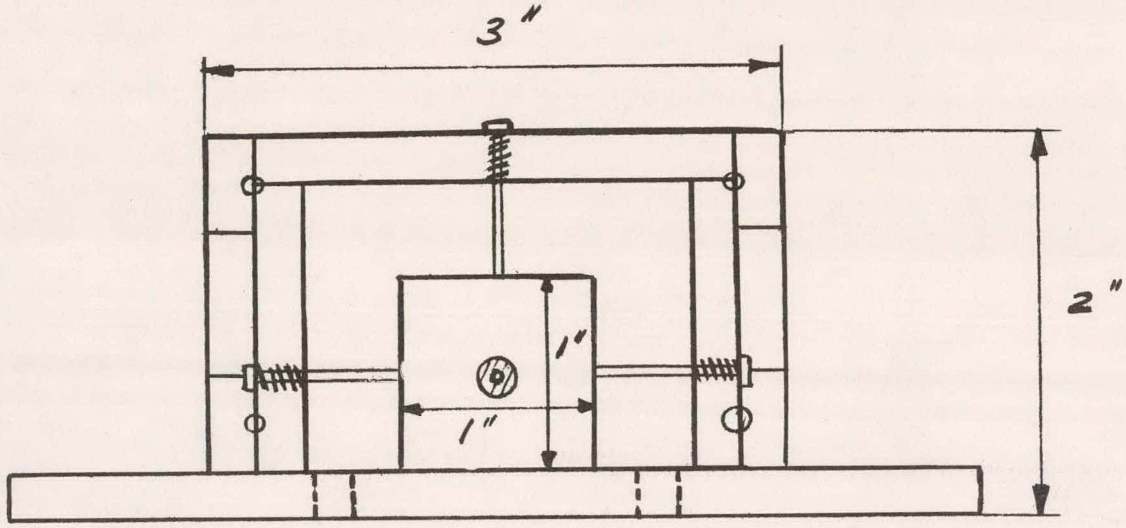


Figure 3 Hot Diffusion Samples Installed In LECO 2600 Furnace.
Left: Controller Connected To Thermocouple



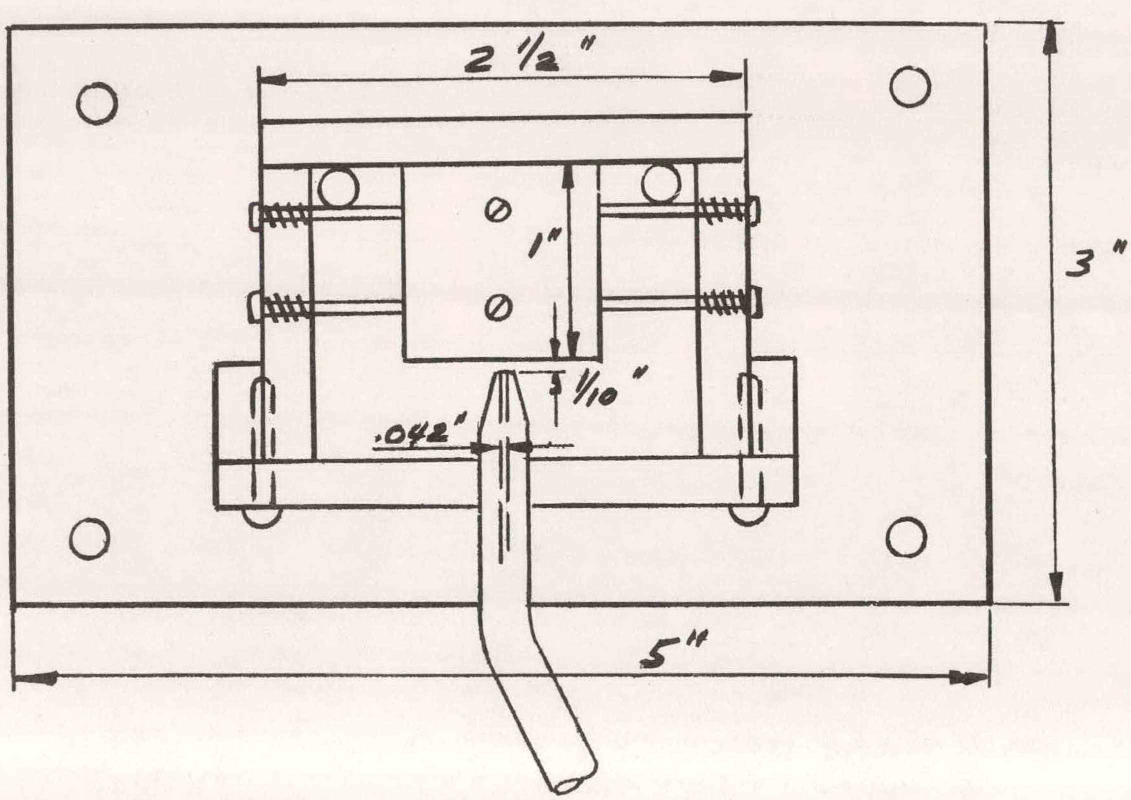


Figure 4 Cold Blast Box

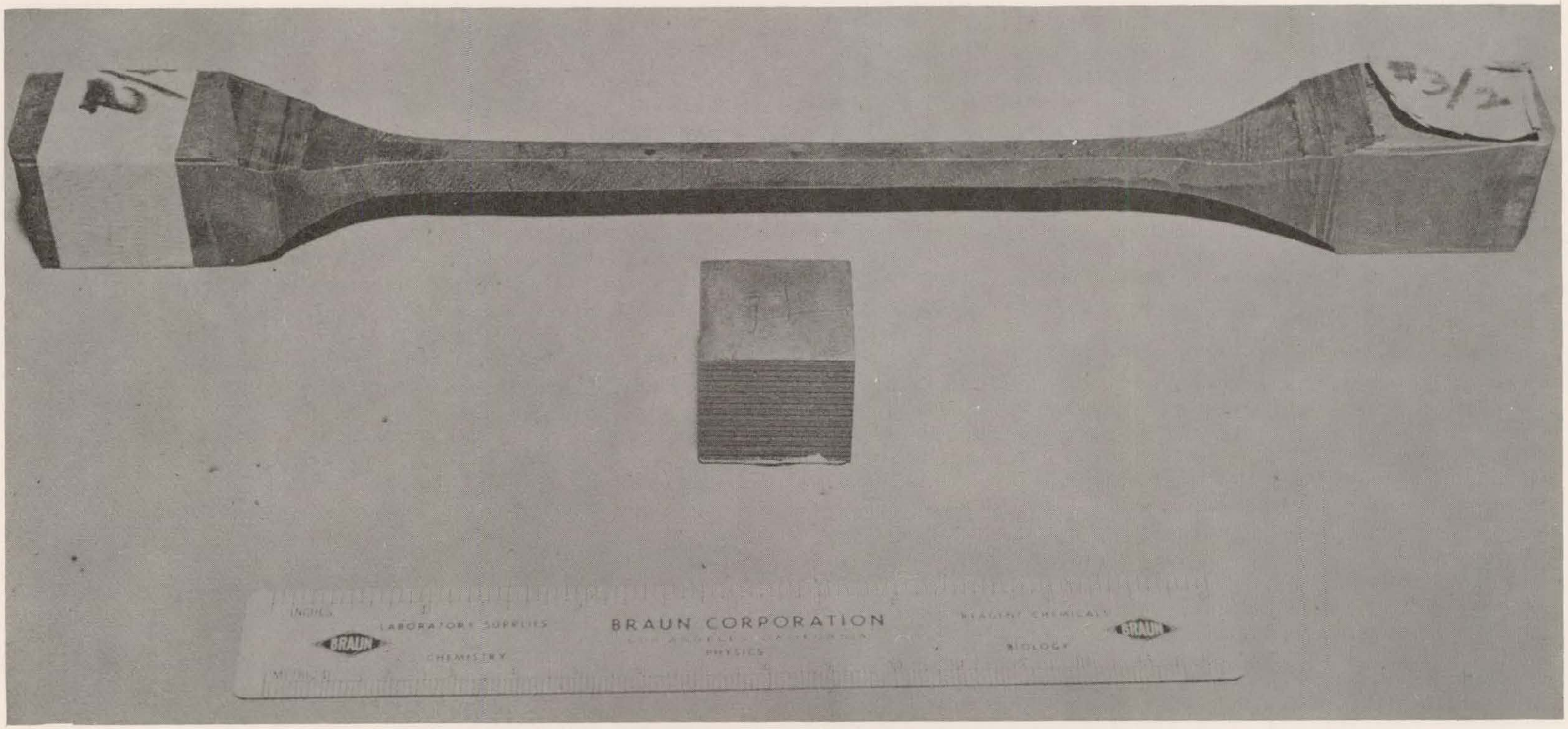


Figure 5 Graphite Erosion Samples
Top: Hot Sample
Bottom: Cold Sample

Figure 6 Setup Of Cold Erosion Test

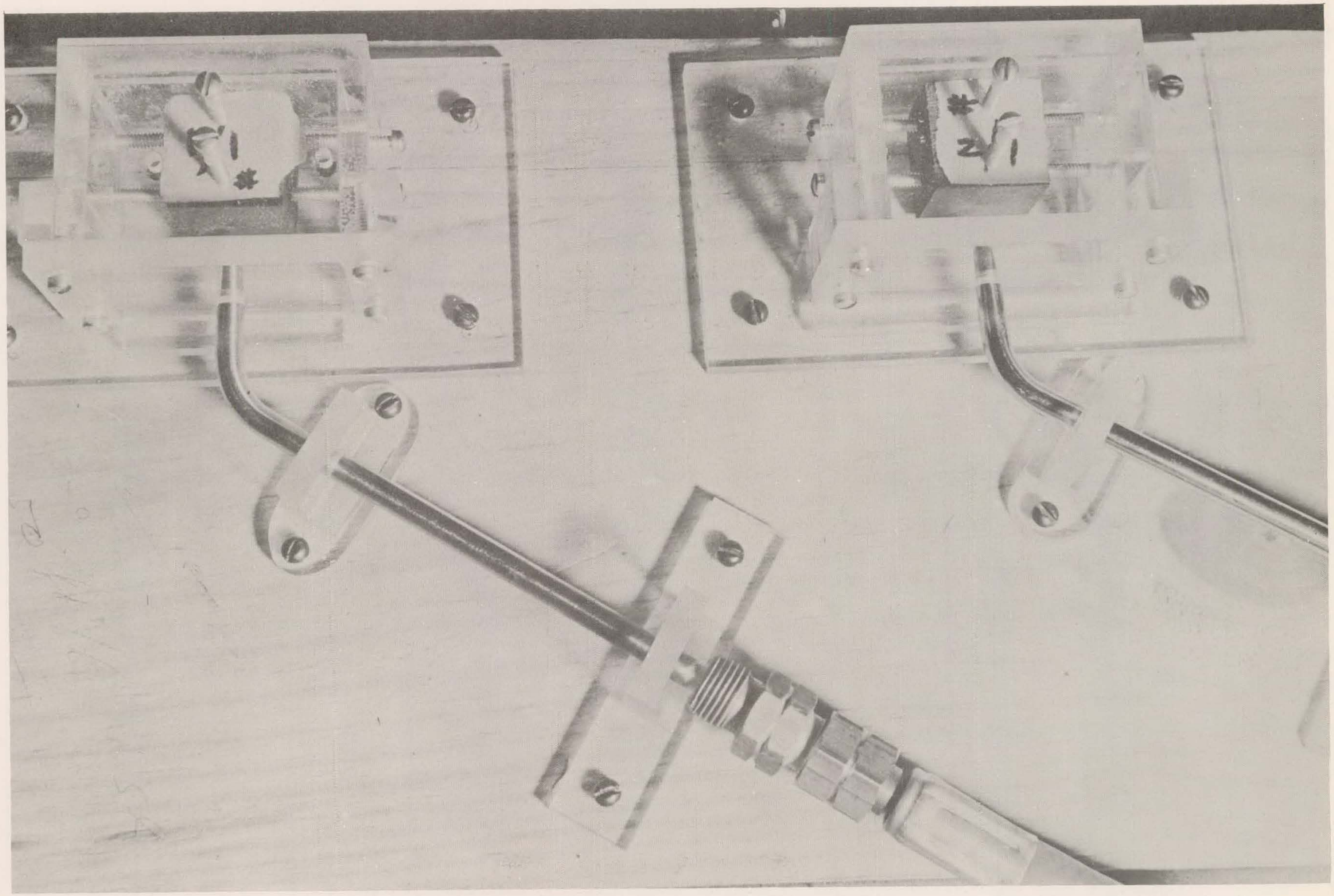


Figure 7 Closeup Of Cold Blast Boxes With Samples Installed

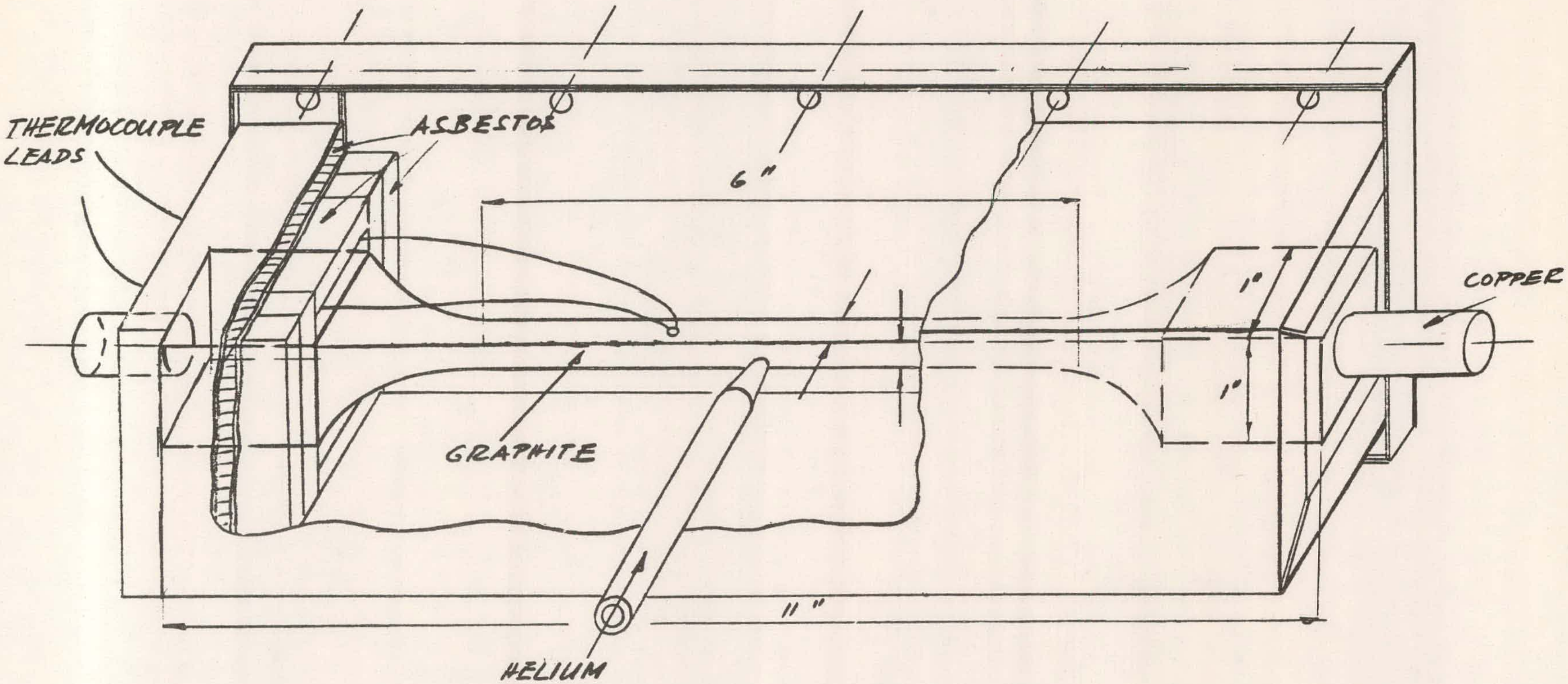


Figure 8 Hot Blast Box

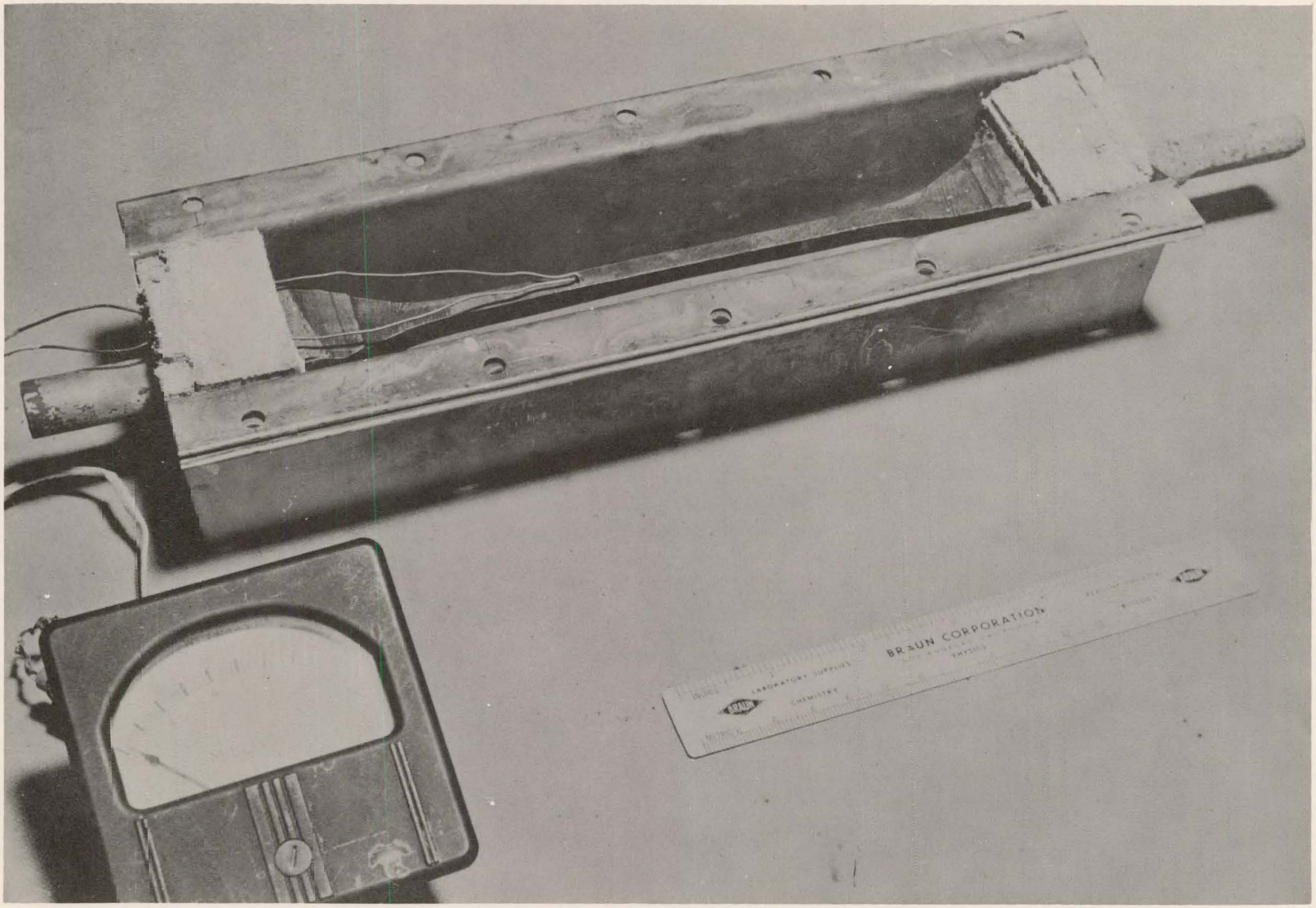


Figure 9 Hot Graphite Erosion Sample Installed Ir
Blast Box, With Thermocouple And Millivcltmeter

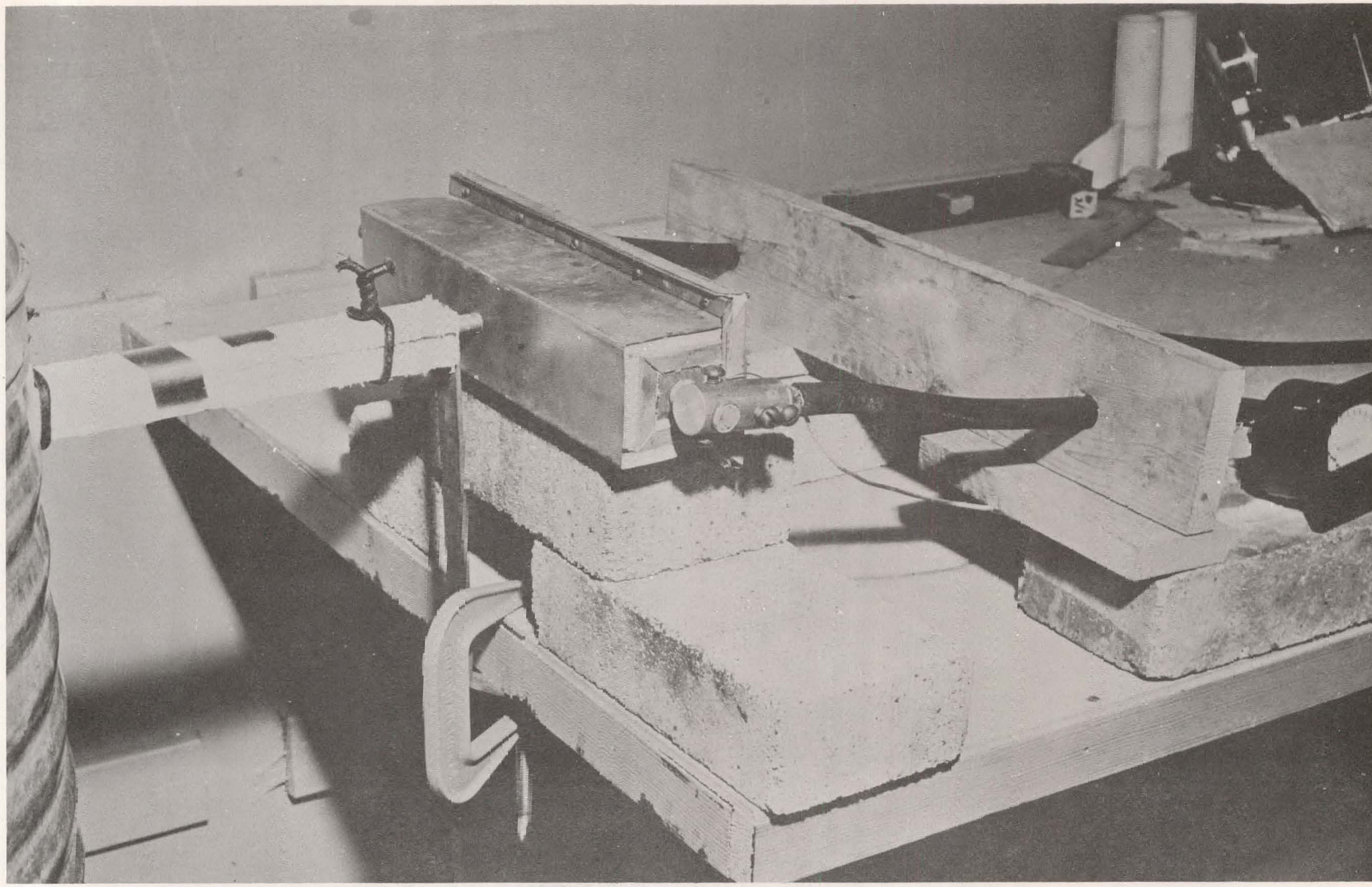


Figure 10 Hot Blast Box With Welder Leads And Helium
Nozzle

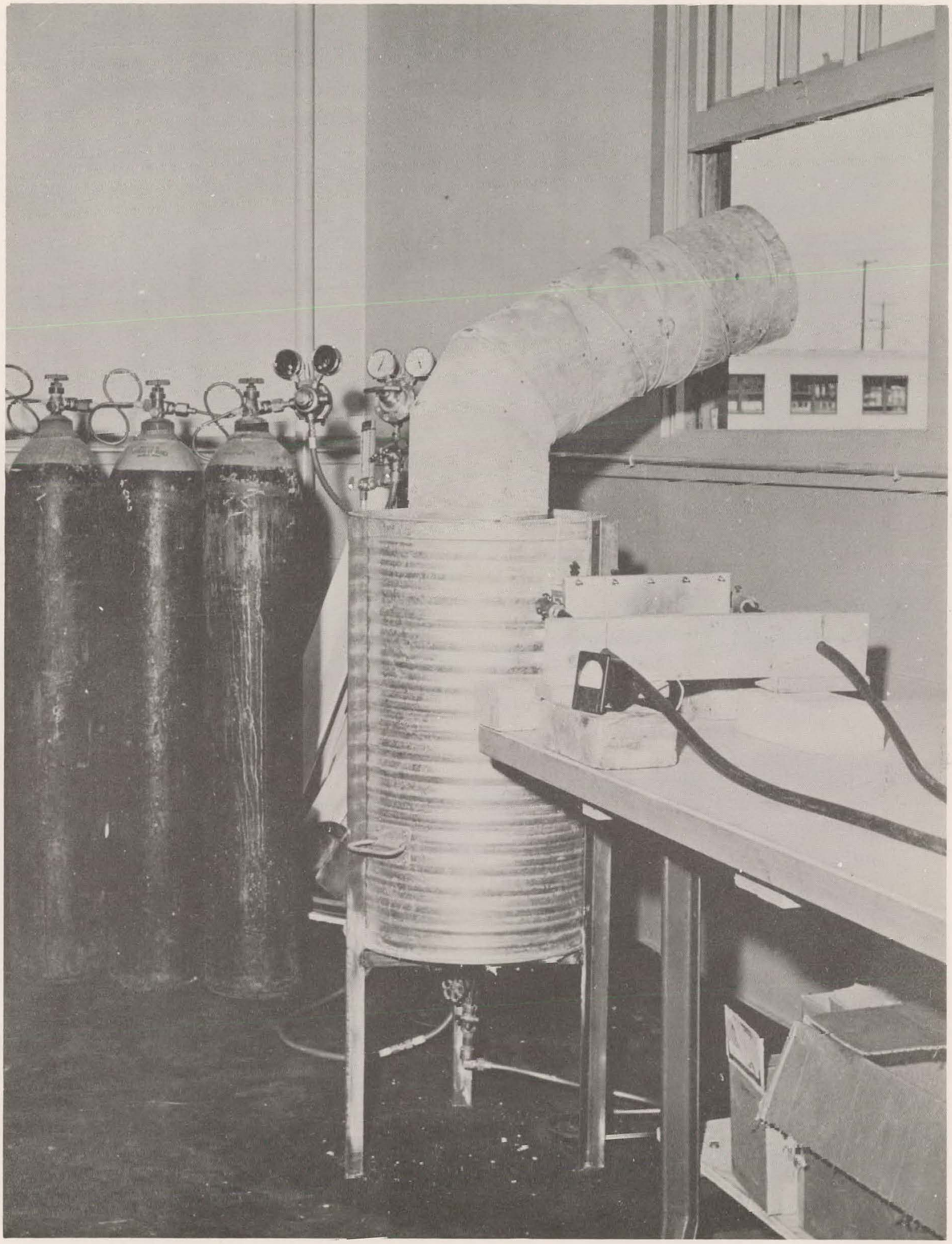


Figure 11 Furnace Used For Heating Of Helium In Hot Erosion Test

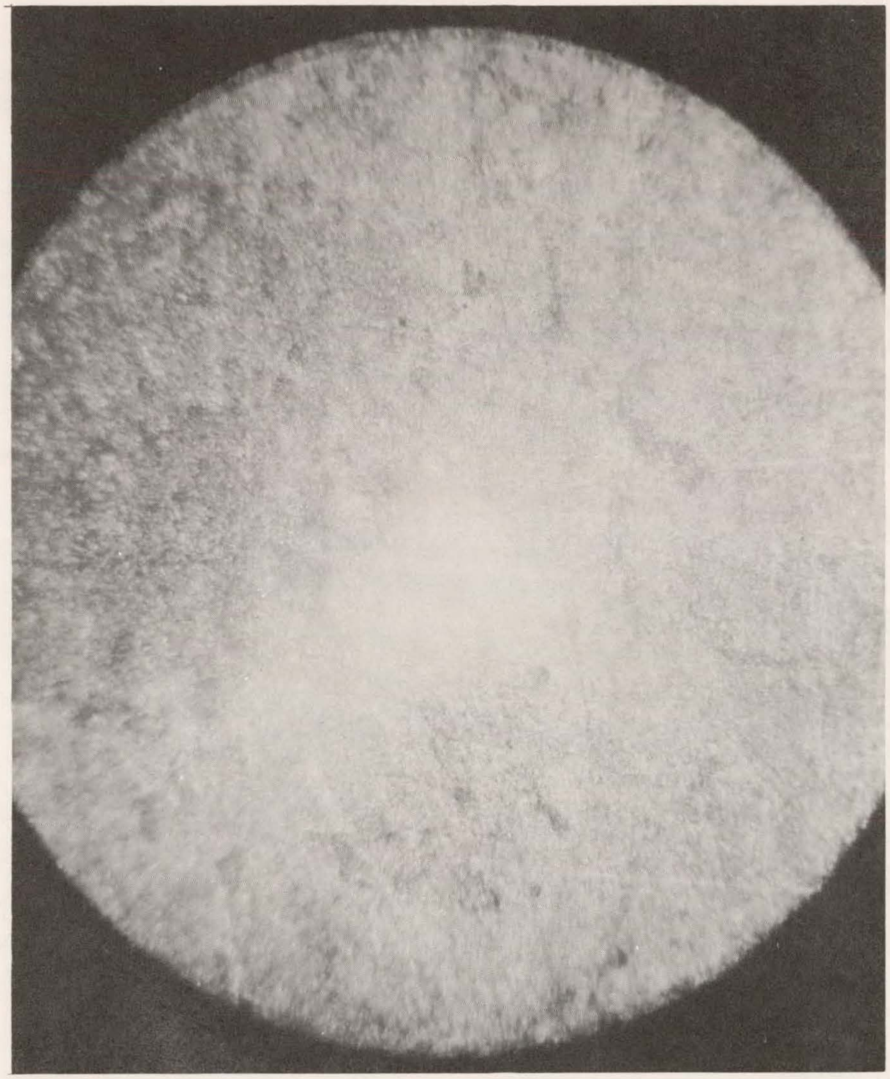


Figure 12 HPC-7: POLISHED SURFACE BEFORE TESTS

50:1

Figure 13 HPC-7: COLD EROSION SAMPLE AFTER TEST
50:1

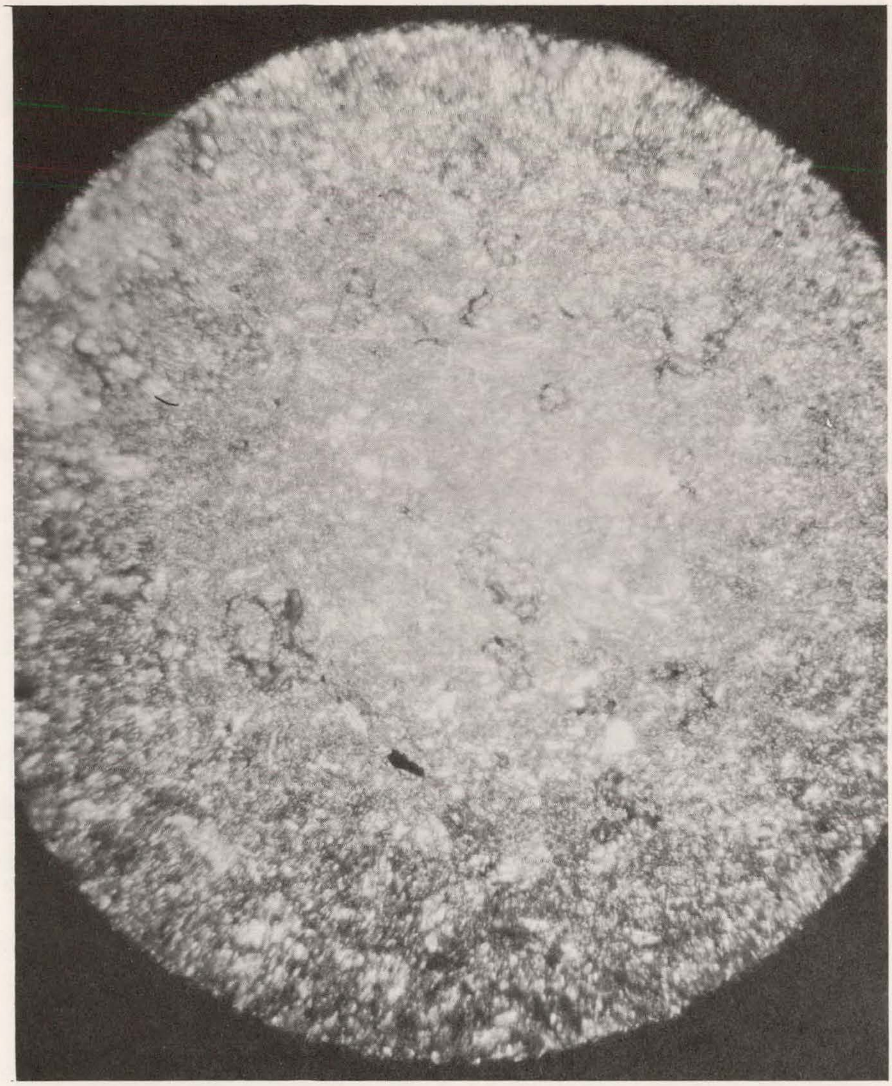


Figure 14 HPC-7: HOT EROSION SAMPLE AFTER TEST

50:1

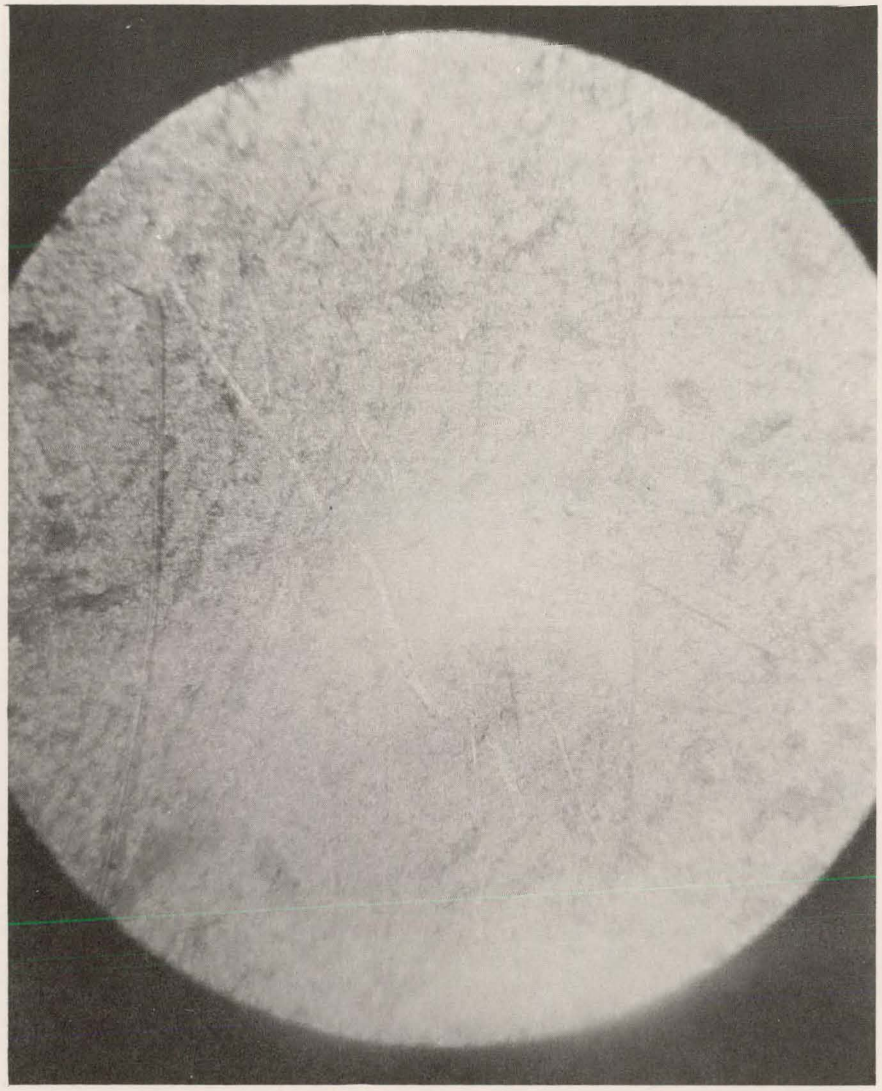


Figure 15 HP3LM: POLISHED SURFACE BEFORE TEST
50:1

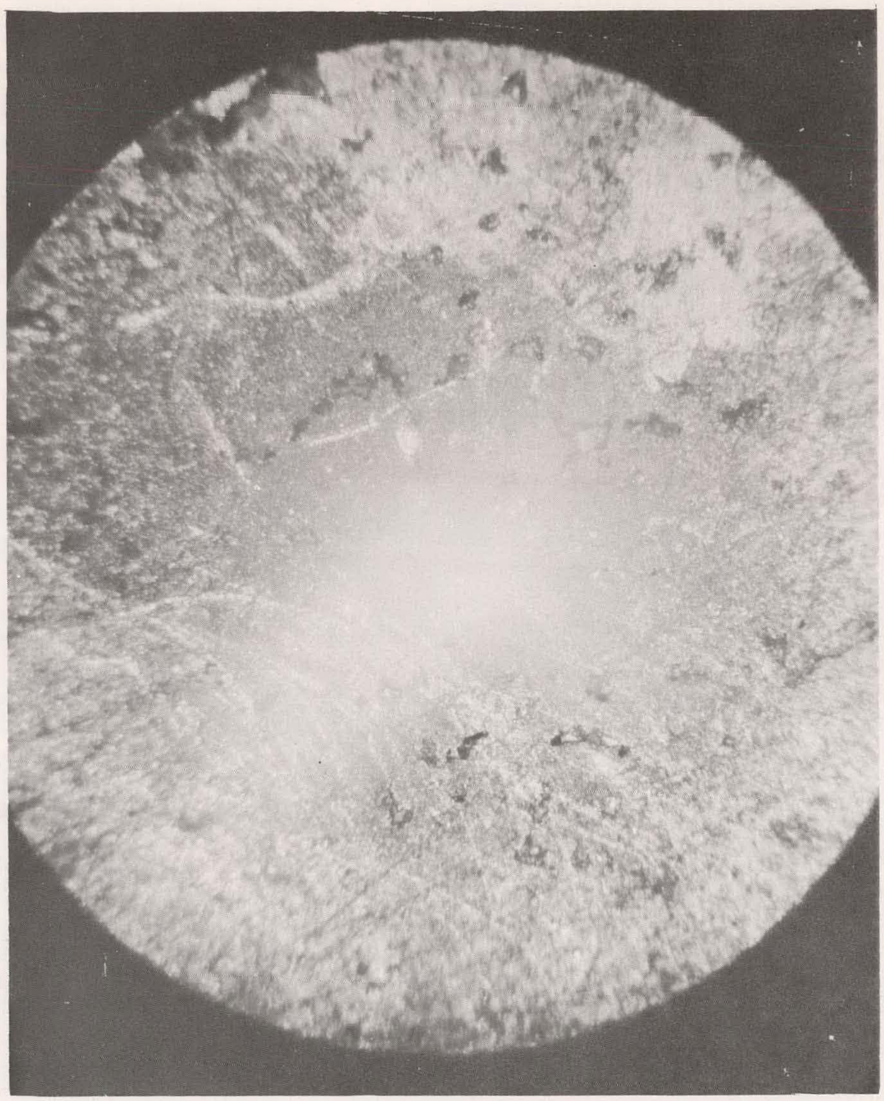


Figure 16 HP3LM: COLD EROSION SAMPLE AFTER TEST
50:1

Figure 17 HLM: POLISHED SURFACE BEFORE TESTS
50:1

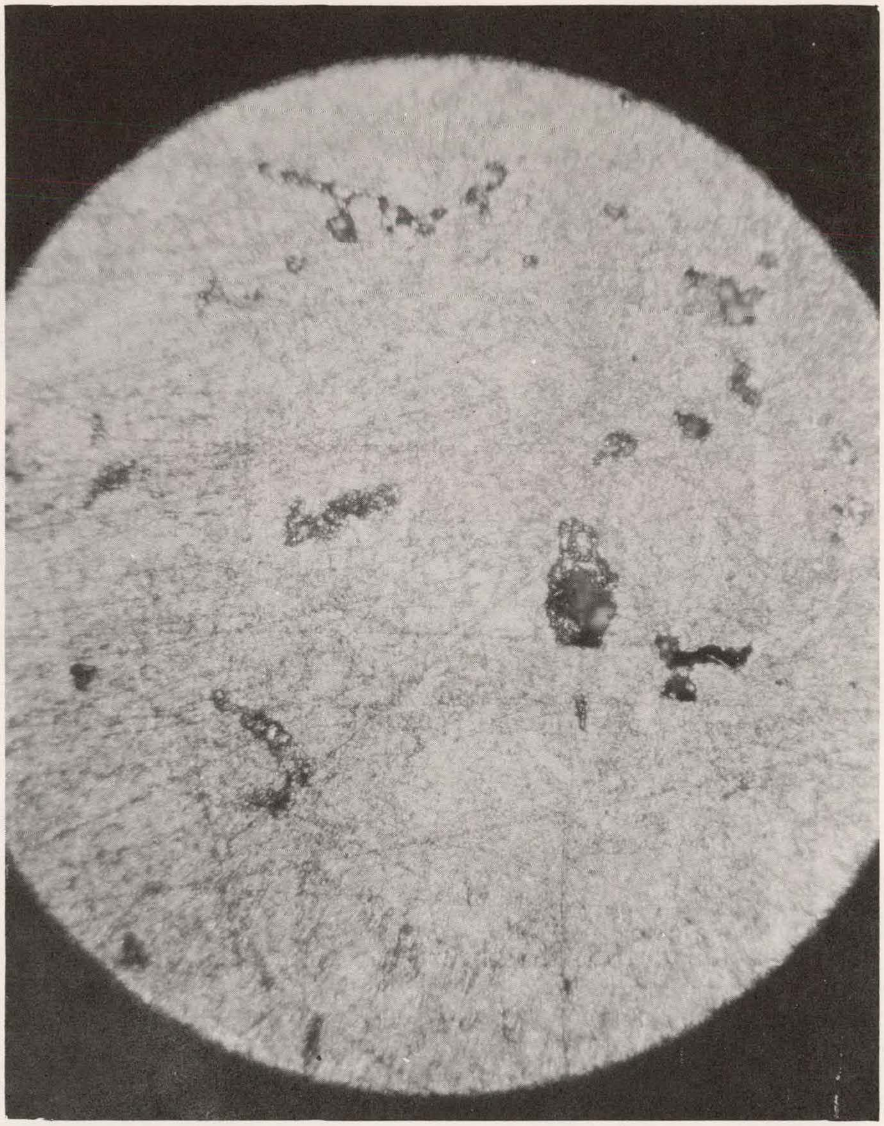


Figure 18 HLM: COLD EROSION SAMPLE AFTER TEST
50:1

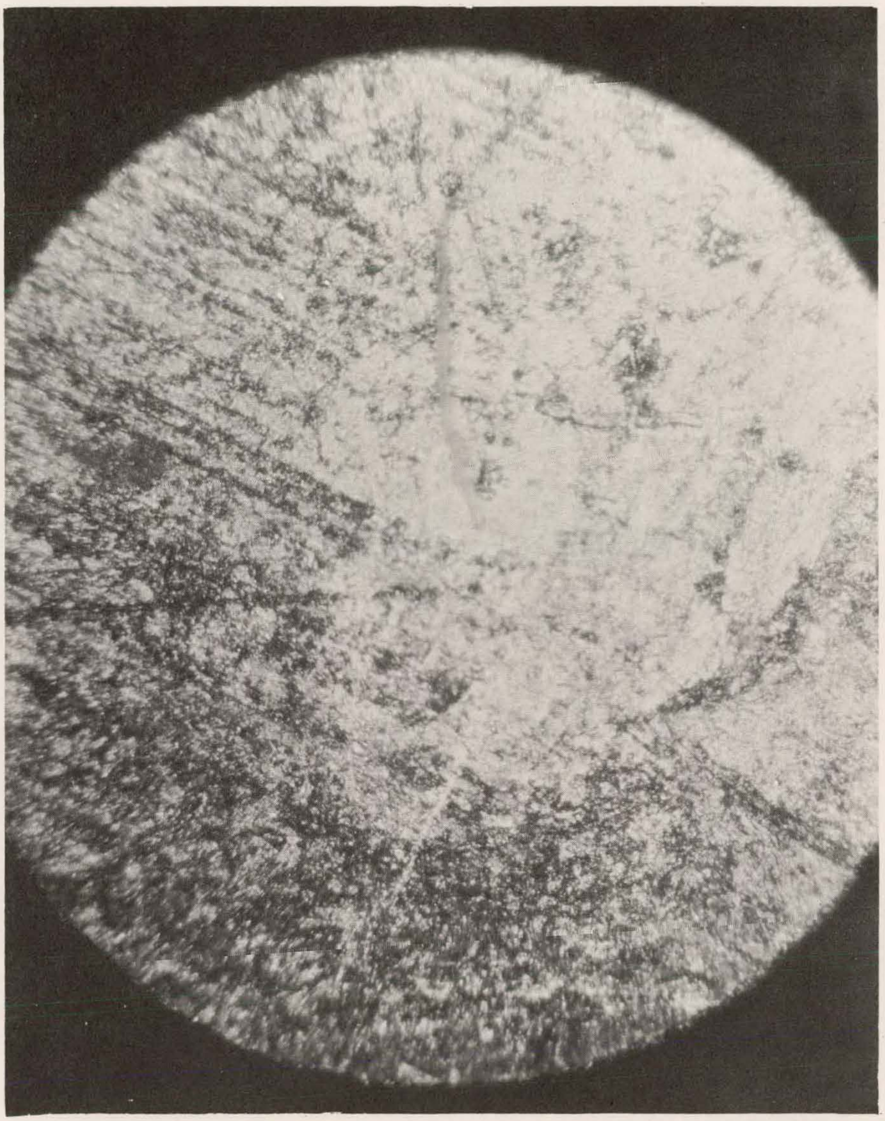


Figure 19 HLM: HOT EROSION SAMPLE AFTER TEST
50:1