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on the

Peaceful Uses of Atomic Energy

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Power Reactors

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PREFACE

The Proceedings of the International Conference on the Peaceful Uses of Atomic Energy are published in a series of 16 volumes, as follows:

Volume Number	*Title*	*Sessions Included*
1	The World's Requirements for Energy; The Role of Nuclear Power	2, 3.2, 4.1, 4.2, 5, 24.2.
2	Physics; Research Reactors	6A, 7A, 8A, 9A, 10A.1.
3	Power Reactors	10A.2, 3.1, 11A, 12A, 13A, 14A.
4	Cross Sections Important to Reactor Design	15A, 16A, 17A, 18A.
5	Physics of Reactor Design	19A, 20A, 21A, 22A, 23A.
6	Geology of Uranium and Thorium	6B, 7B.
7	Nuclear Chemistry and the Effects of Irradiation	8B, 9B, 10B, 11B, 12B, 13B.
8	Production Technology of the Materials Used for Nuclear Energy	14B, 15B, 16B, 17B.
9	Reactor Technology and Chemical Processing	7.3, 18B, 19B, 20B, 21B, 22B, 23B.
10	Radioactive Isotopes and Nuclear Radiations in Medicine	7.2 (Med.), 8C, 9C, 10C.
11	Biological Effects of Radiation	6.1, 11C, 12C, 13C.1.
12	Radioactive Isotopes and Ionizing Radiations in Agriculture, Physiology and Biochemistry	7.2 (Agric.), 13C.2, 14C, 15C, 16C.
13	Legal, Administrative, Health and Safety Aspects of Large-Scale Use of Nuclear Energy	4.3, 6.2, 17C, 18C.
14	General Aspects of the Use of Radioactive Isotopes; Dosimetry	7.1, 19C, 20C.
15	Applications of Radioactive Isotopes and Fission Products in Research and Industry	21C, 22C, 23C.
16	Record of the Conference	1, 24.1, 24.3.

These volumes include all the papers submitted to the Geneva Conference, as edited by the Scientific Secretaries. The efforts of the Scientific Secretaries have been directed primarily towards scientific accuracy. Editing for style has been minimal in the interests of early publication. This may be noted especially in the English translations of certain papers submitted in French, Russian and Spanish. In a few instances, the titles of papers have been edited to reflect more accurately the content of those papers.

The editors principally responsible for the preparation of these volumes were: Robert A. Charpie, Donald J. Dewar, André Finkelstein, John Gaunt, Jacob A. Goedkoop, Elwyn O. Hughes, Leonard F. Lamerton, Aleksandar Milojević, Clifford Mosbacher, César A. Sastre, and Brian E. Urquhart.

The verbatim records of the Conference are included in the pertinent volumes. These verbatim records contain the author's corrections and, where necessary for scientific accuracy, the editing changes of the Scientific Secretaries, who have also been responsible for inserting slides, diagrams and sketches at appropriate points. In the record of each session, slides are numbered in numerical order through all presentations. Where the slide duplicates an illustration in the submitted paper, appropriate reference is made and the illustration does not appear in the record of the session.

Volume 16, "The Record of the Conference," includes the complete programme of the Conference, a numerical index of papers and an author's index, the list of delegates, the records of the opening and closing sessions and the complete texts of the evening lectures.

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The Sodium Reactor Experiment

By W. E. Parkins,* USA

I. INTRODUCTION

The Sodium Reactor Experiment (SRE) is a reactor facility being constructed as part of the five-year program for the development of nuclear power. It is specifically intended for the exploration and improvement of technology associated with the sodium-cooled, graphite-moderated type of reactor. A more inclusive program of work is being carried out concurrently and will make use of the SRE as the major tool in developing this technology. While the SRE is primarily intended as an experimental facility, and not a true pilot plant, it incorporates many of the features believed to be desirable in a full-scale central station nuclear plant of this type.

The schedule being followed calls for completion of the SRE construction early in calendar year 1956. Various criticality and preoperational tests will follow and lead to full power operation at some later date. Costs are estimated to be approximately $3,500,000 for the reactor and its various auxiliaries, and approximately $500,000 for the building and facilities. Cost of development work required to prove out many of the components being incorporated in the SRE is separate. The location is approximately thirty airline miles northwest of downtown Los Angeles, in the Simi Hills. The site is situated at an elevation of 1850 feet and is four miles due south of the small town of Santa Susana, California.

The design and operation of the SRE are the responsibility of North American Aviation, Inc., under direct contract with the Atomic Energy Commission. This project, together with the experimental program for the development of sodium-graphite reactor technology, is being financed by the Atomic Energy Commission, with substantial support directly from North American Aviation, Inc. The design and location of the SRE have been approved by the Advisory Committee on Reactor Safeguards, and as of May, 1955, construction work is well under way (Fig. 1).

*North American Aviation, Inc., Nuclear Engineering and Manufacturing, Los Angeles, California. Work performed by W. E. Abbott, R. L. Ashley, R. O. Crosgrove, T. F. Edziak, D. T. Eggen, R. C. Gerber, W. J. Hallett, M. P. Heisler, E. Matlin, M. Mueller, H. E. Richter, W. Sanders, T. T. Shimazaki, S. Siegel, C. Starr, A. M. Stelle, C. A. Trilling, and other members of the staff of the Nuclear Engineering and Manufacturing Dept. at North American Aviation, Inc.

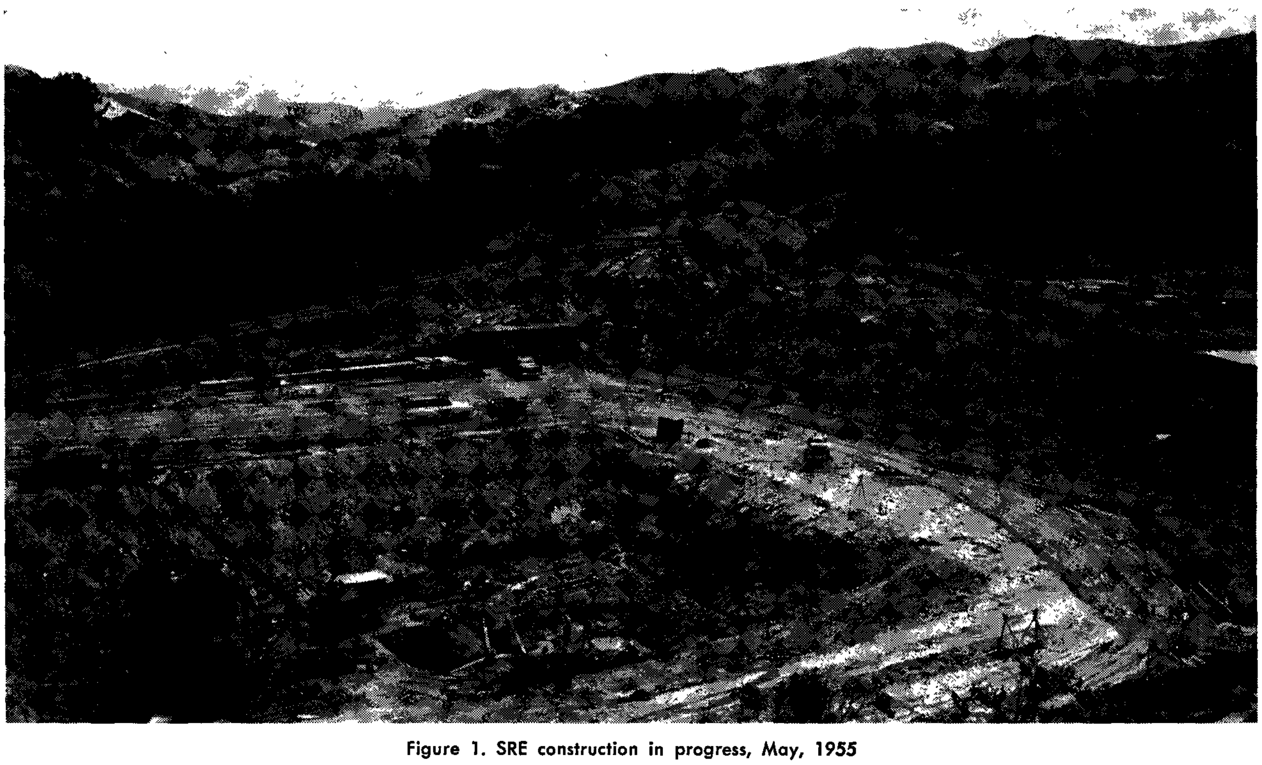


Figure 1. SRE construction in progress, May, 1955

II. GENERAL DESCRIPTION

The SRE is designed for the production of nominally 20,000 kw of heat. No provision is being made in the initial installation for a steam cycle and the production of electrical power. Steps are being taken, however, which will permit such an addition at a later date. The reactor is cooled by sodium, which circulates in a primary system and becomes radioactive. This primary sodium transfers its heat to a secondary, non-radioactive sodium system in intermediate heat exchangers. The secondary sodium rejects its heat to the atmosphere in air-cooled heat exchangers. Both the primary system and secondary system have two separate circulating loops; a main loop, capable of the transfer of 20,000 kw of heat, and an auxiliary loop capable of the transfer of 1000 kw of heat. It is intended that the auxiliary circuit, comprised of the auxiliary primary and auxiliary secondary loops, be operated simultaneously with the main circuit in order to assure heat removal capability in the event of some component failure in either circuit.

Views of the reactor building and the equipment arrangement are shown in Figs. 2 and 3. The reactor is located below grade, with the upper surface of its top shield at floor level in the reactor room. The two primary loops are also below floor level and are installed in separate concrete-walled galleries. Motors for the mechanical sodium pumps and for the control rod drives are located above floor level for easy maintenance. The secondary sodium lines extend from the intermediate heat exchangers to locations above ground level and outside the reactor building to where the air-cooled heat exchangers are located.

A 75-ton handling bridge is designed to move within the reactor room and be capable of supporting lead-shielded coffins used for the removal of radioactive elements from the reactor core. At one end of the reactor room special facilities are installed, again below floor level, for the cleaning and storing of these elements. For the purposes of the experimental program, especially that associated with the fuel development, a hot cell is installed below grade with access holes to receive elements from the handling coffin. Here the fuel elements will be disassembled, inspected, and certain measurements made.

In addition to the sodium coolant and graphite moderator, the SRE will initially use fuel elements fabricated from slightly enriched uranium, containing 2.80 atom per cent U^{235}. These elements, as well as all other elements penetrating the reactor core, are suspended from small plugs in the top shield. The general arrangement of the reactor is shown in Figs. 4, 5, and 6.

The entire core is contained in a stainless steel vessel, 19 feet deep and 11 feet in diameter. The graphite moderator is supported and located on a stainless steel grid plate near the bottom of this core tank. The graphite is in the form of cell-sized hexagonal prisms, placed on a triangular lattice 11 inches between centers. Each prism is 10 feet in height and is clad with thin zirconium sheet. The 10-foot height includes 6 feet for the moderator and an additional 2 feet at the bottom and at the top for reflector. The graphite assemblies making up the core region contain an axial zirconium tube, in which fuel elements are suspended.

Inlet lines from the main primary and auxiliary primary loops enter the core tank above the graphite assemblies and extend vertically downward in double-walled pipes, in order to discharge into a plenum between the bottom of the core tank and the grid plate. This sodium at a temperature of 500°F then passes up

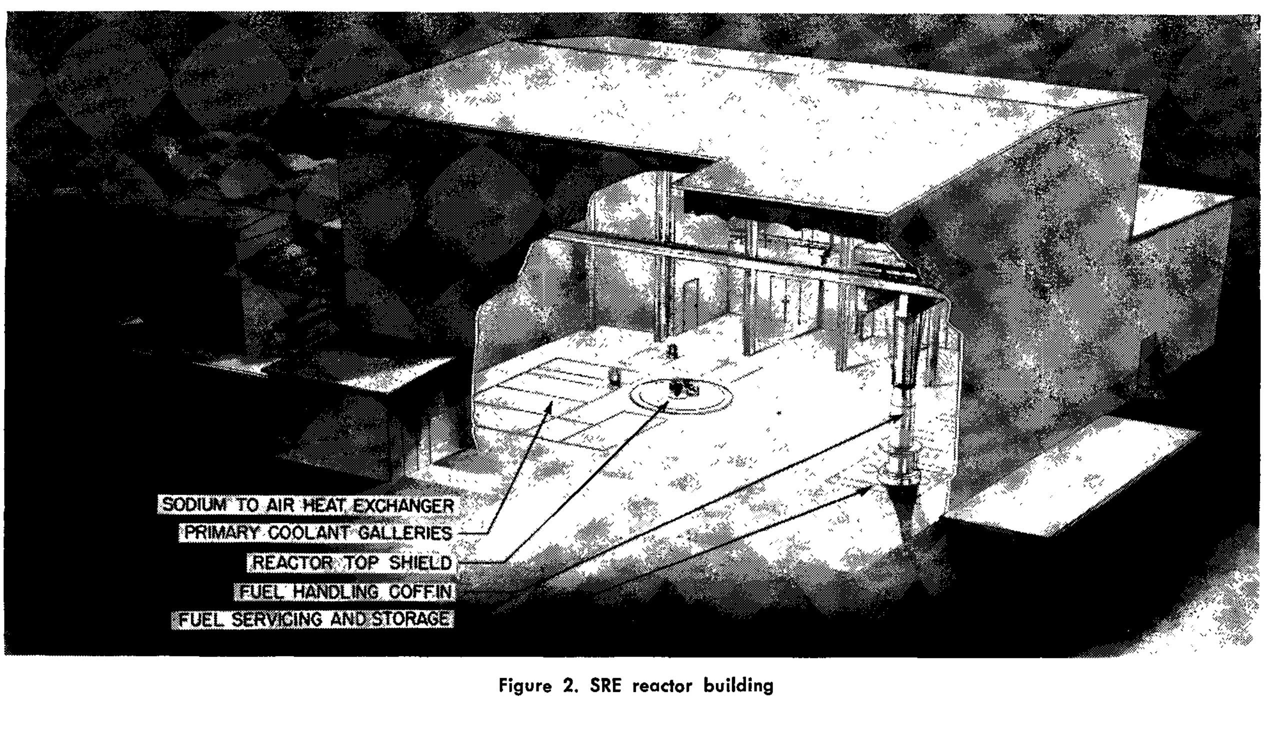


Figure 2. SRE reactor building

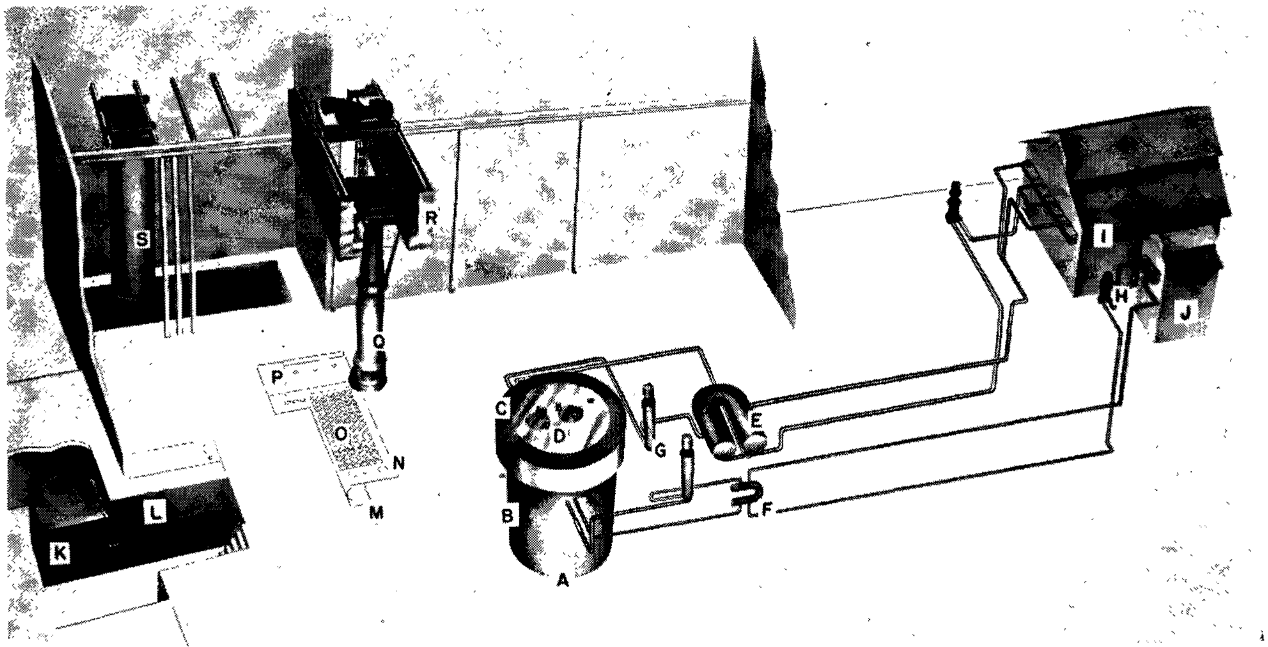


Figure 3. General arrangement of components. A: Reactor; B: Core tank; C: Ring shield; D: Rotatable face shield; E: Main sodium heat exchanger; F: Auxiliary sodium heat exchanger; G: Pump; H: Pump; I: Main sodium-air heat exchanger; J: Auxiliary sodium-air heat exchanger; K: Metallurgical hot cell; L: Primary hot cell; M: Shipping coffin cells; N: Moderator can storage cell; O: Fuel storage cell; P: Fuel cleaning cell; Q: Fuel handling coffin; R: Handling bridge; S: Moderator handling coffin

through the axial tubes, cooling the fuel elements and discharging into a pool 6 feet deep, having a mean temperature of 960°F. Separate outlet pipes for the two primary loops are also located in the core tank above the graphite assemblies.

Surrounding the core tank is a steel thermal shield 5½ inches thick. Immediately outside of the thermal shield is the outer tank intended as an emergency means of containing sodium in the event that a leak should develop in the core tank. The outer tank is surrounded by approximately a foot of thermal insulation. This, in turn, is contained in a tank called the cavity liner. The function of the liner is to serve as a form for the concrete foundation and to aid in the removal of heat from this region. Steel pipe is tack-welded to the outside surface of the cavity liner to provide a means of circulating a fluid for removing the small amount of heat developed in the concrete together with that conducted through the layer of thermal insulation. The fluid circulated is toluene, furnished from a special cooling system intended for many components around the reactor where there might be some conceivable possibility of contact between this fluid and the sodium coolant.

The concrete foundation extends up to the biological shield at floor level, where it is planned to use a special grade of concrete made with magnetite iron ore aggregate. As a closure for the reactor vessel there is a ring-shaped shield supported on a ledge in the cavity liner, and a circular (or rotatable) shield supported on steps on the inside of the ring shield. All of the small plugs permitting access for the core components are located within this rotatable top shield. Large diameter bellows provide a gas seal for the core tank atmosphere and for the atmosphere between the core tank and the outer tank. Separate bellows and diaphragms seal the galleries from the atmosphere between the outer tank and the cavity liner, while permitting thermal expansions of the various pipes penetrating this region.

III. FUEL ELEMENTS

The fuel elements planned for initial operation in the SRE are fabricated in the form of clusters of seven rods, as shown in Fig. 7. Each rod consists of a 6-foot high column of 6-inch uranium slugs in a thin-walled stainless steel jacket tube, thermally bonded by NaK alloy. The slugs are 0.750 inch in diameter, the jacket tube 0.010 inch in wall thickness, and the NaK annulus 0.010 inch in average thickness. The stainless steel jacket material is type 304 and is closed at each end by a welded stainless steel plug. The NaK bonding alloy extends a few inches above the slug column to a free surface above which is helium gas at atmospheric pressure (when at room temperature). This space serves as a volume for the thermal expansion of the NaK and for the accumulation of possible fission gases during irradiation. While sodium might be used as the bonding agent, some difficulty has been experienced in assuring a bond with no voids, since the sodium is a solid subject to fracture at room temperature and since it contracts appreciably upon solidification.

The seven rods forming each cluster are retained at their ends. Support is provided by the fixture at the top, while individual expansion of the rods is permitted by the fixture at the bottom end. Also at the bottom of the assembly is a locating guide and an

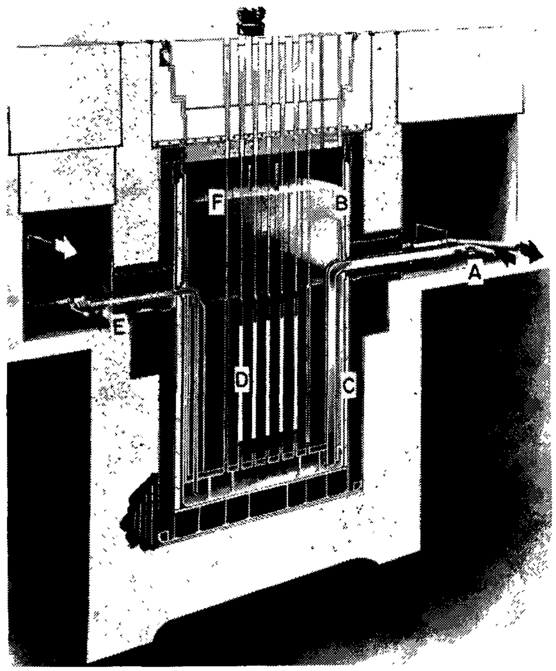


Figure 4. SRE reactor section. A: Main sodium inlet; B: Core tank; C: Thermal shield; D: Fuel element; E: Auxiliary sodium inlet line; F: Sodium level

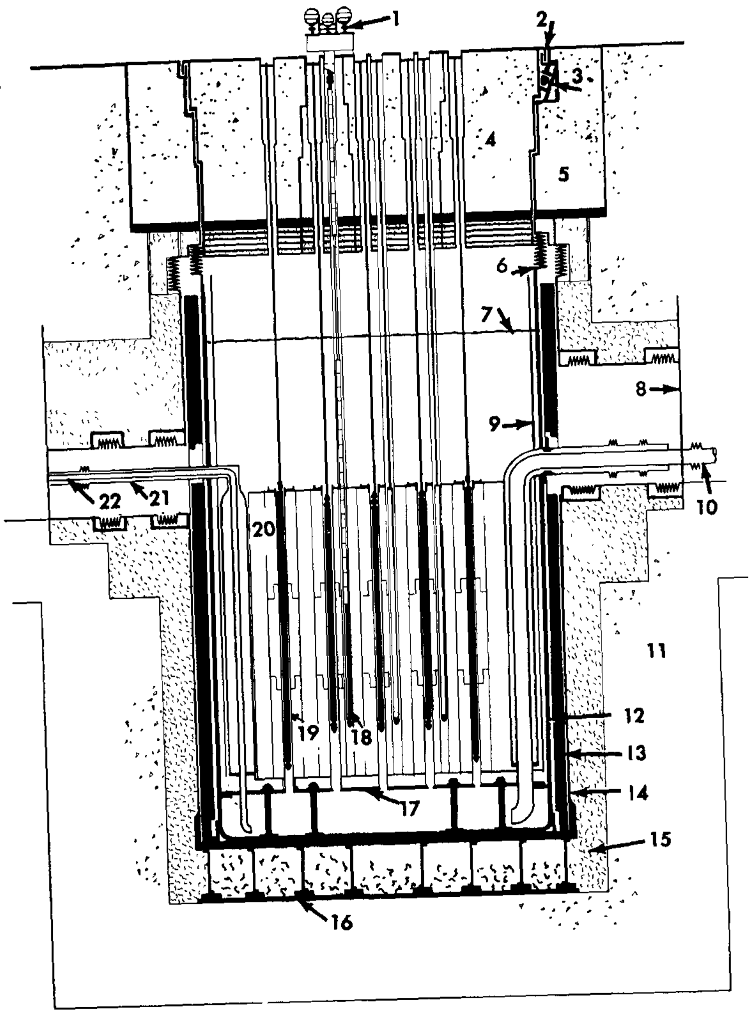
orifice plate for controlling the flow of the sodium, as required for each particular coolant tube. In order to prevent the rods from touching each other or the coolant tube, the six outside rods forming the cluster are spirally wrapped with an 0.092 inch diameter stainless steel wire with a pitch of approximately 10 inches. The direction of wrap is arranged such that adjacent rods are wrapped in counter directions. In order to determine the temperature of the exit sodium from any coolant tube, a thermocouple is inserted down the inside of the stainless steel tube supporting the entire cluster from its plug in the top shield. The junction of this thermocouple is located just above the fixture at the top of the fuel element cluster.

It is planned to operate the SRE so that a maximum uranium metal temperature of approximately 1200°F is attained in each fuel element. This temperature will be determined by the total power level of the reactor, the flow and temperature conditions in the sodium system, and by the selection of orifice plates for the fuel elements. During normal operation there will be a pressure drop of 2.5 psi across the central fuel element and 1.5 psi across its orifice plate, making a total of approximately 4 psi. The flow in this central tube will be 5 feet per second, corresponding to 17,500 pounds per hour of sodium. Flow in a tube requiring the least cooling (near the outside where the average thermal neutron flux is least) will be 3 feet per second. Nominal sodium outlet temperatures, with the 500°F inlet, are 908°F for the central tube and a maximum of 986°F for a tube in the lowest average neutron flux, giving a mixed mean in the sodium pool of 960°F. The maximum heat flux anticipated is about 340,000 BTU/hr-ft^2, which will be accompanied by a heat transfer coefficient of approximately 10,000 BTU/hr-ft^2-°F. Approximate values of the peak-to-average heat flux are 1.35 for any channel and 1.63 for the entire reactor. Under these conditions, the central fuel element will produce 650 kw, corresponding to 340 kw per kg of U^{235}.

To achieve the conditions just described, it is necessary that the flow through the coolant tube and the interstices of the fuel element be such that there is a considerable degree of mixing of the sodium as it moves through the inner passages just surrounding the central rod and through the outer passages surrounding the ring of six rods. Otherwise, the temperature would increase more rapidly in the inner passages, since the total heat flux to sodium cross section is greater there than in the outer passages. The effectiveness of this mixing has been measured by experiments circulating water past a fuel element mock-up and determining the concentration at various locations of a solution of manganous chloride, which was injected into the inner passages near the upstream end of the element. These experiments showed the need for wrapping the wire in counter directions on adjacent rods so that the fluid is most effectively forced outward from an inner passage and inward from an outer passage in alternate openings around the ring of six rods. The pitch of the rotation of the fluid as determined by these chemical experiments is slightly greater than the pitch of the wire wrap, but is sufficiently short to guarantee adequate sodium mixing in the coolant tube during SRE operation. These hydraulic experiments with the fuel element mock-up were also useful in establishing the pressure drops and vibrational conditions. No difficulty is expected from rod vibration even though there is no provision for securing any cluster of rods along its length.

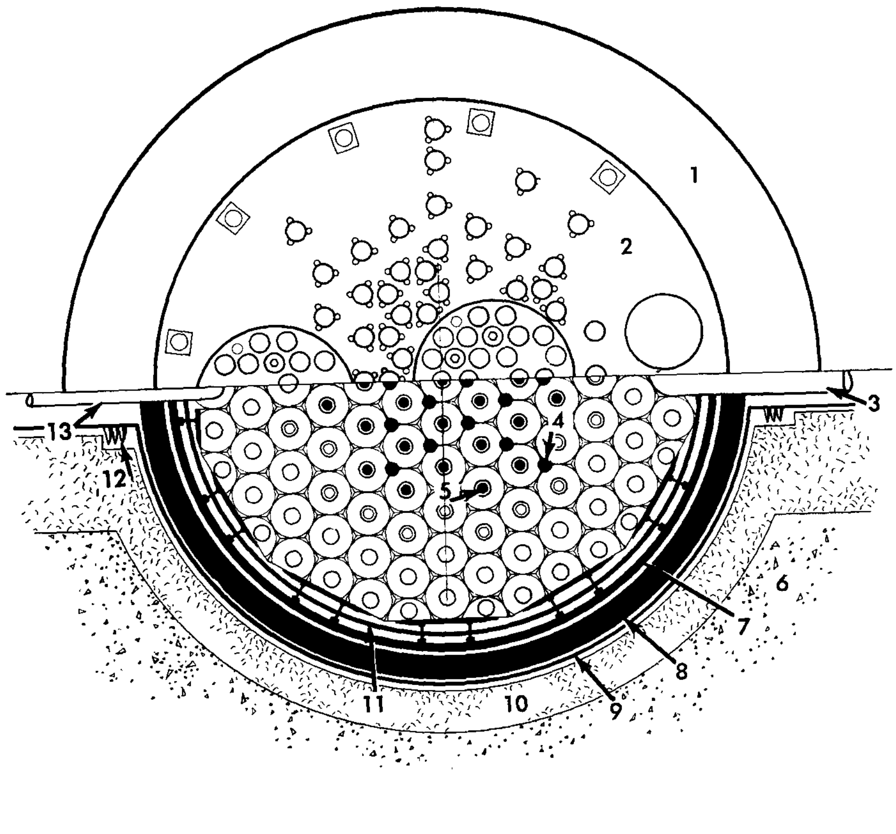
It is estimated that sufficient reactivity for operation will be obtained with a loading of thirty-one fuel clusters of the type described, and employing the 2.8 atom per cent U^{235} enriched uranium. This will provide a core region 6 feet high and approximately 6 feet in diameter, wherein the neutron flux has been calculated to be that shown in Fig. 8. The average thermal neutron flux at the center of the core is estimated to be 2.5×10^{13} neutrons/cm^2-sec. The peak thermal flux in the fuel is estimated to be 1.7×10^{13} neutrons/cm^2-sec. Some of the calculated nuclear parameters for the SRE at design operating conditions are as follows: $\epsilon = 1.034$; $p = 0.799$; $f_{25} = 0.745$; $k_\infty = 1.28$; $L^2 = 137$ cm^2; $\tau = 370$ cm^2; and $B^2 = 4.84 \times 10^{-4}$ cm^{-2}.

With the design conditions of sodium cooling and neutron flux which have been described, the reactor power level with thirty-one fuel elements will be slightly less than the nominal value of 20,000 kw. To achieve this power with design conditions, a loading more nearly equal to thirty-seven fuel elements is necessary. The actual number of fuel elements used will depend upon the results of criticality measure-



1. Control element drive
2. Seal
3. Roller
4. Rotatable shield
5. Ring shield
6. Bellows
7. Sodium level
8. Diaphragm
9. Inner liner
10. Main sodium inlet line
11. Biological shield
12. Core tank
13. Thermal shield
14. Outer tank
15. Thermal insulation
16. Bearing plate
17. Grid plate
18. Control element
19. Fuel element
20. Moderator element
21. Guard pipe
22. Auxiliary sodium inlet line

Figure 5. Reactor elevation



1. Ring shield
2. Rotatable shield
3. Main sodium inlet pipe
4. Control element
5. Fuel element
6. Side biological shield
7. Core tank
8. Thermal shield
9. Outer tank
10. Thermal insulation
11. Inner liner
12. Bellows
13. Auxiliary sodium inlet line

Figure 6. Reactor plan

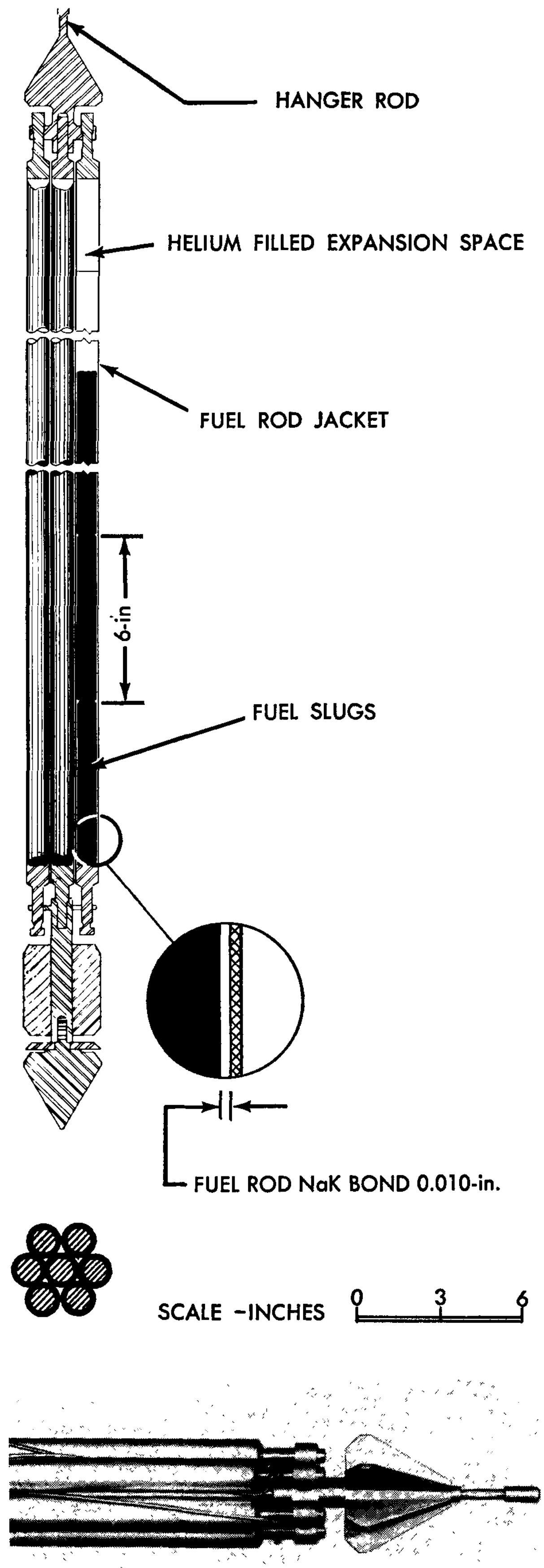


Figure 7. Seven-rod element

ments to be made on the assembled core and upon the requirements of the experimental program.

IV. MODERATOR AND REFLECTOR

The principal problem on the graphite moderator and reflector is that of preventing contact with the sodium coolant. Such contact results in the absorption of sodium into the pores of the graphite, causing excessive neutron losses. Since the amount of metal required for the cladding of the graphite in the SRE design configuration is large, it is impractical to employ stainless steel for this purpose even though stainless steel has been chosen as the fuel jacket material. The graphite is protected by zirconium sheet fabricated into individual can assemblies, shown in Fig. 9. As designed, 0.035-inch thick sheet will be used on the side panels of each hexagonal column of graphite, and 0.10-inch thick stock will be used for the bottom and top can heads. The distance across the flats of each can assembly is slightly less than the 11-inch center-to-center spacing of the triangular lattice. This reduction from 11 inches is sufficient to provide for an average gap between cans of approximately 0.170 inch during normal operation. Such a gap in the form of a thin, flat channel is necessary to permit some heat removal at the can wall by the sodium coolant. Sodium for this purpose enters the core tank in a separate pipe, branching from the main inlet sodium stream and accounting for a total flow of approximately 7 per cent of that in the main stream. This branch line discharges its sodium into a low pressure plenum above the grid, but below the graphite assemblies. This sodium is then free to seep upward through the passages between cans and through special channels provided for core elements other than the fuel elements. It removes heat from these core elements, from the cans, and from the tanks and shielding at the sides of the reactor. Average flow is to be adjusted so that this sodium exits at approximately 960°F.

Each graphite assembly is bolted by means of zirconium studs to a supporting pedestal at the base of the can and to a spacer plate at the top. Both the pedestal and spacer plate are fabricated from type 405 stainless steel. The pedestal at the bottom serves not only as a support for the can, but locates it laterally by fitting into a hole provided in the grid plate. The seal at the grid plate is sufficiently tight to prevent sodium leakage into the plenum above the grid. In addition, the pedestal has a circular channel along its axis which directs the sodium from the main plenum up through the coolant tube in the moderator can assembly. The spacer plate at the top serves as a lifting fixture for the entire assembly and as a means of lateral support. The spacer plates from adjacent cans nest together and are maintained in place by a clamping band around the outside. If it should be necessary at some later date to replace a graphite assembly, this arrangement makes it possible to remove the particular unit by lifting vertically without the complication of unfastening a mechanical connection.

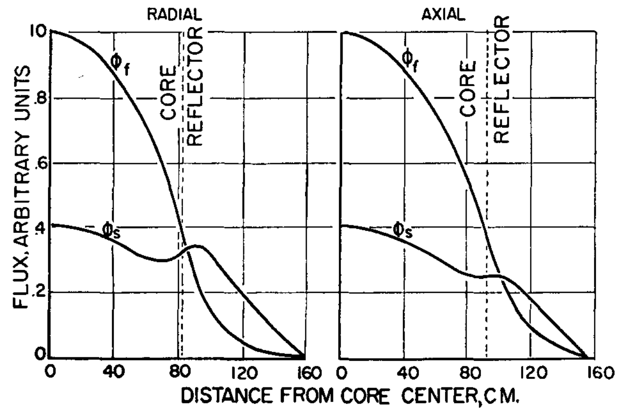


Figure 8. Calculated average neutron flux in SRE core

Another feature, which prevents the moderator assemblies from ever being appreciably displaced in the vertical direction during operation is a tube which extends down from the top shield around each fuel element hanger rod. The lower end of this tube would bear against the spacer plate if the moderator assembly were dislodged.

The zirconium canned graphite assemblies which make up the side reflector do not have axial channels. Each assembly in the core region is penetrated along its axis by a zirconium tube 0.035 inch thick and 2.80 inches in inner diameter, welded to both the bottom and top heads. Such a tube would usually contain a fuel element. At the time of reactor scram, the sodium temperature in the coolant tube would suddenly decrease, causing a contraction relative to the outside walls of the can and producing a deflection in both the bottom and top heads. The stresses resulting here and those from other static and transient operating conditions have been calculated to be safe from the known mechanical properties of zirconium.

One of the problems in canning the graphite is the question of gas pressure which might be produced as a result of long-time operation at elevated temperature in the radiation field. With the can design just described, the side panels are of thin material and sufficiently flexible so that it is only safe to operate with some external pressure collapsing these panels against the graphite. At the time of assembly, the top head is spaced very close to the top of the graphite column, but during operation as the temperature increases, the zirconium expands away from the graphite, necessarily leaving the head unsupported. This makes possible the flexibility required for the scram condition just mentioned, but at the temperatures involved makes it imperative that only a small pressure difference exists between the sodium on the outside of the can and the gas atmosphere on the inside. Since the outgassing properties of the graphite during operation are not fully known and since it is extremely difficult to predict the average temperature of the graphite which will determine the pressure buildup inside any sealed can assembly, provision is being made to control the pressure on the inside by means of a vent tube extending from the bottom of the assembly out through the top head and to the helium atmosphere above the free surface of sodium in the core tank pool. This vent tube is ¼ inch in outside diameter and is protected by a vertical stainless steel guard tube attached to the spacer plate. At the bottom of the vent tube and resting on the bottom can head is a small cup to accept any sodium which enters during the lifetime of the unit, as a result of diffusion of sodium vapor and pressure fluctuations. The diffusion of sodium by this means has been calculated and measured experimentally to be acceptably small. The vent tube arrangement, however, is a mechanical complication which it is hoped may be eliminated in future designs.

The graphite is machined from large logs approximately 13 inches in diameter, but each cell is vertically assembled in three pieces with plug connections. There are no special strength requirements on the graphite except insofar as it must support its own weight and resist lateral earthquake accelerations while supported at the top and bottom ends. Thermal stresses and radiation damage effects have been calculated from available information to be well within safe limits. To provide extra channels for the insertion of control elements, safety elements, and other special units, a number of can assemblies are shaped at the corners to provide circular passages extending down to the grid plate. Each circular channel is formed by 120 degree shaped corners in three adjacent assemblies, making a passage approximately 3¼ inches in diameter.

The amount of heat which flows to the outer walls of the can assemblies varies with position within the reactor. No special provision is necessary to orifice the sodium flow in the channels between cans according to the heat generation rate. If the temperature of the sodium in one channel is increasing in temperature more rapidly than the average in the other channels, the resultant decrease in density automatically increases the flow rate in that particular channel. As a result, this buoyancy effect tends to assure nearly uniform temperature of the sodium throughout any horizontal plane taken through the reactor. This feature is also important in the region outside of the side reflector where considerable heat is received from the core tank and the thermal shield. Calculations show it is necessary for the flat channels between cans to be wider than some amount of the order of 1/16 inch in order to assure sufficiently uniformized temperatures. While these gaps will be maintained by the action of the pedestals and spacer plates, an additional precaution has been taken wherein dimples 0.045 inch in height are rolled into the zirconium panels before can fabrication. During operation, these dimples will prevent the closing off of any channel, as might be caused by misalignments.

V. CONTROL ELEMENTS

There are four control elements in the SRE core, located as shown in Fig. 10. Calculations indicate that this arrangement will permit a total of about 10 per

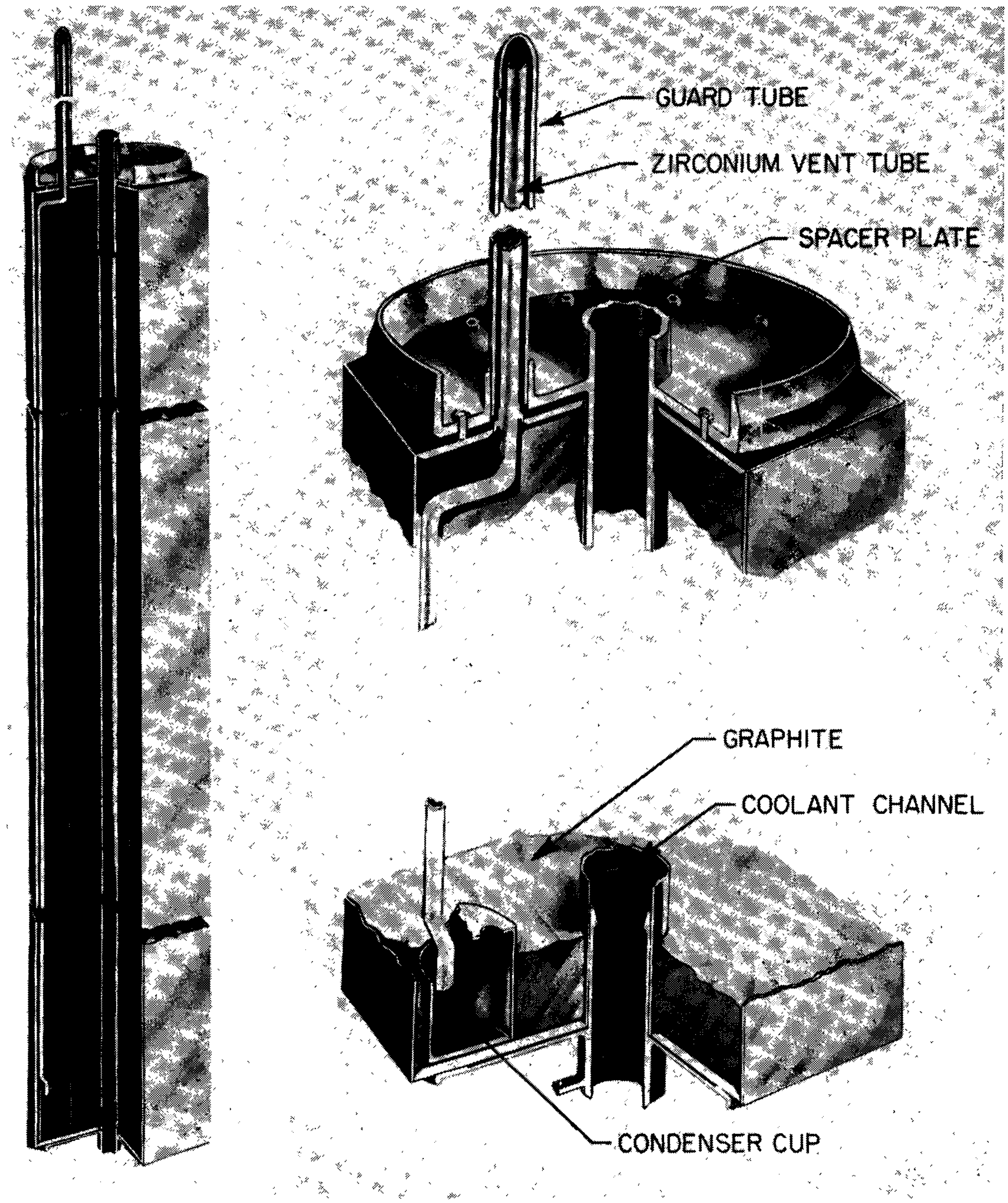


Figure 9. Zirconium-canned graphite moderator assembly

cent control in reactivity. Each control element is contained in a thimble assembly which extends from the top of the rotatable shield to a point just below the core, a total distance of 23½ feet. The thimble material is type 304 stainless steel and in the core region it has a wall thickness of 0.049 inch. This arrangement makes possible a self-contained control rod assembly in which no sodium or its vapor contact the moving mechanical parts. See Fig. 11.

The poison column is made up of a series of eighteen rings of a boron-nickel alloy suspended on a "pull-tube." Each ring is 2½ inches in outside diameter, ⅜ inch in annular thickness, and 4 inches long. The boron concentration in the alloy is approximately 2 weight per cent. Control rod motion is obtained by a ball nut screw arrangement wherein the pull-tube is attached to the nut and a motorized drive mechanism above the top shield turns the screw. The nut is prevented from rotating by guides which move in flutes machined into the inside surface of the heavy wall of the upper portion of the thimble. Suitable shielding is provided within the thimble in the neighborhood of the top shield. Two of the control rods have single motor drives to produce 0.24 foot per minute motion of the control rods for shim action only. The other two rods have dual speed drives for this same shim motion and for a regulating motion at a rate of 3 feet per minute. The latter is mechanically limited to a

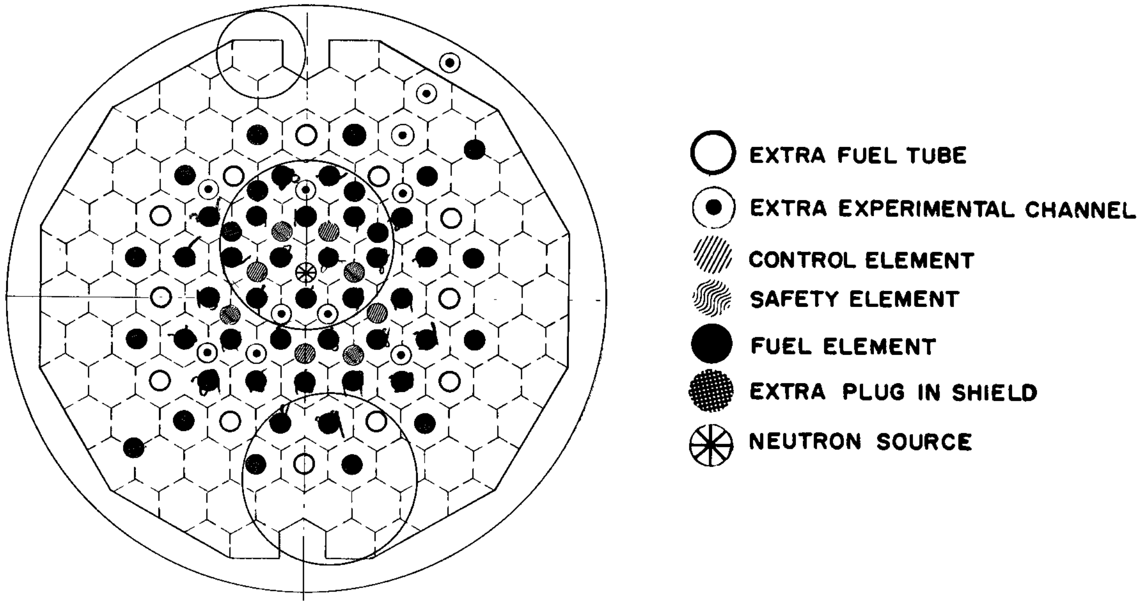


Figure 10. Loading face pattern

range of ±6 inches of travel. The motor drive units rest on the top shield and engage at a clutch several inches below the top surface of the shield. Limit switches and Selsyn motors indicate the control rod positions at all times.

One of the problems with the thimble arrangement is that of transferring the heat generated in the poison material to the sodium coolant on the outside of the thimble. To prevent excessive temperatures in the boron-nickel, helium gas at about 16 psig is used as an atmosphere within the thimble and relatively close clearances are maintained between the rings and the thimble. Experiments have been conducted to establish these conditions. With the maximum heat generation rate anticipated at 8 kw per foot, a maximum temperature of about 1300°F is expected in the boron-nickel. This is for an eccentric condition wherein the ring touches the thimble and the helium annulus has an average thickness of about 0.020 inch. In spite of variations in control rod position and in heat generation rates, the sodium flow on the outside of the thimble will, to a very large extent, automatically adjust itself as a result of density changes with temperature, so that the sodium temperature will increase at the same rate as in the other channels and exit into the pool at about 960°F.

VI. SAFETY ELEMENTS

The SRE core also uses four safety elements, located as shown in Fig. 10. The total reactivity control available in these elements is slightly less than 10 per cent. The design is similar to that of the control elements in that the entire assembly is contained in a thimble extending from the top of the rotatable shield to a location below the core. The thimble material is type 304 stainless steel and in the core region has a thickness of 0.035 inch. The safety element is illustrated in Fig. 12.

Rings of the same boron-nickel alloy as used in the control elements make up the poison column. Each ring is $2^5/_{16}$ inches in outside diameter, $^3/_{16}$ inch in annular thickness and 4 inches long. A total of 14 rings is used, assembled in series, on an internal tube. This tube may be raised by the action of a ball nut screw, but a latch mechanism is incorporated to release the rod at any time while it is being withdrawn or when it is in the fully cocked position. The latch is actuated by a "torque tube" flattened on two sides and extending down the center of the assembly from an actuating solenoid near the top to a position near the bottom of the thimble. The ball screw is driven by a motor located above the top shield as in the case of the control element. To reset the safety rod, the ball nut is driven downward by the action of the screw until engaged to the rod unit by the latch. The direction of the motor is then reversed for withdrawal. Shielding material is incorporated within the thimble in the region of the top shield. There are also electrical contacts arranged to determine the position of the rod as inserted, fully cocked or being withdrawn.

When the safety rods are released to scram the reactor, each latch is disengaged and the rod units fall freely within the thimble under the influence of gravity. In order to decelerate without the production of excessive stresses, the upper end of the tube on which the poison rings are assembled is attached to a piston within a 24-inch long cylinder. As the rod unit falls, the piston and cylinder move with the rod until it reaches a position 24 inches from the bottom of the thimble at which time the cylinder is arrested by a shoulder and the rod unit is then decelerated as the piston is forced to move through a helium atmosphere contained in the cylinder.

In the case of the safety elements, there is not the same problem of heat transfer that exists with the control elements since the introduction of the safety

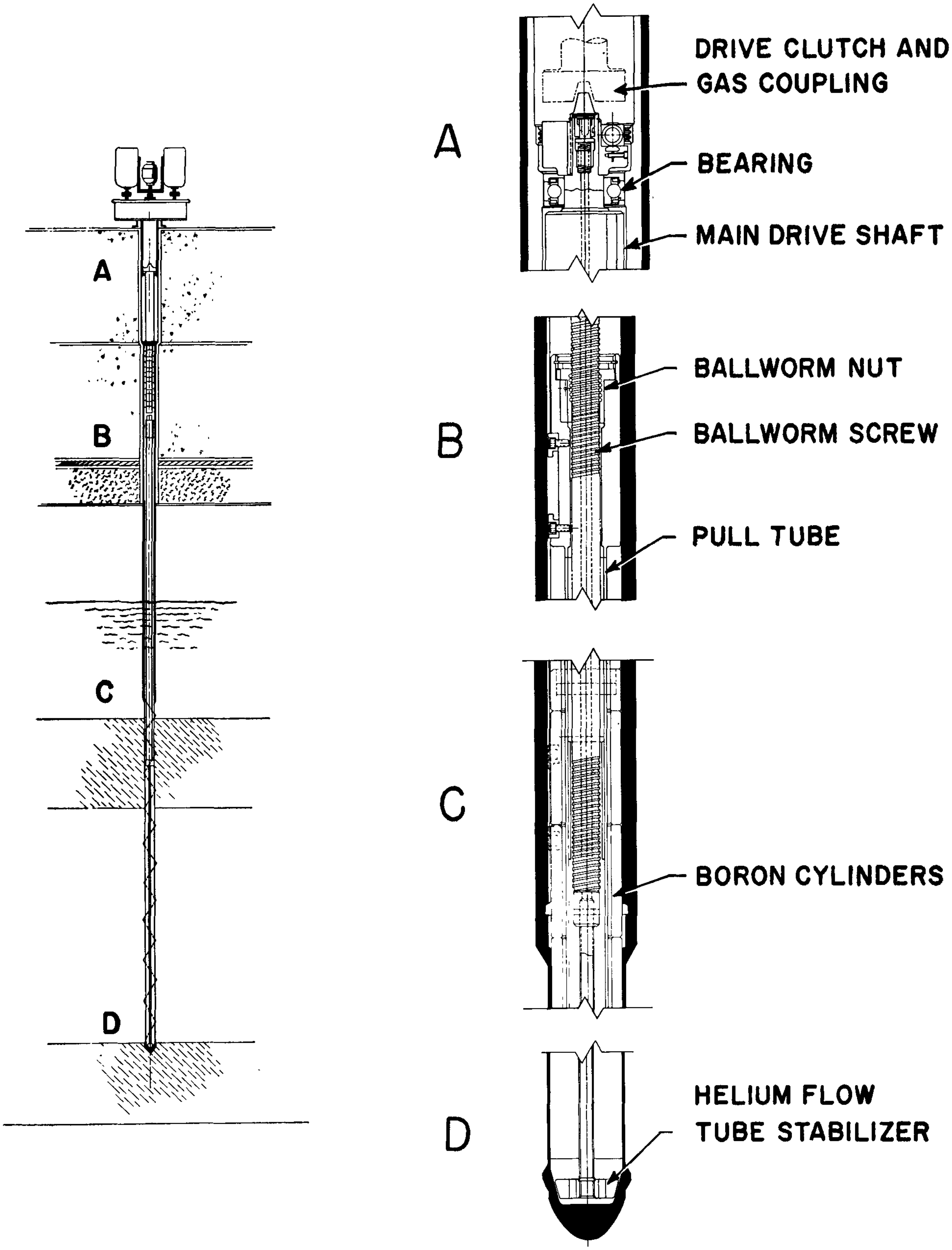


Figure 11. Control element

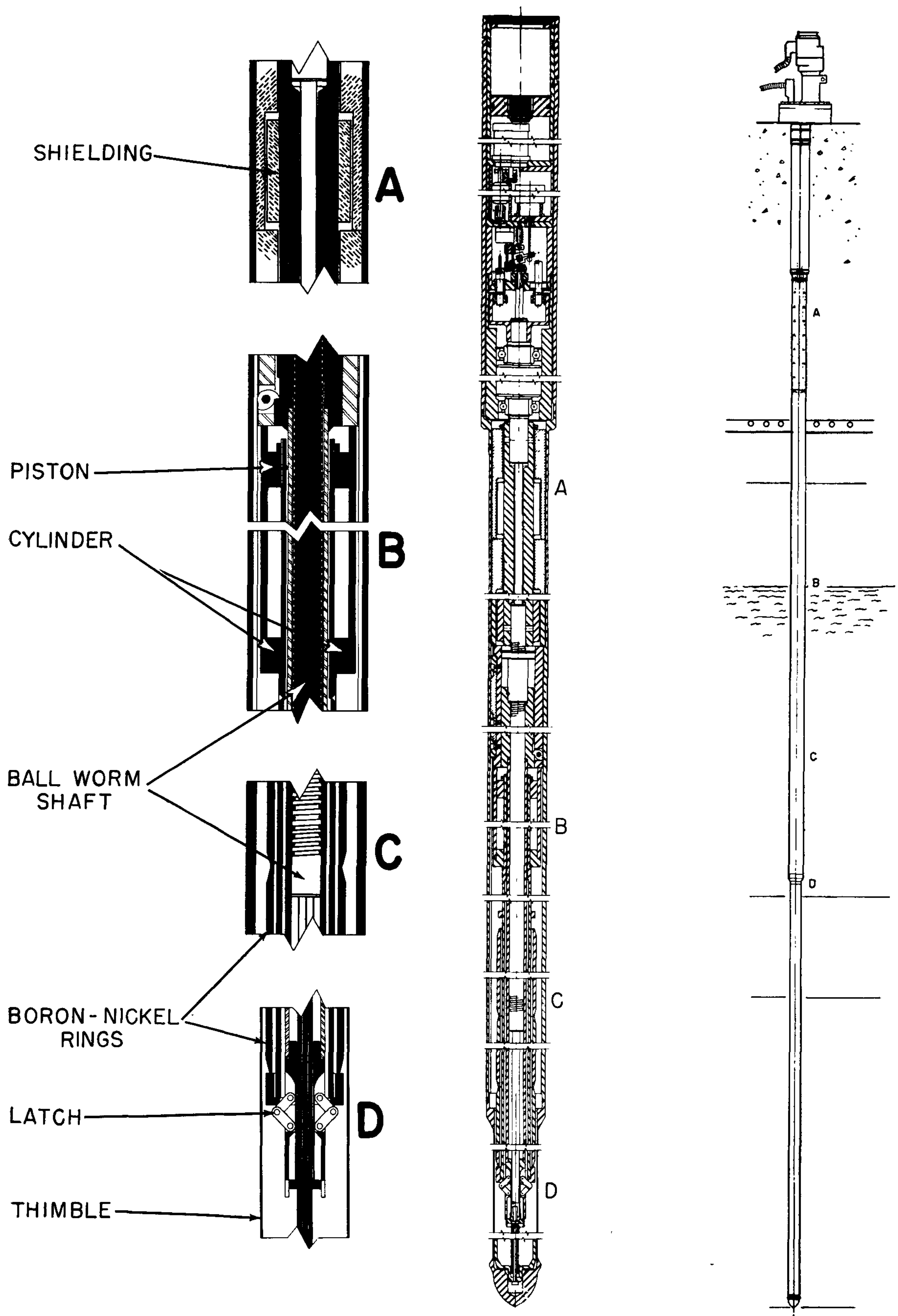


Figure 12. Safety element

rods into the core always results in shutting down the reactor. A much larger annulus, an average of 0.125 inch, exists between the boron-nickel rings and the stainless steel thimble. This has the added advantage of decreasing the possibility of binding due to some misalignment at the time the rods are dropped. An atmosphere of helium at about 16 psig is used within the safety element thimble.

VII. OTHER CORE ELEMENTS

The SRE core is designed so that there are a total of 43 axial tubes in the moderator assemblies capable of receiving fuel elements. There are also 17 channels at the corners of the moderator assemblies capable of receiving control or safety elements or other elements as desired. It is planned to load 31 fuel elements for the necessary criticality, although additional units may be loaded into the remaining 12 fuel tubes if required. This permits a certain degree of flexibility but necessitates the use of "dummy elements" for those tubes not containing fuel elements. The dummy elements serve to replace coolant with moderating material and to prevent the flow of unheated sodium up into the pool above the core. In the same way, those corner channels which do not contain control or safety elements, nine channels as planned, are to be filled with dummy elements or special purpose elements of some kind.

The dummy elements for the fuel tubes consist of graphite cylinders clad in thin zirconium sheet and suspended on hanger rods from plugs in the top shield much the same as the fuel elements themselves. The zirconium can for the dummy elements is 2½ inches in outside diameter, and extends nearly the full height of the core plus top and bottom reflectors. A vent tube is attached which extends up along the hanger rod to a location in the helium atmosphere above the sodium surface. One special feature required for these elements is a weighted plug which hangs from a flexible joint at the bottom of the zirconium can and rests on the upper lip of the pedestal used for support of the moderator assembly. Except for that sodium passing through a small orifice hole which is drilled up the length of the plug, the flow from the main inlet plenum is effectively stopped. The plug assembly is fabricated from type 304 stainless steel and is of sufficient length to seal against a pressure of 8 psi in the lower plenum without the requirement of any other mechanical connection.

The dummy elements for the corner channels are constructed similarly but require no plug at the bottom since the corner channels only open to the low pressure plenum above the grid, which receives sodium for cooling the moderator. Sodium flow in these channels will adjust according to the heating rate so the exit temperatures into the pool are approximately the same as from all other channels which open to this plenum.

One special type element to be used in a corner channel is the neutron source. As designed, this unit consists of a column of beryllium cylinders suspended on a ⅝ inch outside diameter tube extending down from a plug in the top shield. The beryllium cylinders will have the same diameter as the dummy elements and will extend vertically the full 6 feet of the core. A small cylinder of antimony ¼ inch in diameter, 1 inch high and enclosed in a protective graphite capsule, is suspended in a steel tube inserted down the inside of the tube which supports the beryllium cylinders. The graphite capsule with the antimony is sealed in the end of the inner tube and is located at the center of the core. This antimony will be activated before installation in the SRE by exposure in another reactor. With an expected activity of 18 curies, the antimony will produce approximately 5×10^7 neutrons per second by means of the (γ, n) reaction on beryllium. This arrangement permits replacement of the antimony if required, although it should be continually activated during operation of the SRE. While less beryllium could be used without appreciably decreasing the neutron production rate, the design just described has considerable advantage in simplicity.

VIII. REACTOR INSTRUMENTATION

In order to determine the temperatures in the sodium stream cooling the moderator, two of the corner channels will receive special elements containing thermocouples with their junctions at various elevations within the core. These elements are designed very similarly to the neutron source element just described. Each temperature measuring element consists of a column of beryllium cylinders suspended on a stainless steel tube extending down from a plug in the top shield. All of the thermocouples are contained on the inside of this supporting tube and do not contact the liquid sodium.

Aside from thermocouples in the special elements just mentioned, and those within the hanger rods of the fuel elements, no means is provided for measuring temperatures within the core tank. A number of thermocouples are installed to measure temperatures at various locations on the thermal shield, the outer tank, the foundation concrete, and the top shield.

There are some eighteen small plugs in the top shield which are not over corresponding fuel tubes or corner channels within the core. Two of these will be used for sodium level indicators which dip into the pool above the core. The principle of these indicators is a tubular type heater unit to which stainless steel sheathed thermocouples are attached at various heights. The sodium level can be approximately indicated by the recorded temperatures when this unit is being electrically heated.

To determine the neutron flux level, there are eight detecting chambers which can be lowered to positions at the cavity liner in the horizontal mid-plane of the reactor. Figure 13 shows the position of the tubes which permits the insertion of the chambers from wells at floor level in the reactor room. There are four tubes, capped at their lower ends, on each side of the

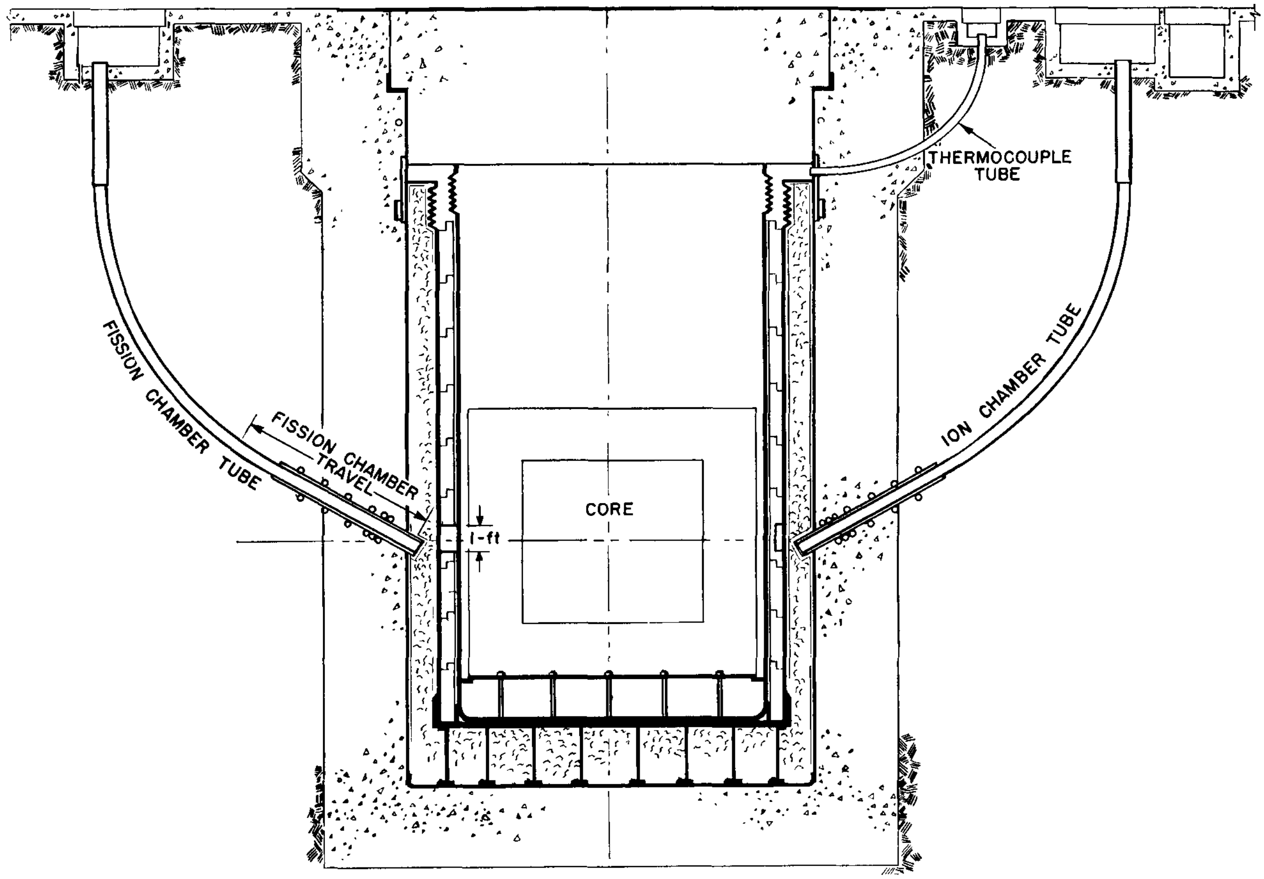


Figure 13. Location of fission and ionization chambers

reactor. They permit access for a total of two fission chambers and six ionization chambers. Before reactor start-up, the neutron source at the center of the core provides sufficient neutrons to obtain an indication from the fission chambers. The operator will use these chambers in coming up to a power of about 3 kw. Three of the ionization chambers, which are gamma compensated, are used for establishing flux level and period in the range from 3 kw to full power. The other three ionization chambers are uncompensated and are used for neutron level in the range from 30 kw to full power. One of these uncompensated ionization chambers provides a signal which automatically regulates a control rod to hold neutron flux level constant. Other chambers are connected to circuits which automatically release the safety rods in the event of excessively high neutron level or excessively short neutron period. In the power range above that intended for use of the fission chambers, these units will be raised from their locations near the cavity liner to positions where the neutron flux is sufficiently low to avoid damage. The tubes which receive the fission and ionization chambers are cooled at their lower extremities by attached piping through which the toluene coolant is circulated. This arrangement avoids the problem of operating the detecting chambers at the temperatures of the primary sodium.

IX. CORE TANK

The large tank which contains the SRE core is 1½ inch in wall thickness and is fabricated from type 304 stainless steel. The tank has no openings below the top of the graphite assemblies. Its discharge line for the auxiliary loop is located just above the graphite level and the auxiliary inlet is about 1 foot above the graphite. A rupture in this loop in the primary auxiliary gallery could not lower the sodium level in the reactor tank below the main discharge line. Openings for the main inlet and discharge, the moderator coolant inlet, the tank drain line, and the gas vent pipe for the core tank are all located approximately 1 foot above the top of the graphite. Ruptures in any of these lines would leave the graphite assemblies covered with sodium and the auxiliary primary loop operative. The inlet pipes for the main and auxiliary loops and for the moderator coolant are of a double wall construction to provide thermal insulation between the inlet pipes and the sodium pool. Helium gas is used to fill the annular space between the pipes.

At the top of the core tank is the large bellows to permit vertical expansion. This bellows is approximately 18 inches high and is fabricated from type 321 stainless steel, 0.10 inch in thickness. A welded flange near the bottom of the core tank supports the 1¼ inch thick type 304 stainless steel grid plate. Additional support for the grid plate is provided by means of staybolts arranged in three circular patterns. The inner circle of bolts is welded to the bottom of the core tank to guarantee sufficient strength against deflection of the grid plate in the event of a pressure surge in the main plenum.

To minimize thermal stresses due to any possible mismatch in temperature of the sodium flowing upward to the isothermal pool, an inner liner has been added. This liner is fabricated from type 304 stainless steel, is ¼ inch in wall thickness, and assures a stagnant layer of sodium 2½ inches thick adjacent to the inner surface of the core tank. A flange welded to both the core tank and liner at a location just above the graphite assemblies provides support. This flange has small weep holes for sodium drainage. The liner extends from 24 inches above the grid plate to the top of the core tank.

To center and support the graphite assemblies in the horizontal plane, a ¼-inch thick band 6 inches high is located at the inner surface of the liner at the elevation of the spacer plates. This band is of type 405 stainless steel, the same material as the spacer plates. It is supported and centered within the liner by means of dowels welded to its outside surface to project radially into holes drilled in the liner. The spacer plates for the graphite assemblies are clamped into position by means of twelve straight bars of type 405 stainless steel mounted on the band. Mechanical means are provided to adjust these bars radially inward to bear against the entire cluster of spacer plates in order to center them and to provide proper clearance upon initial assembly. The type 405 stainless steel was chosen for the components in this region because of its lower coefficient of thermal expansion. This feature helps to minimize the sodium gap which appears between the zirconium cans as the temperature of the reactor is increased. The arrangement just described holds the moderator and reflector rigid and properly located beneath the top shield.

X. OUTER TANK AND THERMAL SHIELD

The outer tank is a flat bottom vessel fabricated from low alloy steel. It is 12½ feet in diameter and 19 feet high. The side wall is ¼ inch thick and the bottom is 2½ inches thick. The bottom serves to support the core tank as well as the thermal shield, which is located in the annular space between the core tank and the outer tank. At the top of the outer tank, there is a transition to a type 321 bellows similar to that used on the core tank. The bellows provides a gas seal for the helium atmosphere used in the annular space between the tanks. Pressure of this helium is nominally 3 psig, which is the same as the pressure in the helium atmosphere above the sodium surface in the core tank. The available space in the annular region between tanks is such that the sodium level cannot drop more than 3 feet in the event of a leak in the core tank. This safety feature, which assures sodium cooling within the reactor core, is the primary function of the outer tank.

The flat bottom of the outer tank is supported on a group of four concentric cylinders 21 inches high made from low alloy steel. These cylinders rest on circular bearing plates attached to the bottom of the cavity liner by means of anchor bolts extending into the concrete foundation below. Since these supporting cylinders are not welded at the bearing plates or where they contact the outer tank, there are essentially no thermal stresses as a result of the approximately 400°F temperature gradient established along these cylinders during normal operation. The three inner cylinders are ¼ inch in thickness whereas the outermost cylinder is ½ inch. This outermost cylinder also serves to maintain the outer tank centered relative to the concrete foundation. This is achieved by means of narrow flanges welded to the outside surface of the cylinder at its bottom and at its top. These flanges are machined to have radial notches which fit over cleats welded to the outermost bearing plate and the bottom of the outer tank. This arrangement constrains the outer tank to radial motion only and assures that the center line of the outer tank will remain fixed, in spite of any thermal cycling occurring during operation. The same principle is utilized in maintaining the core tank centered relative to the outer tank. To accomplish this, there is a skirt which is welded near the bottom of the core tank and which contains milled slots fitting over cleats welded to the top surface of the outer tank bottom.

The thermal shield is fabricated from low carbon steel in the form of a series of seven rings with interlocking joints. The over-all height of this thermal shield assembly is 19 feet and each ring is 5½ inches in thickness. The rings are not welded to each other and the bottom ring simply rests on the bottom of the outer tank while being centered by the cleats located there. Practically all of the heat developed in the thermal shield and tank structure is conducted back into the sodium seeping up along the inner surface of the core tank liner. The maximum heat flux produced in this way is estimated to be 3000 BTU/ft^2-hr. Of this amount, only about 100 BTU/ft^2-hr are estimated to be conducted through the thermal insulation to the cavity liner.

XI. INSULATION, CAVITY LINER, AND FOUNDATION

The region between the outer tank and the cavity liner, approximately 12 inches in annular thickness at the side, and 23 inches at the bottom, contains Superex block insulation. This is a commercial product consisting principally of calcined diatomacious silica and asbestos. In order to decrease the dusting during and subsequent to installation, the Superex blocks are coated with a solution of sodium silicate. In the annular region at the side, the insulation is mounted by wire supports fastened to studs welded to the inside of the cavity liner. An atmosphere of about 3 psig of nitrogen will be maintained in this region during operation.

In order to heat the core tank at the time of sodium filling, tubular type heater units are installed on the underside of the outer tank at the supporting cylinder structure. In addition, at this time, it is planned to use tubular heaters suspended from special small plugs in the top shield. These heaters will extend down into the fuel tubes of the reactor core. No other heaters on the reactor will be required. Those at the base of the outer tank are considered necessary in order to reduce the thermal stresses which would occur with the use of the suspended heaters alone.

The cavity liner which is fabricated from low carbon steel is 1 inch thick at its bottom and ¼ inch thick at its side. There are twenty-eight standard 1-inch steel pipes attached to its outer surface for purposes of cooling. At the bottom, where there is more heat load due to conduction down the supporting cylinders, these pipes are patterned to assure adequate cooling and no excessive thermal stresses. As mentioned above, the cavity liner is secured to the concrete base by means of anchor bolts. The openings in the cavity liner through which the anchor bolts project are filled with weld material to assure that the cavity liner also be leak tight.

Under the cavity liner is a 4-foot thick reinforced concrete pad poured on a sandstone base. Around the side, and extending radially outward from the cavity liner, is a cylinder of reinforced concrete about 3 feet in thickness. The maximum estimated heat production in this concrete is 50 BTU/ft^2-hr from nuclear radiation. This is easily removed at the cooled surface provided by the cavity liner without excessive temperatures in the concrete. Concrete is also poured around special extensions welded to the cavity liner in the region where the sodium pipes extend into the two primary galleries. Since the galleries are to operate with a nitrogen pressure of only a few inches of H_2O, the diaphragms sealing the insulation region from the galleries must be capable of withstanding a pressure differential of approximately 3 psi. These diaphragms are welded to the cavity liner and are arranged to have the necessary flexibility required for thermal expansions.

XII. TOP SHIELD

The concrete in the shielding blocks of the primary galleries, in the ring shield and in the rotatable shield is made using magnetite iron ore aggregate. This ore has a density of approximately 4.6 and is mixed in proportions leading to a final concrete density of about 3.6. All of the shields just mentioned use 6 feet of thickness of this dense concrete. The ring shield is 15 feet in outer diameter and 11½ feet in inner diameter at its top surface. It is encased in a type 405 stainless steel form and weighs approximately 60 tons. A gas seal between the ring shield and the surrounding foundation is made by the use of Cerrobend, a commercial low temperature melting alloy, cast into a suitably shaped trough at floor level.

The rotatable shield also uses type 405 stainless steel as a form and weighs a total of about 75 tons when all of its internal plugs are in place. The top stainless steel plate on the rotatable shield does not necessarily need to be welded to the many casings which extend vertically through the shield, but the plates at the sides and bottom are seal welded throughout since these surfaces come in contact with the helium and sodium vapor within the core tank. There is a total of eighty-one small plugs extending through this shield. In addition, there are three larger plugs, two about 40 inches in diameter and one 20 inches in diameter. These three are located so that there can be access for removal of any graphite assembly within the core tank if the shield is rotated to a proper position and one of these three plugs is removed. For this operation, it is necessary to melt the Cerrobend alloy in a tongue and groove seal at the outer edge of the rotatable shield. The shield must then be raised about ½ inch by means of a hoist on the handling bridge, and the shield rotated. Three built-in rollers serve to center the shield at this time. Certain of these features of the rotatable shield are shown in Fig. 14.

It is estimated that approximately 1000 BTU/ft^2-hr of heat will be received by the underside of the top shield. About 85 per cent of this is due to thermal radiation from the hot sodium pool. The contribution from thermal radiation is reduced by the use of insulation in the form of a series of horizontal thin stainless steel plates suspended from the shield. There are thirteen such plates, $^1/_{32}$ inch in thickness, separated by ¾ inch, and assembled in a number of sections. While it serves as a thermal radiation shield, this plate assembly is not gas-tight and will permit some sodium condensation on the seal plate at the bottom of the rotatable shield. The seal plate is type 405 stainless steel, 1 inch in thickness. On its upper surface it is in thermal contact with a 1¼ inch thick lead layer in which tubing is embedded for the circulation of toluene. Immediately above the lead layer is a 1-inch plate of low carbon steel acting as additional thermal shield for the nuclear radiation. The heavy concrete extends from this surface to the top of the rotatable shield but incorporates no other tubing for cooling. Normally, the bottom of the shield operates at a slightly higher temperature than the circulating toluene and may reach about 140°F. If, after a long period of operation, sufficient sodium condenses on the underside of the shield to short out thermally some of the stainless steel plates serving as thermal radiation shielding, it is possible to increase the temperature to the melting point of sodium by reducing the flow of toluene. At this time, the stresses in the casings for the plugs are at a maximum since the lower seal plate expands radially. As designed, this condition is tolerable, but has been made so by the choice of a low coefficient of expansion stainless steel, and by the use of relieving sleeves which are installed at the lower

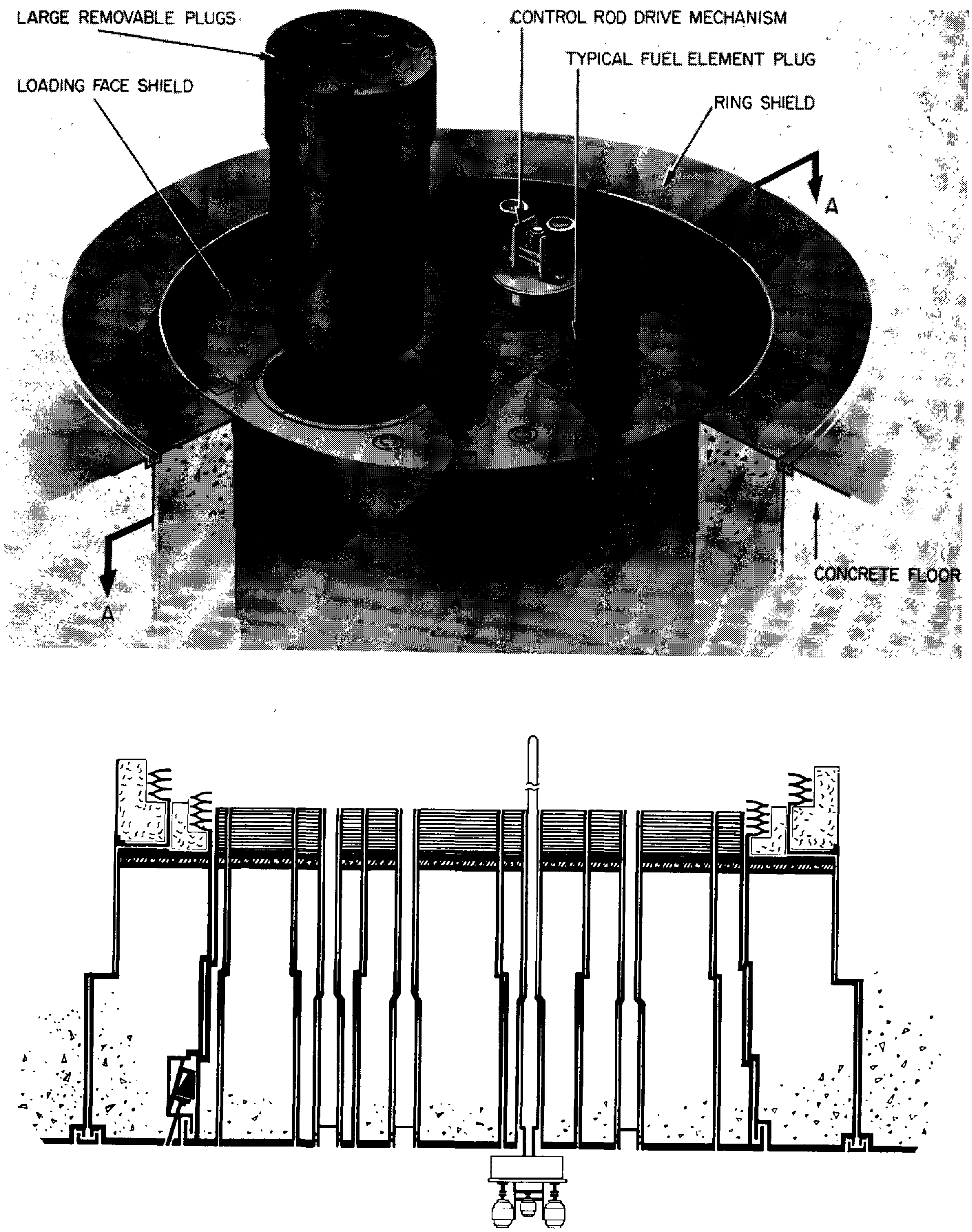


Figure 14. SRE top shield

ends of the casings to prevent seizure of the casings by the concrete in this region.

All of the plugs within the rotatable shield which have been mentioned are stepped to prevent radiation streaming. The step also serves as a means of mechanical support. Sodium vapor is free to diffuse up the annulus between each plug and its casing but should condense in the low temperature region near the bottom of the shield. In all cases the final gas seal on the plugs is made by two "O" rings compressed between the plug and the casing near the top of the rotatable shield. As an extra precaution, a third gasket, depressed by a retaining ring is incorporated at the top lip of each plug, where it is available for maintenance without the necessity of lifting the plug. This retaining ring arrangement is used on all of the plugs in the rotatable shield and performs the additional function of positively locking the plugs in place.

XIII. HANDLING COFFIN

To remove any element from the reactor core it is necessary to use the shielded coffin designed for this purpose. It is carried on the 75 ton handling bridge and may be moved as required within the reactor room. To remove an element from the SRE, the coffin is located over the plug of the element in question. Electrical connectors and the retaining ring with its gasket would have first been removed. A pneumatic mechanism within the coffin then forces a cylinder vertically downward to make an "O" ring seal at the top of the plug casing. Following this, a large lead shield skirt is pneumatically lowered to the surface of the top shield. A gas lock at the lower end of the coffin, where the seal has been made to the casing, is then evacuated by means of a pump and purified helium admitted to the 3 psig pressure existing in the core tank atmosphere as well as in the atmosphere of the body of the coffin. This lock is then opened to the atmosphere within the coffin and a latch mechanism is lowered until it engages the top of the plug of the element to be removed. The direction of the motion is then reversed to raise the plug with its element up into the coffin. A separate mechanism then rotates this entire assembly which performed the lifting operation around to bring a new element into position over the center of the casing. This procedure is then reversed to lower the new element into place, disengage the latch, retract the lifting mechanism into the coffin, close the opening between the lock and coffin body, evacuate the lock, admit air into the lock, raise the shielding skirt, and break the seal made by the lock at the casing. The coffin is then free to transport the irradiated element to the cleaning or storage facilities at the other end of the reactor room.

The coffin just described is 35 feet in over-all height and weighs 50 tons. It has a maximum thickness of 9 inches of lead shielding near its base. This thickness is decreased to ½ inch near the top. No special provision is made for removing heat from the coffin except for a blower which circulates air between the coffin body and the lead shield. Mechanical motion within the coffin is obtained by means of externally mounted motors which rotate shafts sealed at the coffin wall with lubricated "O" rings.

It is planned at a future date to construct another handling coffin, specifically designed for replacement of the graphite assemblies. This coffin can be used on the same handling bridge since side rails are being provided for storage of either or both of the coffins. The storage area will also be used for coffin servicing and will permit the handling bridge to be used within the reactor room with only its general purpose hook attached. The coffin for handling the graphite assemblies will necessarily be larger in diameter and shorter in height but will require a maximum of only about 5 inches of lead.

XIV. CLEANING, STORAGE AND SHIPPING FACILITIES

As an element is removed from the sodium pool in the core tank a small amount of sodium will cling to its outer surface. A special facility is installed near the end of the reactor room for removing this sodium which normally will be radioactive. There are three cleaning cells, each consisting of a steel pipe installed below ground level and equipped with six water spray nozzles. The element to be cleaned is lowered from the handling coffin into a cleaning cell. The plug on the element makes a seal to the cleaning cell casing and also provides radiation shielding. A fine water spray is then used to remove the sodium. Experiments have shown this technique to be effective without leading to excessive heat generation. After the water spray the elements are permitted to soak in the cleaning cell filled with water for perhaps fifteen minutes. The water is then drained to the liquid waste disposal system and the cleaning cell dried by evacuation. Following this, helium is readmitted to the cell. The helium is necessary to provide a heat transfer agent for irradiated fuel elements and to prevent contamination of the coffin atmosphere when the element is removed to another location.

For storage of fuel and other irradiated elements, 96 storage tubes are available. Each tube is a 4-inch low carbon steel pipe, 25½ feet long and capped at its lower end. The tubes are installed vertically and are set in a 2-foot thick layer of dense concrete at their upper ends, which are open at reactor room floor level. The plugs attached to elements in storage serve as additional shielding in this region. It is necessary to use special shielding plugs in storage tubes not containing elements to prevent excessive levels of scattered radiation in the reactor room. In order to remove the after-glow heat generated by irradiated fuel elements, each tube has a cooling pipe welded to its outer surface. The cooling pipe extends from a toluene system manifold near floor level, down, around the bottom, and up the buried tube. This storage facility is necessary not only for the spent fuel, but also to provide a means of storing all elements normally suspended in the SRE core at such time as it may be necessary to rotate the top shield.

The storage tubes just described may be used with all elements in the SRE core, but are not suitable for the canned graphite assemblies. Three larger, 20-inch diameter tubes are installed below floor level as a means of storing such graphite assemblies, if this should be necessary. Also, the three cleaning cells are designed so that they may be easily modified to permit their utilization in cleaning the sodium from these assemblies.

No provision for chemical processing of fuel is being made at the SRE. The fuel elements will be dissembled in the hot cell and then transferred into a specially designed shipping cask by means of the handling coffin. A recess in the floor of the reactor room is available for containing the fuel cask during the loading operation.

XV. SODIUM PIPING AND VALVES

A simplified flow diagram for the SRE cooling system is shown in Fig. 15. All piping is fabricated from type 304 stainless steel with the joints welded to 100 per cent radiographic inspection requirements. Consumable type backing rings are used wherever possible. Piping in the main circuit is schedule 40, 6-inch diameter and in the auxiliary circuit schedule 40, 2-inch diameter. Flexibility to accommodate thermal expansions is achieved by bends. The sodium velocity in the main and auxiliary circuits is nominally 13 feet per second and 5½ feet per second, respectively. Anticipated total pressure drops due to all components are approximately 17 psi and 7 psi in the main primary and auxiliary primary loops, and 36 psi and 17 psi in the main secondary and auxiliary secondary loops. In order to heat the piping and vessels in the cooling system for sodium filling, tubular type heaters and thermal insulation are strapped to all surfaces. Sodium leak detecting cable is attached to the underside of the piping and vessels. Instrumentation in the form of thermocouples, electromagnetic flow meters, and pressure indicators is installed to indicate the performance of the sodium system during operation. Level indicators are also used in the fill tanks and expansion tanks.

Gas vent pipes are connected to the sodium system to permit filling with sodium and to make possible regulation of the gas pressure over the free sodium surface in each expansion tank. The core tank serves as the expansion tank for the primary loops. As a result, the level of sodium above the reactor core is dependent upon the sodium temperature. Over the temperature range to be utilized during operation this sodium level will vary approximately ½ foot. The fill tank for the primary system, which is empty when the sodium is in the reactor, is connected to the core tank by an unobstructed vent line and acts as a ballast to prevent large pressure variations due to temperature or to sodium level changes. The main and auxiliary secondary loops each have an expansion tank in which the gas pressure is regulated to assure a higher sodium pressure in the secondary system than in the

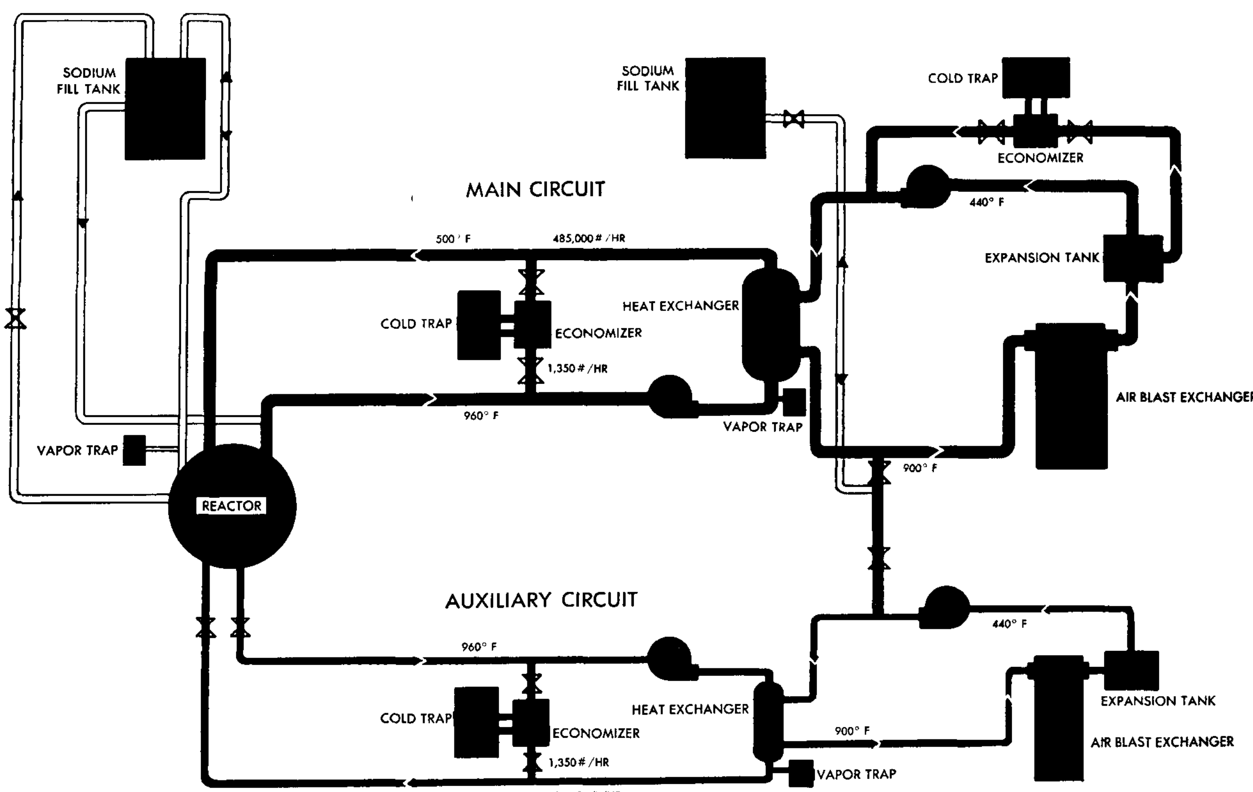


Figure 15. Sodium flow diagram for SRE cooling system

primary. Thus, if a leak should develop between the two systems at either of the intermediate heat exchangers, the flow of sodium would be toward the radioactive primary side. The gas vent lines contain vapor traps to condense any sodium vapor which might otherwise be transported into the gas vent system. In locations where it is possible that liquid sodium could be forced into such a line, a freeze trap is incorporated to solidify the sodium and block the line. These freeze traps have installed heaters which can be used to melt the sodium out at such time as this may be required. The freeze trap serves the function of preventing the sodium from solidifying in parts of the vent line which cannot be conveniently heated to reopen the line for gas flow.

At such time as it may be necessary to empty the cooling system of sodium, drainage will be to the primary and secondary fill tanks. Drain lines and electromagnetic pumps are provided for this purpose in both the primary and secondary systems. In the primary system the drain line reaches to the bottom of the core tank and in the secondary system to the outlets of the intermediate heat exchangers which represent the lowest points in the two secondary loops. Other connections to the sodium system are available for cleaning and flushing the piping, if that should prove necessary.

All valves in the primary and secondary sodium systems use frozen sodium stem seals. Bellows seal valves are restricted to use on drain lines and in the sodium service system. An example of the frozen sodium sealed valve is shown in Fig. 16. A gland circulating liquid toluene replaces the valve packing and causes any sodium in the annulus around the stem to solidify. The valve may be opened and closed by rotating the stem and shearing the frozen sodium. Helium gas is admitted to the region between the frozen seal and the top of the valve to avoid oxidation of the lip of the sodium annulus. There is an additional seal using conventional packing at the valve top to confine the helium gas. Cam lift plug valves are used as blocking valves at the inlet and discharge to the core tank in both the main and auxiliary primary loops. An angle valve is used to throttle the flow of sodium for moderator cooling in the branch line from the main inlet to the core tank. The plug valves in the primary system make it possible for either of the primary loops to be drained independently. Since the secondary loops are separate, they may be independently drained without the use of blocking valves.

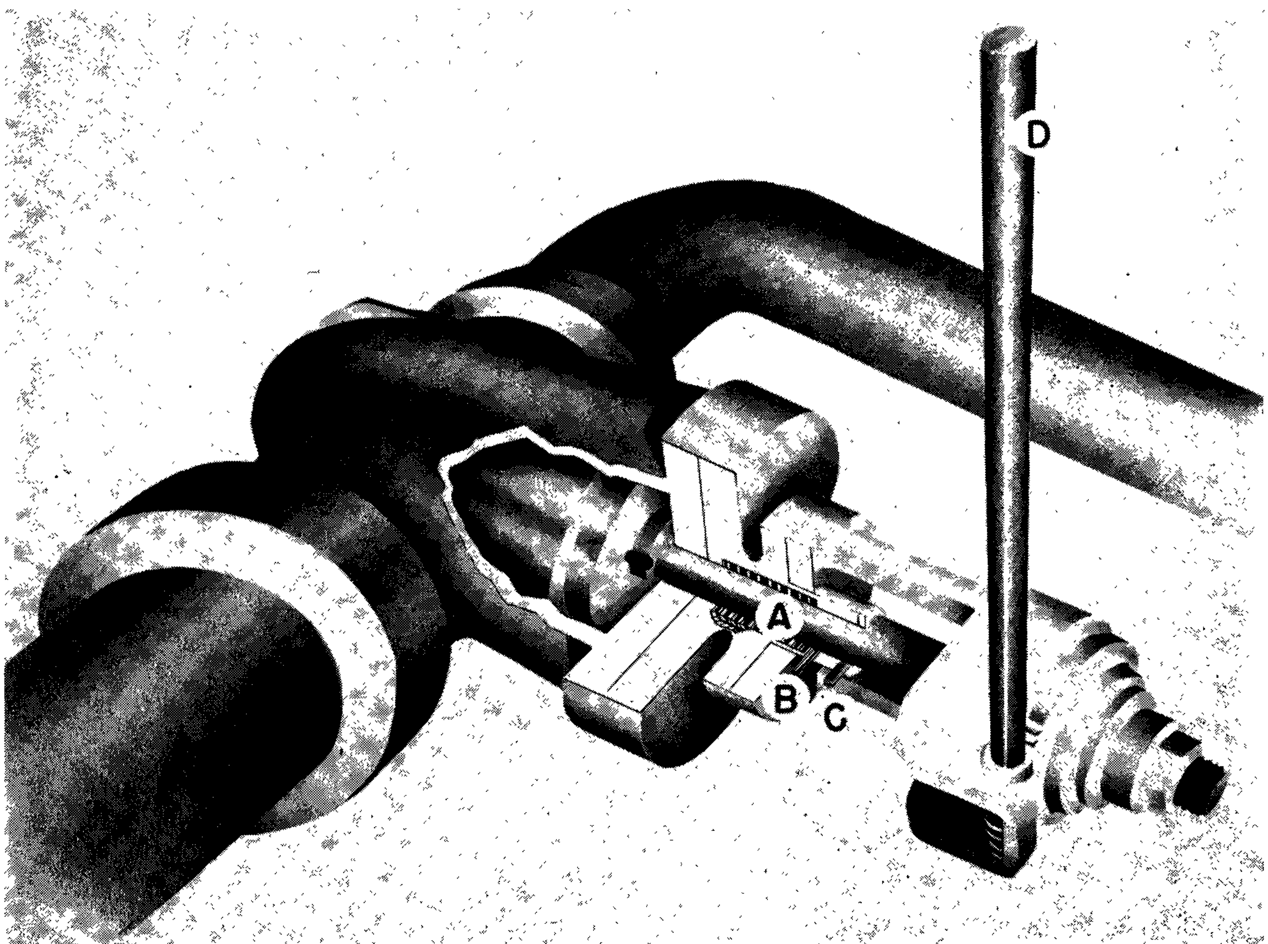


Figure 16. Arrangement for valve with frozen sodium seal

(A) Shaft seal, (B) Coolant line, (C) Gas line, (D) Valve operating shaft

XVI. SODIUM PUMPS

The sodium pumps are modified hot process pumps similar to the type used in refinery service. The principal modifications consist of vertical mounting and the addition of frozen sodium seals at the shaft and at the case. Each of the primary pumps also has its case and drive shaft extended sufficiently to pass up through the gallery shielding to floor level. This arrangement makes it possible to service either primary pump by withdrawing all of its internal parts through the case into the reactor room. Lead shielding is incorporated into this removable assembly just below the floor level.

The case also provides a container for a helium atmosphere maintained at 10 psig, to protect the frozen sodium seal from contact with oxygen. The helium is sealed from the atmosphere by an "O" ring and lubricated face seal at the shaft penetration.

An illustration of the frozen sodium seal pump is given in Fig. 17, and a cross section of the main secondary pump in Fig. 18. Since no shielding is required in the secondary system, the pumps used there do not need the extended case and drive shaft arrangement. The shaft seal is made at a cooled gland inserted in the space normally provided for the pump packing. Toluene coolant is circulated through this gland and through a cooling coil provided at the pump housing for the case seal. Toluene cooling is also circulated to a radial bearing located a short distance above the shaft seal.

Oil for lubrication is fed to this bearing and is continuously removed by being entrained in the helium atmosphere bleed line from a sump immediately below the bearing.

Experiments have been conducted using a prototype of the pump intended for use in the main sodium circuit. With a 50 hp drive, this pump is capable of delivering 1285 gallons a minute against a 130-foot sodium head, while at a speed of 1460 rpm. The shaft power dissipated in the freeze seal is less than 1 kw. Several kilowatts of heat removal are required at both the shaft seal and the case seal in order to maintain the sodium in a frozen condition. The pump is fabricated from type 304 stainless steel except for the shaft and impeller, which are type 316. The diameter of the shaft is 3 inches and that of the impeller is 13½ inches. The impeller is overhung a distance of 15 inches below the lower bearing. Pumps for the auxiliary circuit are similar in design, but of smaller capacity.

The impeller diameter for these pumps is 10 inches.

All of the pump motors are variable speed drive and are accessible during operation. The main primary motor is 25 hp, the main secondary motor 50 hp, and the two auxiliary motors are 2 hp each. During operation of the SRE it will be necessary to adjust the pump speeds as well as the cooling conditions in the air-cooled exchangers to remove the heat generated in the reactor core at required operating temperatures.

XVII. HEAT EXCHANGERS

The intermediate heat exchangers use a shell and tube counterflow arrangement with a "U" type design. This configuration conserves space and minimizes thermal stresses. Both intermediate exchangers use type 304 stainless steel throughout with the tubes being single wall, seamless, ¾-inch outside diameter, 0.058-inch thick wall. The main intermediate heat exchanger has 316 such tubes with a total heat exchange area of 1155 square feet. At the design conditions of 4 feet per second flow in the tubes, the heat transfer coefficient is approximately 985 BTU/hr-ft^2-°F. In the auxiliary intermediate heat exchanger, only 38 tubes are used and with the design flow of 1.5 feet per second will have a heat transfer coefficient of approximately 800 BTU/hr-ft^2-°F.

The air-cooled exchangers in the secondary system have "U" shaped tube bundles installed in protective housings above motor driven fans. The tubes are similar to those in the intermediate exchangers but have mounted type 410 stainless steel fins 2 inches in diameter, spaced eight per inch. All tubes are connected in parallel from headers at one side. The direction of airflow is concurrent to the sodium flow to reduce the danger of freezing in the tubes at low sodium flow rates. Mechanically operated louvers are installed above the tube bundles to assist in the control of the flow of air. Just below the tube bundles are mounted tubular electric heaters for preheating the system and for providing make-up heat during operation at low power. These heaters will maintain a temperature of 350°F in the tubes during shutdown of the reactor.

In the main air-cooled exchanger there are 204 "U" tubes 26½ feet in length. They provide 23,600 square feet of heat removal surface. At normal operating power with the estimated heat transfer coefficient of 8.2 BTU/hr-ft^2-°F, the exhaust air temperature will be 277°F based on air in at 100°F. Below the tube bundles are mounted two 11-foot diameter fans each driven by a 50-hp variable speed motor. The auxiliary air-cooled heat exchanger is similar in design, but has only thirty "U" tubes 8 feet in length. It utilizes a 5-foot diameter fan driven by a 5-hp variable speed motor.

In the SRE the intermediate heat exchangers are mounted in the galleries at an elevation above the reactor core. The air-cooled exchangers are installed at a still higher elevation. As a result, even without forced circulation by the pumps, it is possible to obtain cooling by natural convection in both primary loops and in both secondary loops. This arrangement constitutes an excellent safety feature since it is possible to remove a certain amount of after-glow heat from the reactor core under emergency conditions. It is estimated that with normal operating temperatures, as much as 2000 kw of heat could be removed by natural convection circulation within the complete main circuit, including the cooling by air convection at the main secondary exchanger.

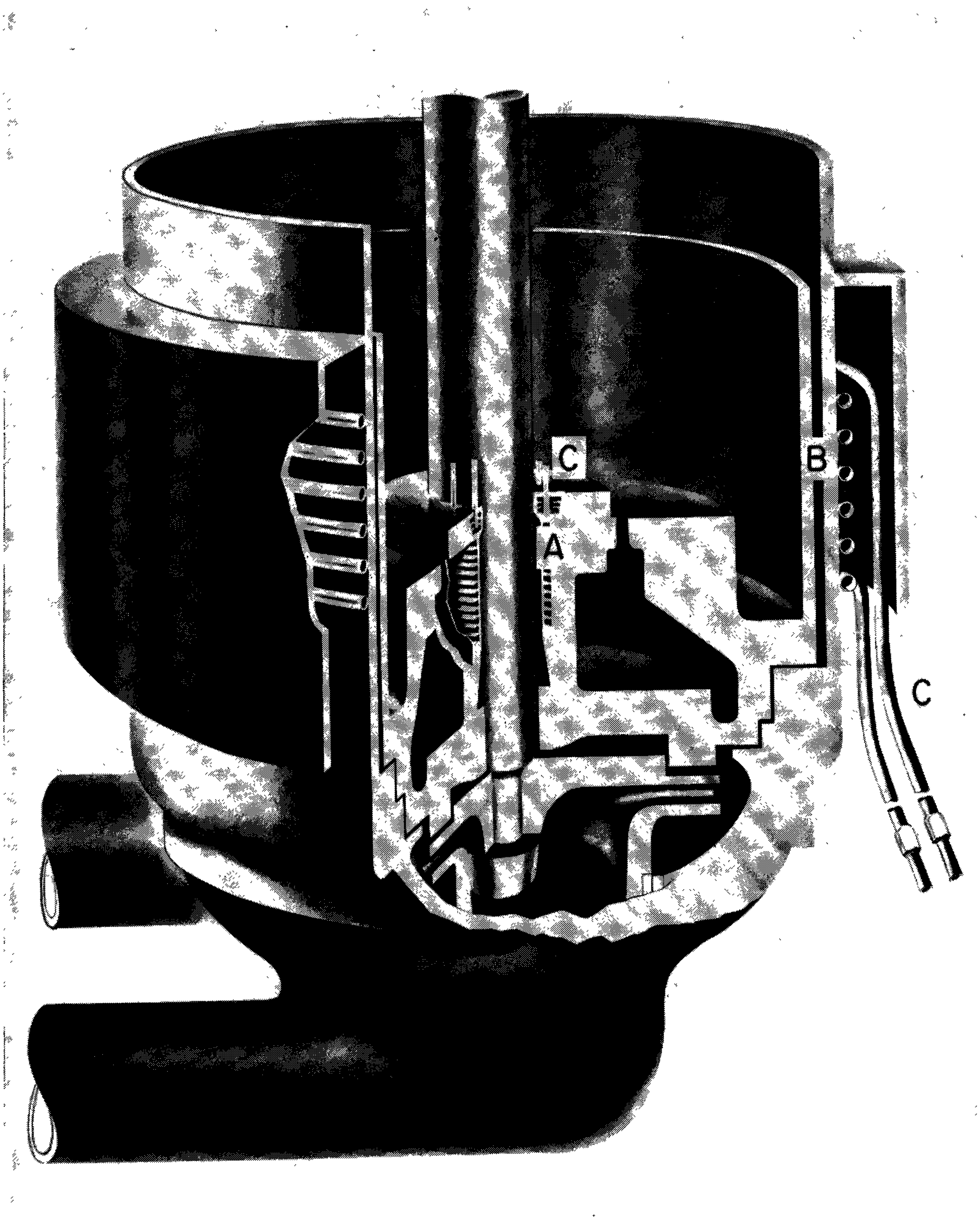


Figure 17. Arrangement for mechanical pump with frozen sodium seals (A) shaft seals, (B) housing seal, (C) coolant lines

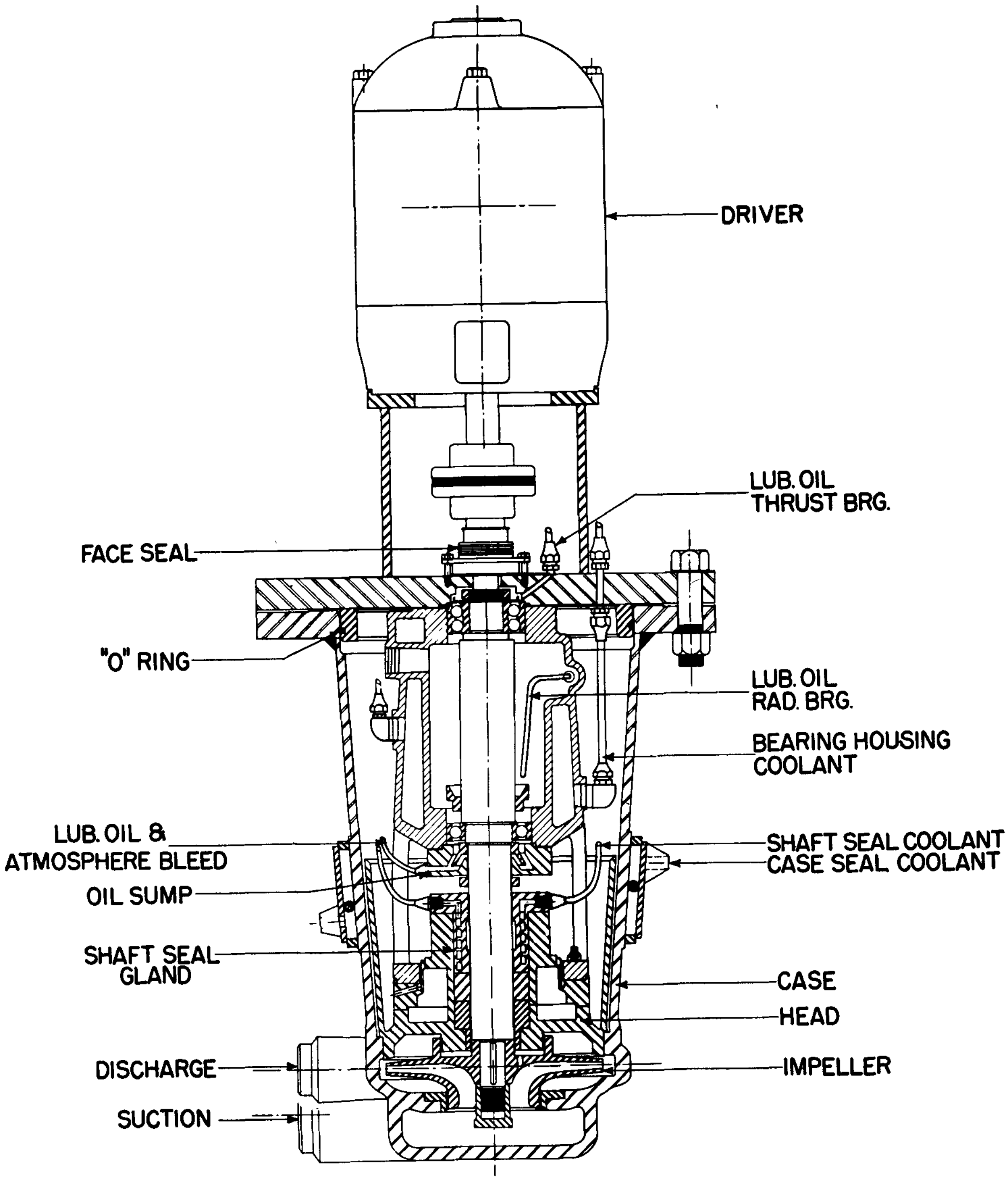


Figure 18. Cross section of main secondary sodium pump

XVIII. COLD TRAPS

Cold traps are incorporated in each of the four loops of the sodium cooling system in order to maintain the sodium oxide content at a sufficiently low value. A special design of circulating cold trap is being developed for use in the primary loops and the secondary main loop. This design is schematically shown in Fig. 19. The principal new feature is the toluene jacket and condensing loop to maintain the low temperature of the sodium in the trap at a nearly constant value in spite of variations in sodium temperature and pressure in the cooling system during operation. The lowest temperature of the sodium in the cold trap will be a few degrees above the 232°F boiling point of the toluene. The toluene which boils from the jacket surrounding the cold trap passes into a condenser and then as a liquid into the reservoir which controls the liquid level within the jacket. Separate cooling for the condenser is provided from the toluene system servicing the various SRE components. Pressure for circulating the sodium through the cold trap is automatically obtained by connecting it across the pump in any one loop. The sodium first passes through an economizer section, and then through the cold trap vessel which is filled with stainless steel mesh. This particular trap is designed for a flow of 1350 pounds per hour of sodium and is expected to maintain the sodium oxide content below 0.005 weight per cent.

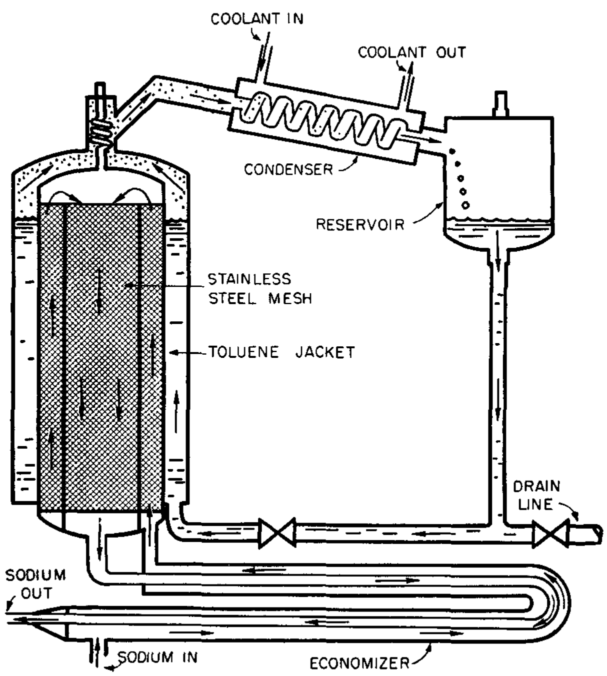


Figure 19. Sodium oxide cold trap utilizing boiling toluene jacket

Where the capacity of the cold trap is less important, it is planned to use diffusion cold traps. One will be attached at the base of the surge tank in the second auxiliary loop and one on the secondary fill tank. These units each consist simply of an 8-inch diameter capped pipe 15 inches long welded to the bottom of the tank. At the base of the diffusion trap a pipe for circulating toluene coolant is attached by welding. Sodium in the base of the diffusion trap may be frozen but the heat losses with this arrangement are negligibly small.

To determine the oxygen content in the sodium, a plugging meter is used in conjunction with each circulating cold trap. The plugging meter is connected in parallel with the cold trap. Its principle of operation depends on the closing of small orifice holes as the temperature of the sodium is lowered and the sodium oxide precipitates out. The sodium oxide content may be determined after calibration by the break in the sodium flow versus temperature curve. In order to speed the rate to which successive plugging meter determinations may be made, a special design is being developed in which the orifices are machined into the seat of a valve. The valve may be opened for a short period to encourage dissolution of the sodium oxide and prepare for another determination. The plugging meter also incorporates an economizer and a cooling line from the toluene system. An electric heater is used to vary the temperature.

XIX. SODIUM SERVICE SYSTEM

The sodium service system comprises equipment for melting, filtering and introducing sodium to the fill tanks as well as for flushing the primary system cold traps. Type 304 stainless steel is used throughout. When the sodium service system is not in use, the lines and vessels are emptied of sodium and a helium atmosphere maintained in the system. All vessels and sodium lines are provided with heaters and insulation in order to preheat to 350°F.

Sodium is first introduced from drums at two melt stations and flows through sodium filters at about 250°F. It may then be directed to either the primary or the secondary fill tank. To transfer the sodium from the primary fill tank into the primary system, it is necessary to use the sodium pumps, once prime has been established by gravity flow. This method is necessary since the primary fill tank communicates to reactor core tank which cannot be pressurized. In the secondary system filling of both the main and auxiliary loops is accomplished by pressurizing the fill tank.

Since the primary system becomes radioactive, special provision has been made for flushing the cold traps in both primary loops to a disposable cold trap in the sodium service system. An electromatic pump and valves are so arranged that hot sodium can be passed through either primary cold trap in order to carry the sodium oxide to the disposable unit. This disposable cold trap in the service system may also be used for removing the sodium oxide from sodium being stored in the primary fill tank, without requiring circulation into the primary system piping. There is no diffusion cold trap on the primary fill tank.

XX. TOLUENE SYSTEM

As a safety precaution all components in the SRE requiring special cooling are served by the system circulating liquid toluene. This material does not chemically react with sodium, but can be pyrolitically decomposed at sufficiently high temperatures. A summary of the components to which toluene cooling is provided is given in Fig. 20. Toluene from a reservoir containing 500 gallons is circulated by two pumps operated in parallel. Throttle valves for each of the major components served are used to regulate the flow to obtain a temperature increase of about 30°F. The toluene is supplied from the reservoir at 95°F. The maximum estimated flow is approximately 360 gallons per minute, which with the temperature rise just indicated, provides for the heat removal of roughly 500 kw. This amount includes 150 kw allowed for the fuel storage tubes. Other principal heat loads are approximately as follows: top shield, 60 kw; cavity liner, 70 kw; main primary loop, 80 kw; auxiliary primary loop, 55 kw; main secondary loop, 45 kw; and auxiliary secondary loop, 40 kw. The return toluene is cooled in two evaporative cooler units operated in parallel and is then piped to the reservoir tank. Low carbon steel pipe and vessels are used throughout. Separate motors drive the two circulation pumps. In the event of power failure these motors will be automatically operated from the emergency electrical system. In order to reduce the load on this emergency system, a gasoline engine is also provided which may be started and coupled to either of the toluene pumps.

There are certain disadvantages to the use of toluene instead of water as a special coolant. Since its flash point is quite low, the use of toluene is accompanied by a fire hazard, and the vapor is also somewhat toxic. Other hydrocarbons have been considered as alternates for the toluene and may be substituted at some later date without requiring any system modification. Two alternates being considered are xylene and tetralin, of which the tetralin appears the more promising. In addition to a higher flash point and lower toxicity, tetralin would have the advantage of a substantially higher boiling point, while still being liquid at room temperature. Other characteristics of the liquids mentioned, including radiation decomposition rate, do not appear to offer special problems.

XXI. INERT GAS SYSTEM

An inert gas system provides helium and nitrogen at various locations around the SRE installation. Both gases are stored in banks of high pressure bottles in an area outside the reactor building. In the case of each gas, the pressure is reduced to 50 psi in standard regulators and is directed to a low pressure storage tank. In order to minimize the oxygen and water vapor content of the helium, this gas is passed through NaK bubblers prior to introduction into the low pressure tank. Distribution from these storage tanks is then made to various points within the reactor building where individual regulators further reduce the pressure to a value suitable for each component serviced. Over-pressure safety valves are provided at various points. The gases are non-circulating during operation except for the helium used to circulate lubricating oil in the sodium pumps.

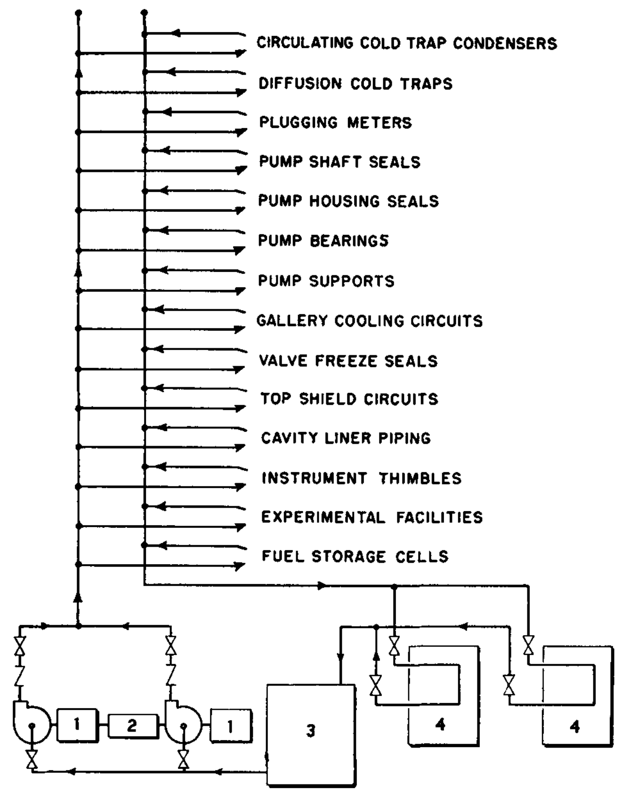


Figure 20. Components serviced by toluene coolant system; (1) motor; (2) emergency gasoline engine; (3) toluene reservoir; (4) evaporative cooler

The helium is used in numerous locations. Examples are the fuel handling coffin, cleaning and storage facilities, sodium pumps and valves, control and safety element thimbles, regions surrounding the core tank and within the double walled inlet pipes, as well as all components in the sodium system which require a free sodium surface during filling and draining or during normal operation. Examples of the latter are the fill tanks, expansion tanks, and cold traps.

The nitrogen distribution is somewhat simpler since this gas is used primarily only for the various galleries and the cavity containing the reactor insulation. To reduce the amount of nitrogen used while maintaining the gallery pressures at a very small value in spite of temperature fluctuations, the gallery atmospheres connect to a constant pressure tank located a short distance from the reactor building. These galleries may be isolated by valves in the piping, but normally will have a pressure of ¼ psig as automatically established by the movable top on the constant pressure tank.

Where helium or nitrogen are provided to regions which will contain sodium vapor only in the event of a leak in the sodium system, it is planned to sample the gas periodically. This method of sodium leak detection is especially useful for components in which electrical shorting detectors cannot be conveniently installed. Examples are the control and safety element thimbles and the double walled sodium inlet pipes. The method of detection to be used involves bubbling of a gas sample through an indicator solution of thymol sulfonthalein. It is not difficult by this means to detect as little as 10^{-7} mol of sodium in the gas sample. Use of this technique does not require heated tubing for transporting the gas containing sodium vapor. Experiments have shown the vapor to be almost quantitatively transported through 30 feet of ¼-inch tubing at room temperature.

XXII. WASTE DISPOSAL SYSTEM

Provision is made for the disposal of both gaseous and liquid radioactive wastes. The gaseous waste problem arises from the possibility of ruptures in the fuel element cladding which might release gaseous fission products into the primary sodium system. As a result, each gas line which connects into the primary system and is required to vent helium for reasons of pressure control, is connected in such a way that if excessive radioactivity is detected, this gas will be pumped into radioactive storage tanks. This is accomplished by means of a radiation detector coupled to valves which automatically shunt the gas stream being vented into a compressor suction tank whenever an excessive level of radioactivity is detected. From the suction tank, one of two parallel compressors forces this gas into a storage tank where it can undergo radioactive decay. There are two shielded storage tanks of 5400 cubic feet at 100 psig capacity. If at some later time the activity level is determined to be sufficiently low, this gas may be bled from the storage tanks and discharged out the building vent line. This line emerges at the roof of the reactor building where the gas released is diluted by the forced circulation from the building ventilation system.

The arrangement just described is shown in simplified form in Fig. 21. As indicated there, the principal components which may require gas venting to the radioactive storage tanks are the core tank and primary fill tank, the gas lock on the fuel handling coffin, and the fuel cleaning cells.

The liquid waste disposal system is somewhat simpler, as shown schematically in Fig. 22. The only components producing radioactive liquid waste are the cleaning cells and the hot cell. Drains from these components will normally carry water with radioactive contaminants directly to a sump. From here a pump will force this liquid up into one of a series of ten 50-gallon hold-up tanks, as determined by valve settings.

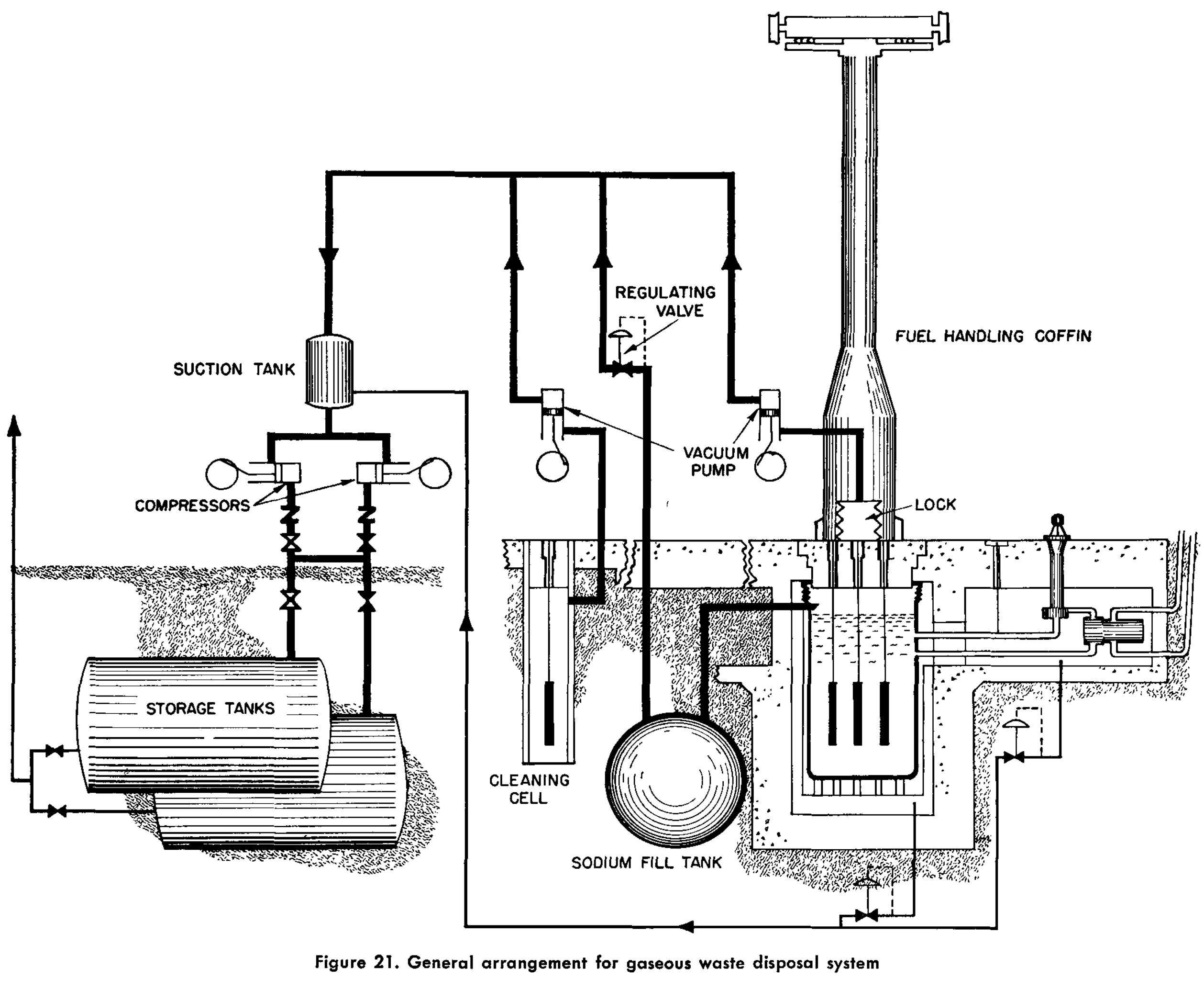


Figure 21. General arrangement for gaseous waste disposal system

Liquid in these tanks may be stored and sampled as required. Depending upon the radioactivity level after this period of hold-up, the waste is directed into either of two 5000 gallon storage tanks, or to the industrial sewage system in the case of very low radioactivity levels. No further provision is made for disposal, although if these large storage tanks should ever become full, it will be necessary to finally concentrate the waste and dispose of it by special means.

XXIII. EMERGENCY ELECTRICAL SYSTEM

In the event of failure of the electrical power provided to the reactor building, there are some SRE components which must be maintained operative on the emergency electrical system. The reactor itself will be automatically shut down, but instrumentation and heat removal equipment must continue to operate. This total emergency load approaches 50 kw, of which almost one-half is required for toluene pumping.

The continuous emergency power comes from a battery driven motor alternator. This set consists of a 100 hp synchronized motor and a 60 kw dc diverter pole generator. It will act as a battery charger and dc power source during normal operation and as a battery driven motor alternator during emergency operation. Because of the continuous operation of the motor generator set necessary to insure emergency power, two identical sets are provided with switching arranged so that either one can be idle for maintenance. The battery is rated at 120 volts and is capable of delivering 480 amperes for ½ hour and 280 amperes for an additional hour.

As a backup for the battery supply there is a delayed emergency power unit in the form of a Diesel engine driven alternator rated at 100 kw. This engine alternator will start automatically at loss of normal power. When up to proper speed and voltage, the output of the alternator will be automatically synchronized with the output of the motor generator and the two will be paralleled. The Diesel set will then take over the entire load and the motor generator will become a battery charger again. Since the toluene pump comprises the largest emergency load a separate gasoline engine is installed there, as described previously. This engine is of greatest value in case the Diesel alternator cannot be started, at which time complete dependence will be on the batteries which have limited life. When normal power returns, the electrical output of the emergency system is automatically synchronized and the load is switched.

XXIV. EXPERIMENTAL FACILITIES

The purpose of the construction and operation of the SRE is not only to carry through the engineering of an experimental sodium-graphite type reactor but to perform measurements and system modifications which will establish performance limits and design improvements. In the previous sections the design of the SRE for initial installation has been described with some discussion of the engineering choices which have been made. This description has been based on a set of normal operating conditions. It is realized that some experience will be required during the early stages of operation to even achieve these design conditions, but it is also hoped that later in the program it will be possible to exceed these conditions and better establish the limits of performance of the various components.

As one step in striving for high performance, an attempt will be made to operate at maximum sodium temperatures approaching 1200°F. As designed, the vessels and piping in the reactor and sodium system

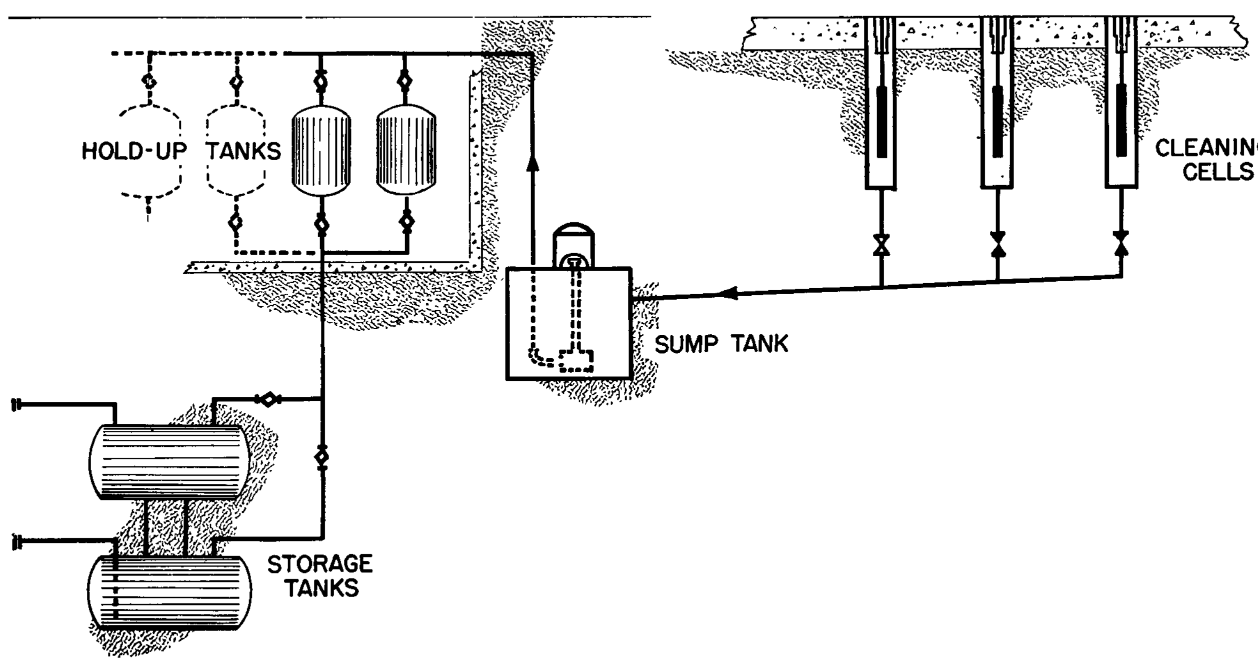


Figure 22. General arrangement for liquid waste disposal system

have been stressed for this temperature condition. To achieve such operation will probably require modifications in the fuel and in the zirconium canned graphite assemblies. Actually, one of the principal objectives of the experimental work is in connection with improvements in the reactor fuel, especially from the standpoints of specific power, operating temperature, and burn-up. Measurements will also be made on the graphite assemblies to determine their outgassing properties and mechanical integrity. In a similar manner, measurements will be made on many other components such as the control and safety elements and sodium pumps.

The hot cell, located below floor level at the end of the reactor room, provides a facility for examination of any radioactive components removed from the reactor. Its principal use will be in connection with fuel element studies, but measurements can also be made on other parts, such as graphite assemblies and control and safety elements. The hot cell is divided into two parts with internal dimensions 5 feet by 15 feet by 16 feet and 5 feet by 8 feet by 11 feet. There are three sets of manipulators and three viewing windows. Components may be inserted into the hot cell through one of three holes at the end of the cell nearest the reactor. Two of these holes are the same size as the small plugs in the reactor shield and can receive any of the elements normally suspended in the reactor core. The other hole is larger in diameter and can accept a graphite assembly. In either case, a handling coffin would be used to transport the radioactive component from the reactor and into the hot cell.

As described earlier, there are twelve fuel tubes and six corner channels in the reactor core which will probably contain dummy elements. It is possible to replace these elements with special units for experimental purposes. If it is necessary to avoid contact with the sodium, such a unit can use a stainless steel thimble attached to a shield plug. Special cooling within the thimble could be provided if it is necessary to perform the exposure at a lower temperature than that of the circulating sodium. The number of experimental units used at any one time will depend upon the reactivity requirements of the core as well as the location and type of experimental units desired. In addition to the channels just mentioned, there are three at center locations of the graphite assemblies at successively greater radii. The tubes associated with these locations do not connect into the lower plenum and hence receive only the moderator cooling sodium. They are intended for the measurement of the radiation levels at various heights and radii within the side neutron reflector, but are also available for other experimental purposes if desired.

Another experimental facility deserving mention is incorporated into the main primary loop. One-inch pipes are welded into the inlet and discharge lines of the main intermediate heat exchanger and carry sodium to two vertical 2½-inch diameter stand-pipes. These two sodium streams are then mixed and returned in a common pipe to the suction side of the primary pump. This arrangement permits insertion of samples from floor level in the reactor room down into the stand-pipes; one of which will normally contain flowing sodium at 500°F, the other at 960°F. During operation, a frozen sodium seal is made at the top of each stand-pipe. Above this is a shielding plug. It is planned to insert samples of different materials into this facility to study corrosion and mass transfer effects under actual operating conditions.