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THE SODIUM GRAPHITE REACTOR POWER PLANT FOR CPPD

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INTRODUCTION

The Sodium Graphite Reactor (SGR) is significant because it has the potential of producing steam conditions which are typical of modern conventional steam plant practice. Current development work and the designs which are evolving indicate the future possibility of SGR nuclear steam power plants utilizing 1400 psig, 1000° F steam. The first full scale version of the SGR concept is to be built for the Consumers Public Power District of Nebraska at their Sheldon Station site near Lincoln, Nebraska. The plant will produce 75,000 electrical kilowatts, using 800 psig, 825° F steam. The reactor will use uranium dioxide as fuel, graphite as moderator, and sodium as the heat transfer fluid. The plant will operate with higher reactor coolant temperatures, higher fuel element surface temperatures, and higher steam pressure and temperature than any other reactor currently authorized in the USAEC Demonstration Power Reactor Program.

The purpose of the Consumers SGR Project, as stated in the contract between the AEC and Consumers, is "to demonstrate the economic and technical practicability of central station power, utilizing a sodium graphite nuclear reactor, by operating the plant for the primary purpose of producing a maximum amount of electrical energy therefrom..." In so doing, it is intended to utilize and extend the knowledge obtained from the Sodium Reactor Experiment (SRE)[1] which Atomics International built and is operating for the AEC.

The current program of work was initiated in November, 1957, and has two prime objectives. The first is to carry out the preliminary engineering so as to provide a complete set of criteria for the final design of the plant by September, 1958. The second objective is to carry out a prototype component development and test program to provide data for the final design of all critical components. The schedule for the project provides for completion of construction of the nuclear facility in January, 1962, and for full power operation of the nuclear power plant in June, 1962. To meet their

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power commitments, Consumers plan to place in operation a coal fired plant in the spring of 1961. The nuclear facility will supply steam to the turbine which is initially installed with the coal fired plant, and subsequently a second turbine-generator may be added for the conventional boiler. Included in the scope of the nuclear facility are the sodium graphite reactor, the sodium heat transfer systems, the sodium steam generators, the overall power plant control system for nuclear operation, the nuclear fuel handling system, and various plant auxiliaries described below.

The economics of the sodium graphite reactor show that these reactors are potentially attractive for power plants with ratings from 100,000 electrical kilowatts and larger, where the relatively large investment costs associated with a physically large reactor and the shielding for a highly active coolant are compensated by high thermal efficiency and the ability to use modern and future steam equipment. A brief review of the properties of sodium and graphite explains the main characteristics of the reactor. The boiling point of sodium at atmospheric pressure is 1620° F, which permits the exchange of heat from the reactor to a steam generator at high temperature and very low pressure. The low pressure essentially eliminates hazards' problems associated with releasable potential energy from pressurization of radioactive fluids and greatly simplifies the containment problems. The low pressure, combined with good heat transfer properties, permits a design with large temperature increase as heat is removed from the core and resulting low pumping power to the sodium systems. The sodium is chemically inert to fuel, graphite, and structural materials in the reactor, thus avoiding the problems of release of chemical energy and of corrosion.

Graphite is a good material to use with sodium coolant since it retains strength at extreme temperatures, is chemically compatible with sodium, is relatively inexpensive, easy to handle, and meets normal moderator requirements.

The sodium graphite system provides many interesting reactor concepts. Four of the types thoroughly studied at Atomics International are shown in Figure 1. The thru-tube reactor was one of the first studied as a minimal variation from the Hanford production reactors. Fuel is suspended in vertical steel process tubes that separate the moderator from the fuel and coolant region. Sodium is brought to the process tubes by a pipe header system under the reactor and removed by a similar header system above the reactor. One version of this reactor provided a neutron shield between the core and the headers. This reactor provides good neutron economy through simple core construction. The principal problems encountered were graphite and process tube replacement in the event of a tube rupture, repairs in the header regions if no neutron shield is incorporated, and the design and analysis of the neutron shield if this is used.

The thimble tube reactor is another version of the use of process tubes in unclad graphite. In this core, a sodium header or manifold tank is suspended above the core on hanger rods from the shielding to supply sodium to the thimble tubes. Hollow-cylindrical fuel elements are hung from the tank so that the fuel element forms a return path for the coolant.

Sodium discharges into a free surface pool in an insulated tray from which the sodium can be pumped to the heat transfer system. Again, neutron shield and non-neutron shield versions were considered. The advantages of the thimble concept are good neutron economy, accessibility of the tubes, and a very high temperature reactor since the main tank and tubes in the reactor see only low temperature sodium. Problems are the protection of the graphite from sodium and sodium vapor, the development of the insulated tray and the design of the coolant system for possible failures.

The other two concepts shown are tank-type reactors where sodium is pumped into a plenum formed by the core tank and a grid plate that supports the core. The sodium then flows upward through process tubes containing suspended fuel elements to discharge into a free surface pool over the core. In the calandria design, graphite is separated from sodium by the calandria structure. Bellows' units are proposed for each tube to allow differential expansion between the tube and the main structure. This design provides good neutron economy coupled with a design concept tested in the SRE with the exception of the calandria structure. The main problem is considered to be the replacement of the tubes, or alternately the sealing of the tubes and protection of the graphite from the amount of sodium that would impair the moderation properties of the graphite.

The main feature of the fourth concept, that used for the Consumers SGR, is the use of multiple cans to separate graphite from sodium. Hexagonal steel-canned graphite prisms with corner scallops form the core region. Process tubes, attached to the grid plate, separate high velocity coolant for the fuel elements from low velocity coolant that removes heat from the individual moderator cans. The SRE version of this concept places the fuel elements at the center of the hexagonal prism. This core design provides segregation of graphite so that one failure in the graphite cladding will poison a small fraction of the core, provides relatively simple replacement of the moderator elements, and uses most of the SRE technology. The main disadvantage is the poorer neutron economy caused by the moderator cladding material and the moderator coolant sodium. An early design for the full scale SGR was described by Chauncey Starr[2] at the International Conference in Geneva, 1955.

OVERALL PLANT ARRANGEMENT

A perspective of the power plant installation is shown in Figure 2. The overall plant is a combination of a coal-fired facility which contains the turbine-generator and a nuclear facility which will provide steam for the turbine in the coal boiler facility. Each of these facilities are in separate buildings. The building in the right-hand foreground houses the nuclear facility.

One possible plan of the nuclear facility is shown in Figure 3. The reactor and radioactive portions of the systems are contained in shielded cells below grade, while the non-radioactive systems are above grade. The building is serviced with a bridge crane, in addition to the gantry crane used for fuel handling. The building is of conventional industrial construction.

Heat is removed from the reactor by a radioactive primary sodium system and is transferred to a non-radioactive secondary sodium system which serves the steam generators. The non-radioactive steam from the three steam generators is fed into a common steam header which serves the single turbine generator in the coal boiler facility building. The sodium heat transfer system is operated at a pressure just sufficient to overcome pressure drops.

A sodium service system is used to fill, drain, and flush the reactor core tank and sodium heat transfer systems. The sodium service system also contains a purification system to remove oxides from the sodium in order to control corrosion rates and to preclude impairment of mechanical and thermal performance of equipment. A helium system is used to provide an inert gas within the reactor and nitrogen is used for the inert gas atmospheres in the shielded cells. There are also systems to remove, store and dispose radioactive gases and radioactive liquid wastes. An organic cooling system provides auxiliary cooling to the concrete biological shielding of the reactor, certain sodium handling components, and the atmosphere in the shielded radioactive cells. Cooling water for the organic cooling system and other services in the nuclear plant is supplied from cooling towers provided for the condensers in the turbine-generator facility.

The fuel handling system consists of a fuel handling cask, fuel storage and washing cells, and all necessary equipment for receiving new fuel, refueling the reactor and shipping spent fuel.

A plant control system is provided which allows automatic operation over the range of 15 percent to 100 percent load, or manual operation at all loads. In addition, a plant protective system allows rapid detection of any off-normal condition in the reactor plant and automatically takes corrective action. This action includes alarm, one-circuit shutdown, setback, and scram. Table I summarizes data for the power plant.

THE REACTOR

The Consumers SGR has two innovations which differ from the SRE. The fuel in the SGR is low enrichment uranium dioxide, and the graphite is canned in stainless steel sheet. The use of oxide fuel is an outgrowth of the lack of stability of metallic fuel at temperatures over 700° F. Investigations to date indicate that burnups in excess of 10,000 MWD/ton are possible for the oxide, and that the cost of power will be less using the oxide. Although zirconium is used to clad the graphite in the SRE, we are uncertain of the life and long-term strength of zirconium or zirconium alloys in an SGR. The current design, therefore, includes stainless steel cladding. The substitution of zirconium alloy for stainless would permit a reduction of more than one percent in the fuel enrichment, with appreciably lower fuel costs, and evaluation and testing of zirconium alloys are continuing.

The core of the reactor consists principally of vertically supported, stainless steel clad, hexagonal graphite prisms with corners scalloped for fuel and control locations. Process tubes, attached to a lower grid plate, are used to channel high velocity sodium from a lower plenum through fuel elements at the fuel locations. Thimbles, hanging from the top shield, separate the control units from sodium at control locations. The grid plate is plugged at these locations to avoid excessive cooling of the control units and moderator cans. Heat is removed from these units by a separate coolant stream that feeds into a small plenum formed by the grid plate and the bottom of each moderator element.

Figure 5 shows a section through the reactor core. The assembly of moderator elements, control units, and fuel elements, is contained in a steel tank and located below floor level of the reactor building. Radially outward from the core is a core tank liner, the core tank, the thermal shield, an outer tank, insulation, the cavity liner, and a concrete foundation that serves as a neutron shield between the reactor and adjacent heat exchange cells. Below the core is a grid plate that supports the moderator and reflector elements and forms the coolant inlet plenum under the core, the tank bottom, a thermal shield, cylindrical steel supporting shells, and foundation concrete. Above the core is an eight foot deep sodium pool, a helium blanket, reflective insulation and the top shield. Sodium is brought to the lower plenum by three, double-walled pipes which enter the core cavity above the moderator elements, are brought down adjacent to the core tank, and terminate at the tank bottom. Sodium is removed from the pool above the core through pipes attached about one pipe diameter above the moderator elements, and then are offset upward so that the suction nozzles remain immersed in sodium if a leak develops in one of the heat exchange loops. The outer tank is built around the main tank to maintain the sodium level above the suction nozzles if a leak occurs in the core tank.

Present operating plans set the inlet sodium at 607° F and the outlet sodium at 945° F. The pressure drop across the core is approximately 20 psi at a sodium velocity of 11.3 feet per second in the fuel elements. The helium gas blanket will be maintained at a slight positive pressure of about 1/2 psig to prevent inward leakage of air. The current design limits all structural material to a maximum temperature of about 950° F. Ultimate plant operation may increase these limits to about 1000° F. Other design data for the reactor are outlined in Table II.

Design studies of reactor components require a "base case" reactor core. The present reactor design is the optimum case for a 19-rod uranium oxide fuel element. The fuel rods consist of 1/2-inch diameter oxide pellets, enriched to 3.5 percent, contained in 0.010-inch wall steel tubes. The core loading contains 243 of these rod cluster fuel elements and 21 dummy fuel elements. The total fuel weight is 56,000 pounds of uranium oxide. The active core forms a cylinder, 14 feet in diameter by 14 feet high, and is surrounded by additional reflector cells about two feet thick. An antimony-beryllium neutron source and approximately 40 shim regulating and safety rods also occupy corner positions in moderator elements. Each fuel cluster, or channel, is individually orificed to enable shaping the channel outlet temperature profile for the reactor. Maintenance cells, accessible from the reactor operating room, will enable changing orifices as necessary. A fuel cluster is suspended from its shield plug by a hanger rod. The outlet sodium temperature for each fuel channel is measured by a thermocouple in the hanger rod, immediately above the fuel.

A moderator element is shown in Figure 6. The element is 14 inches across the flat of the hexagonal block and 18 feet high. The stainless steel cladding is .016 inch thick, and all joints are fusion welded. There are 225 of the hexagonal elements in the reactor, including the reflector elements.

Control and safety elements operate in thimbles extending into the core from the top shield. Heat generated in the poison material of the shim or regulating rods is conducted across a helium annulus within the thimble, through the thimble wall, and into the sodium flowing over the outside of the thimble. About half the rods are for safety and are completely withdrawn from the active core during operation. These rods are released to fall by gravity into the core after a scram signal from the protective system. Development tests are in progress to determine the feasibility of a combined shim-safety unit, which would reduce the number of control units to approximately 20.

The top shield and loading face of the reactor is a plug of high density concrete, weighing about 400 tons, contained in a 24 foot diameter, 6 foot high shell of stainless steel. Supported on a roller bearing, the shield can be rotated, when the reactor is shut down, to permit access to the moderator cans if necessary. This design is an extrapolation from that used on the SRE which is 16 feet outside diameter. Ports and plugs penetrate the shield for each process channel in the core. Four ports, about two feet wide and four feet long, enable access to any moderator can in the core.

Fuel handling, for loading and unloading the reactor, and for the transfer of irradiated fuel between the wash cells, the storage cells, and the shipping or disassembly cells, is carried out with a shielded cask which is supported and moved with a gantry crane. The cask weighs about 140 tons and will traverse an area of the operating room, including the loading face, about 25 feet by 100 feet. It is tentatively planned that the SGR fuel will be reprocessed at Hanford, and we are working with the AEC to establish the extent to which the spent fuel elements will be disassembled for shipment. Based upon the experience with the SRE fuel cask and the present preliminary design, we estimate the annual plant down time attributable to fuel changing to be four or five days. If replacement, or investigation of moderator cans becomes necessary, internal parts of the fuel cask can be lowered into a service pit and the cask adapted to handle moderator units.

CORE DESIGN CONSIDERATIONS

At the present time (March, 1958) the core design is in the preliminary stages of optimization. The concept and general operating conditions are established as described above, but the detailed dimensions can still be adjusted for three to four months on permanent core items and until about January, 1960, on the fuel elements for the first core loading. The fuel elements under study are a 19-rod UO_2 element, currently considered the base case, a double ring UO_2 element, a 19-rod low alloy uranium metal element, and a 7-rod thorium-uranium element. The design approach is to establish a fuel element configuration from core design and heat removal limitations, predict nuclear performance and select optimum dimensions from economic studies.

Thermal considerations vary, depending on characteristics of the fuel element materials. The design assumption is usually made that the reactor size should be a minimum--consistent with the capacity to produce the specified thermal power and with the temperature limitations imposed by the physical properties of the central fuel element. Since the metallic rod type fuel elements are designed so that the temperature drop is low from the fuel surface to the coolant, the total heat production per unit length of fuel element is set by temperature difference in the fuel and independent of rod diameter. Assuming a 1200° F central temperature limitation for low alloy uranium, the only variable for the central fuel elements is sodium temperature at the vertical location of the maximum fuel temperature. This difference is maximized by tailoring the radial flow distribution across the reactor resulting in low channel exit temperature in the center of the reactor and high channel exit temperature in the outer region. A maximum design temperature is also placed on the exit channel sodium temperature by the design of the moderator elements and the reactor structure. The general result is that the inner fuel channels are limited by the fuel temperature and the outer by maximum coolant temperature. The maximum heat generation rate at the center of the reactor is approximately 300 kw/ft of fuel element for a Mo-U alloy now under study. The dashed lines on Figure 7 show the results of a heat removal calculation for a fixed power removal cross-plotted on nuclear curves discussed in the following paragraph.

Nuclear survey calculations are performed for a series of rod sizes at a fixed excess reactivity, plotting reactor size and initial conversion ratio vs. lattice spacing with enrichment as a parameter. The results of a typical rod size are shown on Figure 7. The cross-plot with the heat removal curve results in a series of reactor cores that meet the design bases for a specified rod size. An economic analysis of several cases then determines the minimum power cost for the given rod size, and when repeated for several rod sizes, results in a set of recommended core dimensions for the specific fuel element design.

The approach to the optimum core conditions for UO_2 differs slightly due to the heat removal characteristics of oxide fuel elements. If an oxide rod sealed in a helium atmosphere is considered, the temperature drop from the fuel to the cladding is no longer negligible and the total heat removal is partially dependent on rod diameter. Separate heat removal calculations are desirable for each rod diameter considered, as well as separate nuclear calculations. A simplifying condition exists, however, in the conditions for a desirable radial exit channel temperature. In this case, with UO_2, the temperature drive within the fuel is approximately 3200° F, and a relatively small range (50° F) in the radial exit temperature profile has little influence on the reactor size or total power removal. The assumption of a constant radial temperature allows a reduction in the maximum sodium temperature resulting in improved stress conditions, an increase in mixed mean sodium temperature, and a welcome simplification in heat removal calculations. The results of these studies for the 19-rod UO_2 fuel elements optimize at the base case design of approximately 210 kw/ft of fuel element in the center of the reactor.

The other oxide concept currently under consideration is a double-ring fuel element. Since the heat removal capability of a thin annular ring at a fixed temperature drive varies inversely with ring thickness, the number of cases needed to establish an optimum configuration is greatly expanded over rod designs, and digital codes are being used to reduce calculation time. The basic approach of heat removal, nuclear and economic studies is being followed, and the results follow similar trends. The double ring shows heat removal capabilities from 300 to 500 kw/ft.

Another current effort is to find a combination of dimensions for permanent parts of the core to allow a wide range in fuel element choice after the reactor is completed. The 19-rod metallic uranium fuel element optimizes at a small core with a large lattice spacing, the 7-rod thorium at a small core with small lattice spacing, the 19-rod oxide at a large core with small lattice spacing, and the double ring oxide at intermediate dimensions for both core size and lattice spacing.

The choice of core arrangement, after design and safety requirements are satisfied, is made by economic considerations. The economic ground rules being used are based on the assumption that the plant is owned and operated by a private utility in the United States. A first-year basis with a capital charge of 15 percent and an 80 percent plant factor is used. A lifetime basis with a capital charge of about 10 percent and a plant factor between 50 and 60 percent would result in approximately the same fixed charges. Maintenance and operating expenses, other than fuel costs, are assumed to be independent of the core arrangement. Fuel costs, including the lease charges for fissionable material, show the greatest differences when comparing various fuels and core arrangements.

The economic analysis will be repeated for each fuel concept described above. A fuel cycle is first described and unit costs estimated for each step. For example, the fuel cycle for rod-type uranium metal fuel elements includes the purchase of UF_6, conversion of UF_6 to derbies and derbies to slugs with appropriate scrap recovery steps, the manufacture of fuel elements, storage at the reactor, burnup in the reactor to an average 3000 MWD/ton U, decay storage, chemical processing to nitrates, and the conversion of nitrates to plutonium metal and to UF_6. A lease charge of 4 percent of the cost of the uranium at its initial enrichment is used for total uranium hold-up in the reactor and in the other steps of the fuel cycle. The time outside the reactor is estimated at 1.5 years. Average time in the reactor is one year for metal fuel and 3.5 years for oxide fuel. The cost of burnup for a given case is found from the nuclear characteristics, the UF_6 costs published by the AEC in November, 1956, and an assumed credit for plutonium. In most cases, both $12/gm Pu and $30/gm Pu are used for the plutonium credit. The capital cost estimate includes only the portion of the reactor facility that will vary with the fuel or core arrangement being considered. This usually includes the entire reactor structure, the fuel handling equipment, and the fuel storage facilities.

The planned design procedure is to complete the phase described above and define core dimensions, use these design conditions to conduct exponential experiments with two enrichments that are expected to bracket the final enrichment, and use the results of the exponential experiments plus additional analysis based on the experiments to set the enrichment for the first core loading. A critical experiment may also be run after the final enrichment is set to assist in planning startup and adjusting numbers of fuel elements and control units within the restrictions of the fixed reactor design.

SODIUM SYSTEMS

The sodium systems consist of the sodium heat transfer system and the sodium service systems.

The sodium heat transfer system, Figure 8, consists of three independent circuits with separate connections to the core tank. Each circuit handles one-third of the design full load sodium flow, transferring 85,000 thermal kilowatts, and consists of a radioactive primary loop and a non-radioactive secondary sodium loop. The primary sodium circuits transfer thermal energy from the reactor to intermediate heat exchangers, and the secondary circuits transfer thermal energy from the intermediate heat exchangers to the steam generators. Each intermediate heat exchanger is in a separately shielded cell, which enables isolating any single primary loop while the remaining two are operating.

There is a stop valve, flow control valve, and a check valve in each primary loop. The check valve vents sodium backflow in case of a primary pump failure. There is also a control valve in each secondary loop outside of the shielded cells. The control valves are used only to control the sodium natural convection flow during the period of dissipating reactor decay heat.

The expansion tanks in the secondary circuits have a regulated gas pressure to maintain the pressure of the secondary circuits higher than that of the primary circuits at the intermediate heat exchangers. This prevents radioactive contamination of the secondary circuits in the event a leak develops in an intermediate exchanger. All piping and components in contact with sodium are fabricated from 18-8 austenitic stainless steel, using welded joints throughout. All inaccessible sodium piping has instrumentation to detect sodium leaks. Sodium piping and equipment have electric heating to preheat the system before admitting sodium and also to keep the sodium in a fluid condition when the reactor is not in operation. All sodium piping and equipment are thermally insulated.

The pumps are of the centrifugal type and are rated at 7200 gallons per minute against a 150 foot sodium head. The pump drivers are constant speed, A.C. motors, connected through variable speed electromagnetic couplings.

Each of the three steam generator units consists of a boiler and steam drum, a superheater, and a desuperheater. A third fluid monitoring system is included with duplex tubes in both the boiler and the superheater. The boiler manufacturers have been asked to propose either 304 stainless on the sodium side and chromium-molybdenum steel on the water-steam side, or chrome-moly throughout. In the latter case, the superheater would be limited to 5 Cr-$\frac{1}{2}$ Mo, Ti, and $2\frac{1}{4}$ Cr-1 Mo could be used in the boiler. Water treatment will be the same as for a conventional boiler.

The sodium service systems, Figure 9, provide the means to fill, drain, flush and purify the sodium in the reactor core tank and the sodium heat transfer systems. In the filling operation, sodium is received in drums or in a railroad tank car. The sodium is melted, filtered, and distributed to the primary or secondary fill tanks. The reactor and the primary loops can be drained collectively, or separately, by the primary service pump to the primary fill tanks. The secondary circuits are drained to the secondary fill tanks by inert gas pressure or the secondary service pump. Sodium flushing is used to reduce the activity of a primary loop before the heat exchanger cells can be entered. The loop to be flushed is drained to a primary fill tank and then filled with fresh sodium from another of the primary fill tanks.

Cold traps are used to remove oxides from the sodium, and plugging meters are used to measure the oxygen content of the sodium. The cold traps are of the circulating type in which the sodium is cooled with tetralin to permit the deposit of the oxides in the traps. The primary system and the secondary system each have their own cold traps. The cold traps can be used to purify the sodium in the fill tanks, or in the heat transfer system.

AUXILIARY SYSTEMS

The reactor auxiliary systems include the organic cooling system, inert gas systems, radioactive vent system, and radioactive liquid waste system.

The organic cooling system provides cooling for all components in the reactor plant that are to be maintained at controlled temperatures. Because the system serves areas where possible leakage might result in inter-action with sodium, a coolant such as tetralin which is relatively inert to sodium is used. This system is essentially divided into two circuits. One circuit is composed of items which require uninterrupted coolant. The pumps in this circuit are also on the emergency power system to insure continuous operation under all conditions. The second circuit is composed of items which can stand uninterrupted service between the time there might be a power failure and the startup of diesel-generated power.

The inert gas systems consist of a helium system and nitrogen system. The helium system is used to establish an inert gas atmosphere within all piping vessels and equipment containing sodium. The nitrogen system is used to establish an inert gas atmosphere in the shielded areas where radioactive sodium might come in contact with the atmosphere in the event of equipment rupture. Individually pressure-controlled stations and relief valves maintain the gas pressures within the desirable limits for each user. The exhaust from relief valves on systems which may be radioactive are connected to the radioactive vent system.

The radioactive vent system collects all gas that is radioactive and delivers it to shielded storage tanks.

The radioactive liquid waste system collects, stores and disposes of the radioactive waste resulting from the cleaning of fuel elements or from decontamination operations in the maintenance cells. The fuel cleaning operation consists essentially of washing the spent fuel element with water, removing the hydrogen gas evolved into a radioactive vent system, and draining the waste water and sodium hydroxide formed to decay tanks where the liquid is held until the activity is reduced for final disposal.

REACTOR INSTRUMENTATION

The nuclear instrumentation, Figure 10, is divided into nine channels: two count rate, two log N, and five power level channels. These instruments provide adequate flux measurements from source level to approximately 150 percent of full power. In addition to providing reactor control signals, the nuclear instrumentation provides signals to the plant protective system. The neutron detectors are positionable in the instrumentation thimbles which are located in close proximity to the reactor core. The count rate and log N channels are used for reactor startup. Two power level channels (Nos. 8 and 9) are used for automatic power regulation. The remaining three power level channels (Nos. 5, 6, and 7) are used in the protective system through coincident circuits. The flux signals from all of the channels are indicated in the control room.

Thermocouples are used to measure fuel channel sodium exit temperatures. These temperature signals are used for normal reactor control as well as for actuating the reactor plant protective system. All fuel channel thermocouples are monitored. Additional thermocouples are located throughout the reactor vessel and biological shielding; these temperatures are monitored by means of a selector and indicator.

PLANT CONTROL

The plant control system, Figure 11, is intended to provide load following automatic operation over the range of 15 to 100 percent load, or manual operation at all loads. The control system is designed to provide operation as flexible as that of a conventional steam power plant.

The temperature program at steady state load conditions is shown in Figure 12. It is noted that the reactor inlet and outlet sodium temperatures are held fairly constant, thus minimizing transient stresses in the reactor. Also, the steam temperature and pressure to the turbine is held constant. The saturated water temperature in the steam drum decreases slightly on partial load to compensate for the reduced steam pressure drop, so that the steam pressure at the turbine will be constant at all loads. The secondary sodium temperatures, feedwater temperature, and the steam temperature at the superheater outlet are allowed to vary.

The regulating elements and controlled variables are as follows:

Regulating Elements	Program Variable	Reset Variable
Control rod position	Reactor power	Sodium outlet temp.
Secondary sodium pump speed	Secondary sodium flow	Steam pressure at turbine
Primary sodium pump speed	Primary sodium flow	Reactor inlet temp.
Feedwater valve	Feedwater flow	Water level in steam drum
Attemperator (desuper-heater) valve	Water to attemperator	Steam temp. to turbine
Turbine governor	Steam flow to turbine	Plant output

The flow signal from the steam line is an anticipating signal which will cause the sodium flow and reactor power level to respond rapidly to the turbine load demand.

In Figure 11 the steam dump is shown as flowing to the turbine condenser. In the final design, the steam dump could be to the turbine condenser (or a separate condenser), to the atmosphere, or a combination of both. The steam dump system would be used in the following operations: Reactor startup; to remove decay heat during a long shutdown; as a backup method for steam pressure control; as a partial relief during a turbine trip; and finally, during a load transfer from the coal-fired boiler to the nuclear plant while power is being generated. It is anticipated that the steam dumping system, which does not include the capacity of the safety valves, would be designed for approximately 25 percent of the plant rating.

REACTOR PLANT PROTECTIVE SYSTEM

The reactor plant protective system consists of an instrumentation and control system which will rapidly detect any off-normal condition in the reactor plant and, when necessary, automatically reduce reactor power level or scram the reactor. The reactor plant protective system is in addition to the plant control system, which serves to automatically regulate power generation and reactor temperatures during normal operation. The reactor plant protective system provides the following types of action in the event of off-normal conditions: Alarm; shutdown one sodium heat transfer circuit; reactor setback; reactor scram. By a combination of these various types of protective action, reliability is insured while maintaining the maximum practicable level of safety.

SAFETY CHARACTERISTICS

The reactor and the sodium systems operate under low pressure. High pressure is found only in the steam and feedwater systems. The reactor and its associated radioactive components are located in shielded areas. Containment of radioactive fluids is provided by welded steel containers and piping, steel liners on the concrete shielding, and sealed joints on the removable shielding. An inert gas atmosphere is used in all areas containing primary system piping and components. Within the reactor there is no releasable potential energy from pressurization or chemical reaction.

An automatic plant control system provides for safe normal operation of the reactor power plant. A plant protective system provides additional automatic control of the reactor systems for safety against failure or improper operation of any component. The protective system is designed for reliability, with reactor scrams reduced to a minimum, and in general supplemented by means which provide merely a power reduction or a shorter shutdown time.

The regulating and shim rods, which serve to control reactor power level during normal operation, have a rate which is limited to ensure that excessive reactivity cannot be inserted by erroneous action. Safety rods which fall by gravity into the reactor core provide scramming action in the event other protective system action cannot adequately assure plant safety. The strong negative prompt fuel temperature coefficient of reactivity provides reactivity control over upsurges in fuel temperature.

During a plant shutdown, one or more of the three independent sodium heat transfer circuits may be used for cooling the reactor to remove stored and afterglow heat. Natural convection is more than adequate to transfer the afterglow heat after full power operation.

COMPONENT DEVELOPMENT PROGRAM

Due to the very considerable scale-up in power output and physical dimensions of the Consumers SGR reactor core and plant components over the SRE, and the desirability of incorporating improved materials and designs, the USAEC is supporting a broad component development program. It is a major objective of this program to confirm the suitability of each major component for its intended application before construction of the reactor plant. This program is testing both sub-components and full scale prototypes under simulated reactor operating conditions. The component development program covers fuel elements, control and safety rods, control instruments, moderator and reflector assemblies, sodium instrumentation, pumps, valves, and other sodium system components. It also includes the development by suitable in-pile tests of improved fuel and control poison materials. Hydraulic studies on models will be made to provide design data for the fuel elements and core structural components. Experimental determination of temperature gradients and stresses will be made in components subjected to high thermal stresses. Handling techniques for removal of all reactor core components will be developed with a full scale mockup. All of the information to be derived from the above work is scheduled for completion early enough in the overall program to support the final design and construction effort. During the construction period on the plant, proof tests in sodium at design temperatures and pressures will be run on all production units of such components as mechanical pumps and control and safety rods before they are installed in the plant. In addition, it is planned to run life tests on spare production units of these components.

Given below is a summary of the development work currently planned and in progress on major components:

Fuel Elements - Consumers SGR fuel elements will operate at higher surface temperatures than those for any other large scale civilian power reactor in the present AEC Demonstration Program. Due to the desire to achieve high burnup and obtain minimum fuel cost, attention is being concentrated on the use of high density UO_2 fuel clad in a thin jacket of stainless steel. An in-pile test program is underway utilizing both the Materials Testing Reactor and the SRE to determine dimensional stability and fission gas release from a variety of prototype fuel elements. Figure 13 is a sketch showing two types of in-pile test capsules being used in the MTR reactor. One type of capsule is designed primarily for determination of fission gas release after high burn-ups, while the other is intended chiefly to determine central temperatures of the UO_2 by incorporation of special thermocouple instrumentation in the test assembly. These capsules will test elements with fuel up to a .356" in diameter by 18" long. Larger scale tests of fuel element assemblies in the SRE will be made using the standard fuel lengths of six feet. Fuel burnups in the range of 10,000 to 20,000 MWD/ton will be achieved in this program.

Moderator-Reflector Assemblies - In addition to the increase in size of the graphite moderator units, serious consideration is being given to the use of .016 inch thick type 304 stainless steel cladding rather than the .035 inch thick sponge-zirconium cladding used in the SRE. A fabrication development program involving forming of the required shapes to close dimensional tolerances and joining by heliarc welding to extremely high quality requirements is underway. A number of full scale prototypes are being made for a test program aimed at determining the adequacy of design with respect to pressure and thermal stress. In the full scale tests it is planned to thermally cycle a group of seven moderator units in a large sodium tank. In addition to these full scale tests, tests will be run to prove out some of the detailed design features. Figure 14 shows a photograph of a partially assembled two foot long test moderator section used for experimental stress analysis in one of the early stages of this program.

Control - Safety Rods - The development program on control and safety rods is following two lines of approach; first the test of scaled-up units of the type proven in SRE experience involving separate control and safety rods, and second, the development of units which combine the functions of shim control and safety. This latter approach is highly desirable to reduce the total number of units in the core but involves heat transfer and materials' problems. In order to obtain necessary design information required for this concept, an in-pile test has been run in the SRE employing a special test assembly as shown in Figure 15. Cylindrical poison rings of several materials and of various dimensions and clearances within the thimble were irradiated with temperature and flux instrumentation to confirm analytical calculations of temperature profiles within the poison material. The alternate poison materials being reviewed include boron-nickel alloy, boron carbide, hafnium, hafnium oxide, and several rare-earth oxides.

Mechanical Pumps - Full size prototype centrifugal pumps of two types, both of 7200 gpm capacity at 150 foot sodium head, are being procured for exhaustive tests. One type employing a frozen sodium shaft seal is illustrated schematically in Figure 16. Another type involving a free sodium surface and incorporating a hydrodynamic radial bearing supplied with sodium through a by-pass line from the pump discharge will also be tested. The prototype pumps will be installed in separate loops of 12 inch pipe size which contain valves, flow meters, and instrumentation required to determine pump characteristics and performance. It is planned to test each pump for several thousand hours before specifying the type to be used in the SGR plant. Life tests of the selected unit will then be run concurrent with the manufacture of the production units.

Valves - In the SRE, valves up to six inches, employing both bellows seals and frozen sodium stem seals, have been used. Operating experience to date with valves which were largely of conventional design, with some minor modifications, has indicated the desirability of incorporating improved features for the larger valves up to 14 inches required for the SGR. By procurement and test of a number of standard commercial and specially designed valves, the best design features of each will be determined and incorporated in the specifications for the production valves. The valve test program will involve determination of across-seat leakage, both before and after thermal cycling, stem leakage and life tests of the valve operators. Figure 17 shows a test rig used in determining effectiveness of various frozen sodium stem seals by cycling pneumatic stem operators. The experimental valve stems illustrated are air-cooled and employ an alkali-resistant, conventional packing as a backup for the frozen sodium seal.

Steam Generator - To verify the suitability of the Consumers SGR steam generator, a quarter-size model (approximately 20,000 kilowatts or 60,000 pounds of steam per hour) will be tested in the SRE plant. Testing of this model steam generator will be done in two phases; first, with the steam generator following the reactor power level, and finally, after successful operation in the above manner with the reactor-steam plant complex operating as a load following system. Operation of the model steam generator will provide design information and heat transfer data under both steady state and transient conditions. It will provide assurance that the steam generator and reactor comprise a compatible system, and in addition, will serve to verify the adequacy of the fabrication and quality control procedures used in manufacture. Fabrication of the model steam generator is scheduled for completion by the summer of 1959. This will provide approximately a full year of testing before fabrication of the Consumers' units will be started.

Cold Traps - Operation of the SRE has indicated several areas where significant improvements can be made in the cold traps which are used to reduce the oxygen content of the sodium coolant to an extremely low value. In the primary sodium system, as a result of the free sodium surface in the core tank, some contamination with oxygen is unavoidable during handling operations on core components. For this reason, the Consumers' prototype cold traps are being designed to operate efficiently and rapidly to reduce the oxygen content of "dirty" sodium. In order to determine the operating performance of the cold trap, a full scale prototype will be tested in a loop in which oxygen will be continuously added to the sodium before it enters the trap---in this way simulating the conditions that would prevail during initial cleanup of a large quantity of sodium.

System Pre-heating - It is necessary to develop a reliable, inexpensive system for initial preheating of the main coolant and service system piping prior to filling with sodium. In the SRE this was done with a rather elaborate system of electric resistance heaters and controls. An experimental investigation of two alternate heating methods is being planned for the 12 inch pump test loops previously described. In one of these loops experiments will be conducted using a direct transformer heating method in a balanced-bridge circuit which is aimed at elimination of leakage currents to connecting lines and grounds. In the second pump loop, experiments on low frequency induction heating will be performed using various thicknesses of carbon steel wrapped around the stainless steel piping. It is hoped that this experimental program will lead to the development of a simpler and less expensive installation for the Consumers SGR.

Sodium Instrumentation - Specialized instrumentation is required to measure level, pressure and flow of the sodium at various points in the reactor plant. The development program will extend existing instruments to the larger sizes required and will attempt to obtain improved reliability and lower temperature sensitivity than the instruments now used in SRE. Encouraging progress has already been made in the development of high temperature insulation, permitting level coils to operate for extended periods at temperatures of 1000° F. Considerable effort will be required on flow meters for the indicated 14 inch piping size to develop accuracy over the full flow range and to eliminate calibration changes with time due to temperature. Work is also underway on an improved air-cooled plugging meter device used for determination of the oxygen content of the sodium.

TABLE I

Power Plant Data

Nominal Rating	75,000 electrical kilowatts
Design Net Electrical Power	80,000 kilowatts
Design Gross Electrical Power	86,000 kilowatts
Reactor Thermal Power	254,000 kilowatts
Steam at Turbine Throttle	
Pressure	800 psig
Temperature	825° F
Steam Flow to Turbine	752,000 lb. per hour
Condenser Pressure	1.5 inches Hg abs.
Boiler Feed Water Temperature	304° F
Net Plant Heat Rate	10,800 BTU/kilowatt hr.
Net Plant Thermal Efficiency	31.5%

TABLE II

Reactor Data

Fuel		
Material	Enriched Uranium Dioxide (density 10.2 gm/cc)	
Enrichment		.035 U-235
Inventory UO_2		56,000 lb. (25,500 Kg)
U		49,400 lb. (22,500 Kg)
U-235		1,709 lb. (781 Kg)
Burnup		10,000 MWD/ton, Avg.
		16,000 MWD/ton, Max.
Average Core Lifetime		3 years
Form of Fuel Element	Cluster of 19 rods, ½ in. dia., 14 ft. long	
Cladding	Stainless Steel Tubing, .010 in. wall thickness	
Core Design		
Power		254,000 thermal kilowatts
Active Core	Diameter	14 ft.
	Height	14 ft.
Reflector		2 ft.
Overall Core Dimensions	Diameter	18 ft.
	Height	18 ft.
Form	225 hexagonal prisms of graphite	14 in. wide, 18 ft. high, clad in stainless steel
Process Channels		
Inside Diameter		3.3 in.
Total Number Available for Fuel, Control, Source, Instrumentation		300 (Approx.)
Lattice		14 in., hexagonal
Fuel to Moderator Volume Ratio		.051
Axial Flux Peak to Average		1.4
Radial Flux Peak to Average		1.72
Initial Conversion Ratio		.65
Excess Reactivity (Poison, Depletion Temperature)		7.5%
Sodium Flow Rate		8.46×10^6 lb./hr.
Sodium Inlet Temperature		607° F
Mixed Mean Sodium Outlet Temperature		945° F
Temperature Rise through Reactor		338° F
Central Fuel Element		
Maximum Fuel Temperature		4000° F
Power Output		2100 kilowatts
Maximum Heat Flux		300,000 BTU/hr.ft.^{2}
Maximum Rod Power		12 kilowatts/ft.
Life, Full Power Days		547

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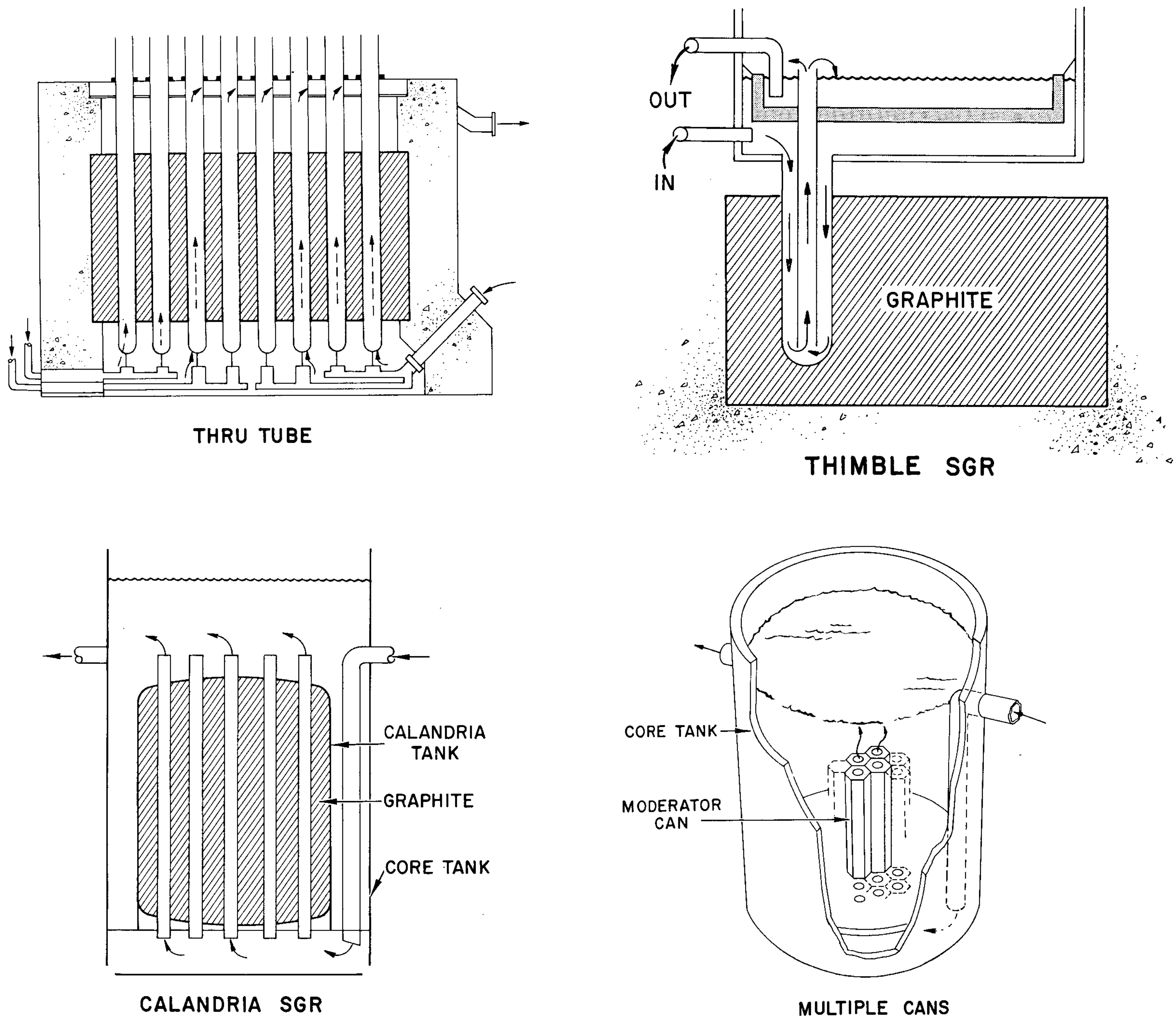


Fig. 1. Outline Drawing of SGR Arrangements

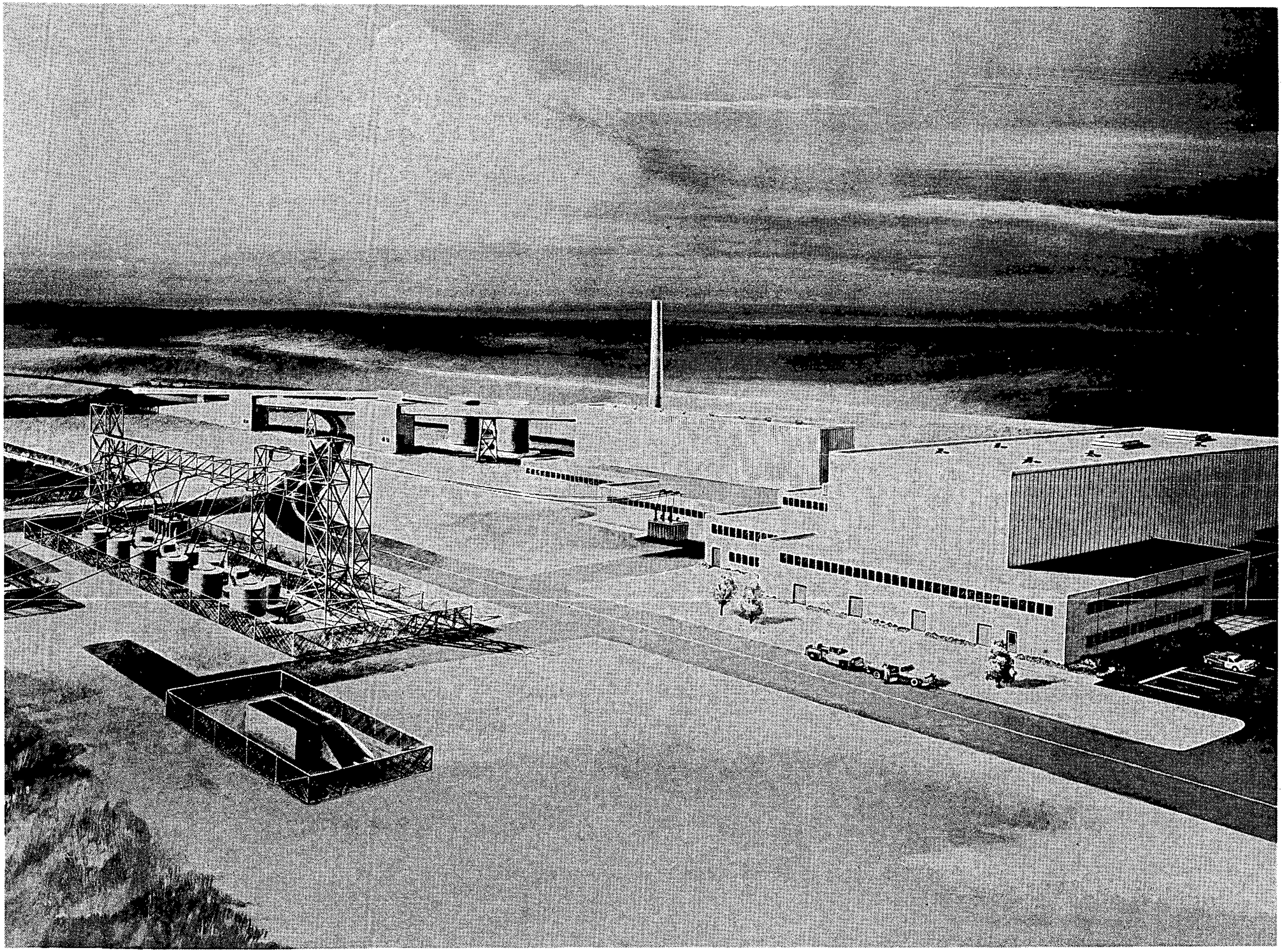


Fig. 2. Perspective of CPPD SGR Nuclear Power Plant

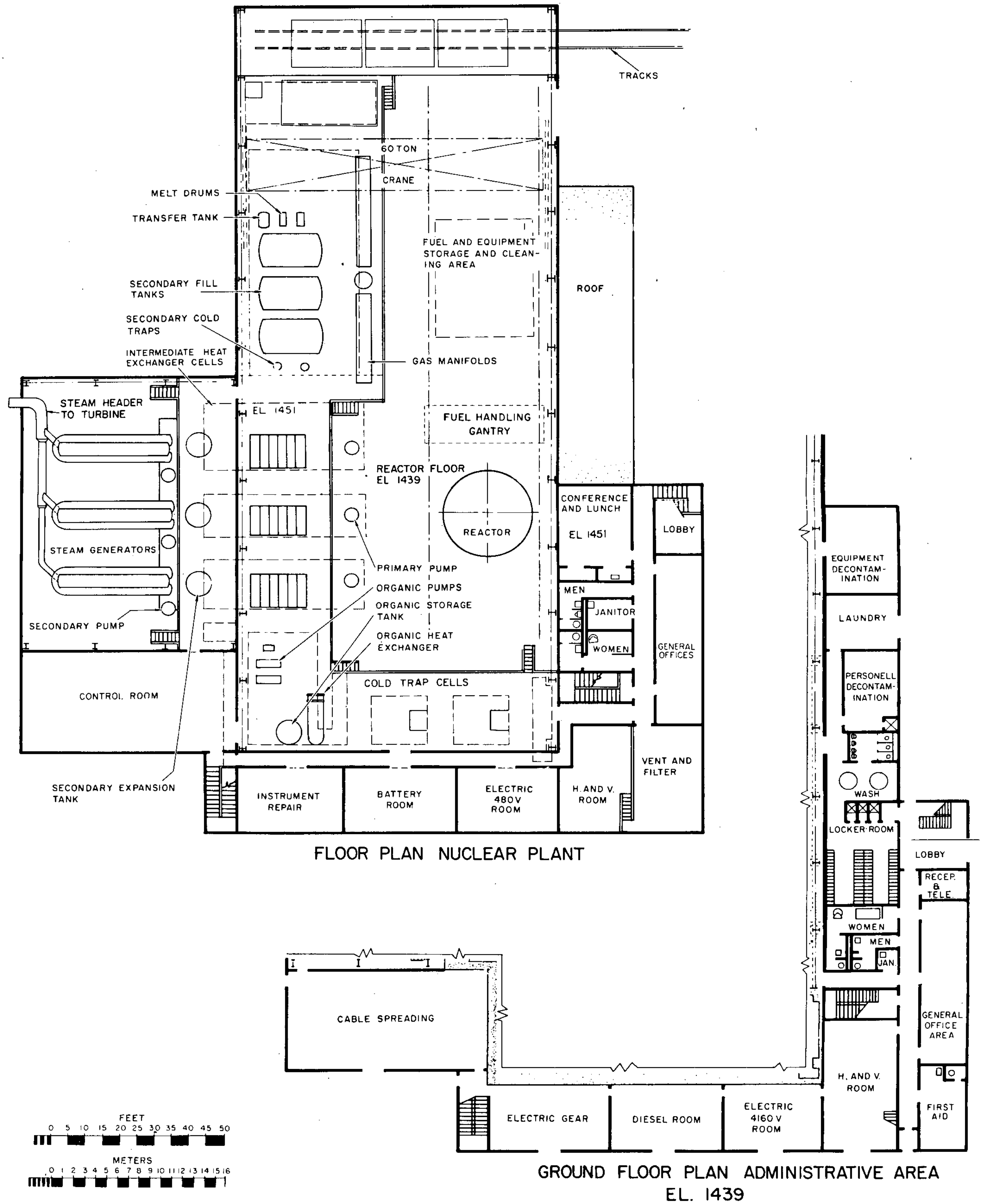


Fig. 3. Plan of the Nuclear Facilities

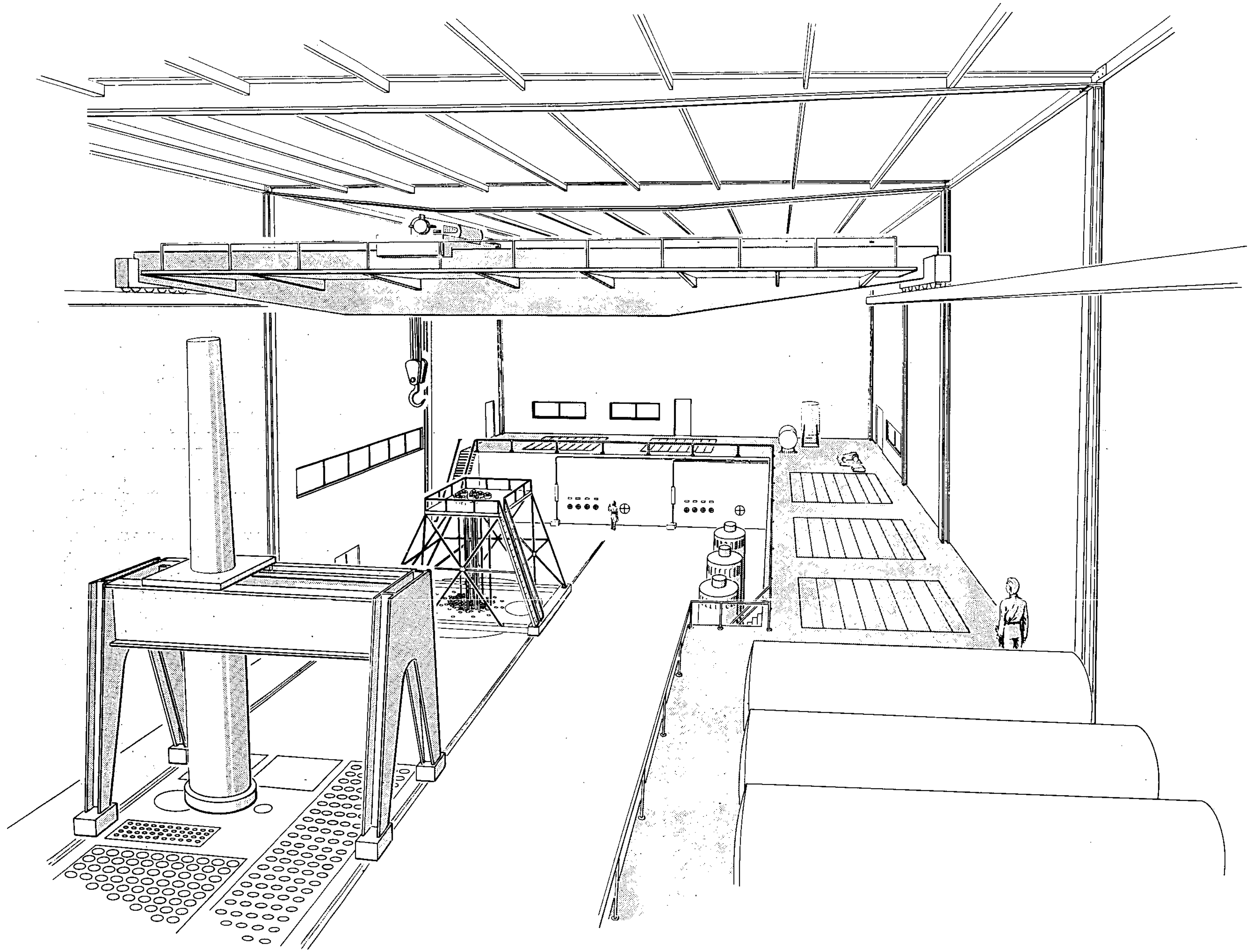


Fig. 4. Perspective of Interior of Reactor Room

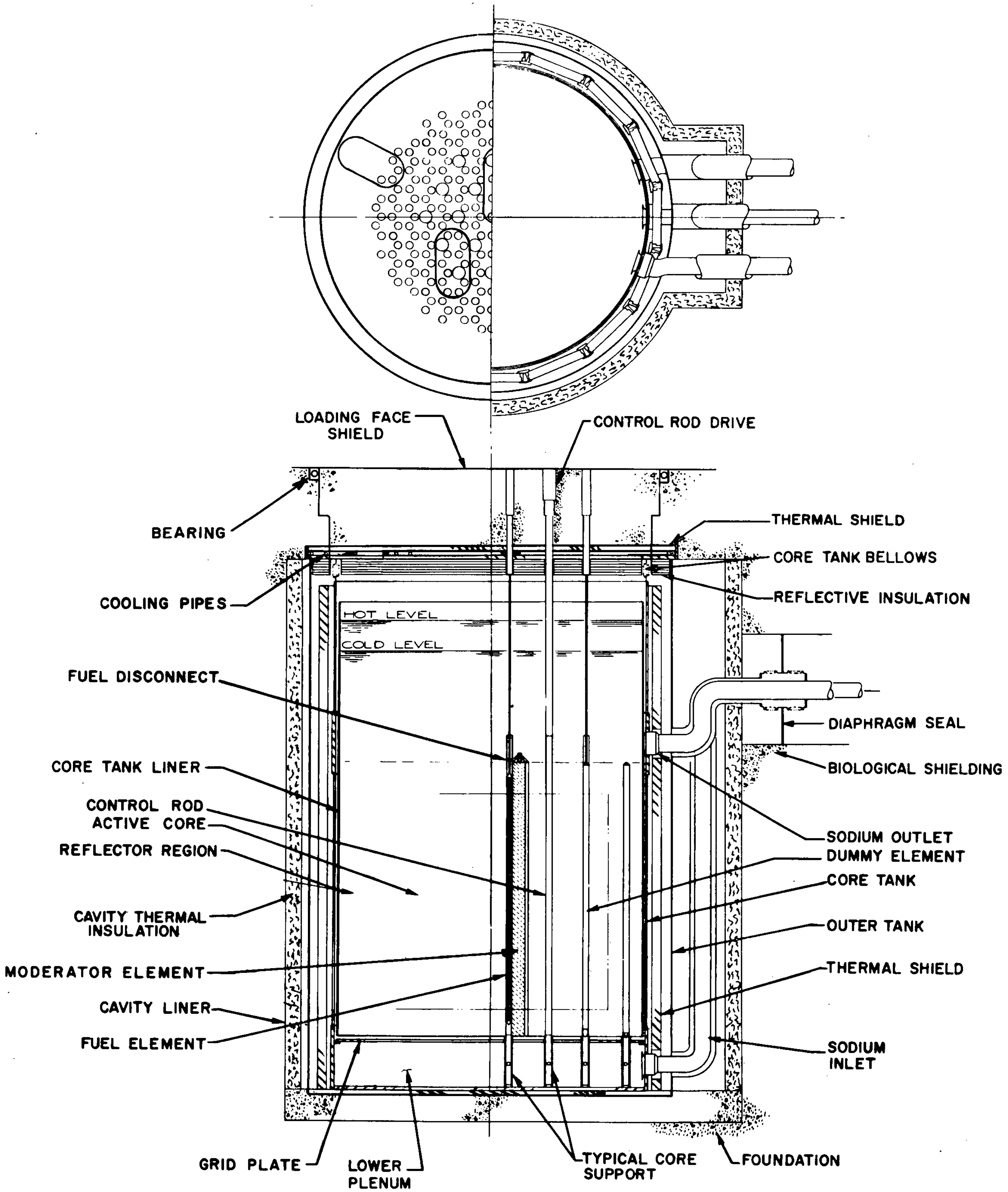


Fig. 5. Cross-section of the SGR

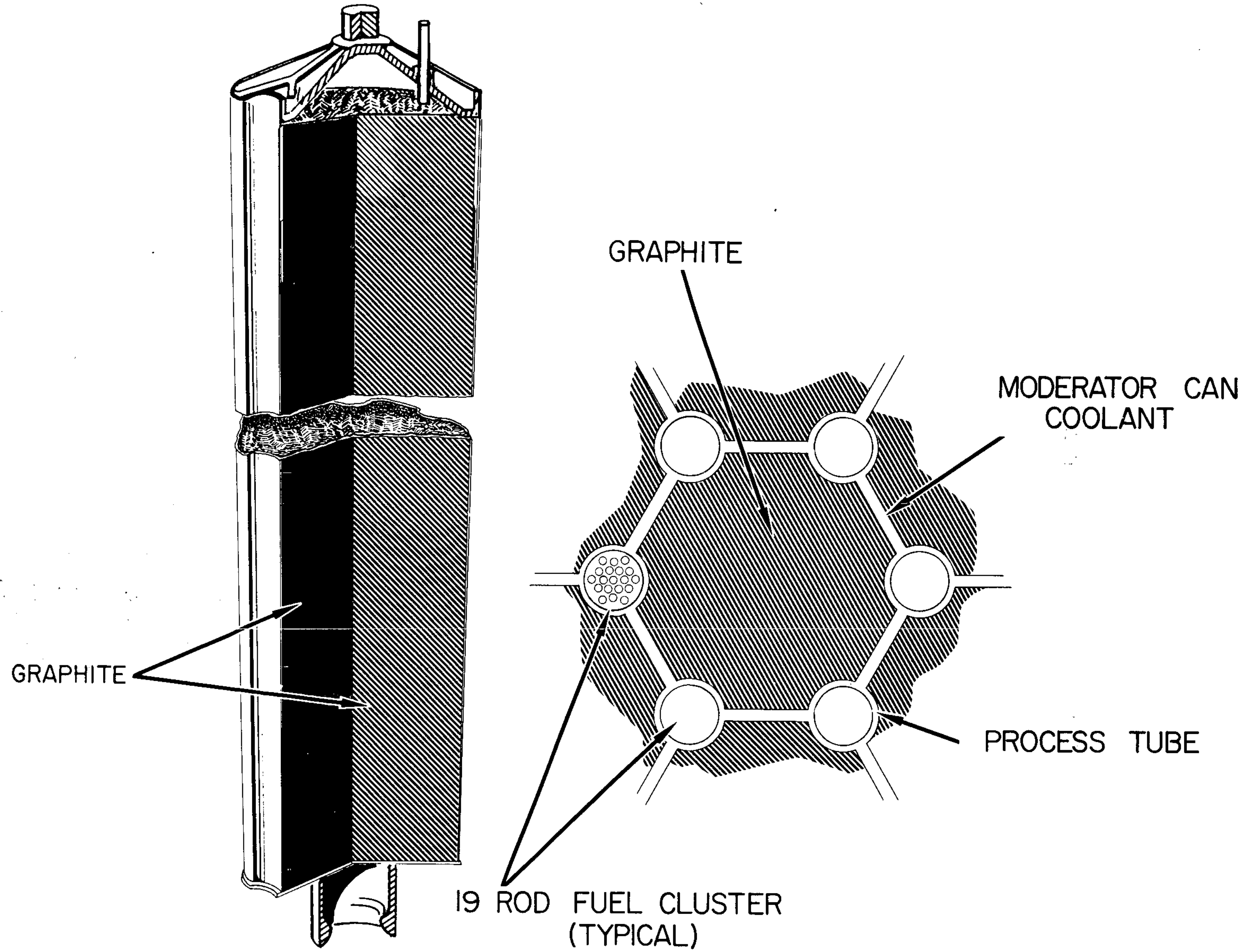


Fig. 6. Moderator Can

NUCLEAR-HEAT TRANSFER PARAMETER CURVES

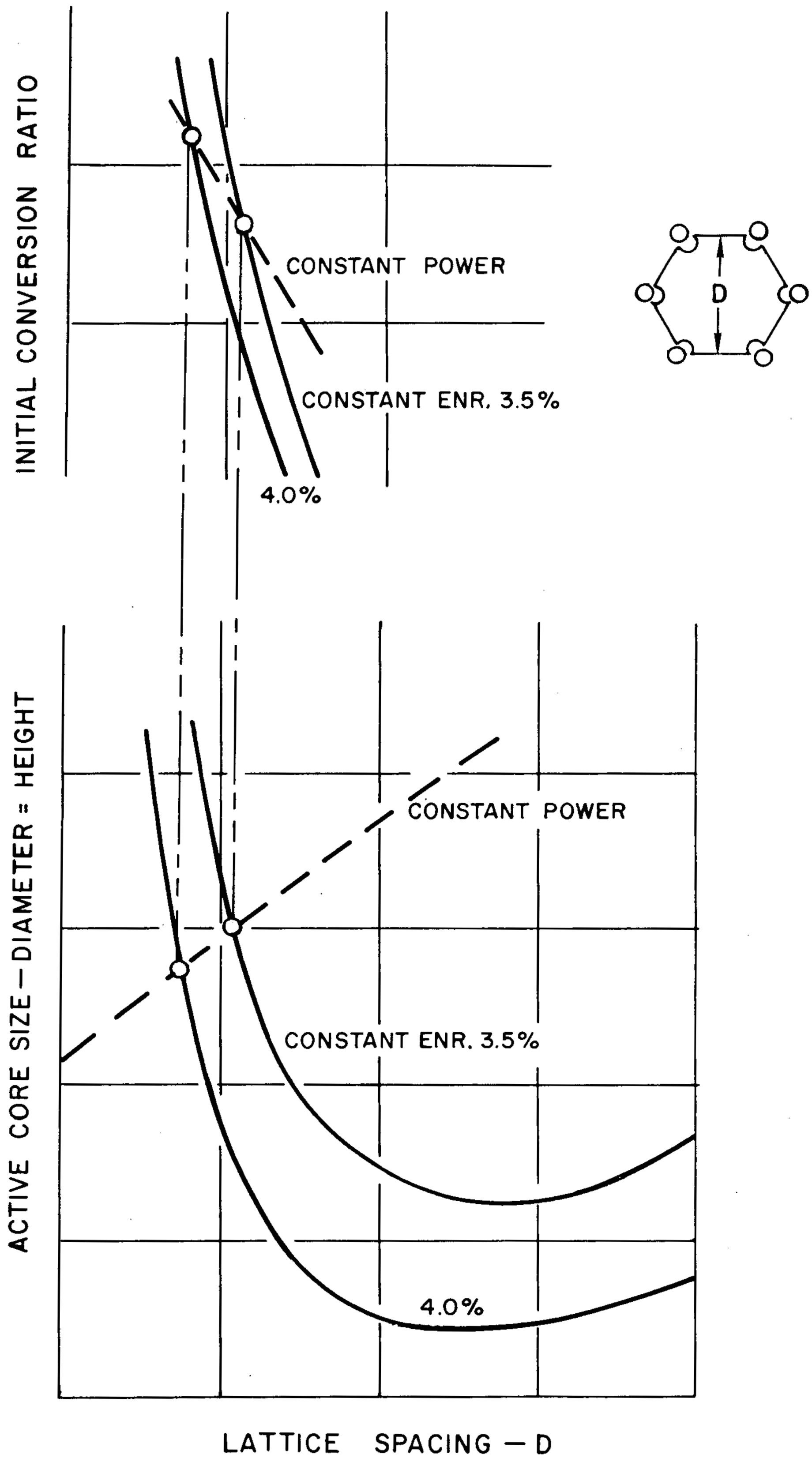


Fig. 7. Nuclear Heat Transfer Parameter Curves

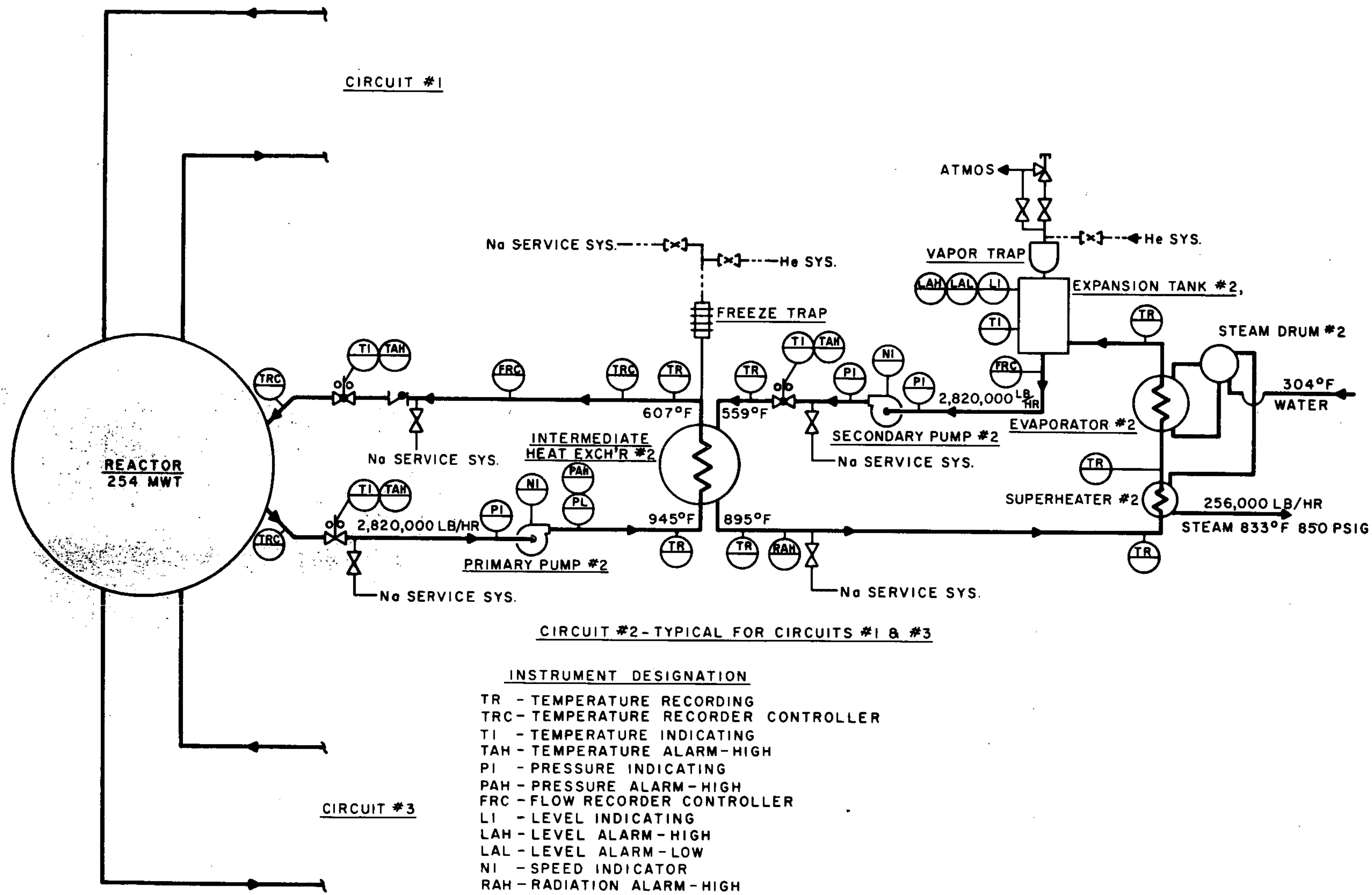


Fig. 8. SGR Heat Transfer System

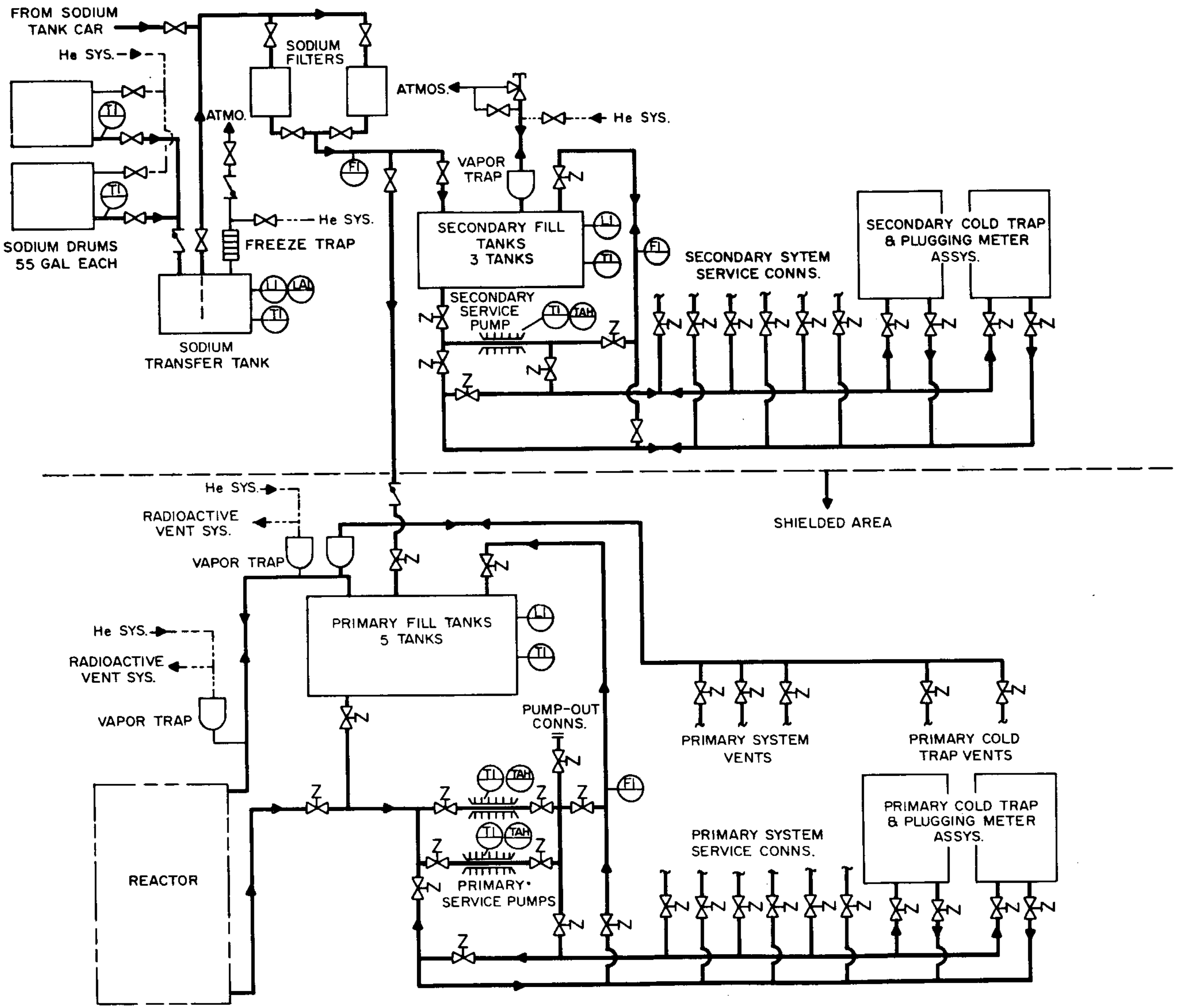


Fig. 9. SGR Sodium Service System

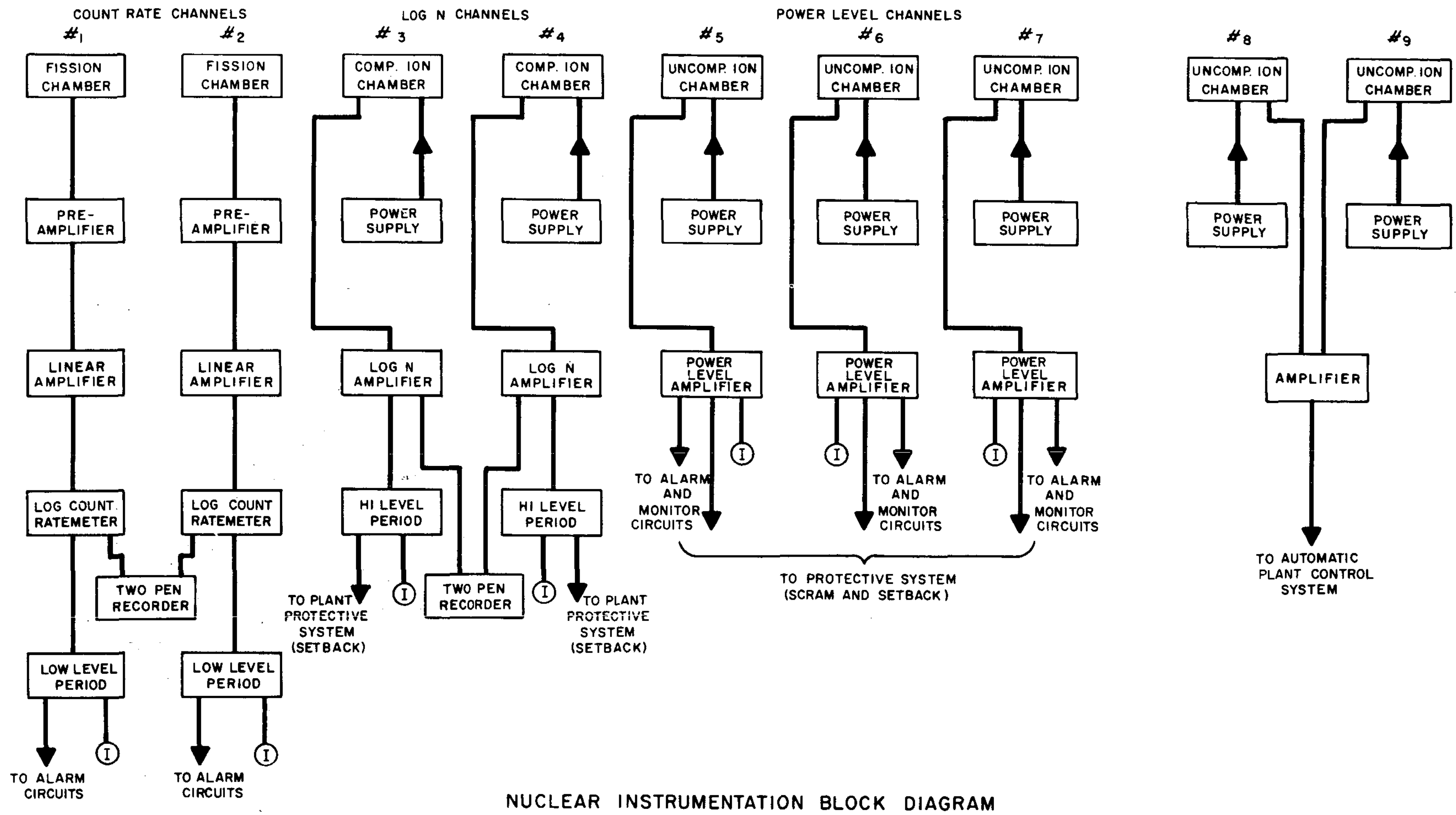
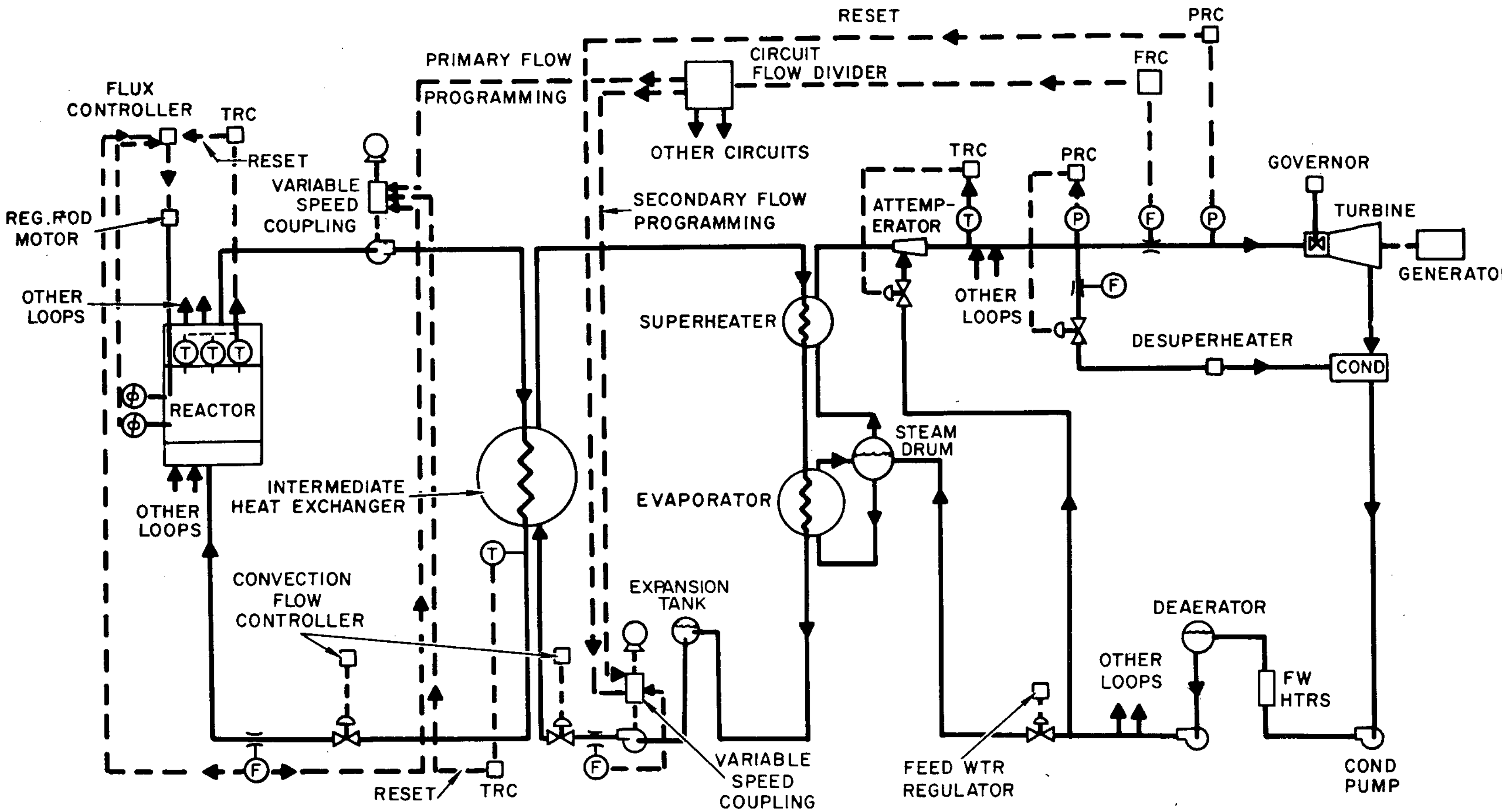


Fig. 10. Nuclear Instrumentation Block Diagram



(T) TEMPERATURE
(P) PRESSURE
(F) FLOW
(Φ) NEUTRON FLUX

FRC - FLOW RECORDER CONTROLLER
TRC - TEMP. RECORDER CONTROLLER
PRC - PRESS. RECORDER CONTROLLER

Fig. 11. SGR Plant Control System

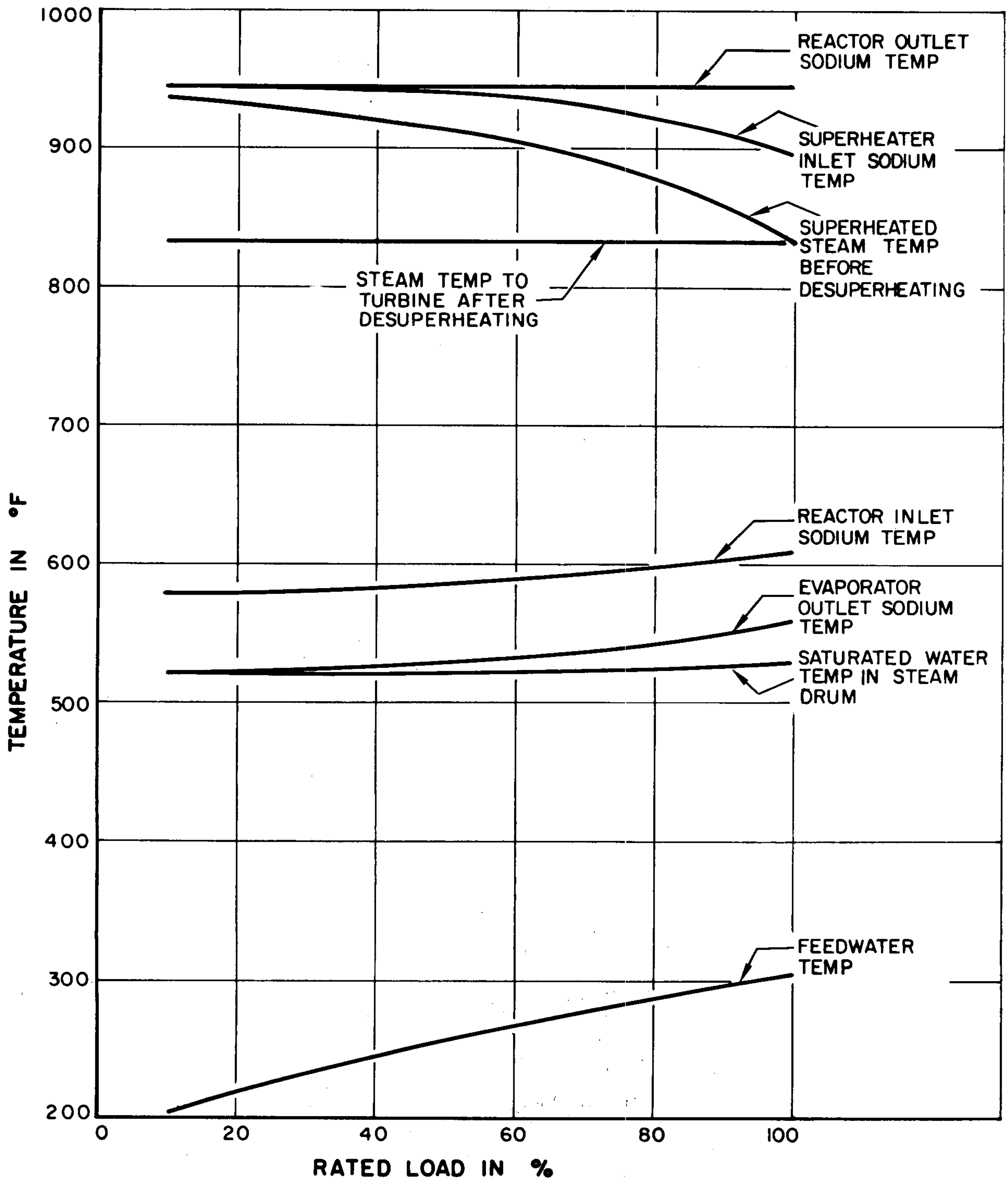


Fig. 12. SGR Plant Temperature Program

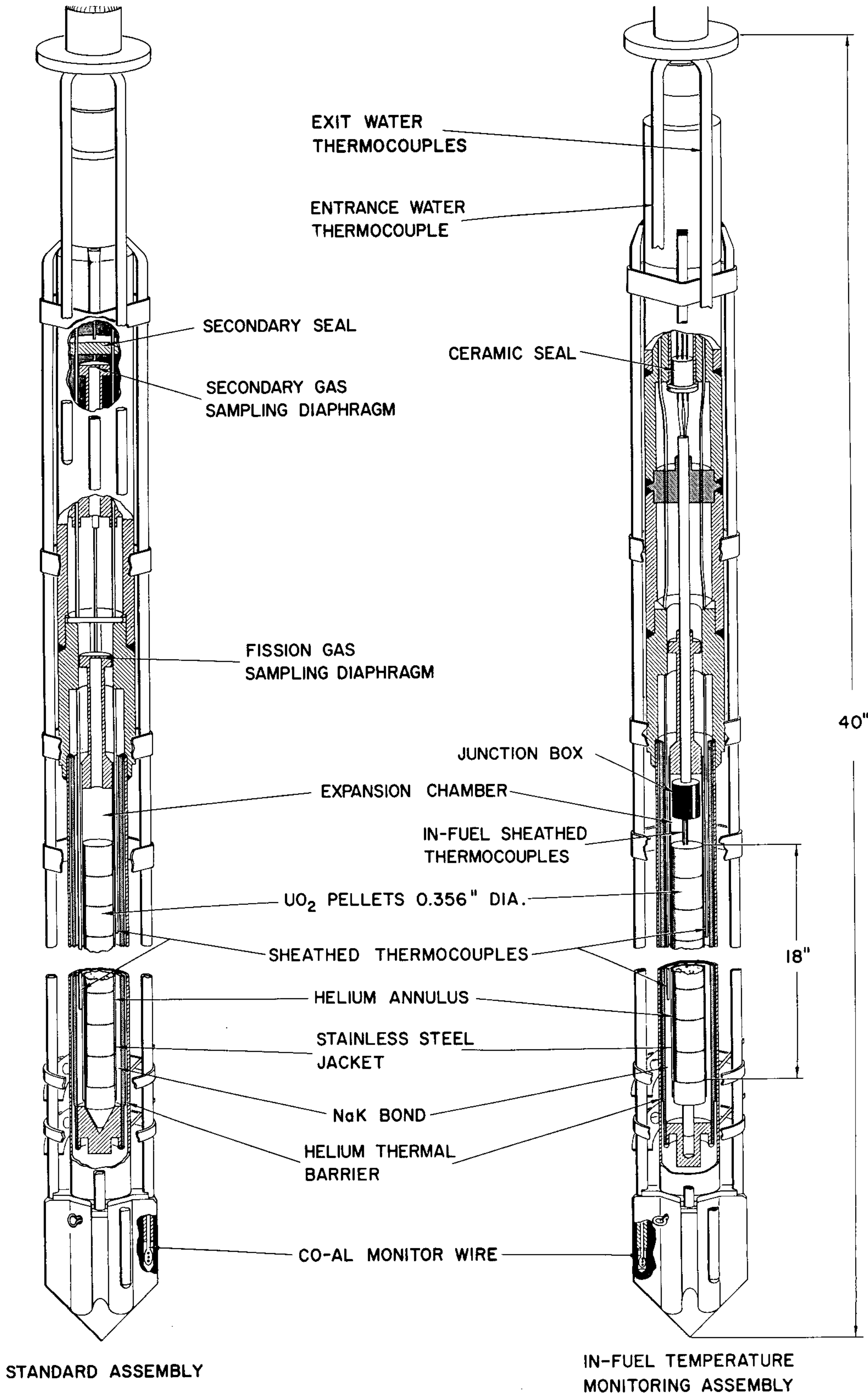


Fig. 13. UO_2 Fuel In-Pile Test Assemblies

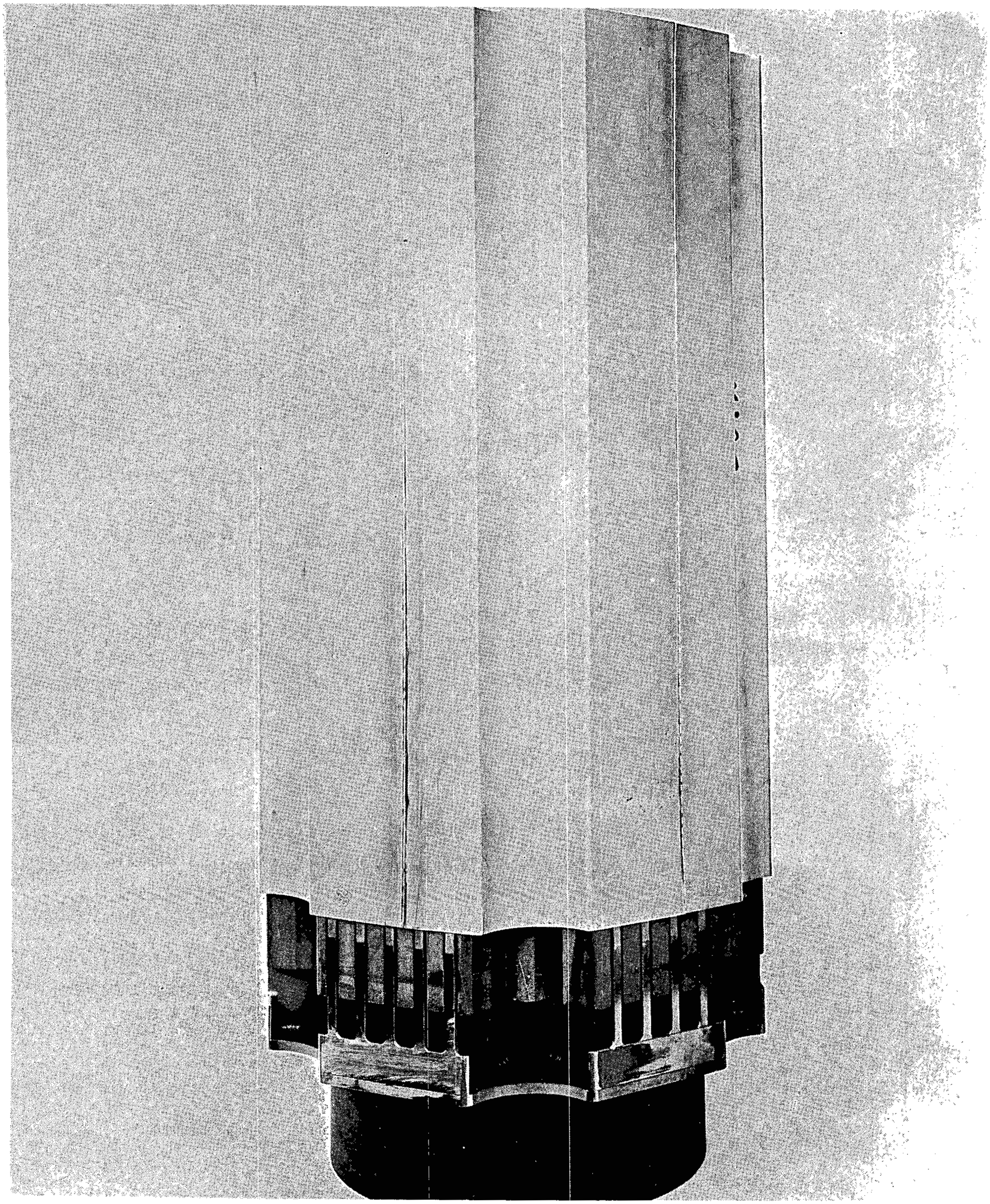


Fig. 14. Moderator Can Test Section for Experimental Stress Analysis

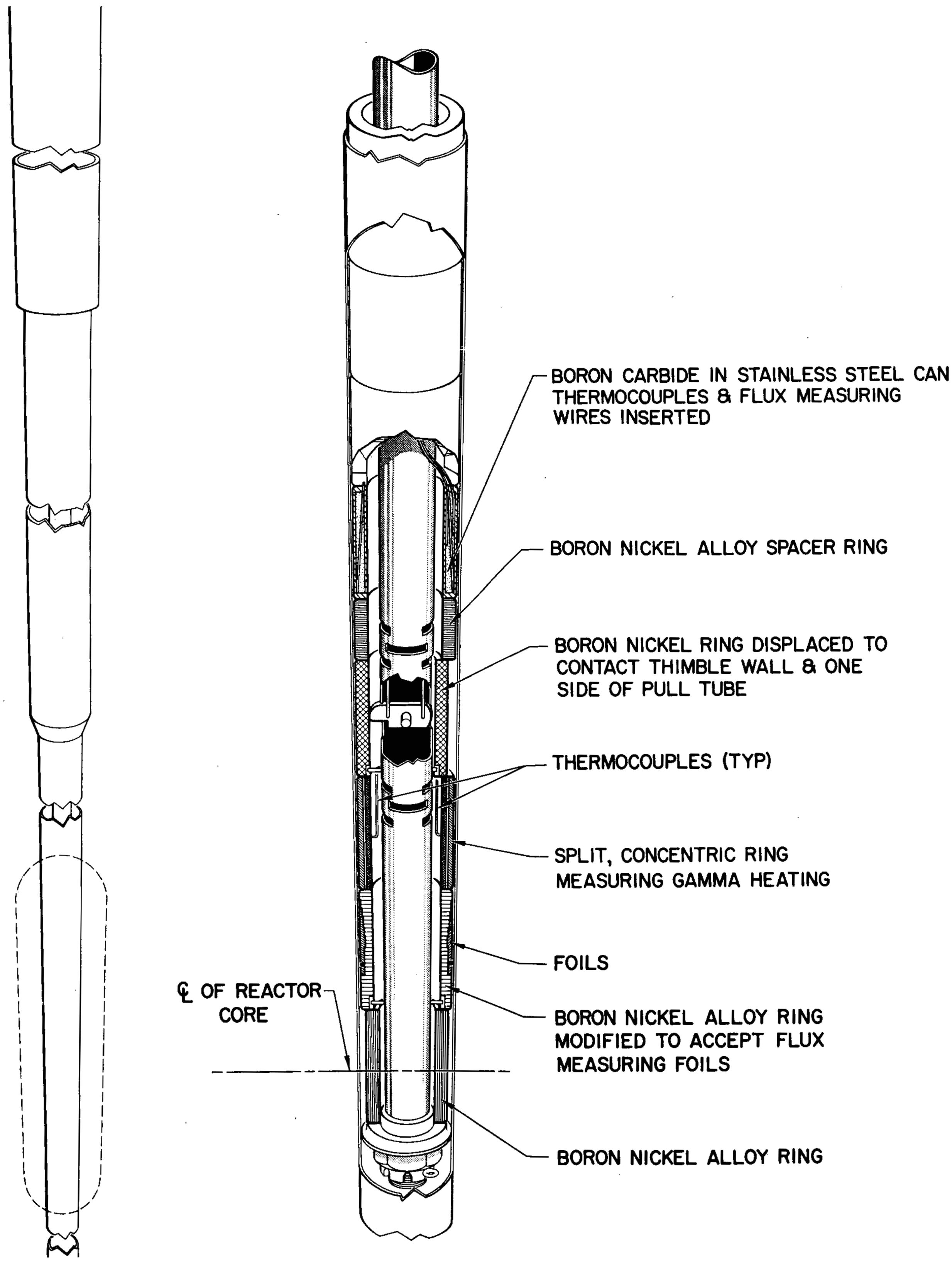


Fig. 15. In-Pile Test of Control Poisons

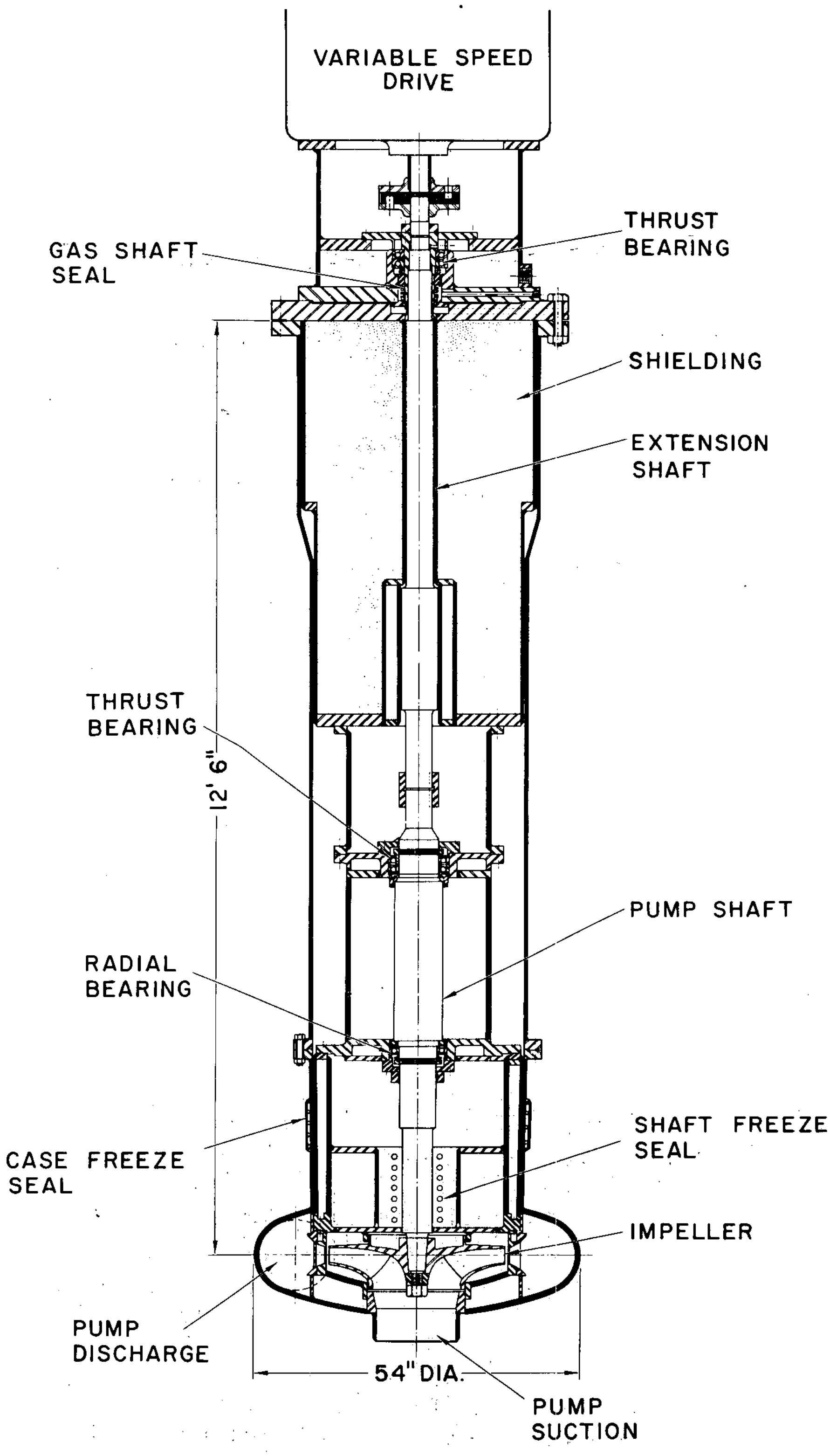


Fig. 16. Prototype SGR Sodium Pump

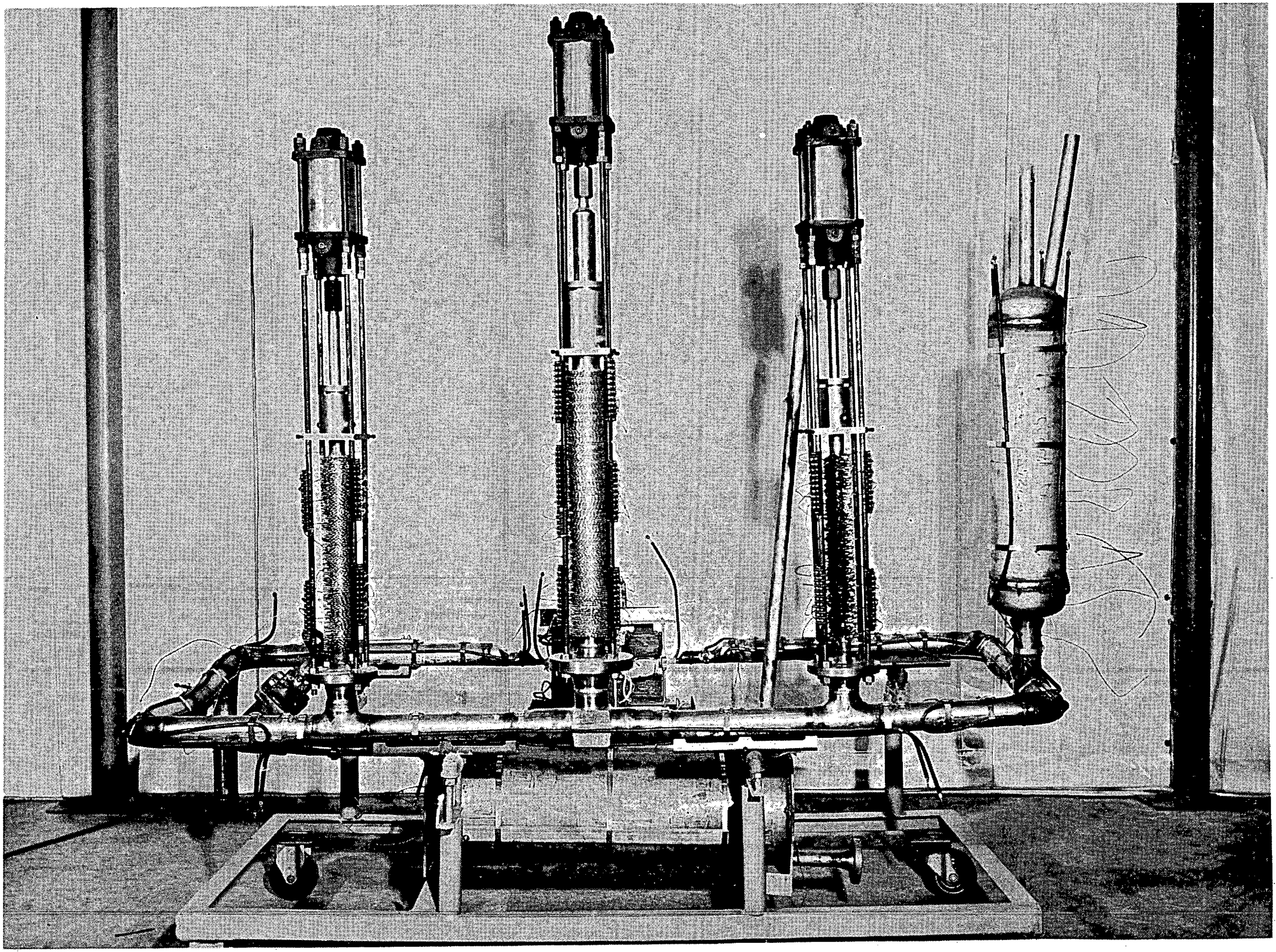


Fig. 17. Sodium Valve Stem Freeze Seal Test

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