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PROCEEDINGS OF THE SRE-OMRE FORUM

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.

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PROCEEDINGS OF THE SRE-OMRE FORUM

Held at Los Angeles, California

November 8 and 9, 1956

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
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PREFACE

In furtherance of the Atomic Energy Commission's policy to assist industrial, municipal, and other groups in keeping currently informed on the status of the Civilian Nuclear Power Program, the Division of Reactor Development sponsored an information forum on the Sodium Graphite and Organic Moderated Reactor development programs on November 8 and 9, 1956. The Atomics International Division of North American Aviation, Inc., who has the responsibility for these programs, was the host at this forum, which was held at the Institute of Aeronautical Sciences Building, 7660 Beverly Boulevard, Los Angeles 36, California. During the first day, the subject of discussion was concerned with the development, construction, and operation of the Sodium Reactor Experiment, which is now nearing completion. The morning of the second day was devoted to a discussion of the Organic Moderated Reactor Experiment, on which construction has just commenced at Arco, Idaho. In the afternoon, a tour of the SRE Site was conducted.

This report includes the papers presented at this forum, together with reproductions of the slides shown during the two sessions. Each group of slides is numbered separately and is included at the end of the contributing author's paper. Included also is the verbatim record of the discussion periods for the two sessions.

A brief discussion of those questions which could not be answered during the session because of time limitations is presented at the conclusion of the record of discussion for November 9, 1956.

ENTIRE CONFERENCE ... ASLM-00976

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SESSION I

SODIUM REACTOR EXPERIMENT

NOVEMBER 8, 1956

SRE ENGINEERING DESCRIPTION

W. E. Parkins

Background information on the project location and schedule are briefly presented. The engineering considerations leading to general reactor arrangement, together with present plans and objectives, are discussed.

A brief description of the installation with references to the previous forum and to the Geneva papers is given.

The primary departures from earlier design are taken up in some detail. These include subjects such as: use of tetralin as the organic coolant, control and scram modifications, pressure regulating means for the galleries, hot trap for control of sodium impurities, addition of collars in the zirconium cans, grease-packed bearings and seals in the pumps, and check valve in the primary auxiliary line.

Several points of principal concern are discussed. The physical properties of zirconium cans and long-time pump performance are two of these. Other engineering problems such as the life of the control and safety elements, thermal convection transients following scram, and oxide build-up in top shield thermal insulation are covered.

I. INTRODUCTION

The Sodium Reactor Experiment is being constructed as a part of a somewhat more general program with the Atomic Energy Commission for the development of the sodium-cooled thermal type of power reactor. Within this more general program, experiments and theoretical studies are being conducted to further the understanding and technology of this particular approach.

Emphasis is being placed on the use of graphite for the reactor moderator and reflector and on the use of slightly enriched uranium as fuel. Some consideration is being given to the possibility of other moderating materials and to the use of alternate fuels. As an example of the latter, there is considerable interest in the eventual use of U^{233} alloyed in thorium, which may make possible the application of a thermal breeding cycle.

The proposal for the construction of the SRE and the accompanying experimental program was an outgrowth of several years of engineering study at Atomics International. This study included an examination of possible power reactor types and a comparison of their suitability on the basis of known technology, estimated performance, and future potential. The contract between Atomics International and the Atomic Energy Commission for the conduct of this program, which was aimed primarily at the sodium-cooled, graphite-moderated type of reactor, was entered into on June 25, 1954. Prior to that time, a number of preliminary designs had been considered. These led to a decision that the SRE would be of a design arrangement which has been referred to as the "tank type." It would also use graphite canned in zirconium sheet as the moderator and reflector. This arrangement provided the experimental flexibility desired as well as a design which could be adapted to full-scale power reactors.

The SRE has been designed for the production of 20 Mw of heat. The original plans provided only that these 20 Mw of heat be dissipated to the atmosphere in air-cooled heat exchangers. Although the air-cooled heat exchangers have been installed, provision has also been made for a parallel-connected steam plant which will make possible the generation of electrical power. This steam plant addition has been designed and constructed by the Southern California Edison Company. It will permit the generation of more than 600 kw of electricity, which will be fed into the Southern California Edison distribution system.

II. GENERAL DESCRIPTION

I would now like to describe the design of the SRE. It will be necessary to cover the various parts of the plant very briefly. Where appropriate, I will attempt to bring out significant changes which have been made in its design. These will represent departures from earlier descriptions of the SRE design given at the time of the last Forum on this subject and in a paper prepared for the Geneva Conference on the Peaceful Uses of Atomic Energy.

The general layout external to the reactor building is shown in Fig. 1. This picture was taken on October 3 of this year. The steam plant with its cooling tower can be seen at the upper right-hand side. Pipe lines, more recently installed, will circulate high-temperature sodium over to a steam generator at the steam plant. The air-cooled heat exchangers and associated components in the secondary sodium system are located at the end of the building nearest the Edison plant. The building itself is of tilt-up concrete construction and houses the reactor, its primary system, the control room, the fuel handling facilities, and associated equipment.

As is apparent, the SRE building is not designed for containment in the sense that it can withstand an internal pressure and still be tight against leakage. The lack of a requirement for a specially constructed containment shell is an important aspect of this sodium-graphite type of reactor. No accident can be foreseen in which for chemical or nuclear reasons an increase in pressure could occur which, in turn, would demand an external shell for purposes of containment. It is true, however, that for the low pressures present during normal operation and for those which might result from possible accidental circumstances, containment is provided by the location of the reactor and the entire primary radioactive system in shielded compartments below floor level in the reactor building.

Figure 2 shows the arrangement inside the reactor building. Heat is removed in a main circuit which consists of a primary loop and a secondary loop, each circulating sodium and each capable of the removal of the full 20 Mw of heat. A similar auxiliary circuit rated for 1 Mw is to be operated simultaneously. This auxiliary circuit is required only to provide some heat removal capability in the event of a failure of some component in the main circuit. The auxiliary primary and main primary loops are contained in separate vaults adjacent to the reactor

and below floor level. A lead-shielded cask operates on a handling bridge within the reactor room. It is capable of the removal of any of the elements penetrating into the reactor core and transporting them to cleaning, storage, or hot cell facilities located at the other end of the building. Provision can be made to operate another cask on the same handling bridge for the purpose of removing and transporting the moderator assemblies from the reactor. A preliminary design of this moderator cask has been prepared, but until its need is more definite, no final design or fabrication will be undertaken.

A cross section of the reactor is shown in Fig. 3. The canned moderator and reflector assemblies are supported on a grid plate within the stainless-steel core tank. This tank is 11 feet in diameter and 19 feet high. Sodium enters in double-walled pipes from the main and the auxiliary galleries and is discharged into a plenum at the base of the tank. This sodium flows upward through channels provided in the graphite assemblies, removing heat from fuel elements suspended there. The hot sodium then flows from a six-foot deep pool above the top reflector through outlet pipes leading to the main and the auxiliary primary loops. Proceeding radially outward from the core tank, there is a 5-1/2-inch-thick thermal shield of cast steel, a 1/4-inch-thick steel outer tank, approximately 12 inches of thermal insulation, a 1/4-inch-thick carbon steel liner, and 3 feet of ordinary concrete. The purpose of the outer tank is to contain the sodium in the event of a leak in the core tank. The liner serves as a form for the concrete and has attached cooling pipes to remove the small amount of heat generated in the concrete as well as that conducted across the thermal insulation layer.

Since the core tank and the outer tank are each supported at their bases, a flexible metal joint for thermal expansion is installed at the top of each tank and provides a low pressure gas seal. The top shield is comprised of a ring section and a central "loading face" section containing all of the individual plugs installed above the fuel and other elements penetrating into the reactor core. These shields are approximately six feet thick and contain concrete made with magnetite ore aggregate. The bottom surface of each shield contains a layer of cooling coils and a steel plate thermal shield.

In order to heat the system prior to filling with sodium, electrical heaters are attached on all piping and sodium components. For the reactor itself, this preheating will be performed by immersion heaters suspended on special plugs

installed in the top shield and by electrical heaters affixed to the underside of the outer tank. The immersion heaters will be removed and the outer tank heater turned off once the primary system is filled and operating. The only special cooling for the reactor other than that provided by the sodium within the core tank is due to a fluid circulated in pipes attached to the outside of the cavity liner and others located near the bottom surface of the top shield. This fluid is tetralin, chosen because of its particular physical properties and because of the fact that it will not react with the liquid sodium in the event of an accidental leak.

The loading pattern for the SRE is shown in Fig. 4. Thirty-one fuel elements are located on a triangular pattern at the centers of the moderator assemblies. Some 12 extra fuel channels are provided for flexibility in loading. In addition to the channels at the centers of the moderator assemblies, channels are provided at corner locations for special elements. These include control elements, safety elements, the neutron source and experimental assemblies. The elements in these corner locations, as well as the outside surfaces of the canned graphite assemblies, are cooled by slowly-flowing sodium which returns from the main primary loop through a branch line. This sodium, which amounts to approximately seven per cent of the sodium flow in the main primary loop, is admitted at very low pressure into a plenum above the grid plate but below the graphite assemblies. The branch line flow is adjustable, by a throttle valve, and will normally be set so that this sodium will attain approximately the temperature of the exit pool in passing up through the interstices of the core.

III. FUEL ELEMENTS

The SRE fuel element is illustrated in Fig. 5. It consists of a cluster of seven rods, each fabricated by sealing a six-foot-high column of slightly enriched uranium slugs in a thin stainless-steel-jacket tube with liquid NaK contained within the tube for thermal bonding. The six outer rods are wrapped with a helical wire to prevent physical contact of the rods with each other or with the zirconium tube which forms the sodium channel along the axis of the graphite assembly. Alternate rods are wrapped in counter directions to produce a maximum mixing of the flowing sodium. An orifice plate is located at the bottom of the fuel cluster to permit operation of the uranium fuel at a maximum central

metal temperature of approximately 1200° F for the particular channel occupied by the fuel element. A thermocouple contained within the fuel element hanger rod has its junction near the top of the fuel element cluster in order to measure the temperature of the exit sodium.

IV. MODERATOR AND REFLECTOR

An illustration of the canned graphite assembly being used in the moderator region is given in Fig. 6. This design uses a hexagonal column of graphite approximately 11 inches across the flats and 10 feet high, clad in zirconium sheet. The zirconium of the side surfaces and the axial channel tube is 0.035 inch in thickness. For strength requirements, 1/10-inch-thick zirconium is used at the bottom and top heads. A stainless-steel support pedestal is bolted to the bottom head, and a stainless-steel spacer plate and lifting fixture is bolted to the top head. In order to control the pressure inside the can at a known safe value, a vent tube communicates to the helium plenum above the free surface of the sodium. This vent tube is protected by a stainless-steel guard tube attached to the top spacer plate.

While considerable work was performed in developing satisfactory manufacturing and inspection techniques, the zirconium-canned graphite assembly design has not been changed significantly from earlier concepts. One change which was proved necessary by experimental work and which has been incorporated is the addition of a thick-walled collar section at each end of the axial tube. These collars are welded into the top and bottom heads and give necessary added strength against fatigue failure for the stresses which are encountered at the tube-to-head welds at the time of reactor scram. The collar sections are made from 1/10-inch-thick zirconium and are each 1-1/8 inches long. This modification has been made on all assemblies containing an axial tube.

V. CONTROL ELEMENTS

The control element consists of a column of boron-nickel alloy rings assembled on a tube which is moved in a stainless steel thimble extending down from the top shield into the core. This element is illustrated in Fig. 7. The original

design included a ball-nut screw assembly to provide the vertical motion wherein the column of poison rings was attached to the nut, and the screw was driven through a clutch at the upper surface of the top shield by a motor and gear unit.

The many requirements for this control element, which consisted of provision for a helium atmosphere within the thimble, limit switches, and shielding in the upper section, contributed to a rather complex design for the space available. A number of experiments have been carried out to appropriately modify and prove out this proposed design. These experiments have been successful at ordinary temperatures but have indicated a limited life for a control element which must operate with the ball-nut in a high temperature zone. The calculated temperature requirement for this mechanism in the SRE is 1100° F. The present design using MoS_2 lubricant, Haynes 25 alloy screw, and nut and Haynes cast No. 6 Stellite balls appears adequate for control shim action, but not for the purposes of the regulating rod, where motion must be nearly continuous. As a result, a Mark II design of the control rod has been undertaken to locate the critical parts above the reactor top shield in a lower temperature zone. Present plans call for completing two of these units in sufficient time to install them in the initial SRE assembly prior to the critical experiment. The Mark II units will be used for regulating action, whereas the Mark I design is to be retained for the two shim rods.

VI. SAFETY ELEMENTS

The safety element for the SRE is similar to the control element in that it uses a column of boron-nickel alloy poison rings, a stainless steel thimble, and a ball-nut screw mechanism. The safety element, however, has the additional complication of a latch mechanism for dropping the poison column and a snubber mechanism for its deceleration (see Fig. 8). The service requirements on the ball-nut screw assembly are much less severe than in the case of the control rod, so that the same difficulty of temperature limitation on materials has not been so important. Other difficulties associated with the rapid release and the snubber actions have been encountered during life tests on prototype safety-element units operated at temperature. The results of these tests have indicated that simpler solutions to the mechanical problems would be obtained by again locating the

critical mechanisms in a region above the reactor top shield. On this basis, a Mark II safety element design has been prepared, but present plans call for initial start-up of the SRE using all four safety elements of the Mark I design.

VII. TOP SHIELDING

The top shield for the reactor is illustrated in Fig. 9. The central loading-face portion, some 140 inches in diameter, contains stepped casings for plugs over each of the core channels requiring fuel or other special elements. In addition there are three intermediate sized plugs - two 40 inches in diameter and one 20 inches in diameter - to provide access to the various canned graphite assemblies without removal of the loading face itself. Access to any particular assembly can be achieved by rotating the loading face to a proper angular position within the ring shield and by removing the appropriate intermediate sized plug. Cooling lines and electrical leads are routed across the top of the loading face just beneath a steel cover plate and within a two-inch-thick layer subsequently filled by steel shot. This layer of shot rests on the top surface of the dense concrete filling the loading face shield, and is provided to prevent lateral "shine" from an irradiated fuel element as it is being removed from the reactor into the handling cask.

On the underside of the loading face is attached a layer of thermal insulation. It is comprised of 13 thin stainless steel plates, spaced approximately 3/4 inch from plate to plate. The helium atmosphere with sodium vapor is free to communicate within this assembly of reflective insulation. The undersurface of the top shield is intended to operate at approximately 150° F as determined by the tetralin cooling coils just inside the shield. If it should ever be necessary to melt out condensed sodium, it is possible to permit the underside of the shield to be warmed to a temperature above the melting point of sodium.

VIII. SODIUM COOLING SYSTEM

A simplified flow diagram for the SRE cooling system is shown in Fig. 10. Nominal temperatures are 500° F for the return sodium from both the main and auxiliary circuits to the reactor. The sodium exit temperature from the reactor is nominally 960° F. The secondary loops operate approximately 60° F lower than these temperatures.

Each of the four loops contains a mechanical pump for circulation of sodium. Cold traps are also provided for the purpose of reducing the oxygen contamination in the sodium. The primary loops and the main secondary loop have circulating cold traps. Simple diffusion cold traps are used on the secondary fill tank and on the expansion tank in the auxiliary secondary loop. In order to permit clean-up of the cold traps in the radioactive primary system, a disposable cold trap has been installed in a separate shielded vault. Proper valving will permit the circulation of sodium through the "in-line" cold traps in the primary loops to the disposable cold trap. Proper adjustment in temperatures will result in the transfer of the sodium oxide to the disposable unit.

Expansion tanks are contained in both of the secondary loops. None are required in the primary system since the reactor vessel itself serves as an expansion tank. A fill tank is provided for each of the secondary and primary systems. The primary fill tank and all connecting lines are shielded, since it may be necessary to drain the radioactive primary sodium into this tank for the purpose of performing repairs after the reactor has been in operation.

An intermediate sodium-to-sodium heat exchanger is contained in the main and the auxiliary circuits. In addition, each circuit has a forced-convection-cooled sodium-to-air exchanger in its secondary loop. The addition of the steam plant by the Southern California Edison Company has required the incorporation of a sodium-heated steam generator. This change has been made in the main circuit only and valves have been arranged so that the sodium can be circulated to either the steam generator or the main sodium-to-air exchanger.

Another important change in the SRE sodium system is the addition of a hot trap. Its function is also to remove contaminates from the sodium, but its operation is based on a different principle than that of the cold trap. The hot trap operates at a high temperature of approximately 1200° F and getters the sodium by the action of a large surface area of thin zirconium sheet contained in the trap. Two hot trap units are located in the disposable cold trap vault where either may be used at any time to pass a small flow of the primary sodium. This hot trap addition has been made for the sole purpose of affording better protection for the zirconium cans in the SRE core against oxygen and other contaminates in the sodium.

The valve arrangement for the sodium system has undergone minor revisions. As mentioned before, valves were provided to divert sodium to the steam generator. In addition, it has been necessary to incorporate a check valve in the auxiliary primary loop. A study of emergency conditions in the SRE indicated that a failure of the auxiliary primary pump would result in a backward flow of sodium in the auxiliary primary loop. This, in turn, could produce excessive thermal stresses in the auxiliary intermediate heat exchanger. This backward flow is prevented by the added check valve. All other valves, both blocking and throttle type (Fig. 10), are of the frozen stem seal design. This type of sodium valve seal has proved very satisfactory in practice, and is used on all valves located in the two primary and two secondary circulating loops. Bellows seal valves are used in other locations (not shown in Fig. 10) involving lines for system draining, filling, and sampling, for trap operation, and for other functions. Many of these lines also contain vapor traps and freeze traps. The vapor traps are for the purpose of condensing out sodium vapor to prevent its deposition in the gas lines. They also serve to reduce the amount of radioactivity carried into the lines by sodium vapor from the primary system. The freeze traps serve to prevent surging of liquid sodium up into unheated lines where it might solidify and make a removal difficult. The freeze trap localizes the frozen region from such an occurrence in a region where heating can be provided to restore the sodium level to normal.

The primary main pump is illustrated in Fig. 11. The basic principles of this and the other mechanical pumps have remained unchanged, but some improvements have been incorporated. The primary pumps were originally designed for lubrication at their lower radial bearings by a slow flow of oil which was carried out of the pump casing by a continuous bleed of helium gas - also used to provide an inert atmosphere in the casing. Lubrication by oil flow was considered undesirable because of the decomposition resulting from the primary sodium gamma radiation. This design has now been modified to an arrangement assuring greater operational reliability. The lower bearing is to be packed with a grease which exhibits good stability against decomposition by radiation. To assure an adequate life even of this grease, it has been necessary to add a small amount of special shielding consisting of approximately 2 inches of steel to the underside of this bearing. The upper bearing is situated above the top plate of the pump casing

and above the steel-shot shielding in the upper region of the casing. It will be lubricated by packed grease as was planned in the original design. Minor design improvements have also been made in the gland at the frozen sodium shaft seal and at the shaft seal for the helium gas at the top plate of the pump housing. Another design improvement in the form of "anti-convection" rings in the sodium annulus at the casing seal have been incorporated to reduce the heat load in that region.

The intermediate heat exchanger uses a shell and tube counter-flow arrangement of a "U"-type design. The main intermediate heat exchanger is illustrated in Fig. 12. No important changes in its design have been made over earlier concepts. The same is true of the forced draft sodium-to-air exchanger used in the secondary loop. An illustration of the main sodium-to-air exchanger is given in Fig. 13. A description of the new sodium-heated steam generator is covered in a subsequent paper contributed by the Southern California Edison Company.

The arrangement being used for the circulating cold traps is shown in Fig. 14. Considerable experimental work has been performed on this component and has resulted in certain dimensional changes, especially in the economizer section. The principles of operation, however, including the provision for temperature control by boiling toluene, have been unchanged. The design of the disposable cold trap is similar to that of the "in-line" cold traps except for a provision to open the cold trap vessel at its top head in order to remove the stainless-steel entrainment mesh. All of the "in-line" cold traps have plugging meters connected across them for the purpose of measuring the oxygen concentration of the sodium. This plugging meter consists of an economizer section and a plugging section wherein the plugging orifices are formed by holes in a valve plug.

The hot trap, mentioned earlier, is illustrated in Fig. 15. It is designed for the circulation of approximately 24 gpm of sodium at 1200° F. The circulation path in the hot trap consists of a single pass, but the inlet and outlet lines flow through an economizer (not shown in Fig. 15). The packing within the hot trap uses 320 pounds of 0.004-inch-thick corrugated zirconium sheet.

IX. ORGANIC COOLING SYSTEM

The special cooling of components in the SRE is provided by a system which circulates liquid tetralin. An organic fluid which will not react with liquid sodium was chosen as a safety precaution in the event that these two fluids should be accidentally brought in contact. The organic fluid which had been chosen for this special cooling system at an earlier date was toluene. For several reasons, principally because of the higher boiling point and reduced hazards associated with toxicity and flammability, this choice has been changed to tetralin. The boiling point of tetralin is 405° F, compared with 232° F for toluene. It has been unnecessary to significantly modify operating conditions in the organic cooling system with this change in fluid. Maximum heat load conditions will require a flow of approximately 350 gpm with a change in the tetralin temperature from 100° F to 120° F. A schematic flow diagram for this organic cooling system is given in Fig. 16. Principal components serviced include pumps, valves, cold traps, fuel storage cells, and cooling circuits in the galleries, as well as the cavity liner and top shield at the reactor. All other cooling required at the SRE is provided by the sodium system itself.

X. HELIUM SYSTEM

Many locations around the SRE installation require a helium inert atmosphere. A special piping system, shown in Fig. 17, has been provided for this purpose. Helium from supply bottles is passed through a NaK bubbler for purification, stored in a 50-psi tank, and made available to a number of components. In most instances, the helium is used to provide an inert atmosphere over a sodium surface. It is used in certain other locations where a leak might bring oxygen into contact with the sodium. For reasons of pressure control, a number of the components serviced by the helium system have pressure-regulated outlet lines connected into the vent system. The actual gas discharged is released from the building stack or directed to the radioactive gas storage, depending upon its radioactivity level.

XI. NITROGEN SYSTEM

Gaseous nitrogen is provided to several other locations in the SRE. These are the two primary galleries, the primary fill tank vault, the disposable cold trap vault, and the cavity at the reactor containing the thermal insulation. This nitrogen system is shown in Fig. 18. Again, gas from supply bottles is stored at 50 psi and made available through separate pressure-reducing valves to the five locations just enumerated. An important change has been made in the design of the nitrogen system by the elimination of a constant-pressure tank. This tank and its connections had been designed to maintain a nearly constant pressure in the various galleries and vaults in spite of temperature variations occurring. Such an arrangement, requiring a tank of a maximum of 1000 cubic feet, would reduce nitrogen gas consumption while maintaining the pressure in the correct range. This tank has been eliminated in favor of separate pressure relief valves connected into the gas vent system. These valves are set to relieve at 0.4 psig, whereas the inlet valves from the gas supply admit nitrogen at 0.25 psig. A number of reasons contributed to the change from the constant pressure tank to the gas relief arrangement, one of the more important being the impracticality of applying the constant-pressure-tank concept to a full-scale central station plant.

XII. WASTE DISPOSAL

The gaseous waste disposal system, referred to previously, is shown schematically in Fig. 19. This system accepts vented gases from many of the components serviced by the nitrogen and helium systems, in the event that radioactivity levels in the vented gas are above some predetermined magnitude. The radioactive gas is then stored in either of two shielded tanks of 5400 cubic feet at 100 psig capacity. If, at some later time, the radioactivity level is determined to be sufficiently low, the gas may be bled from these storage tanks, diluted, and discharged out the building stack.

The SRE also uses a liquid waste-disposal system, shown in Fig. 20. This waste is principally from fuel cleaning and hot cell operations. It is pumped from a sump to one or more of ten 50-gallon "hold-up" tanks. If long half-life activities are involved, the waste is eventually stored in two shielded 5000-gallon tanks.

XIII. REMARKS

The foregoing discussion very briefly describes the different components and the design of the SRE. There are some portions of the installation which have not been covered. These include the control system, the emergency electrical system, fuel storage and fuel cleaning facilities, the fuel handling cask, and the hot cells. Minor modifications have been made on the control system as a result of the addition of the steam plant. Also, some changes have been made in the circuits which produce "scram" and "set-back" of the reactor during operation. In the fuel cleaning equipment and fuel handling cask, small changes in the form of mechanical and operational improvements have been made as dictated by experimental development work.

At this date, the construction of the SRE is essentially complete and pre-operational testing has been in progress during the last month. From past experience, which as yet does not have the benefit of actual plant operation, there are many features of the SRE on which there is confidence of the adequacy of the design. This is true when evaluated from the standpoints of the engineering analysis, experimental development work, component fabrication, and field installation performed. There are other portions of the SRE on which some difficulty has been experienced. It is hoped that the design of these parts, as finally installed, will also prove to be satisfactory. In many of these cases it is true that the design has been purposely compromised. This has occurred for a variety of reasons such as budgetary limitations, lack of sufficient development information, and difficulties of manufacturing. At the same time, many of these compromises are in keeping with the original premise of the SRE, intended as a reactor experiment where certain engineering risks would be taken with the objective of developing, within a short period, a maximum of information on this type of plant.

Fig. 1. Photograph of SRE

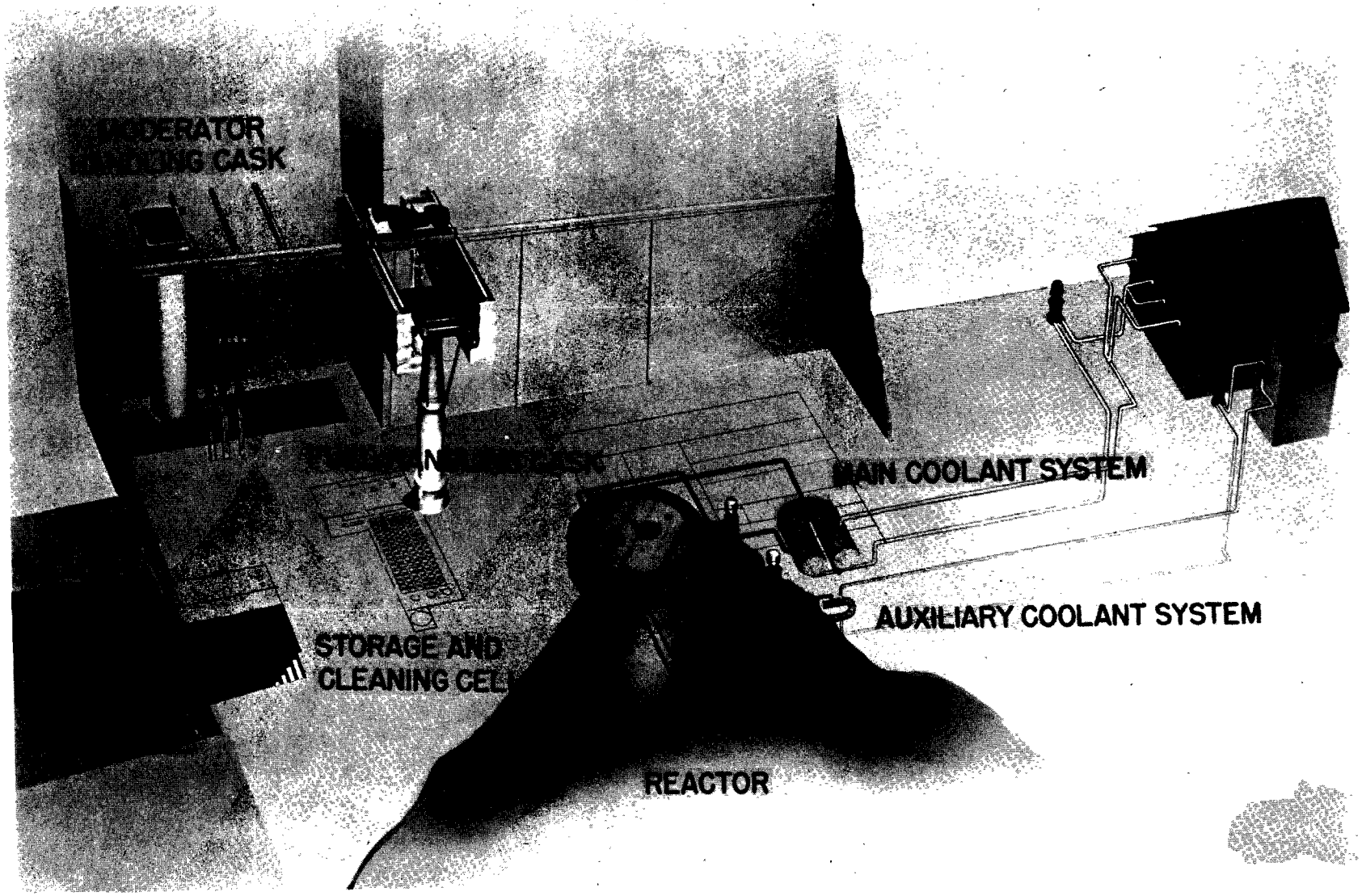


Fig. 2. Arrangement of Components

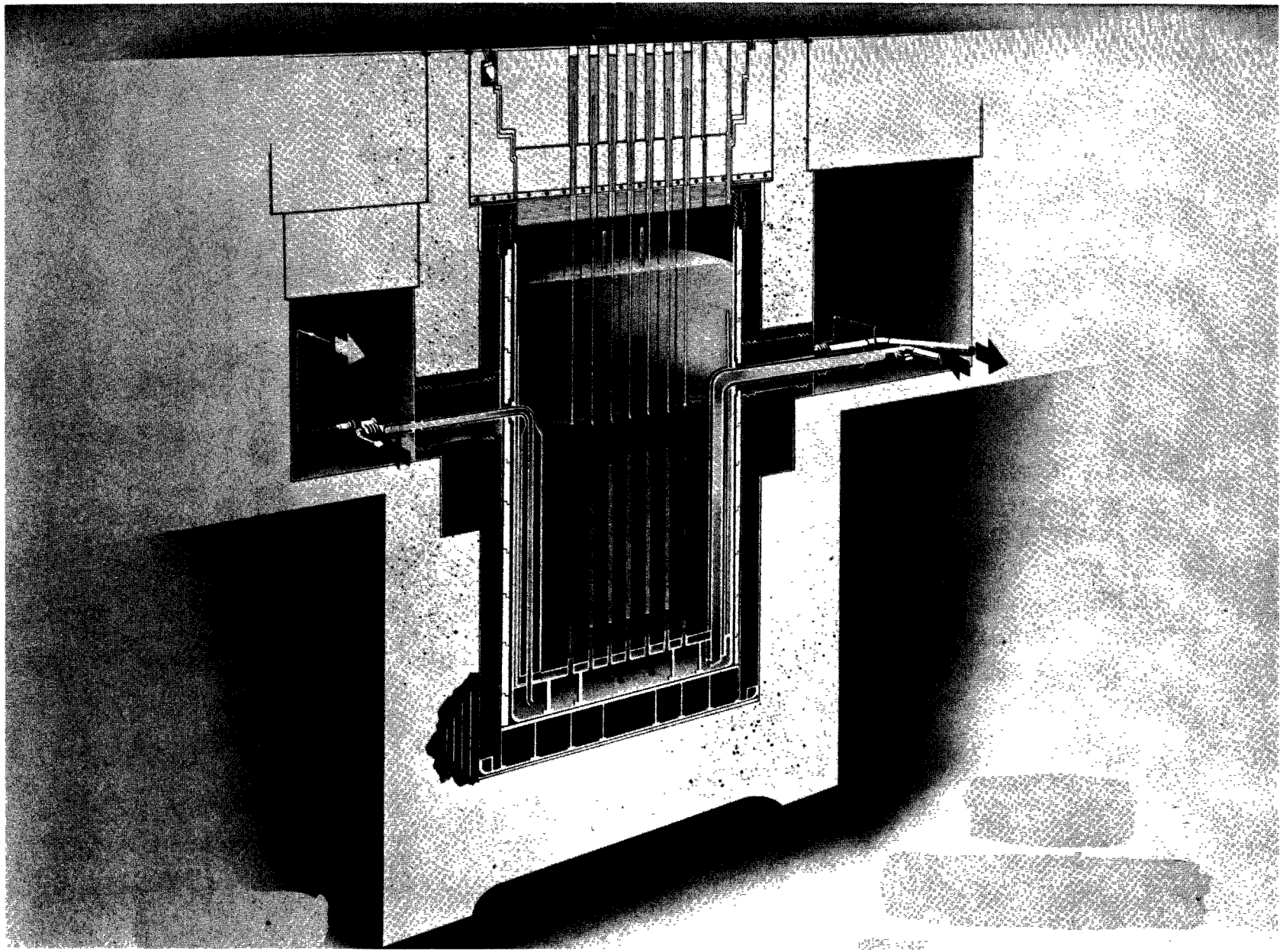


Fig. 3. SRE Cross Section

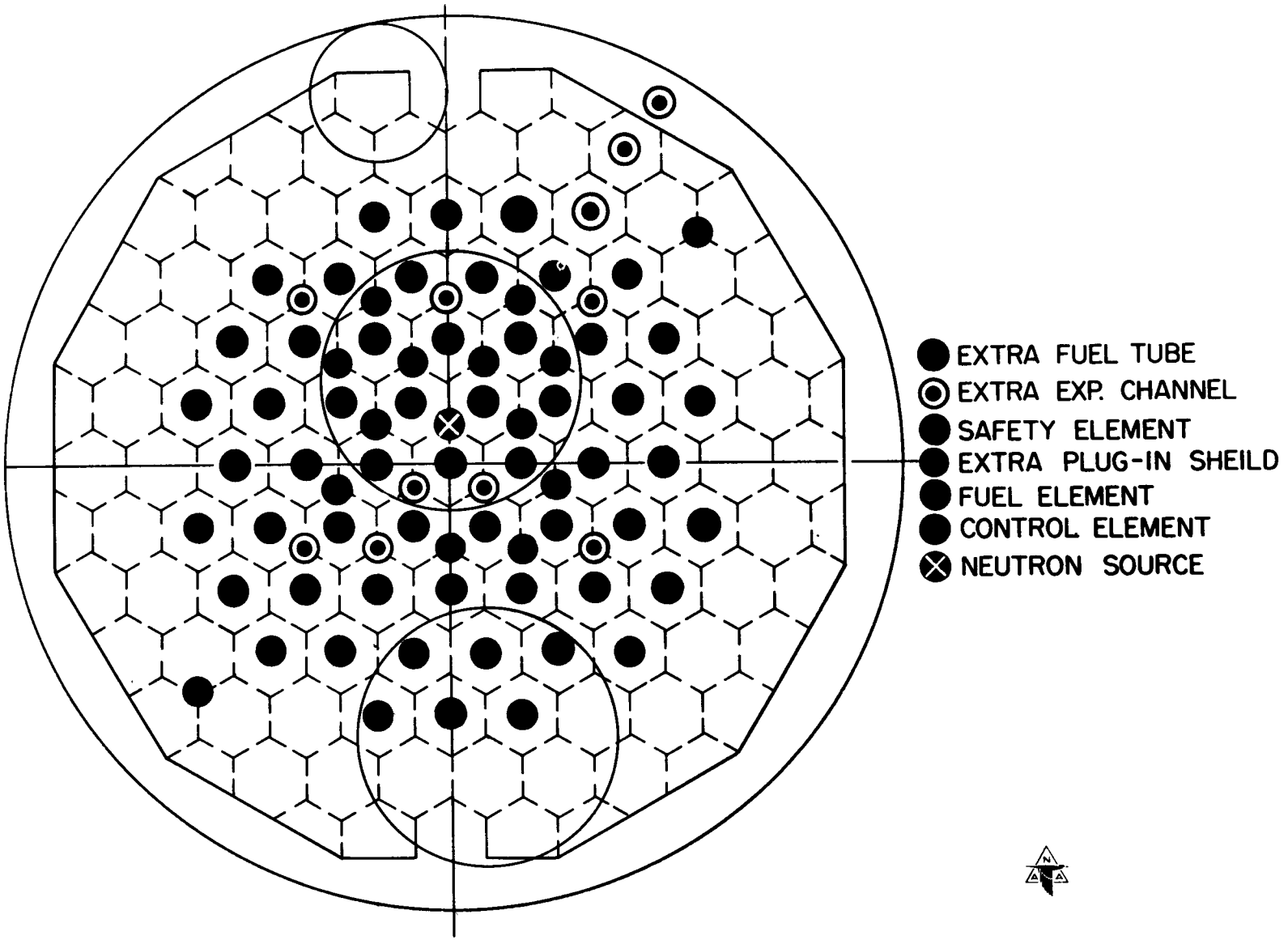
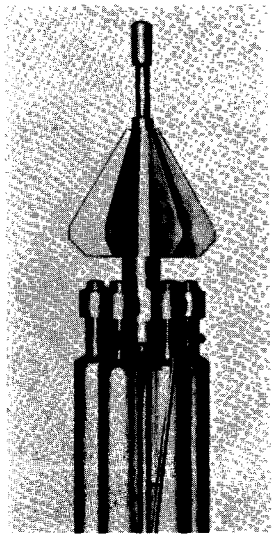
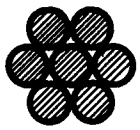
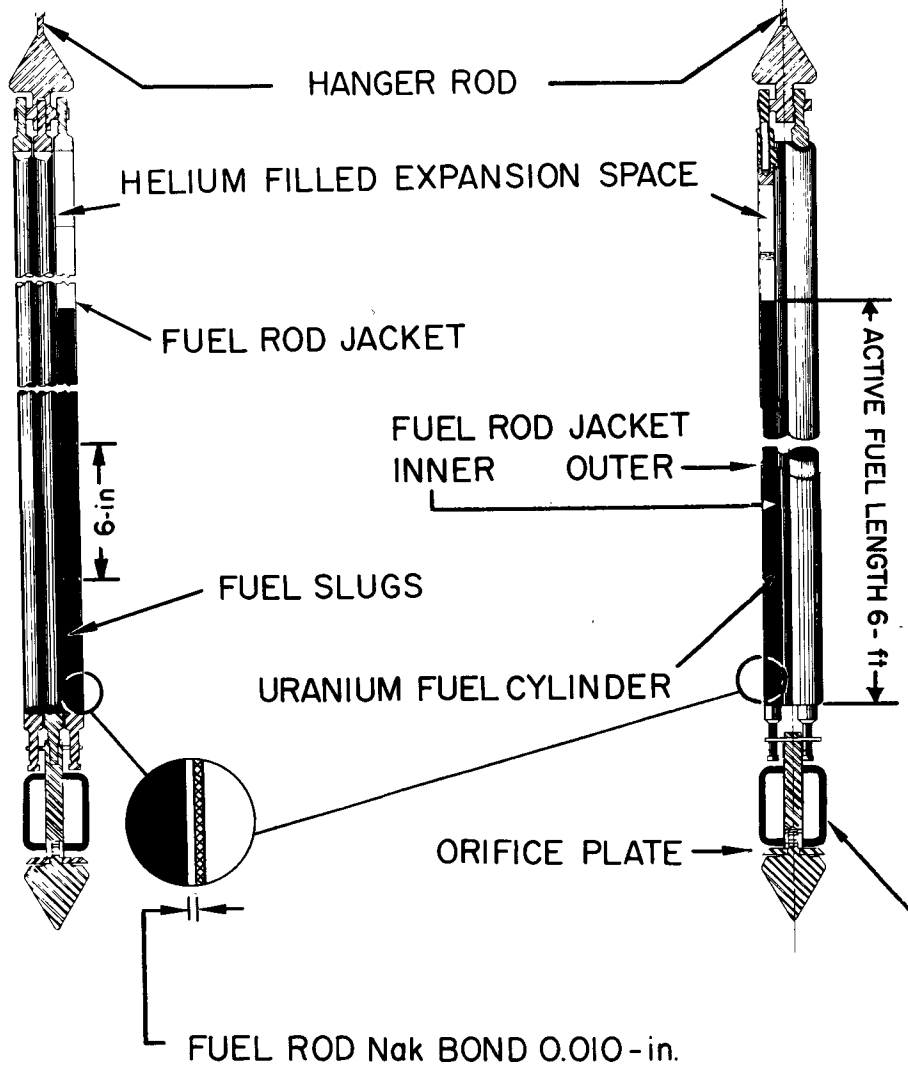


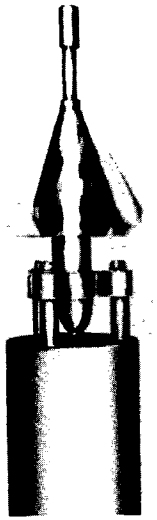
Fig. 4. Loading Pattern

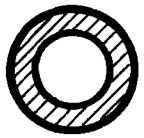




7-ROD ELEMENT







HOLLOW ELEMENT

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SCALE-INCHES

Fig. 5. Fuel Element

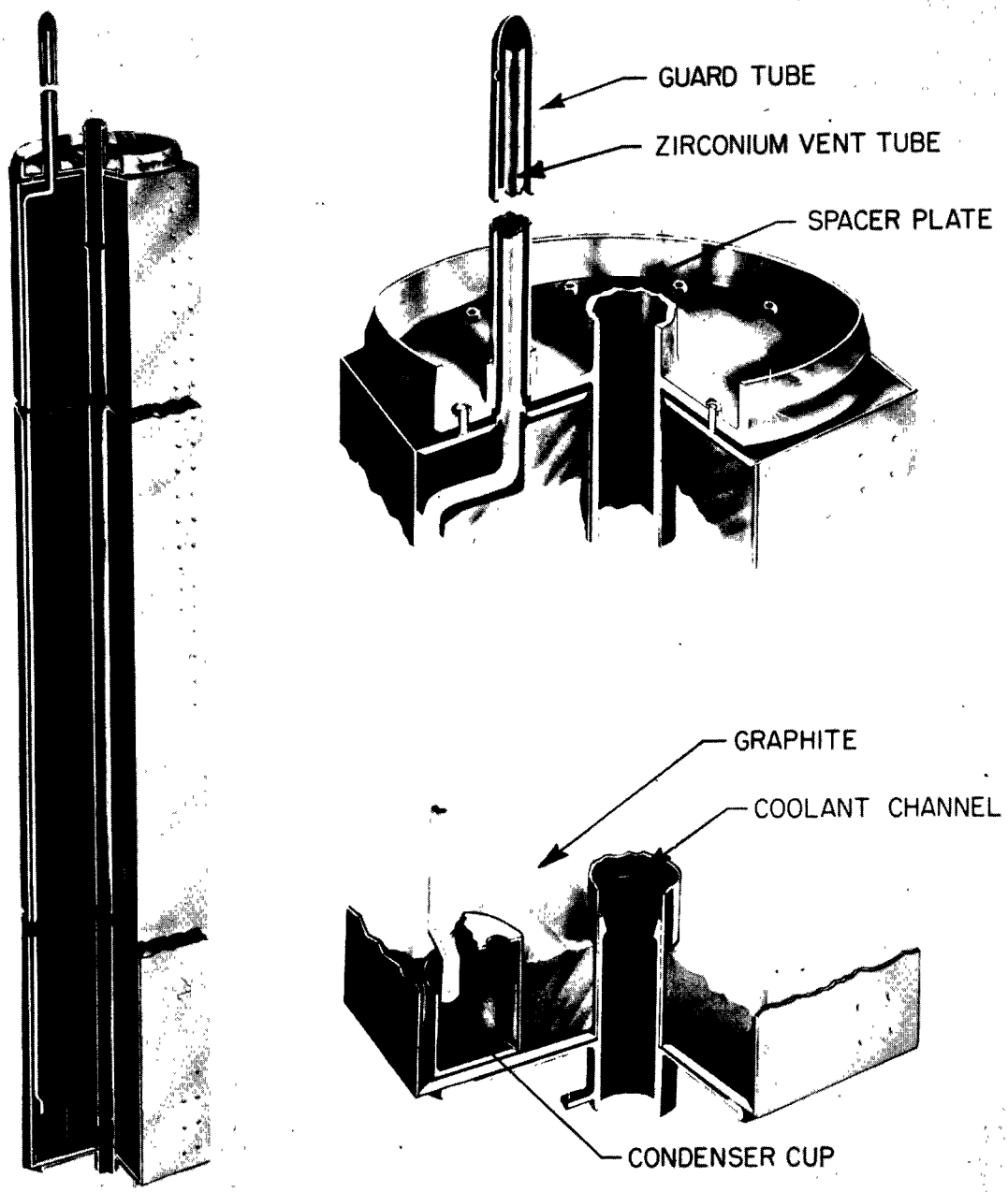


Fig. 6. Moderator Can

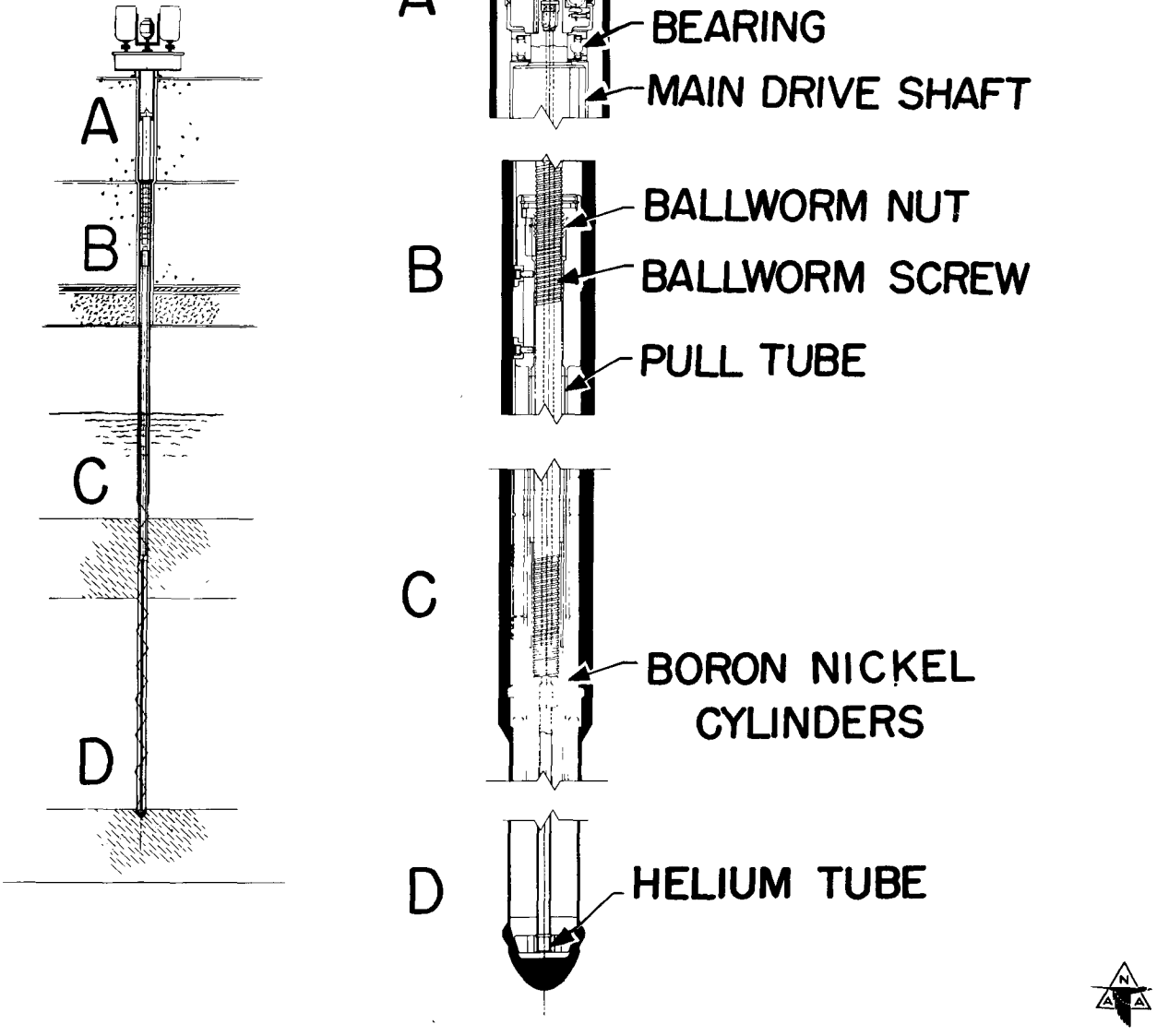


Fig. 7. Mark I Control Element

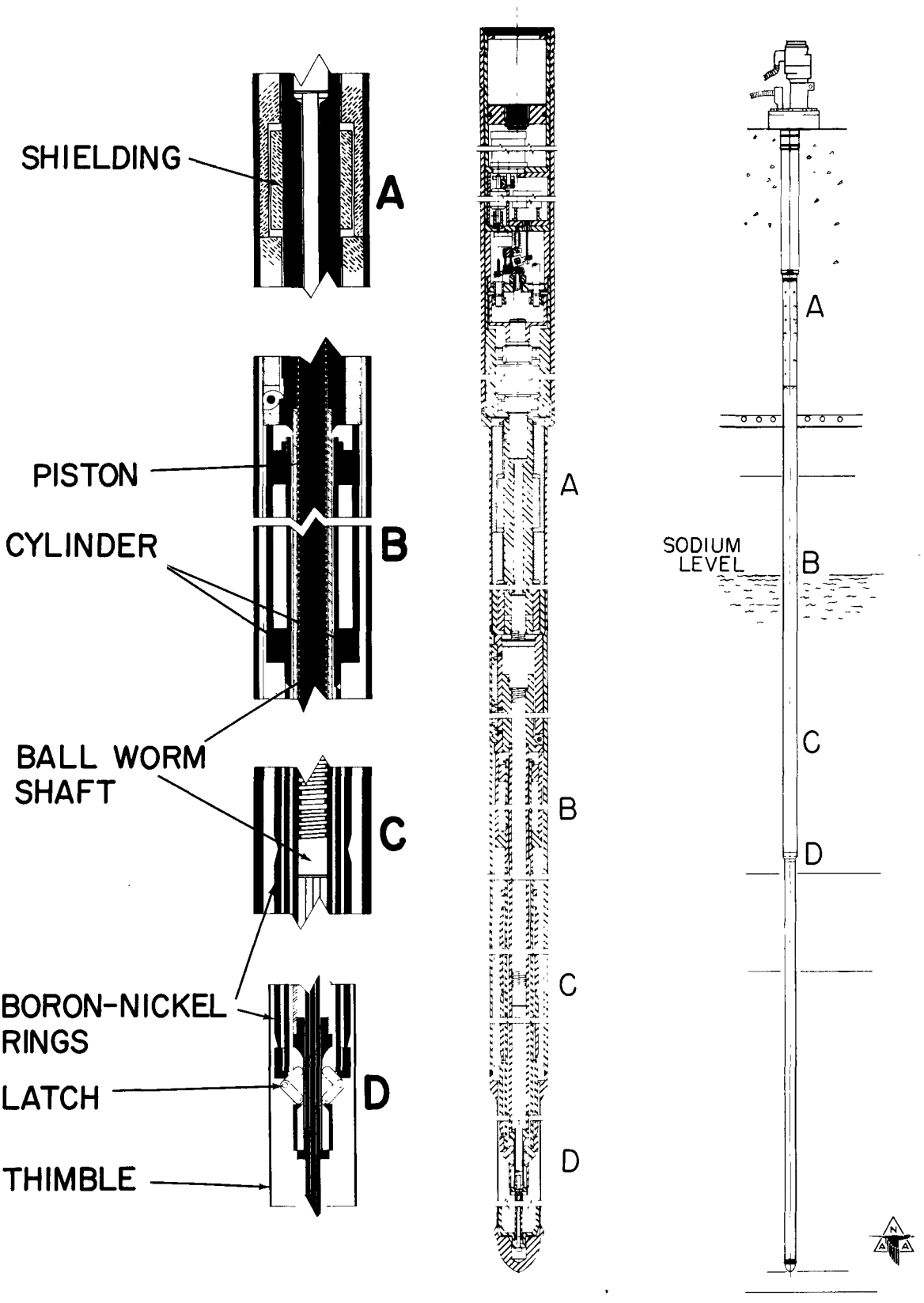
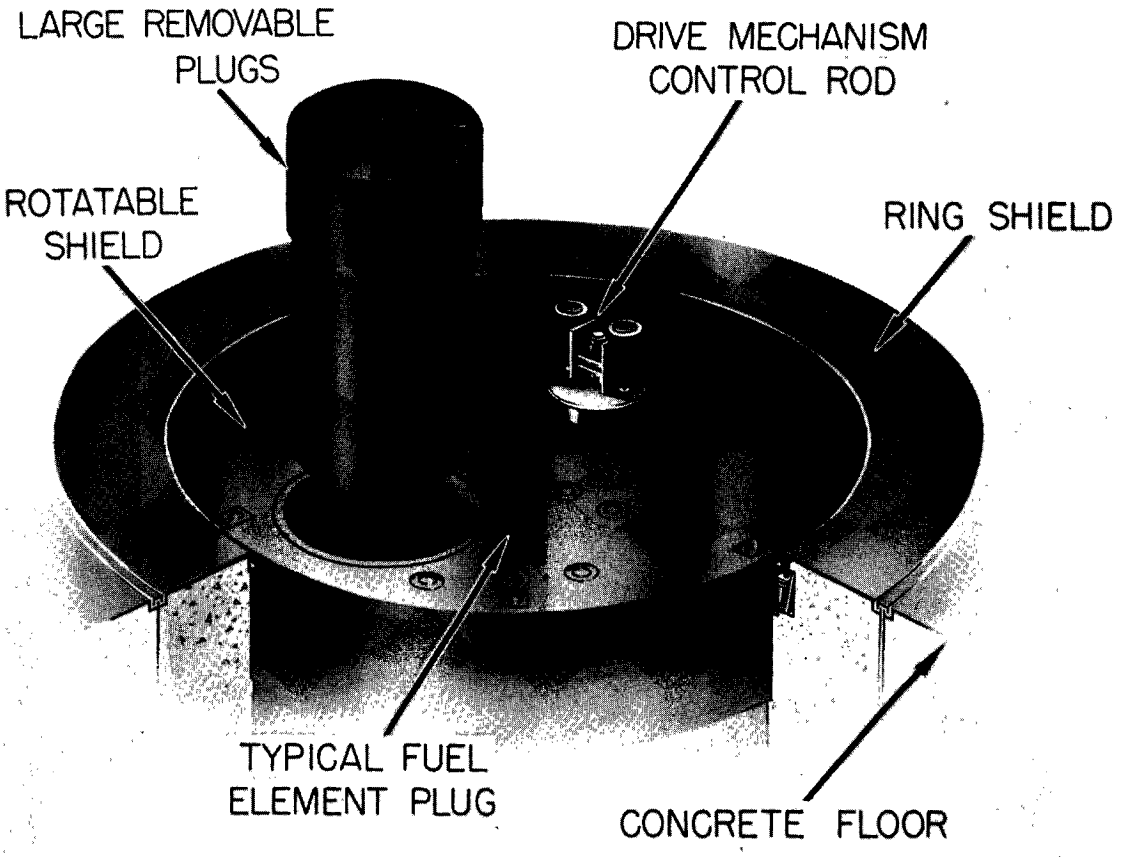


Fig. 8. Safety Element



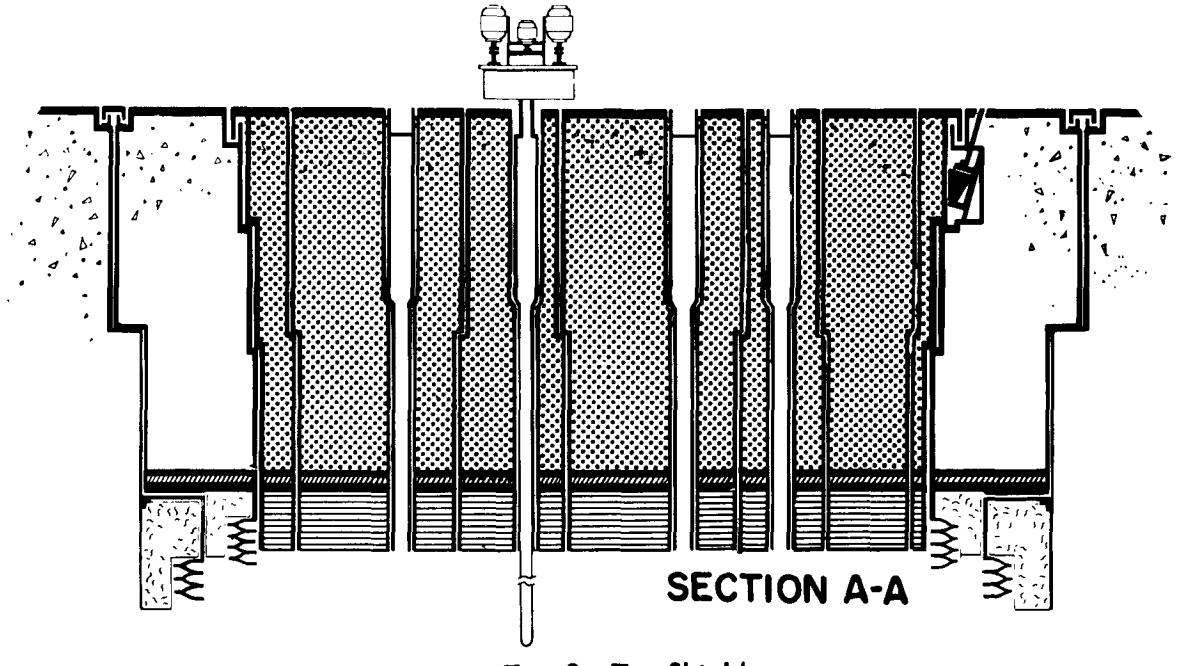


Fig. 9. Top Shield

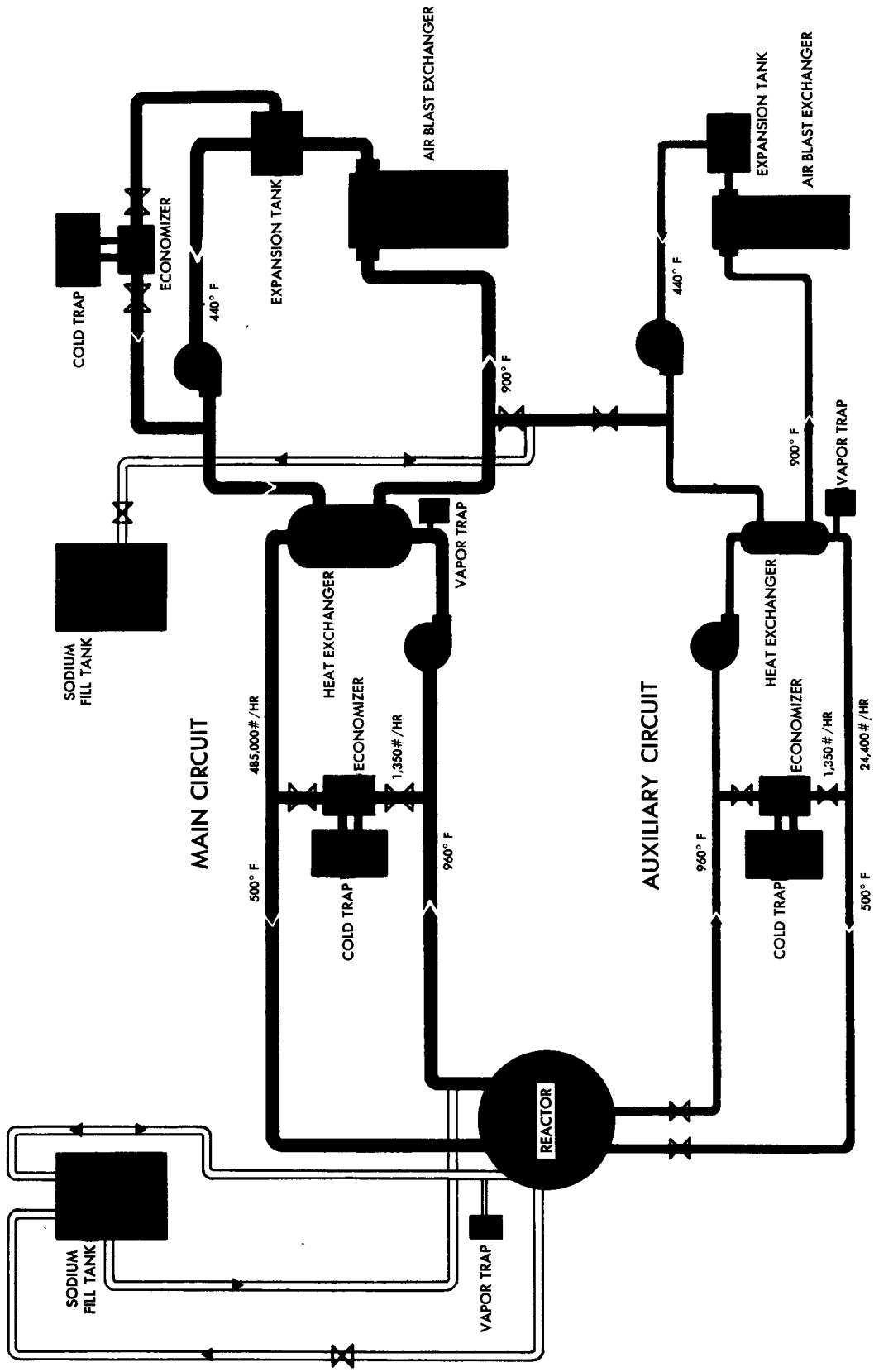


Fig. 10. Sodium System Flow Diagram

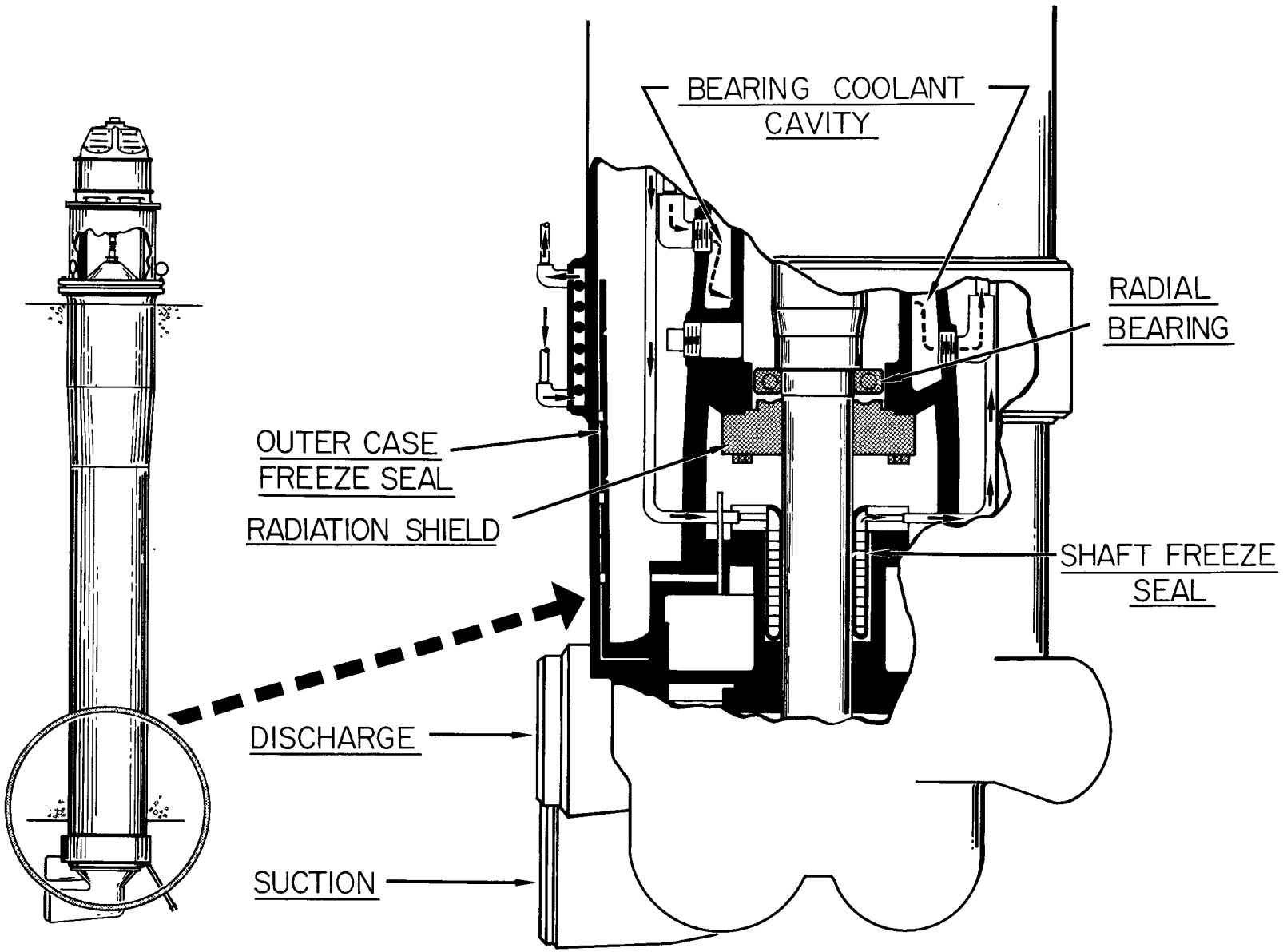


Fig. 11. Main Primary Pump

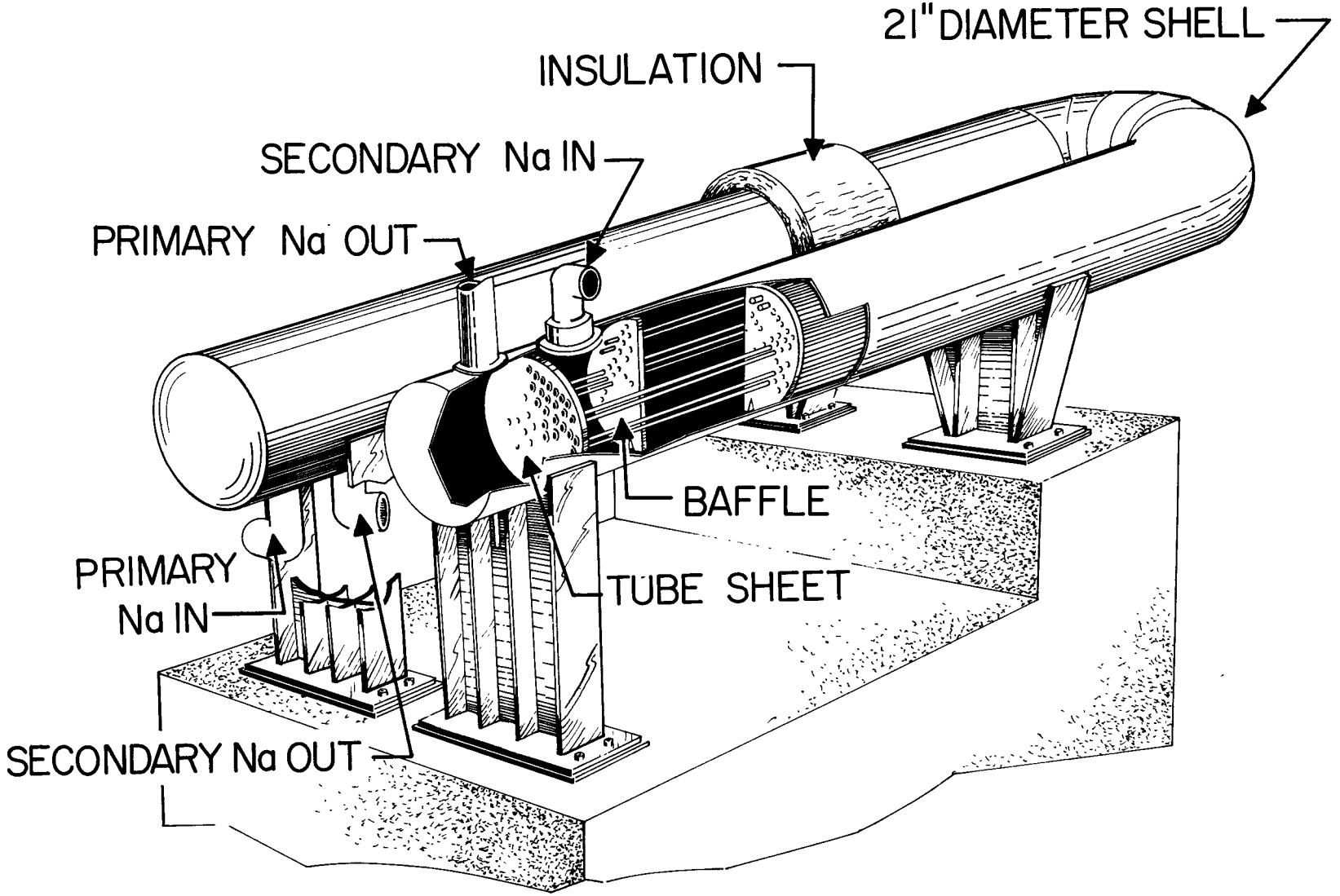


Fig. 12. Main Sodium-to-Sodium Exchanger

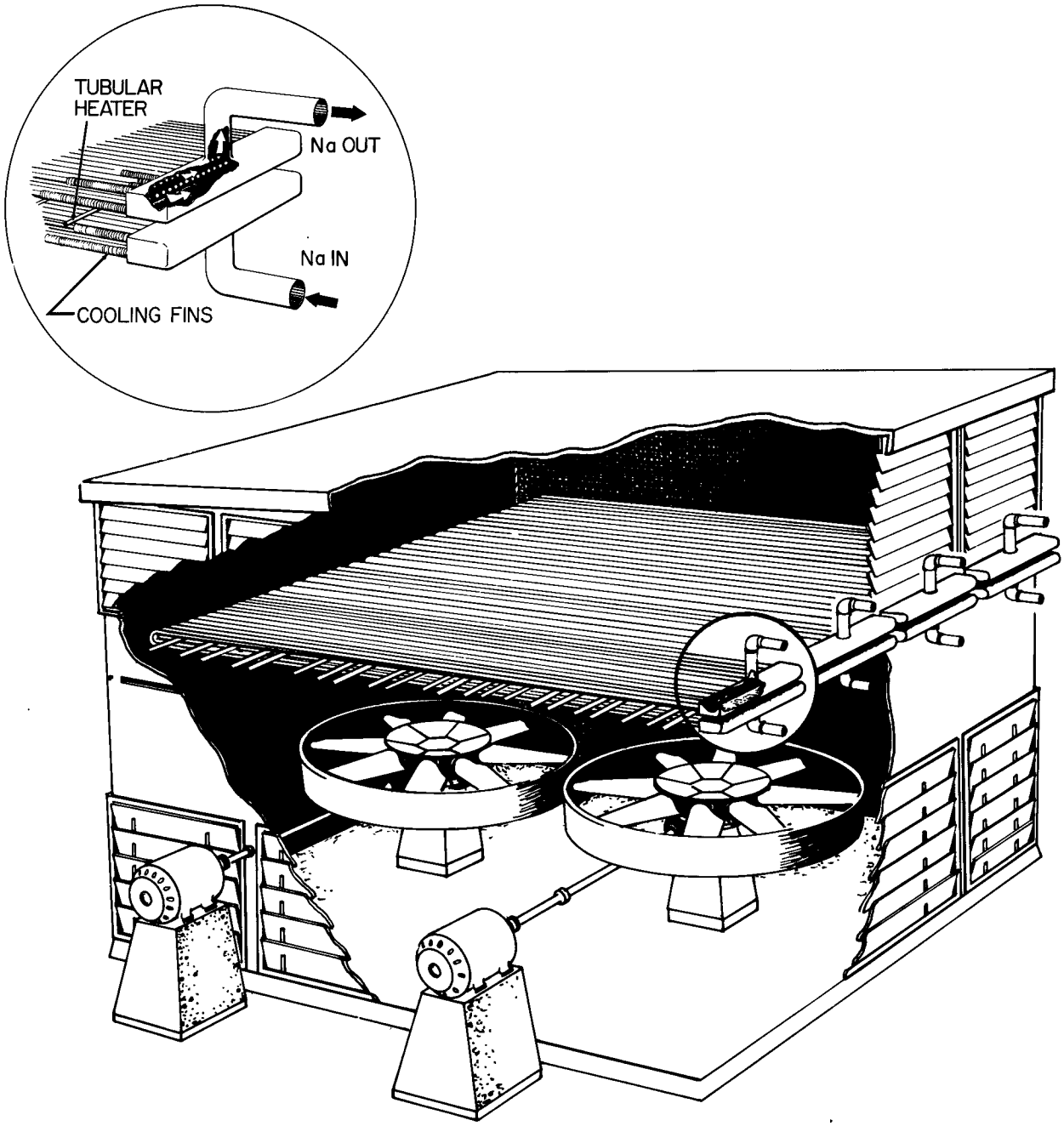


Fig. 13. Main Airblast Heat Exchanger

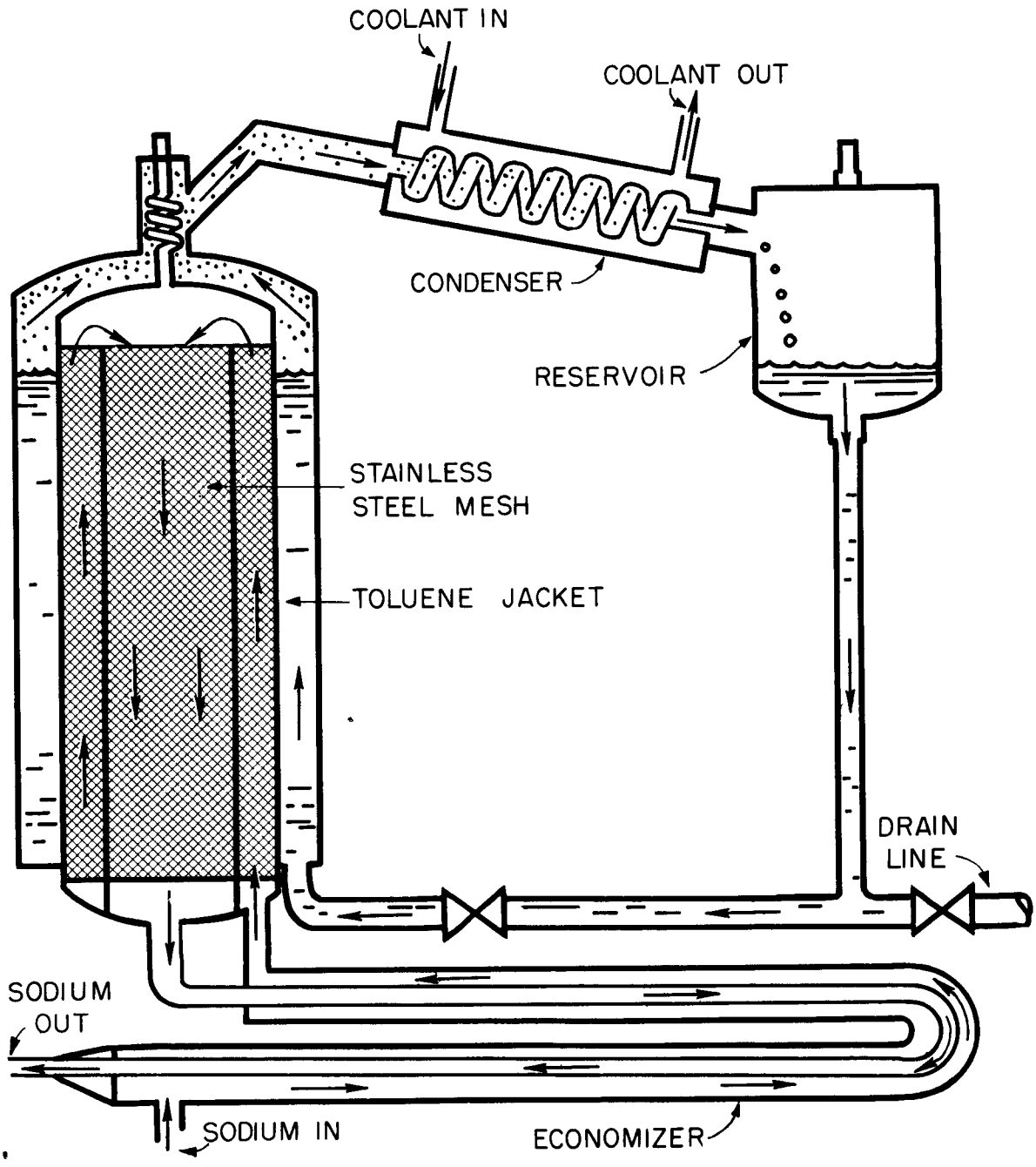


Fig. 14. Circulating Cold Trap

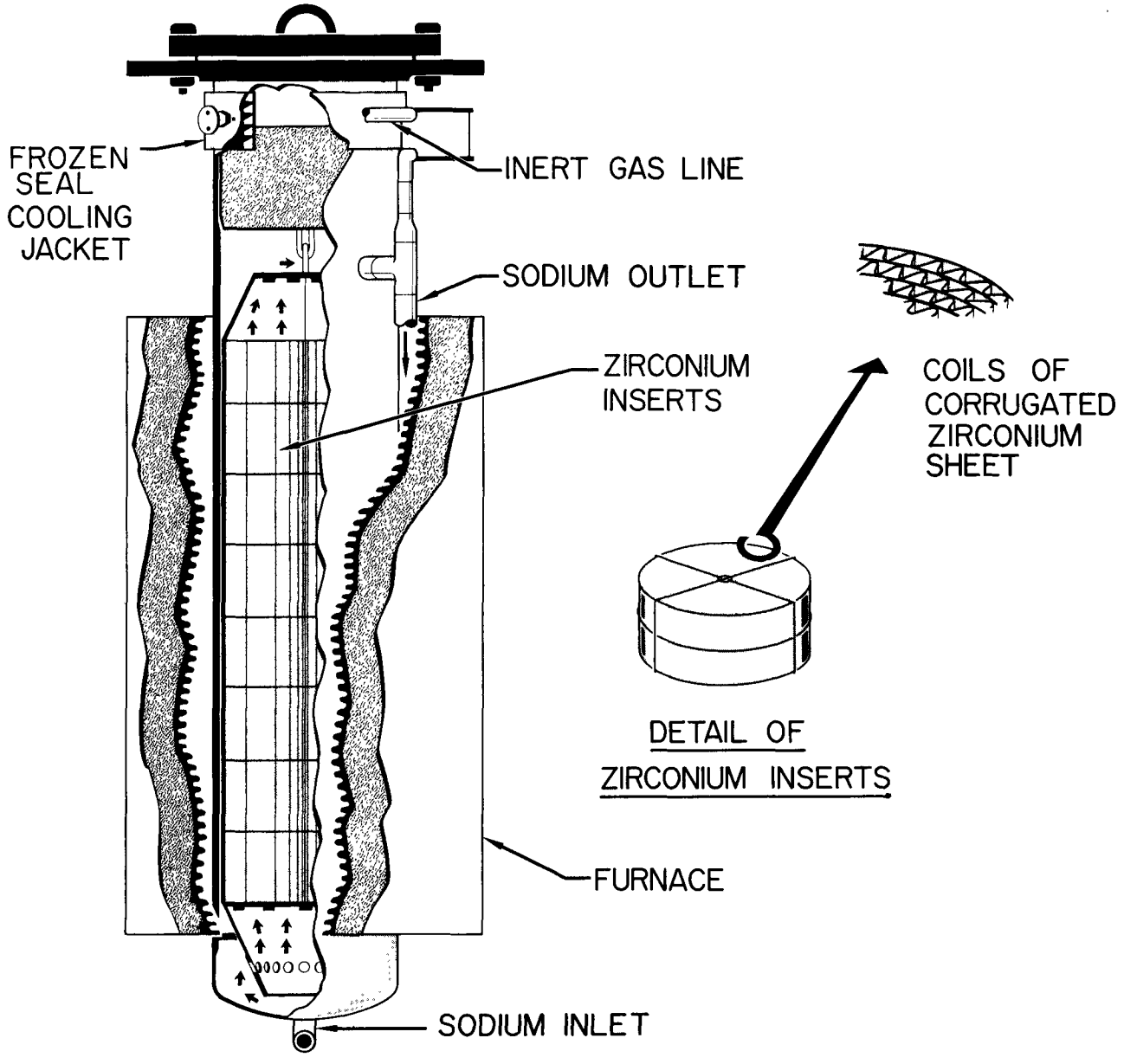


Fig. 15. Hot Trap

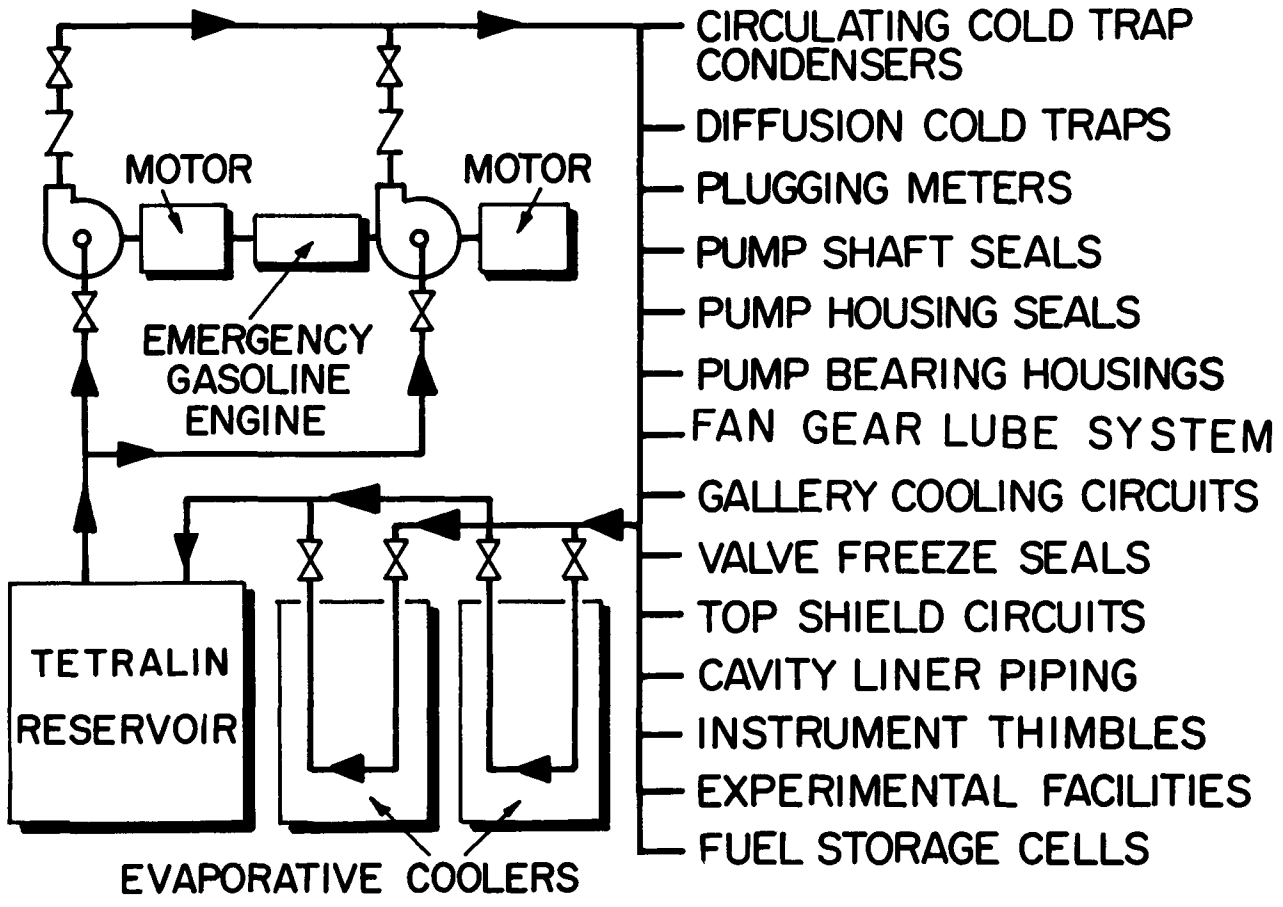


Fig. 16. Organic Cooling System

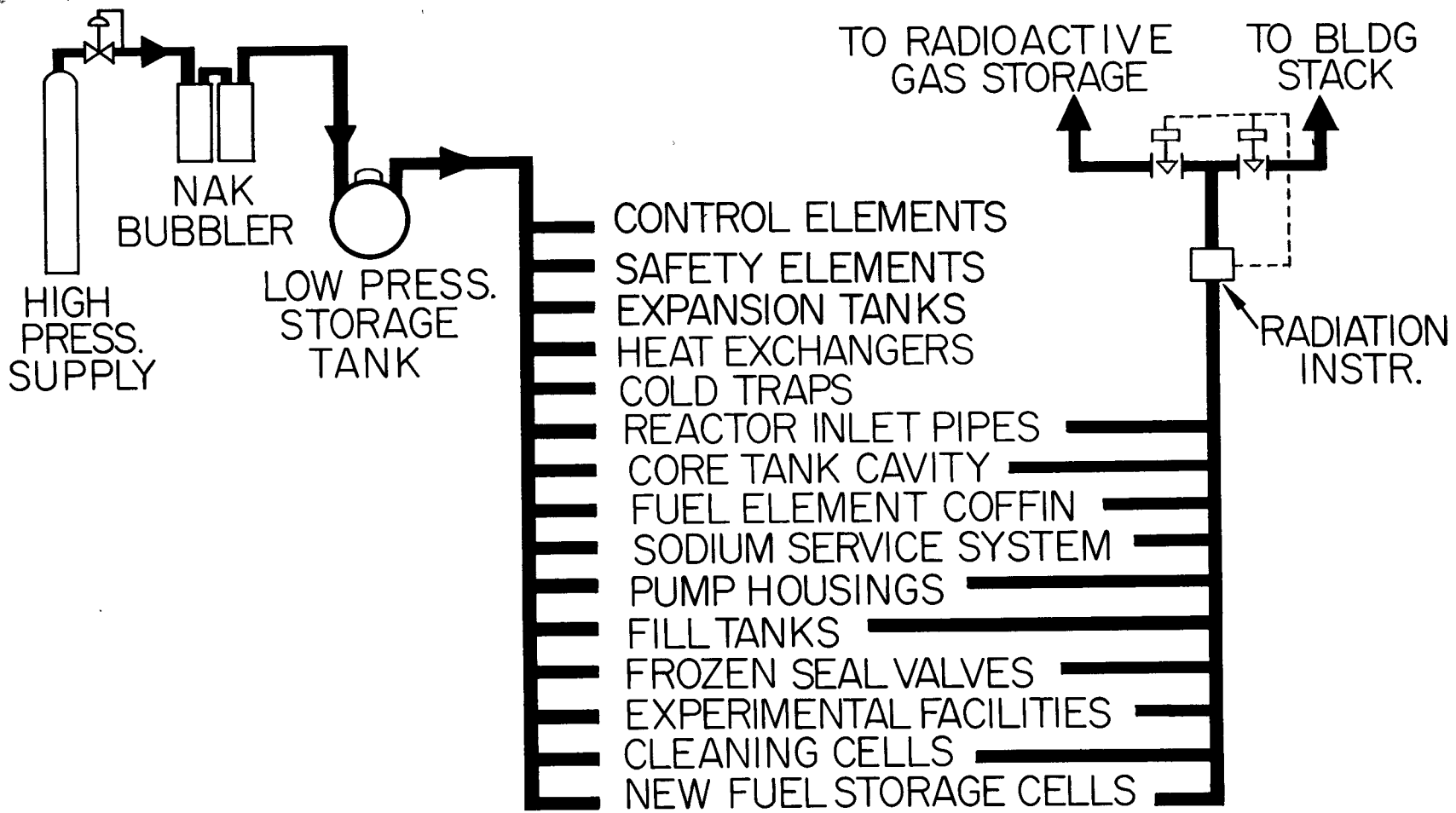


Fig. 17. Helium System

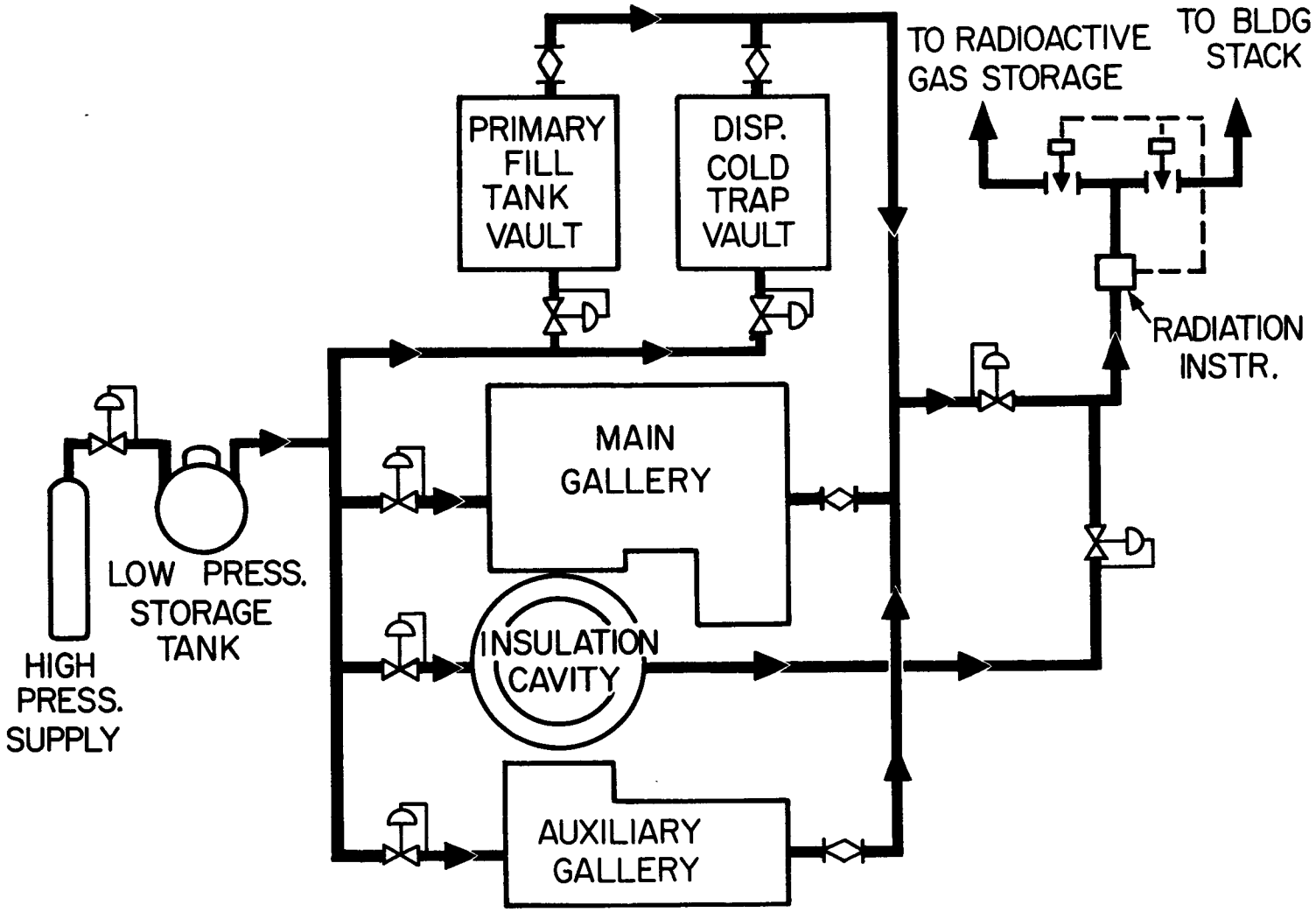


Fig. 18. Nitrogen System

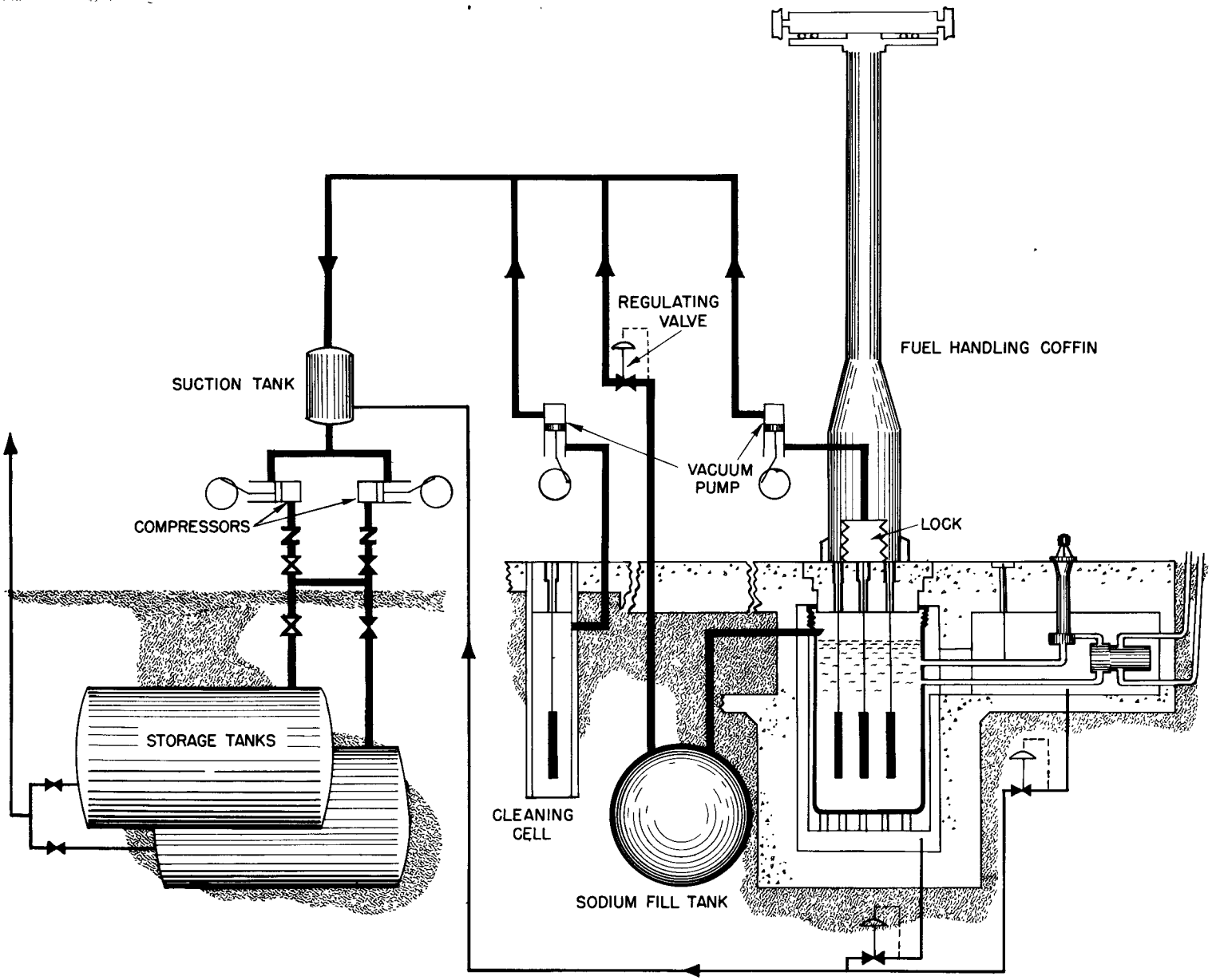


Fig. 19. Gaseous Waste Disposal System

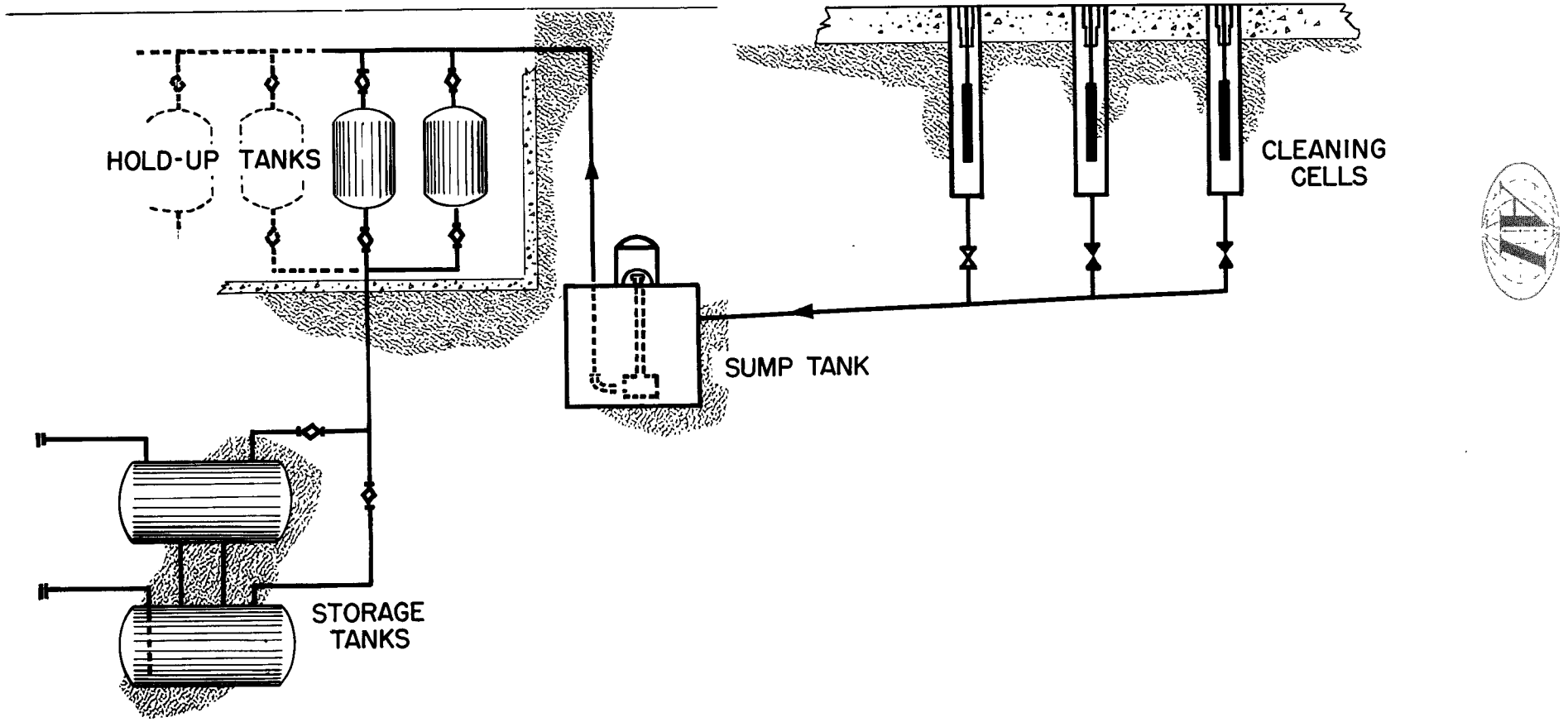


Fig. 20. Liquid Waste Disposal System

PROBLEMS OF COMPONENT MANUFACTURE

D. N. Petersen

The application of standard manufacturing methods and processes, as well as the development of new methods for the manufacture of components for the Sodium Graphite Reactor, are presented.

Problems and methods of specific component manufacture are presented. The following components in particular are discussed: core tank, loading face and ring shield, moderator and reflector cans, sodium hot trap, sodium pump systems, control and safety rod mechanisms, and fuel handling systems.

I. INTRODUCTION

The components which will be discussed from a manufacturing standpoint are: the loading face shield and ring shield, core tank control- and safety-rod thimbles, and the moderator and reflector elements - or "cans" as they are commonly called. Although these components appear to have no unusual configurations, they are very large and their dimensional relationship to each other is critical. Therefore, unique fabrication techniques were necessary to achieve the required dimensional specifications.

II. LOADING FACE SHIELD

Located above the core tank is the loading face and ring shield. The important consideration in fabricating these components was in maintaining the near-perfect alignment relationship between the core-element channel holes in the loading face and the corresponding holes in the grid plate, some 24 feet below.

The loading face shield assembly forms the lid to the core structure (Fig. 1). It is approximately 11-1/2 feet in diameter and 7 feet high. The assembly consists of a stepped stainless-steel shell, a stainless-steel top and bottom plate and an intermediate carbon-steel plate. It also contains two 40-inch and one 20-inch removable plugs as well as 81 core-element channel plugs and a maze of cooling coils imbedded in lead between the intermediate and bottom plates.

In accomplishing this fabricating task, it was necessary that the top, bottom and intermediate plates by machined flat and bored concentric to each other (Fig. 2). Boring was accomplished through the three plates by match machining on a horizontal boring mill.

It was then necessary to fabricate the cooling coils to be installed on the lower side of the intermediate plates. In doing this, templates were used. This was done by forming the tubes over the templates, as shown in Fig. 3.

These tubes were then buried in one half inches of lead on the lower side of the intermediate plate. Figure 4 shows the lead being installed by utilizing the technique of lead burning.

Since the plate was to be subjected to heat in the lead burning process, it was necessary to fabricate this special table and weld the intermediate plate to the table in order to prevent the plate from warping during the lead burning operation.

After burning, the lead was finish machined to ensure thermal contact with the bottom plate.

The walls of the loading face were then machined (Fig. 5). The seats for the top, intermediate and bottom plates were machined at right angles to the centerline and then the plates were installed in the shell. The bottom plate was welded in position and the intermediate plate containing the lead was bolted and doweled to it. The top plate was then located optically within a tolerance of 0.001 inch and doweled. The top plate was then removed and all of the core element casings installed.

Figure 6 illustrates the installation of the core element casings. It emphasizes the complexity of the inner portions of the loading face. The top plates was then placed back into position and the casings were welded in place.

Optical checks on the alignment were again made and the plug assembly was finish machined.

Figure 7 shows the loading face shield being set up for finish machining on a vertical boring mill. All bores were held parallel to the centerline and squared with the top of the shield and seats. Maintenance of close tolerances and strict welding specifications on this large and complex assembly were our prime considerations.

Figure 8 shows the completed assembly of the loading face loaded for shipment to the job site.

Figure 9 illustrates the two 40-inch plugs and the one 20-inch plug that fit into the loading face. These plugs were fabricated in much the same manner as the loading face itself.

The ring shield is a stepped cylindrical assembly, approximately 16 feet in the outer diameter and 6 feet high. Figure 10 shows the assembly just prior to finish machining on a vertical boring mill. The encasement material is Type 405 stainless steel. Welding was again a primary consideration and was accomplished

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according to the same specifications and following the same methods used in fabricating the loading face shield. At the completion of welding, all critical surfaces of this ring shield were machined to dimensional tolerances considered extremely close for an assembly of this size.

III. CORE TANK

The core tank (Fig. 11) was fabricated from Type 304 stainless steel plate, 1-1/2 inches thick. It is 11 feet in diameter and approximately 19 feet in length. It weighs approximately 26 tons.

The tank was fabricated in three separate circular sections and a bottom section. These pads were welded on so that the bottom could be machined, giving a true horizontal surface for the tank to set on. Shaping, forming, and assembly of the tank did not present any unique fabrication problems. However, the tolerances are much closer than normally required in standard tank fabrication. Arc-welding methods were employed, using lime- and titanium-coated electrodes. The tank was welded in a horizontal position in order to employ down hand welding techniques. As in the case of the loading face and ring shield, back-chipping and grinding was accomplished after each root pass and dye-penetrant inspection was performed. Every inch of the final welds was checked by radiograph inspection and a final freon leak test was made on the completed vessel.

IV. CONTROL AND SAFETY ROD THIMBLES

The importance of the control and safety element operations is recognized. However, as the size and depth of the core tank increases, so do the problems of manufacturing. This was manifested in the construction of the control rod thimbles (see Fig. 12). These thimbles, which are over 280 inches in length, necessitated machining and welding of varying diameters and wall thicknesses. The outer stainless-steel tubing was machined and welded in sections. Each weld had to be radiographically sound, and the inside diameters had to be straight and concentric. Shrinkage and warpage had to be overcome. Careful attention and the utilization of special fixtures during the welding of the various sections enabled us to accomplish an assembly within the required tolerances without additional machining operations after welding.

V. MODERATOR AND REFLECTOR CANS

The most unusual set of problems we encountered were those associated with fabrication of the components made of zirconium for the moderator and reflector cans. A typical assembly consists of a graphite stringer, 0.035 sheet zirconium for the hexagonal tube or "can", zirconium plate (0.100) for the upper and lower heads, 2.875-inch-wall zirconium tube for the fuel channel tube, and 1/2-inch outside diameter by 0.035-inch-wall zirconium tube for the pump-out tube used to charge the assembly with pressurized helium. Without the vent and guard tubes, the overall height of an assembly is approximately 120 inches.

The outside width across the flats (Fig. 13) is approximately 11 inches ±0.025. The zirconium in each assembly weighs 40 pounds, and, as described in the film, the completed assembly weighs 760 pounds.

At the time the sodium reactor experiment was initiated and the pilot run of cans was being made, very little was known of the behavior of zirconium. The need for this information sparked a vast production development program which began with the processing of ingots of zirconium at the mill.

Early in our program we experienced difficulties with the production of mill products, as did other users of zirconium and its alloys. A process was finally developed which our experience shows will produce the largest yield of good quality products. Our experience emphasizes that double arc melting under vacuum at both stages produces soft ingots capable of rendering higher yields in fabrication. Careful preparation of the material at all stages is necessary to achieve satisfactory product yield. The basic sequence and preparations at each stage are shown in this figure.

Six panels make up the hexagonal tube forming the outer shell of the can (Fig. 14). These panels were formed by blanking 0.035 stock strips of zirconium, which had been ultrasonic tested for defects, into strips 6-1/2 inches wide by 124 inches long.

The panels were formed with profile dies on a power brake at room temperature (see Fig. 15). After forming, they were clamped in a special stress-relieving fixture and then stress relieved at 1040° F with a 15 minute soaking period. After stress relieving, the parts were placed on a planer.

The longitudinal edges of the panels were dry-machined to the desired dimension with an alloy steel cutting tool that had a 3° lip angle (Fig. 16). A cutting speed of 40 feet per minute was used. After planing, the panels were ready for assembly into the hex shape of the can.

A welding machine and fixture were used to weld the panels into the hexagonal shape (Fig. 17). To achieve technical requirements, it was necessary to shield both sides of the surface being welded with an inert gas such as argon or helium. This machine automatically accomplished this with gas jets which automatically shielded all surfaces being welded with an inert gas. It was also necessary to utilize adequate chill blocks to dissipate the welding heat as rapidly as possible. Prior to welding, the edges were cleaned by light buffing with emery cloth and then wiped with acetone. The panels were placed on the machine and positioned to the stake with a locating gauge. The ten-foot butt-welds were made by a power-driven automatic welding head, moving at four inches per minute, while the stake locates and supports the panels. All welds were made without filler material. The welder employed a tungsten electrode, 3/32 inch in diameter, tapered to a sharp point. Close inspection was required at all stages of cleaning, fit-up tacking, and welding of the components. The initial loading of zirconium cans now in the reactor represents approximately 4200 pounds of zirconium and two miles of zirconium welding.

The machining and assembly of the graphite stringers presented no major fabrication problems (Fig. 18). As described in the film, all work was accomplished in an area where precautions are taken to avoid all forms of contamination. Even the cleanliness of the workmen's clothing is considered, by requiring them to be laundered in boron-free soap. The preservation of the graphite during the storage period, prior to assembly operations, was accomplished by employing an aluminum foil wrapper, which has proved to be an excellent moisture-gas-proof barrier.

The graphite stringers were assembled, pinned and moved horizontally into the hexagonal can shell (Fig. 19). This was accomplished by a loading fixture which we developed specifically for this purpose. The final assembly operation comprised insertion of the fuel channel tube and can heads (see Fig. 20). These units were manually welded in place, using an inert-gas-shielded tungsten arc. Although there is a considerable difference between the basic thickness of the can

panels and head, the welding closures were made without the use of filler material. All welds are inspected by dye penetrant and X-ray.

Upon completion of welding, the cans were moved to a pump-down area where they were evacuated and back-filled with helium, and then leak tested with a helium mass spectrometer (see Fig. 21). After inspection, the stainless-steel pedestal and spacer casings were fitted into position on the can heads, using alignment fixtures to ensure concentricity. The stud nuts were firmly secured and a lock wire was employed.

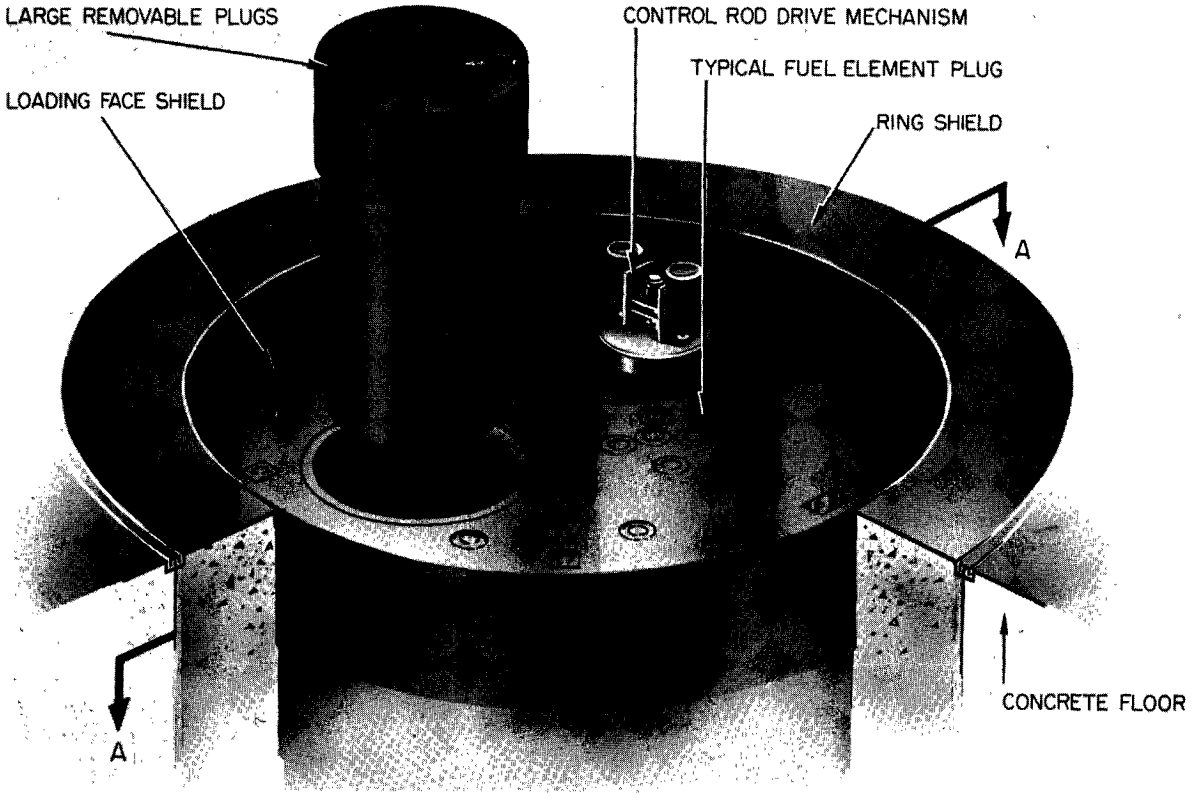
The fabrication process from zirconium ingot to the completed assemblies consumed approximately two years of production development and fabrication time. With our present methods and experience we are capable of assembling eight cans per shift.

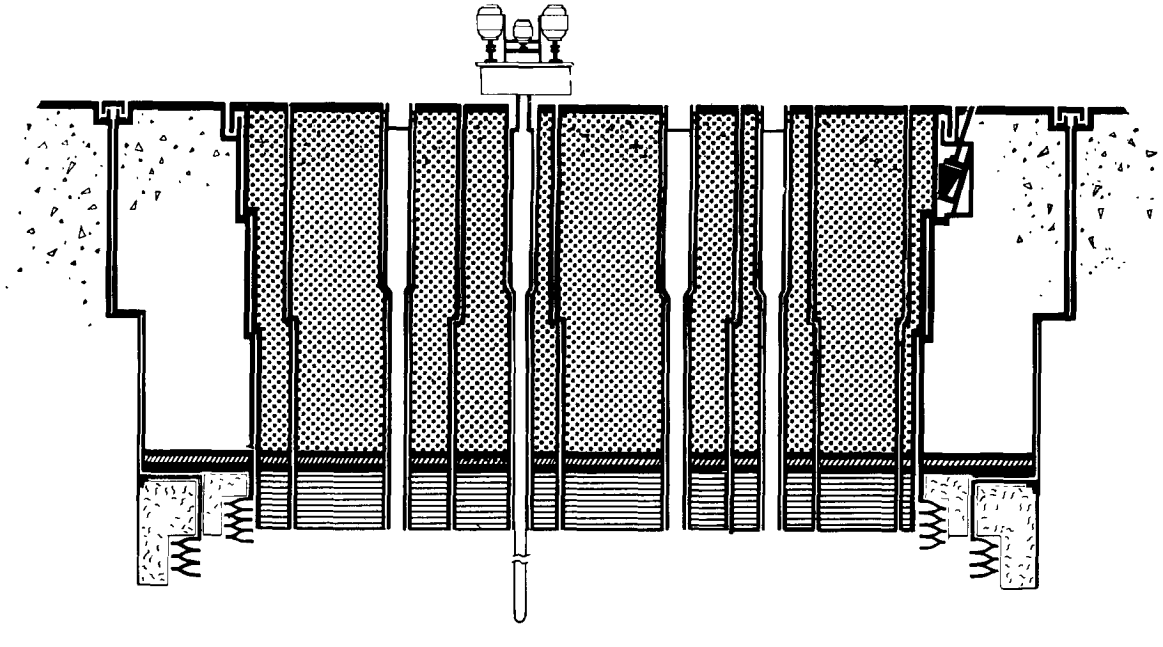
Completing the assemblies is not the end of our problems. There remains the task of transporting these very fragile, 760-pound odd-shaped assemblies to the remote location of the job site. Impact-o-graph analyses of road conditions leading to the site were made.

As a result of these studies, special boxes were designed to minimize road shocks and shocks encountered in normal handling. The boxes were designed so as to act as part of a loading fixture which inserts the cans in the core of the reactor, thereby ensuring safe delivery and reducing and simplifying handling at the job site.

VI. REMARKS

The manufacture and construction of reactors, unlike most products, varies greatly with the size and configuration resulting from each design concept. Reactors vary in size from large power-plant-types, involving precision machining of large vessels and associated assemblies, to small research reactors whose manufacturing problems are reduced only by the decrease in size. As a result of this range of size from small to very large components, manufacturing facilities encompass very small, high-precision machine tools to heavy-duty machine-shop equipment, and special fabrication techniques in the fields of machining, welding, assembly, and testing are continuously being developed to meet the more rigid specifications to which we must work.





SECTION A-A

Fig. 1. Loading Face Shield

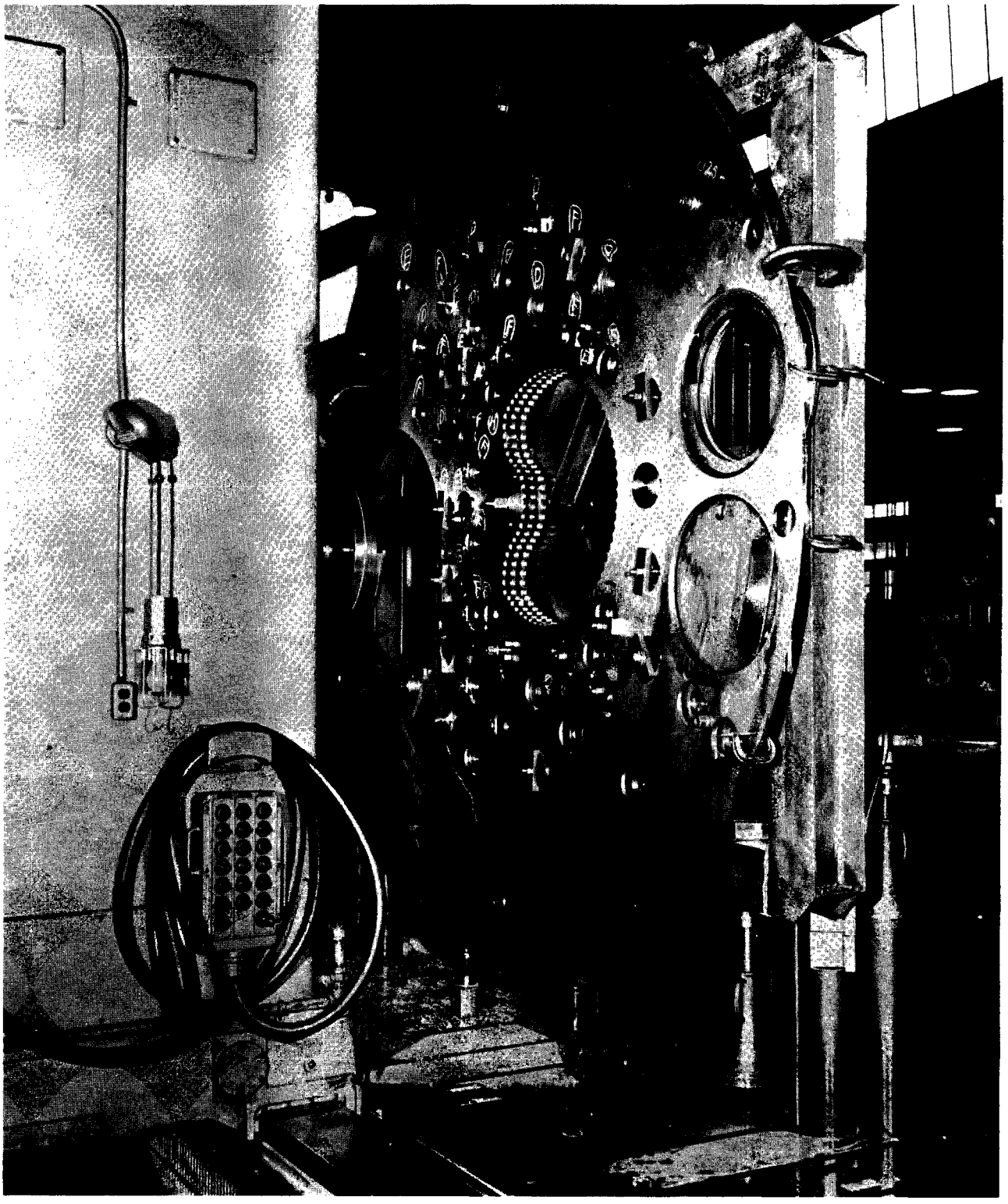


Fig. 2. Machining and Boring of Plates

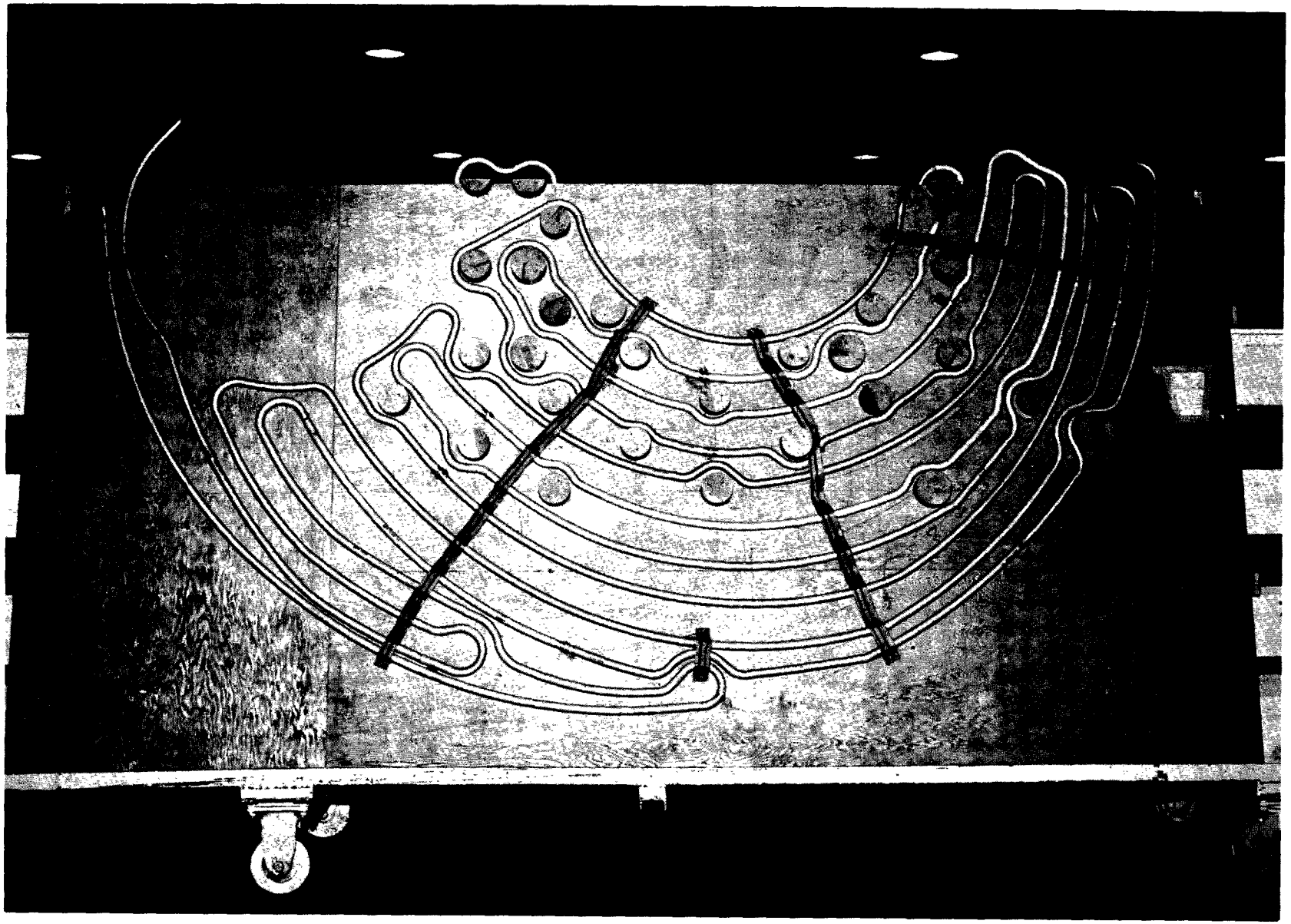


Fig. 3. Fabrication of Cooling Coils

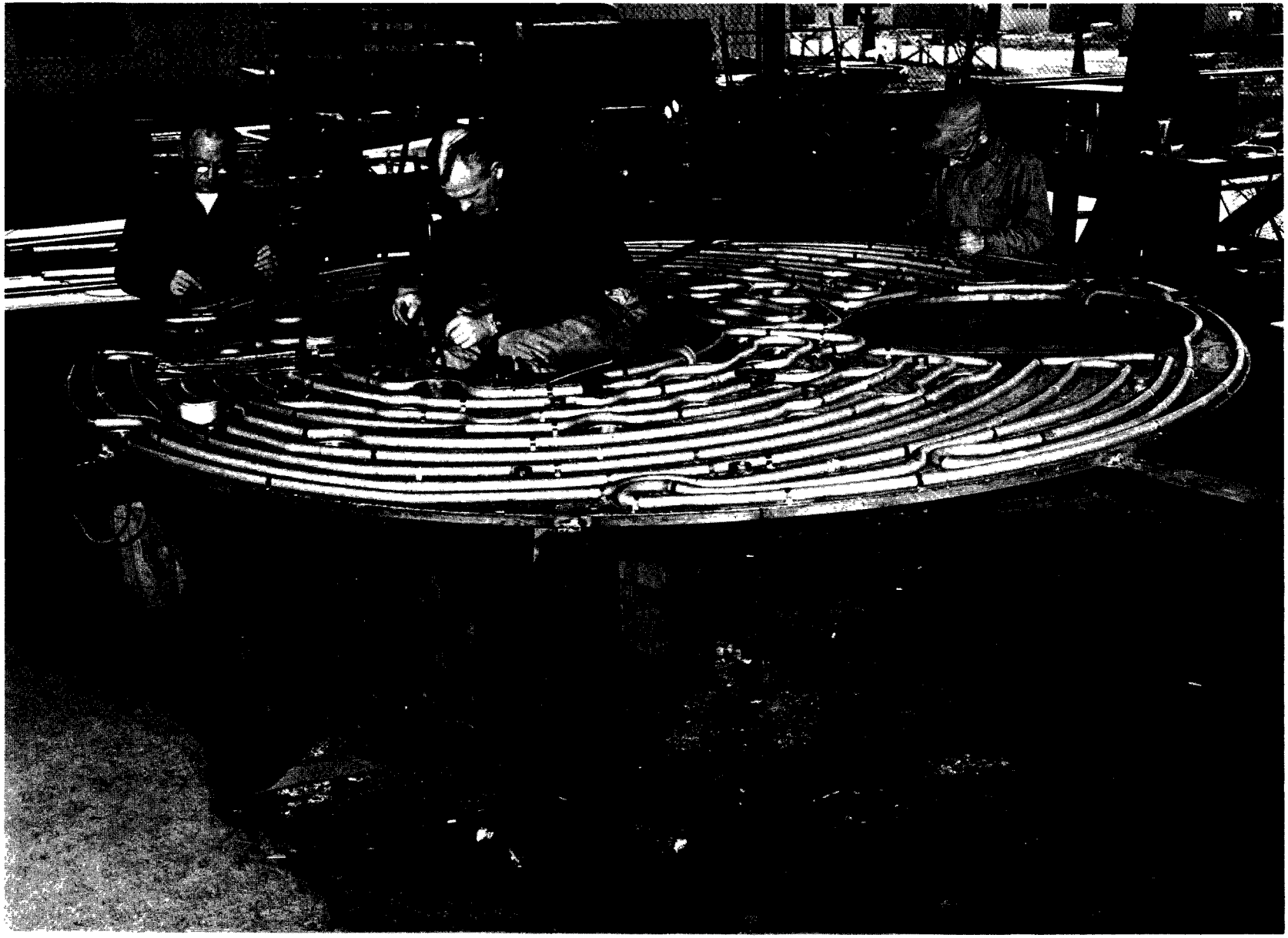


Fig. 4. Installation of Lead on Tubes

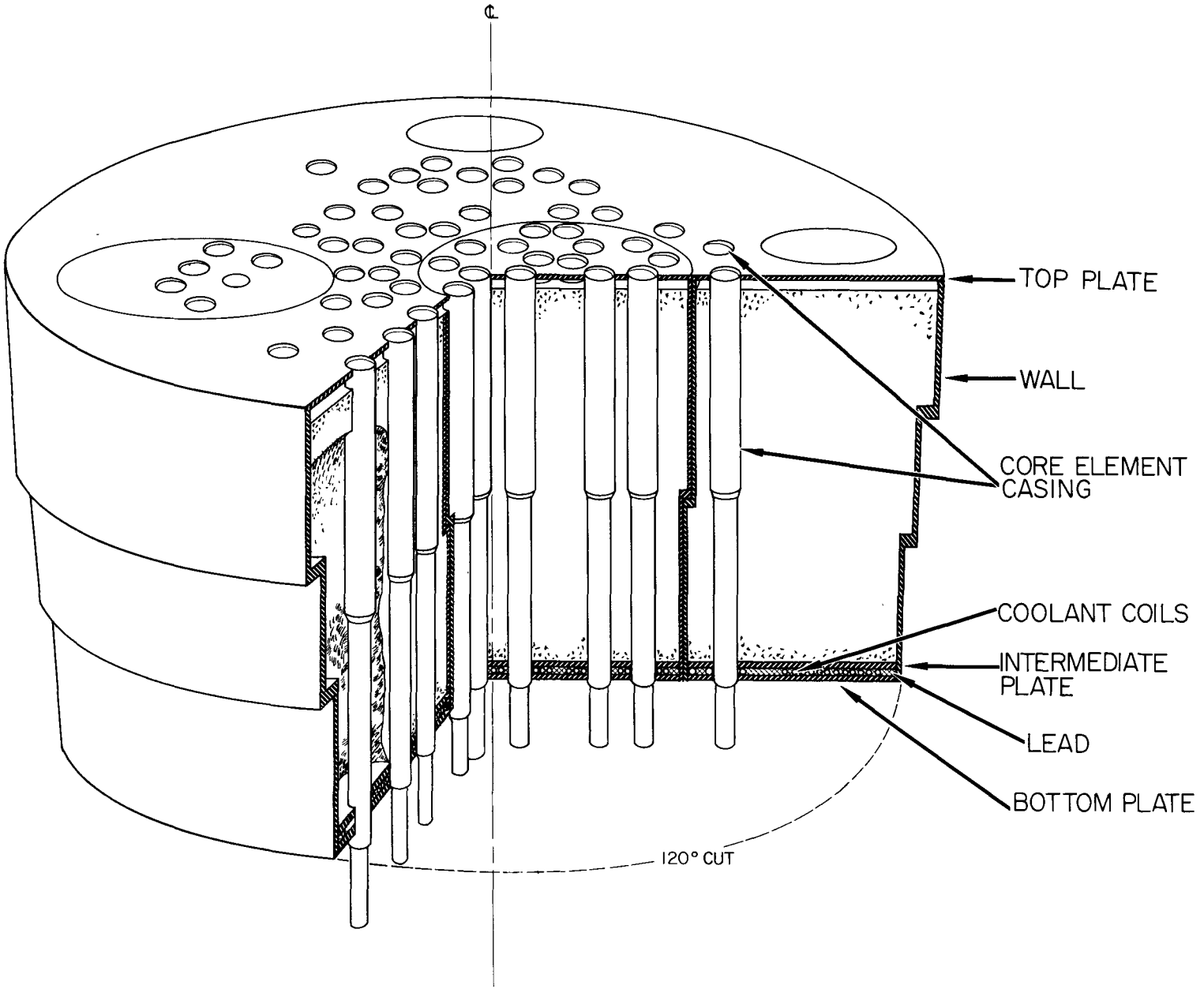


Fig. 5. Sectional View of Loading Face Shield

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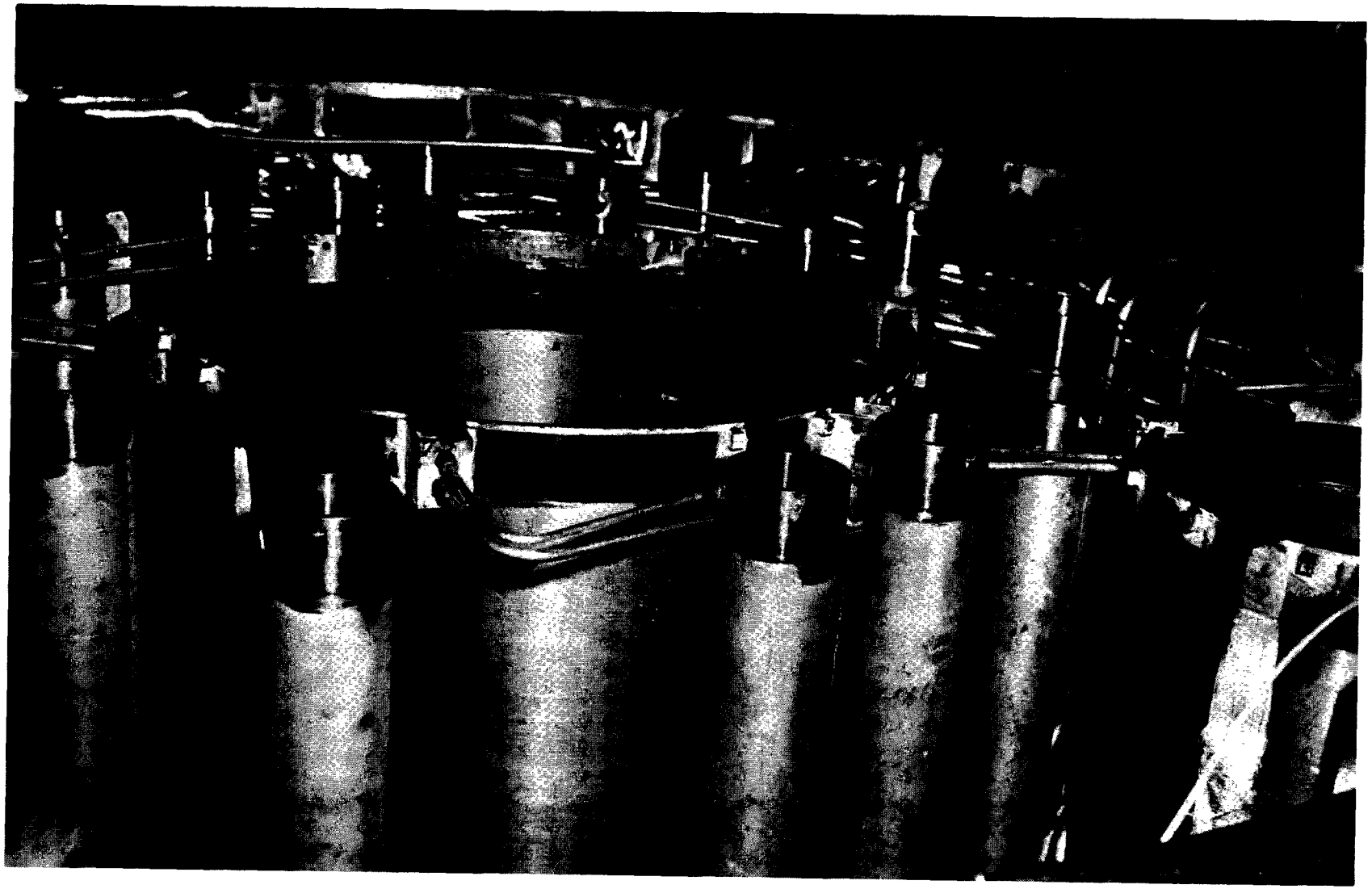

Fig. 6. Installation of Core Element Casings

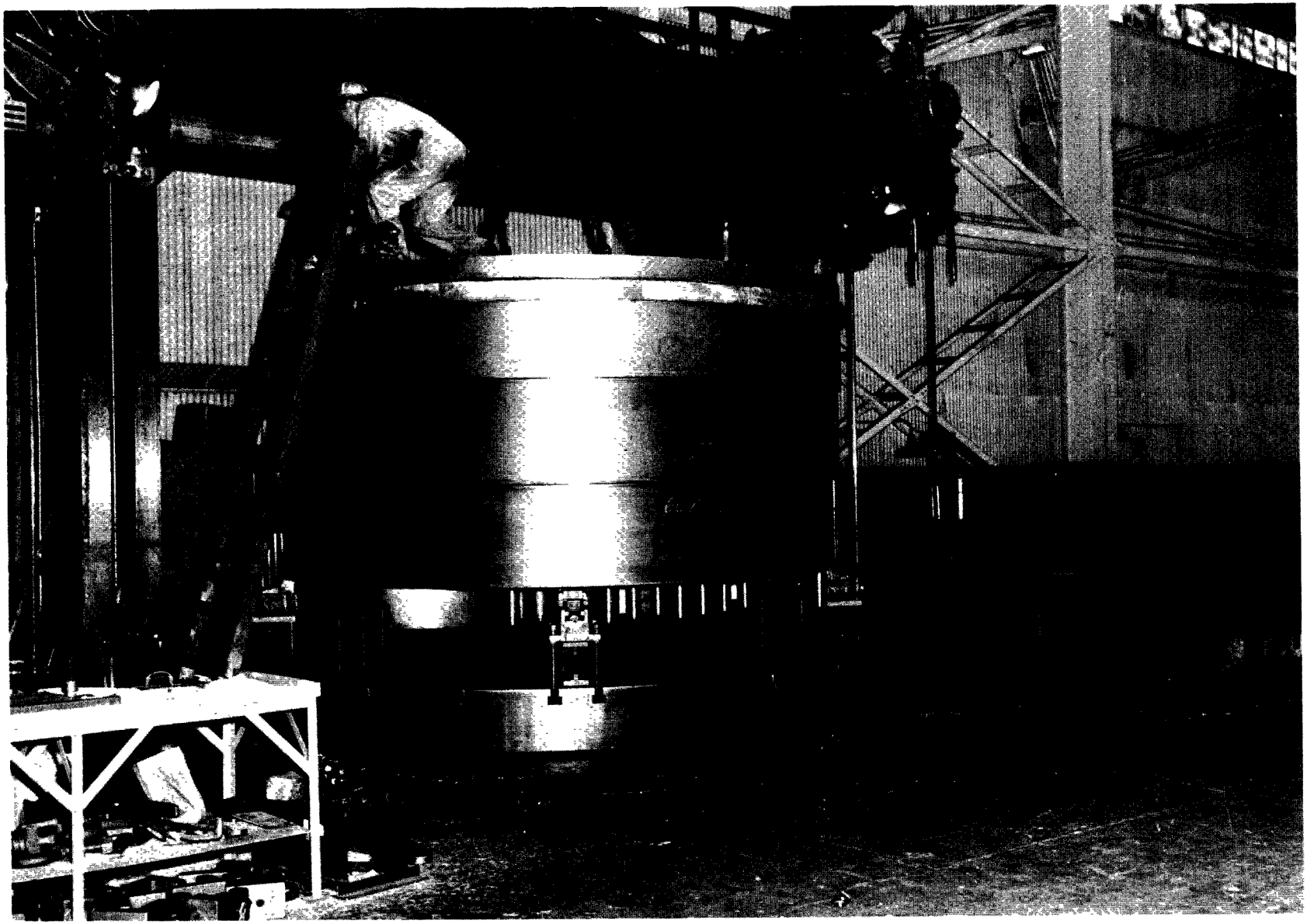

Fig. 7. Setup for Finish Machining

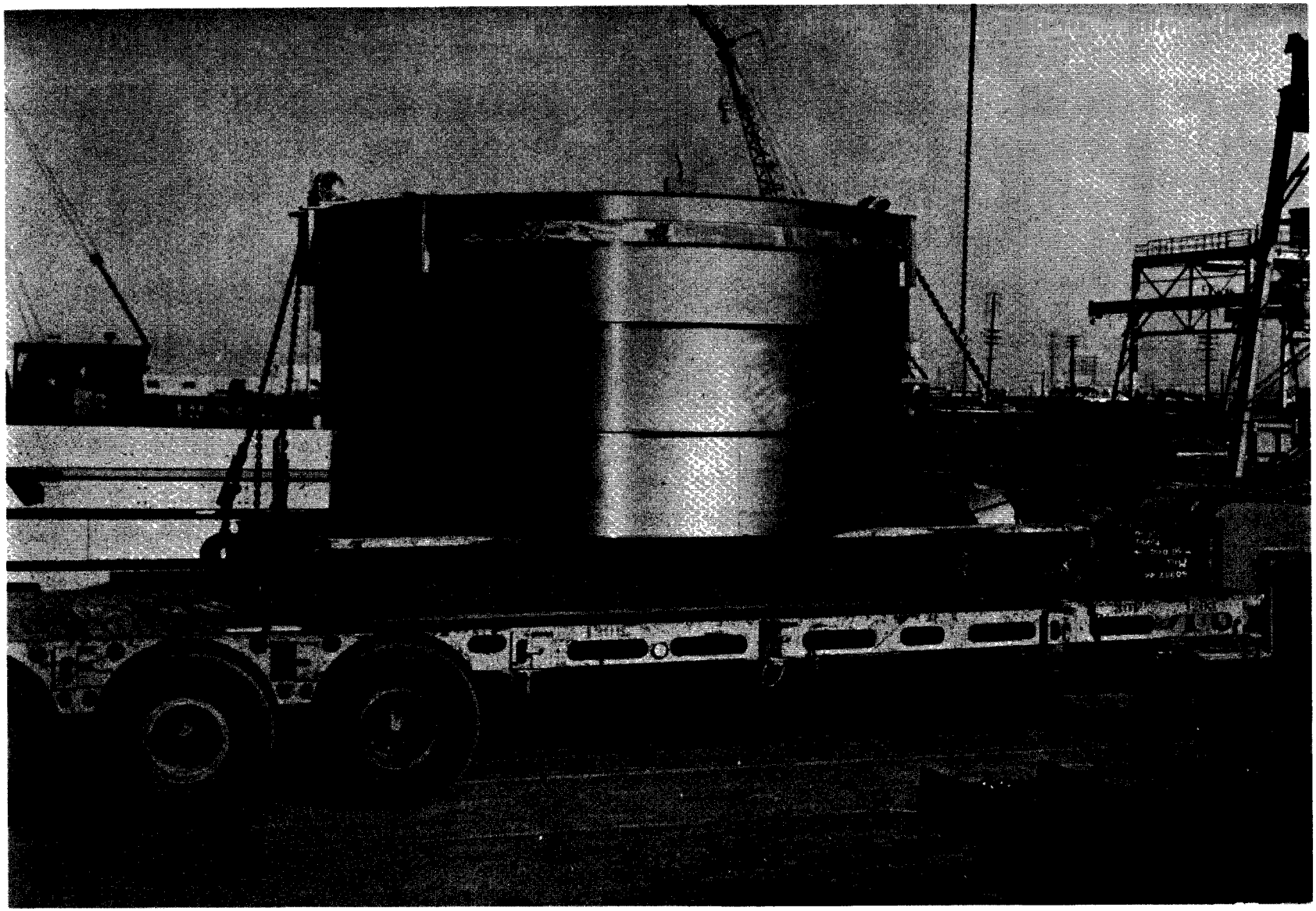

Fig. 8. Loading Face Shield Loaded for Shipment

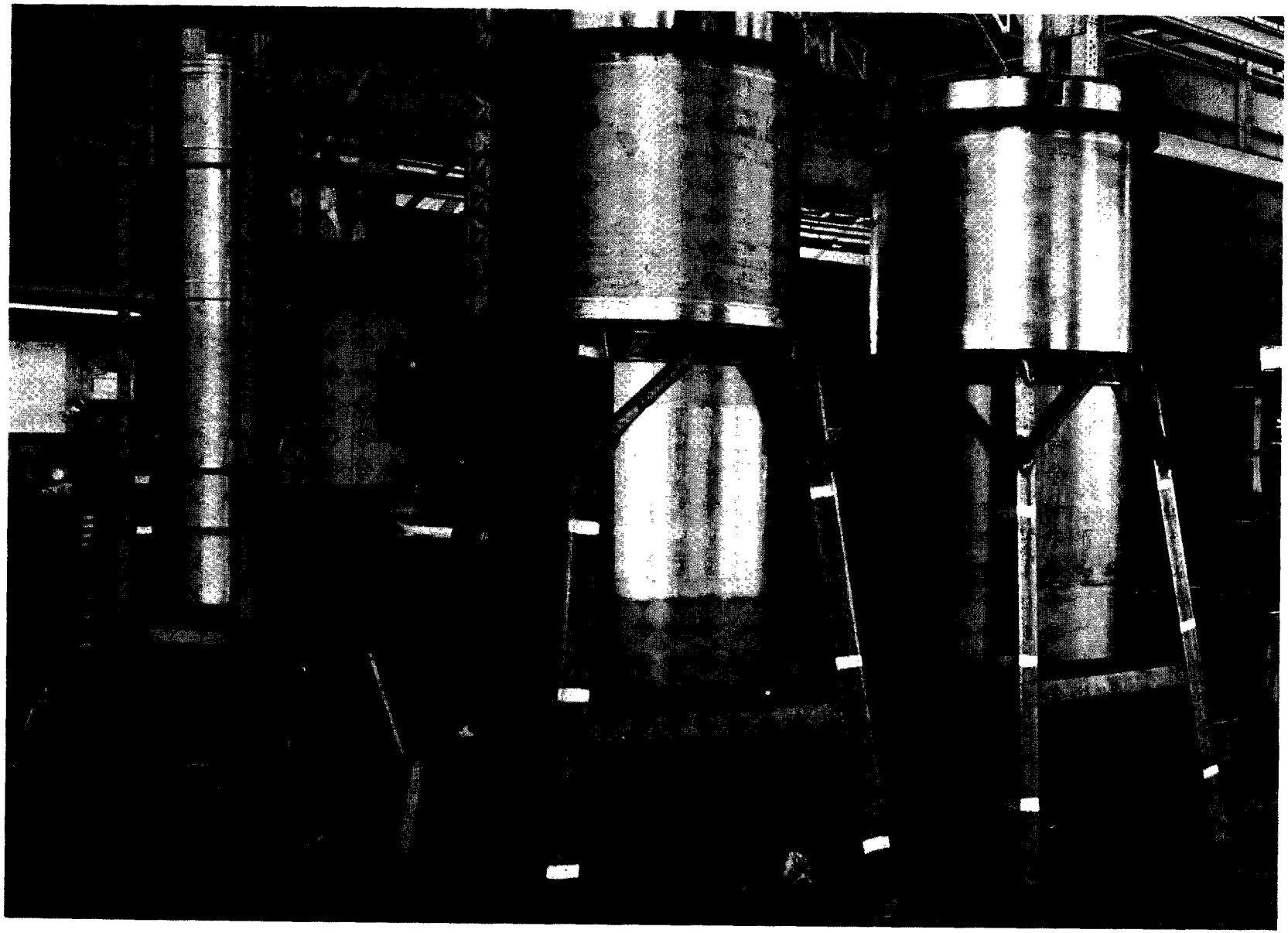

Fig. 9. Loading Face Plugs

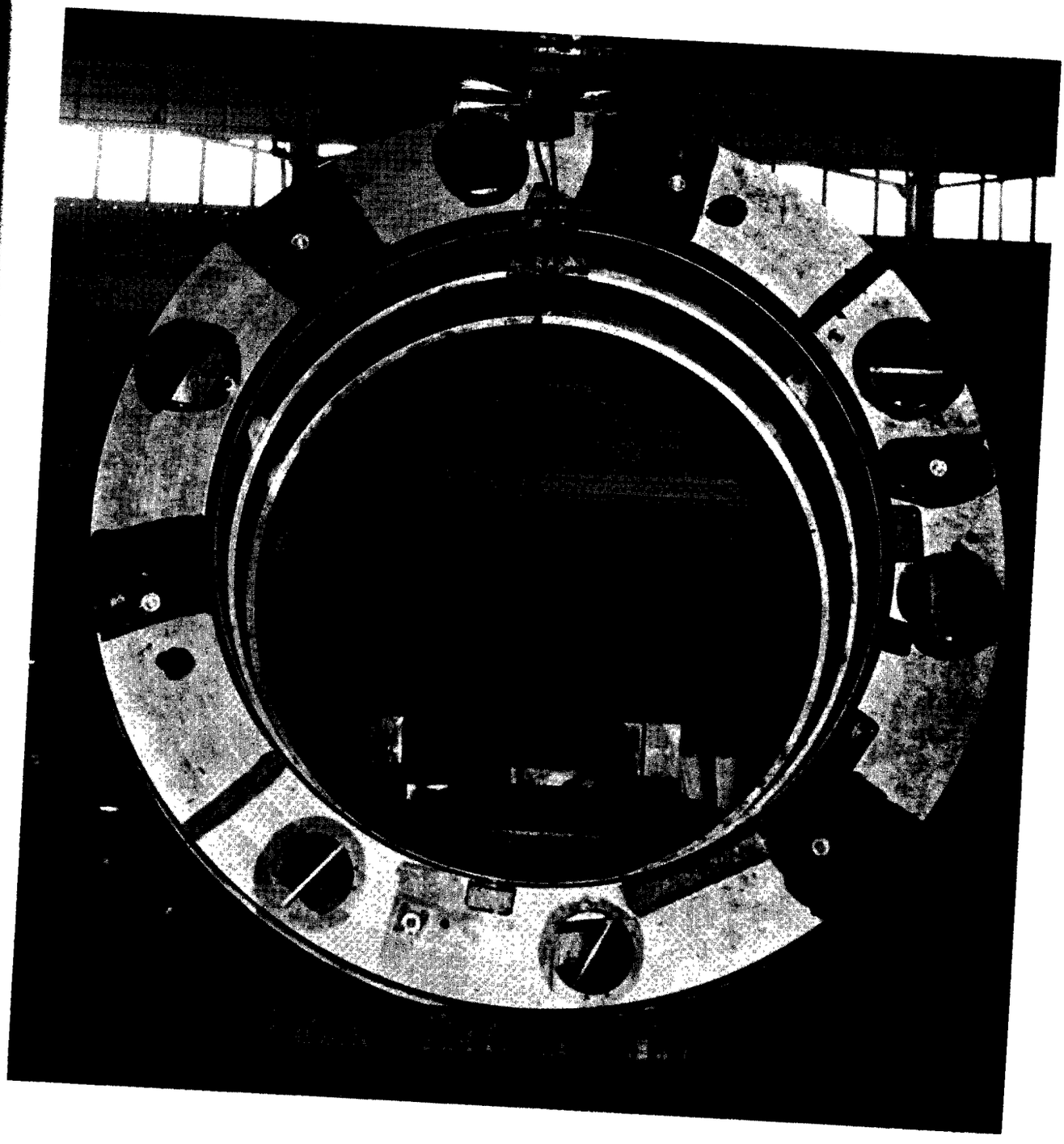

Fig. 9. Loading Face Plugs

Fig. 10. Ring Shield Assembly

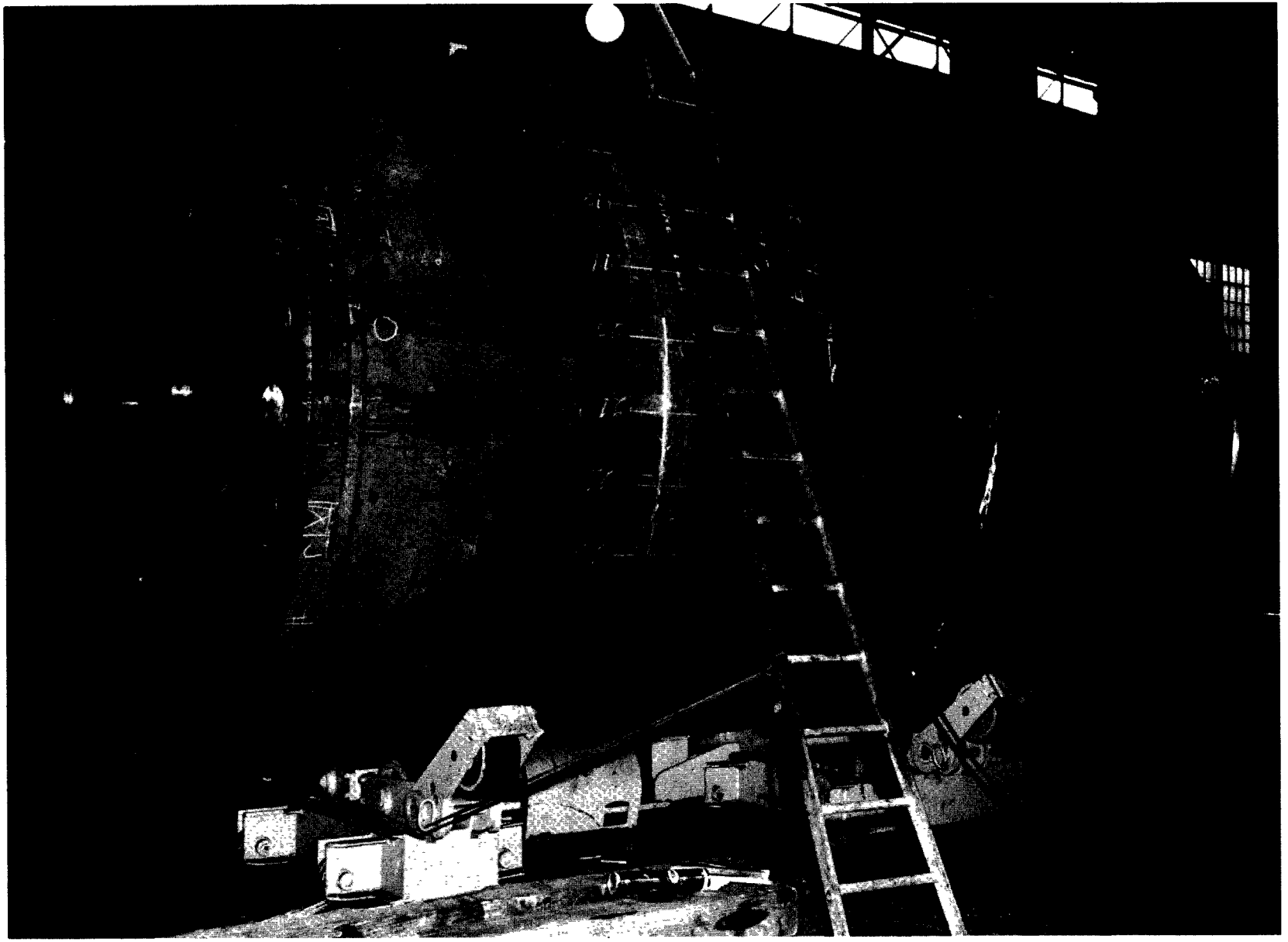

Fig. 11. Core Tank

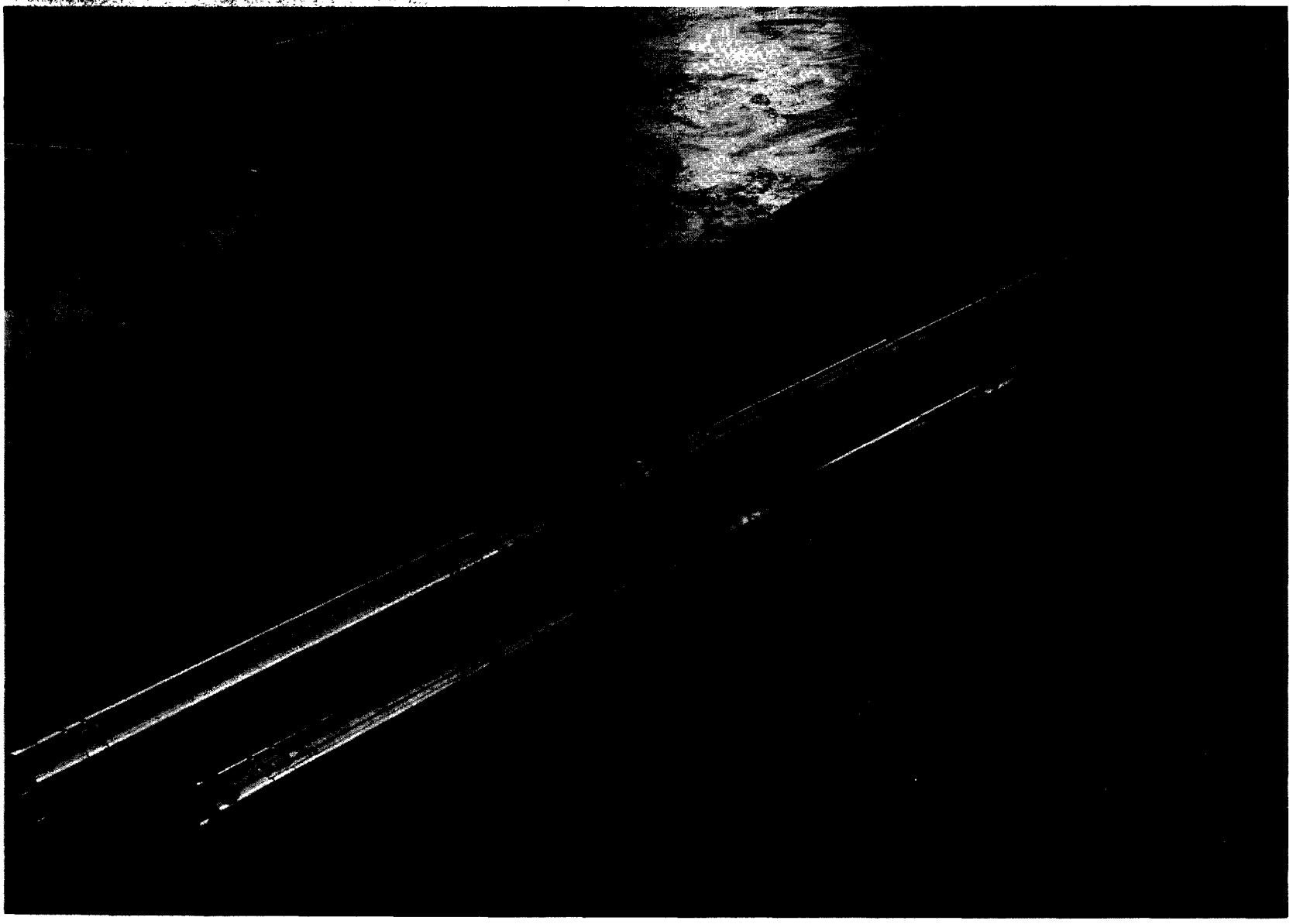

Fig. 12. Control Rod Thimbles

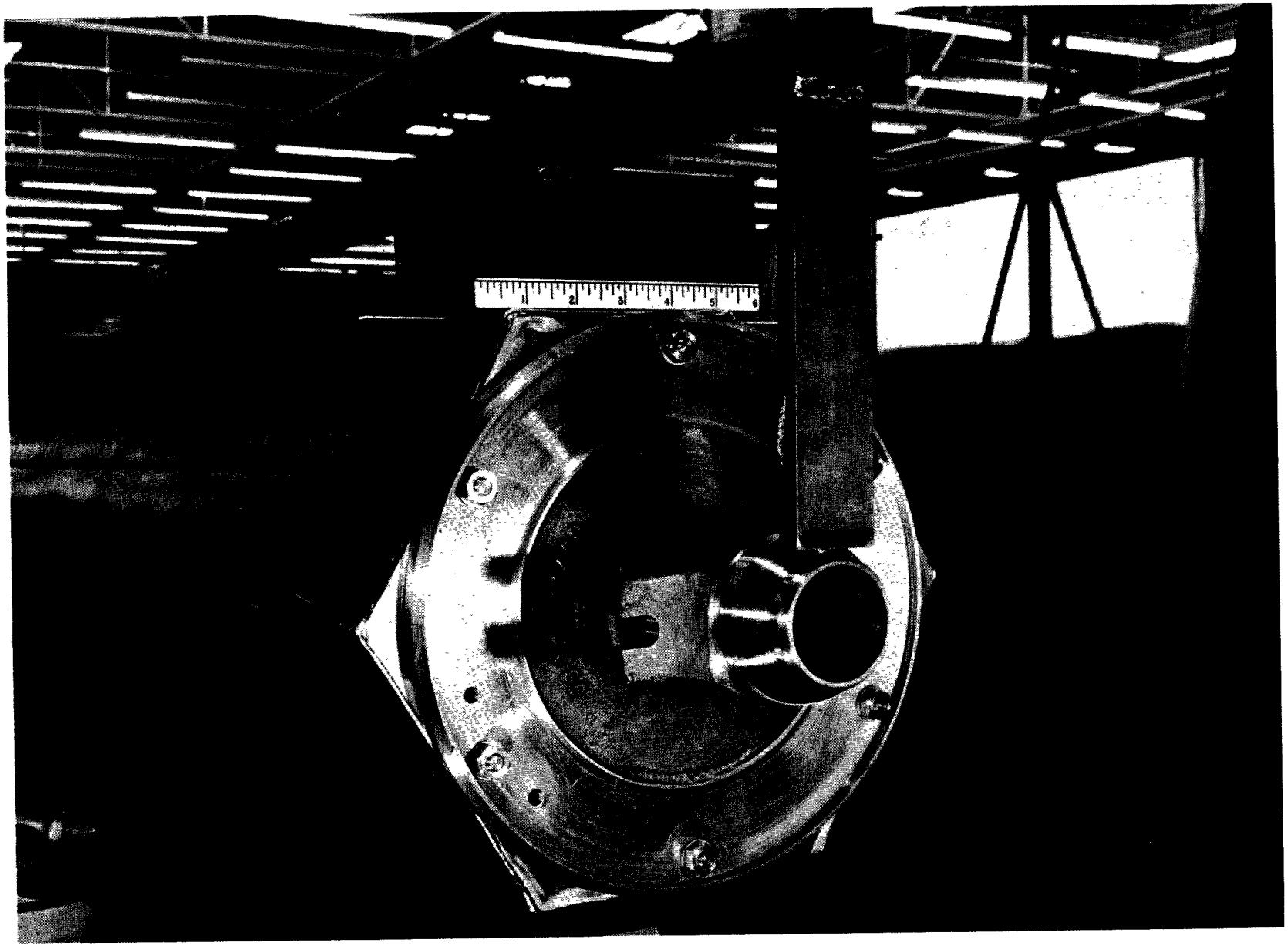

Fig. 13. Measurement of Zirconium Can

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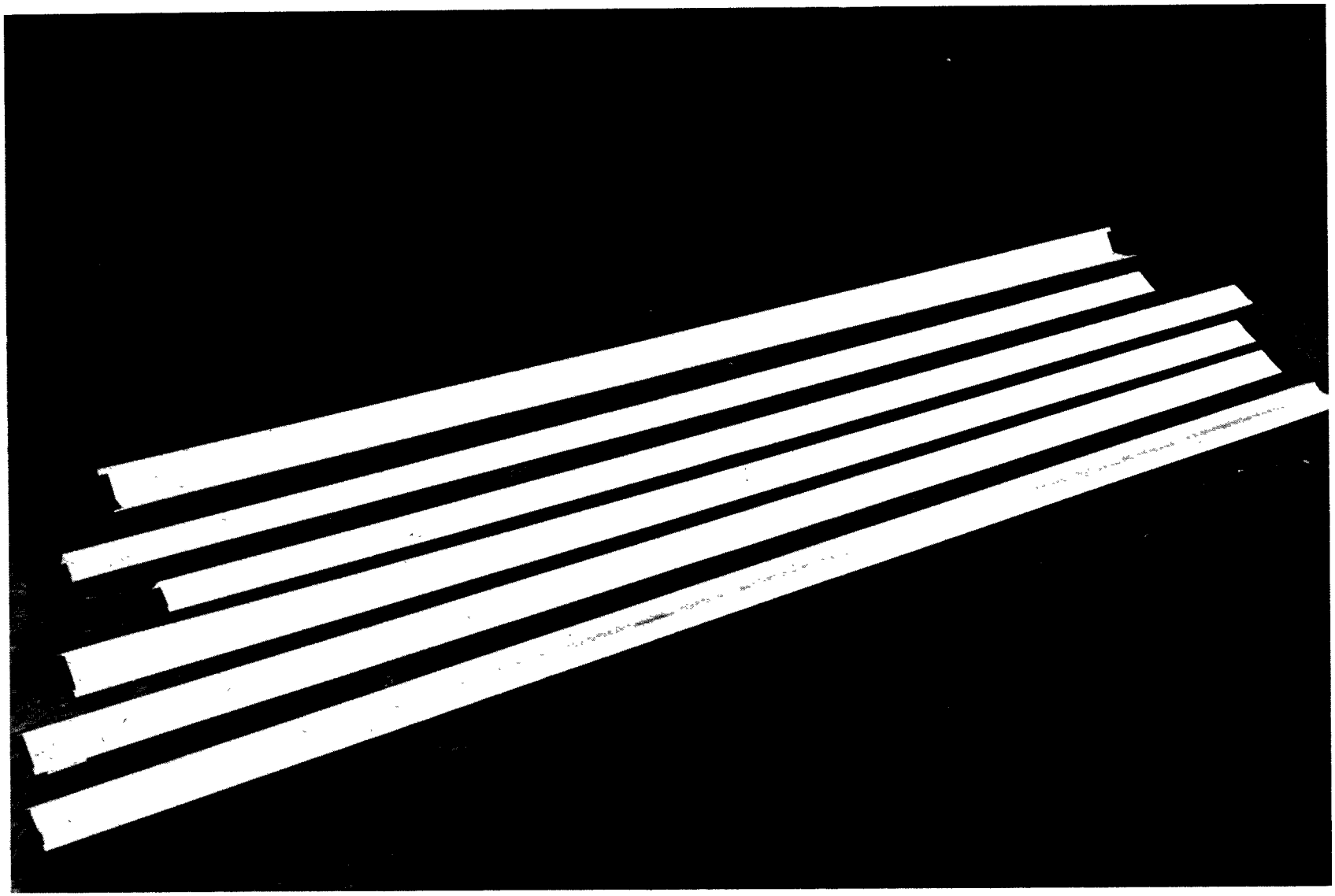

Fig. 14. Zirconium Panels for Hexagonal Tube

Fig. 15. Forming of Panels

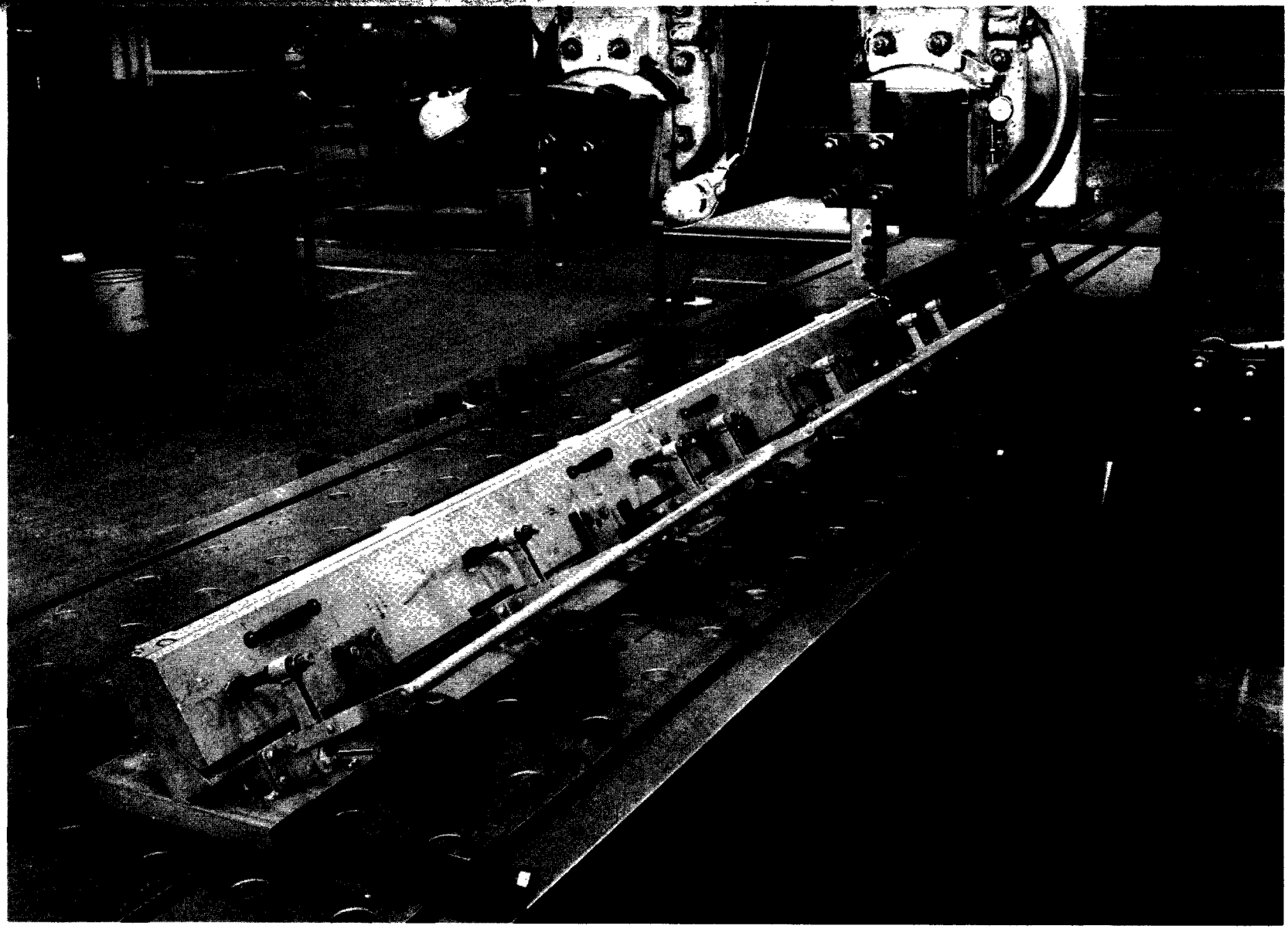

Fig. 16. Dry-Machining of Panel Edges

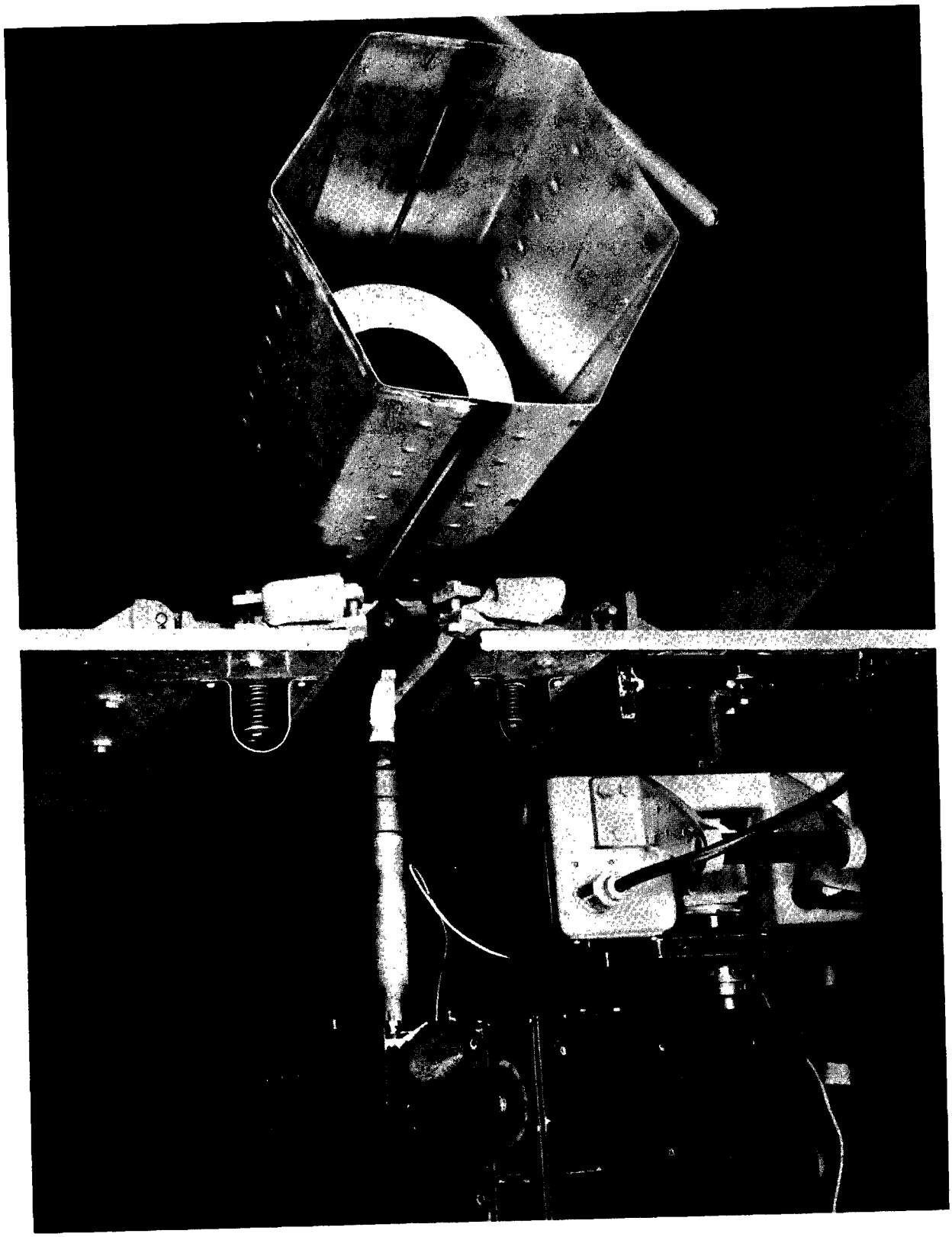

Fig. 17. Welding of Panels into Hexagonal Tube

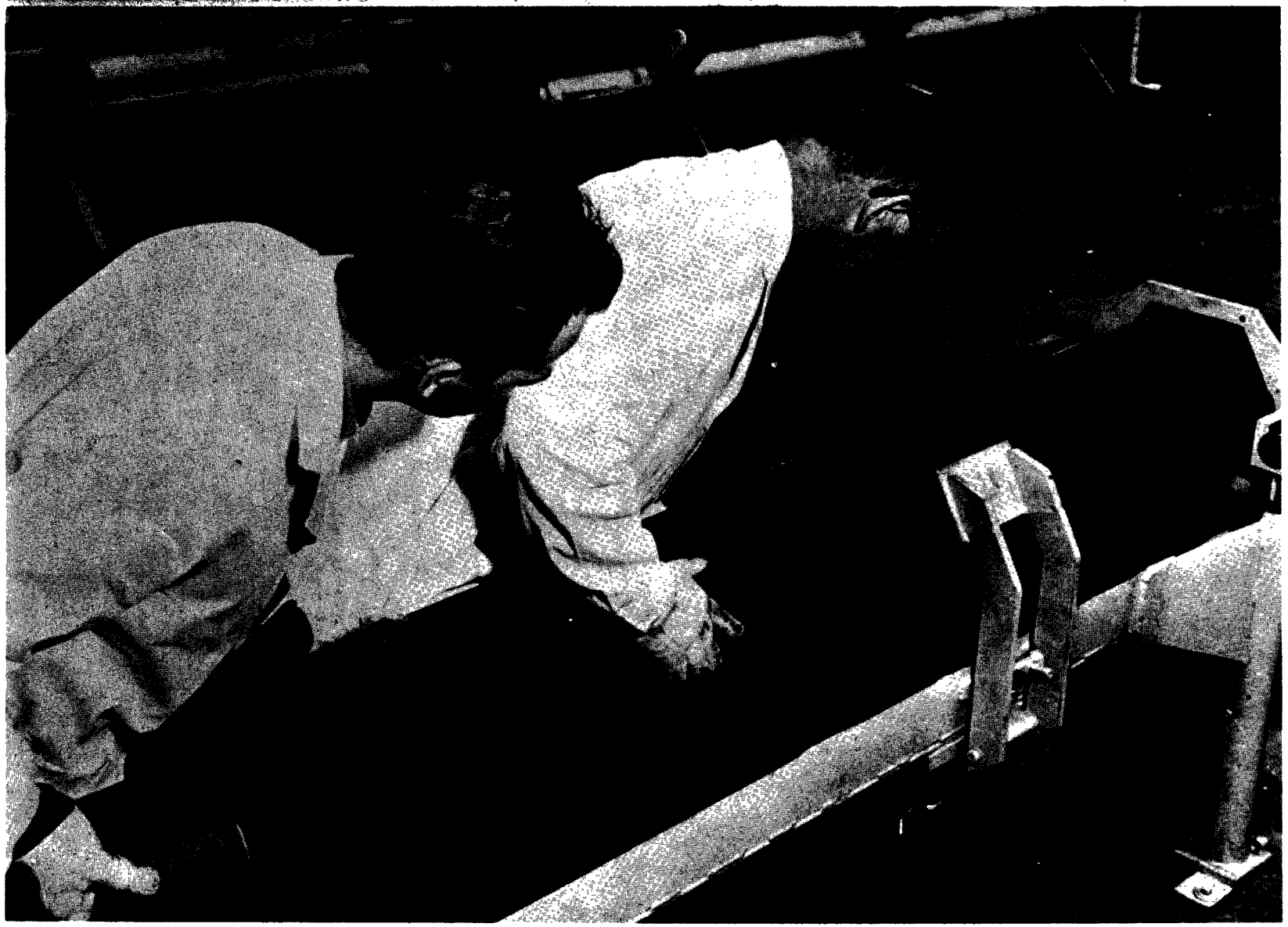

Fig. 18. Machining and Assembly of Graphite Stringers

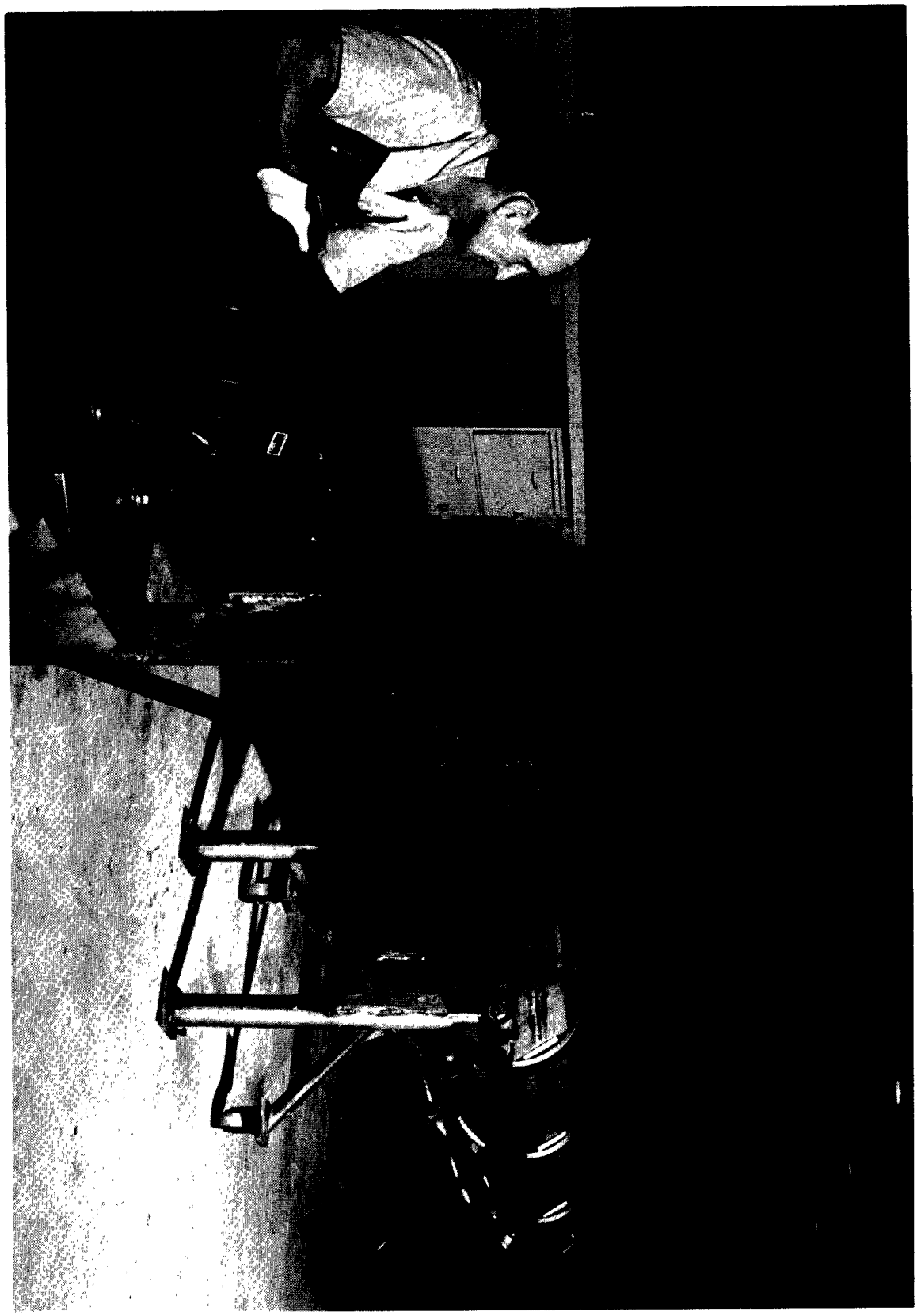

Fig. 19. Assembly of Stringers in Can Shell

Fig. 20. Insertion of Fuel Channel Tube and Can Heads

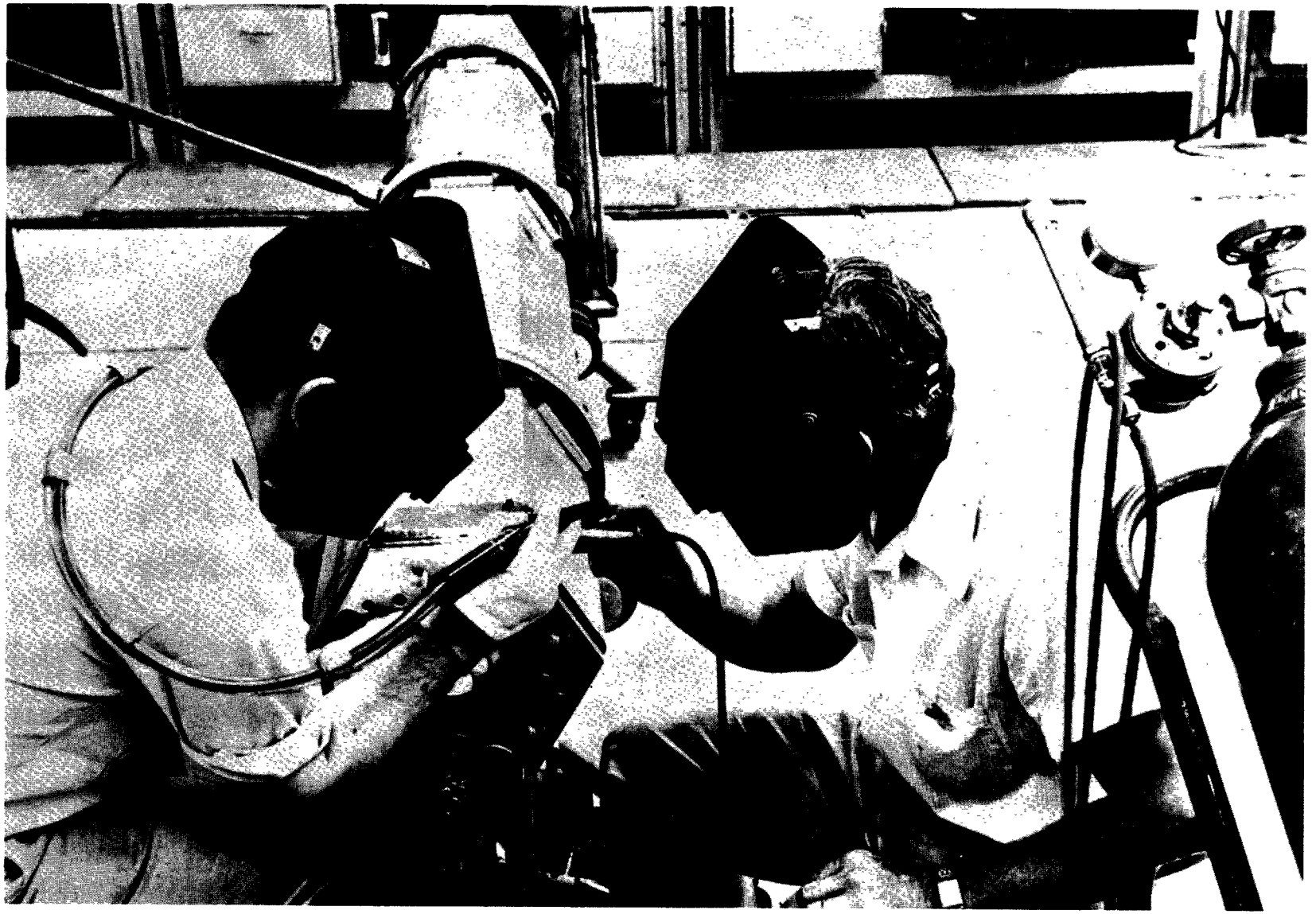

Fig. 21. Leak Test of Hexagonal Cans

SRE PHYSICAL PROGRESS

J. C. Cochran

Construction progress on the Sodium Graphite Reactor at Santa Susana is depicted from the beginning through the use of illustrations. The status and progress in each of several areas are discussed as of the date of the presentation of the paper. Particular construction problems and difficulties encountered, together with lessons learned, are discussed as they occurred during the building of the reactor.

I. BUILDINGS

The Sodium Reactor Experiment is located in the Santa Susana Mountains approximately ten miles northwest of Canoga Park, California (Fig. 1).

The excavations for the main building were made about 18 months ago in January and February, 1955 (Fig. 2). This was followed by the forming and pouring of the foundations for the hot cell, the fuel storage cells, the fuel cleaning cells, and the primary fill tank vault.

These basic foundations had been completed by July, 1955 (Fig. 3). By November, 1955, all foundations had been completed (Fig. 4). The slabs which were to be used for the sides of the building had been formed and were lying read for use. The erection of the steel for the support of the bridge crane had begun. The outer tank and the core tank had been delivered to the site.

By the following spring, the main building construction had been completed (Fig. 5). The sodium service building had almost been completed. The pouring of foundations in the secondary area had begun. The equipment for the circulatio and cooling of tetralin had been installed.

At the present time, all facilities construction has been completed, includi the paving of the outside area. The fence has yet to be installed.

II. REACTOR

The construction in the reactor area itself can also be traced. The founda tions here were begun in March, 1955 (Fig. 6). Three months later, the core cavity liner, together with its surrounding coolant lines, had been installed, as well as the fuel storage cells (Fig. 7).

By November, the outer tank and the thermal rings also had been put in place (Fig. 8). This was followed by the installation of the core tank. However, it was necessary to remove this last component in February, 1956 to effect repai made necessary by the use of improper welding rod during the initial fabrication (Fig. 9). These repairs were completed and the core tank was reinstalled in May, 1956 (Fig. 10).

The ring shield (Fig. 11) was delivered in August, filled with heavy concrete, which increased the weight to 75 tons, and put in place over the core tank.

The next step taken was to install the bellows (Fig. 12) between the outer tank, the core tank, and the ring shield. The loading face plug was delivered in September (Fig. 13). Prior to its being filled with heavy concrete, an additional leak test was made of the coolant coils used to circulate tetralin in the loading face plug. Although these coils had been tested several times during fabrication, as well as just prior to delivery to the site, it was found on this test that one of them leaked in an inaccessible place. It was then decided to install additional coils both in the loading face plug and in the plugs which are inserted in the loading face plug. This work is currently in progress (Fig. 14).

While these repairs were being effected, the 119 moderator and reflector cans were installed.

In the reactor area (Fig. 15), our schedule calls for installation of the loading face, leak testing, and turnover of the area to the operating personnel on December 14. The hot cell has been completed and turned over to operating personnel.

III. AUXILIARY COMPONENTS

The radioactive waste area is scheduled for completion during the second week in December.

All components in the main primary gallery (Fig. 16) and the interconnecting piping have been installed. Only a very few heaters have yet to be placed on the pipe lines and to be covered by insulation (Fig. 17 and 18).

The present items of work left to be accomplished are installation of valve operators and tying-in of the electrical heaters to the panels outside of the gallery. This is also the general status of the auxiliary primary gallery (Fig. 19 and 20).

In the secondary area (Fig. 21), all components have been installed, but completion of the piping has been delayed by the late receipt of stop valves required by the interconnection to the Southern California Edison steam generating equipment. These valves have now been received.

The sodium service area will be the last to complete (Fig. 22). We have yet to receive the hot trap components for installation. These are promised by the middle of November. If they are received, the schedule calls for completion of this area by January 6, 1957.

The control room and the motor generator room have been completed and turned over to the operating personnel (Fig. 23 and 24).

IV. REMARKS

In summary, the estimated physical completion of the entire installation is 98 per cent. If our present schedule is met in all areas, it is intended to commence loading sodium and to conduct fuel loading and a critical experiment in the early part of 1957.

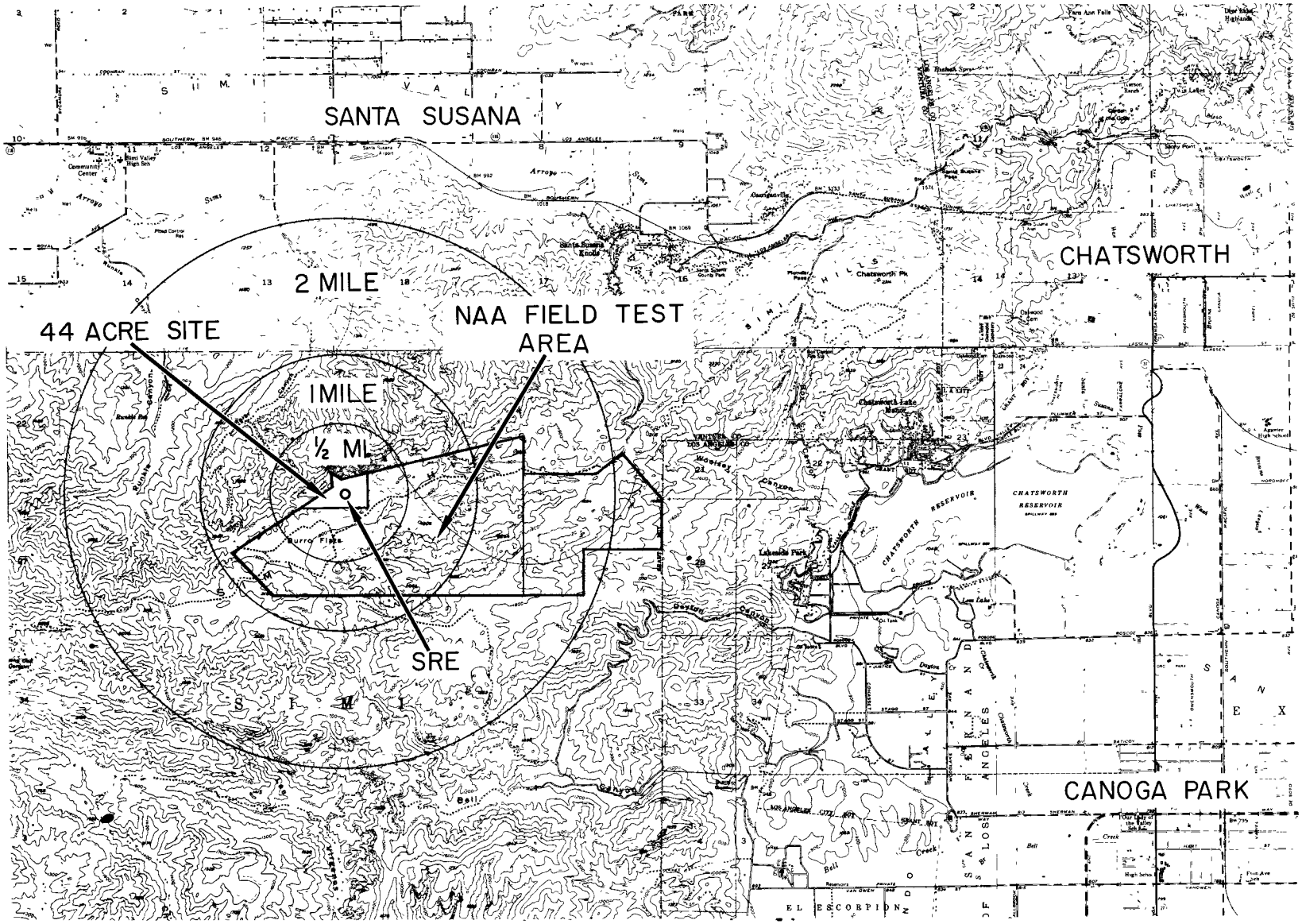

Fig. 1. SRE Site Location

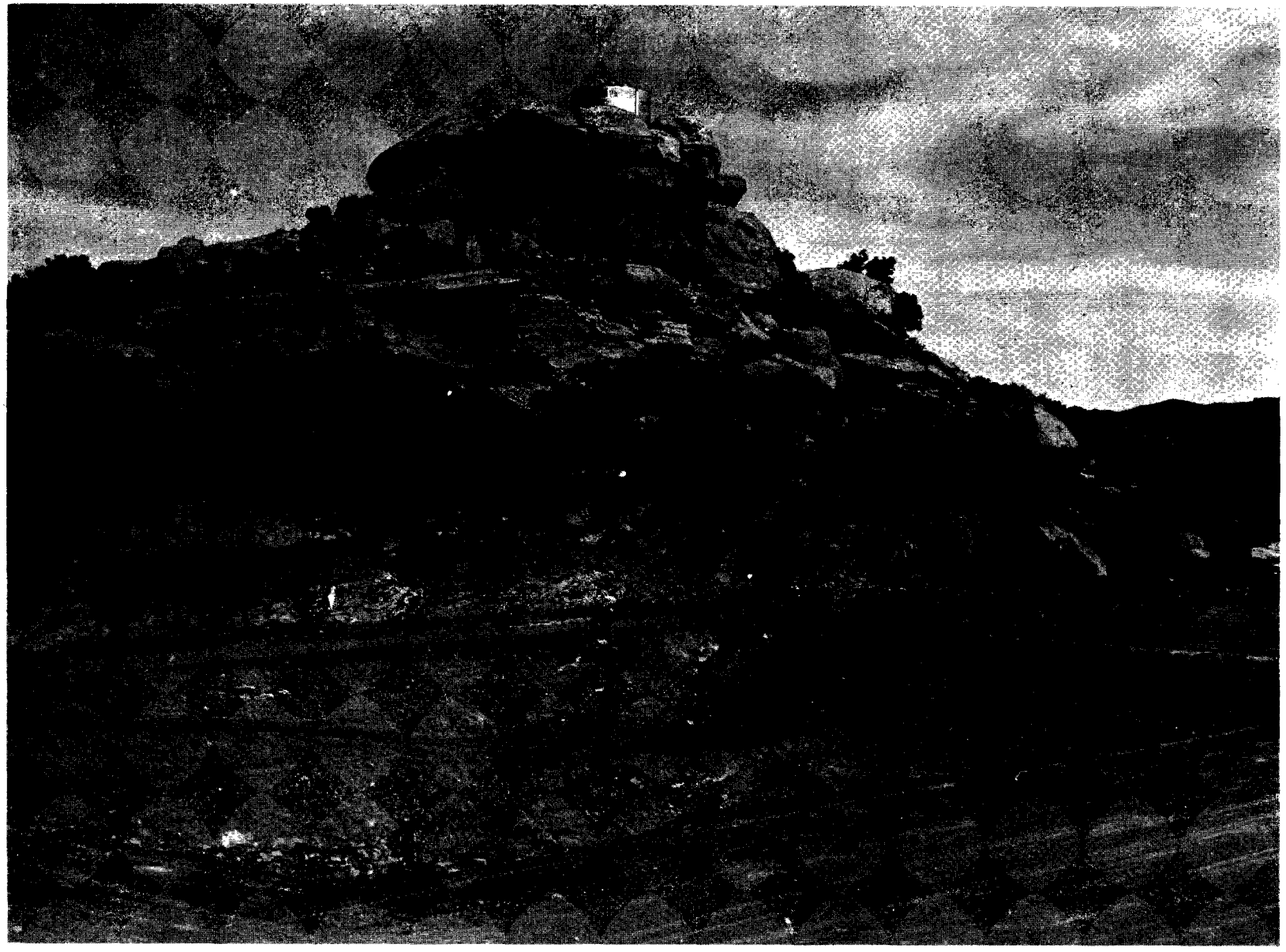

Fig. 2. Excavations For Main Building

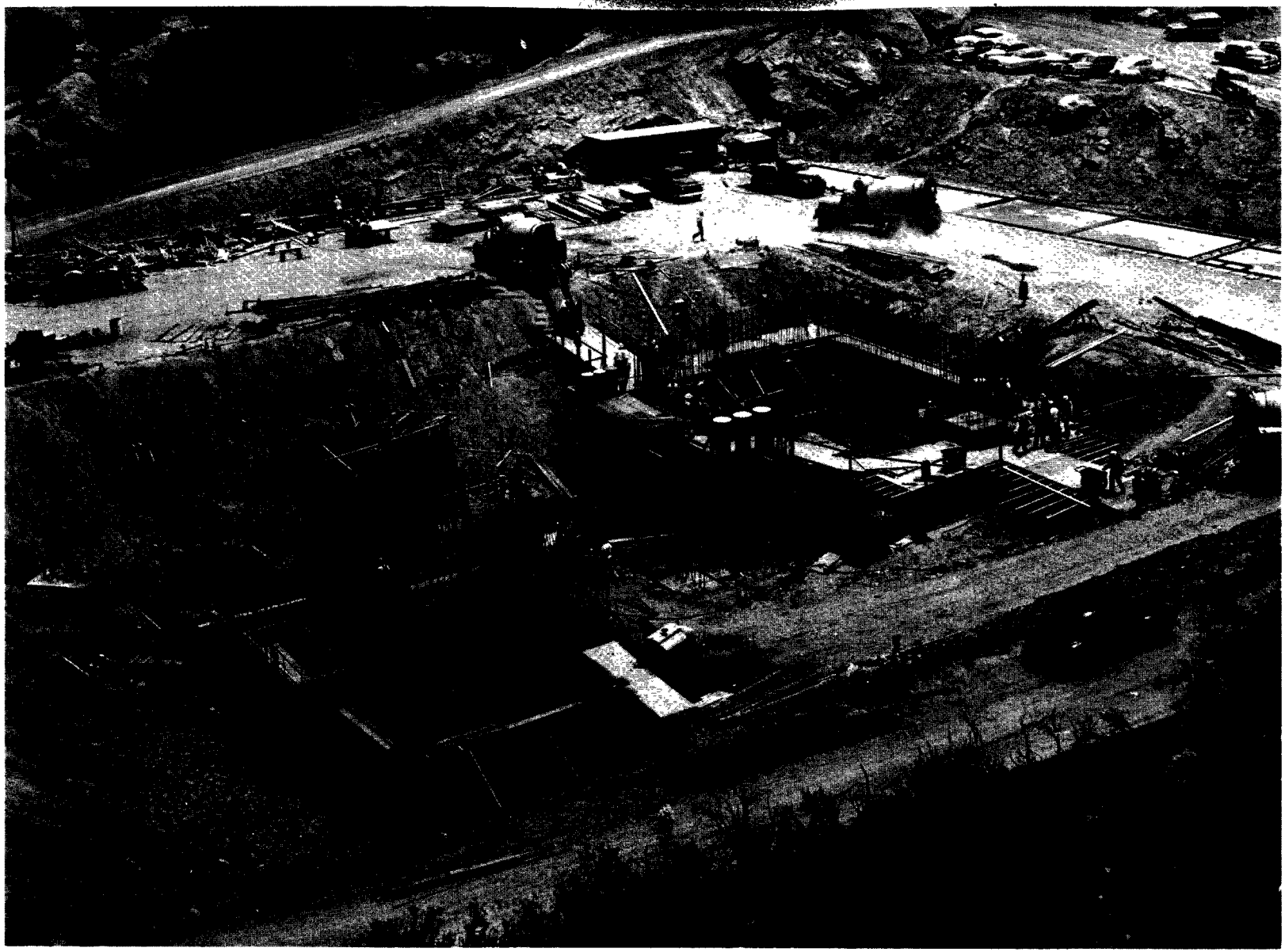

Fig. 3. Foundations (July 1955)

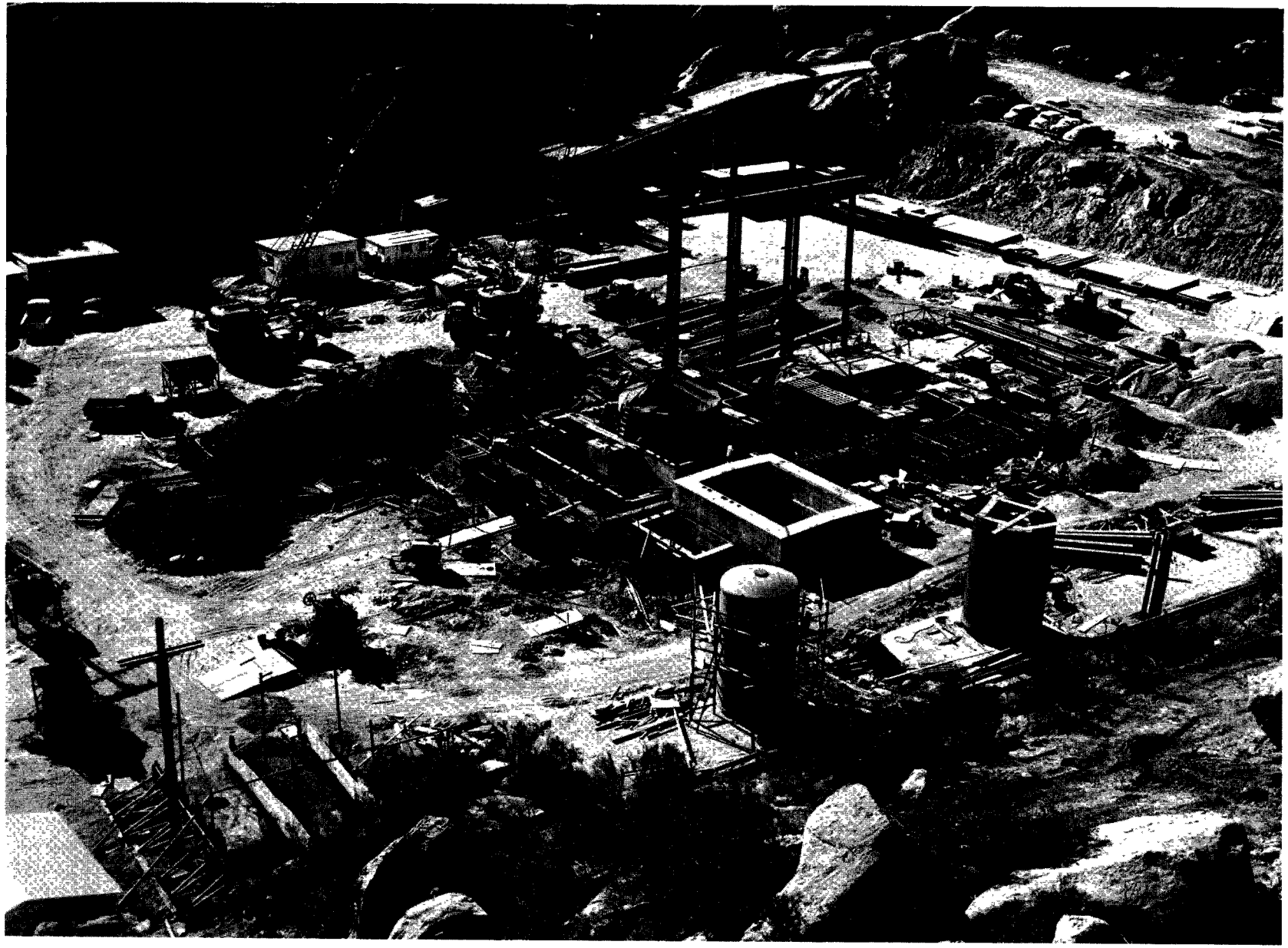

Fig. 4. Foundations (November 1955)

Fig. 4. Foundations (November 1955)

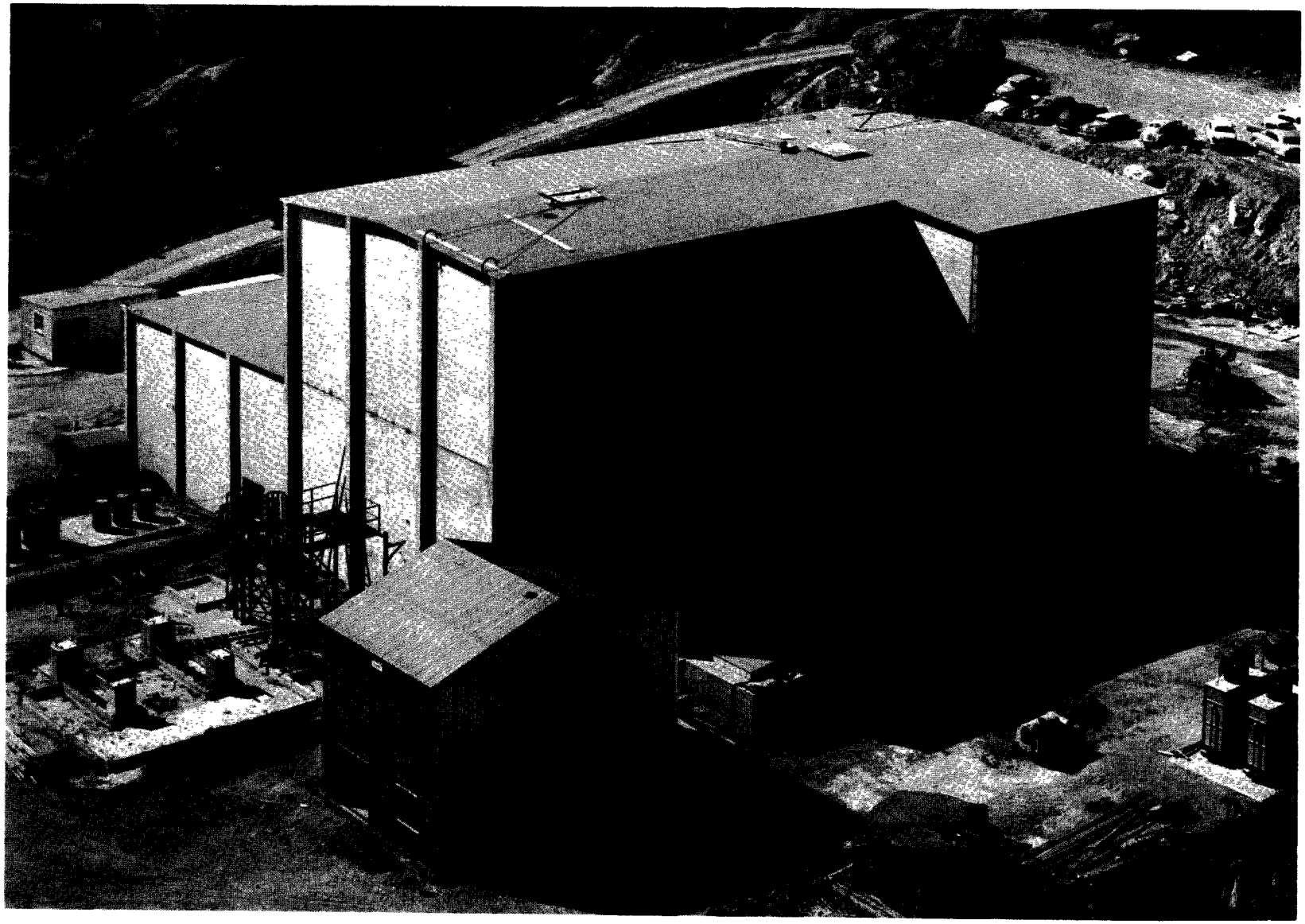

Fig. 5. Main Building (April 1956)

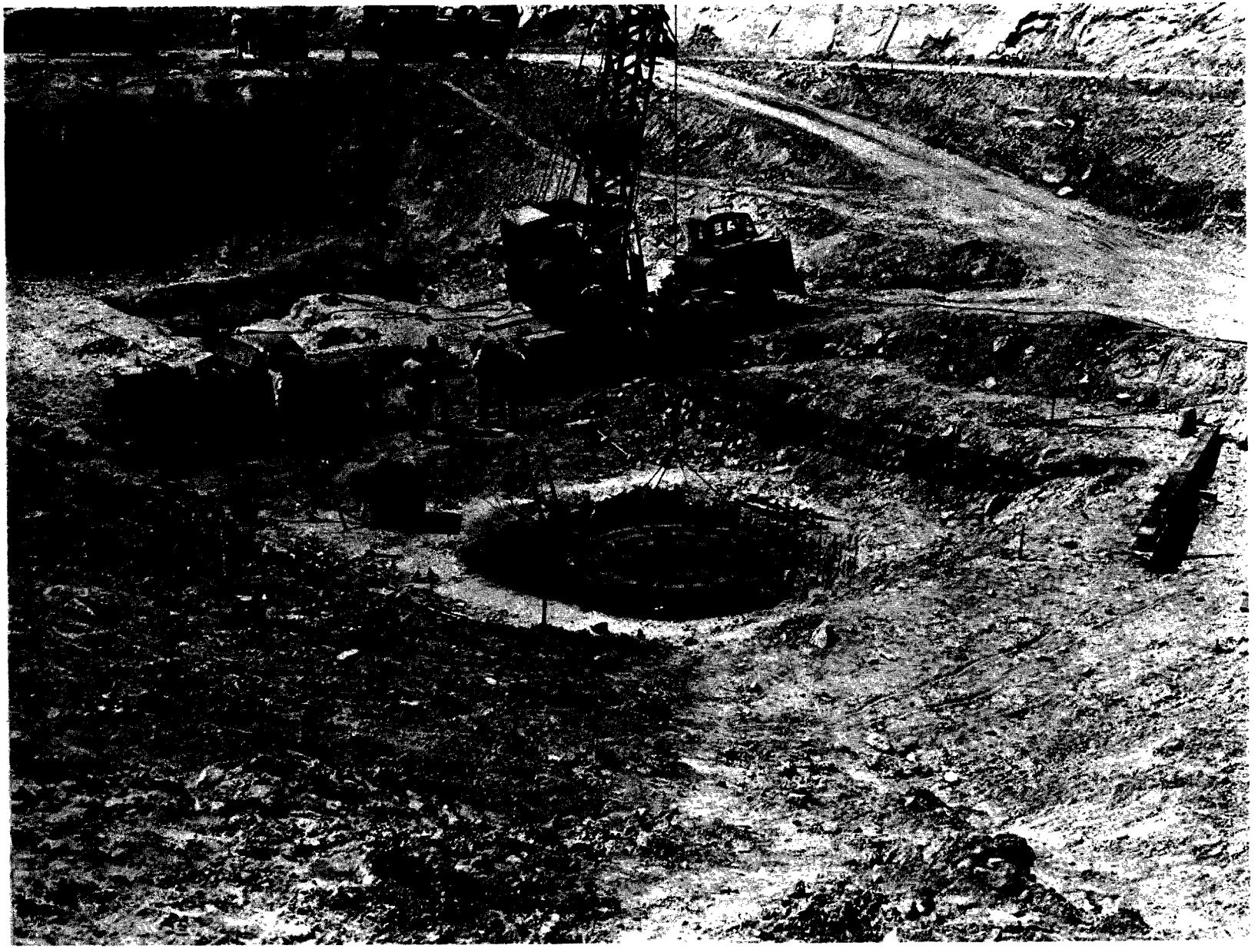

Fig. 6. Reactor Foundations (March 1955)

Fig. 6. Reactor Foundations (March 1955)

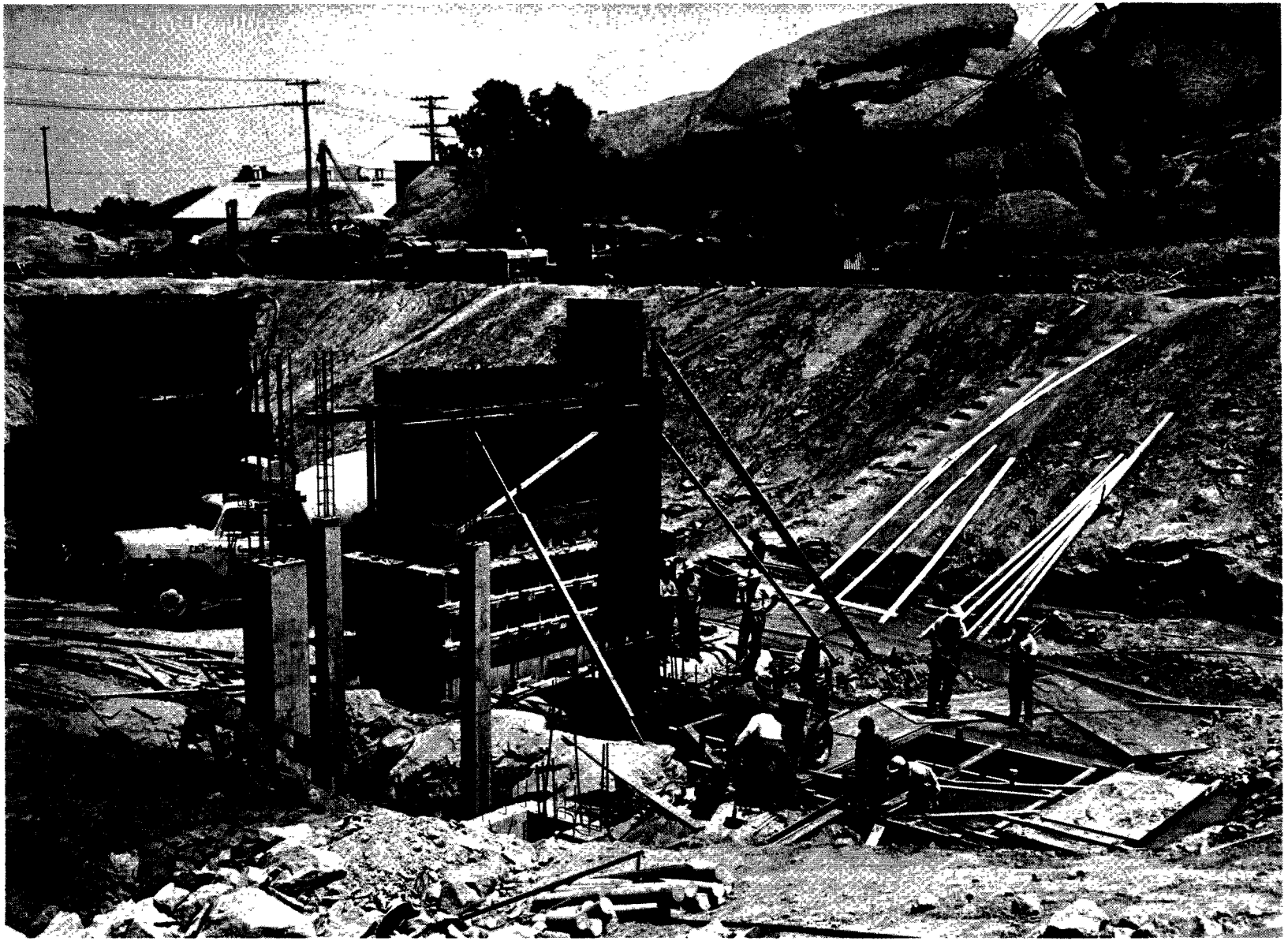

Fig. 7. Core Cavity Liner, Coolant Lines, and Fuel Storage Cells (June 1955)

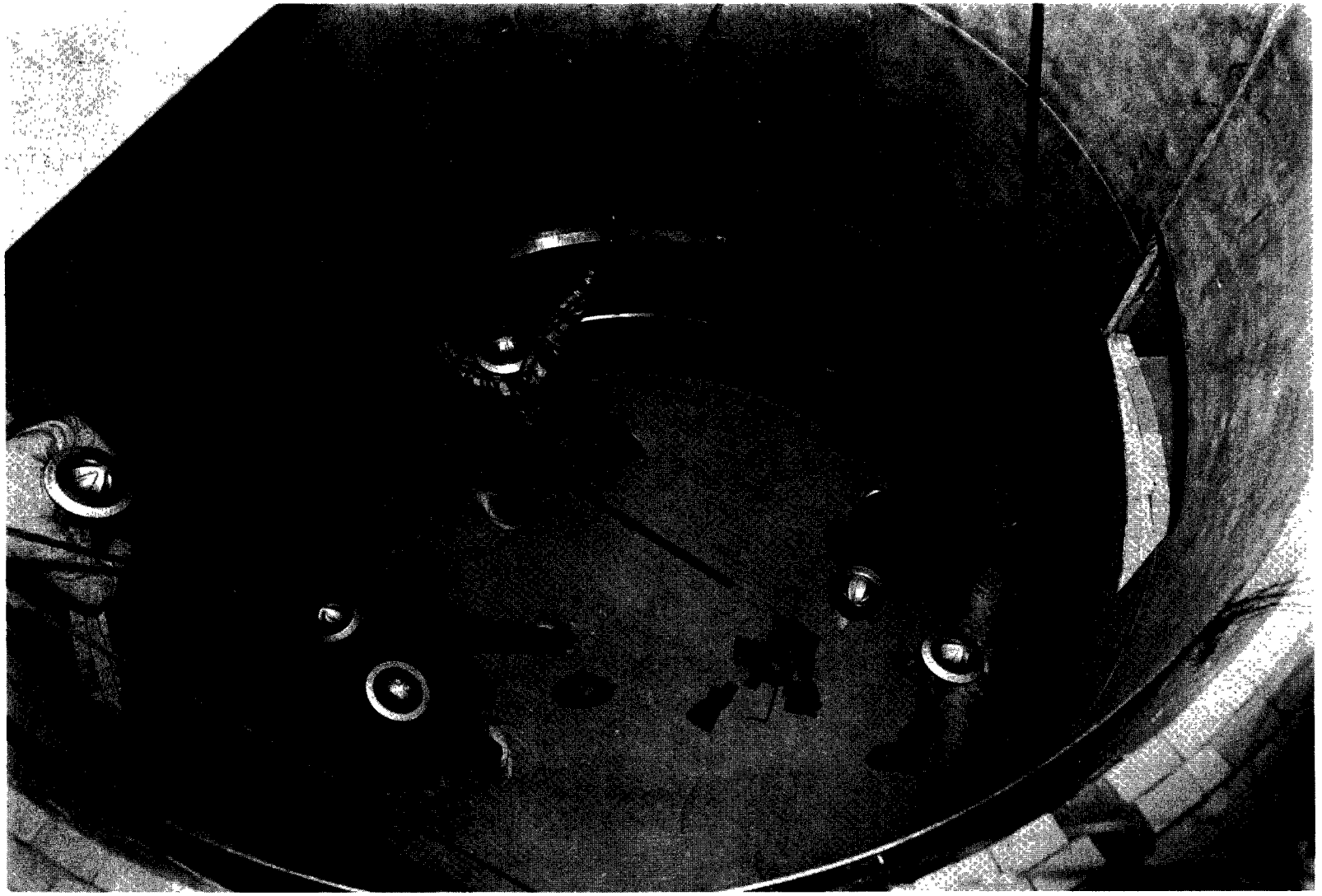

Fig. 8. Installation of Outer Tank and Thermal Rings (November 1955)

Fig. 8. Installation of Outer Tank and Thermal Rings (November 1955)

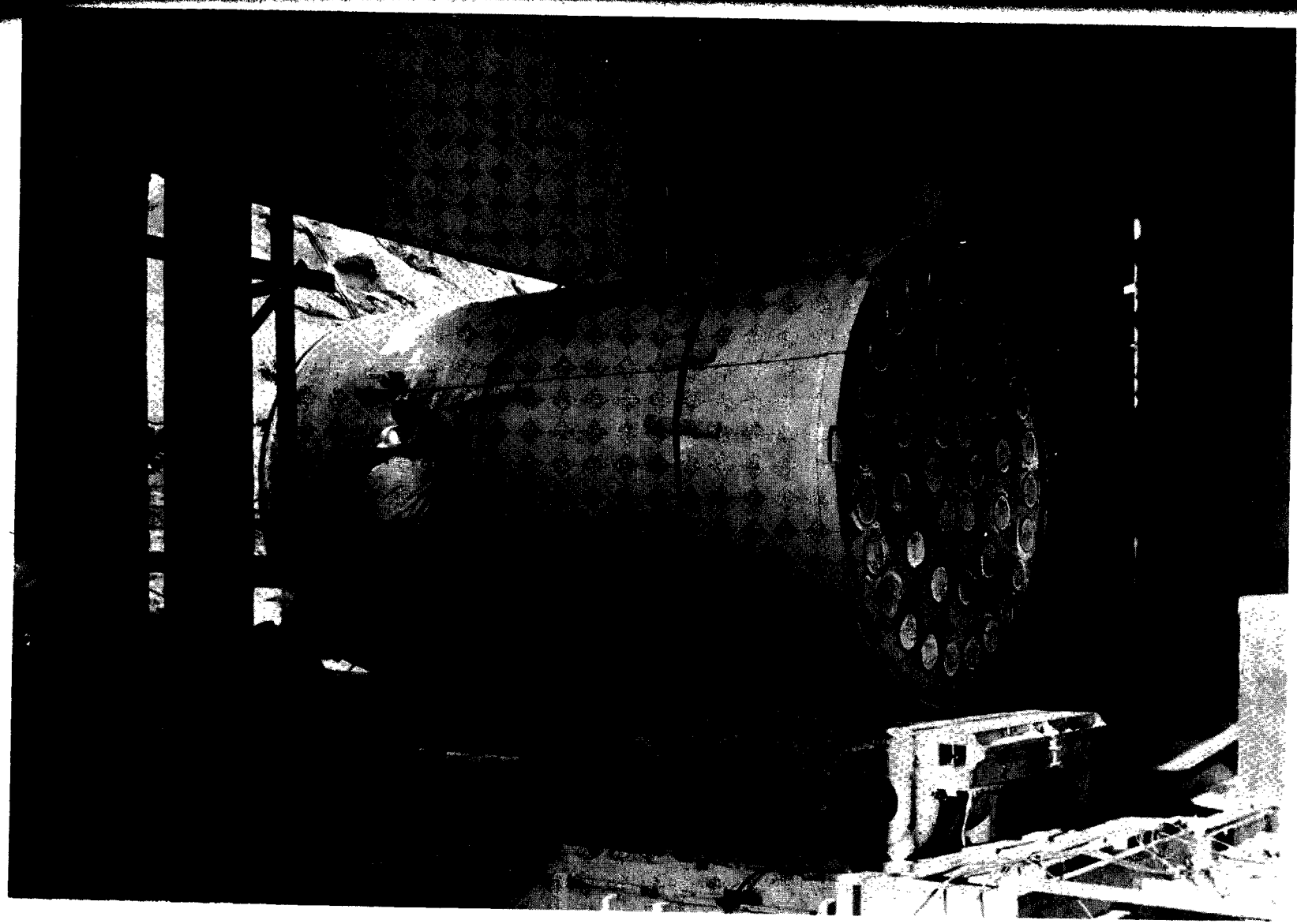

Fig. 9. Core Tank Removal (February 1956)

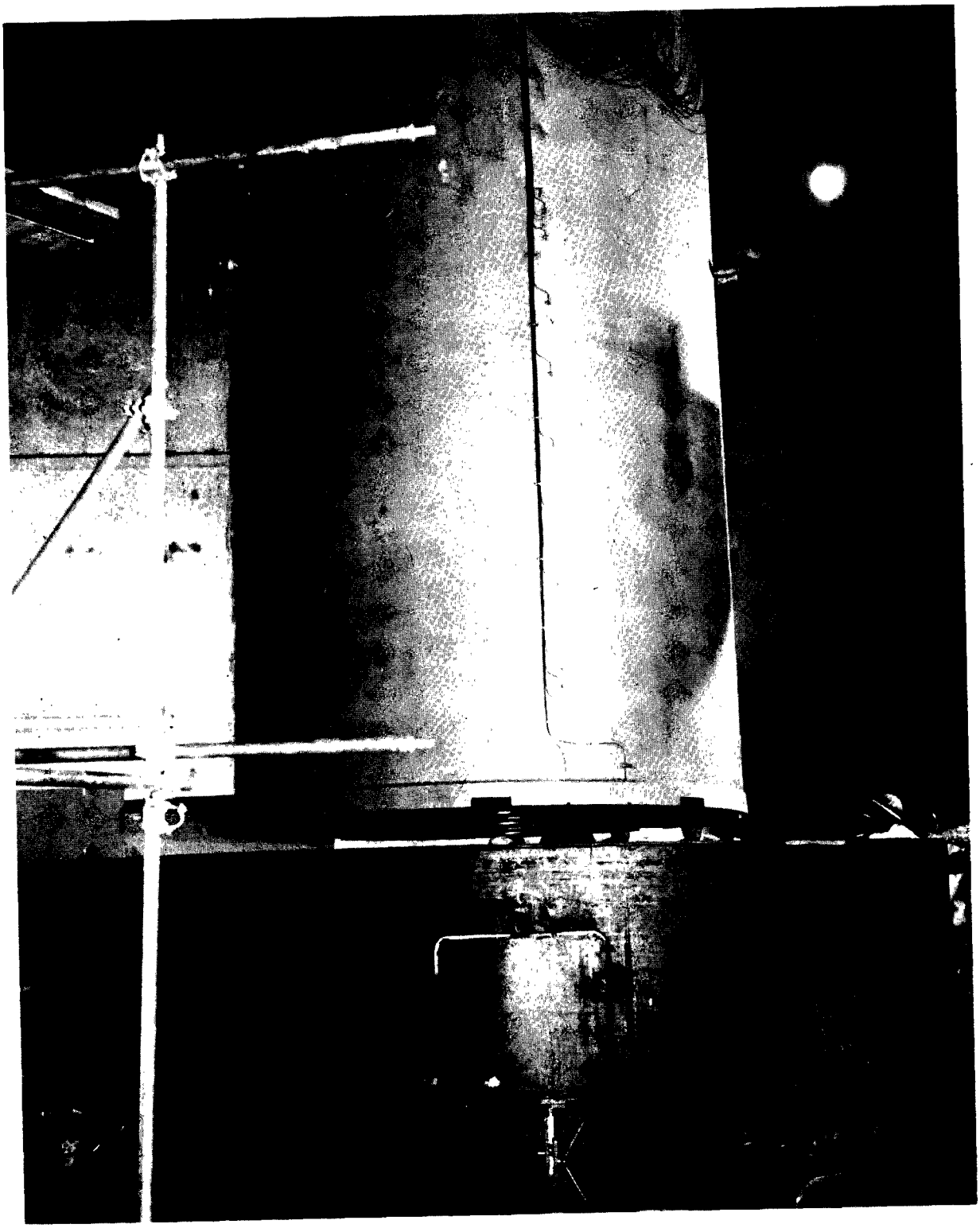

Fig. 10. Core Tank Reinstallation (May 1956)

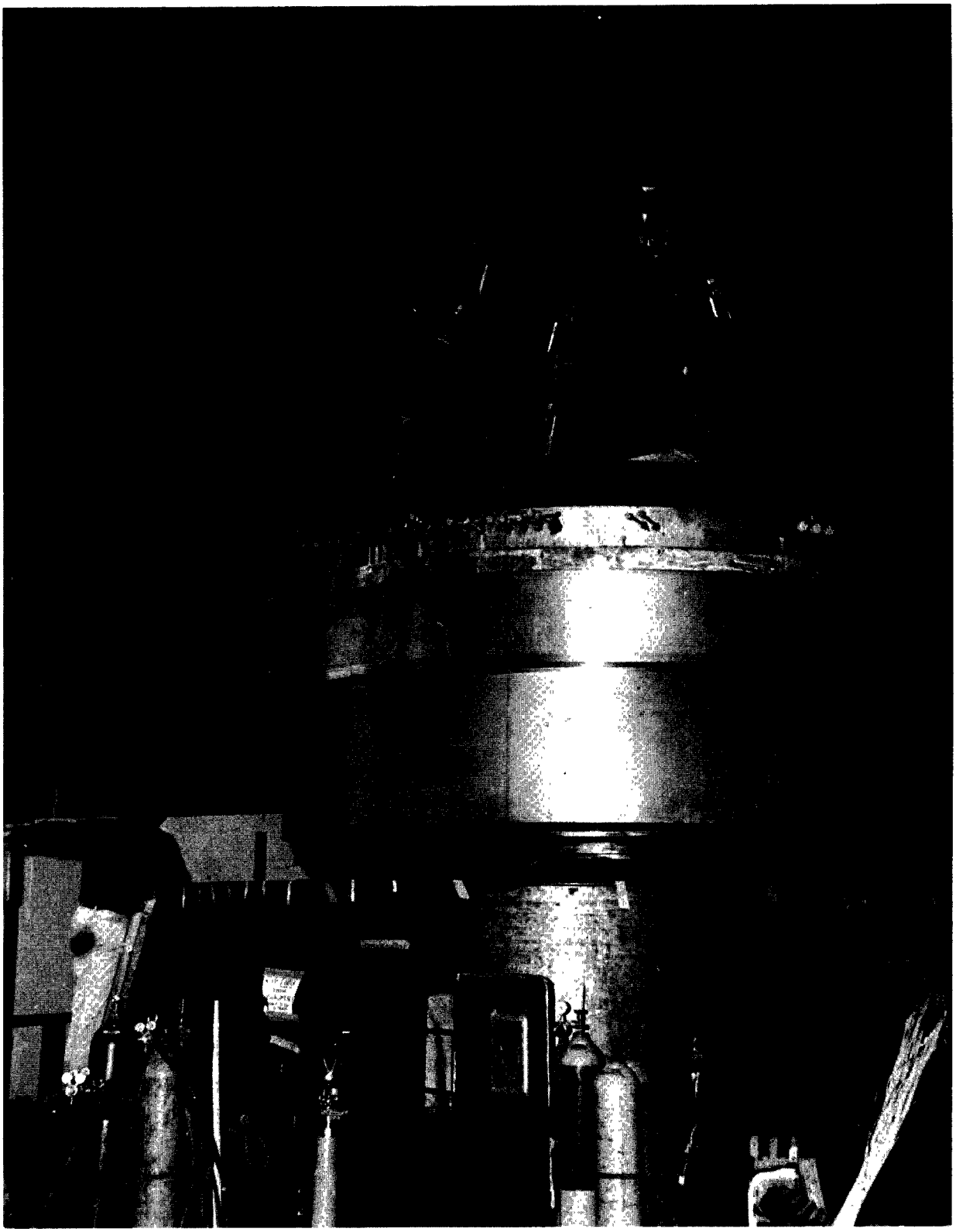

Fig. 11. Ring Shield Installation (August 1956)

Fig. 12. Installation of Bellows (August 1956)

Fig. 12. Installation of Bellows (August 1956)

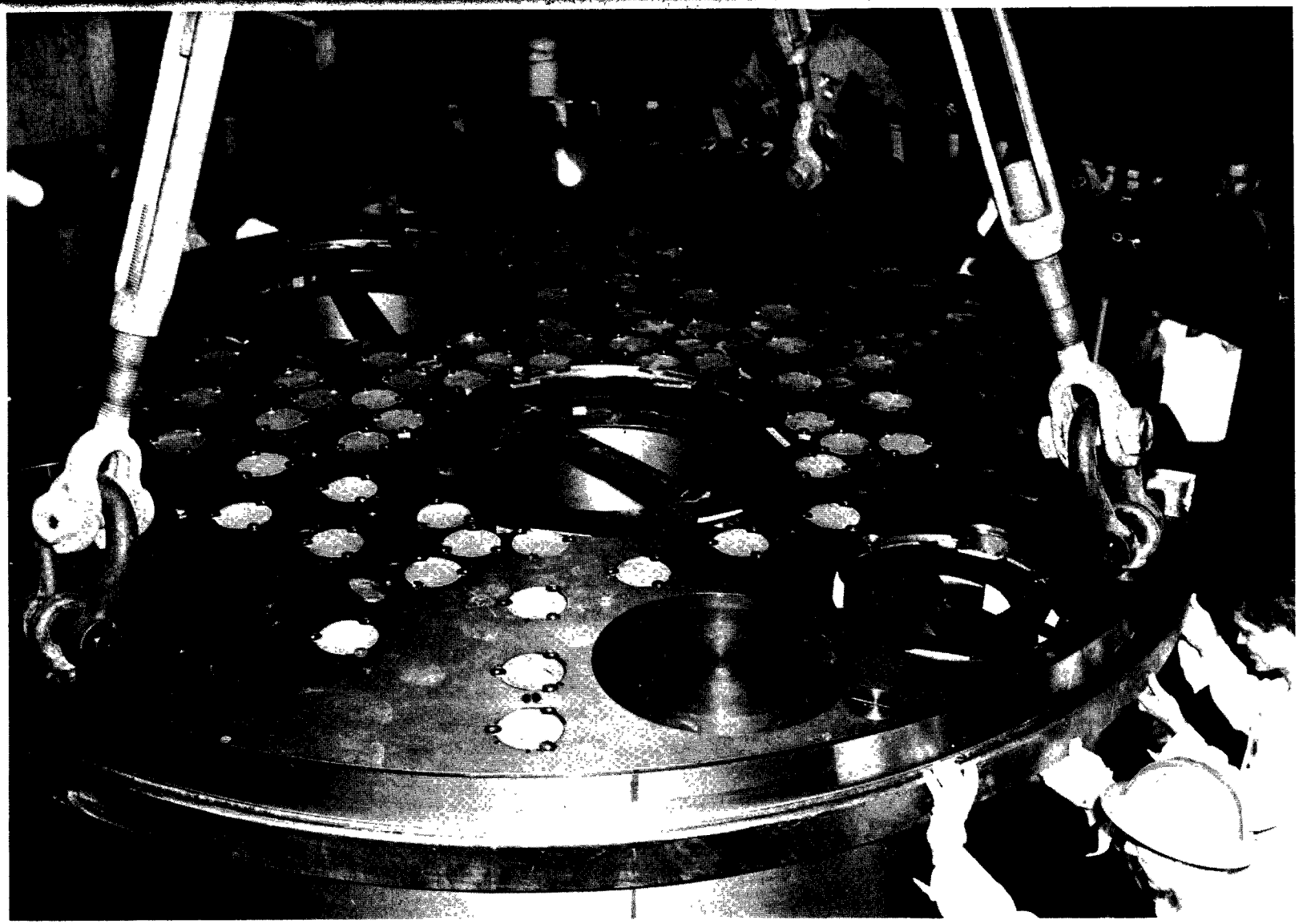

Fig. 13. Installation of Loading Face Plug (September 1956)

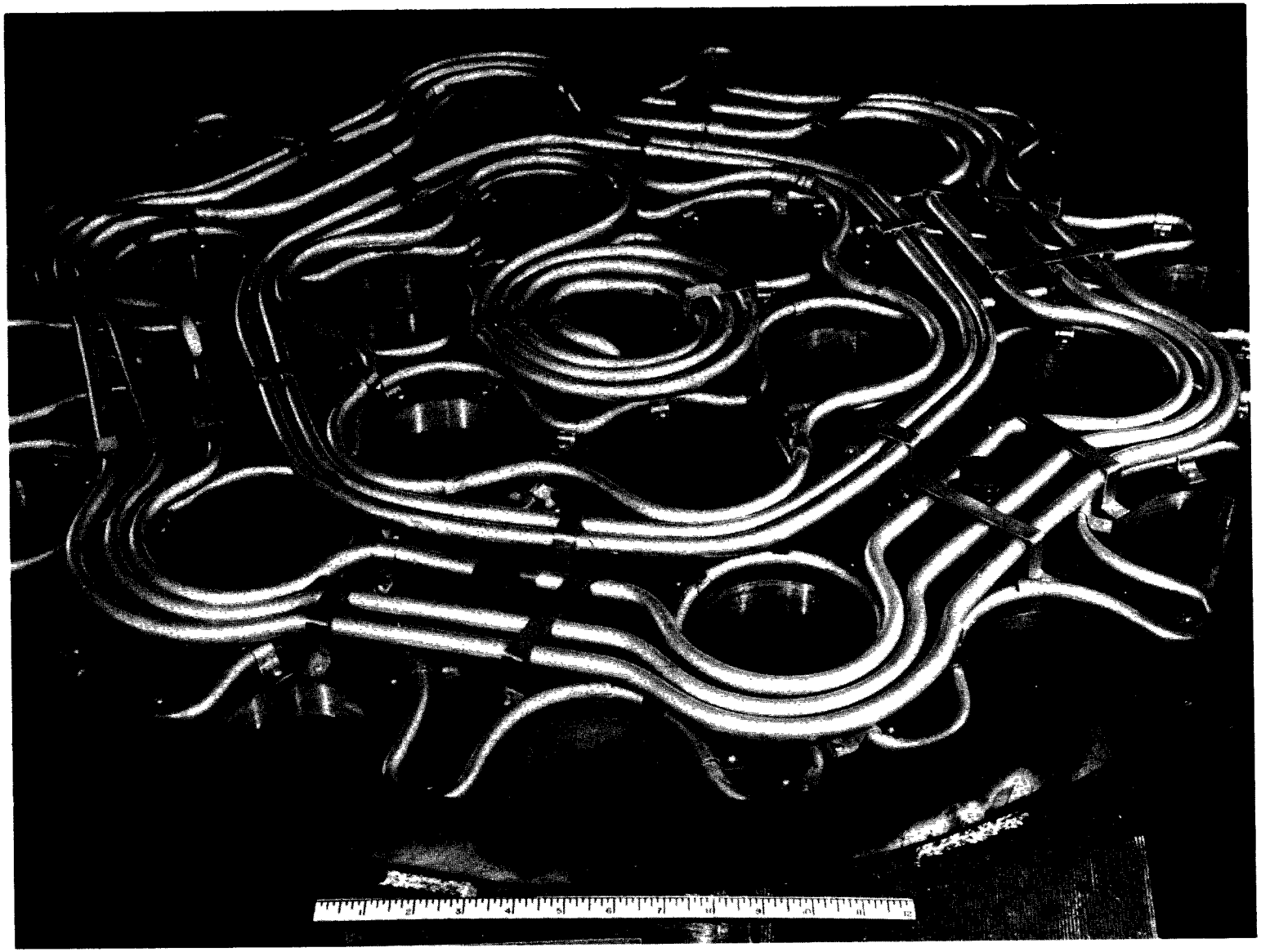

Fig. 14. Loading Face Cooling Coils

Fig. 14. Loading Face Cooling Coils

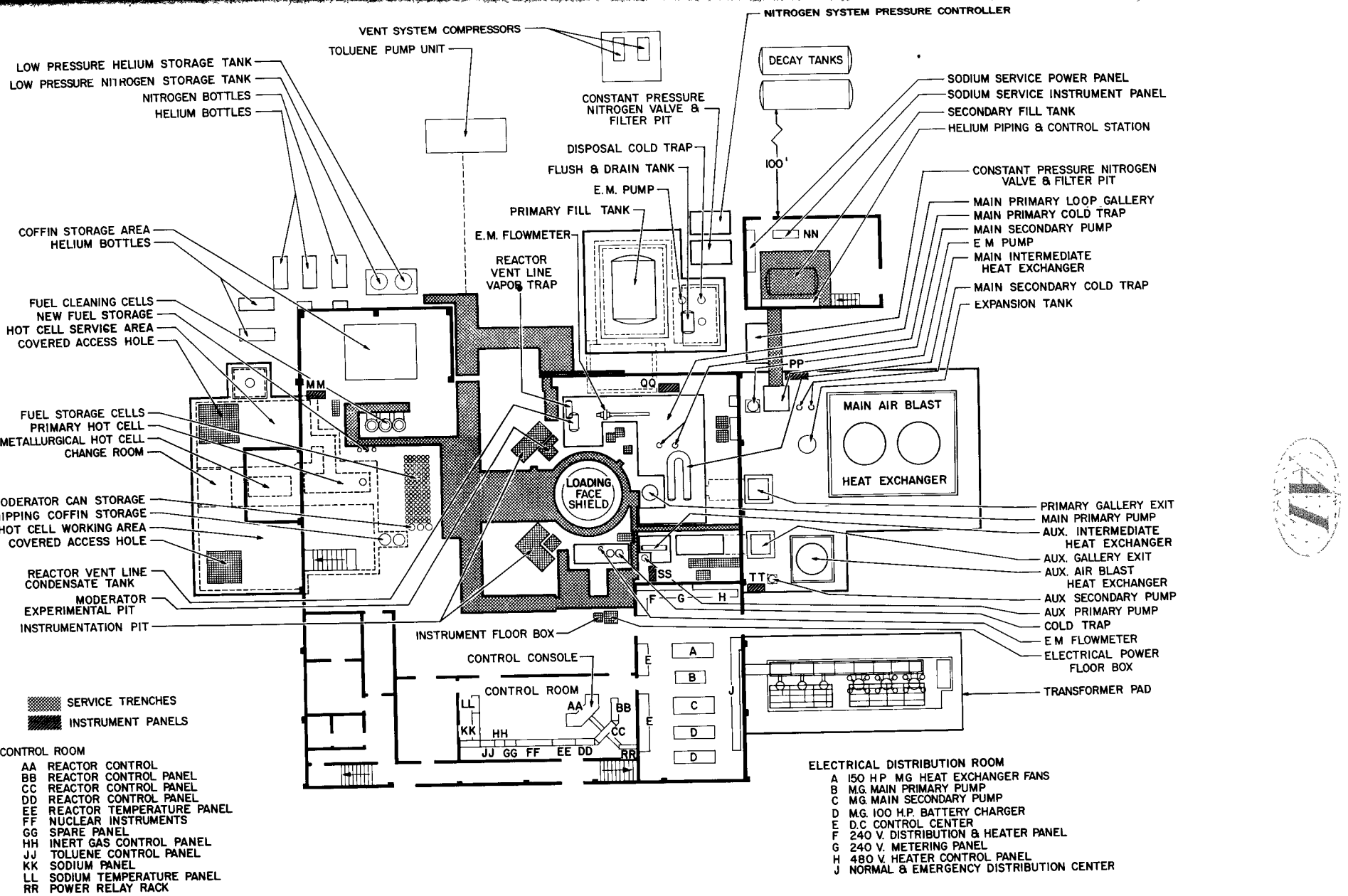

Fig. 15. General Layout of Reactor Area

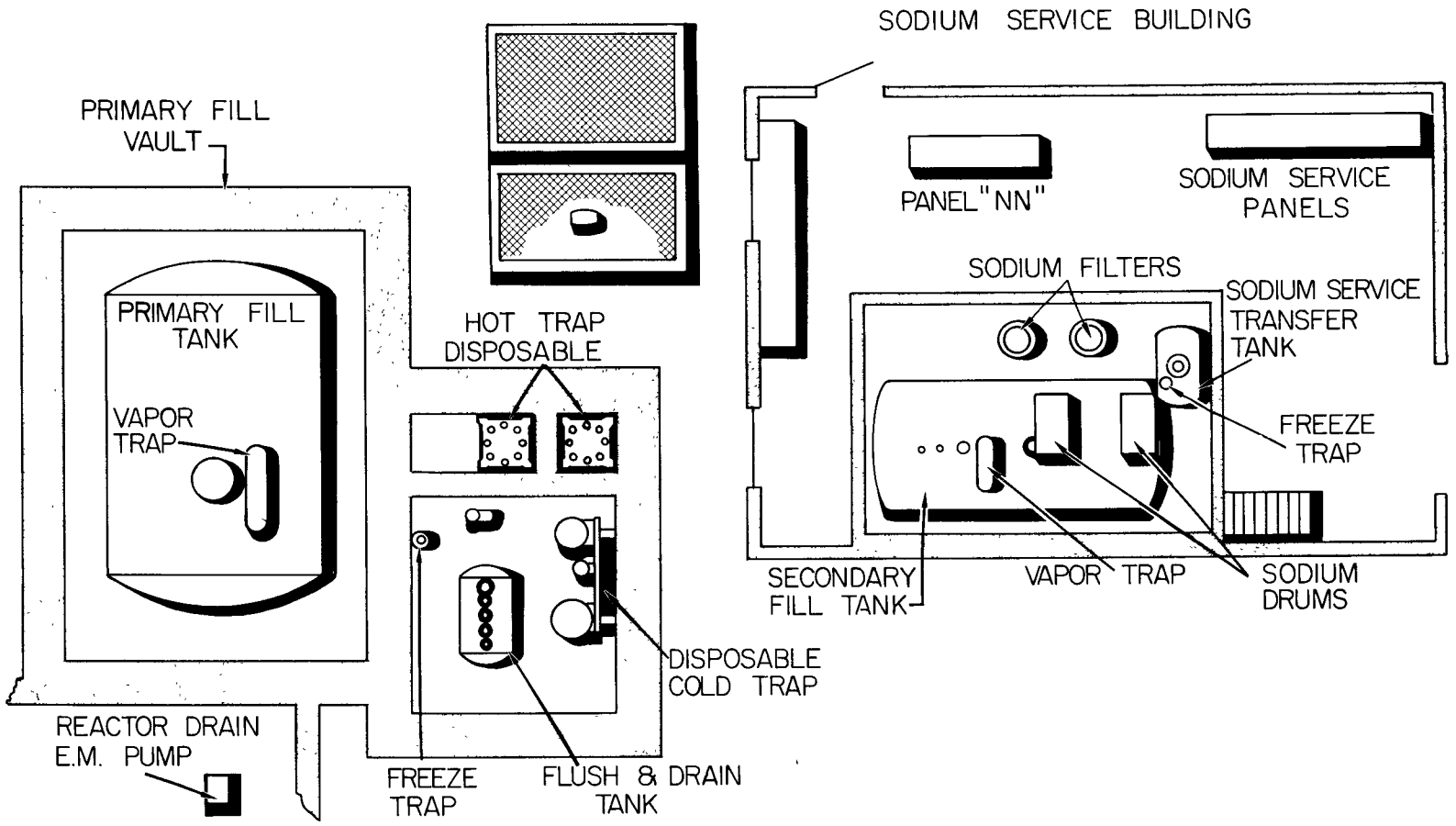

Fig. 16. Main Primary Gallery Components

Fig. 16. Main Primary Gallery Components

Fig. 17. Installation of Heaters on Fill Tank

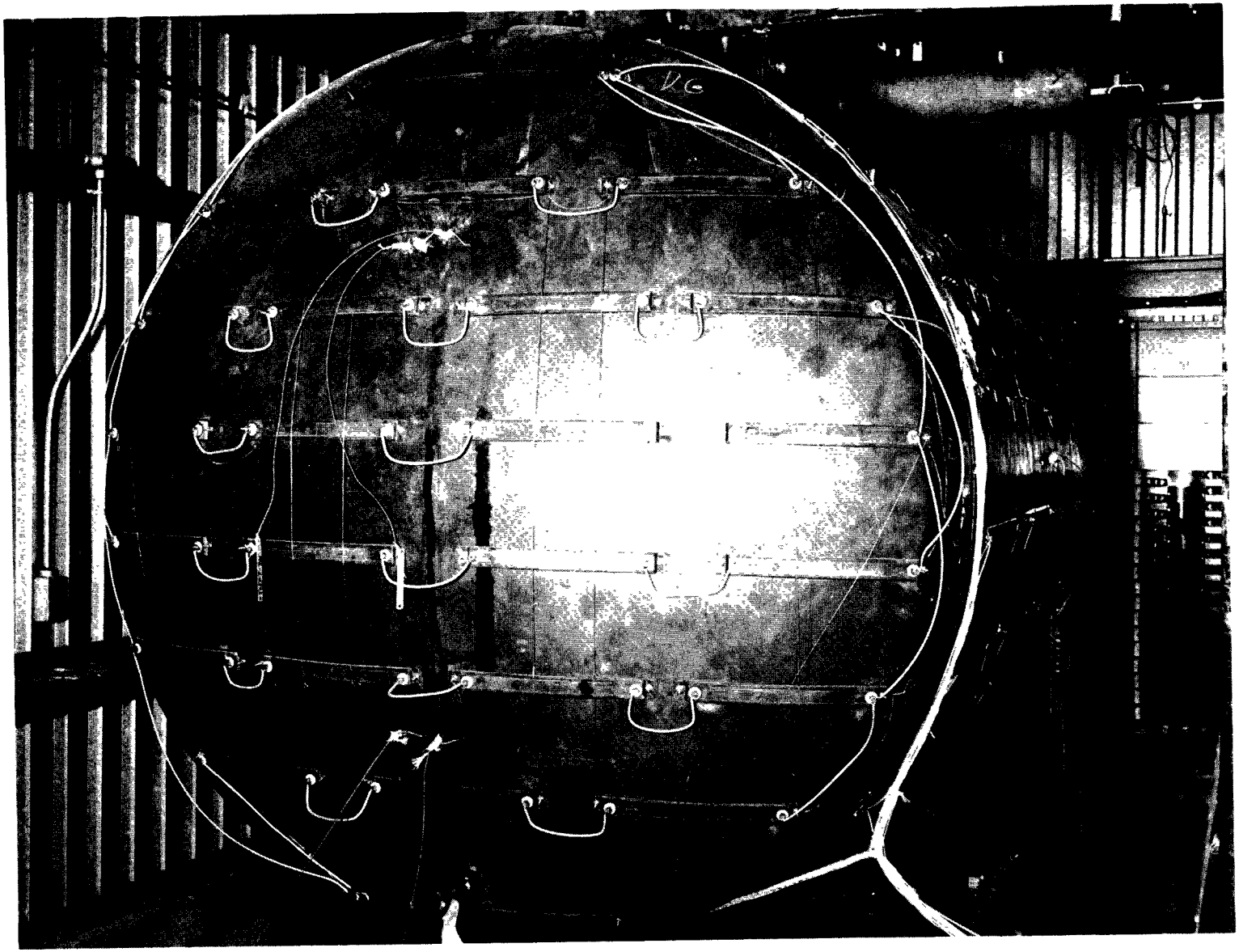

Fig. 18. Fill Tank, Showing Heaters

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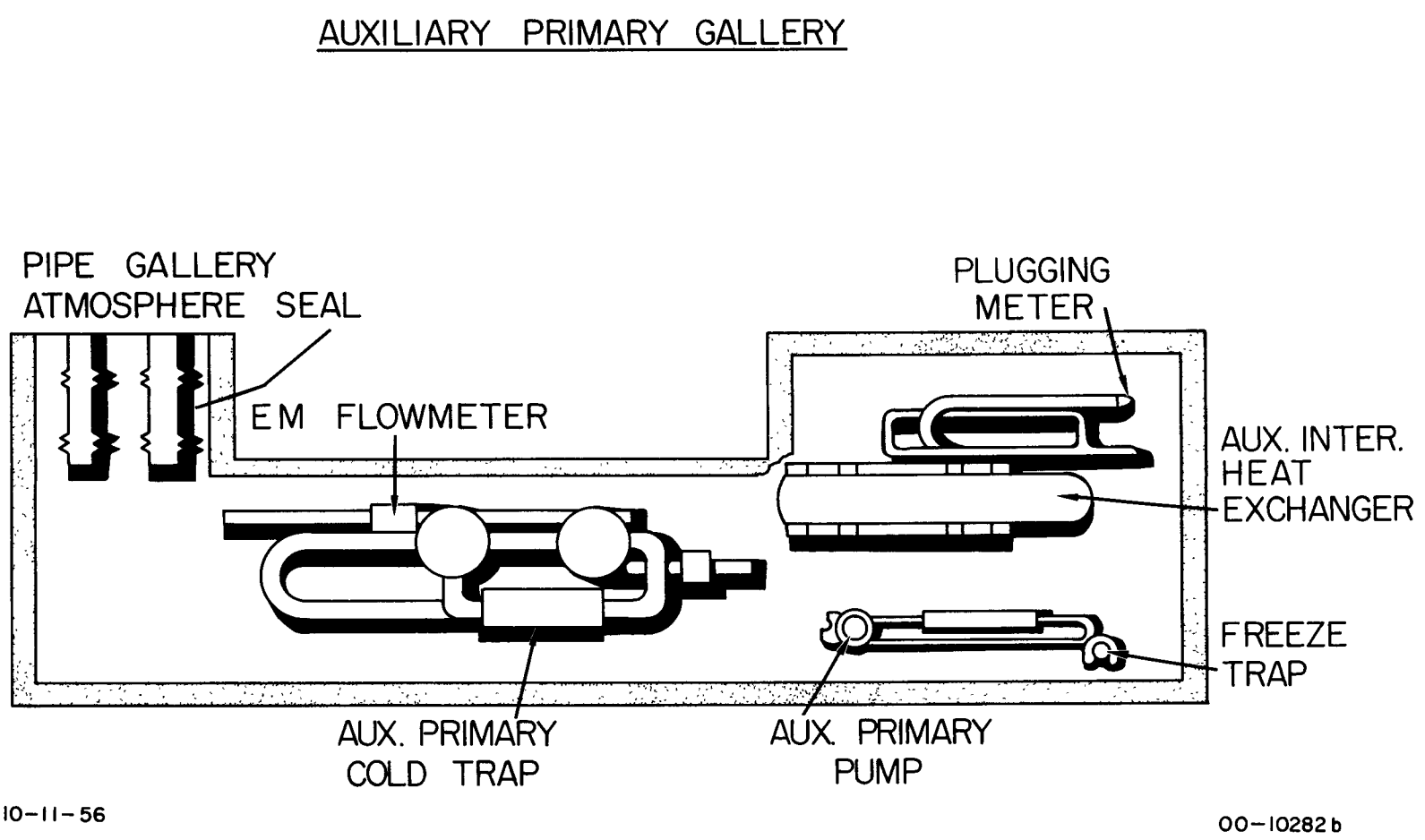

Fig. 19. Auxiliary Primary Gallery Components

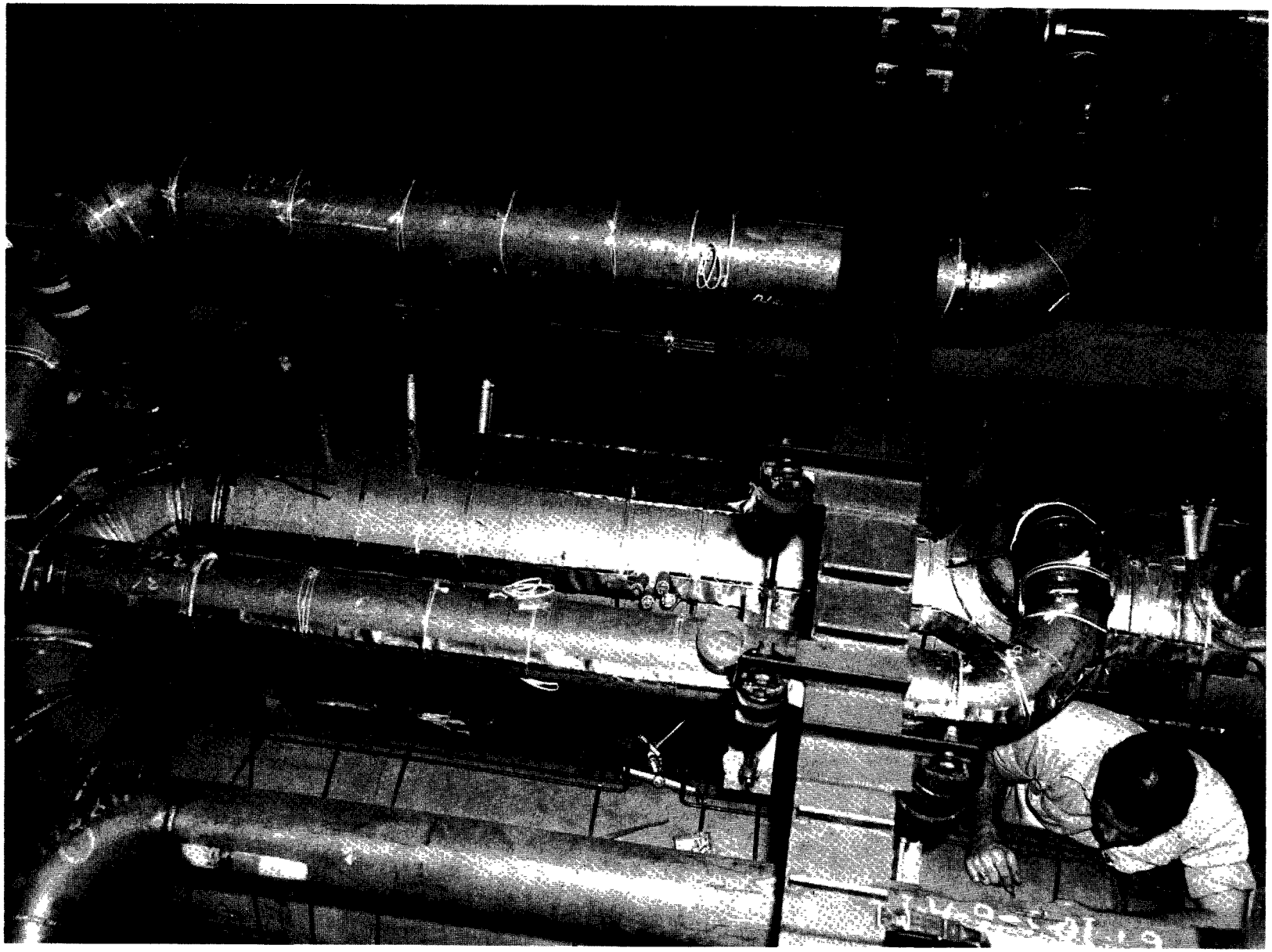

Fig. 20. Main Intermediate Heat Exchanger

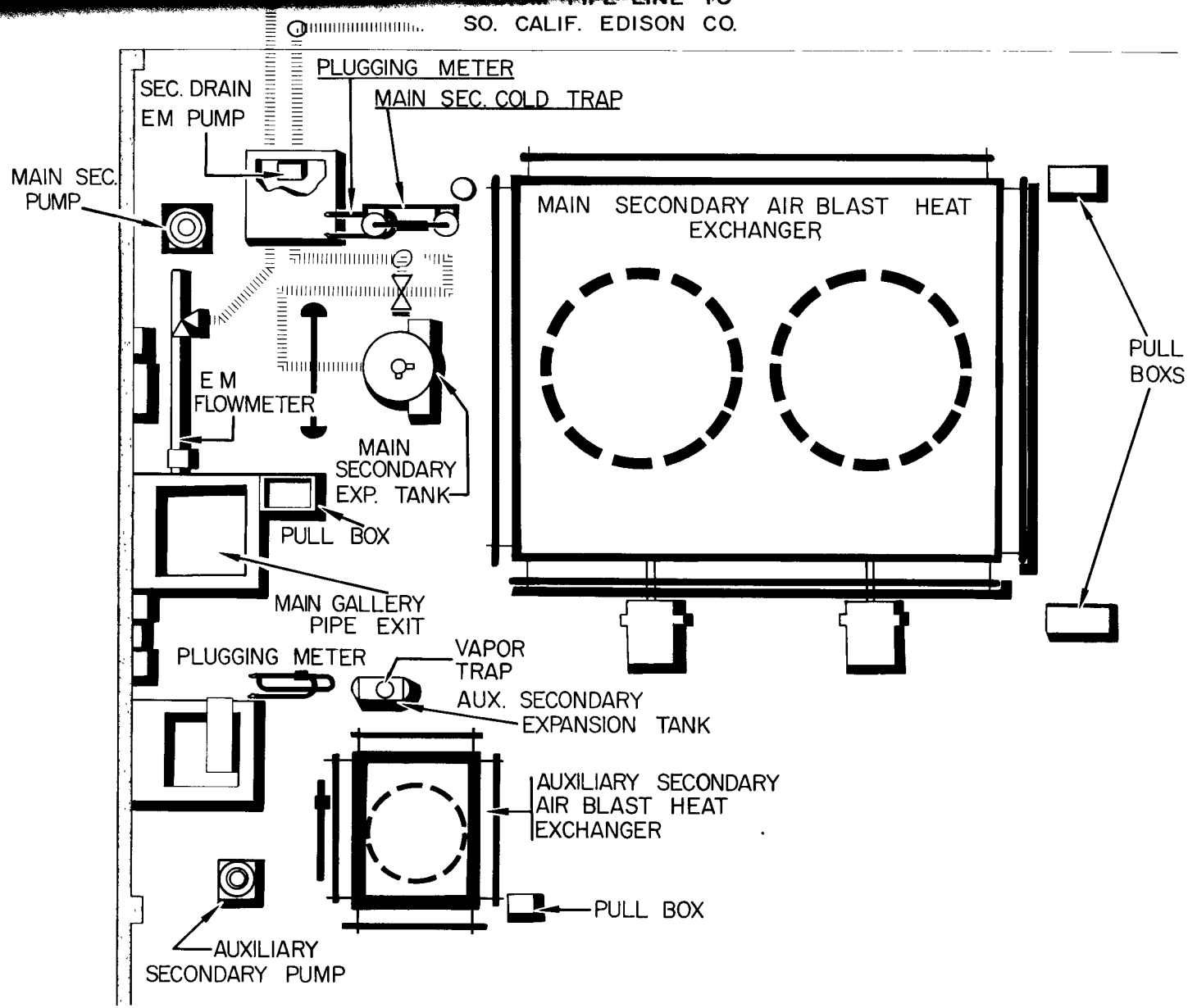

Fig. 21. Secondary Area

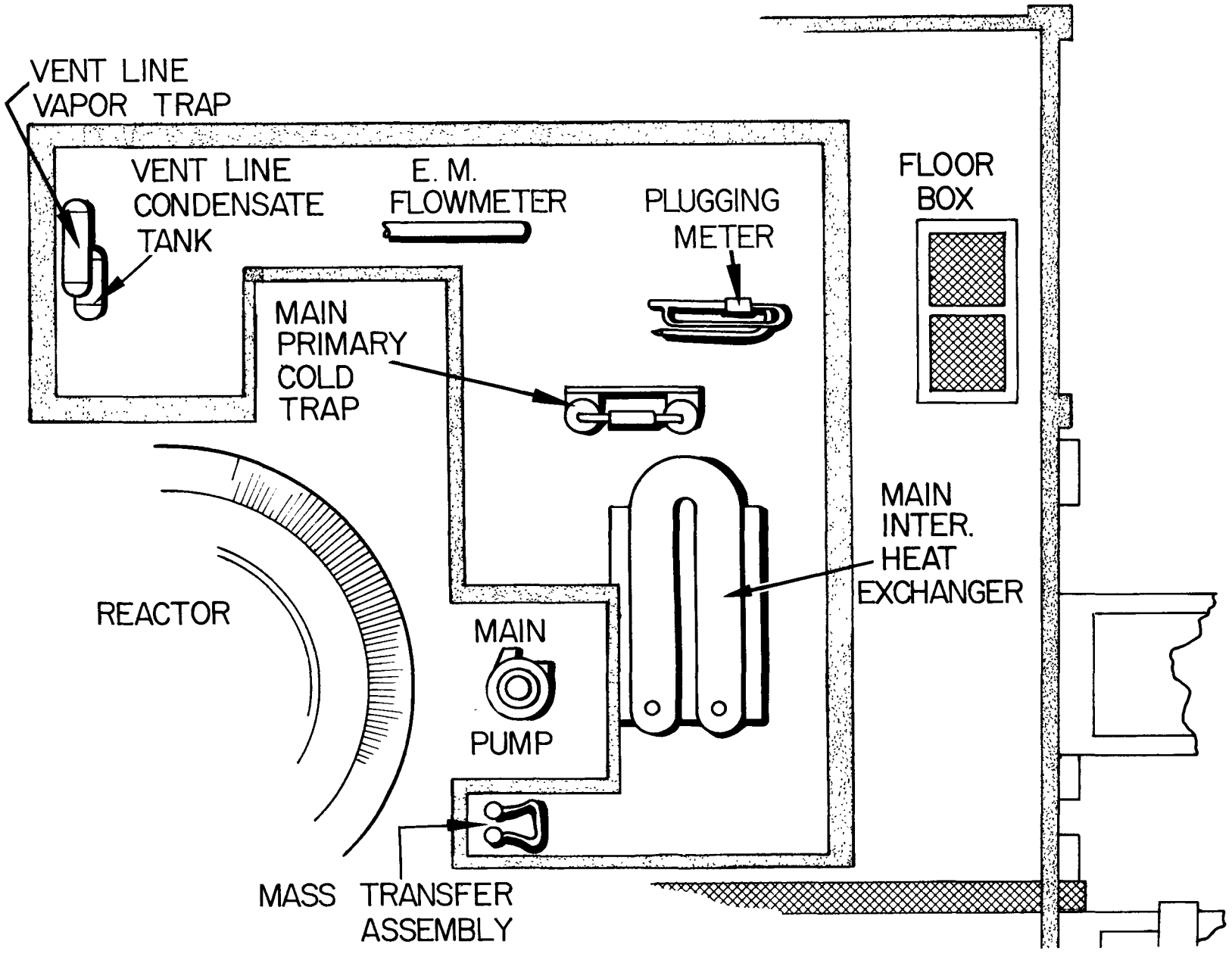

Fig. 22. Sodium Service Area

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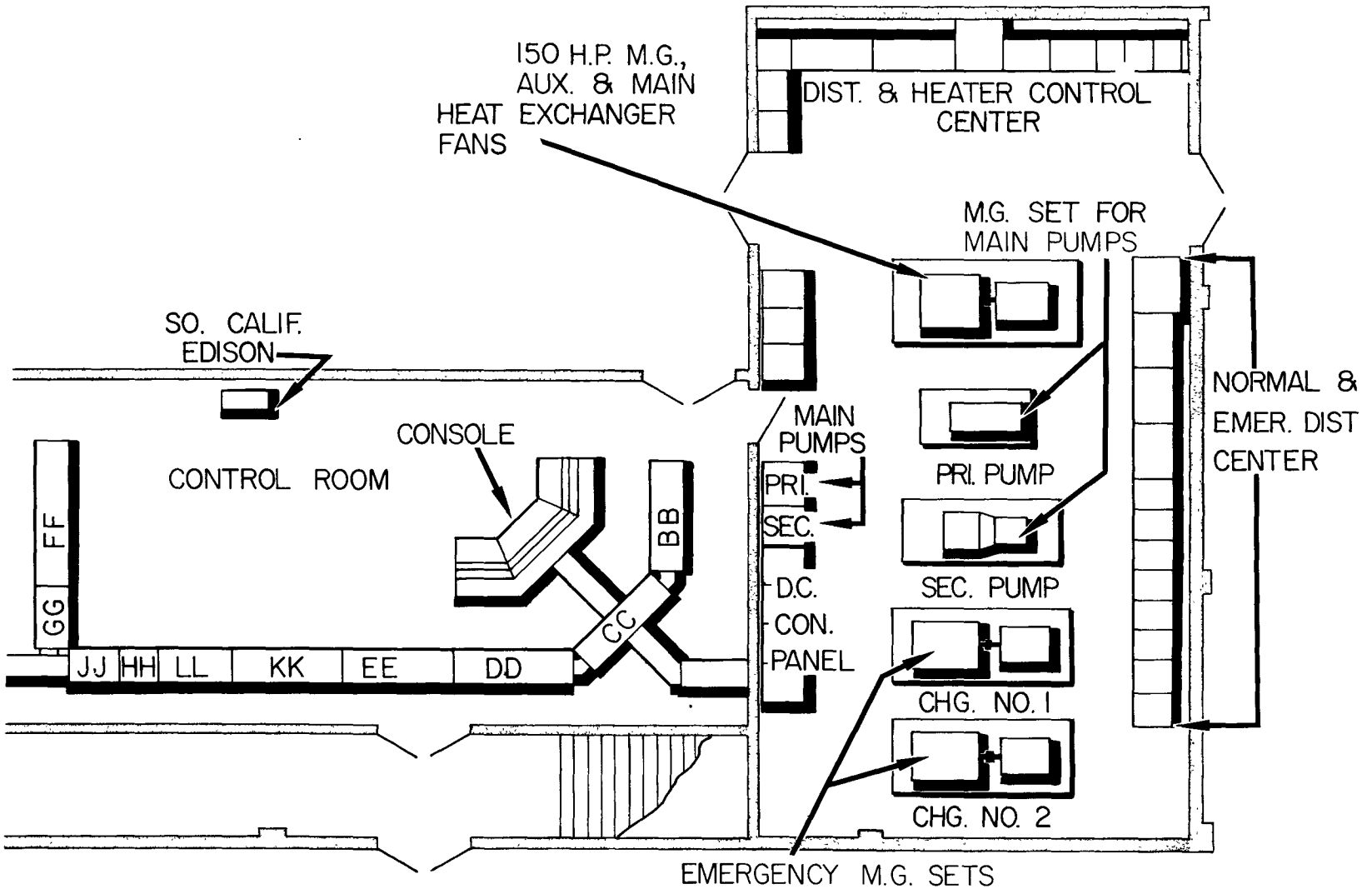

Fig. 23. Control Room and Motor Generator Room

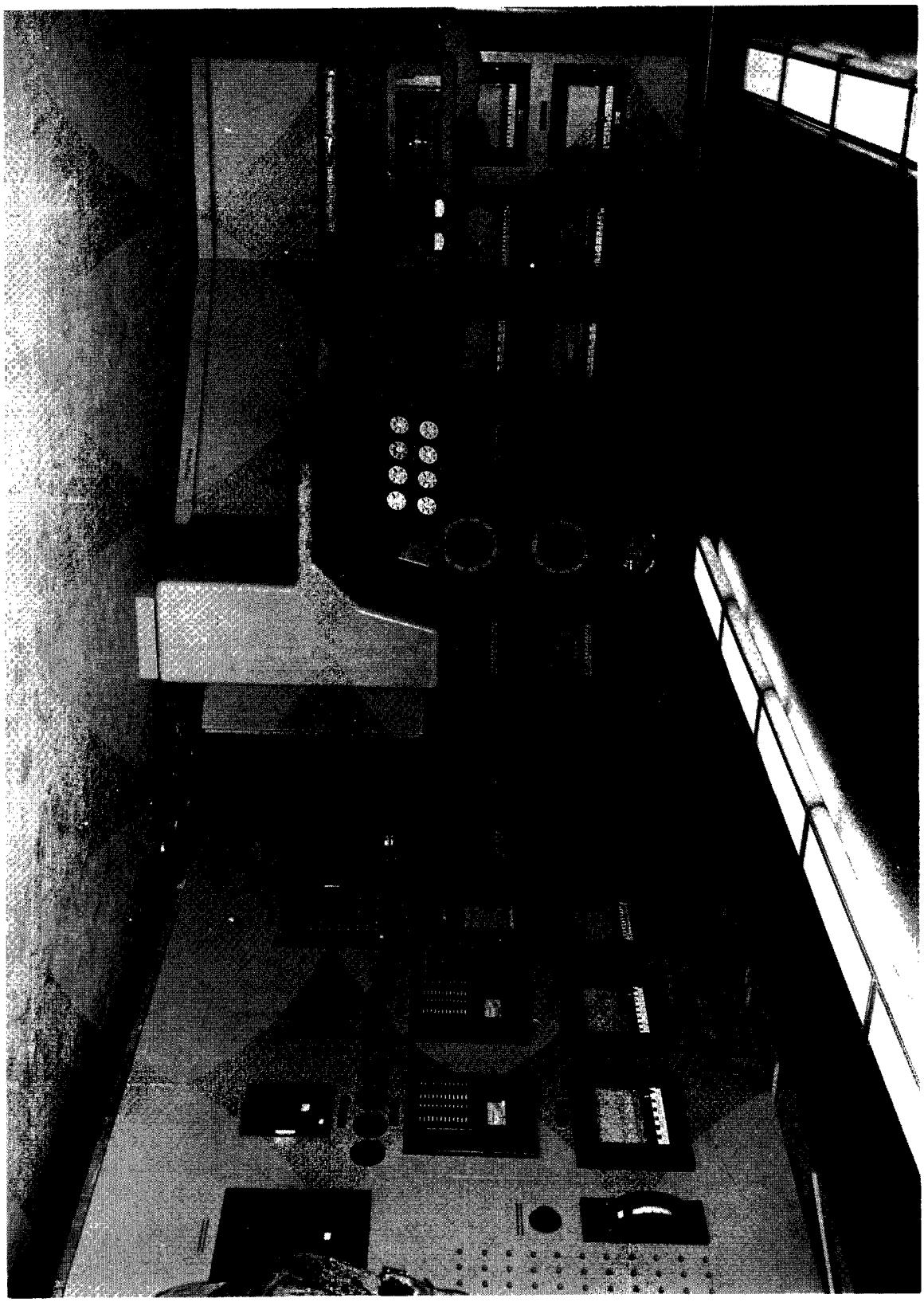

Fig. 24. Control Panel

SRE OPERATIONS

F. Faris

The manpower requirements and the procedures for the operation of the SRE are presented. The training program for operating personnel includes actual operations training, lectures on reactor technology and the SRE systems, and "on-the-job" training. A thorough pre-operational testing program extending over a period of approximately 3 months is being carried out prior to actual operation. The schedule for bringing the reactor to full power initially provides for operation at successively higher power and temperature levels, with tests and analyses of test data each time to ensure that the safety of the reactor will not be impaired upon proceeding to the next level. The normal startup and shutdown procedures, the procedures to be followed in case of a "scram", and the design of the control and safety instrumentation are based on systems analysis studies, the results of which are described.

Fig. 24. Control Panel

I. INTRODUCTION

The purpose of this section is to provide information on plans for the operation and testing of the Sodium Reactor Experiment. The beginning of a formal operations program for the reactor may be said to have started approximately two years ago with the establishment of the group which is responsible for the operation and the testing of the reactor. The group was formed early in order to provide an ample opportunity for contribution to the design of the reactor and for careful planning of the operating and experimental programs. Even before the formation of the responsible group, a considerable amount of systems analysis work, which is of course quite important for operational planning, had been performed as part of the engineering of the overall reactor system.

During the two years of the group's existence, the members have been participating in a reactor training program. At an early date, personnel received instruction regarding the SRE systems and pertinent aspects of reactor technology in two series of lectures, one planned primarily for the engineers and the other planned primarily for the technicians. A number of engineers also received practical experience in reactor operations as a result of training trips to other sites.

The most important phase of the training program is training received on the job. In addition to knowledge gained as a result of design review and operational planning, personnel who will be operating and testing the reactor will have learned a great deal about the SRE by assisting in some of the component development experiments and in the construction of the reactor. In the initial stages of the preoperational testing program, all members of the group were provided an opportunity to familiarize themselves with the various reactor areas by comparing the as-built systems with the blueprints. We consider that training must be a continuing process, and it is the responsibility of each supervisor to see that his men are continually trained on the job as work with the reactor progresses.

The personnel required for normal operation have been determined from studies of the location of instruments, valves, and other equipment in the reactor area and of the functions to be performed. On this basis, it has been estimated that four crews of six men each together with relief personnel will be required for continuous operation of the reactor. Each crew will be made up of a shift

supervisor, a chief reactor technician, two reactor technicians, an instrument man, and a health physicist. In addition to the operating crews there will be a number of engineers and technicians engaged in maintenance work, the testing program, and improvements and modifications of the reactor system.

II. PRE-CRITICAL TESTING PROGRAM

The pre-critical tests actually began during the construction phase. During this period, various items, such as the location of thermocouple and heater wires, which would later be inaccessible, were checked. Also checked during this period was the ease with which various pieces of equipment, which later might be relatively inaccessible because of radioactivity, could be maintained. Examples of this latter activity include a check on the location of piping in the coolant galleries to make sure that it would not interfere with accessibility to major pieces of equipment and the ease with which moderator cans and pump components could be removed.

The second series of tests, which are now under way, include the calibration of all instruments and the checking of all electrical, mechanical, and process systems.

Upon the completion of the tests with the system dry, tests will begin with the sodium systems filled. The systems will first be heated to 350° F. Sodium will then be introduced and circulated for a sufficient period of time to establish confidence in the pumps and the other components of the sodium systems. During the functional tests, the sodium will be maintained by heaters at a temperature of approximately 700° F, which is close to the average temperature expected for the sodium in the reactor tank during normal operation. While hot sodium is in the core, the operation of the control and safety systems will be carefully checked. At the completion of the circulating sodium tests, the reactor and the secondary drain systems will be checked to make sure that they are functioning properly. The reactor will then be ready for the critical experiment.

III. CRITICAL EXPERIMENT AND LOW POWER PHYSICS EXPERIMENTS

As a result of the rather large difference which has been calculated for the critical loading with the system dry (approximately 14 fuel elements) as compared to the critical loading with sodium in the core (approximately 24 fuel elements), it has been decided to dispense with a dry critical experiment. There is planned instead a subcritical loading with the system dry, the results of which may be compared with data from the wet critical experiment in order to estimate experimentally the effect of the sodium on the number of fuel elements require for criticality. The subcritical work will proceed by the symmetrical addition of fuel elements from the center outward until a multiplication (the ratio of the count rate from the neutron detectors with fuel present to the count rate with only the neutron source in the core) of approximately 25 is achieved. It is planned to perform the dry subcritical experiment with the core heated to approximately 350° F with electrical heaters.

The wet critical experiment will also be carried out at approximately 350° F. During the wet critical experimental and also during the dry subcritical loading previously described, the four safety rods will be in their upper position and will be cocked ready for insertion if the occasion demands it. All four control rods will be completely inserted whenever fuel is being added. A measurement of the multiplication after the insertion of a given amount of fuel will be made with three control rods fully withdrawn and with the fourth control rod in various positions including full withdrawal from the core. When measurements made during the previous additions of fuel indicate that the last fuel element required for criticality has been added, the control rods will be slowly withdrawn, and multiplication measurements will be made at frequent intervals during the withdrawal of the last rod until criticality is achieved.

After criticality has been achieved, a series of measurements at low power levels will be made. Preliminary control rod calibrations will first be made. The additional fuel elements required for normal operation will then be added, one at a time, with multiplication measurements at appropriate control rod positions. After all of the fuel elements have been added, data on the following subject will be obtained:

1. Calibration of the control rods
2. Effectiveness of the safety rods

3. Neutron flux as a function of position in the core, the reflector, and the shield adjacent to the reflector.
4. The effectiveness of fuel elements in various core positions
5. The effectiveness of different kinds of experimental fuel elements
6. The effect of temperature (at core equilibrium) on the position of the control rods (i.e., a measurement of the temperature coefficient of reactivity).

IV. STARTUP AND NORMAL OPERATION OF THE REACTOR

Upon completion of the critical experiment and the low-power physics measurements, the reactor will gradually be brought to full-power operation. During this period, there will be a series of tests at successively higher power levels to provide more information about the reactor and to gain confidence before proceeding to the next higher level. At each level, particular attention will be given to the collection of data on temperature distribution, from which thermal stresses will be determined before advancing to higher power levels and larger temperature gradients.

A tentative program for the normal startup of the reactor during this period is indicated in Fig. 1. During the normal startup, and also during the normal shutdown of the reactor, the rate of change of temperature will be maintained below 30° F/hr, which is a conservative value designed to keep thermal stresses in the system at a low value. After the reactor system has been heated to approximately 350° F and the safety rods have been withdrawn, the reactor power will be increased to ~2 per cent of its full value by withdrawal of the control rods. During this time, the sodium flow will be maintained at 100 per cent in all loops, a procedure designed to provide maximum safety during the initial phase of starting up the reactor. The main airblast heat exchanger fans will be turned on, and the motor operated louvres will be opened by steps.

The electrical heaters will be turned off until the cooling achieved at the airblast heat exchanger is increased to take care of the 2 per cent power generated in the reactor, and equilibrium is established. The reactor power will then be increased in small steps until all of the electrical heaters are off and the louvres are open. The fans will still be operating at minimum speed. The reactor power

will be slowly increased until the outlet sodium temperature from the airblast heat exchanger reaches 440° F, at which point the main airblast heat exchanger temperature controller will be turned on to provide automatic control of this outlet sodium temperature. The reactor power will be held constant at this point and the sodium flow will be reduced to approximately 15 per cent of its full powe value. The temperature gradient across the reactor core will then be establishe by increasing the power to approximately 15 per cent. When the outlet sodium temperature from the reactor core reaches 960° F, the thermal power of the reactor and the sodium flow will be alternately increased until such time as the desired operating power is established. Once this power has been achieved, the reactor may be placed on automatic control, so that the control rods will be mai tained at the position required to hold power constant. During the entire proced the auxiliary system will be adjusted to keep the temperature of the sodium retur ing from it to the reactor within 50° F of the temperature of the sodium returnin from the main system.

Scram, setback, and alarm circuits are provided to protect the reactor against unusual incidents which might damage it. Many of the systems analysis studies were directed at determining the time delays which could be tolerated in these circuits. As an example, Fig. 2 shows the effect on the outlet temperatur at the fuel channels of a delay in the scramming of the pumps after the reactor power is suddenly reduced. It is desirable to minimize rapid temperature chang in the fuel channels because of the thermal stresses produced in the moderator can heads when a difference in temperature exists between the channels and the periphery of the cans. On the basis of this work, it was decided to arrange for a pump scram which would be very nearly simultaneous with the drop of the safet rods. Figure 3 shows the conditions which will give rise to a scram with four safety rods dropped, a scram with one safety rod dropped, and a setback with th control rods motorized into the core. The set points given in the figure are thos for the anticipated normal operating conditions. The decision to have a one-rod scram in addition to a four-rod scram is based upon systems analysis studies wh indicated that the rate of change of temperature at the outlet of the fuel channels would be considerably less as a result of the slower rate of power decay. The four-rod scram is thus reserved only for those conditions under which it is neces sary to reduce power at the maximum possible rate of speed in order to protect

the reactor. Figure 4 summarizes the results of the studies of the temperature transients following the two different scram conditions. In the figure a reactivity change (δk) of 10 per cent corresponds to the simultaneous dropping of four rods, and a reactivity change of 2 per cent approximates the dropping of one rod.

Also shown in Fig. 4 is a comparison of the effect of final flow on the fuel-channel outlet temperature following a scram. The thermal driving head in the main system plus the pump in the auxiliary system provide a flow approximately 15 per cent of that during full-power operation, whereas the thermal driving head alone provides a flow approximately 10 per cent of the full-power value. If the results of these studies are borne out by tests on the reactor, it will probably be decided to set the valves in the auxiliary system for a very low sodium flow during normal operation. If anything were to happen to the main system in the case of such operation, the valves in the auxiliary system could be opened in sufficient time to take care of after-glow heat in an adequate fashion.

V. POST-CRITICAL TESTING PROGRAM

After the reactor is operating satisfactorily at a relatively high power level, additional measurements relating to the physics of the reactor (e.g., the effect of xenon and temperature on reactivity) will be made; and a variety of engineering experiments (e.g., heat balances and studies of system temperature changes in response to mild transients) will be performed.

Many of the tests which will be performed after a reasonable thermal gradient is established in the reactor core will be directed toward determining the extent to which the predictions of the systems analysis studies are valid. A number of assumptions (for example, the extent of mixing in the upper plenum of the reactor and the effect of certain sodium-system components on flow decay) were of necessity employed in the studies. In many cases, analyses were performed for several different assumed values of a particular parameter. The results, of course, are no better than the assumptions themselves.

Let us now look at a few typical results of the systems analysis work and at the experiments they suggest. In Fig. 5 we see a comparison of flow decay and power decay for the case of a four-rod scram with the valves in the auxiliary loop

completely open. Temperature transients following a scram are markedly dependent on the extent to which flow decay matches power decay.

Unfortunately, there is some uncertainty in the predicted value of flow deca and the measurement of flow as a function of time after shutting off the pumps wi be one of the early experiments. Based upon the flow decay expected from the thermal driving head in the main system only, and based upon the assumption of perfect mixing in the upper reactor plenum, we find in Fig. 6 various system temperatures predicted for the reactor following a scram. The temperature transient indicated here for the fuel channel is somewhat milder than that indicated in Fig. 4 because an estimate of the effect of heat stored in the graphite has been included. After the reactor reaches a reasonable power level, but befo the level is so high that thermal stresses are likely to be of serious concern, system temperatures such as those shown in the figure will be measured followin a scram. Some of the temperatures are quite sensitive to the degree of mixing in the upper plenum, and we are particularly interested in determining the extent of the thermal inertial provided by the pool in the upper plenum.

VI. REMARKS

We certainly hope that our experiments will show that we may proceed to full power without any particular concern regarding thermal stresses following a reactor scram. If the results indicate that such stresses would be a problem, corrective action to minimize them can be taken. Typical of the sort of thing which it is possible to do, if necessary, is the addition of a sodium throttling valve in the main primary system. Such a valve could be controlled automatically so that the sodium flow decay would match very closely the power decay.

Routine operation will be carried out simultaneously with the post-critical tests, and a definite attempt will be made to separate clearly the costs associated with the testing program from those associated with the operation of the reactor. In this way, we hope to obtain a good basis for estimating the cost of operating and maintaining a full scale sodium-graphite reactor plant.

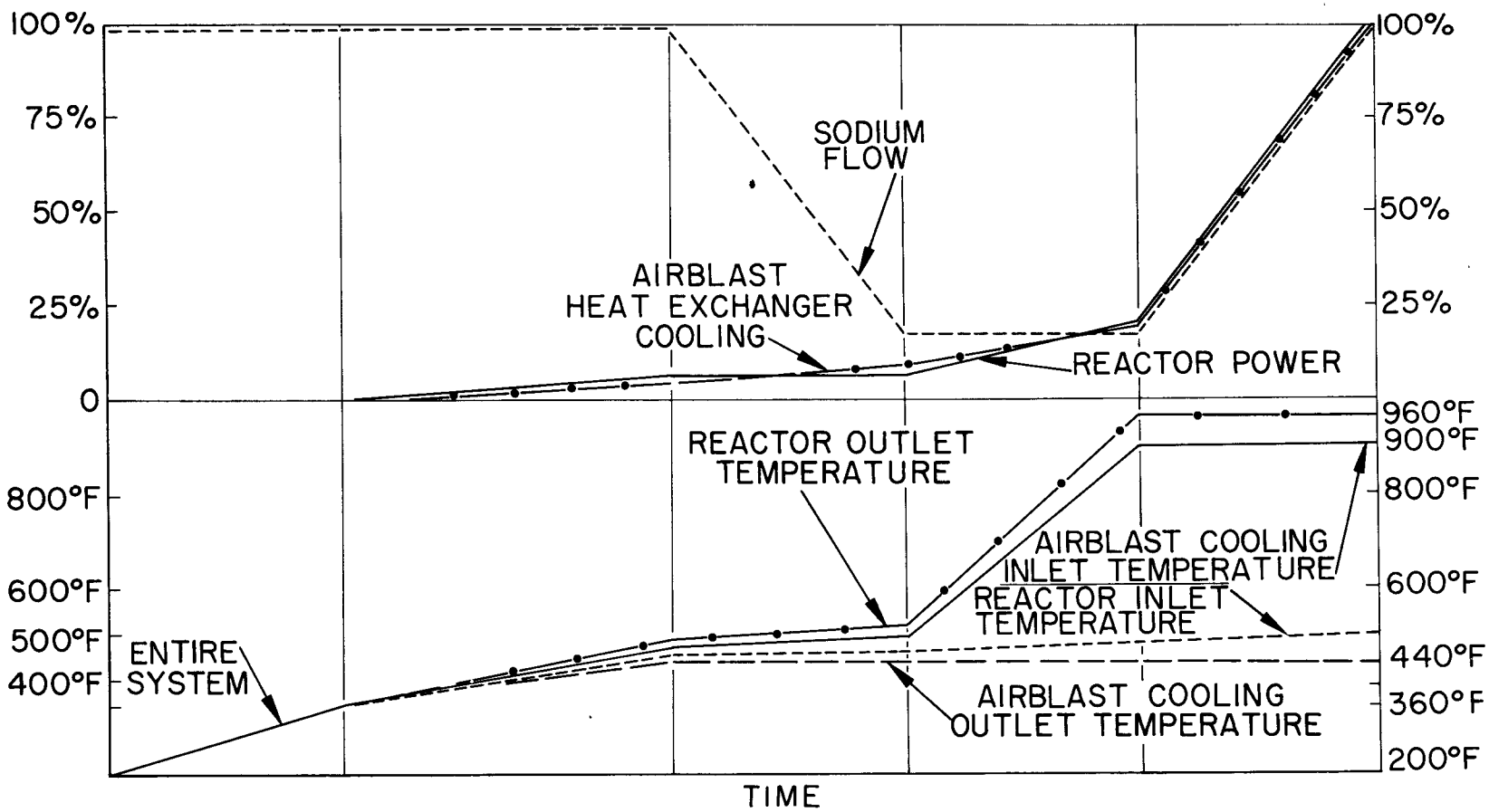

Fig. 1. Programming of the Startup of the SRE

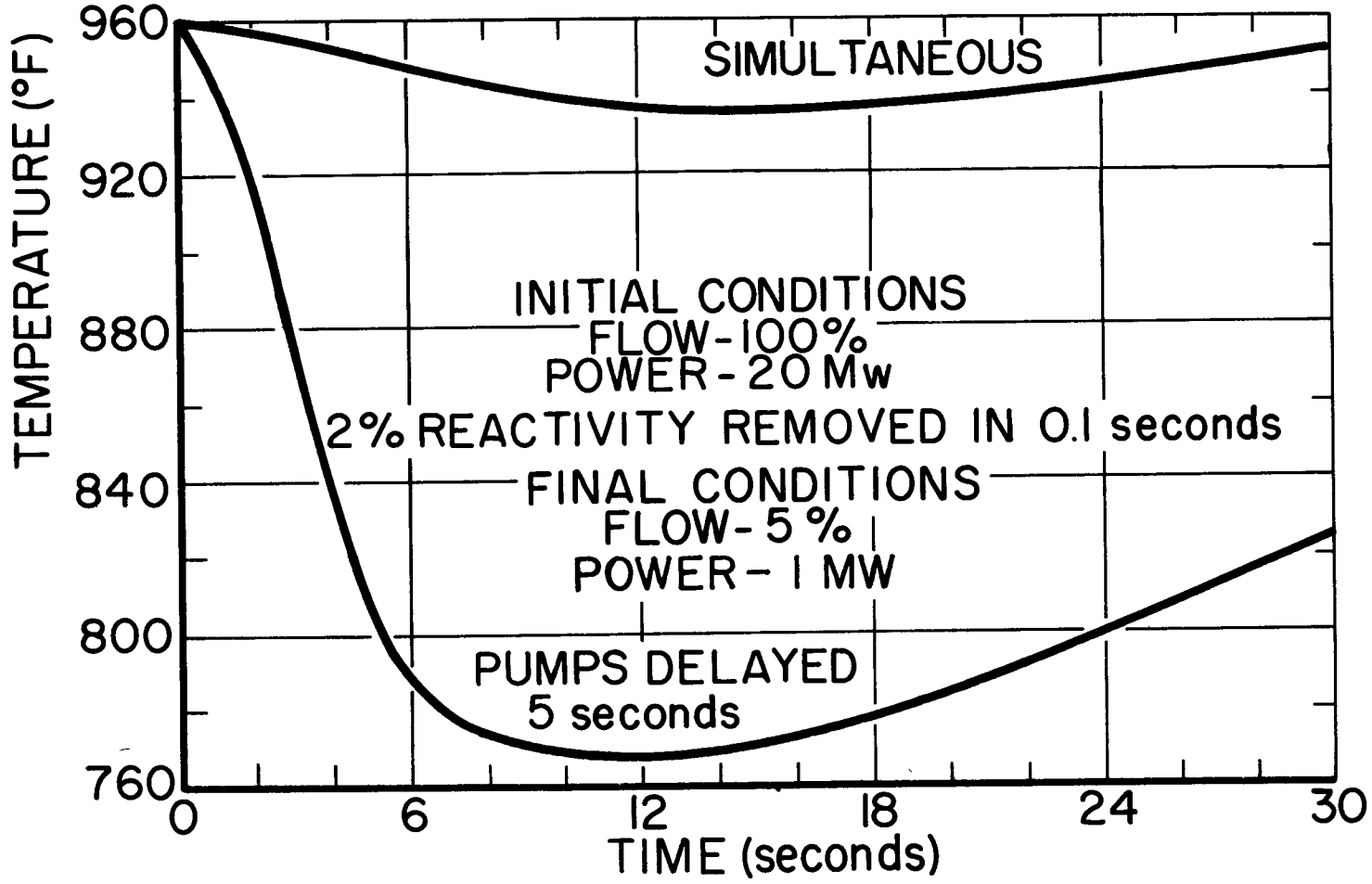

Fig. 2. The Effect of Pump Delay Time on Fuel Channel Outlet Temperatures Following a Scram

MANUAL	X		
EARTHQUAKE	X		
ELECTRICAL POWER FAILURE		AFTER 2 SEC. DELAY	
REACTOR PERIOD	< 10 SEC.		< 15 SEC.
NEUTRON LEVEL	> 125% NF		> 115% NF
MAIN PRIMARY SODIUM FLOW		< 90% SET FLOW RATE	
MAIN SECONDARY SODIUM FLOW		< 90% SET FLOW RATE	
FUEL CHANNEL TEMPERATURE	> 1050°F	> 1040°F	> 1030°F
LOSS OF THE MAIN AIRBLAST FANS		< 85% SET FAN SPEED	
LOSS OF EITHER AIRBLAST FAN		< 80% SET FAN SPEED	
LOSS OF FEEDWATER		< 600 psi PLUS 8 SECOND DELAY	
MAIN PRIMARY COLD LEG TEMPERATURE		> 600° F	
MAIN SECONDARY COLD LEG TEMPERATURE		> 550° F	

Fig 3. Conditions Giving Rise to Scrams

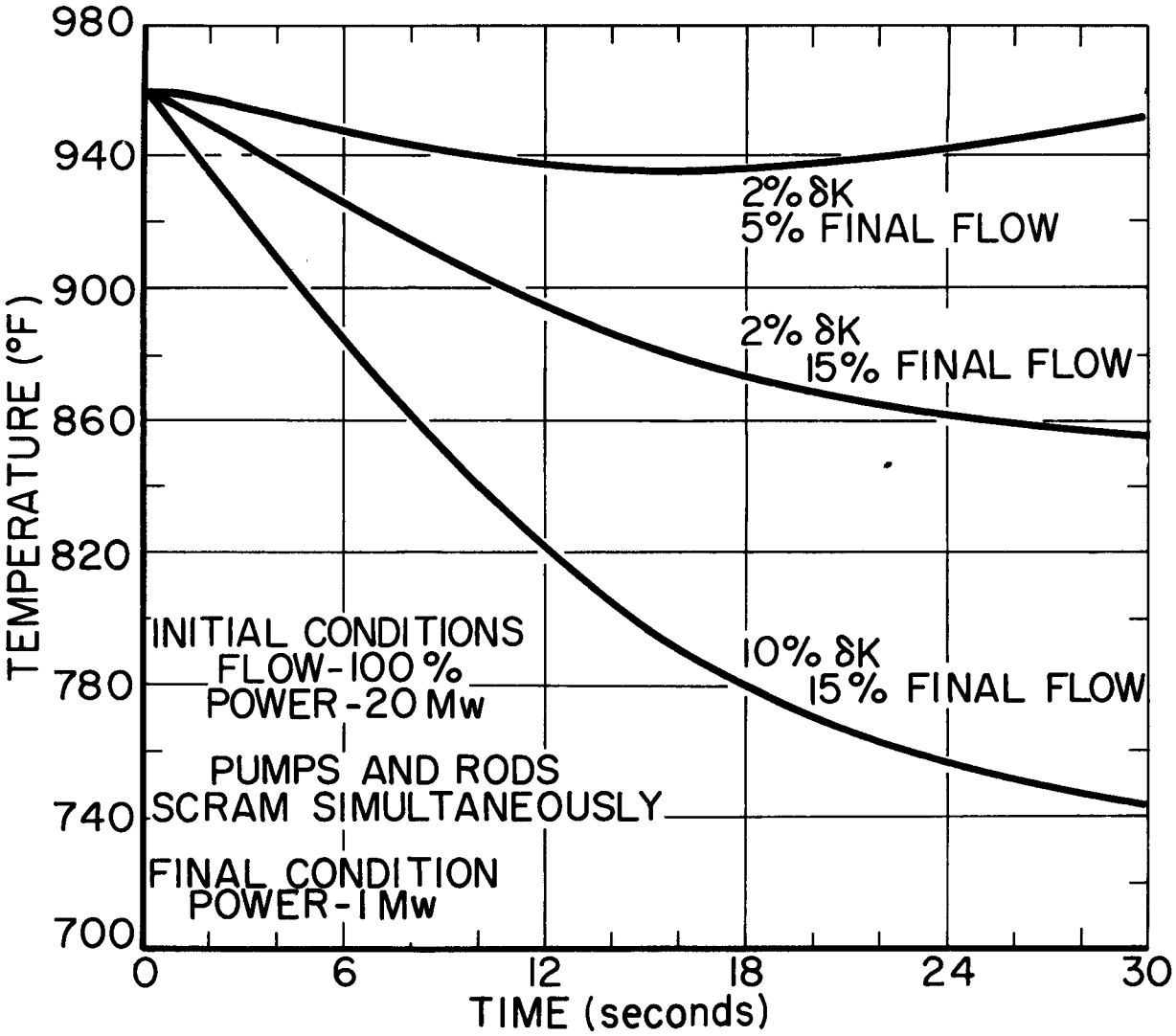

Fig. 4. Fuel Channel Outlet Temperatures Following a Scram With the Amount of Reactivity Removed and the Final Sodium Flow at Parameters

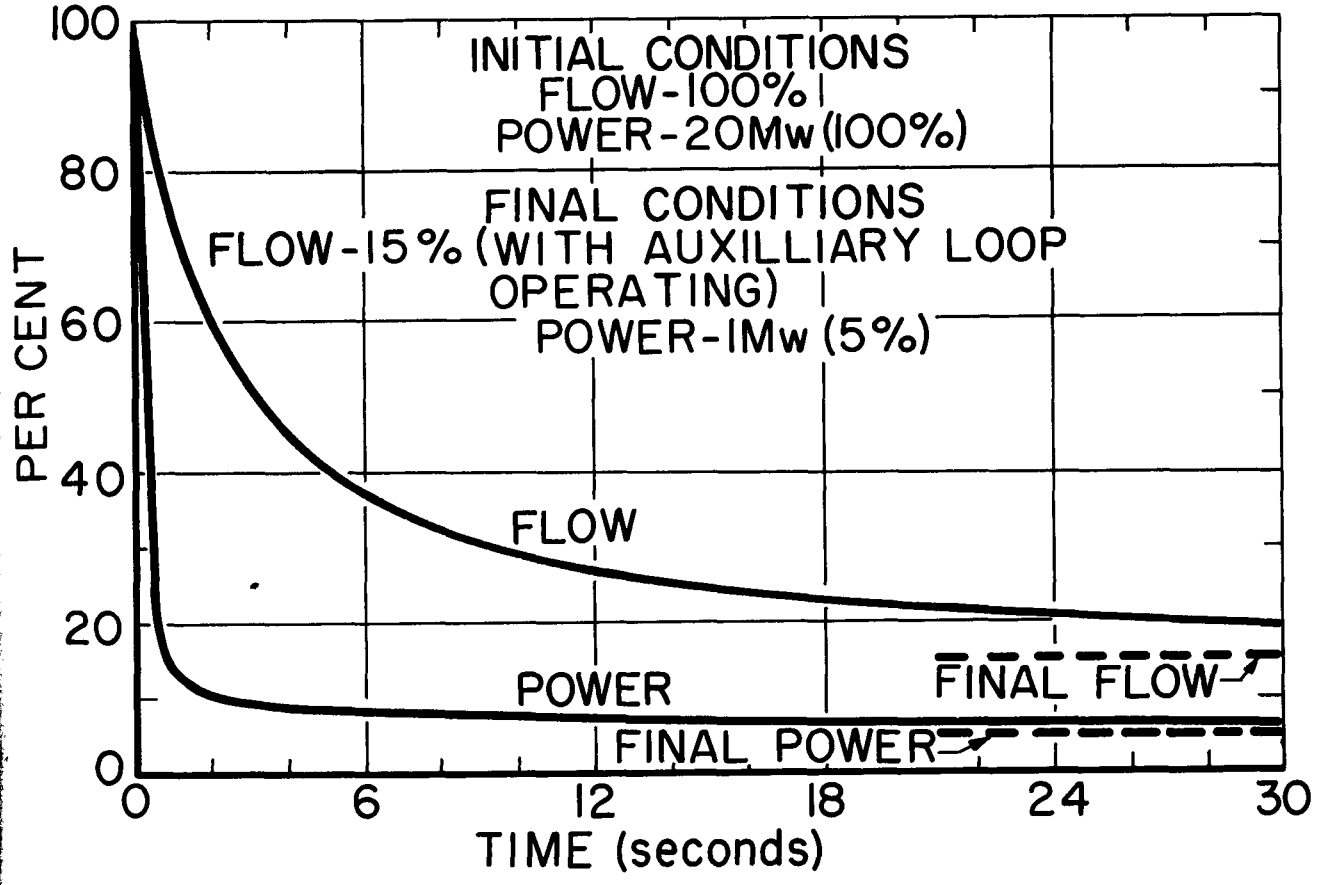

Fig. 5. Comparison of Power Decay and Flow Decay Following a Scram

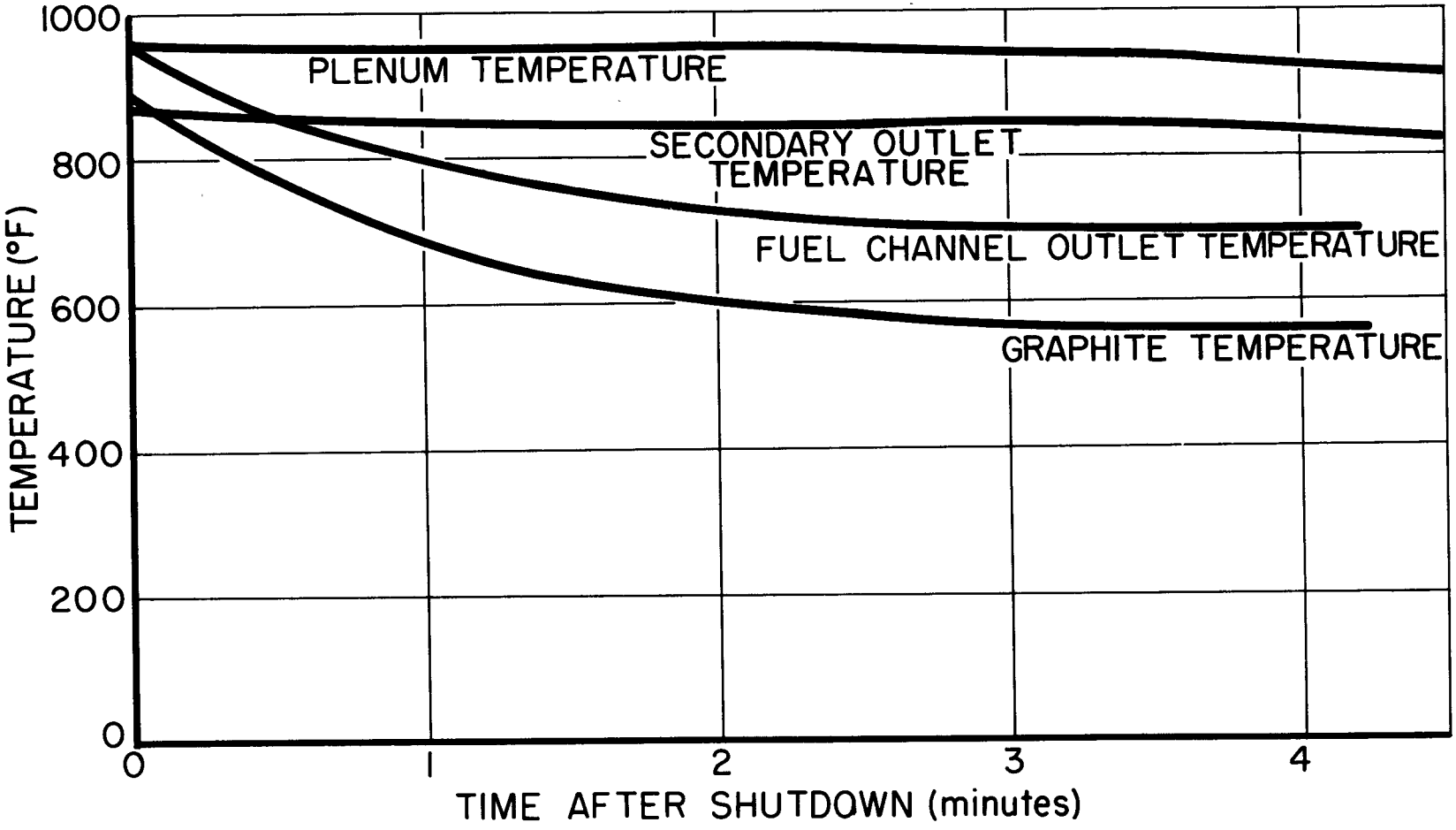

Fig. 6. System Temperatures Following a Scram

THE CALIFORNIA EDISON COMPANY STEAM ELECTRIC PLANT

A. C. Werden, Jr.*

The Southern California Edison Company is planning to install and operate, ~~at its sole expense~~, an experimental 7500-kw steam-electric plant, for the purpose of utilizing the heat developed by the SRE.

The estimated cost of the installation is $1,200,000. Surplus heat ~~will be~~ purchased at 45¢ per million Btu. Energy from the plant ~~will be~~ distributed to the customers of the Company, now ~~totaling~~ nearly 1,400,000, on a system-wide ~~basis~~.

Of major importance ~~will~~ be the performance of the once-through steam generator. This liquid metal-to-water boiler is designed for a dual sodium inlet rating of 900° F and 1140° F.

Initially, controls ~~will~~ be connected such that the steam plant will ~~follow~~ the reactor output.

Edison's objective in cooperating with Atomics International and the Atomic Energy Commission in this venture ~~is~~ to develop basic knowledge in the use of nuclear energy for the generation of electric power.

*Southern California Edison Company

I. INTRODUCTION

The Southern California Edison Company, like most electric utilities, is keenly interested in the use of nuclear energy for the generation of electric pow Consequently, we are appreciative of the opportunity to participate here with th SRE and for the privilege of presenting our plans and objectives at this meeting today.

As many of you perhaps know, the Edison Company has nearly completed the installation of a 7,500 kw steam-electric generating station located in the Santa Susana Mountains adjacent to the SRE. The SRE, or Sodium Reactor Expe ment, is part of the Atomic Energy Commission's Nuclear Power Development Program. The nuclear reactor for this project is being constructed by Atomics International, a division of North American Aviation, Inc. The function of this Experimental Station, as it is called, will be to convert the SRE heat energy into electric energy for distribution over the Company's system.

Figure 1 is a general view of the Edison steam plant with the reactor show in the background. The Experimental Station is a modern outdoor-type plant. It is conventional in all respects consistent with a reactor heat source.

The control room building is seen in the foreground. To the right is the steam generator. Directly behind the control house is the turbine generator and condenser. Further on to the left is the metal-clad switch gear, power transformer, and 69 kv take-off tower. In the left foreground is the water demineralizer equipment.

The installation is estimated to cost $1,200,000, all of which is the sole expense of the Edison Company. In addition, the Company will pay 45¢ per million Btu for heat energy that is used to generate electric energy.

From a public interest point of view, this installation will be the first nuclear-heat plant in the United States to generate commercial electric energy from a nonmilitary reactor and it will bring the first nuclear-generated energy to the West Coast.

Recognizing that this kind of installation could not be an economical power supply under experimental conditions, the Company's objective was to cooperate with our government in the research, experimentation, and development of low-cost nuclear power.

Further, it affords an opportunity to join with North American Aviation, Inc., a customer and neighboring industrial organization, in a cooperative project seeking basic knowledge on the use of atomic energy for power generation.

Technically the Company was keenly interested in the desirable characteristics of the sodium-cooled reactor, particularly as it afforded temperature and pressure throttle conditions closer to those currently in use in modern fossil-fuel burning plants than many other reactor types. Of further technical interest was the opportunity to test a promising reactor type and an associated liquid metal-to-water heat exchanger on a pilot plant basis, and, hence, minimizing the costs from obsolescence incurred by prototype experimentation.

Because continuity of operation is a rather important and desirable factor in the reactor experiment, the steam plant facilities were designed to operate with a reliability comparable with that of a major conventional steam plant. Obviously, on a system such as Edison's, now totaling nearly 2-1/2 million kw of generation, the addition of 7,500 kw of interruptible capacity is a small fraction of one per cent and its interruption not significant. Likewise, the energy output, estimated to be in the order of 50 million kwh for the contract period, will be a relatively small part of the system requirement. In transmitting this block of colorful energy to 1,400,000 customers located in ten counties, it would hardly amount to more than one day's use. On the other hand, if this nuclear-generated energy could be isolated, it would serve one of the picturesque, smaller communities, such as Santa Susana, which is located in the Simi Valley to the west.

II. GENERAL DESCRIPTION OF SYSTEM

Figure 2 indicates the manner in which the plant equipment will be connected to the reactor heat source. This figure is a schematic diagram of the SRE cooling system. The radioactive sodium in the main primary system transfers its heat to the non-radioactive sodium in the main secondary system by means of the intermediate heat exchanger. The non-radioactive sodium than is circulated through the shell side of the Edison liquid metal-to-water heat exchanger (or steam generator, as we usually refer to it).

Figure 3 shows the basic flow diagram of the steam plant thermal cycle. The two lines entering from the left represent the sodium piping which conveys

the non-radioactive sodium to and from the steam generator. These are 6 inch, stainless-steel pipe lines, traced with calrod electric heating units. The sodium, in transferring its heat to the water, enters at 900° F and leaves at 440° F. At rated load of 20 Mw of heat, the flow rate will be 485,000 lbs per hr. The steam generator, in conjunction with its attemperator, will yield 620 psi steam, superheated to 825° F, to the turbine throttle.

There are two lines tapped to the main steam header. One supplies a steam jet air ejector. The second passes through a pressure reducer and desuper heater directly to the condenser. This bypass arrangement around the turbine will facilitate stabilization of steam conditions with the reactor prior to admitting steam to the turbine. It will also serve to accommodate reactor operations at heat outputs too low for satisfactory turbine operation. The bypass will be automatically controlled and is designed to route up to 5 per cent of full load steam directly into the condenser.

The main steam turbine is a 3,600 rpm unit with a guaranteed heat rate of 11,544 Btu per kwh at full load. It has two stages of extraction for feed water heating. The first stage is used in the closed-feed water heater. The second stage extraction is utilized in a Cockrane Tray Type Deäerating Heater. Although this equipment serves as a feed water heater, its primary function is to release the oxygen content in the condensate. The specifications for this equipment required a manufacturer's guarantee that the oxygen content of the effluent will not exceed 0.005 parts per million.

Returning again to the main steam flow, the turbine exhausts into a 7,000 sq ft, two-pass, surface condenser. This condenser is supplied with water at 75° F by twin circulating pumps from an induced-draft cooling tower. The condensate is pumped from the hot well by dual-demineralized condensate pumps through the steam-jet air ejector and deäerator. Dual-feed water pumps then pump the condensate back through the closed-feed water heater to the steam generator at a temperature of 297° F.

III. DESCRIPTION OF STEAM PLANT COMPONENTS

Figure 4 is a closeup view of the major steam plant components. It was taken prior to the installation of the interconnecting sodium piping and thermal insulation

The steam generator again is in the right foreground. Connected to its right-hand leg is an attemperator. Between the two legs is the mercury expansion tank. The elevation of the steam generator is determined by the gradient requirement of the interconnecting sodium piping.

Of vital importance and particular interest in the operation of the steam plant will be the performance of this once-through steam generator furnished by Babcock and Wilcox Company. To the best of our knowledge, it is the first of this type ever to be placed in commercial operation in the United States. Not only will this piece of equipment perform the duties of the conventional economizer, evaporator, and superheater, but in addition, it is designed to accomplish this under varying heat input temperatures. In order to meet certain elevated experimental temperature conditions imposed by the reactor, the design of the steam generator required that it be capable of supplying constant temperature steam over a range of inlet sodium temperatures of from 900° F to 1140° F and for a corresponding range of outlet sodium temperatures of from 440° F to 670° F.

Figure 5 shows a sketch of the steam generator and a simplified cutaway view. Connected to and shown immediately above the steam generator is the attemperator and mercury expansion chamber. The steam generator is an all-stainless-steel heat exchanger formed in the shape of a "U." Its developed length is approximately 80 feet. The tubes are double-wall construction with sodium on the shell side and water flashing to steam in the inner tubes. The inner tubes are 1/2 inch and the outer ones 11/16 inch in diameter. The annular volume between the tubes is filled with mercury pressurized at 190 psi. It thus serves as a pressure monitoring fluid for the purpose of detecting tube failures. Approximately 5 tons of mercury were required. In the event of a tube failure indication, provisions have been made to positively identify the faulty tube prior to removing the exchanger heads. The means for doing so will be of interest.

Figure 6 illustrates an end view of the tube assembly and construction at the inner tube sheet. If an outer tube failure is suspected, positive determination and tube identification can be made, after draining and cooling, by pressurizing the shell side with helium gas and then "sniffing" between tube walls at the inner tube sheet. The sniffing is accomplished with a probe inserted through removable plugs which can be noted around the tube sheet periphery. Each hole is indexed with a definite row of tubes. If an inner tube failure is suspected, similar

investigation can be made at the outer tube sheet through a 4-1/2-inch removable plug in the end of the exchanger head.

Figure 7 is a construction view of the tube bundle with the shell removed. Prior to assembly, tubes with the slightest wall defects were eliminated by ultrasonic tests. During fabrication, all welds were zyglo-tested or radiographed. Finally, a mass spectrometer leak test was made with helium gas, followed by a hydrostatic test. In order to protect the tubes against stress corrosion difficulties during operation, a high degree of water purity has been specified as follows:

1. The water pH value shall be 9.5 to 9.6
2. Iron content shall not exceed 0.01 ppm
3. Total solids shall not exceed 0.5 ppm
4. Free oxygen shall not exceed 0.005 ppm

To meet these conditions, twin mixed-bed demineralizers, each with a capacity equal to 30 per cent of the full condensate flow, have been installed. They are designed to maintain total solids at or below 0.1 ppm. This equipment will be connected to polish a percentage of the condensate flow as required to maintain the specified water purity. Also, as previously mentioned, an aereator will hold the oxygen content to low values, in the order of 0.005 ppm.

In order to minimize thermal shocks in the steam generator, normal operating practice will limit temperature changes to a rate of 30° F per hour.

IV. OPERATION OF STEAM PLANT

During start-up operations, the steam generator will be preheated to 350° F by means of an auxiliary electric water heater. Sodium at 350° F will next be admitted and gradually raised to 440° F. As the inlet sodium is then gradually increased on up to 900° F, feed water will be admitted as necessary to hold sodium outlet temperature at 440° F.

In order to secure a constant steam temperature at the throttle, an attemperator was installed. It is designed to maintain a constant throttle temperature of 825° F by introducing feed water into the steam through a venturi section. When operating at a heat load of 100 million Btu/hr and with sodium entering at

900° F and leaving at 440° F, this evaporator, together with the attemperator, will generate 88,700 lbs per hr of steam at a pressure of 620 psi and 825° F. It is anticipated that this steam flow will be realized with no attemperation.

When operating at the higher sodium temperature levels, the steam flow conditions to the throttle will remain the same except that about 11,000 lbs per hr will be introduced through the attemperator while generating 77,700 lbs per hr of steam in the evaporator.

V. CONTROL AND INSTRUMENTATION

The steam plant facilities will be operated from the Edison control room, which is located remotely from the reactor control room. In general, the controls are typical of a modern, conventional plant. The thermal cycle equipment and instrumentation are mostly pneumatically operated. They are served by one 77-cfm station, air compressor, and a second 10-cfm instrument air compressor at 100 psi. The electrical equipment is operated and controlled from the station service bus and a 130-volt battery.

Due to the fact that the primary purpose of the SRE is to develop basic reactor technology, the controls are connected initially to cause the steam plant to follow the reactor output, similar to a stream-flow hydro plant. Accordingly, the steam plant control system is designed to remove heat automatically from the secondary sodium loop as required in order to maintain constant sodium temperature return to the intermediate heat exchanger. This is accomplished by controlling the position of the feed-water valve by means of a quick response, resistance-type temperature device located in the sodium outlet line of the steam generator. In order to partially compensate for the time constant of the steam generator and time lag in the temperature control relays, an inlet sodium flow signal has been included as an anticipatory device. The feed-water control valve is equipped with an airlock to maintain its position in the event of control air failure.

During normal operation, the output of the turbine generator will be controlled by a pressure device in the main steam header. This device will open or close the steam turbine throttle as required to maintain constant pressure.

The steam temperature to the turbine will be automatically controlled by regulating the flow of water from the feed-water pumps to the spray type attemperator.

Although the SRE and Edison control rooms are physically separated, there will be interconnecting telephone and instrumentation circuits. SRE information displayed on the Edison control board will include the following:

1. Neutron flux level expressed as per cent of power
2. Reactor scram alarm
3. Reactor set back alarm
4. Sodium inlet and outlet temperatures

Steam plant information displayed in the SRE control room is as follows:

1. Kilowatt output
2. Steam pressure
3. Feed-water flow
4. Alarms on low feed water, sodium outlet temperature, throttle trip, and monitoring fluid pressure

Most important of all these interconnected alarms is the signal initiated by the loss of feed-water pressure to the steam generator. This signal is included in the SRE scram loop, and will, upon complete loss of feed-water pressure, initiate a reactor scram operation.

Such an occurrence, however, is considered to be very unlikely because of the reliability that has been built into the station. Dual automatic pumps on the feed water, condensate, and condenser cooler should give first-class service. Equally important is the high degree of reliability furnished in the source of station light and power as well as in the construction of the auxiliary power syste

VI. ELECTRICAL

Figure 8 is a view of the station, with the cooling tower in the background, which shows the electrical equipment to better advantage.

The turbine generator equipment is a conventional 7,500-kw package unit supplied by the General Electric Company. The entire assembly is mounted on a 7,000 sq ft, two-pass, surface condenser manufactured by Ingersoll-Rand. The

enerator is an air-cooled, outdoor-type unit served by a direct-connected exciter. ts neutral is connected to ground through a potential transformer to provide for round alarm indications.

On the line side, the generator connects through an air circuit breaker to three-phase power transformer and station service bus. The step-up transformer raises the voltage from 4,160 volts to 69 kv for transmission onto the Edison system.

Connected to the station service bus and supplied from either the generator r the system is the station auxiliary transformer. It serves the auxiliary equipment at 480 - 240 and 120 volts.

Air circuit breakers and outdoor-type, metal-clad switch gear are provided hroughout for both the 4-kv and 480-volt busses. All circuits are protected with vercurrent relays, and, in addition, the generator and power transformer have ifferential relay protection.

On the extreme left-hand side of the figure you will see a portable boiler hat will be used to test the turbine generator unit. With this boiler, the turbine ill be brought up to speed for balance and vibration checks. The generator will e phased-in electrically and synchronized to the system.

In view on the extreme right-hand side of the figure is the terminal pole of e 69-kv circuit which will transmit the energy to, first, the Edison Chatsworth ubstation and thence over 69-kv transmission lines to one of the major 220-kv tations for distribution over the entire Edison system.

VII. REMARKS

In conclusion, let me repeat that the Southern California Edison Company ncerely hopes that this experimental installation will contribute materially toward e development of low cost electrical energy from the sodium graphite reactor.

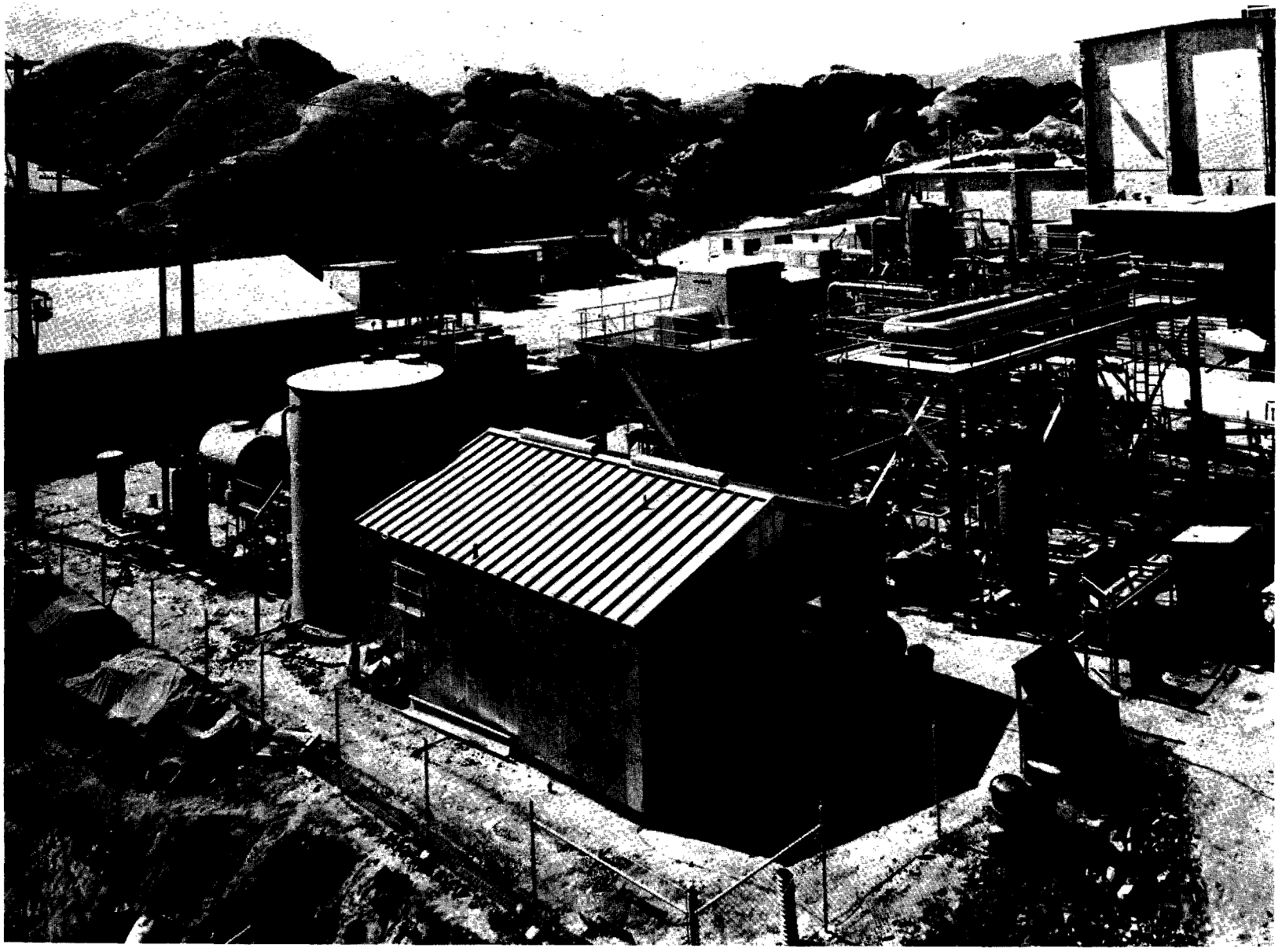

Fig. 1. General View of Edison Steam Plant

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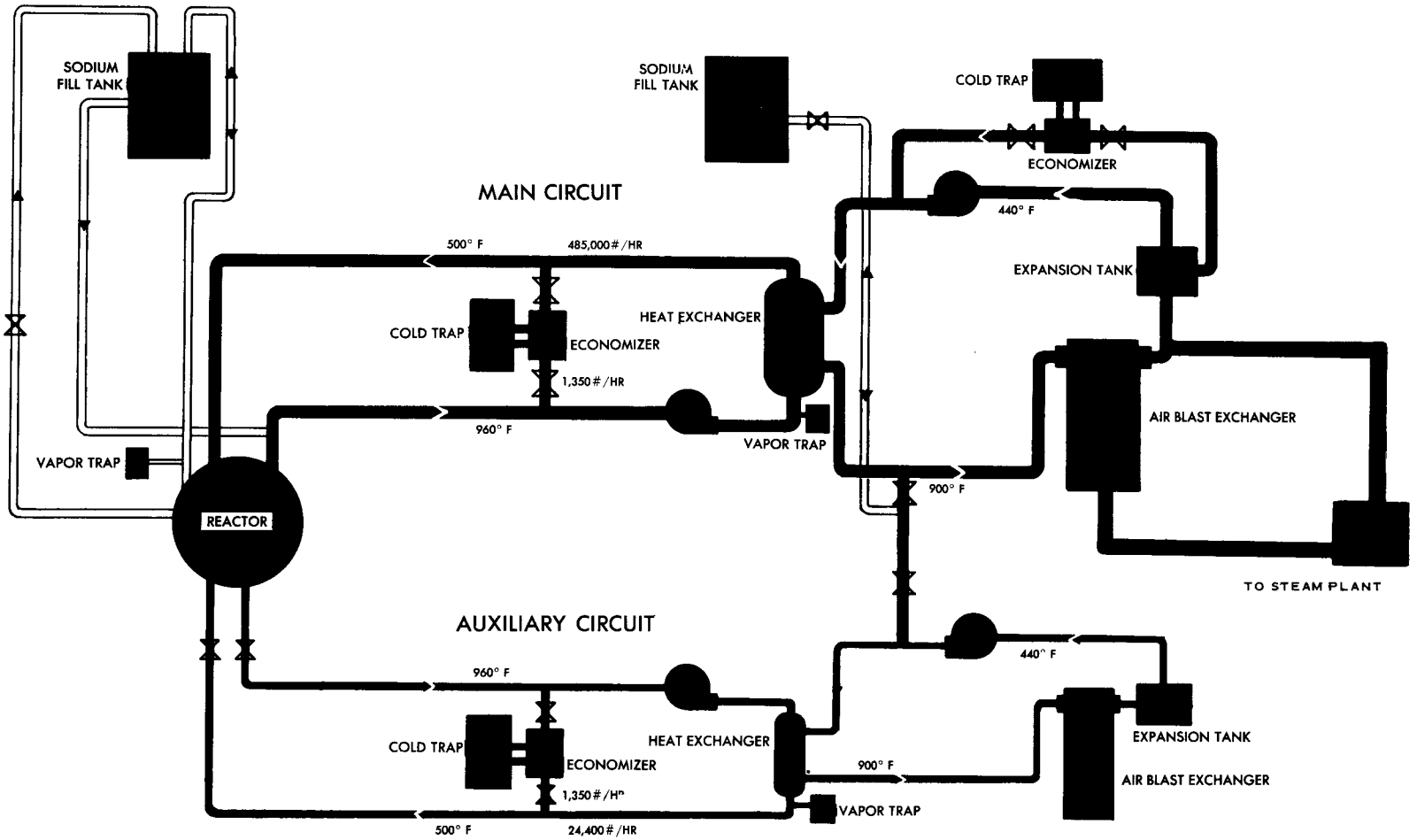

Fig. 2. SRE Cooling System Diagram

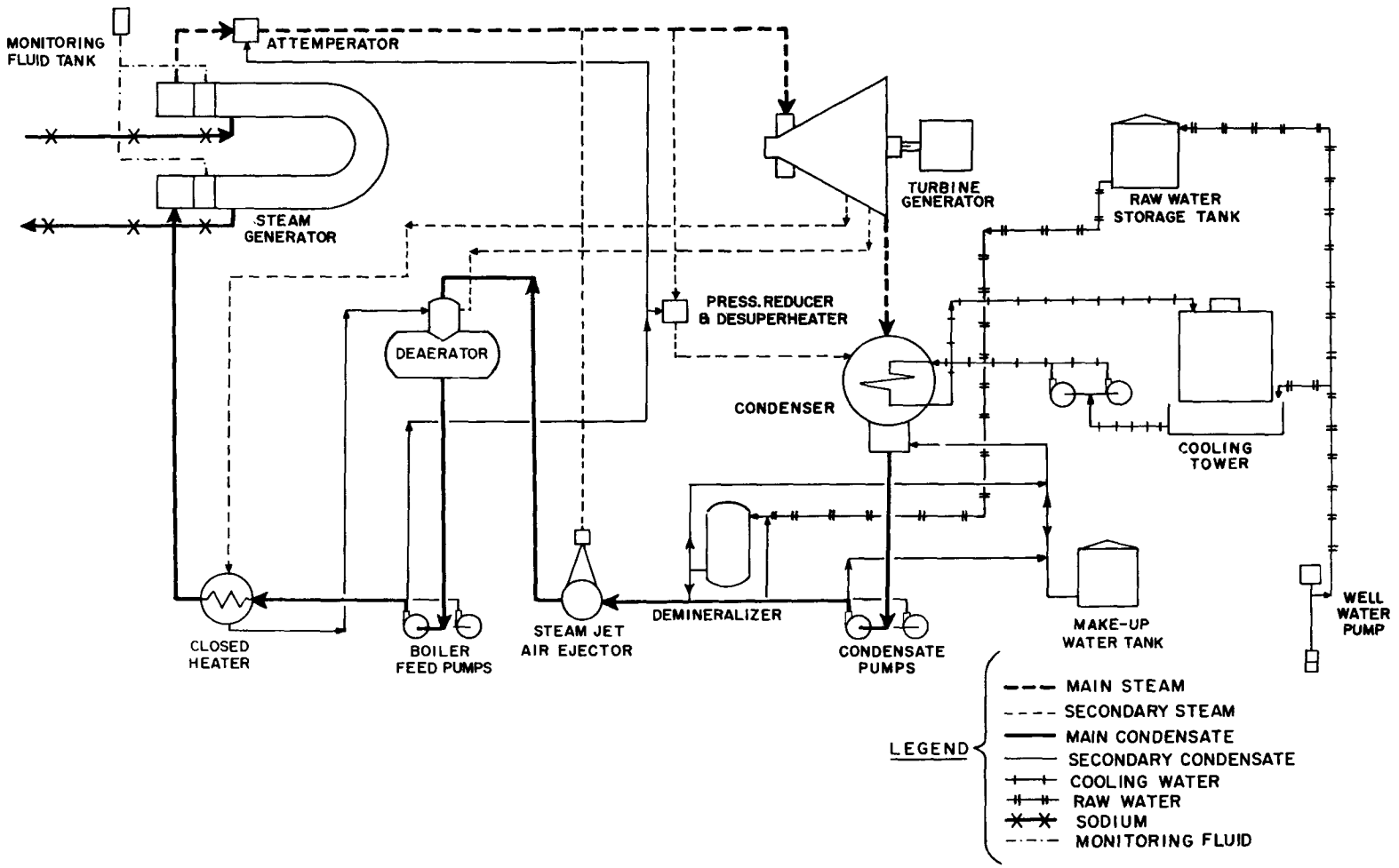

Fig. 3. Basic Flow Diagram of Steam Plant

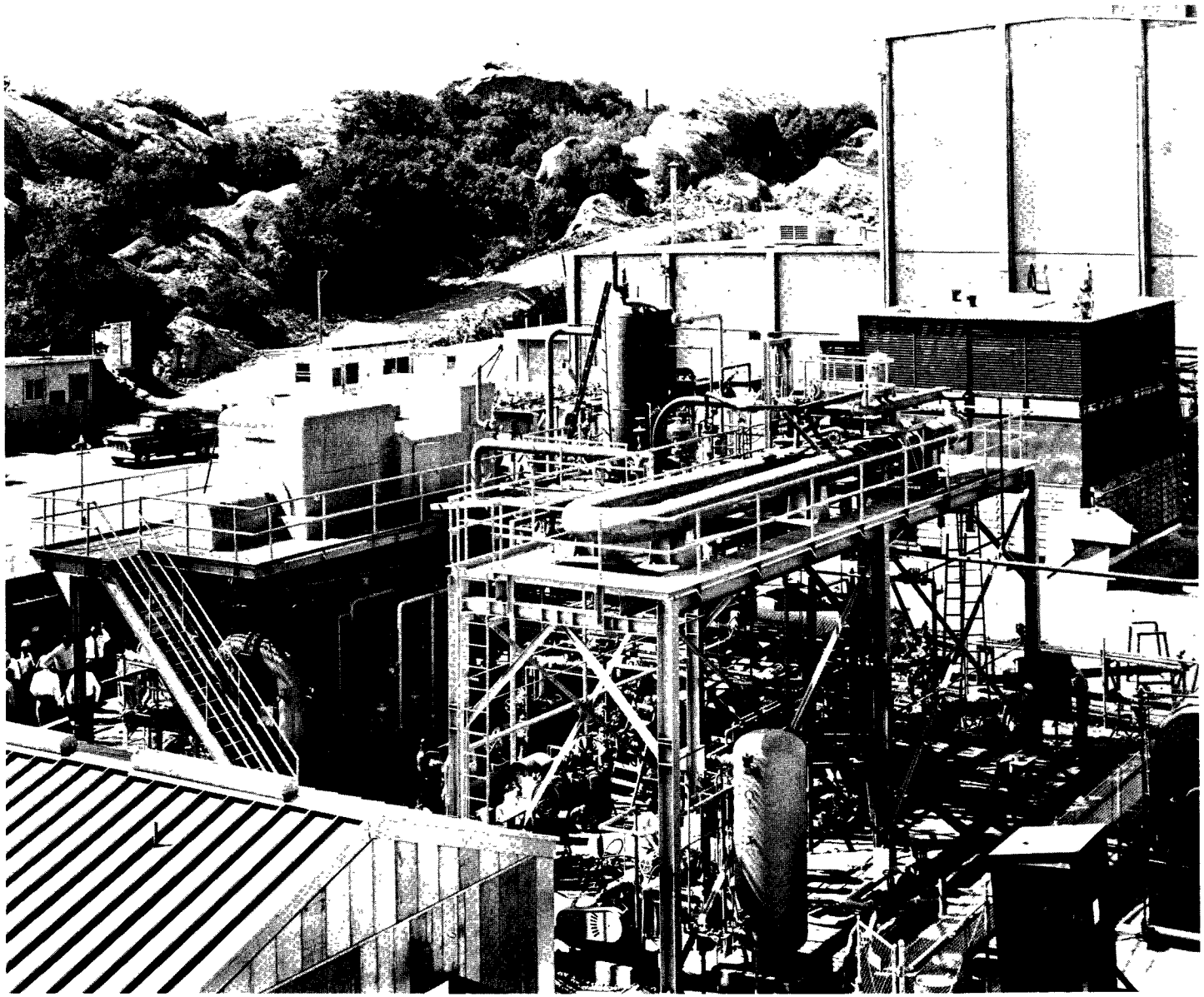

Fig. 4. Major Steam Plant Components

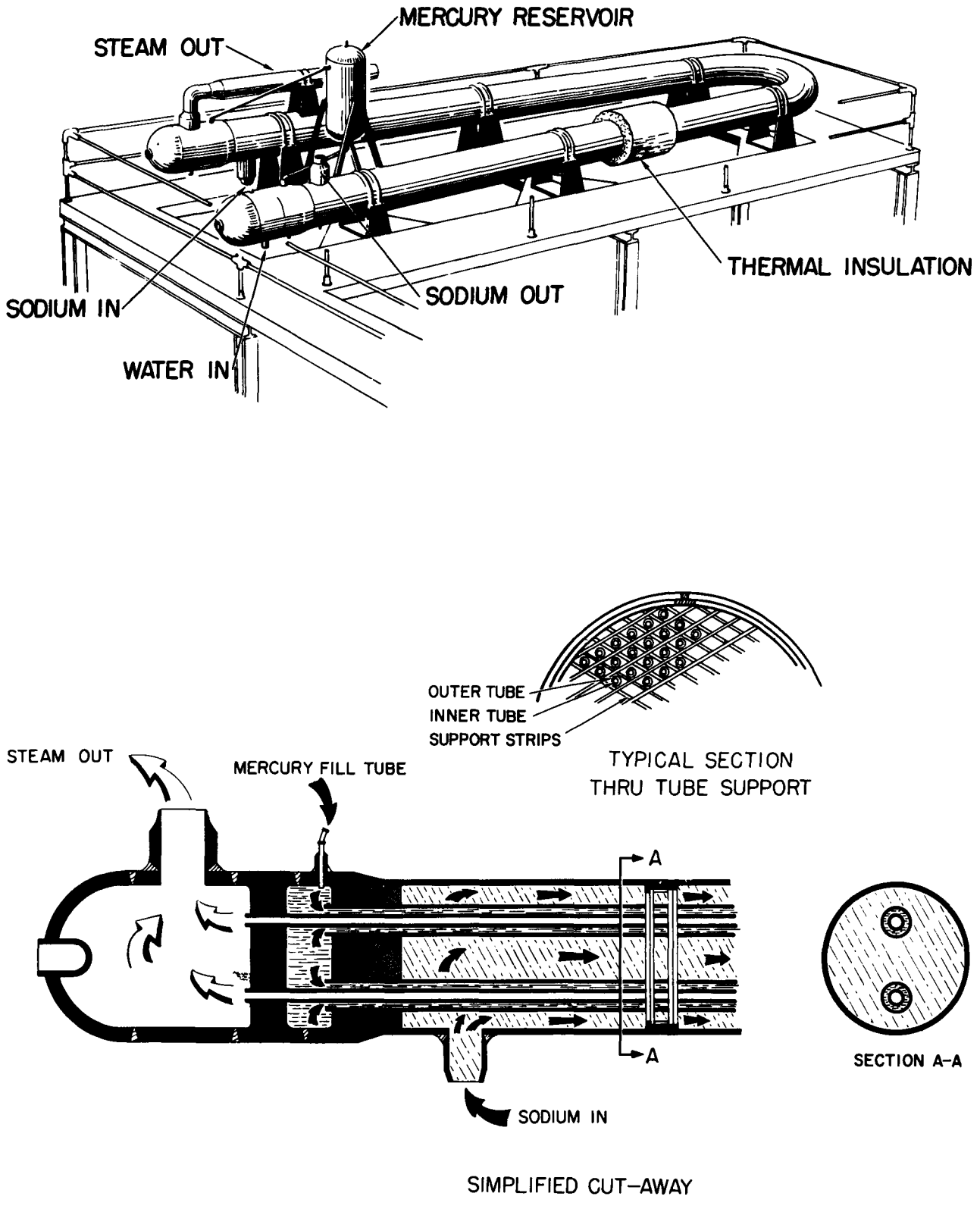

Fig. 5. Steam Generator

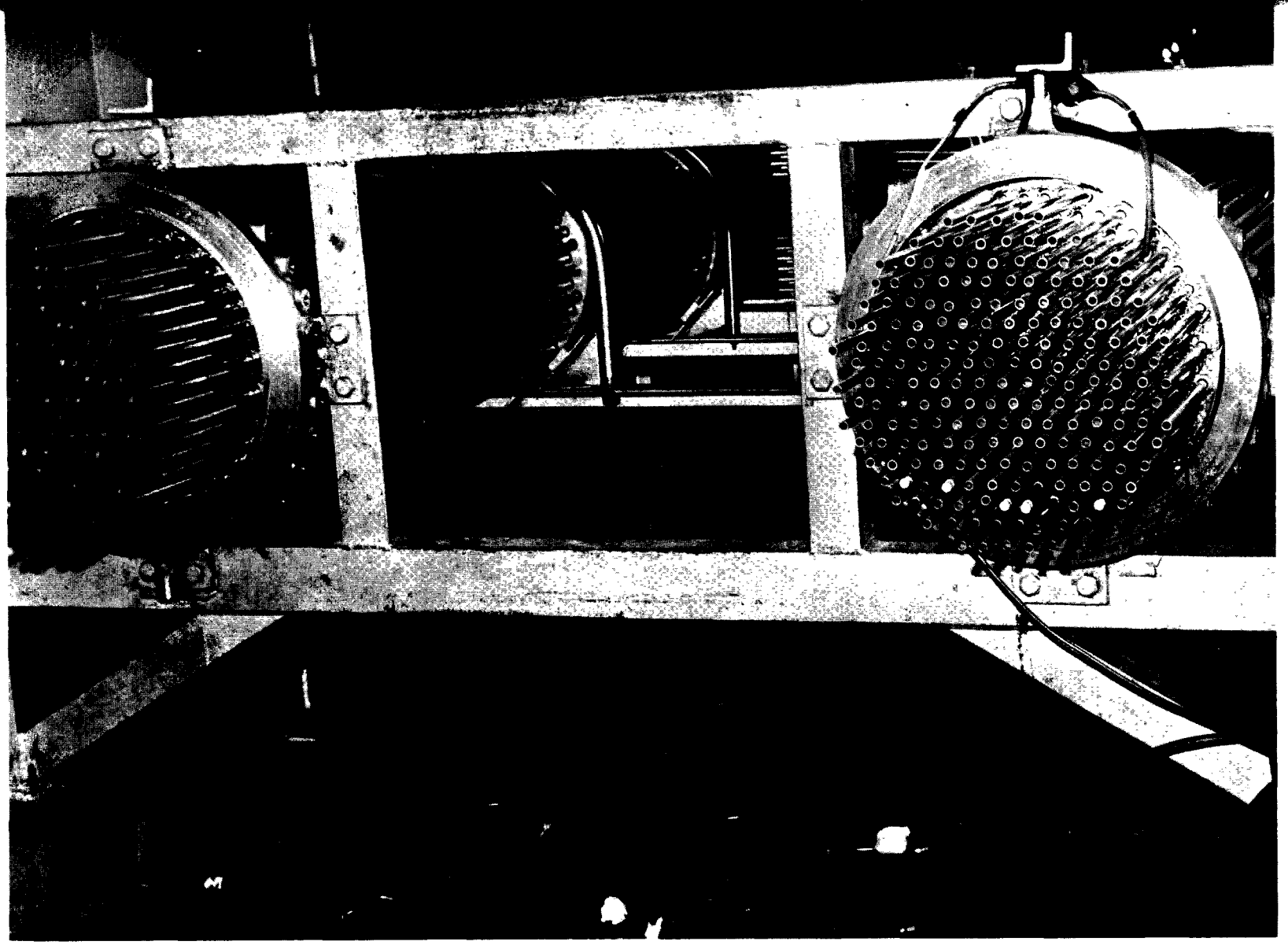

Fig. 6. End View of Tube Assembly

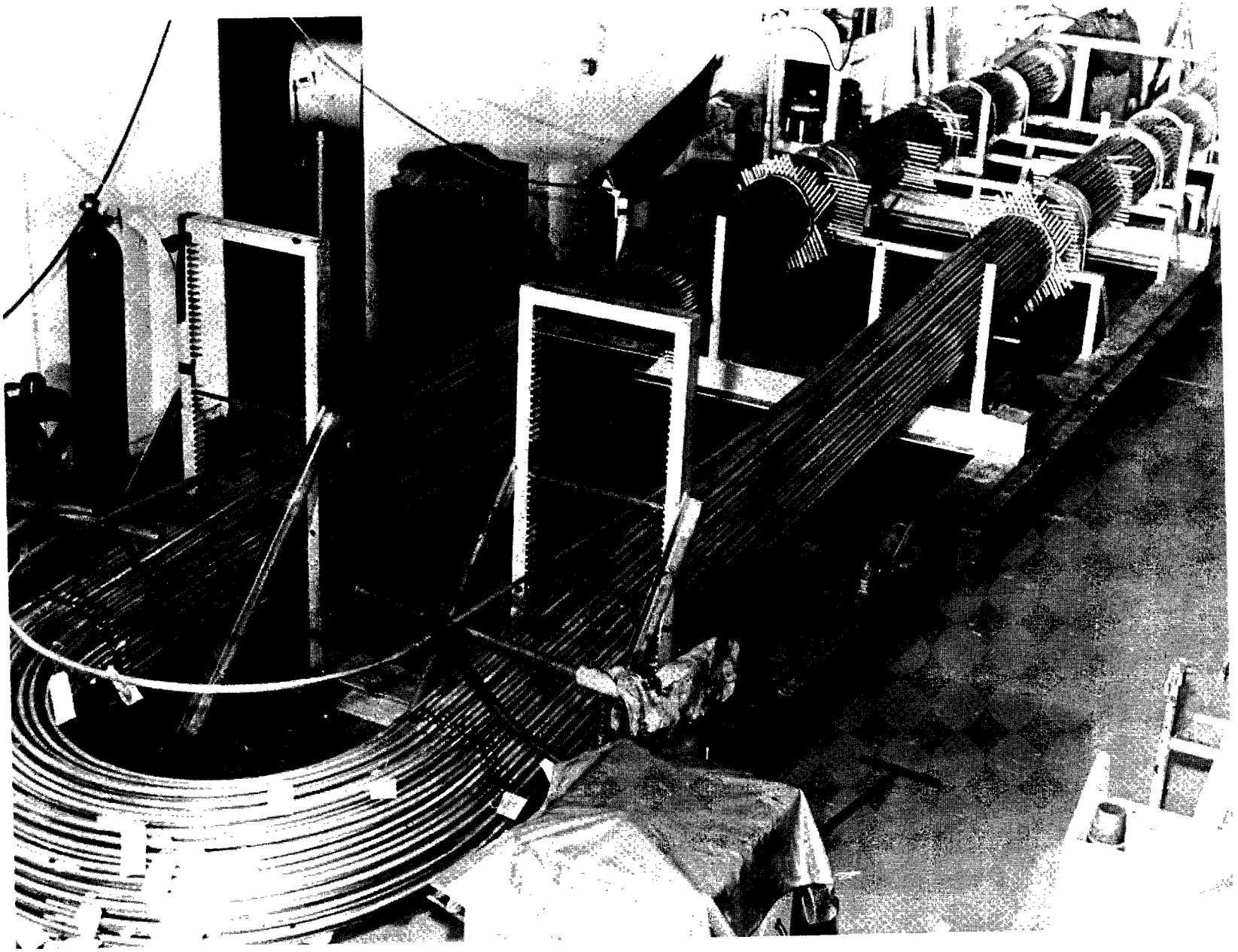

Fig. 7. Tube Bundle

Fig. 7. Tube Bundle

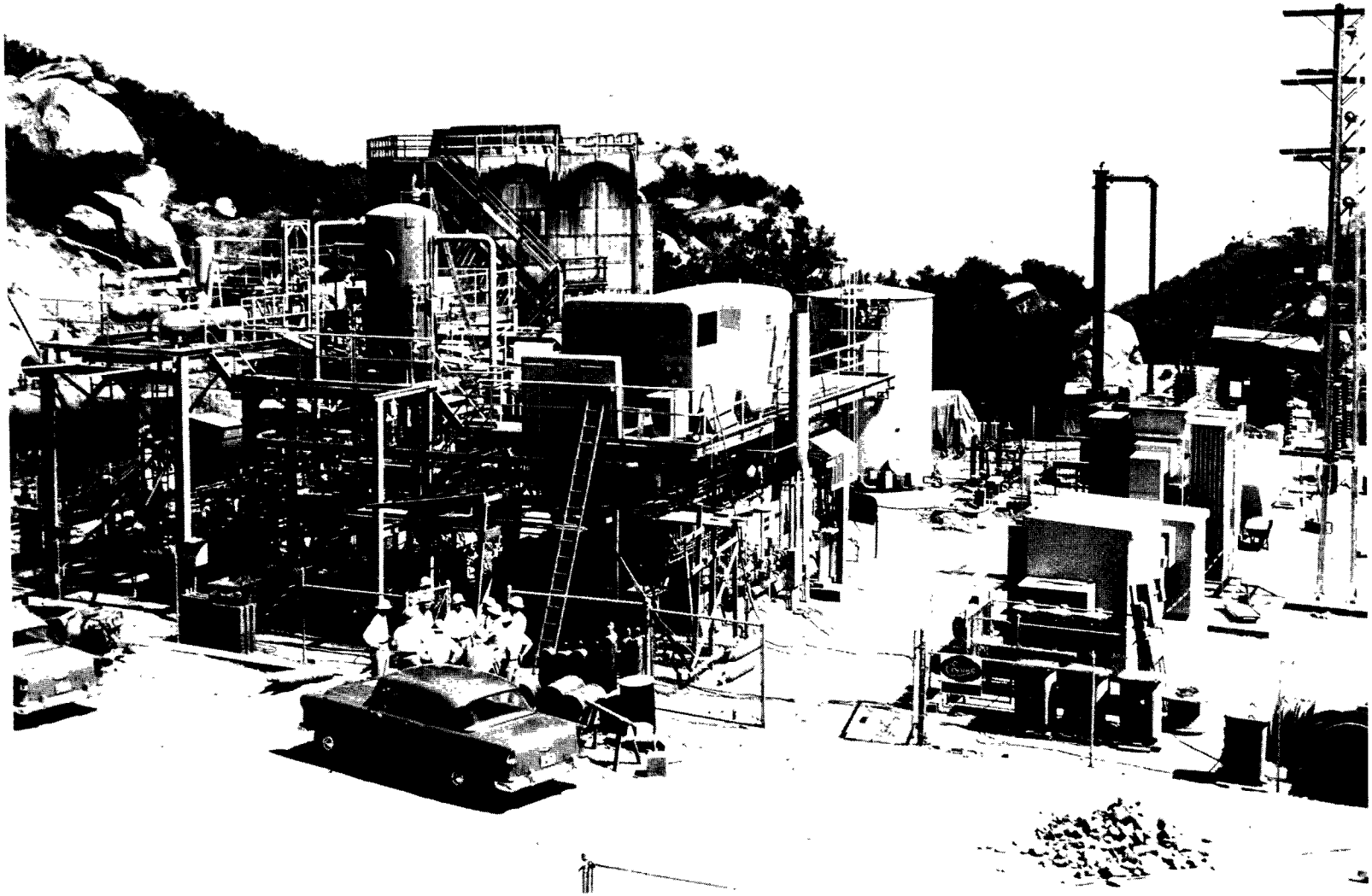

Fig. 8. Electrical Equipment

REACTOR COMPONENT DEVELOPMENT

D. T. Eggen

During the last three years of the SGR program, several of the reactor components have been developed and tested for the SRE. These include fuel elements, moderator-reflector elements, safety rods, control rods, and level indicating devices. In addition, certain of the heat transfer components have been developed, including the static and rotating freeze seals for the pumps and valves. The operational characteristics of the cold trap, tetralin system, and pumps have been studied. Experiments have been performed on various insulating materials, shielding materials, and commercial items which will be incorporated into the SRE, and recommendations for their use have been made. A program of development of maintenance and handling techniques has been pursued. Hydraulic flow studies have been conducted on the fuel assembly and fuel element configuration and on the core configuration to determine the adequacy of the sodium coolant mixing for heat-transfer results and reduction of thermal stress and thermal shock during reactor transients.

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I. INTRODUCTION

An active SRE development program has been in progress, aimed at pro-ding proved components for reactor use. This work has covered the fields of at-transfer system components and instruments, reactor core components, and xiliary devices. The development program is intended to expand our area of formation covering sodium-graphite technology, to experimentally demonstrate e reliability of reactor components, to extend their performance limits, and to pply the information developed to further improvements of the SRE.

A general description of each of the reactor components was presented in e section entitled SRE Engineering Description; therefore, the illustrations resented here will be used merely as reference in conjunction with the discus-on of the development work.

II. HEAT TRANSFER SYSTEM COMPONENTS

The heat transfer system of the SRE includes pumps, valves, and auxiliary uipment, as well as the heat exchangers and associated piping. Experience in mping sodium in large quantities at high temperature was not extensive in the rly stages of the SRE work and the feasibility of operation of the heat transfer stem had to be established.

PUMPS

A mechanical pump for sodium service has been under development, because the high cost and low efficiency of electromagnetic pumps. A major effort was ced on the development and testing of a centrifugal pump shaft seal known as reeze seal. This seal is formed by cooling a relatively thin annulus of sodium ound the shaft until a solid plug is formed by freezing of the liquid metal. gure 1 shows a cutaway of the SRE pump. Two freeze seals are used: the shaft al and a case seal which prevents leakage of sodium into the upper shell. Cool-load measurements on the case seal are reduced from 10 kw to 2.5 kw by the dition of anti-convection rings.

Details of the shaft seal are shown in Fig. 2. The coolant, tetralin, enters e seal at the upper flange, progresses through a helical path around the shaft

to the lower end of the seal and returns in the same way to the discharge. The shaft seal is made by machining the annular cooling passages from a solid piece of steel.

Tests performed on this seal were coolant pressure drop, temperature distribution, heat removal rate, distortion from thermal cycling, pump starting and running torque, and the time required for seal failure after loss of coolant flow. A typical temperature profile is shown in the figure for conditions of 1000° F bulk sodium temperature and a shaft speed of 1250 rpm. The location of the frozen sodium is approximately one inch below the top of the seal. A typical heat removal rate for these same operating conditions is 3 kw with a flow of tetralin of 4 gpm.

B. VALVES

A valve using this same principle of sealing is used in place of a packing gland or bellows-sealed valve stem for sodium service. Figure 3 shows a 6-inch plug cock with a freeze seal on the valve stem. To lengthen the heat conduction path, thus decreasing the coolant requirements, the freeze seal is removed several inches from the packing gland. Loss of coolant flow can be sustained for as long as 20 minutes without failure of the freeze seal.

The precipitation of sodium oxide in a freeze seal could lead to difficulty of operation, since the solid particles of oxide could cause binding of the shaft or stem. This can be avoided by maintaining a low concentration of sodium oxide dissolved in the sodium in the system. Long life of moderator cans also depends upon maintaining a low concentration of oxygen in the sodium.

C. COLD TRAP

In order to keep the sodium oxide concentration at a suitably low level, a toluene-cooled cold trap is used to precipitate and filter sodium oxide from the reactor coolant. Figure 4 shows the general arrangement. Toluene was selected to cool the sodium since the boiling point of 231° F corresponds to the temperature at which the solubility of sodium oxide in sodium is at an acceptable value. The cold-trap temperature is maintained at a constant value by the boiling of toluene and the rate of heat removal is varied by adjusting the flow of toluene.

A full-scale prototype cold trap (Fig. 5) was constructed and tested to determine operational characteristics prior to startup of the SRE. In addition to obtaining thermal data to evaluate overall heat-transfer coefficients of the cold trap assembly, information has been obtained on rates of oxide removal as a function of the cold-trap operating variables.

Oxygen was deliberately added to the circulating sodium and the cold trap was operated at the design throughput of 3 gpm sodium.

The residence time was sufficient to permit the exit liquid sodium to be reduced to its equilibrium "saturated" oxide concentration at the cold trap temperature.

The mesh filter within the cold trap retained 100 per cent of the oxide precipitated from the flowing sodium stream.

Utilizing these results, it has been possible to calculate the cleanup time for cold trapping which will be required in the SRE. For example, assuming an initial oxide concentration of 100 ppm in the 52,000 lb of sodium in the primary loops, the two cold traps provided will be able to reduce the oxide concentration to 10 ppm in a period of 58 hours.

D. PLUGGING METERS

Figure 5 shows a plugging meter which is used to measure the concentration of oxide in circulating sodium. The operation of this device is based upon gradual cooling of the sodium until the sodium is saturated and sodium oxide is thereby precipitated. These solid particles of sodium oxide agglomerate are filtered out on a perforated disk through which the sodium flows. As the temperature is lowered in the tetralin HX, the flow rate of the sodium through the disk is restricted, since the holes become plugged with Na_2O. The temperature at which the flow rate initially decreases can be correlated with the oxide content of the sodium. Development work included optimization of the size and number of holes in the disk and incorporation of this disk as the plug of a valve for operating simplicity and for ease of cleanup following a plugging temperature determination.

E. FLOWMETERS

The measurement of the flow of sodium is done by means of electromagnetic flowmeters. Meters for use in 6-inch, 2-inch, and 1-inch pipe lines were calibrated and the magnets were stabilized by thermal treatment and mechanical shock.

III. FUEL ELEMENT DEVELOPMENT

A. FUEL ELEMENTS

Figure 7 shows the details of construction of a standard SRE fuel element.

The 6-inch long uranium fuel slugs are contained in a 10-mil-wall stainless steel jacket with NaK added as a bonding material to improve heat transfer. Purified helium occupies the expansion space above the column of fuel slugs. To prevent the escape of fission products into the coolant stream, the closure welds at the ends of the jacket must be of the highest integrity. One of the principal problems in the loading process was this welding of the thin wall tubing to the relatively massive section of the end caps.

Procedures and techniques were developed for assembling the fuel rods while maintaining the system free of oxygen (Fig. 8). Equipment was constructed which permitted the assembly of 21 rods in a batch, the closure weld being made by a remotely operated semi-automatic welding device.

Testing procedures have been evolved for detecting voids in the NaK bond, imperfections in the tubing, and leaks in the welds.

B. HYDRAULIC STUDIES

Hydraulic studies have been carried out to ascertain that adequate coolant is present throughout all regions of the fuel element assembly. Pressure drop across the fuel element has been determined over the full range of anticipated coolant flow rates. Orifices have been calibrated and the pressure drop across the orifice determined for flow rates of from 10 to 100 gpm.

The wire wrap configuration was established for each of the outer rods of the standard elements by use of chemical injection techniques. The proper

onfiguration ensures circulation of coolant from the vicinity of the outer rods o that of a central rod to prevent overheating of this central rod.

A hydraulic model of the core tank (Fig. 9) was used to study the flow distribution in the fuel channels, pressure distribution in the upper and lower plenum hambers, and deflection of a fuel element during fuel loading.

IV. MODERATOR CANS

Figure 10 shows the construction of a moderator can assembly. The principal problem encountered during the development of the moderator cans was that f the thermal stress. The maximum thermal stress occurs just after a reactor cram. At that time, the can head acts as a diaphragm and must bend downward o allow for the contraction of the coolant tube. This motion imposes a severe ending moment on the head to coolant tube joint. The investigation of this problem constituted a major part of the program of producing acceptable moderator ans.

Trouble with the original joint was first encountered in a Can-Head Flexure est (Fig. 11). In this test, an 8-inch long section of moderator can was fabricated using two "top" heads arranged so that these could be mechanically flexed a manner similar to conditions expected in the reactor. Repeated failure of e head to tube joint occurred in parent material adjacent to the weld. Investiation proved that the welds were sound but that the joint was not sufficiently igid to withstand the stress level imposed by the bending moment. Several esign alterations were investigated with the use of "Stress-Coat" and the most romising was thoroughly stress analyzed. Four head-to-tube joints of the new onfiguration, using a reinforcing collar, which is shown in Fig. 12, were fabicated into can flexure specimens and were successfully tested for extended eriods. One effect of this redesign was a new stress distribution in the can uch as to increase the stress level at the head to sheath joint. This joint is still onsidered safe, since the calculated stresses are below the ultimate strength r the material.

An additional part of this program was the testing of two full-sized cans a vacuum retort, where it was possible to obtain a temperature gradient

equivalent to that in the reactor. A small amount of outgassing of the graphite moderator was observed during these tests.

V. CONTROL RODS

The design of the shim and regulating control rods is shown in Fig. 13. Problems connected with this assembly were poison ring development, heat transfer from these rings, and the selection or development of materials suitable for high temperature operation.

Centrifugally cast rings made from a modified hardfacing material containing 1-1/2 per cent boron were successfully fabricated. Thermal cycling tests demonstrated the ability of these rings to withstand reactor conditions.

Since the heat generated in the rings from nuclear reactions will be as high as 8.3 kw/ft, a heat transfer experiment was required to demonstrate that adequate cooling could be obtained. This experiment established the maximum diametrical clearance between the poison rings and thimble to be approximately 0.040 inch.

The choice of materials for components which must undergo unlubricated sliding contact at operating temperature without galling proved to be a major problem. The ball nut and screw used to move the poison rings in the core received considerable study and testing. Haynes 25 alloy has been selected for the screws as the best available material. This could be improved for longer life by proper cold-working of the ball races. It has been found that high hardness at high temperature is necessary for long time operation. Successful operation with cast Stellite #3 balls in the ball nut has been accomplished.

Tests of a prototype control rod have indicated an expected life of 10,000 cycles using presently available materials. This is adequate for use as shim rods but not for regulating rods. The Mark II control rod shown on the right of the figure is designed to permit a majority of the moving parts to be located above the loading face in a cool region, thus alleviating the severe materials problem.

VI. SAFETY ROD

Figure 14 shows some of the features of the design of the SRE safety rod. Tests of individual components were made to experimentally demonstrate the suitability of each part for its prescribed function.

The snubber, which absorbs kinetic energy of the falling rod by compression of trapped helium, was found to be effective, provided piston rings were used. This is illustrated in Detail A. Some materials problems have been experienced due to the close sliding fits at high temperatures.

The latch shown in Detail B was held at temperature under the anticipated load for periods of time up to 500 hours and then released to demonstrate that pressure welding did not occur.

The torque tube, used to release the latch and drop the poison rings into the reactor, was tested for ability to withstand the torsional stresses involved.

A full scale prototype has been tested for time response and operating life. Material problems similar to those in the control rod are evident, but due to the infrequent use and its fail safe features, it will be used for low power operation. A backup design, wherein the moving mechanisms are placed above the loading space, will undergo tests in the near future.

VII. SODIUM LEVEL INDICATOR

Sodium level determination at high temperature has led to the development of a line of gages using electrical-solenoid type coils. In this design (Fig. 15), the turns of the coil are magnetically shunted by the rising level of sodium around the gage thimble. This reflects a decrease in the inductance of the coil and is measured by an increase in current for a constant voltage. These have the advantage of not requiring penetration of operating parts into the sodium and sodium vapor atmospheres, thereby reducing the problem of installation and maintenance. The major development problems were met in finding an electrical coil which would operate for extended periods of time at elevated temperatures and which would be sensitive to varying levels of sodium surrounding the coil thimble. A double layer of Refrasil-insulated copper wire coiled on a split hollow-iron core has proved satisfactory for this application.

As you have noted, all the SRE components which are inserted into the core through the loading face shield are very long, from 20 to 30 feet, and of small diameter.

Figure 16 shows the component test tower in which these various items have been tested. A heated tube containing sodium is used for all high-temperature prototype tests of control rods, safety rods, and fuel elements. A mock-up of the SRE fuel-handling coffin has been used to establish handling procedures and to verify the principles of operation of the coffin.

The remotely-operated coupling device shown in Fig. 17 is used to connect the lifting mechanism in the coffin to the fuel or other element for removal from the reactor. This was tested and its satisfactory operation verified. Also shown are the cleaning cells in which the components are washed in water after immersion in sodium.

VIII. REMARKS

The primary purpose of the development program has been to provide information on the operation of various components, to recommend improvements, and to increase the level of confidence in the equipment. This purpose has been pursued over the course of the program in successful operation of many pieces of new apparatus installed in the SRE.

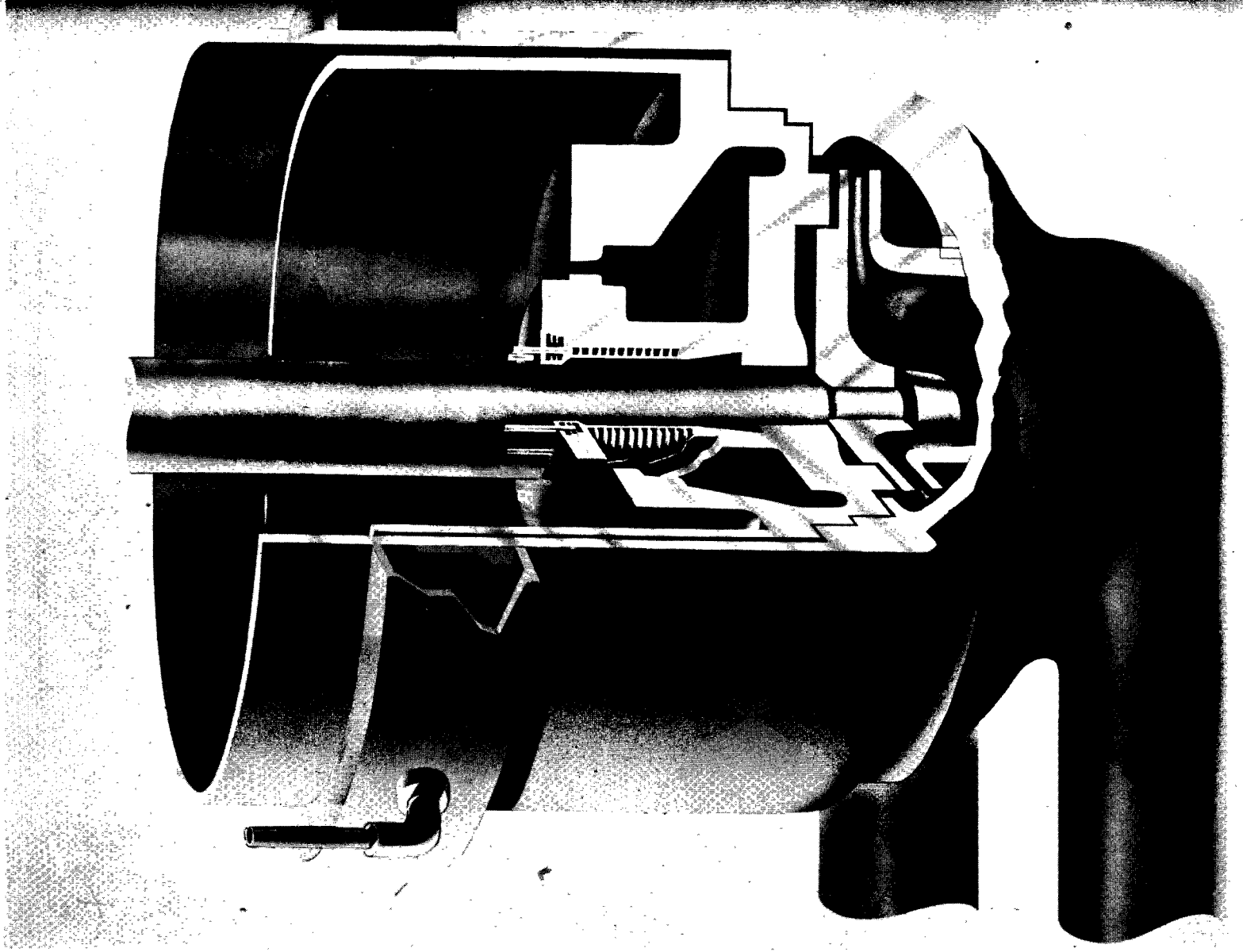

Fig. 1. Cutaway View of SRE Pump

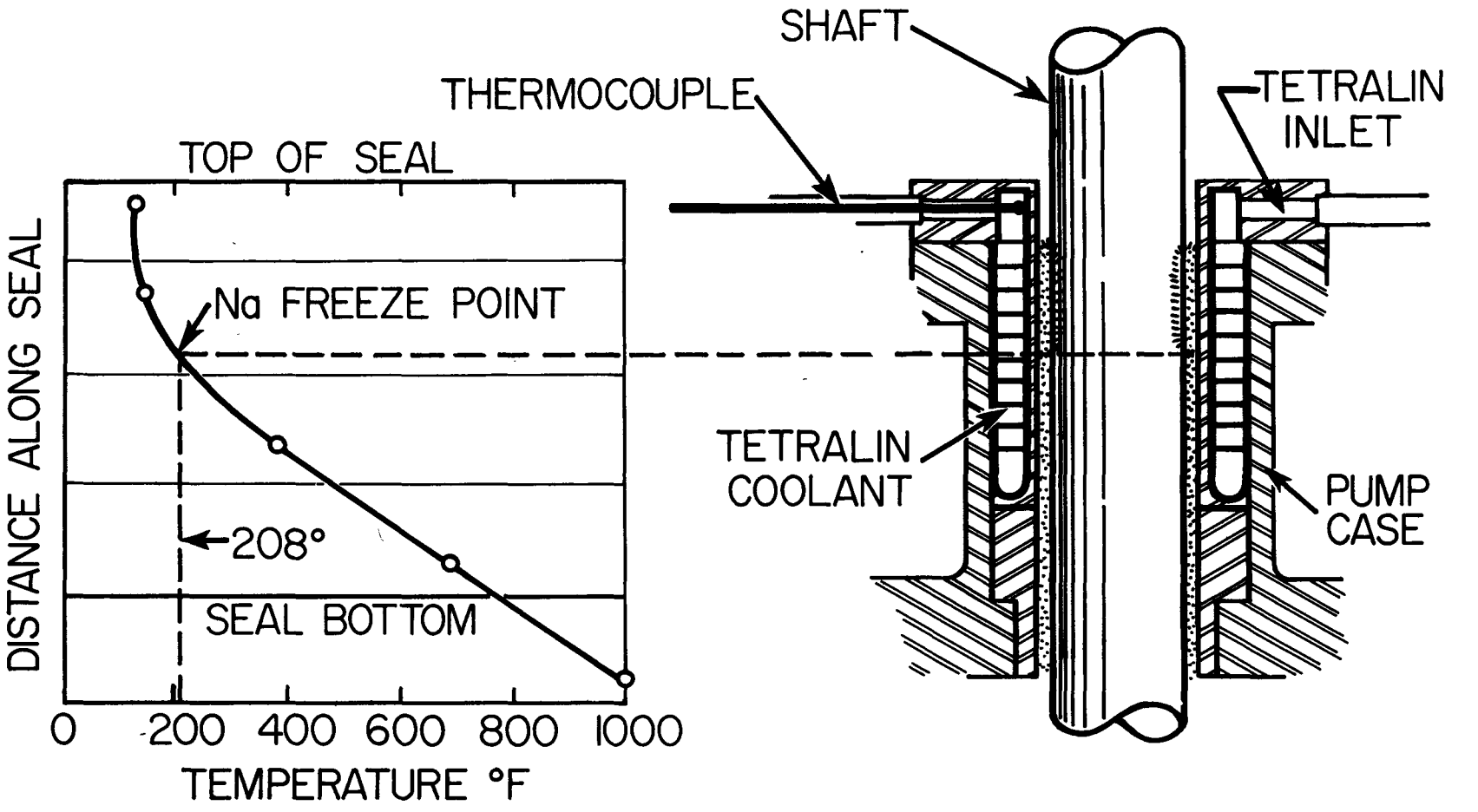

Fig. 2. Pump Shaft Freeze Seal

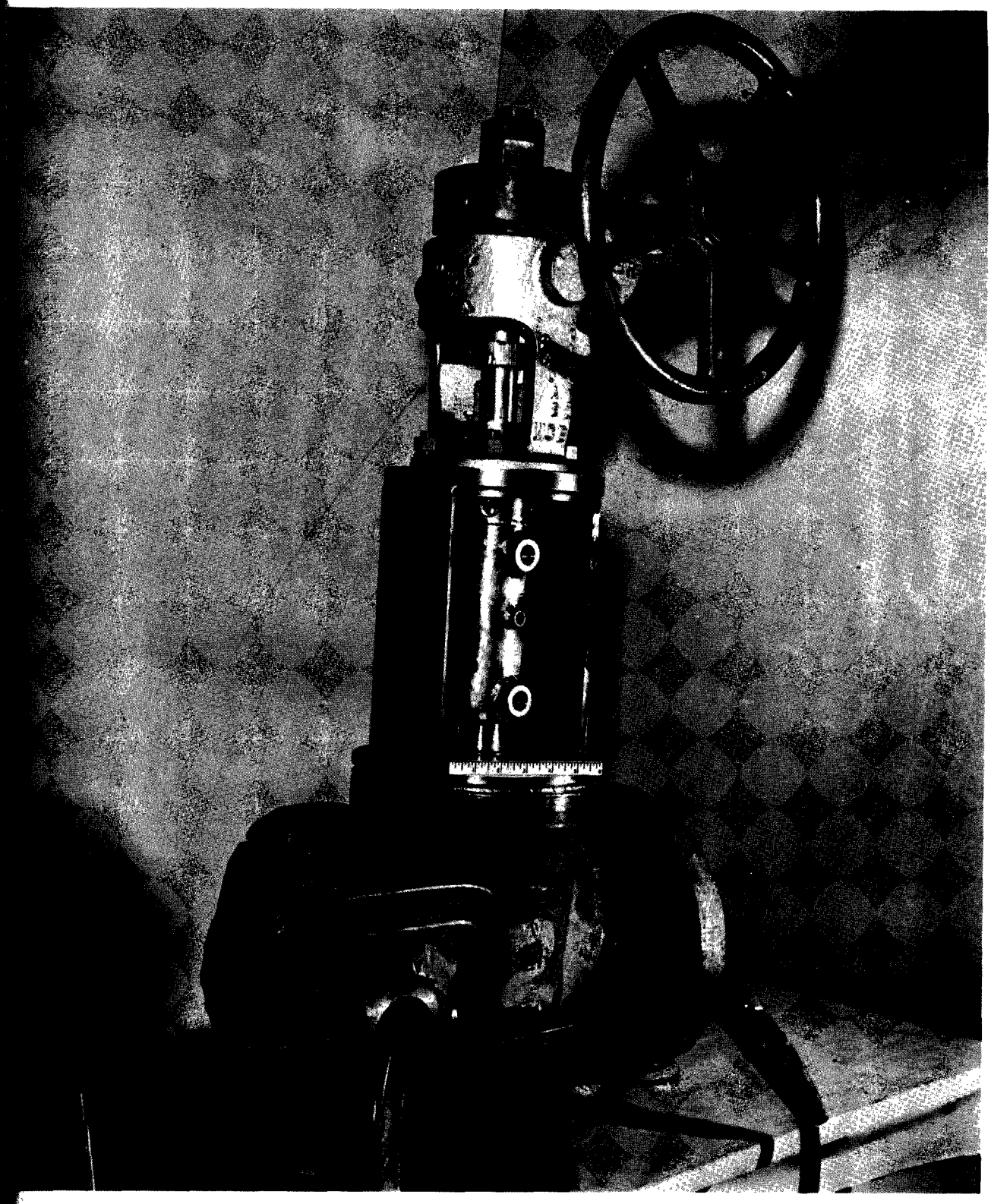

Fig. 2. Pump Shaft Freeze Seal

Fig. 3. Six-Inch Plug Cock With Freeze Seal on Valve Stem

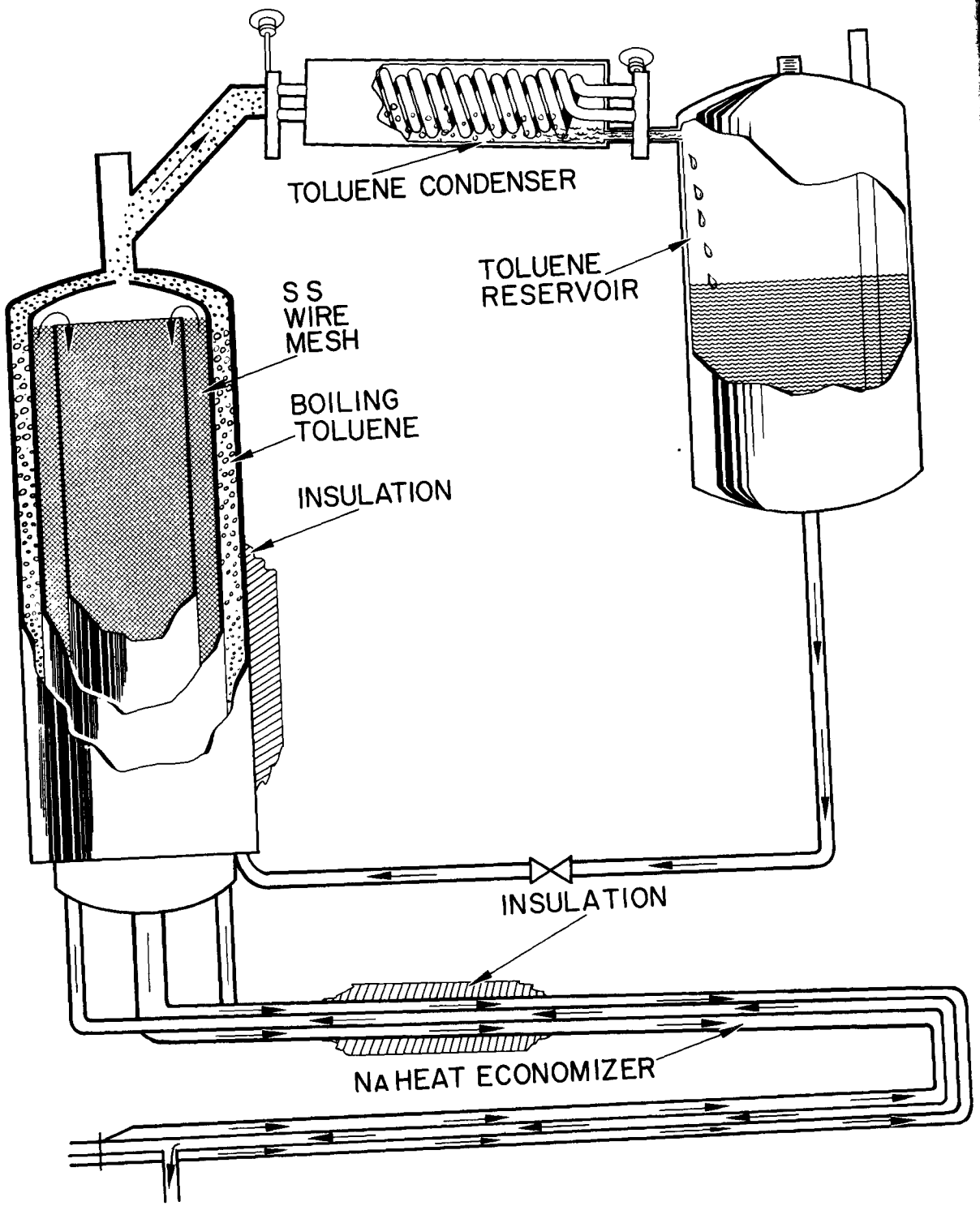

Fig. 4. Sectional View of SRE Cold Trap

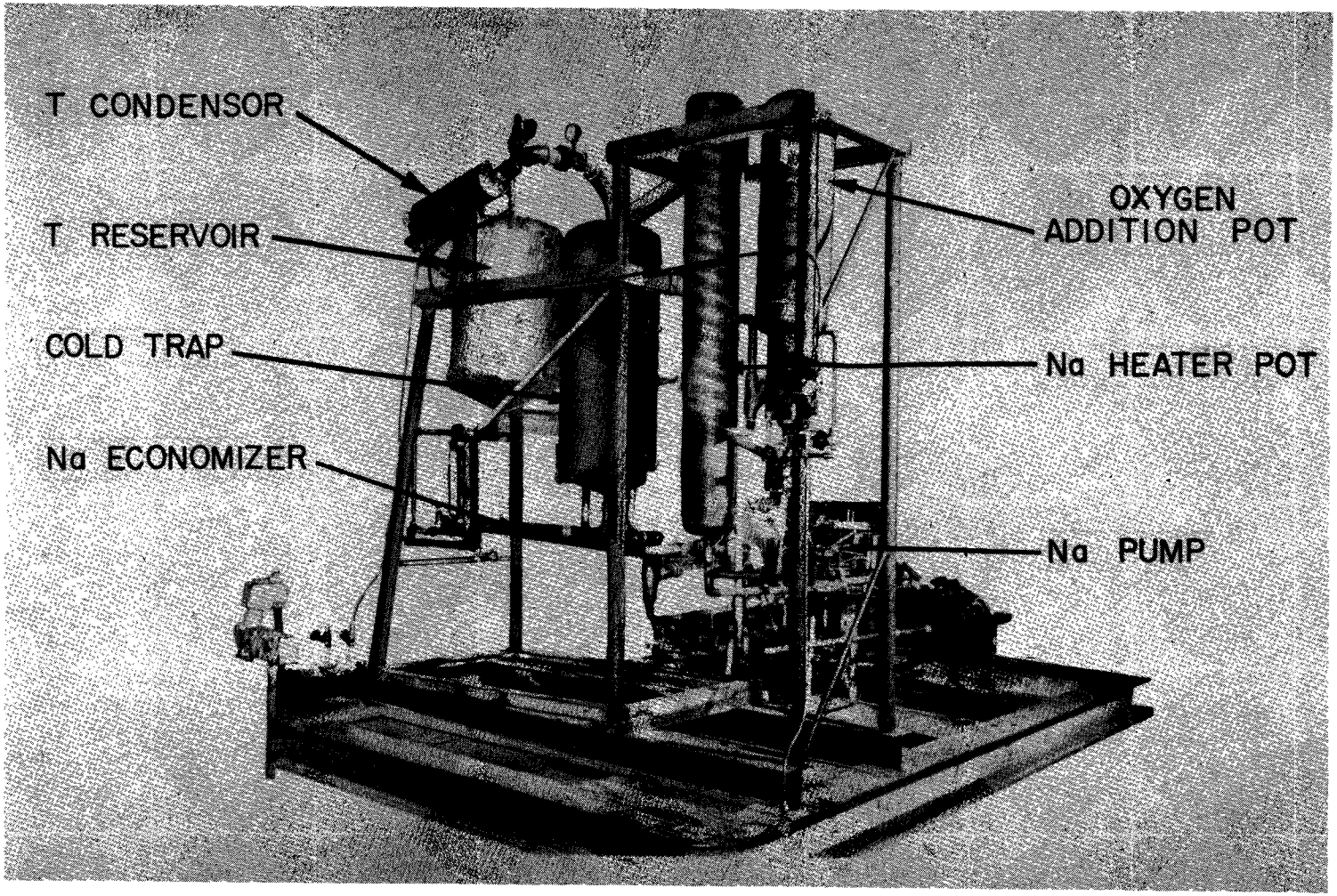

Fig. 5. SRE Cold Trap (Prototype)

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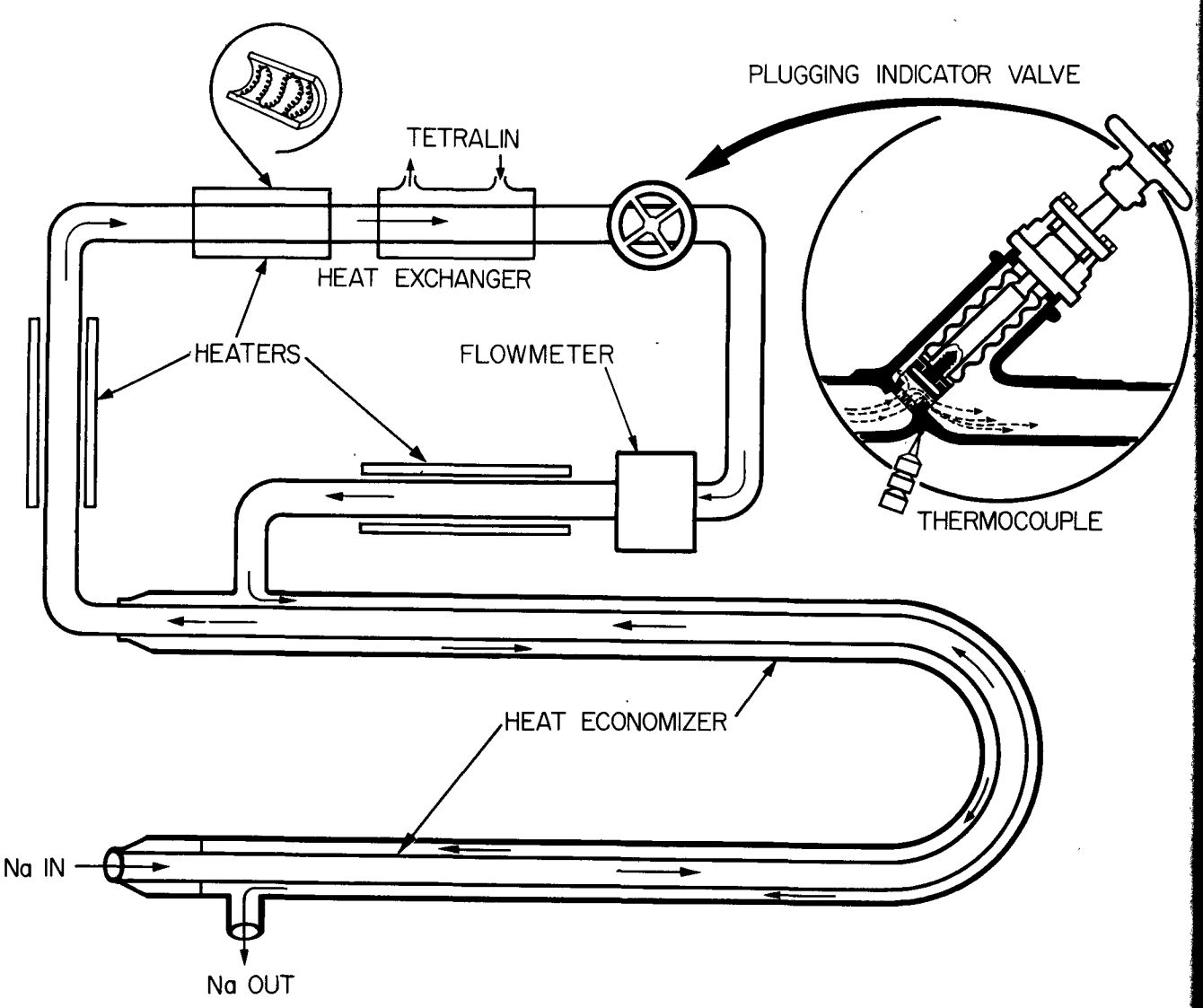
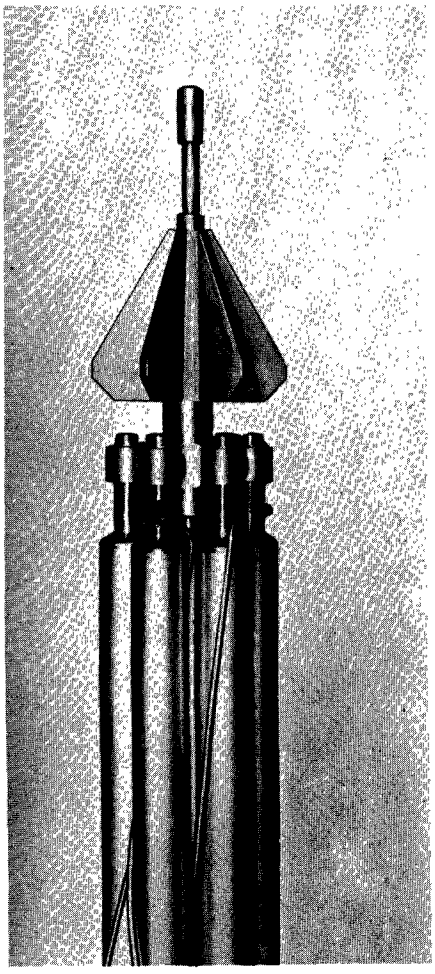


Fig. 6. SRE Plugging Indicator



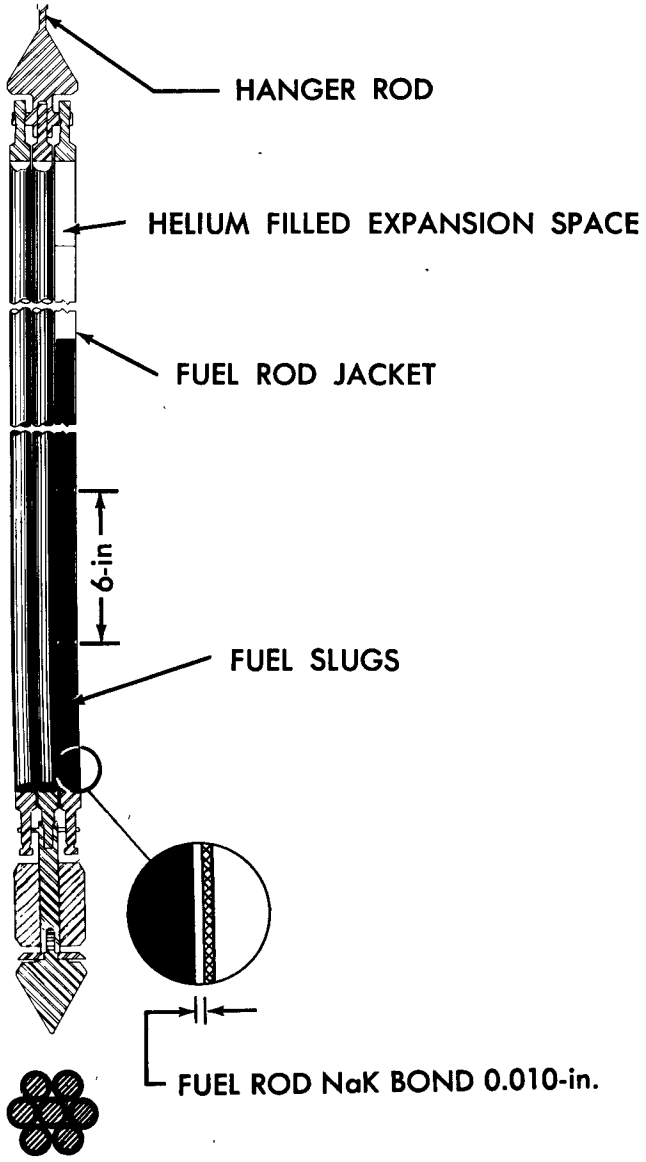


Fig. 7. SRE Fuel Element

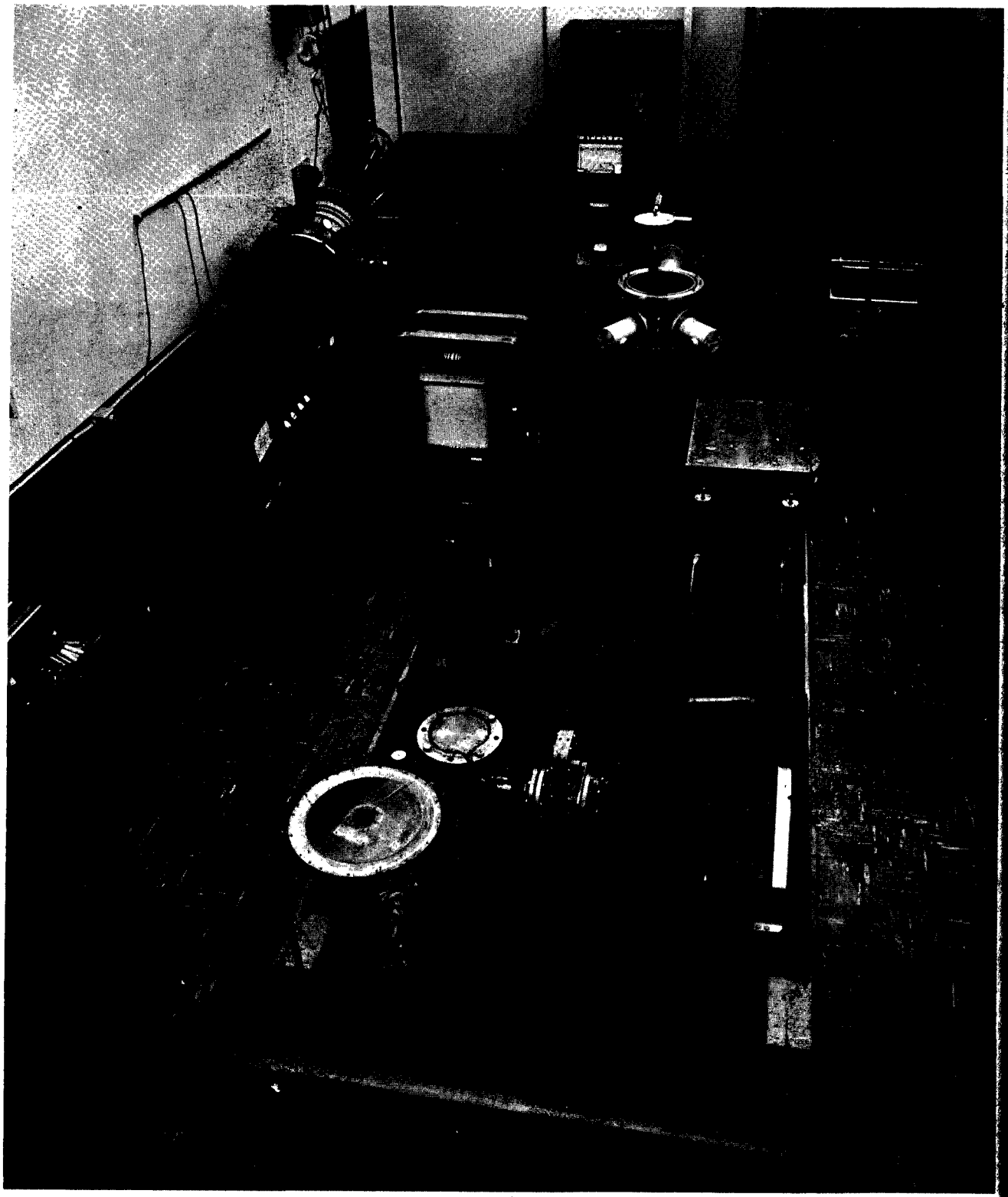


Fig. 8. Fuel Assembly Room

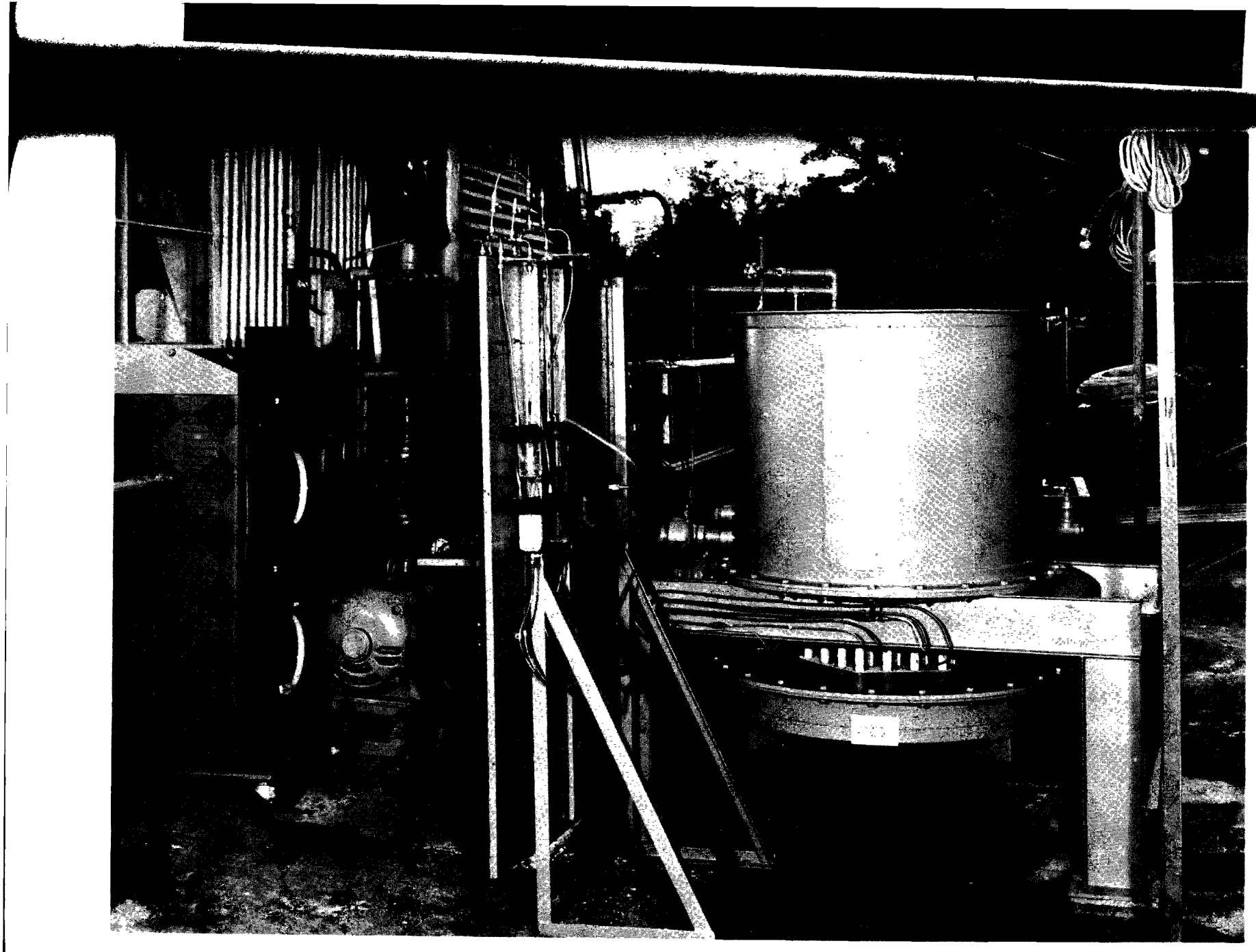


Fig. 9. Hydraulic Model of Core Tank

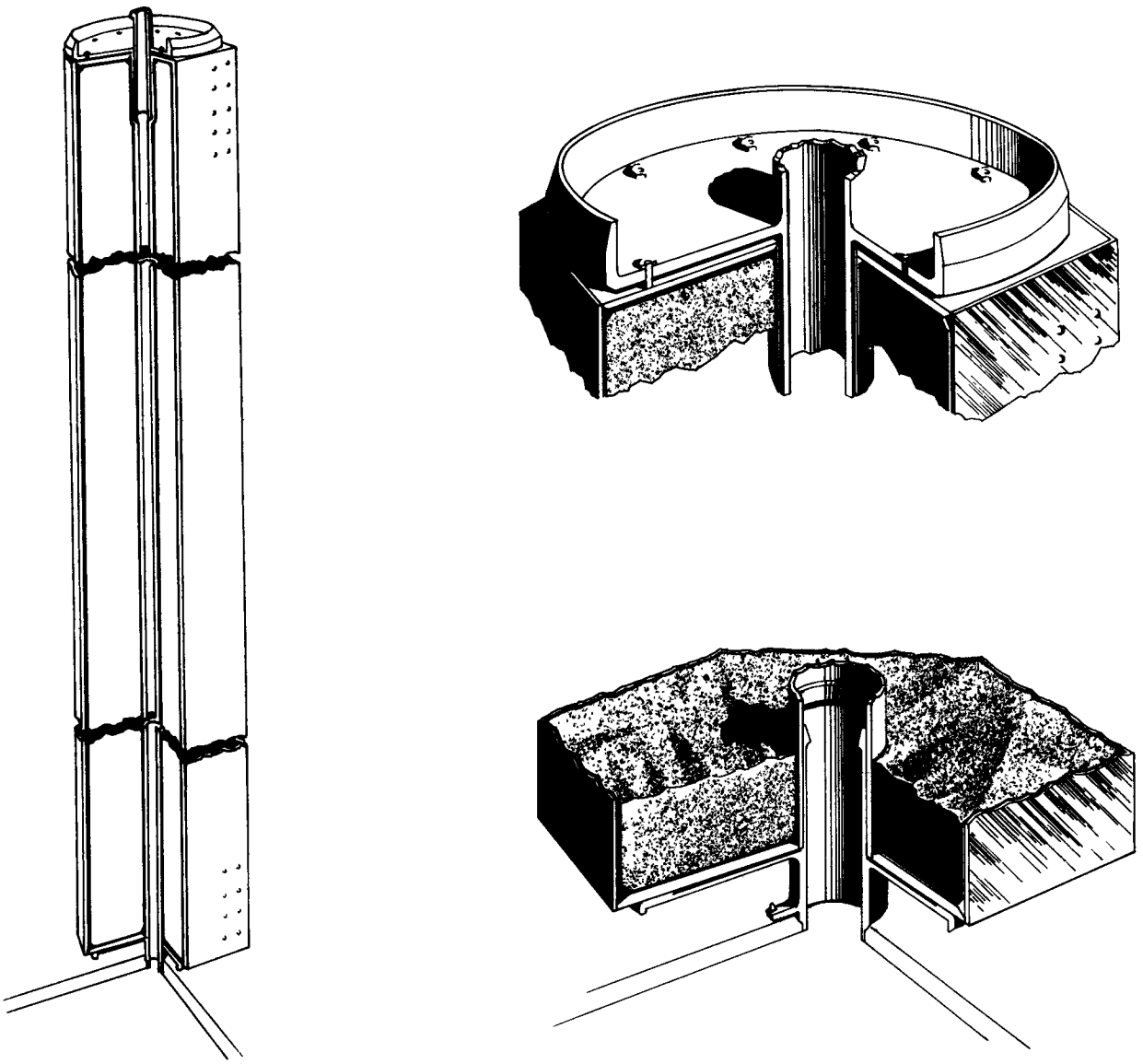


Fig. 10. Cutaway View of Moderator Can Assembly

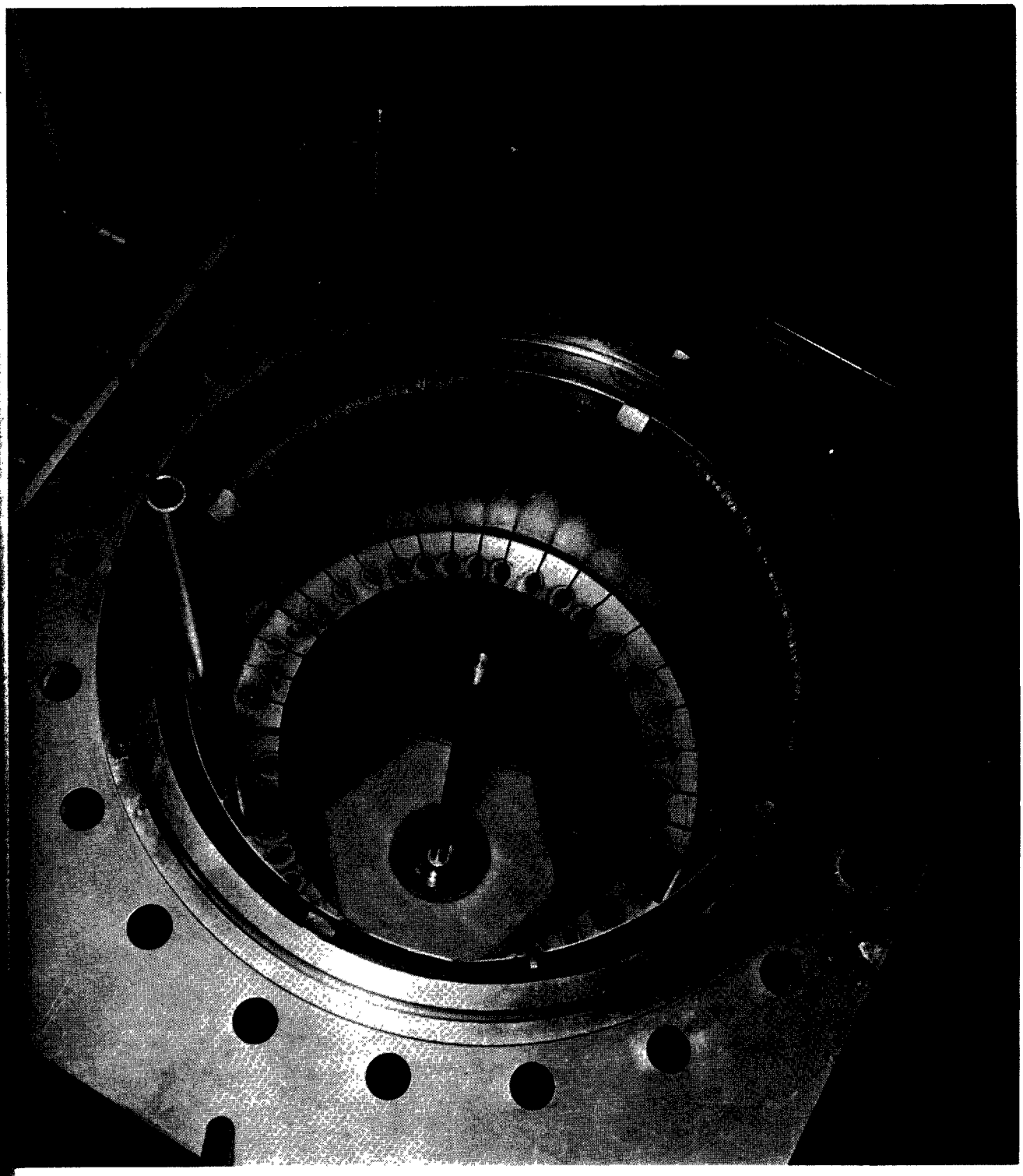


Fig. 11. Can Head Flexure Test

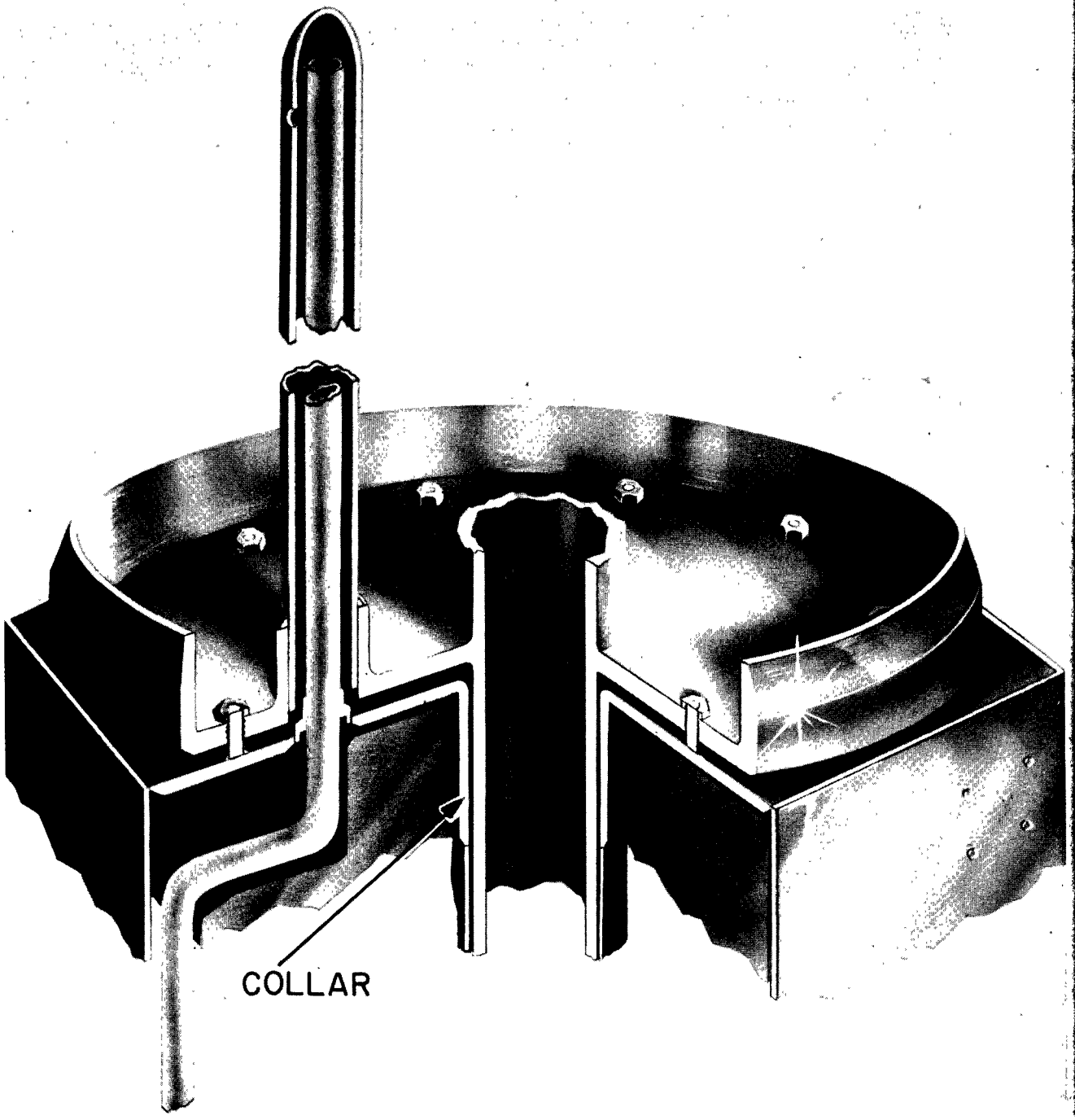


Fig. 12. Top Detail of Moderator Can

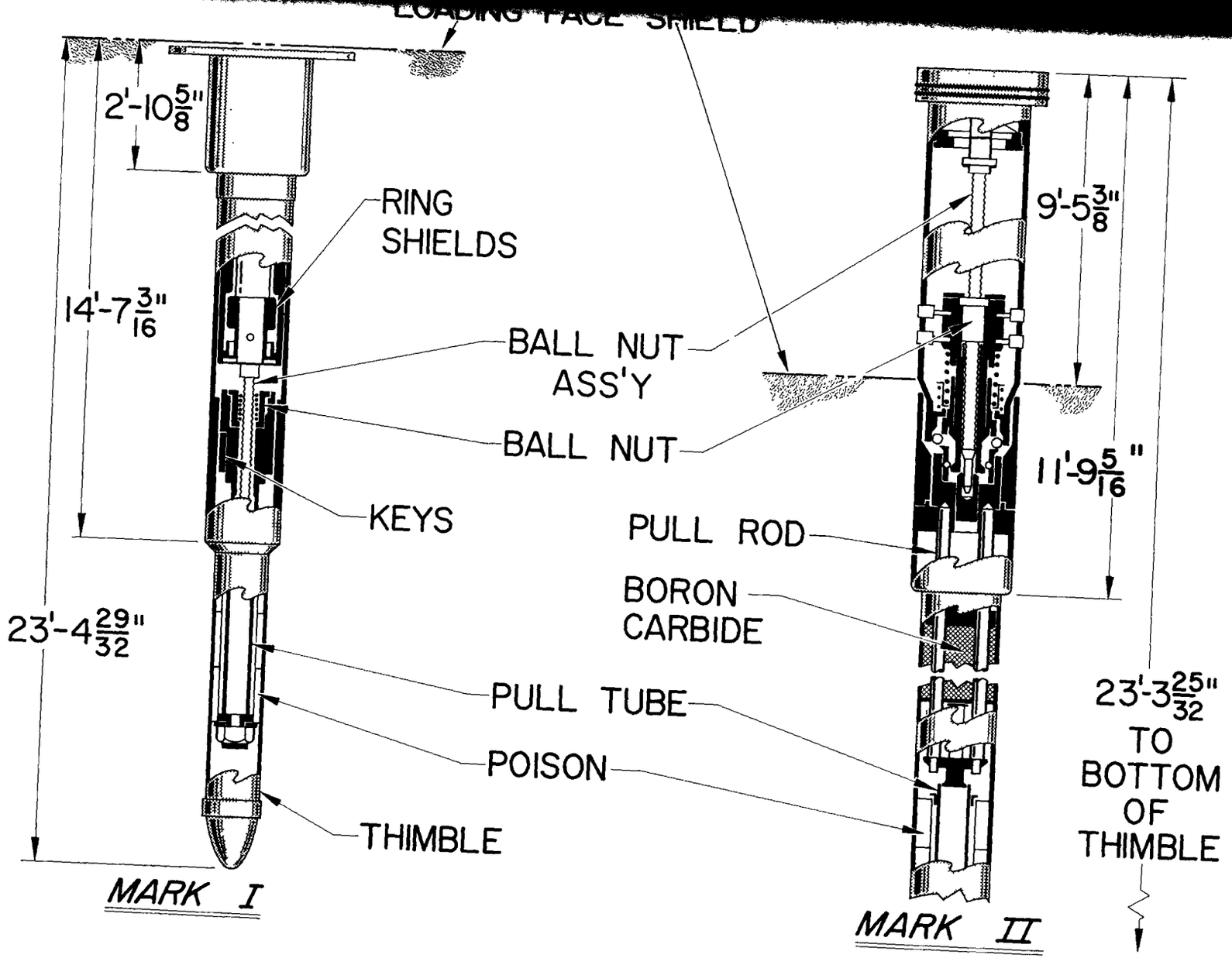


Fig. 13. Sectional View of SRE Control Rod

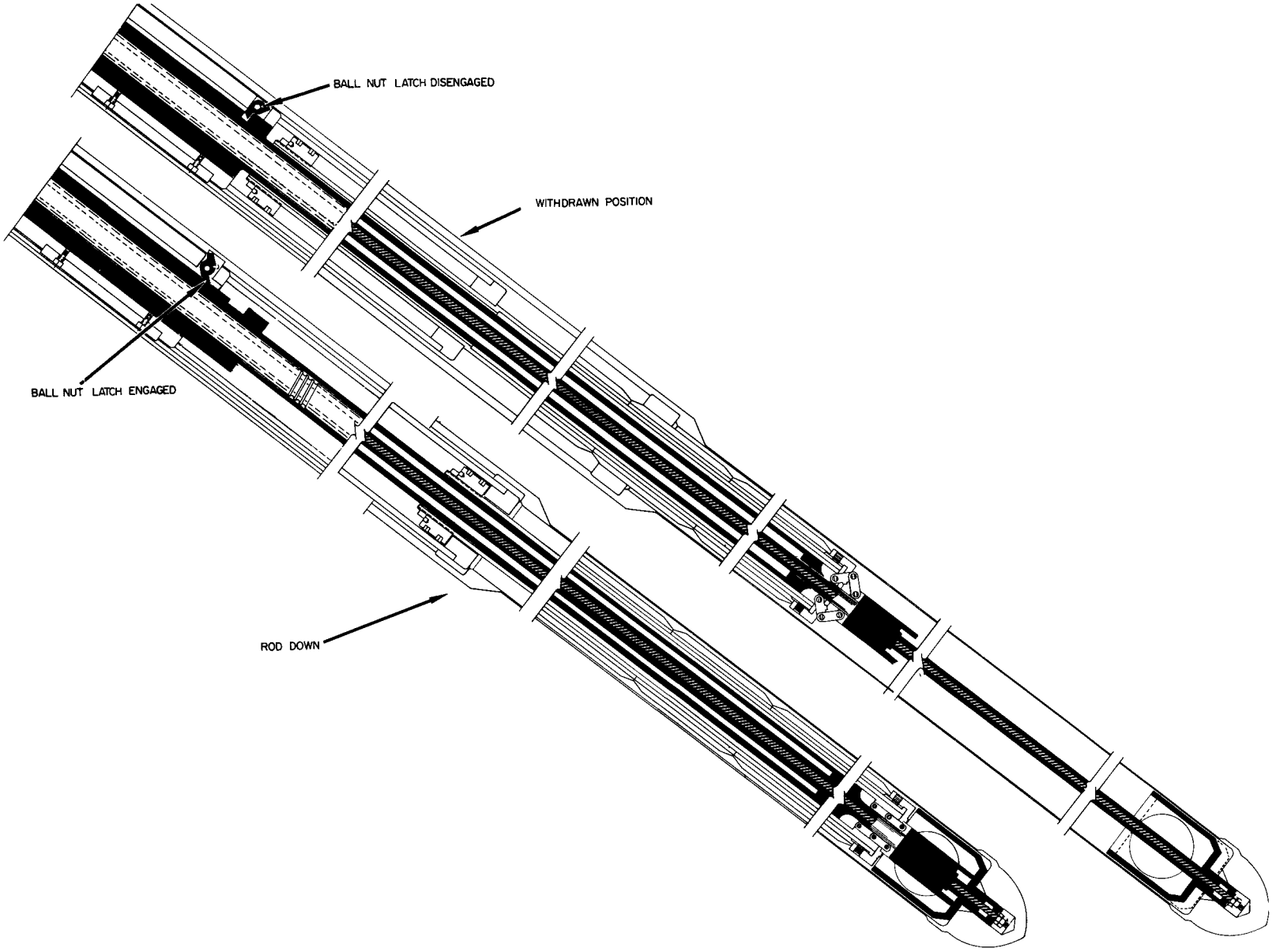


Fig. 14. Sectional View of SRE Safety Rod

Fig. 14. Sectional View of SRE Safety Rod

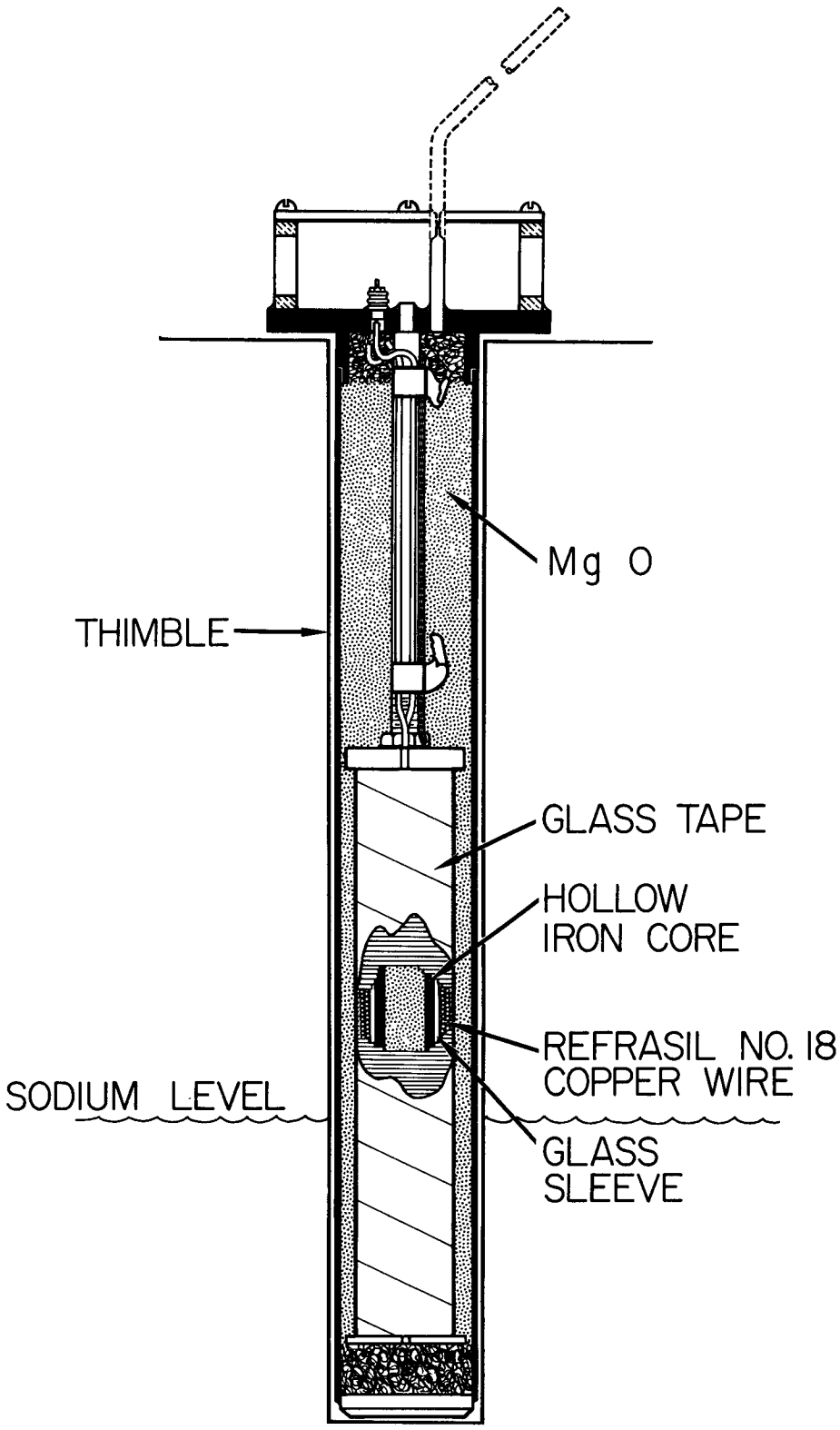


Fig. 15. Sodium Level Indicator

Fig. 16. Component Test Tower

Fig. 16. Component Test Tower

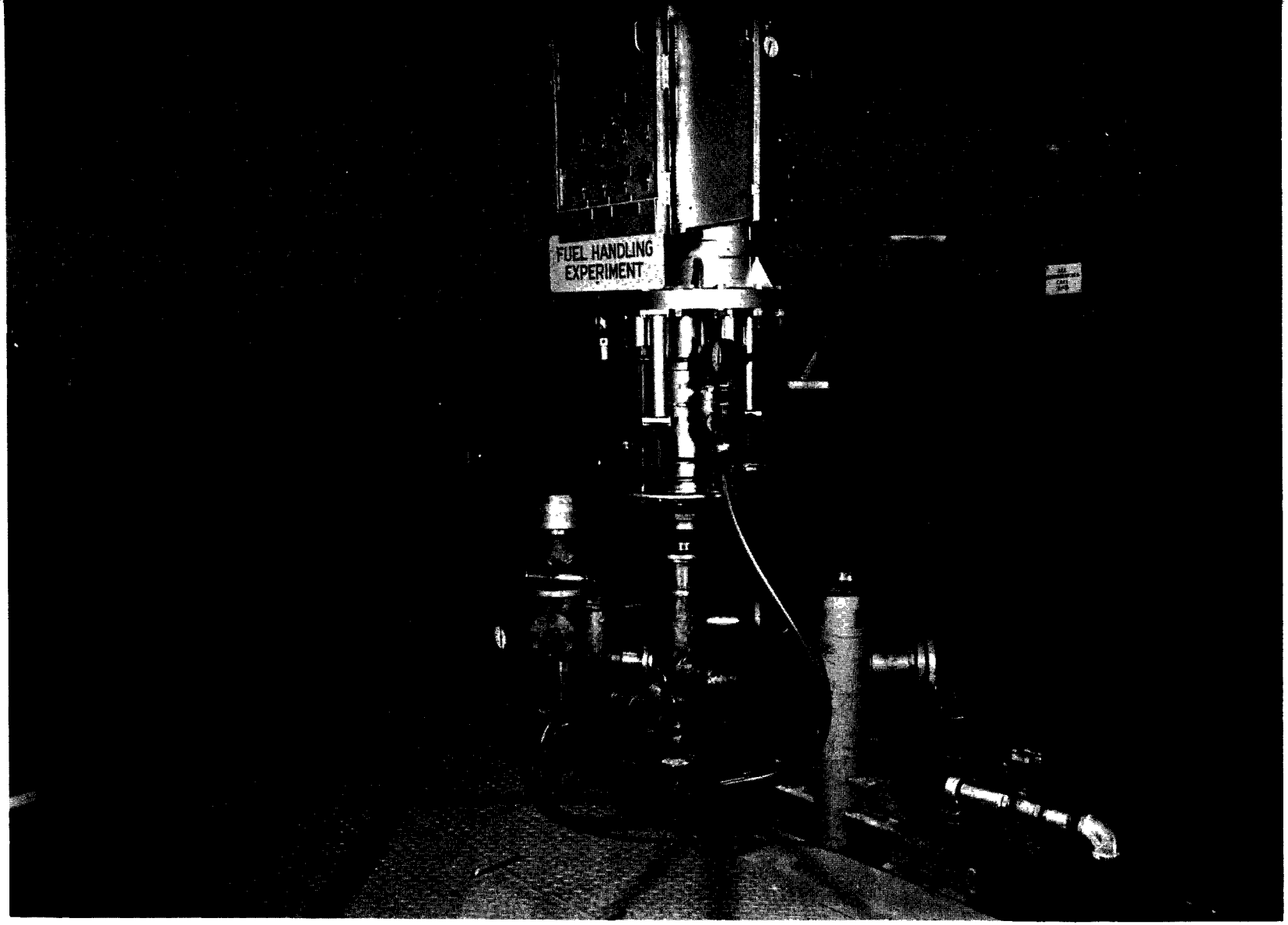


Fig. 17. Remotely Operated Coupling Device

EXPERIMENTAL FUEL MATERIALS

C. C. Woolsey

Both uranium base and thorium base metallic, slug type fuels have been prepared on a semi-production scale for study in the sodium graphite reactor. The experimental fuels include both design and material (alloy) variations. The major fabrication methods used were powder metallurgy, centrifugal casting, and extrusion. For each type of fuel material, a thorough chemical and metallurgical fabrication history has been obtained. A maximum fuel temperature of 1200° F will be used for most operating conditions. In addition, a few fuel elements will be tested at temperatures up to 1800° F. Fuel elements will then be removed at scheduled increments of burnup and examined for dimensional and metallurgical changes in the adjacent hot cell.

I. INTRODUCTION

A major economic consideration in any reactor system is that one associated th the behavior of the fuel elements. The nature of the fuel materials pose rtain limitations on the temperatures at which the reactor may be operated and the length of time during which the fuel element may be left in the reactor. ese have been covered in great detail in the literature and will be passed over re. The objectives of the SRE experimental fuel program are to evaluate these nitations for the basic project-grade fuel material, wrought and beta-heat-treated anium, and to develop improved fuel materials, both from the standpoint of in-eased fuel element life (or burn-up) and from the standpoint of increased operat-g temperatures.

The experimental fuel program will make extensive use of the SRE as a fuel radiation facility; in fact, after initial criticality and low-power operational udies, about 25 per cent of the fuel in the SRE will be experimental. These fuel aterials have been designed to carry their own weight from a reactivity stand-int and should neither detract from nor add to the nuclear performance of the actor in more than a superficial manner.

II. DESCRIPTION AND LOCATION OF FUEL ELEMENTS

The SRE fuel elements have been described in the sections entitled "SRE ngineering Description" and "Reactor Component Development." With one ex-ption, the experimental fuel elements use this same seven-rod-cluster type lements, merely substituting a different fuel material for the wrought and beta-eat-treated uranium slug. In the experimental elements, each rod of the ele-ent will carry a different experimental material. At intervals of fuel burn-up, element will be removed, disassembled, and the fuel slugs carefully examined r degree of damage. The elements are so designed that any of the six external el rods may be replaced; thus, if it appears that one particular experimental el material may be behaving erratically, that one may be replaced on other lements and their irradiation may be continued.

Figure 1 gives a plan of the reactor core, showing the different types of xperimental elements and their location in the core. The central fuel element,

which will achieve the most rapid burn-up, will be made up entirely of thorium - 5.4 per cent uranium alloy, the uranium being fully enriched in U^{235}. This fuel element will be used primarily for nuclear physics studies of such a fuel materia but it will ultimately become available for metallurgical studies after it has been irradiated to a relatively high burn-up.

The first row of fuel elements around the central one will include five experimental elements and one standard element, the latter also being primarily for nuclear physics studies. Several types of experimental elements will be included in the program:

1. The element described in the upper right of this figure is designed to permit a study of several experimental fuel materials as well as to allow monitoring of the standard fuel material up to 2500 Mwd/ton burn up and at temperatures below the alpha-beta phase transformation. It will be noted that all experimental fuel slugs have the same dimensions as the standard fuel except those made by powder metallurgy techniques For the seven-rod-cluster type element, the latter are only 1 inch in length, thus requiring six times as many slugs for a fuel rod.
2. The element described in the upper left corner of the figure has been designed to attain higher burn-up but with the same temperature limitation as the first series of elements. It will be noted that the standard beta-treated uranium fuel and the experimental cast unalloyed uranium fuel slugs have been replaced by alloy materials.
3. The element described at the lower left is designed to attain high burn-up and to operate at temperatures above the alpha-beta phase transformation. The increased central temperatures will be achieved by increasing the enrichment of the alloy fuel slugs to 4.5 per cent U^{235}, thus increasing the heat generation in these slug. To maintain uniform power generation for the element, only four high-temperature rods are used, the other three rods using normal uranium. It is expected that central fuel slug temperatures as high as 1350° F will be attained.
4. The fuel element described at the lower right represents the departure from the seven-rod-cluster type. This is a single, hollow element which has the same total volume of uranium is in a seven-rod element. It is made up of a series of hollow slugs, each 4 inches in length, mad

by hot pressing of unalloyed powdered uranium, and is clad both internally and externally with thin-walled stainless steel tubing, with a NaK bond between the fuel and the cladding.

5. An additional type of experimental element not described in the figure will be one to examine the behavior of thorium-uranium fuel materials at temperatures up to 1800° F. These higher temperatures will also be obtained by increasing the enrichment - in this case, by increasing the concentration of fully enriched uranium in the thorium-uranium alloy. Again, uniform power generation in the element will be achieved by using only a few high-temperature rods with the balance made up of unalloyed thorium.

III. BURN-UP SCHEDULES

Figure 2 gives a schedule of fuel element residence in the reactor, both for e experimental elements and for the standard wrought and beta heat-treated ements. The burn-up schedule assumes full-power operation with one shift per ek shutdown time. Time zero for this schedule is at the beginning of full-power eration or very nearly so. It is not expected that the experimental fuel clusters ll be inserted until reactor operations reach that point, as it is desired that ry little low-temperature exposure be accumulated on the experimental fuels.

The central fuel element, in Channel 44, is fueled with thorium-5.4 per cent anium and will be used primarily to study changes in reactivity with burn-up. is expected that it will receive about 16,000 Mwd/ton burn-up in the two-year hedule presented here.

The next six channels listed, 45 down through 56, which are those immediely surrounding the central one, will be used for most of the experimental fuel posures. Average burn-up for uranium fuel materials in this ring of elements ll be at the rate of about 3400 Mwd/ton per year. The designations MC-1-1, C-2-1, MC-3-1, etc., refer to the mixed cluster-type elements described rlier. Referring to Fig. 1, the MC-1 series was that type shown in the upper ht corner, the MC-2 series in the upper left corner, and the MC-3 series in lower left corner. The MC-1 and MC-2 elements will permit the study of the havior of fuels at temperatures below the alpha-beta phase transformation.

These elements will be removed for evaluation at burn-ups of 750, 1800, 3000, 4500, and 6750 Mwd/ton. The MC-3 clusters will contain the more highly enrich uranium alloys and will operate with peak temperatures up to 1350° F. One will be removed for evaluation at 750 Mwd/ton and a second one will be allowed to go to 4950 Mwd/ton.

The HC elements in Channel 56 are the hollow cylinder elements described at the lower right corner of the Fig. 1. Since these are fueled with unalloyed uranium, burn-up will be limited.

Channel 46 and those standard fuel elements represented by Channel A constitute the second ring of fuel elements. Uranium fuel material in this ring will burn-up at an average rate of about 3000 Mwd/ton per year. Channel 36 and the Channel B standard elements constitute the third ring, where uranium fuel burn-up will be at an average rate of about 2900 Mwd/ton per year. The high temperature thorium-uranium fuel elements will be exposed in Channel 36.

The selection of types of slugs and of alloying materials for the experimen program was based on a number of considerations. First of all, for the uranium base fuel materials, it was desired to keep the uranium atomic density as high as possible, for nuclear reasons. Maximum alloy content was arbitrarily set at 5 atomic per cent. With this limitation, it was calculated that the effect on uranium enrichment was so slight that no adjustment need be made. Thus, the feed material for preparing these alloys was the same material that was used for the standard wrought and beta-treated slugs.

The particular uranium alloys included in the program were selected as a result of a fairly extensive thermal-cycling program carried out in our laboratories and of irradiation tests reported by other laboratories, particularly those from Argonne National Laboratory. The thermal-cycling work has been reported earlier* and will not be gone into in detail here other than to mention that the investigation was directed toward learning the effects on the fuel materials of the thermal cycling that occurs in reactor operation. Included were studies of therm cycling into the beta phase.

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*B. R. Hayward, "Dimensional Changes Resulting From Alpha-Beta Thermal Cycling of Uranium and Uranium Alloys," NAA-SR-1434, March 15, 1956.

IV. FABRICATION TECHNIQUES

Figure 3 lists the experimental fuel materials, together with their mode of fabrication, their alloy composition in weight percentages, their enrichment, and the numbers of pieces that will be used in the program.

Centrifugal casting was selected as the technique for producing most of the experimental uranium alloys because of the excellent results reported by ANL on the behavior of such cast slugs in the EBR. Additional advantages of the casting technique are its low cost and its potential as a method for the remote refabrication of slugs from fuel material which has been partially decontaminated by pyrometallurgical reprocessing techniques.

National Lead Company of Ohio cooperated with AI in the development of techniques for casting slugs of the SRE dimensions, patterned somewhat after the method developed at ANL for casting fuel slugs for the EBR. The group of experimental fuel slugs, using uranium of the same enrichment as the standard wrought fuel slugs, have been cast. These include unalloyed uranium and uranium alloys containing 2 per cent zirconium, and 1.2 per cent and 1.8 per cent molybdenum.

Some uranium alloys prepared by powder metallurgical techniques have shown excellent behavior both in thermal-cycling and irradiation studies. It appears that future potential costs would be greater for these than for cast fuel slugs and that they would be competitive only if reactor performance were appreciably better. Thus, only one alloy prepared by powder metallurgy will be studied. This one, containing 1.2 per cent molybdenum, was fabricated at Sylvania Electric Products.

At the time of planning for the hollow slug type fuel element, powder metallurgy techniques seemed to be the best advanced along this direction. Thus, these slugs were prepared by hot-pressing of powdered uranium, also at Sylvania. Since costs might be improved by using casting technique, some effort is being directed toward developing a satisfactory process for casting such hollow slugs for future experiments.

All slugs at the 2.78-per cent enrichment level have been received and soon will be loaded into fuel rods.

The last two uranium alloys are to be made by centrifugal casting, with th uranium enriched to 4.5 per cent U^{235}. As mentioned earlier, these will be use to obtain fuel temperatures above the alpha-beta phase transformation. These slugs will be used somewhat later in the program; procurement has been initiate only relatively recently.

The final group of fuel materials in this figure are the thorium-uranium alloys, which are being studied for possible use in a thermal breeder reactor. These alloys are being prepared by extrusion techniques primarily because there had been some previous experience in preparing such alloys in this manner; thus a minimum of development work has required to produce satisfactory fuel slugs for the program. The group of slugs containing 5.4 per cent uranium has just been completed at Nuclear Metals and will soon be received for loading into fuel rods.

The two other thorium-uranium alloys will be used to study the behavior at higher temperatures - up to 1500° F in the case of the one containing 11.2 per cent uranium and up to 1800° F for the one containing 16.4 per cent. Again, as with the high temperature uranium alloy studies, these will be used somewhat later in the program; procurement arrangements are under way.

V. HOT CELL OPERATIONS

After removal from the reactor, the experimental elements will be cleaned of sodium and then transferred to the hot cell in the reactor building. The hot cell is so arranged that the fuel element can be lowered into it directly from the handling coffin in the same manner that it would be lowered into the reactor or into fuel element storage cells. The working face of the hot cell area is shown in the Fig. 4. One pair of master-slave manipulators is shown; two more pair will be installed in the holes seen to the right and left of the existing ones. Three hig density lead glass windows can be seen; the vertical one at the extreme right is directly in front of the position into which the fuel element is lowered from the handling coffin.

The fuel rods will be disassembled from the element in this hot cell and the fuel slugs will be decanned, using the device shown in Fig. 5. On removal from

he element, a fuel rod is fastened to an angle-iron fixture to support it for handing purposes. It is then laid in the trough of the decanner, immersed in kerosene or Ultrasene, and the jacket is slit longitudinally by an abrasive wheel mounted on the carriage which rides along the track behind the trough. The individual slugs are removed and cleaned in an ultrasonic cleaning bath. They may then be examined for surface condition, measured for distortion, and photographed. Selected slugs from each rod will be prepared and examined for effects of the irradiation on hardness and microstructure.

This picture was taken with the decanner in place in the mock-up hot cell, a wooden structure of the same dimensions as the hot cell itself. Remote operations are developed and initial "cold" operations are carried out in this cell prior to installation in the hot cell at the reactor building. With the use of these mock-up cells, equipment development for the hot cell operations has been under way for more than a year and will be reasonably complete by the time the first elements are ready for examination.

VI. IRRADIATION EXPERIMENT AT MTR

The lack of information on the behavior of fuel materials on irradiation at SRE temperature conditions stimulated the design of an experiment to obtain such information, using the MTR at Idaho as an irradiation facility.

An assembly of six capsules, seen in Fig. 6, each containing a 3/8-inch diameter by 1-1/2-inch long specimen of an SRE fuel material, is at present in the MTR. The heat-transfer characteristics of the capsule design are such that the relationship between fuel central and surface temperatures should be similar to that in the SRE at the point of maximum fuel temperature. Each specimen has a thermocouple located at its center. This experiment failed to reach the 1200° F temperature desired but has attained a peak temperature of 1050° F. It will be removed on accumulation of 0.5 per cent burn-up, expected to be attained about the middle of December, and will be brought to the SRE hot cells for examination. This will give a preview of what we might expect in the behavior of both the standard SRE fuel material and some of the experimental fuel materials, and will guide the planning of the exposure schedule. The SRE itself, of course, will be the most important experiment.

VII. REMARKS

In conclusion, it might be mentioned that already additions to the experimental fuel program are being planned. There is need for additional development work to improve methods for fabrication of the fuel materials in the present program. Along this line, work is under way to develop injection casting for direct casting of SRE type slugs both in uranium and thorium base alloys. Fuel materials being considered for future investigation include UO_2 in thorium and urania-thoria ceramic bodies.

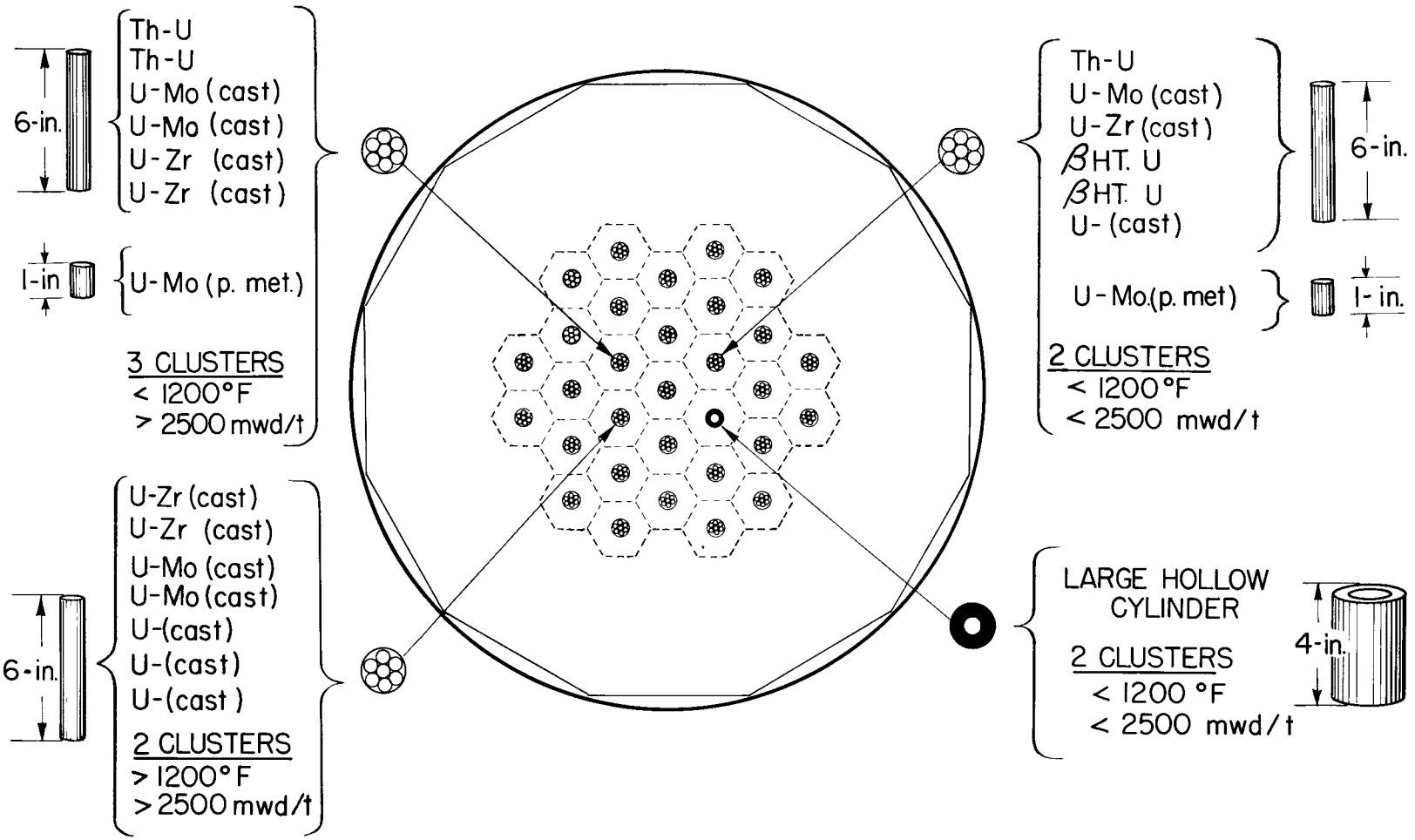

Fig. 1. SRE Core Plan, Showing Experimental Elements

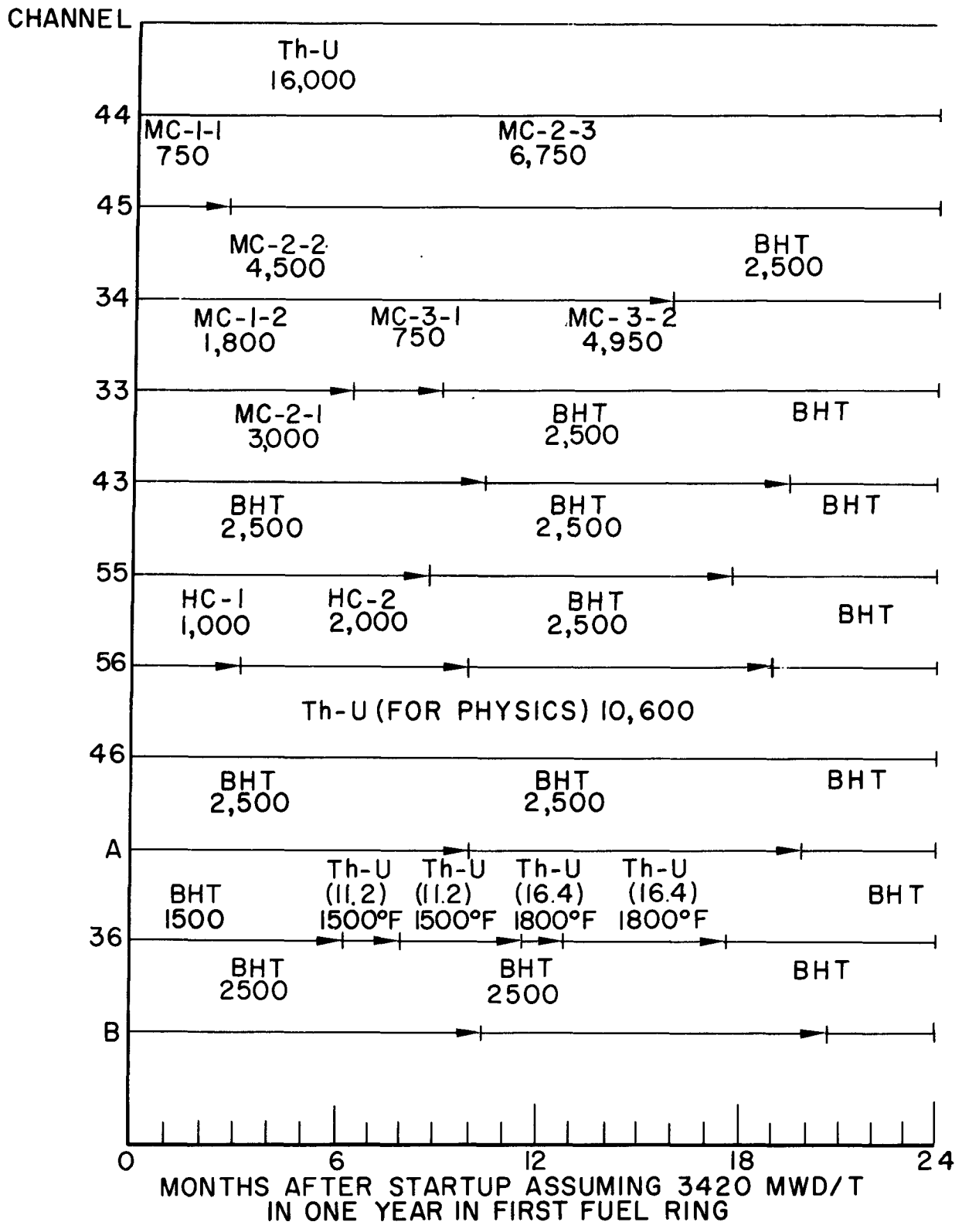

Fig. 2. Schedule of Fuel Element Residence

TECHNIQUE	MATERIAL	QUANTITY	U ENRICHMENT
Powder Compacted	U-1.2 % Mo	515	
	U-Large Hollow	36	
Cast	U-2.0 % Zr	215	2.78%
	U-1.2 % Mo	75	
	U-1.8 % Mo	65	
	U	25	
	U-1.8 % Mo	80	4.5%
	U-2.0 % Zr	80	
Extruded	Th-5.4 % U	450	Max.
	Th-11.2% U	110	
	Th-16.4 % U	75	

Fig. 3. Experimental Fuel Materials

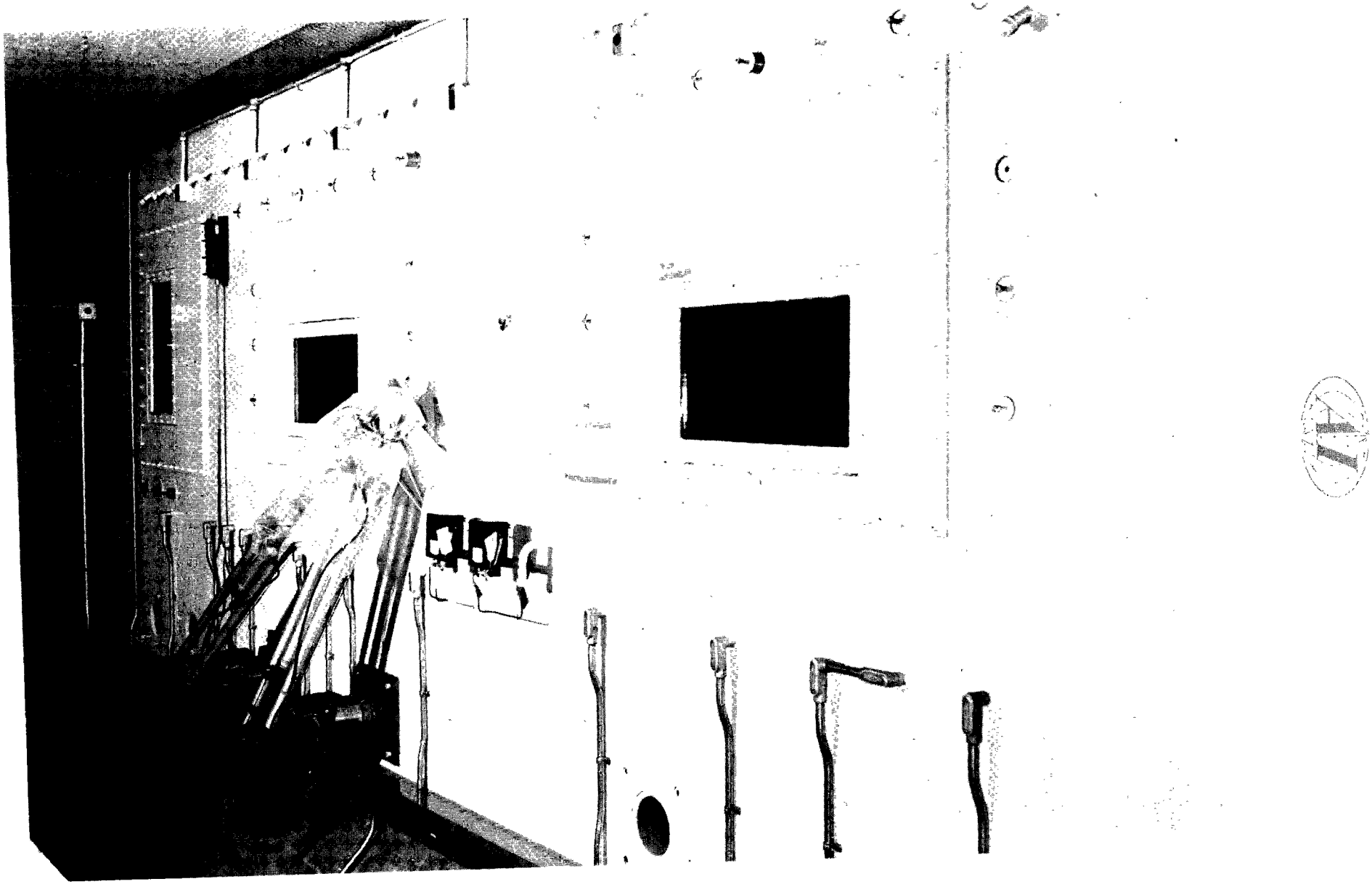

Fig. 4. Working Face of Hot-Cell Area

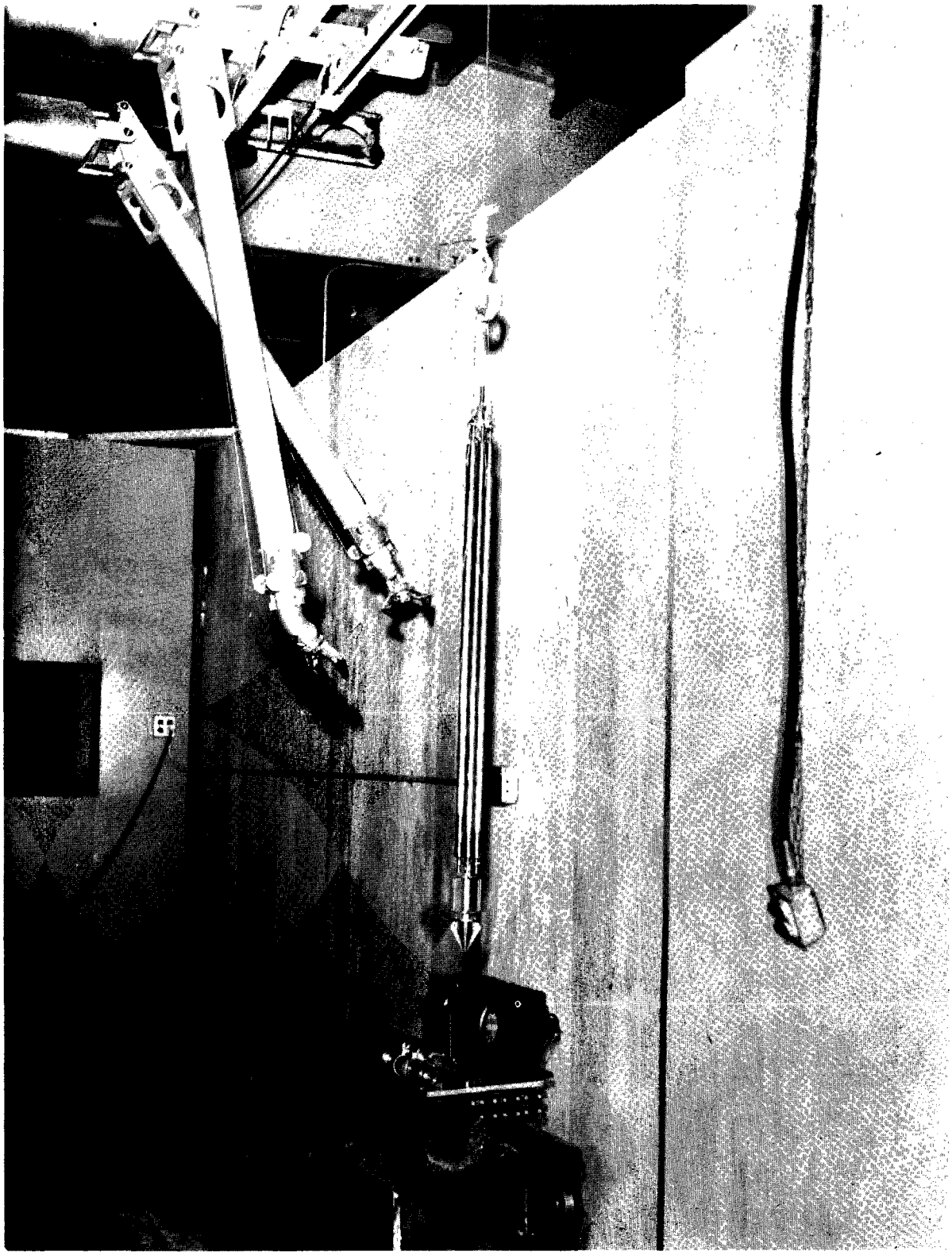

Fig. 5. Fuel Rod Disassembly Equipment

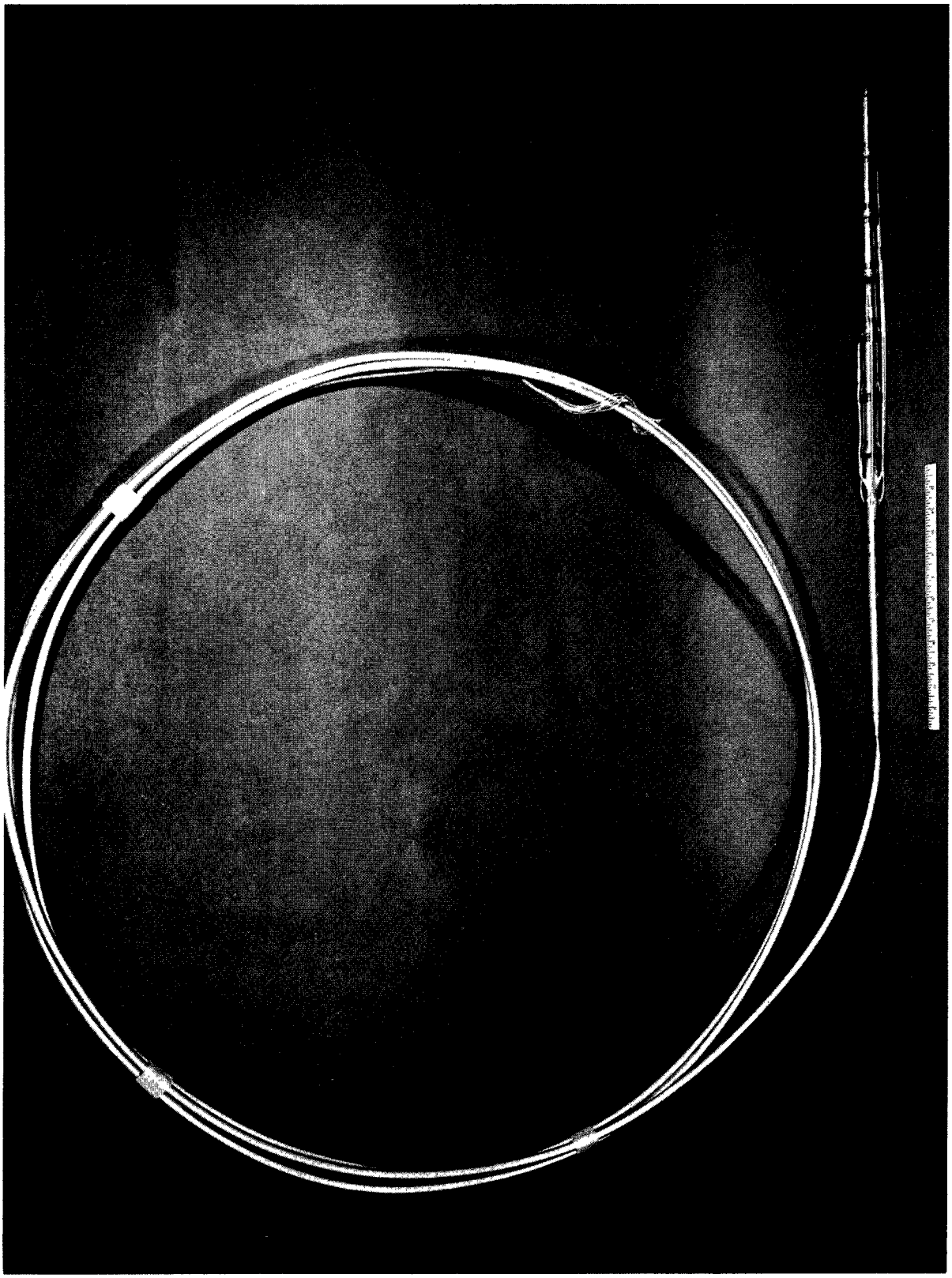

Fig. 6. Experimental Irradiation Assembly

SRE ZIRCONIUM PROBLEMS

R. L. Carter

Calculations of stresses in the SRE moderator cans show critical regions t the tube-ends and the heads of the scallops. Engineering tests have shown rratic and early fatigue failures in these regions. The erratic behavior is be- leved to be due to stress-risers; the early failures to ill-defined property changes nduced by fabrication treatments.

Tensile strengths and fatigue limits of oxygen-oxidized and oxygen-rich- odium-oxidized zirconium and of hydrogen-containing zirconium have been easured and correlated. Grain-coarsening effects on these properties have lso been studied. Surface oxide above 0.2 mg/cm^2 or grain sizes above 10 mils ause a severe reduction in the fatigue limit.

Parabolic rate law behavior is observed for oxygen absorption by zirconium xposed in liquid sodium below 1000° F with an oxygen concentration above 20 ppm. he observed activation energy is about 29,400 calories per mole. Hydrogen bsorption kinetics appear to be similar, but are complicated by the presence f oxygen in zirconium.

In the Sodium Reactor Experiment, the unwanted consequences of the grain rowth process are being controlled through the limitation of sodium temperatures o values below 950° F where exaggerated grain growth has not been observed, nd through the use of cold traps to precipitate hydrogen from sodium and hot raps to getter oxygen from sodium.

169-186

LM-00984

I. INTRODUCTION

Previous speakers in this Forum have discussed at length the characteristi of the SRE, its structure, the testing of its components, and, to some extent, the nature of the shortcomings which these either have or might easily develop. As pointed out by Dr. Parkins, certain changes have been made in the initial design of the moderator cans. Dr. Eggen detailed in a more specific way the reasons for these changes, the tests which have been made on prototype elements having revealed the points at which maximum probability of failure occur. He also discussed to some extent the analytical stress studies which have uncovered the points at which difficulties are most likely. Mr. Peterson mentioned the proble of 0.004-inch thick zirconium fabrication for the SRE hot trap device which Dr. Eggen described in some detail. The rate of oxygen pick-up by the zirconiu containers which surround the moderator elements is reduced by the employment of this device.

It is the purpose of this paper to discuss the problems that are met in the use of zirconium metal in contact with reactor coolant sodium, the ways these problems have been studied, and the results which have been obtained to date. It became evident very early in the consideration of the materials problems confronted in the SRE that any physical failure threatening the zirconium metal woul have as its source a combination of flexural fatigue effects and tensile stresses. Our principal plan of action therefore became a judgment of the seriousness of any particular zirconium composition change - first, in light of loss of fatigue life, and second, in consideration of the reduction in tensile strength of zirconiu To this end, samples of zirconium were subjected to the particular conditions of interest simulated in sodium test loops.

II. SODIUM-ZIRCONIUM STUDIES

This program was carried out through the use of dynamic sodium loops with experimental exposure facilities in them. In these loops, sample tabs of 35-mil can stock 1/2 inch in width by 2 inches in length were exposed to sodium at temperatures ranging from 750° F to 1200° F. In these loops, it was possibl by means of cold traps and hot gettering traps, to vary the concentration of oxyg

d hydrogen in sodium (as well, possibly, as the concentration of other trace ntaminants). In order to examine the changes in mechanical properties that curred in these samples during exposure, a device was set up for fatiguing mples by repeated bending at any temperature from room temperature to 1000° F. tensile test device permitted the determination of the yield point and the failure int for samples which had been exposed under various circumstances in the posure loops. The changes in these physical properties were taken to have a rect relation to the changes experienced in large bodies composed of zirconium et, parts of which will suffer corresponding degrees of fatigue and tensile ess under operating conditions in the SRE.

In order to make it possible to identify the materials in the sodium whose nsfer to zirconium will lead to significant changes in the mechanical properties he zirconium, vacuum-fusion analytical equipment was set up. Thie equipnt permits accurate determination of the amount of oxygen and hydrogen in conium samples before and after exposure in the experimental loop facilities. ng this device in conjunction with the actual measurement of weight changes he samples, it has been possible to develop an expression for the rate of conination by oxygen from sodium. Activation energies for oxygen gain have n determined. It is anticipated that diffusion data will be obtained in the near ure. Metallographic studies of the changes accompanying exposure to loop ium, and after flexure fatigue experience, have been conducted.

III. SODIUM EXPOSURE FACILITIES

The sodium exposure facilities set up for this study are elaborate devices ich in many ways resemble the actual circulating loops present in the primary d secondary sodium cooling systems of the SRE. Figure 1 shows a photograph a sodium loop (with the thermal insulation removed) in which the components d their relative disposition are shown. Figure 2 shows a line diagram of a ld-trapped sodium loop. We observe that the sodium is pumped through a series heaters and a vessel in which exposure samples are placed so they can be adily removed during operation of the loop. In the balance of the loop are loted the devices for trapping out oxygen, lowering the temperature of the sodium, d measuring the concentration of oxygen present in the loop (accomplished by

means of the formation of an oxide plug in a set of small orifices). This facility has been used to obtain a great deal of the data upon the changes in properties of zirconium samples during exposure to liquid sodium over long periods of time, ranging from a few days to several months.

A very similar facility has been used to complement these data (see Fig. 3) and has been referred to as a "static pot." However, the sodium within it is not static because of the very strong convection currents which cause continual circulation through the turret which holds the zirconium samples being exposed. The data obtained with this loop, which is cold-trapped like the one illustrated in Fig. 2, have been quite consistent with the data obtained with the former facility; in fact, so much so that recently the latter experimental facility has been diverted to a special set of experiments concerned with the growth of grains with various types of samples over long periods of time. This facility ought to be well accommodated for this because of the fact that it is fool-proof and easier to operate and yet the data obtained are much superior to those obtained through use of samples encapsulated within relatively small volumes of sodium. In the case of such small sodium volumes, one runs a severe risk of varying the conditions of exposure during the time that exposure is taking place, through the removal of a significant portion of the total quantities of impurities initially present in the sodium by the getter action of the samples.

Figure 4 shows a line diagram of the hot-trapped sodium exposure facility. This is essentially similar to the cold-trapped dynamic loop illustrated previously with the exception of the fact that a hot-trap gettering device containing a zirconium titanium 50-50 alloy* is included in the system. All of the sodium pumped through the system must pass through this device. Recently there has been added to this system a cold trap so as to simulate a little more closely the circumstances present in the SRE, in which it will be possible to use both hot-gettering device and cold trap to remove impurities from the sodium. In the case of the hot-trapped loop, of course, the sodium must be cooled when leaving the hot trap, since the hot trap is generally the highest temperature point in the system.

*Kindly supplied by the United States Bureau of Mines, Albany, Oregon.

IV. RATE DATA

Data have been collected over a period of nearly a year through the use of these loop exposure facilities. The changes experienced by zirconium samples when exposed to hot sodium under a welter of circumstances were determined. The earlier data had to be used with some caution because relatively little knowledge was available on the portion of the weight gain which was attributable to hydrogen and the portion attributable to oxygen. Such data, obtained with cold-trapped sodium, were found to fit well a simple behavior with time. For exposures ranging from 750° F to 1000° F, the weight gains varied parabolically with time, according to the expression $w = c \exp(-A/kT)t^{1/2} = (Kt)^{1/2}$. The rate factors of these curves were determined on the basis of best fit, and these, when plotted logarithmically against the reciprocal of the absolute exposure temperature (Fig. 5) were found to yield a reasonably consistent activation energy. This activation energy, presumably for the pickup of oxygen, was found to be about 29-1/2 kilocalories per mole. Some confusing deviations were found, however, for the case of material which had been prepared specifically for the moderator cans and the surface of which had not been chemically cleaned before the test began (Fig. 6). However, it was possible to show that this material apparently had a surface contaminant, the nature of which is still not known, but which entered into the reaction to the extent of causing an early inordinately high weight loss, as shown in the figure. It is seen by comparing the weight gained by a sample of this type with that gained by a non-vapor-blasted sample of the sort upon which the earlier data were based that there is no inherent difference in the weight-change rate other than the early loss of a small quantity of material, ($\sim$0.35 mg/cm^2). A small amount of weight-gain data was also obtained for the zirconium-titanium alloy, enough to indicate that the rate of contamination here was a factor three to four higher than that for zirconium metal of equal surface area in the range of temperatures examined. Samples of Type 304 stainless steel exposed showed virtually no effect after exposure ranging up to 2000 hours.

Recent data obtained with the vacuum fusion analytical device have supported the belief that in technical grade of as-fabricated zirconium the weight loss experienced is apparently a true loss of material originally present on the sample. The increase in oxygen and hydrogen content found upon post-exposure sample

analysis equals the weight gain in the case of all samples not showing the weight loss anomaly. In the anomalous cases, an increase in oxygen and hydrogen is observed even though there may be a net weight loss.

V. HOT TRAP EFFECTIVENESS

Variation in the oxygen content from 20 ppm to 60 ppm was found to have little effect upon the rate of weight gain, particularly at lower levels (Fig. 7). However, analyses of zirconium samples exposed in a loop facility utilizing a hot-trap getter have shown no measurable increase in oxygen content after 1000 hours exposure at 1000° F. Under these circumstances, there is measurable transfer of hydrogen, with an apparent tendency toward achievement of hydrogen equilibrium in the system. Hydrogen shows the highest concentration at points of lowest temperature, as long as rates are high enough to permit equilibration.

VI. FATIGUE RESULTS

Fatigue data were obtained by setting up a rather arbitrary procedure in which a standard size sample of 30 mil thickness was oscillated at a fixed rate; this rate, which was about one cycle per second, had an amplitude which was als arbitrarily chosen to be approximately ten per cent beyond that at which yield occurred at room temperature. These tests were made first at room temperatu as a function of the amount of weight gain experienced by the samples upon exposure in a sodium loop; they were also made for the case of the weight gain experienced upon exposure in gaseous oxygen for somewhat shorter time. In both of these cases, it was determined that the fatigue life dropped rapidly with the presence of an oxide surface coating which accompanied both types of exposure to oxygen. In a manner shown in Fig. 8, these tests were extended to 500° F an later to 985° F, whereupon a less extreme variation to reduced fatigue life was likewise experienced with increasing oxygen content of the samples. Since it wa noticed that the fatigue life was still somewhat in excess of that which is being experienced currently in the can-head test program, the severity of these tests was increased by increasing the amplitude by 30 per cent. The curves obtained

re similar to those shown in Fig. 8, but the values fell to somewhat lower over- fatigue life, agreeing in magnitude with the engineering test-life expectancy.

At about the time these results were being neatly tied up, we looked at some mples which had been exposed in the hot-trap loop. It was found that little or weight gain was experienced which was not attributable to hydrogen. Now, ving the effect of hydrogen up to discussion shortly, I want to point out that an tially unaccountable decrease in fatigue life was experienced. Upon metallo- phic examination, this change was found attributable to a radical growth of casional grains which was found to have occurred in samples which had been posed to temperatures of around 1000° F for periods of the order of 300 hours more. Upon the discovery of this phenomenon, a rather extensive study of ain growth and its causes in this case was undertaken. It was determined that effect of grain growth and oxide coat upon reduction of the fatigue life expect- cy were cumulative, both being definitely undesirable from this point of view. wever, it was found, both from the results of loop exposures and from the sults of tests made upon samples deliberately contaminated with hydrogen, that tle or no fatigue life reduction was attributable to the presence of hydrogen, rticularly if this hydrogen were dissolved.

It is already evident that the problem of grain growth is a very serious one, nce it cannot be readily controlled by introducing devices such as traps to re- ove impurities. It is, rather, an inherent property of the material, depending imarily upon the length of time for which the material is held at an elevated mperature. Investigation of this process, the conditions leading to it, and the ssible steps that could be taken to alleviate it, are still underway. One study eks to determine precisely what the expected lifetime of a structure containing rge grains would be in terms of the extent to which this grain growth has pro- eeded. Another study attempts to relate this grain coarsening to the extent of working of the material.

Two mechanisms of grain coarsening are found to occur in zirconium at mperatures approaching the recyrstallization range: uniform grain growth and rminative grain growth. The first phenomenon is caused by the tendency to duce the interfacial energy which exists between grains in a polycrystalline tal. In germinative grain growth, stored energy existing in strained crystals uses new stress-free crystals to nucleate and grow, replacing the distorted

grains. Adjacent relatively low-energy grains do not in themselves grow, but will be consumed by the growing new stress-free crystal.

VII. TENSILE RESULTS

Tensile tests of material which had been exposed to various environments showed that neither the presence of an oxide layer, the presence of dissolved hydrogen, nor the presence of large grains in themselves had a radical effect upon the tensile strength of samples of zirconium. Rather, it has been determin that the principal contributory cause to reduced tensile strength is an annealing of the material, which appears to be dependent only upon the time-temperature history of the zirconium sample. Despite the fact that the material which was tested was reported to have been stress-relieved following rolling, it was found that further stress relief and consequent loss in tensile strength could be accom plished by holding this material at a temperature of 980° F for 300 hours. Figure 9 shows this loss of tensile strength as a function of annealing time at temperature. Recent studies of material removed from the component can-head test at the points of greatest flexure have shown grain growth at these points. T grains had grown inordinately large and there had been incipient failure of the material in these regions.

VIII. SUMMARY

In summary, it can be said that the incorporation of adequate trapping devices in the sodium reactor design has, we believe, rid us of problems of con tamination and the accompanying severe loss in material integrity. There re- main, however, at least two important problems with the SRE zirconium. One arises from stress concentrations and the consequent relief of the resultant strains by means of growth of large grains at elevated temperatures. It is pos- sible that the fairly frequent repetition of these stresses during the process of the grain growth may interrupt this growth. As a consequence, grain growth m not have an opportunity to continue to the extent that has been observed in mater which has been first stressed and then exposed in the loops at high temperature The other critical matter, in a sense, pertains to the consequences of the occur rences of the opposite effect. In this instance, we are talking about the tendenc

r material which is not being worked during reactor operation (namely, that aterial on the surfaces of the moderator cans) to dead-anneal to the point where has a minimum tensile strength. Such material would have the greatest tendency creep under stress to a shape which may conceivably result in some mechanical roblems. In the long run, the solution to both of these problems might well be und in the selection of a suitable alloy of zirconium, following the practices that ve been employed in the alloying of other structural materials which, in the re form, may show one or both of these characteristics. We believe that there a good chance that success can be found in this direction, and that these prob-ms will vanish.

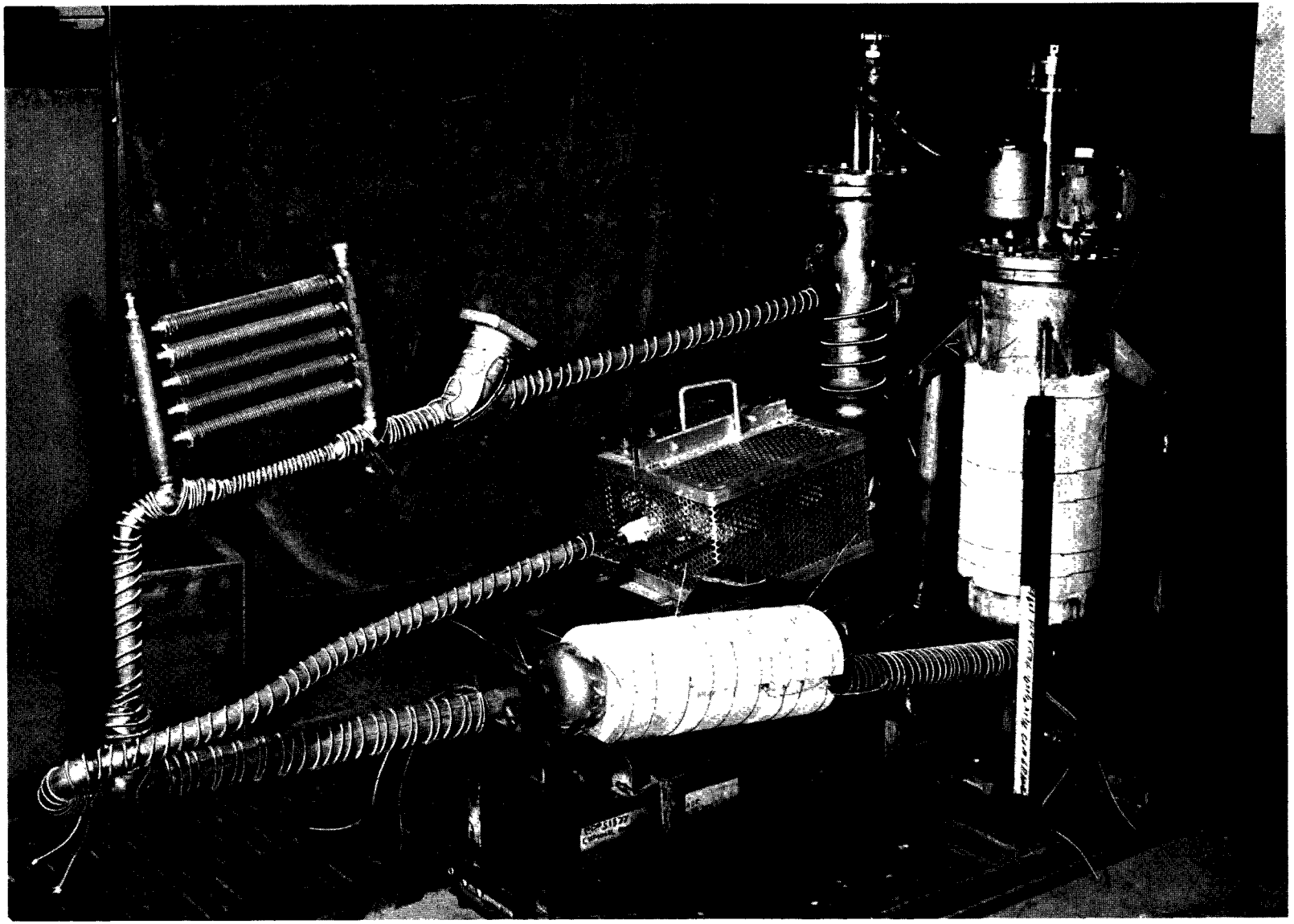

Fig. 1. Sodium Exposure Loop

Fig. 1. Sodium Exposure Loop

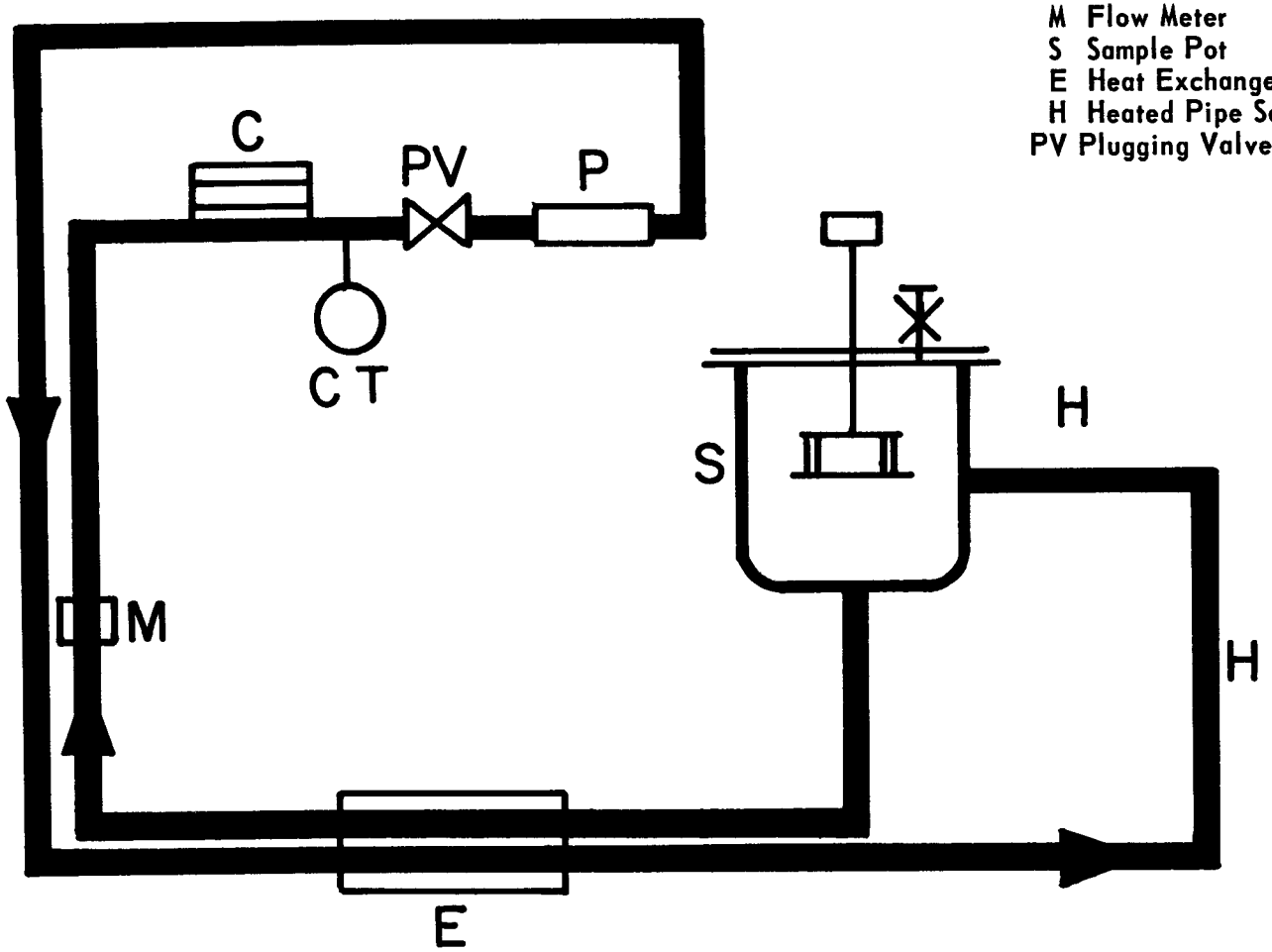

Fig. 2. Line Diagram of Cold Trap Sodium Loop

S Sample Pot
CT Cold Trap
H Heated Pipe Sections
D Dump Tank
SR Sample Rack
SC Sample Changer

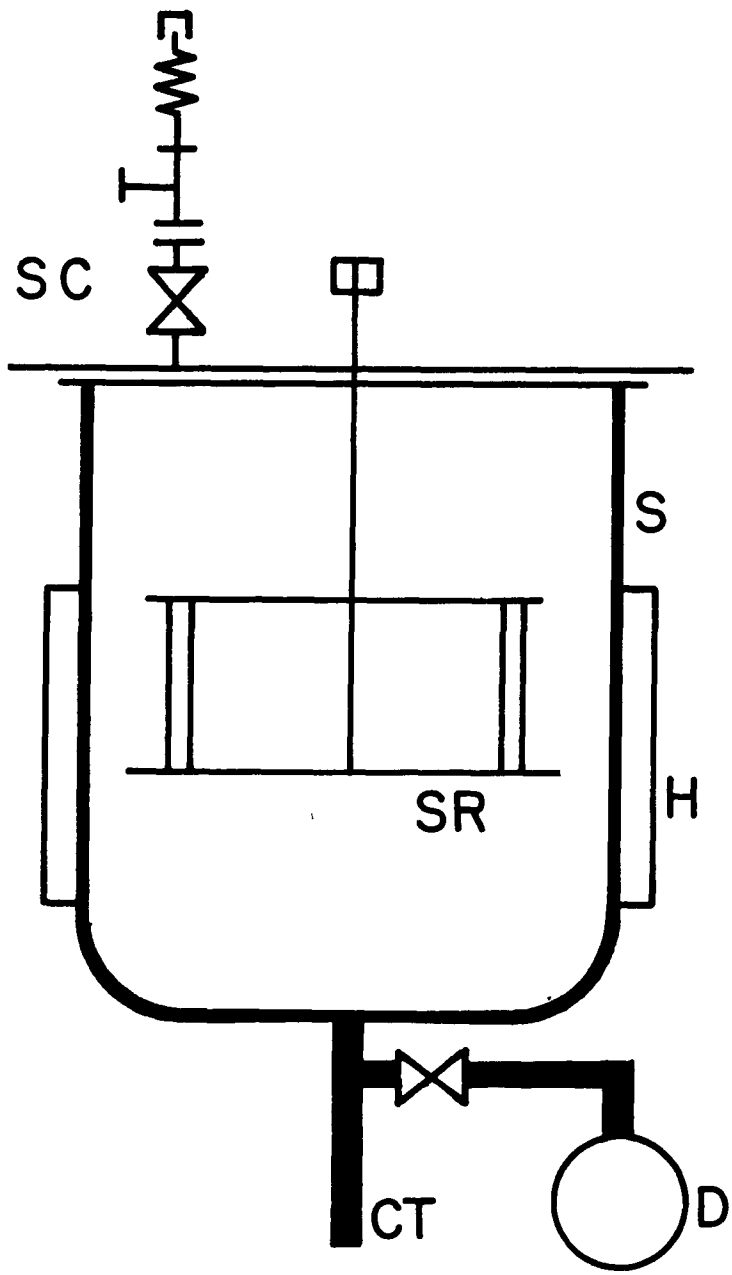

Fig. 3. Static Pot Facility

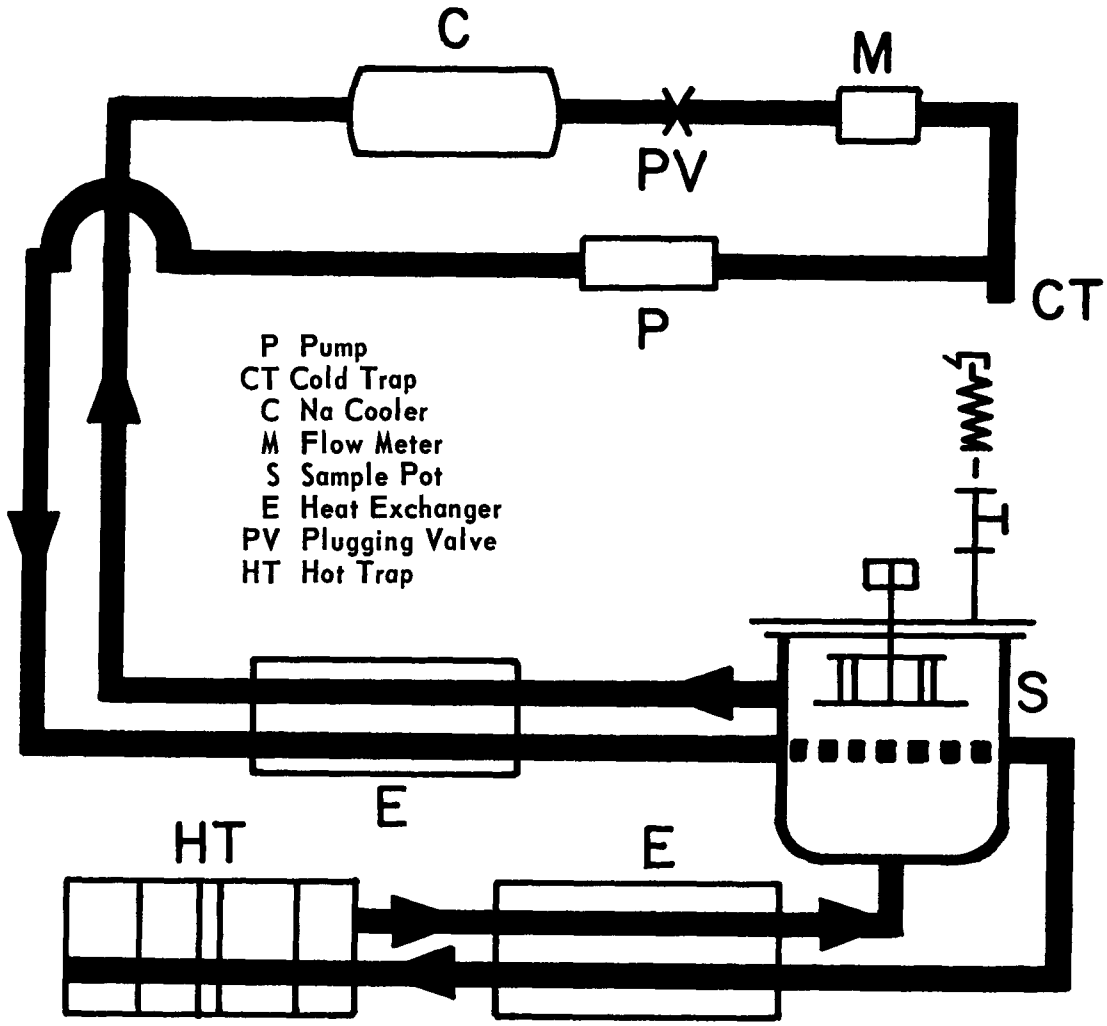

Fig. 4. Line Diagram of Hot Trap Sodium Exposure Facility

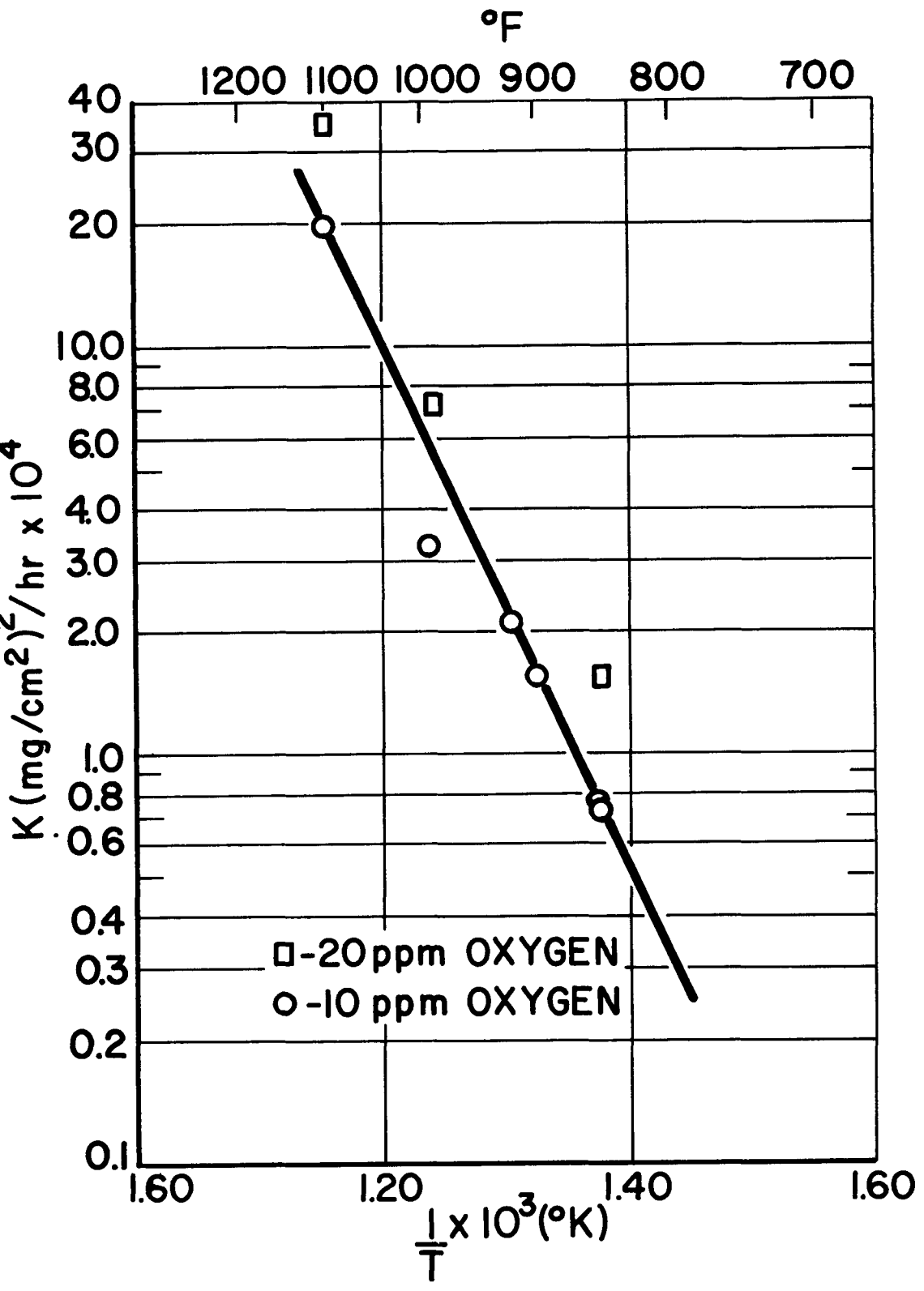

Fig. 5. Rate Data for Zirconium

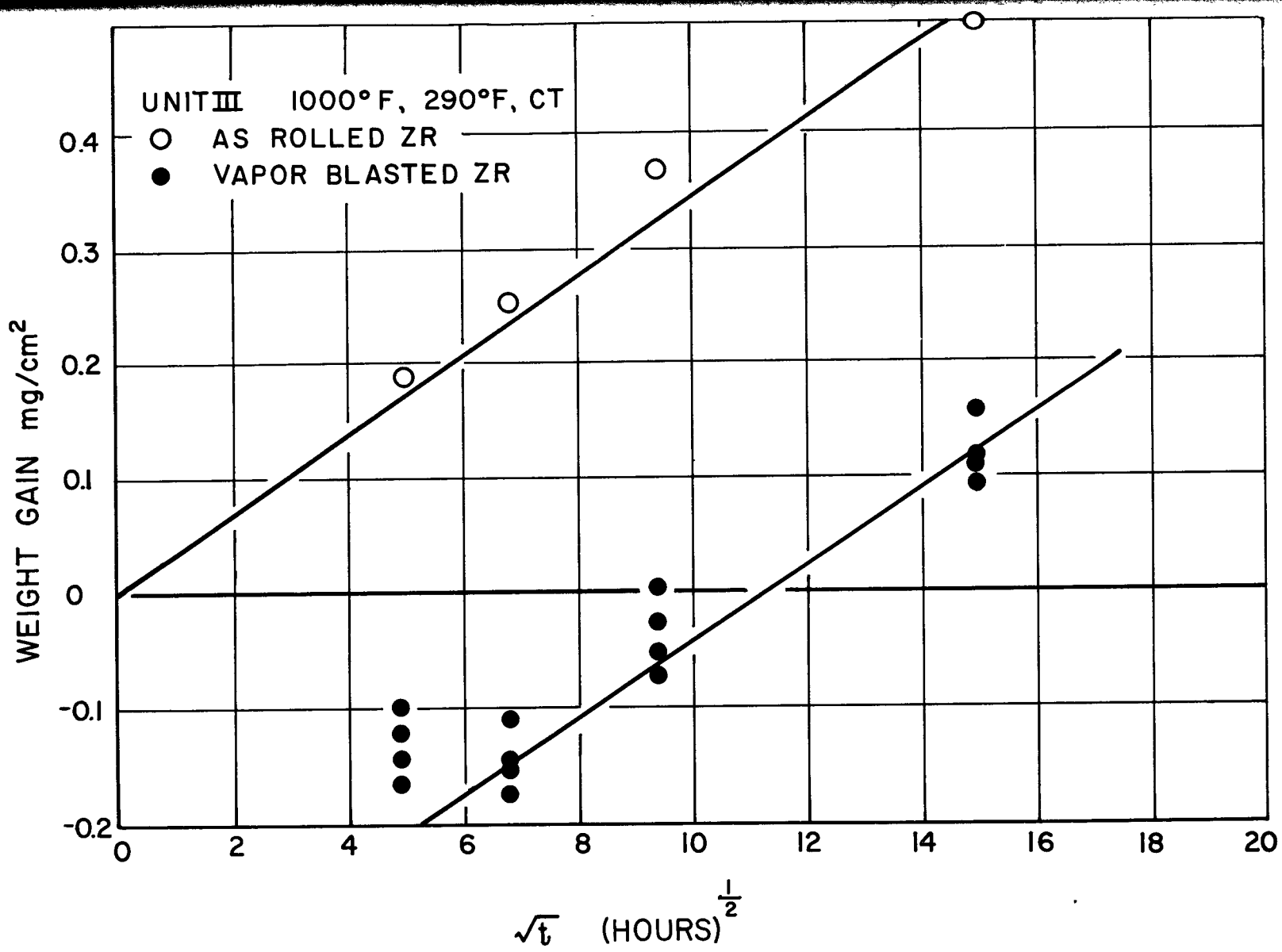

Fig. 6. Weight Gain Data for Unclean Zirconium

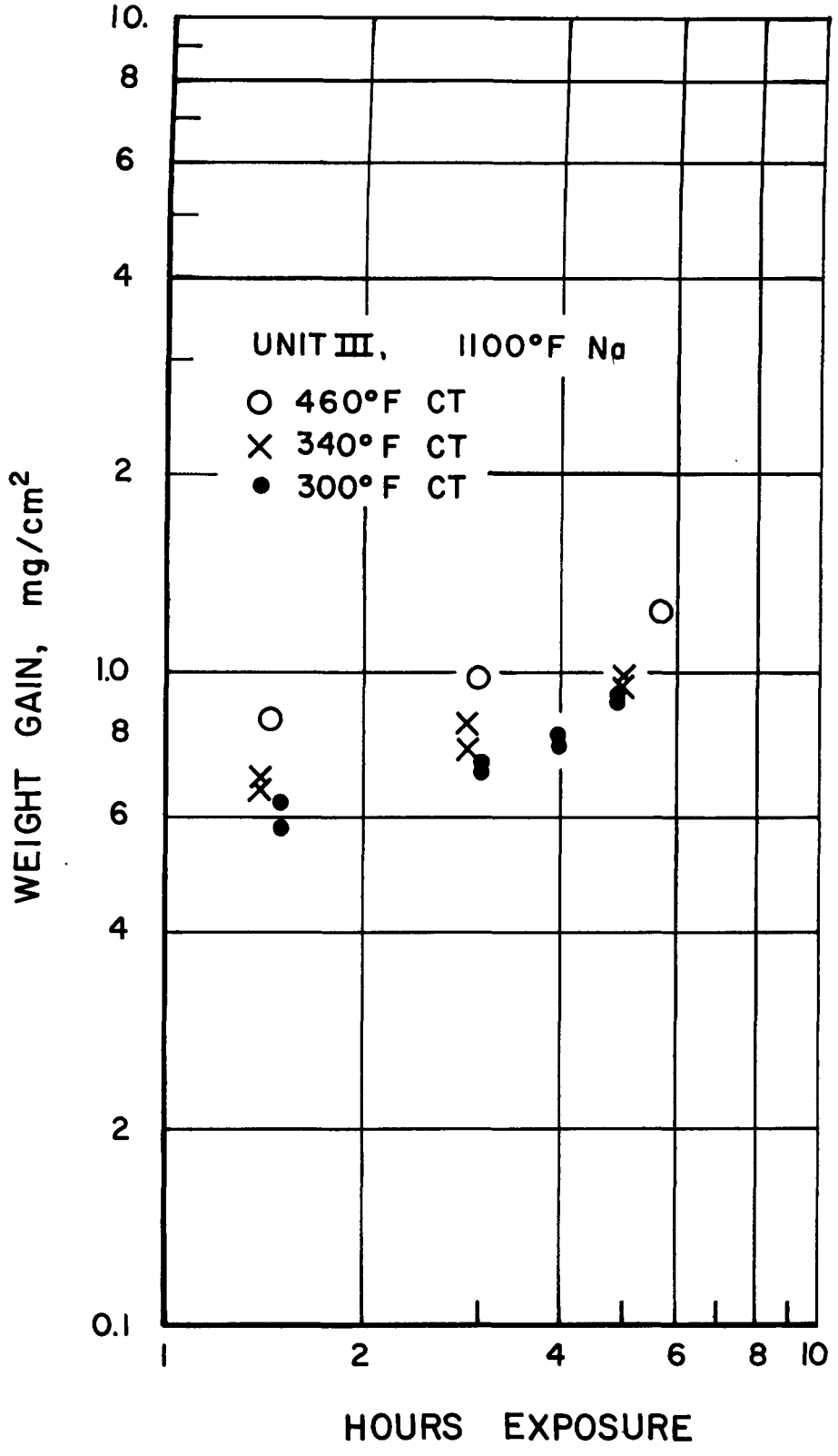

Fig. 7. Hot Trap Effectiveness

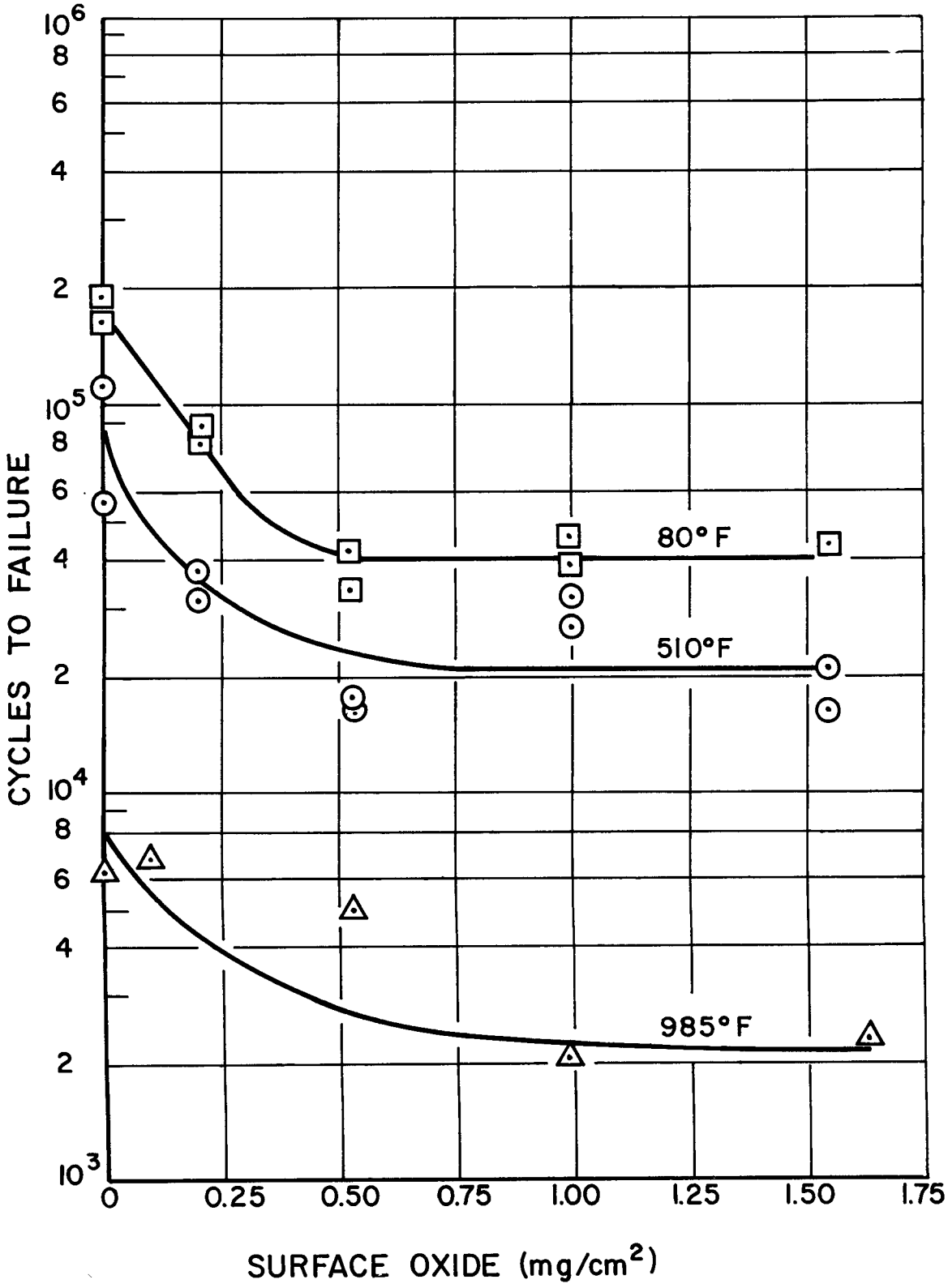

Fig. 8. Fatigue Test Data

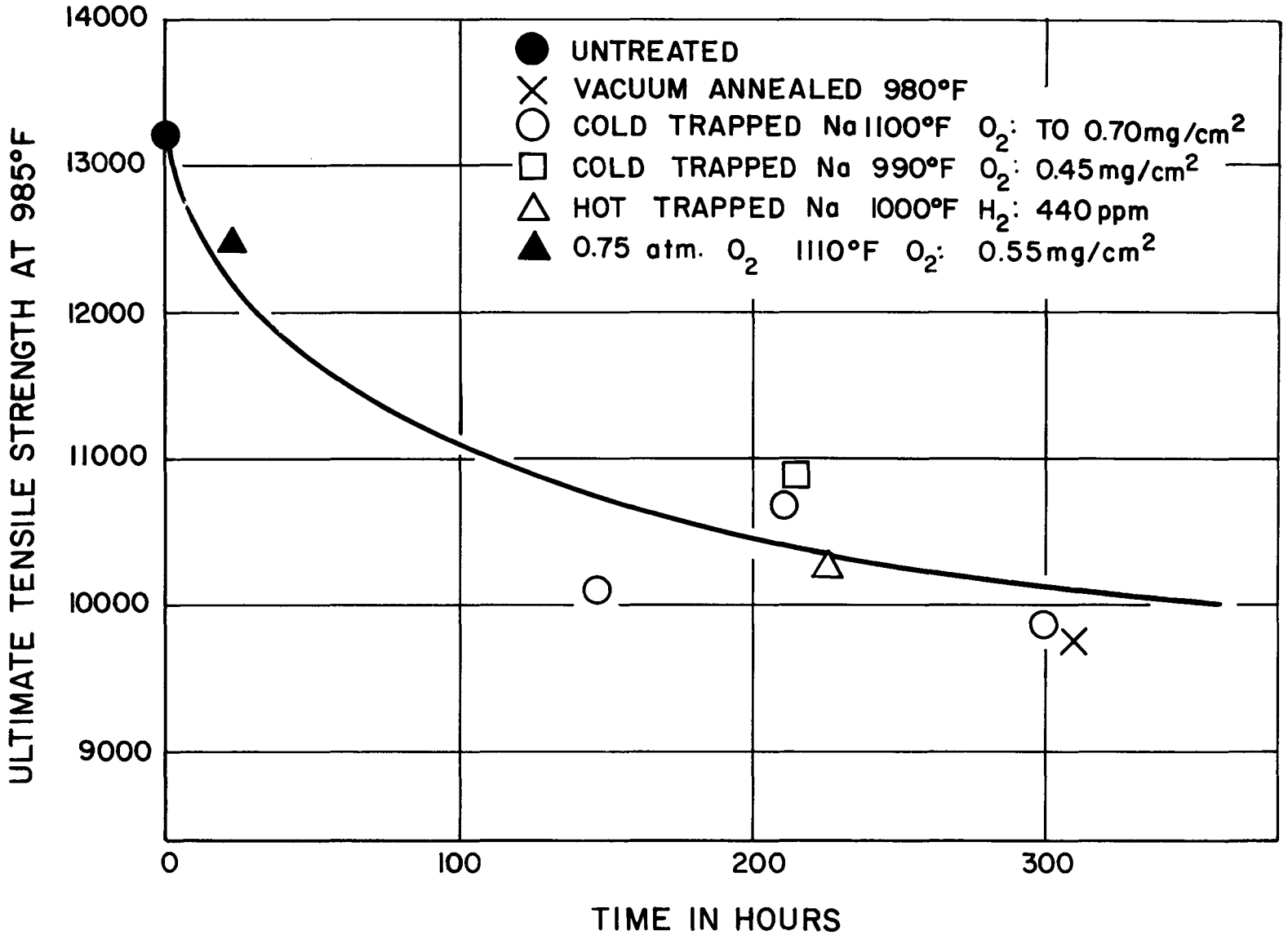

Fig. 8: Tensile Test Data

FULL SCALE SODIUM GRAPHITE POWER PLANT DESIGN

R. C. Gerber

The paper covers changes in the full-scale plant since the last forum as well as alternate designs that show interesting possibilities.

The core is now sized to use either uranium or thorium fuels. Stainless steel is being considered as the moderator canning material. The heat-exchanger system has been changed to three loops, requiring larger pumps, heat exchangers, piping, and steam generators than those previously discussed.

Alternate designs include thimble and through-tube versions to reduce structural material in the core and to reduce the size of the surrounding reactor structure.

(Publication of this paper has been temporarily withheld. Copies will be made available to interested parties at a later date.)

RECORD OF DISCUSSION

NOVEMBER 8, 1956

189-211
LM-00985

QUESTION: What is the design pressure for the stainless steel (1-1/2-inch ck) reactor vessel or core tank?

S. SIEGEL: The maximum pressure above the reactor is only a few pounds square inch. However, I think Bill Parkins can tell us better about the design ssure.

W. PARKINS: The core tank is designed to withstand a pressure of psi at a temperature of 1200° F. Prior to installation, the tank was given a rostatic pressure test at room temperature and a helium leak check. For purpose, a top head was welded to the core tank and then subsequently re- ved prior to installation.

QUESTION: Who built the sodium pump and was a special fabricating plant up? What components were built there?

S. SIEGEL: As was indicated in one of the talks, the sodium pumps are dified hot-oil pumps. The pump manufacturer is Byron-Jackson. The modi- tion, in one sense, consists of the use of a different material--the substitution Type 304 stainless steel for the conventional material which would be used in hot oil industry.

The other modifications are in the mechanical construction. The pump body been substantially extended so as to permit shielding between the pump im- ler and the region near the top of the pump where the motor is installed. And, haps most important, there is the pump seal, which is entirely different from conventional arrangement. This pump seal will be discussed later, but I will ntion briefly now that the seal is accomplished by an annulus of frozen sodium.

As far as I know, no special fabrication plant was set up by the Byron- kson Company for manufacturing these pumps, although, because of their her substantial length, we have, internally at AI, set up a special manufacturing ility to make the modifications that we carried out.

QUESTION: What are the relative merits of stainless steel and zirconium fuel elements in moderator cans, other than cross section? What radiation- istant grease is used in the sodium pump?

S. SIEGEL: Of course, the principal difference between the stainless steel zirconium is in the cross section. The properties of zirconium under our

operating conditions are satisfactory, provided we maintain adequate purity in the liquid sodium. The principal impurity we are concerned about is oxygen; at one of the later talks, this particular point will be expanded upon.

The strength of stainless steel at a temperature in the neighborhood of 1000° F is better than zirconium. In the SRE, however, we have chosen zirconium because of its better nuclear properties.

The radiation resistant grease is used near the bottom of the lower bearing of the sodium pump. I cannot specify it as it has been developed by the California Research Corporation.

QUESTION: What about energy stored in the graphite due to irradiation Can this require operation of the auxiliary cooling system after shutdown?

S. SIEGEL: There is a later paper by Bob Carter which will expand on this point (the energy storage in graphite). I will merely say that it isn't much of a problem, because at the operating temperatures, very little energ is stored due to the irradiation of graphite.

Inasmuch as there is no problem, the answer to the second part of the question is "no".

QUESTION: Is the sodium reactor type the most dangerous type reactor with respect to the surrounding public?

S. SIEGEL: As is evident from the design, both we and the commission a convinced that this is an extremely safe type reactor. The building, as you hav seen, is not a containment type building. The objective in the building design is to provide containment in the sense of restricting the circulation of the air in an out of the building to some modest rate of, say, 2 per cent a day. It is not a co tainment type building. We believe that this reactor is an extremely safe type o reactor, and there are basic reasons for this belief.

The entire system which contains radioactive material is a low pressure stem. There is no mechanical energy stored within the reactor which, upon lease, can disburse radioactivity.

I think we need to emphasize that all fission-type reactors contain large ounts of radioactivity, and this is the origin of the danger. If this radioactivity nnot be disbursed, then the danger, we believe, is minimal. In the sodium aphite reactor, the entire radioactive system is, first of all, a low pressure stem. Secondly, there is no means for disbursing the radioactivity by the lease of chemical energy. The constituents of the reactor are materials which e chemically inert with respect to each other: sodium, stainless steel, zircon- m, and graphite; these materials do not react chemically to each other, so that abnormal operations, there is no opportunity for the release of chemical energy ich can serve as a means of disbursing radioactivity.

The nuclear features of the reactor are such as to lead to rather slow anges under all circumstances that can be envisaged. And I think that this is aracteristic not only of this reactor, but also of many other thermal-neutron actors. There is a substantial negative metal-temperature coefficient in this actor, so that as the fuel elements increase in temperature, the reactivity of reactor decreases. All of these reasons lead to a stable reactor which does t have within it means for disbursing the radioactivity present in it.

As an additional safety factor, we contain the reactor and its cooling system a succession of local containment reservoirs. Also, we have provided for all- lded stainless-steel construction for the reactor and the cooling system. Finally, have two concentric tanks surrounding the reactor core, so that we cannot lose olant from the reactor core itself. All of these design considerations lead to ituation where a malfunction of the heat-transfer system still leaves the reactor oded with sodium coolant.

There are other aspects which I will not dwell on, but basically, the point view is that although there are certain types of malfunctions that can lead to ernal damage to portions of the reactor itself (particularly the fuel element), re are no circumstances, we believe, under which this radioactivity can be leased and be a danger to the surrounding public.

QUESTION: How do you detect a fuel-element rupture in the reactor which releases radioactivity products into the sodium?

S. SIEGEL: We do not provide means within the reactor to detect ruptured fuel elements. Since the metallic uranium is basically stable with respect to the sodium, there are no chemical reactions. Any rupture of a fuel element will not lead to any pronounced malfunction of the reactor itself. It will be a minor nuisance in the sense that we could get small amounts of fission products into the sodium and thereby render maintenance of the sodium system more awkward than it would be with merely radioactive sodium present. But, other than that, there is no major problem.

We do have means for detecting the presence of a ruptured fuel element in later stages of operation. After the fuel element has been removed from the reactor and has gone through the washing operations which get rid of any clinging sodium, we can then detect the presence of fission products by monitoring the wash water which is held up and stored. If fission products are present in this water, we can thereby conclude that we have a fuel element with a ruptured cladding (which we store in a slightly different way than we would otherwise). But we provide no means for detecting ruptures in the reactor.

QUESTION: What precautions are taken to prevent excess reactivity in the event of loss of sodium in the core? Will safety and control rods have enough worth to prevent criticality in this event?

S. SIEGEL: I think that I have already touched on part of this: the fact that the reactor configuration is such that it is not really conceivable that you can lose sodium from the core. The primary reactor container is the 1-1/2 inch stainless steel tank, and the sodium level is about 5 feet above the fuel elements. Actually it is more than that, under normal circumstances. If one conceives that the primary tanks break, then the sodium level falls but is contained by the outer tank at a level still covering the fuel elements, so that even under these unlikely circumstances we still have sodium in the reactor core.

To answer the second aspect of the question - Will safety and control rods have enough worth to prevent criticality in this event (that is, the postulated event of draining the sodium out of the core)? I believe so, although perhaps Frank Faris would like to comment on this.

F. FARIS: As I mentioned in my talk, the dry reactor has a critical loading n the order of somewhere between 12 and 15 fuel elements, while the reactor ith sodium in it at temperature will have somewhere in the neighborhood of about 0 to 35 fuel elements. Theoretical calculations indicate that the safety and control ods can handle a difference in reactivity represented by about 30 fuel elements. his, of course, is more than sufficient to prevent criticality in the event of total ss of sodium from the core.

S. SIEGEL: We are even safer than I thought we were.

QUESTION: What lubricants are used on the pump bearings? Where is the rease obtained which is used to lubricate sodium pumps? How long can it be ed before radiation renders it useless? What is the bearing operating tempera- re and what is the radiation level in the pump-bearing area?

S. SIEGEL: I have mentioned the lubricants already. I believe that the emainder of the question which has been raised can be answered a little bit later n during Dr. Eggen's talk. Since the question has been raised, he will be in a sition to answer it and it will come out in the normal part of his talk.

QUESTION: Why do you use two types of inert gases? Why not use one gas nly?

S. SIEGEL: It is principally a matter of conservatism. In all areas where e have a gas in contact with sodium, we use helium, which we can obtain with ite a high degree of purity. This purity is necessary in order to keep the oxygen ntent in the sodium down below 10 parts per million. The other gas is one that e use to blanket the sodium-containing equipment in the primary galleries in der to prevent a fire in the primary galleries if we have a leak of any equip- ent there. We use nitrogen because it is cheaper and because it serves the rpose adequately.

QUESTION: Is the reactor flow orificed? What are the maximum-to-average utron flux conditions? What will be the maximum fuel-element surface tempera- re?

S. SIEGEL: I think the first of this series of questions has already been swered. It has been pointed out that there is no throttling valve at present in e main system, but that the reactor flow is orificed among the individual fuel

channels. The individual fuel elements have orifice plates near their lower end which serve to distribute the flow among individual fuel channels so that we get a reasonably uniform exit coolant temperature from all of the channels. The central fuel element, of course, produces the largest amount of power; the birth row fuel elements produce a smaller amount of power. In order to avoid compromising performance, we want the exit coolant temperature from all the channels to be approximately the same, and so we individually orifice against the fuel elements.

As I recollect, the maximum neutron flux is on the order of 4×10^{13}, and the maximum-to-average flux is about two. These are rough figures, but I think they're close enough for our present purposes.

The maximum fuel element surface temperature is somewhere in the neighborhood of 1000° F under the conditions where our pool is at 960° F. A more important limitation is the central metal temperature in the uranium, which, because of the alpha-beta transformation in uranium, is held below 1200° F. This is one of the design conditions - that the central metal temperature be below 1200° F.

QUESTION: What is the expected enrichment? What is the weight of the U_{235}? What is the U_{235} burnout? In other words, what is the physics aspect of the reactor?

S. SIEGEL: The enrichment is 2.8 per cent. The total weight of uranium in the reactor is around 2000 kilograms. The number I have here is 2500 kg for 37 fuel elements, which is somewhat larger than the critical loading, or even power loading. The weight of U_{235} can easily be caluclated from the two figures I have: 2.8 per cent enrichment and 2000-2500 kilograms total uranium.

The burnup, of course, varies. The reactor will be loaded with various types of fuel elements. As will be brought out in one of the later talks, the basic type of fuel material is essentially project-grade beta heat-treated unalloyed uranium. With that material we expect to get into the neighborhood of 2000 megawatt-days per ton of burnout in those units.

There will be a large number of experimental fuel materials of various types. The initial loading of about 30 to 35 fuel elements will contain about 11 experimental

sters containing elements of various materials, including uranium-thorium; h these, of course, we expect to demonstrate substantially greater burnup.

QUESTION: Is any protection, such as a blast mat, to be placed around the ium-water heat exchanger?

S. SIEGEL: No, this will not be done. When we first contemplated the tallation, we gave some thought to a blast mat, but finally decided that it would be necessary. One of the reasons is that the design of the steam generator kes it virtually impossible to get quantities of water and sodium together in umes that could produce a serious explosion. Altogether, the water content in steam generator is only about 200 pounds. Another consideration is that the ter is at 600 pounds and the sodium is at 40 pounds, so that any leak from water sodium is going to flash into steam; and, of course, steam mixing with sodium far less violent than water mixing with sodium.

QUESTION: What is the thermal efficiency at 825° F.

S. SIEGEL: The thermal efficiency of the plant is 11,544 Btu per kilowatt hour, ich corresponds to an efficiency of about 29 per cent. Of course, there isn't much s in efficiency in a steam generator. Radiation loss is the only loss you have there.

QUESTION: Explain how the design of the tubes in the steam generator events steam and water from coming in contact with sodium if a leak through e tube occurs.

S. SIEGEL: The principal protection there, of course, is a double tube. ere is sodium on the shell side in the outer tube; in between the two tubes, ich are concentric, there is mercury; and then inside the inner tube there is ter flashing to steam. The mercury is pressurized at 190 psi and the water erates at 600 psi; the sodium has a pumping pressure of about 40 psi, so that would expect to find a leak very quickly in either the inner tube or the outer be by a change in mercury pressure long before you could get the sodium and ter together. Incidently, in this connection, we are investigating other means monitoring this type of leak. Initially, we started out with pressure. We are etty certain now that we will add a volumetric measurement as well, which is ore sensitive by far than the pressurized mercury system, and we are further oking into methods for monitoring the sodium (and possibly the steam) in some y to detect minute leaks in any of the combinations.

QUESTION: What is the size of the station battery? Can it operate the fe pump?

S. SIEGEL: The station battery will be rated at about 200 ampere-hours and will be used largely for controls. It cannot operate the feed pump; these pumps are ac pumps and require more battery capacity than we will have.

We feel that we have a very reliable source of station light and power. Th station (which is right at the site) ties in to our system from 66-kv transmissio lines. We consider that a very reliable circuit. Also, the station light and pow can feed off of the generator itself.

QUESTION: Is there any possibility that a wasting or wearing away of a shell or tube material could develop, especially at the bends of the B & W steam generator, from abrasive action caused by flow of sodium through the generator

A. C. WERDEN: I can hardly answer that. The manufacturer would have to answer that. Apparently he did not think there would be any wear there. You understand, of course, that the sodium is on the shell side. Since it isn't flowi through the tube, we certainly don't expect any wear.

QUESTION: This question is confined to the B & W steam generator. Is there the possibility that caustic embrittlement could develop in the shell or the tube material directly in contact with the sodium?

A. C. WERDEN: My understanding of a caustic embrittlement is that sod hydroxide is formed. Normally, very pure sodium is in contact with one side o the tube in the shell. You wouldn't expect to get embrittlement unless water get in there and makes a hydroxide out of it. This could not happen unless that you got the impossible thing of two very small leaks in both tubes that would, in so way or another, let water and sodium get together. Nevertheless, we are study ing the monitoring of that very thing.

QUESTION: Will you please clarify the reasons for elevating the steam generator?

A. C. WERDEN: Atomics International has designed the whole sodium system to drain by gravity. The SRE airblast cooling system was designed long before California Edison came into the picture, so that at the point of elevation

which it was feasible to us to connect, we had to get our equipment higher, turally, to allow gravity draining and proper filling.

QUESTION: How is the spacing of the heat-exchanger tubes controlled?

A. C. WERDEN: The outer tube is held in a series of lattices around the ngth of the tubes; however, as far as the two tubes themselves are concerned, e inner tube floats in the outer tube with mercury monitoring fluid between. So is just a floating proposition as far as the inner tube to the outer tube is con- rned.

There is no harm in the two tubes coming in contact, to our knowledge. he mercury would be there to detect a leak, and this, of course, is the purpose having them separated.

QUESTION: In the steam generator, what is the steam pressure, mercury essure, and sodium pressure?

A. C. WERDEN: The steam pressure is 620 psi, the mercury pressure is 90 psi, and the sodium pressure is roughly the pumping pressure of 30 to 40 psi.

QUESTION: Assume that the B & W steam generator is operating under ll load; i.e., steam at 600 psi, mercury at 190 psi, and nitrogen-sodium at 0 psi. If a steam tube and a sodium tube ruptured at the same instant, what ould take place in the generator?

A. C. WERDEN: Naturally, it would depend on how severe the rupture as, but as I have tried to indicate, with the design and monitoring we expect to ave in this installation, we expect to detect minute leaks long before there is ny quantity of mixture of sodium and water of a magnitude that could cause any- ing approaching a violent explosion.

QUESTION: It is understood that the steam generator tubes are stainless teel. If so, what is the permissible chloride content allowed in the feed water? s stress corrosion cracking likely?

A. C. WERDEN: That is a serious consideration on this piece of equip- ent. As far as the permissible chloride content is concerned, that hasn't been pecified exactly. The manufacturer of the steam generator said that we should

keep total solids within 0.5 parts per million. The people that are furnishing the water-demineralizing equipment guarantee a total solid content of less than 0.1 pa per million. Although they haven't made any specific determinations, it was thei opinion that the chloride content is that sort of a situation should certainly be les than 0.01 parts per million.

We don't feel that stress corrosion cracking is likely when we are keeping the impurities of the water down to the degree that we are and when we are keeping the chloride content of the water down to 0.005 parts per million. It is recognized that this type of stainless steel is susceptible to a chloride stress corrosio if you do have a high oxygen content and if you do have a high chloride content. We expect to keep both of those so low that we won't experience that difficulty.

QUESTION: Have any design considerations been given to a possible sodiu water reaction in the event the monitoring system fails?

A. C. WERDEN: I would say that inherent in the design in that case is the fact that there is a double-walled tube which, in itself, prevents any great amoun of sodium and water from getting together.

QUESTION: What thermal efficiency is expected from the steam plant at 825° F?

A. C. WERDEN: An efficiency of about 29 per cent - in other words, abou 11,544 Btu per kilowatt hour.

QUESTION: Please discuss the mechanical specifications of the steam generator: i.e., the heads and tube sealing.

A. C. WERDEN: Essentially, it is an all-welded construction. The tube i rolled into the tube sheets and welded on the outside of each tube sheet, and then the heads of the steam generator themselves are welded to the body of the heat exchanger.

QUESTION: Why are UO_2 elements excluded from the SRE fuel-element development program?

C. C. WOOLSEY: Initially, they were excluded merely because we had to limit ourselves within the overall budget of the total SRE program to items which

ere of immediate interest. As I mentioned at the very end of my talk, we are w giving consideration to UO_2, both in connection with thorium metal and in nnection with thoria as a ceramic body.

QUESTION: Are any special precautions taken to avoid alloying of uranium ith iron in the stainless-steel fuel element cladding to form a low-melting eutectic?

C. C. WOOLSEY: The limitation that we place on the central fuel element mperature, (1200° F for the standard fuel slugs and 1350° for the high-temperature anium-alloy experimental fuel slugs) will result automatically in limiting the mperature at the interface or at any point where the fuel slug actually touches e stainless-steel jacket to a temperature that is below the melting point of this w-melting eutectic. (The low-melting eutectic that is formed melts at around 325° F.) The expected maximum temperature at any point of contact between anium and the stainless-steel cladding would be, at the very most, 1050° F to 100° F. We are aware of this as a problem and as a limitation should we go to igher coolant temperatures, using uranium clad in stainless steel.

QUESTION: Who is the supplier of standard fuel slugs?

C. C. WOOLSEY: The National Lead Company of Ohio supplied our standard uel slugs. For your information, the National Lead Company of Ohio operates e seed-material production center FMPC, which is an AEC plant. They supplied oth the standard material and the cast experimental materials.

QUESTION: Please describe techniques for encapsulating fuel elements, hen NaK completely fills the space between the element and the sheath through e core section.

C. C. WOOLSEY: Dr. Eggen described in some detail the development ork that was done as far as the canning or encapsulating is concerned. There s a free surface of NaK above the fuel column. There is about two feet of tainless-steel tubing between the top of the fuel column and the top plug which ttaches to the hangar assembly. About half of this space is filled with NaK. he remaining half of the space is filled with helium, before the individual rod s filled up.

QUESTION: How does the hollow fuel element compare with the seven rod cluster, as far as fabrication problems are concerned?

C. C. WOOLSEY: We would anticipate the handling of a smaller number of parts, which would give us a cheaper fabrication. Particularly, this would be true in the case of remote refabrication of reprocessed or partially decontaminat reprocessed fuel.

QUESTION: Regarding the consideration of urania plus thoria for the SRE: Is there a reaction of urania or thoria with sodium at high temperatures?

C. C. WOOLSEY: We know of none. I don't know of any extended experiments that have been undertaken. I can see no reason why there should be. We have not experienced it with uranium metal, which, in general, is more reactive if anything, than the oxides are. I don't believe that there is any reduction, if this was the thought in mind in this particular question.

QUESTION: What do you anticipate will be the effect of using unalloyed or slightly alloyed uranium fuel elements with our fuel element design?

C. C. WOOLSEY: As I mentioned previously, we have examined these under a fairly extensive thermal-cycling program. We feel that one of the reaso that unalloyed uranium breaks down on thermal cycling is that its strength is ver poor at high temperatures. In the case of the alloys, we feel that the strength is appreciably better and that the outer skin is able to withstand the stresses tha are developed in the thermal cycling process, as well as the stresses that will be developed under the temperature gradient conditions that will be in the reacto

QUESTION: What are the relative advantages of presealed sodium valves and bellows-sealed valves?

D. T. EGGEN: As you all know, there has been considerable trouble with bellows-sealed valves; one of the problems is that the range of activation of a bellows should not be extended beyond a certain percentage (less than 15 per cen of its free length. They have a limited life, due to continual flexure. In the pre sealed valves, the torque necessary to open the valve is kept small, since the shear strength of sodium is low. The preseal is quite reliable, inasmuch as it i not a continuously moving body. It is in a solid frozen state, rather than in a thin liquid film, as in the case of the shaft seal, and it is considerably cheaper.

can take a particular example in our experimental work. We have gone over to he use of two-inch presealed valves, using a standard off-the-self stainless-steel hrottle or gate valve. These would cost, for a two-inch size, somewhere around 50.00 or $60.00; by taking the top bonnet off and with about a $75.00 machine job, ou can put a new free stem on the top with air cooling which will operate without ny other auxiliary cooling except ambient air. These have operated quite satis-ctorily with no leakage for periods of three to four months, without any attention t all. You don't have the problem then of a bellows in a liquid sodium system.

QUESTION: What problems do you anticipate with tetralin?

D. T. EGGEN: This is a good question. We expect less problems than we ould with toluene or water. With water, of course, the main problem would rise in a place such as a preseal where a high thermal stress exists. The pos-ibility of cracking and the possibility of throwing water into the sodium system lways poses a problem similar to the one brought up with relation to the steam enerator. However, we have operated the experimental sodium pump loop with ater as a preseal. We have had breaks in this system. In some of our earlier eal designs which broke, water entered with gentle popping; also, of course, a onsiderable amount of oxide was formed which was cleaned up by filtering. We ve had cracks with tetralin. Of course, no violent reaction occurred. There a slight reaction. The tetralin decomposes slightly in sodium at high tempera-res and carborizes, forming sodium carbide and some carbon (which, of course, not good for stainless steel systems). The advantage of tetralin over toluene its higher flash point, its higher boiling point, its better radiation stability, d some others. So I have perhaps answered this question in a negative fashion - u asked what the problems were and I have given the advantages. I told you hat the problems of the other materials were.

QUESTION: What types of thermal insulation are used in the reactor and the primary sodium loop, and why were they selected?

D. T. EGGEN: We were using Johns-Manville Super X or the equivalent. ay that because I am not completely certain; I am relatively sure since I saw me Super X boxes at the site during the installation. We are using this because ran several tests, using most of the commercially available insulating mate-ls that we could get hold of at the time; these included mineral wool blankets,

diatomaceous earth, and several others. We used blocks of sodium (I believe we reported on this at our last forum), brought these up to a temperature of 1000° F to 1500° F, and let them soak for a period of time; then we noted the amount of reaction that had taken place under these conditions. We found that Super X did not saturate as quickly. It formed a protective crust, and we thought that it was the most satisfactory of those we tested for the price. Of course, reflective insulation would have not been attached at all, but it is pretty expensive to instal in adequate insulating quantities.

QUESTION: Are provisions made to clean up fission products in the prima sodium loop?

D. T. EGGEN: As far as I know there are no provisions for cleaning up the fission products, with the exception of the provision for dumping the sodium out of the system and flushing it with clean sodium and then disposing of this sodium in the liquid waste system. So it is a matter of purging the material out. In the event that fission products did attach to other parts of the system (although it doesn't look as though there will be much occasion for mass transfer) then it would be necessary to remove these parts in the system by remote maintenance techniques.

QUESTION: To what extent is the increased use of nonmetallics being considered? What development program, if any, is now under way?

D. T. EGGEN: I am not sure what you mean by non-metallics. Ceramics are being considered for the moderator. There is the possibility of using beryllium oxide in ceramic fuel bodies, such as in the thoria or uranium-oxide system These are the only things that are being considered. There is no development work at the present being carried on actively at AI, with the exception of the uranium oxide program. At no other place is there any ceramic work being carried on.

QUESTION: What about erosion of metal under sodium flow?

D. T. EGGEN: We have had operating systems for some time with sodium flowing. We have not witnessed erosion under conditions of clean sodium flow. We had one case where some erosion or corrosion had taken place, but this was due to circumstances of an extremely dirty system. As far as I know, sodium

a much better material for a coolant from this standpoint, than almost any of e other standard coolants. (If you want to consider sodium, with the 5 or 10 ears of experience behind it, as a standard coolant material.)

QUESTION: What materials are used in the pipe?

D. T. EGGEN: We are using Type 304 Austinetic stainless steel exclusively roughout all of the process system. There are a few exceptions. None, however, hich are in contact with sodium.

QUESTION: Why is the primary sodium pump on the hot side?

D. T. EGGEN: We put it on the hot side so that it would be in close con- nction to the reactor tank, and permit better control of the pressure on the inlet ide of the pump. Control is accomplished by the static head in the reactor tank, hich maintains conditions such that cavitation does not occur in the pump. If we ad put it on the back (cold) side of the heat exchanger, the drop-through would ave forced us to go to a higher pressure inside of the reactor tank; we did not ant to do this at the time.

QUESTION: How many thermal cycles have the pump freeze seals sucess- ully undergone?

D. T. EGGEN: We ran a test on the actual prototype seal that will be used n the reactor by thermal cycling it from 500° F to 1000° F on a two-hour cycle which was the fastest we could conveniently do) for 300 cycles. Before this, we ade critical measurements (dimensions of the preseal); afterwards we took the ame measurements over again and found no thermal distortion of the preseal and o evidence of fatigue failure over 300 cycles; this is, as far as thermal-cycling s concerned, perhaps 4 or 5 years of operation of the preseal.

QUESTION: You stated that the net goal cost is 2.2 mills per kilowatt hour nd includes a credit for plutonium. How many mills per kilowatt hour does the lutonium credit represent?

S. SIEGEL: 2.9 mills per kilowatt hour.

QUESTION: Could you discuss the instrumentation and control equipment? s it unusual in any way?

S. SIEGEL: This, of course, is a substantial topic which we have perhaps slighted. There are nuclear instrumentation fission chambers and ion chambers there is process instrumentation for the sodium system flowmeters, pressure indicators, level gauges, and hundreds of thermocouples. The nuclear instrumentation circuits (period and flux level circuits) are used to observe the reactor operating conditions, and they control scram or indicate power and flux level, in the way that Dr. Faris indicated. From our point of view, the equipment is not particularly unusual. The nuclear instrumentation is relatively conventional. The sodium instrumentation is special to sodium systems and has its own problems but to us it does not appear to be particularly unusual. We have level gauges of an unconventional source, such as was previously described. We have commercially available pressure gauges; we have commercially available flowmeters; and so on. I think a discussion of the philosophy of instrumentation and control would take a rather extended period. So I will set that aside for the moment. If there is detailed interest in this subject, we might attempt to expand on it later.

QUESTION: Why not a "spiked" core in a sodium graphite reactor?

S. SIEGEL: This question implies the use of an arrangement of a few highly enriched rods, and the remainder natural rods. We have considered the economics of such a configuration and in our studies there appeared to be no particular advantage to that approach. Hence, our reactor design for both the SRE and the larger reactors are primarily along the lines of essentially uniform enrichment throughout the reactor.

QUESTION: Was all machining performed by North American Aviation?

S. SIEGEL: No, of course not. A great deal of the machining, particularly of the larger components, was done in other shops. The large core tank was made in another shop, the quite complex extremely exceptional loading base shield that Mr. Peterson discussed this morning was made in other shops, and, of course, many of the other components were made in other shops. However, many of the parts were made in our own shop. The control rod assembly, the moderator can, much of the work on the pumps, the final assembly of the pumps - numerous other things I could mention - were made in our own shops.

QUESTION: What is the cost of the SRE?

S. SIEGEL: It is somewhat difficult to give an unambiguous answer, since e are so thoroughly involved in a construction and development program. I ink that a number something like six to seven million dollars would be a fair rice for the construction cost of the SRE as we have built it. This is not to be garded as a reproduction cost by any means.

QUESTION: Do you expect difficulty due to dimensional instability of aphite under irradiation? What can we say about graphite growth?

S. SIEGEL: The principal point to make is that the graphite in this reactor erates at a higher temperature than probably any other material in the reactor - rtainly above 1000° F. There is a very extensive amount of information on the havior of graphite under irradiation at various temperatures, and this infor- ation can be rather reliably extrapolated. Furthermore, there are actually perimental data under similar conditions which suggest that the amount of owth to be expected due to irradiation is quite small; and what small amount s been expected (which is less than 1 per cent) has been taken care of in aspects the moderator-can design.

QUESTION: During normal operations, at what rate is helium consumed?

W. E. PARKINS: It is nominal. There is a slight amount of breathing of e region above the core as temperatures changes occur; this gas would be vented the waste gas exhaust system where it would be retained. I can't say quantita- vely what the rate is, but it is nominal. There is no continuous purge of helium rough any part of the system.

QUESTION: During initial operation, what effect, if any, is the effect on e system of the moisture which is picked up by the graphite before it is canned?

S. SIEGEL: Of course, significant amounts of moisture in the graphite could ad to a serious problem; and because of that very obvious possibility, extended ecautions were taken in the way that the graphite was originally manufactured the supplier: in the way the graphite was packed during storage; in the way is handled; and finally, of course, in the last stage of fabrication, which is a mp-out operation and a sealing of the moderator can. (The moderator cans as

they are presently installed have been pumped down to a partial pressure of heli and sealed with a fusible plug, so that they will remain sealed until the tempera ture rises to the point where these fusible plugs will be molten.)

We have done a considerable amount of experimental work (which Bob Car did not mention) on the release of gases from graphite handled in just this sort of way. We know the gases that can be expected to be released - I think Bob mentioned that we can look for hydrogen and carbon monoxide, principally.

QUESTION: Concerning the oxygen problem, what is the anticipated or de rate of addition of oxygen to the SRE system, as it fixes design of the cold trap and hot trap.

S. SIEGEL: Well, the answer to that is simple: it is zero. We believe th for the principal continuous possible source of oxygen after the system has been cleaned up, the procedure will be to fill the system with sodium and essentially self-clean it. This will be accomplished by circulating sodium at a temperature at which we can clean up the system and get the oxide into the cold trap before w reach temperatures where we can endanger the zirconium. We have two materi that we are concerned about: one is stainless steel and one is zirconium. The stainless steel can live in a far higher oxide content in sodium than can the zirconium, particularly at lower temperatures, so that the first procedure will be to clean up the system by circulating sodium at some moderate temperature and then to get the oxide so collected in the sodium and out into the cold trap. We have several cold traps in the system. We can deliver that oxide, as it were, into what we call the disposable cold trap in the sodium service area. During normal operation, after the system has been first cleaned up, one potential sour of oxygen contamination would be the refueling operation. We have done a considerable amount of work at the tower facility on the coffin mockup, and our beli is that the amount of oxygen we will introduce by this means under normal circumstances is exceptionally low. I haven't been quantitative on this; I think this is a point where we will have to gain operating experience to see how much oxyg we do introduce into the system during normal operation. In all of our loops, th problem of erratic introduction of oxygen is generally quite minor.

QUESTION: What are limits on hydrogen in the sodium? Does the hot tra control this?

R. L. CARTER: As it stands right now, the hydrogen in the sodium system distributes itself between the hot trap and all the other zirconium pieces in the system. The amount of hydrogen in the sodium system is particularly low, so that the concentration distributed around these pieces is low and certainly below the 200 parts per million level that I indicated gave negligible change in the original plastic or tensile properties of zirconium.

QUESTION: Why is it necessary to can the graphite?

S. SIEGEL: Graphite is a porous material. If it is left in immediate contact with sodium, it will soak up approximately 25 per cent of its macroscopic volume with sodium, and this would kill the neutron economy of this reactor. There is no significant chemical reaction between sodium and graphite. There would be, however, in consequence of this filling of the pores of the graphite, a moderate amount of microscopic expansion on the order of a fraction of 1 per cent. Although this is of slight concern, the real concern is the nuclear one that I mentioned. We simply could not tolerate that much sodium in the graphite. A further reason is that, particularly at the higher operating temperature we are striving for, graphite in contact with sodium contained in stainless steel systems would lead to carborization of the stainless steel. So, for these reasons, we have to physically separate the graphite from the sodium.

QUESTION: How will the hot waste products, such as the wash water, be disposed of?

S. SIEGEL: I think this was covered in Dr. Parkins' talk. There is a liquid waste disposal system in which the contaminated wash water, under normal circumstances, contains radioactive sodium which is clung to the fuel elements and which is washed in the washing operation. But under abnormal circumstances, (i.e., a fuel element rupture), the wash water may contain small amounts of fission products. This wash water is held up in the liquid waste storage system, and if it contains sodium only, it is stored in one of the tanks of the liquid waste hold-up system until the sodium activity has decayed to a point where this water can be disposed of. If there are long-lived activities in this water, it will have to be held up for extended periods, in which case we may then have to provide further facilities for concentrating this waste and otherwise getting rid of it. We have not yet provided for this because we do not know what operations will reveal.

Normal operations should lead to a relatively straightforward problem of letting the radioactive sodium decay. If we have other kinds of abnormal operations taking place, then we may have to handle the problem somewhat differently. But we're waiting to see, since this is an experiment.

QUESTION: What per cent of the reactor heat can be used to produce stea That is, what are the heat losses?

S. SIEGEL: The reactor will nominally produce 20 megawatts of heat; abo 19 megawatts is handled in the primary system and about one megawatt in the auxiliary system. Essentially, all of the primary heat appears in the steam generator. Very little of the nuclear heat produced leaks out of the reactor or the heat-transfer system. Probably only a few per cent at most of the heat loss are from the reactor or the heat transfer system (two per cent in a large reacto

QUESTION: Must a 7500-kilowatt core tank be field constructed?

S. SIEGEL: We will have to give you an ambiguous answer to this. In son discussions of this problem, it has appeared that the tank might be field constru primarily for transportation reasons. Of course, this is a tank which is substantially larger than the one for the SRE.

QUESTION: What degree of leak tightness is required in the sodium syste

S. SIEGEL: The sodium system, by which we mean the heat transfer syst and the tanks which contain the sodium, is entirely welded. An extreme amount of care is used in ensuring that these welds are leak tight, by a very careful sequence of inspection steps. The route passes are heliarc and there is a serie of inspections at that point. The later passes use filler metal, but again there is a stepwise series of inspections. There is complete radiography of the weld, and finally the system is piece-by-piece helium-checked to ensure leak tightnes So that what we do is contain the sodium in a very good vacuum system, close it here and there by top plugs which have frozen metal seals and which use multipl O-rings, etc., and end up with what is an extremely tight sodium system. For that reason, we feel that the oxygen contamination will probably be quite low.

QUESTION: What is the refueling procedure?

S. SIEGEL: Consider the loading-face shield with its large number of ugs - one plug for each fuel element. From the bottom of each plug, the fuel ement hangs by means of a hanger rod down into the core of the reactor. The ffin that was described (and of which you saw pictures) can be moved on the idge crane to be located exactly over one of these plugs. The coffin interior rmally contains a helium atmosphere similar to the reactor-blanket atmosphere. hen the coffin is in position over a fuel element which is to be removed, part the mechanism near the bottom of the coffin (which was described in Dr. Eggen's lk) is lowered to engage with the top of the reactor (the loading face shield, sentially) to form a gas lock. The gas lock is purged until there is a helium mosphere in it, also. Then a mechanism engages the plug and raises the fuel ement out of the reactor up into the coffin. This sequence of operations, of urse, is carried out with a reactor at some moderately reduced temperature - rhaps 500° F to 700° F - and with the reactor not at power. The reactor is utddown and the temperatures are somewhat reduced. In this fuel removal eration that I have just described, the fuel element is brought up into the coffin, tated about 180 degrees in the coffin to bring a fresh fuel element into position rectly above the hole in the top shield, and is then lowered into place. After at, the lock is essentially opened and the fuel-element handling coffin moves er to the cleaning and fuel-storage area.