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THE SODIUM GRAPHITE REACTOR: TOMORROW'S POWER PLANT

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520 001

CONTENTS

		Page
I.	Introduction	1
II.	SRE Operating Experience	4
III.	Hallam Nuclear Power Facility	6
IV.	Current Status of SGR Concept	7
V.	Advanced Sodium Graphite Reactor	8
VI.	Description of 256-Mw ASGR Power Plant	9
	A. Reactor	11
	B. Sodium Heat Transfer Systems	15
	C. Electrical Generating Plant	18
	D. Plant Control System	18
	E. Economics of 256-Mwe Advanced SGR	19
	F. Development Program for ASGR	20
	References	35

FIGURES

1.	Sodium Reactor Experiment (SRE) in Full-Power Operation	21
2.	Large Component Test Loop (LCTL) for Environmental Testing of Full-Scale Sodium System Components	22
3.	Hallam Pump Test Facility With Two Types of Sodium Pumps Under Test	23
4A.	Construction Progress of the Hallam Nuclear Power Facility (HNPF - October 1959)	24
4B.	Construction Progress of the Hallam Nuclear Power Facility (HNPF) - February 1960	25
5.	Plant General Arrangement (plan elevation 95'0")	26
6.	Plant General Arrangement (sections)	27
7.	Calandria	28
8.	Intermediate Heat Exchanger	29
9.	250-Mwe-ASGR Steam Generator	31
10.	Preliminary Heat-Balance Diagram	33

520 002

I. INTRODUCTION

At the start of the United States power reactor program, it was recognized that the extraordinary heat transfer properties of liquid sodium made it a very promising reactor coolant. It has since been demonstrated that use of liquid sodium results in a compact, low-pressure reactor system which requires a minimum of pumping power and is capable of operation at high temperatures required for generating steam used in modern turbine-generators.

At the present time, the types of sodium-cooled reactor systems being developed are: graphite-moderated thermal reactors, and intermediate- and fast-breeder reactors of various configurations. This paper describes only the sodium graphite reactor (SGR) type of nuclear power plant

Technology of the SGR concept has advanced beyond initial construction and operating phases, permitting fairly accurate predictions of future costs. As technology of steam cycles continues to advance, interest in the SGR as a heat source for generation of high-quality steam should increase. Several single 500,000-kw steam-turbine units and one 600,000-kw unit are currently on order in the U. S. for installation in conventional power plants. The power industry is looking forward to installing units of 750,000 to 1,000,000 kw within a period of a few years. Net plant heat rates are expected to approach 8,000 Btu/kwh. It appears unlikely that units of such size and refinement will be designed for anything except high-pressure, high-temperature steam in multiple-reheat cycles. As a safe nuclear heat source to produce this high-quality steam, the SGR has an unmatched potential.

Inherent safety of the SGR is due to its (1) unusually low operating pressure (only a fraction of an atmosphere over the core), (2) absence of chemically incompatible materials in the core, and (3) excellent stability under static and dynamic conditions. Experience acquired at the Sodium Reactor Experiment (SRE) has proved that liquid sodium can be safely handled and will not constitute an important hazard factor. Design of the Hallam Nuclear Power Facility demonstrates the safety of the SGR concept, since it will be the only nuclear power plant to have been built in the U.S. without a containment vessel.

The first part of this paper presents a brief history of the SGR concept, a summary of experience gained from the SRE, a review of the major features of the SGR prototype plant being constructed at Hallam, Nebraska, and a brief summary of the current status of the SGR concept. The body of this paper is a description of a full-scale plant incorporating advanced SGR design features and capable of generating 255,000 kw net electrical power. This plant shows promise of competing with conventional thermal plants at present-day costs of fossil fuels.

The United States Navy pioneered the use of sodium as a reactor coolant in its nuclear submarine program. A prototype sodium-cooled reactor was built at West Milton, New York, and a second unit was installed in the submarine Sea Wolf. Both of these were epithermal reactors, but incorporated liquid-sodium heat-transfer systems which were direct forerunners of those being used today. The naval program made extensive contributions to the technology of using liquid metals. For example, the temperatures at which they could be routinely handled were substantially advanced from the 400-500°F range of the chemical industry. However, the severe requirements of submarine construction (limited space, mechanical shock, and warship accessibility) eventually led to abandonment of sodium as a coolant for naval reactors. Nevertheless, in two years of operation, the Sea Wolf is reported to have suffered no defects in reactor-compartment operating machinery, with the exception of leaks in its steam superheaters.

The Experimental Breeder Reactor I (EBR I), built near Arco, Idaho, was not only the first fast breeder in the U.S.A., but also the first nuclear reactor known to have generated electrical power. Cooled by a low-melting-point sodium-potassium alloy (NaK), it attains temperatures up to about 800°F, but is not subject to shipboard mechanical limitations. Operating history of the reactor has been notably successful and reliable. EBR I began operation in 1952 and is still operating with the original heat transfer equipment.

The SRE was authorized in 1954 as part of the U.S. Atomic Energy Commission's Five-Year Development Program. It was designed as a developmental facility which utilized the most advanced SGR technology known at that time. Sodium coolant temperatures were advanced to 1000°F. Construction of this 20-Mwt plant was begun by Atomics International at Santa Susana, California, in 1955. The SRE achieved initial criticality in April 1957.

520 004

Although the SRE was originally designed to dissipate its heat directly to the atmosphere, the program was broadened in 1956 to permit generation of electricity. Arrangements were made for Southern California Edison Company to install steam generating equipment and a 7500-kw turbine-generator adjacent to the reactor. The SRE has demonstrated its ability to produce superheated steam which exceeds the turbine design conditions of 600 psi and 825°F.

Before SRE construction was completed, the U.S. AEC began negotiations with Consumers Public Power District of Columbus, Nebraska, and with Atomics International for the design and construction of a 76,000-kwe sodium graphite reactor power plant. This plant will be known as the Hallam Nuclear Power Facility (HNPF) and will demonstrate the economic and technical practicability of central station power using an SGR as the heat source. Estimated cost of power from this plant compares quite favorably with that of other prototype nuclear power plants.

Final design of the plant began in September 1958, and ground was broken for construction in April 1959. A brief review of construction progress and important design features of the HNPF project follows discussion of the SRE.

520 005

II. SRE OPERATING EXPERIMENT

Other publications have given detailed information on design and construction of the SRE (see References at end of paper).

After an early shutdown for installation of eddy-current brakes (to limit temperature transients following rapid shutdowns), the system enjoyed over a year of very successful performance. Although operated as a component test facility, which necessitated frequent shutdowns for removal and examination of test items, the system has generated over 15,000,000 kwh of electricity. This activity, combined with experimental and operational scrams, has resulted in startup experience equivalent to 20 years of normal operation. The SRE is shown in Figure 1.

Ability of the system to produce modern steam conditions has been proved. In May 1959, the SRE completed a 1,000,000-kwh high-temperature run during which the steam temperature was maintained at 900°F, with a gross power output of 5.6 Mwe. In a special test on May 22, 1959, the system produced steam at 1000°F for a period of 55 min. Bulk sodium temperature at the reactor outlet during this test was 1065°F, with power output limited to 1 Mwe to reduce the temperature differential through the steam generator. During this run, it was necessary to reduce the degree of steam superheat to permit utilization by the turbine-generator, which was designed for 825°F.

Reactor stability has also proved to be outstanding under both static and dynamic conditions. During a 144-hr period of steady-state operation at full power, integrating timers showed a total of only 3.5 min of control rod movement. Dynamic stability of the reactor at all frequencies measured (up to 20 cps) has been demonstrated by a series of pile oscillator tests at various power levels.

In July 1959, failure of the freeze-seal on a main coolant pump permitted leakage of organic liquid (tetralin) used for auxiliary cooling into the primary coolant system. Coking of this organic material apparently resulted in clogging and subsequent overheating of several fuel elements. As a result, 12 of 43 fuel elements and several of the 119 moderator cans sustained some damage. Fuel and moderator handling required to remove the damaged elements has caused the

reactor to remain shut down since July. In the course of this operation, new tools and improved core maintenance techniques have been developed which will be valuable in the maintenance of all sodium-cooled reactors. Use of tetralin as a freeze seal coolant has been eliminated, and the SRE is expected to resume full-power operation with its second core in the spring of 1960. With Th-U alloy fuel elements used in the new core loading, the SRE will resume its role as a test reactor capable of exposing fuel elements at high temperatures.

520 007

III. HALLAM NUCLEAR POWER FACILITY

Construction of the HNPF was authorized by the U.S. AEC as part of the first round of its Power Demonstration Reactor Program. While this reactor will incorporate basic SRE design technology, it will provide an important scale-up in the size of various components. The first HNPF core will be fueled with an alloy of uranium with 10 wt % Mo, which has good stability under radiation. This material, however, while satisfactory as an interim fuel at HNPF operating conditions, is not considered economical for more advanced sodium graphite reactors. Current fuel development work is directed toward utilizing uranium carbide elements to replace the first core in 1963. Some experimental uranium carbide elements will probably be included in the first HNPF core.

Substantial success has been achieved in the component development program for the HNPF project. Full-scale control rods, moderator cans, bellows, and pumps have been tested for prolonged periods at 1000°F in large liquid-metal facilities constructed for that purpose at Santa Susana, California. Moderator cans and other core components have been given thermal shock tests that greatly exceed the conditions expected in the reactor. These tests were conducted in the Large Component Test Loop (Figure 2.) Two types of sodium coolant pumps have been extensively tested at the pump test facility (Figure 3), and one of these has been selected for installation at Hallam. The fuel handling equipment has been assembled over a reactor mock-up and will be checked out under non-radioactive conditions prior to shipment to the site.

Construction of the HNPF is proceeding well. The substructure and the core-tank enclosure are substantially complete, as are the framework and crane for the reactor building. Completion of construction is scheduled for June 1961, with full-power operation expected in mid-1962. Two photographs showing construction progress are shown in Figures 4a and 4b.

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IV. CURRENT STATUS OF SGR CONCEPT

The HNPF reactor will operate with sodium temperatures similar to those of the SRE, and will therefore generate steam for a similar power conversion cycle. Since the HNPF is the first SGR designed primarily for power production, its design is somewhat conservative, and does not include the more recent advances in SGR technology. The plant, however will make a major contribution to the development of large sodium-cooled power plants.

A preliminary design and cost estimate was recently completed for a 300-Mwe SGR which incorporates many of the recent advances in sodium reactor technology. This design is termed the current-status SGR. It compares with the basic design of the HNPF as follows:

1) The same fuel material (U - 10 wt % Mo) was assumed, so the sodium temperatures (945°F at reactor outlet) are essentially identical.

2) A significant saving in capital cost is effected through an adaptation of the steam generator to be used in the Enrico Fermi Fast Breeder Reactor plant. This generator utilizes a single-wall tube vs the double-wall tube to be employed at HNPF.

3) Cost of the primary heat transfer system is reduced by close-coupling of the reactor, pumps, and primary heat exchangers. This is effected by using bellows in the low-pressure sodium piping rather than expansion loops.

4) Because of the greater plant size, use of more equipment in the steam cycle is economically justified. Use of reheat and improvement in steam cycle efficiency result in a calculated net plant thermal efficiency of 36.8% vs 31.7% for the HNPF.

In overall economics, the 300-Mwe current-status reactor would compete favorably with other current reactor types. An independent detailed cost estimate has just been completed, by one of the largest engineering firms in the U.S., indicating that a plant of this design could now be built for approximately $200/kw, including interest during construction.

520 009

V. ADVANCED SODIUM GRAPHITE REACTOR

The current-status SGR is likely to be outmoded in the near future, however, by a more advanced sodium reactor, designated the Advanced Sodium Graphite Reactor (ASGR). Based on the use of fuel elements permitting higher sodium temperatures and greater burnup, the ASGR would provide fully modern steam conditions with net plant efficiencies over 40%. Significant savings in fuel cost are thus expected. Capital cost would also be lowered by increased efficiency and reduced sodium flow requirements. These benefits, however, would be somewhat offset by the more expensive steam generators required for the modern heat conversion cycle (2400 psi/1050°F/1000°F).

The key to improved fuel element performance lies in the development of a fuel material with good thermal conductivity and resistance to radiation damage at high temperature. Development work, now being performed by Atomics International under the AEC's Advanced Sodium-Cooled Reactor Program, indicates that uranium monocarbide possesses both these properties. Capsule irradiations of UC are now being performed for typical reactor fuel configurations. Prototype elements are also being designed, with performance tests scheduled for initiation in the near future.

Upon successful completion of this work, the ASGR is expected to be competitive with conventional fossil-fueled power plants in most industrial areas of the world where units of 200 Mwe or larger can be used.

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VI. DESCRIPTION OF 256-MW ASGR POWER PLANT

The following sections of this paper describe the preliminary design of an advanced, sodium-cooled, graphite-moderated nuclear power plant sized to produce a net output of 256 Mwe. Steam is generated at 2400 psig, superheated to 1050°F, and reheated to 100°F after partial expansion in the turbine. Net plant thermal efficiency is 42.2%. Estimated cost of this plant is $160-190/kw, resulting in an estimated energy cost of approximately 6 mills/kwh.

Work which followed completion of this preliminary design has shown that a reduction of 60% in auxiliary power requirements can be attained by (1) using mechanical pumps in both primary and secondary sodium loops, and (2) using steam-driven instead of motor-driven boiler feed pumps. Net output of the plant would thus increase to 265 Mwe, and the thermal efficiency would reach approximately 43%.

Some additional development work is required before design of the nominal 250-Mwe ASGR can be finalized, and this is summarized at the conclusion of this paper.

The reactor is thermal in neutron spectrum, fueled with uranium carbide, cooled by sodium, and moderated with graphite. A calandria vessel contains the graphite moderator and reflector, and is pierced by through-tubes (process tubes) which contain the fuel elements. The calandria itself is located in a tank of sodium.

Superheated steam at 2400 psig and 1050°F is produced in the steam generators and flows to a common header which serves a single three-cylinder turbine. A reheat-regenerative cycle is utilized; the steam is partially expanded through the high-pressure turbine cylinder and returned to a sodium reheater. Steam is resuperheated to 1000°F in this reheater and expanded again through the lower pressure turbine stages. Steam is extracted at seven stages in the turbine for regenerative feedwater heating.

A sodium purification system is used to fill, drain, and flush the sodium heat transfer system. Sodium is noncorrosive if the oxide content is kept low. The purification unit removes sodium oxide to a point (<10 ppm) at which corrosion is negligible. An inert atmosphere is used within the reactor container and

at all points where a free surface of sodium exists. Waste disposal systems are provided to remove, store, and dispose of radioactive wastes accumulated in the equipment cleaning facilities and personnel decontamination room. A water cooling system provides auxiliary cooling to the concrete biological shielding and to certain sodium handling components.

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

The hypothetical site is a level area adjacent to a river in a temperate zone with highway, railraod, and river barge services available. The reactor, reactor service systems, fuel handling system, and control and office areas are enclosed by a single structure; the turbine-generator is located outdoors. The reactor and radioactive portions of the systems are contained in shielded cells. The reactor bay of the building is serviced by the gantry crane used for both fuel handling and reactor equipment installation and maintenance. A 75-ton bridge crane services the reactor auxiliary bay and steam plant. The high-voltage switchyard, cooling water intake and discharge structures, and liquid-waste treatment system are remote from the building.

General plant arrangement is shown in plan and vertical sections in Figures 5 and 6 respectively. A summary of Plant Data is presented in the following table.

PLANT DATA

NOMINAL 250-MWE ASGR

Gross Electrical Power	270 Mw
Net Electrical Power	255.9 Mw
Main Steam Pressure	2400 psig
Main Steam Temperature	1050°F
Reheat Steam Pressure	504 psig
Reheat Steam Temperature	1000°F
Throttle Steam Flow	1.835×10^6 lb/hr
Number of Feedwater Heaters	7 including a deaerator
Feedwater Temperature	540°F
Condenser Back Pressure	1-1/2-in. Hg absolute
Gross Turbine-Cycle Heat Rate	7550 Btu/kwh

520 012

PLANT DATA (continued)

Net Turbine-Cycle Heat Rate	7750 Btu/kwh
Steam-Raising Plant Efficiency	98.5%
Net Plant Heat Rate	8-10 Btu/kwh
Reactor	
Reactor thermal power	606 Mw
Reactor coolant	Sodium
Coolant temperatures	650°F inlet; 1200°F outlet
Coolant flow	12.6×10^6 lb/hr
Maximum velocity in core	23.6 ft/sec
Core pressure drop	27 psi
Number of primary loops	4
Secondary coolant	Sodium
Coolant temperature	600°F inlet; 1150°F outlet
Coolant flow	12.6×10^6 lb/hr
Fuel	
Material	Uranium carbide
Configuration	1/2-in. dia x 1-in. UC pellets
Jackets	Stainless steel tubes
Tubes per element	19-rod cluster
Number of fuel elements	151
Initial enrichment	2.75 at. % U^{235}
Initial excess reactivity with equilibrium Xe and Sm poison	0.127
Initial conversion ratio	0.52
Average fuel burnup	17,100Mwd/MTU*
Uranium in core	16,200 kg
Number of control-safety rods	18
Active core size	12-ft dia x 12-ft high
Reactor vessel size	17-ft dia x 31-ft high
Peak-to-average power ratio	2.87
Specific power (average for new core)	1,400 kw/kg U^{235}
Maximum fuel temperature	2250°F
Average heat flux	425,000 Btu/hr-ft^2
Average thermal neutron flux in fuel	8×10^{13} n/cm^2-sec

A. REACTOR

1. Reactor Core

The reactor core and reflector are contained in a calandria tank as indicated in Figure 7. The calandria tank is set in a stainless steel reactor vessel 17 ft in diameter by 31 ft high. The vessel is sealed by a concrete shield plug

*Mwd/MTU = megawatt days per metric ton of uranium.

520 013

similar to that used in the SRE Designed for an operating pressure of only a few pounds above atmospheric, the reactor vessel has a wall thickness of approximately 1-1/2 in. and is provided with a flat bottom head.

The moderator and reflector are graphite in the form of vertical logs of hexagonal cross section. In the active section of the core, each graphite log is pierced by a vertical process channel. These channels contain the fuel elements, the control elements, and the neutron source.

Primary sodium enters the calandria bottom plenum through inlet nozzles which penetrate the reactor vessel wall. After passing up through the process channels, the coolant enters the hot sodium pool above and surrounding the calandria. It then exits through nozzles in the upper part of the reactor vessel.

2. Calandria Vessel

The calandria vessel which contains the moderator and reflector logs is 16.5 ft in diameter by 19 ft high. The dished-top head and vessel wall are 3/4 in. thick, and the bottom is 1 in. thick. The top head and vessel bottom each contain 187 nozzles on a 10-in. triangular spacing. Two additional nozzles for vent and pumpout lines are located in the top head. The vessel, inlet plenum, and process channels are made of type 304 stainless steel and are of all-welded construction. A small-size mockup of a similar calandria vessel has been fabricated. Model tests on similar calandria tank-head configurations have established not only the practicability but the conservatism of this design.

The graphite moderator logs are hexagonal in cross section with holes centered axially for the process channels. Pedestals, which are integral parts of the bottom nozzles, support the individual graphite logs at the bottom, and each log is located at the top by the top nozzles. All graphite logs are 9.91 in. across the flats and 16 ft in length, machined from crucible-grade graphite.

The graphite reflector logs are identical to the moderator logs except that the center hole is omitted. These logs are located and supported at the bottom on dummy nozzles and pinned at the top to each other and to the adjacent moderator logs.

Each process channel consists of a lower nozzle, process tube, top nozzle, bellows, bellows guard, and entry guide. The process tube is 3.33 in.

520 014

of ID and has a 0.030-in.-thick wall. This tube extends from the lower nozzle through the top nozzle and is welded to a bellows. The bellows are provided to allow for differential thermal expansion between the calandria vessel and the process tubes. Guards surrounding the bellows are used to stagnate sodium and reduce thermal shocks. An entry guide extends above the process tube and provides a channel for inserting the orifice tube (for fuel element channels) or a control rod thimble. The orifice tube is a loose-fit sleeve within the top of each process tube and extends upward to the top shield, acting as a guide for fuel element insertion and as the mobile member of the variable orifices. Of the 187 process channels, 168 are for fuel or dummy fuel elements. The calculated loading is 151 elements. One channel is for the neutron source, and the remaining 18 are for combination control-and-safety elements.

An inlet coolant plenum below the calandria vessel provides coolant flow distribution through the 168 fuel channels (including 17 spares). This plenum is a short, cylindrical tank, 13 ft in diameter with wall and bottom of 1-in.-thick stainless steel, and is welded to the calandria vessel. Sodium within the inlet plenum is under a pressure of 40 psig. To carry this pressure load, the bottom of the inlet plenum and bottom of the calandria vessel are tied together by 1-in.-diameter rods on a 20-in. triangular spacing.

3. Fuel Element Assembly and Orifice Tube

The fuel element is a cluster of 19 fuel rods suspended vertically in a process tube from a plug in the top shield. The cluster is supported at the top by a ring; each rod is allowed to expand downward independently of the others. Thin sheet-metal spacers, located every 12 in. along the length of the fuel rods, separate the individual rods and provide lateral support. Three equally spaced wires, which run the length of the rods, tie the metal spacers together, preventing them from sliding out of position.

The fuel rods are 1/2-in.-diameter UC slugs sheathed in 0.52-in.-ID by 0.010-in.-wall stainless steel jackets. The rods are approximately 14 ft long with the lower 12 ft containing the fuel slugs, and the remaining 2 ft providing space for fission gases. A sodium bond is used between the UC slugs and the jacket. End caps welded to the jackets seal the fuel slugs and sodium under an inert gas.

520 015

The fuel element is suspended in the process channel by a hanger rod. This rod is welded to the bottom of the shield plug and extends down to the fuel element. A disconnect near the bottom of the rod allows the fuel handling cask to divide the fuel element assembly (overall length 48 ft) into two equal segments, thereby reducing the height of the fuel handling cask. Since the fuel element assembly is always inside the close-fitting orifice tube and process channel, a simple sliding tongue-and-groove-type disconnect is employed.

An orifice tube extends from the floor level down through the loading face shield and into each of the process-channel entrance guides. Orificing of coolant flow is accomplished by a rectangular opening in the orifice tube which is located vertically to match a triangular opening in the process-channel entrance guide. Effective orifice area, and therefore coolant flow, is varied by rotating the orifice tube by means of a spanner wrench from the reactor floor.

4. Control-Safety Rod Assembly

Combination control-and-safety rods operate in thin-walled stainless steel thimbles which extend from the top of the loading face down through the core. These thimbles are positioned in process channels and cooled by natural convection. The poison rings are suspended on a steel pull tube and moved vertically by a drive mechanism located in the upper portion of the thimble.

The drive mechanism incorporates a collapsible rollernut and translating screw whcih imparts linear motion to the poison pull tube. The mechanism is rotated for normal movement of the control rod by a low-frequency a-c reluctance motor. To provide emergency shutdown of the reactor, negative reactivity is quickly inserted by de-energizing the motor, allowing the rollernut to collapse and the control rod to fall into the core under the force of gravity. Four hydraulic snubbers are provided to absorb kinetic energy of the falling rod.

The entire drive mechanism, with the exception of an upper thimble, is located below the reactor loading face in order to facilitate refueling. Since the drive screw is withdrawn above the loading face when the reactor is in operation, the upper thimble is provided to contain the mechanism and poison column in a helium atmosphere. For refueling, the upper thimble is readily removed.

520 016

Since the reactor has a small isothermal temperature coefficient of reactivity, only 18 control-safety elements are required. Stability is assured by reactor prompt negative fuel-temperature coefficient as demonstrated by operation of the SRE.

5. Miscellaneous Core Elements

The miscellaneous core elements include a startup neutron source, dummy elements, sodium level and temperature indicators, and neutron instrumentation. These elements are suspended from shield plugs and can be used in any channel in the reactor.

6. Reactor Physics

The core contains 151 fuel elements with the fuel enriched to 2.75 at.% U^{235}. This configuration gives a calculated hot, clean k_{eff} of 1.112 and an initial conversion ratio of 0.52. The average fuel exposure attainable with this initial reactivity is about 17,000 Mwd/T (average exposure of elements removed).

7. Fuel Handling Equipment

The fuel handling cask is approximately 35 ft high and weighs about 60 tons. It is carried on a gantry crane which travels over the reactor on rails embedded in the reactor floor. The cask is designed to divide the fuel element from the shield plug and hanger rod. This feature permits the cask height to be held to a minimum. The shield plug and hanger rod are re-used with a new fuel element. The cask consists primarily of lead shielding, a rotating internal cylinder, grapple and hoisting equipment, a gas cooling system and a gas lock.

A forced-convection gas cooling system is provided in the cask to cool the hot fuel element. A periscope viewer is provided to view fuel element assemblies during withdrawal.

B. SODIUM HEAT TRANSFER SYSTEMS

1. General Description

Primary sodium circulates through the reactor and leaves the reactor vessel at 1200°F. Sodium pumps taking suction from the reactor deliver the sodium to an intermediate heat exchanger which transfers the heat to the secondary system. Primary sodium leaves the heat exchanger at 650°F and returns to

520 017

the core. There are four primary heat transfer loops. Nominal sodium flow in each primary loop is 3.15×10^6 lb/hr.

The secondary sodium system consists of four loops of 16-in. piping constructed of type 304 stainless steel. Sodium is pumped through the intermediate heat exchanger to the steam generator and back in a closed circuit. The intermediate-heat-exchanger outlet temperature is 1150°F, and the inlet temperature is 600°F. It enters the superheater of the steam generator at 1145°F and leaves the evaporator-preheater section of the steam generator at 600°F.

A portion of the sodium flow is diverted from each loop to the steam resuperheaters, and the sodium from the reheaters rejoins the main stream at the superheater-evaporator junction.

Reactor power is controlled automatically over a load range of from 1/4 to full load. Manual control is provided for ratings from startup to full power.

2. Sodium Pumps

Primary and secondary sodium pumps are of the vertical centrifugal free-surface type. Both primary and secondary pumps are installed in the primary-heat-exchanger cells, with motors above the reactor room floor. Pumps are designed so that the internal mechanism can be removed without removing pump case or piping. Pump capacity is 7650 gpm, and one pump is installed in each loop. Each pump is driven by a wound-rotor motor operating on a 3-phase, 60-cycle, 4160-volt power supply. Pumps are of type 304 stainless steel construction.

3. Intermediate Heat Exchangers

The intermediate, sodium-to-sodium heat exchangers separate the primary and secondary systems. There are two parallel sodium-sodium heat exchangers per loop. Each unit is of shell-and-tube, counter-flow design with an expansion bellows in the shell. Primary sodium flows through the tubes, and secondary, nonradioactive sodium flows through the shell. Design pressure drops are 10 psi both on tube and shell sides. The units are fabricated entirely of type 304 stainless steel and are of all-welded construction. Additional details are shown in Figure 8.

520 018

4. Steam Generator

Four steam generators are provided, one per secondary loop. When provided with feedwater at a temperature of 540°F, the generators are capable of producing 1,835,000 lb/hr of steam at 2450 psig and 1055°F.

The steam generators are once-through, shell-and-tube exchangers, with sodium on the shell side and water and steam on the tube side. The units are installed vertically and are supported and braced by an enclosing structure which also receives all piping reaction loading from boiler feed, main steam, and other interconnecting piping systems.

Each steam generator consists of an evaporator-preheater unit, a separating drum, and a superheater unit. The separating drum is used to eliminate carryover of solids, as well as to assure delivery of dry saturated steam to the superheater. Several rupture discs are installed on the shells of the evaporator and the superheater in such positions that there can be no possible entrapment of a pressure surge that would cause permanent damage to the unit. The rupture-disc discharges are connected to a sump tank which can be disconnected, scrapped, and replaced in event of tube rupture, eliminating any costly cleanup problems.

The steam generators are of all-welded construction and fabricated of chrome-moly steels. Bellows expansion joints are used on the shells, as shown in Figure 9.

5. Resuperheater

Two steam resuperheaters are provided for the plant. The units are designed to resuperheat steam from 504 psig at 650°F to 454 psig at 1000°F. Steam is then returned to the intermediate pressure cylinder of the turbines and expanded to exhaust. The resuperheater is also a once-through, shell-and-tube exchanger with sodium on the shell side and steam on the tube side. Sodium flow of each unit is in parallel with that of the superheater and is returned to the superheater at the superheater-evaporator junction.

The unit is similar in design and construction to the evaporator, except for the tube-side design pressure of 600 psig. Fabrication is of chrome-moly steel with all-welded construction. Rupture discs are provided as for the steam generators.

C. ELECTRICAL GENERATING PLANT

The turbine is a tandem-compound, three-cylinder, reheat, condensing unit. Throttle operating conditions are 2400 psig and 1050°F with 1000°F reheat. The high-, intermediate-, and low-pressure sections of the turbine operate on the same shaft at 3600 rpm. The turbine generator has a capability of 265,000 kw at 30-psig hydrogen pressure in the generator and 1-1/2 in. Hg. back pressure. Plant heat balance is presented in Figure 10.

The seven feedwater heaters comprise two low-pressure heaters, a direct-contact deaerating heater, and four high-pressure heaters. The high-pressure heaters are equipped with desuperheating and drain cooler sections, the low-pressure heaters with drain cooler sections.

The generator rating is 316,000 kva at 0.85 power factor, and voltage is 23.0 kv. The generator is designed for outdoor operation and is complete with hydrogen coolers, lube-oil coolers, lube-oil reservoir, turning gear, and a-c, d-c, and shaft-driven lube-oil pumps.

D. PLANT CONTROL SYSTEM

A plant control system allows automatic operation over the range of 20 to 100% load, or manual operation at any load. The plant is designed to operate on a load-following basis. Feedwater flow is controlled by a three-element controller with feedwater flow, steam flow, and sodium outlet temperature from the steam generator as controller variables. Steam pressure ahead of the turbine throttle is used to control secondary sodium flow through the steam generators, thereby controlling the rate of steam generation. Bulk temperature of the coolant out of the reactor is used to control reactor power by positioning control rods. A plant protective system provides rapid detection of any off-normal condition in the reactor plant and automatically takes corrective action. This action may involve an alarm, setback or scram. All power plant control is from a central area which integrates both reactor and turbine-generator instrumentation. Monitoring systems, with alarms to indicate radiation levels in excess of predetermined values, are provided.

520 020

E. ECONOMICS OF 256-MWE ADVANCED SGR

Improved performance of the ASGR, in terms of reduced sodium flow and improved thermal efficiency, should result in a construction cost lower than that estimated for the current-status SGR. Although no detailed estimate of plant construction cost has been completed for the 256-Mwe ASGR, it appears quite probable that this cost, based on present-day prices, will fall within the range of $160-190/kw, exclusive of the cost of the first core.

Before fuel cost of the ASGR plant can be predicted with accuracy, it will be necessary to obtain further verification of the performance and cost data of the UC fuel materials. Capsule tests have already been conducted to obtain irradiation data, although the temperatures reached were lower than those anticipated for the ASGR. These capsules have each contained several specimens of UC. Burnup has varied up to approximately 12,000 Mwd/MTU and center temperatures have ranged up to about 1500°F. Results of these tests have been very encouraging. Although some cracking of the carbide does occur, there has been no crumbling, and the heat transfer characteristics have not been affected appreciably. Fission-gas retention is excellent; the measured release is approximately that which is expected from surface recoil. Many additional capsules are currently being irradiated or are under preparation, with burnups to 25,000 Mwd/MTU and center temperatures above 2000°F scheduled.

Based on the cost and performance data obtained to date, the estimated fuel cost has been calculated using the following assumptions:

Average heat flux	425,000 Btu/hr-ft^2
Maximum center temperature	2250°F (1232°C)
Average exposure	17,000 Mwd/MTU
Maximum exposure	26,000 Mwd/MTU
Average core life at an assumed 80% plant factor	19 months
Out-of-core recycle period	15 months
Processing and fabrication cost (UF_6 to fuel element)	$65/kg
Inventory use-charge	4% year
Net plutonium credit	$12/gram (less losses and recovery cost).

These assumptions lead to an estimated net fuel cycle cost of 1.7 mills/kwh, and a total power cost of approximately 6 mills/kwh for an 80% plant factor. The

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fuel cycle cost is not particularly sensitive to variations in fuel performance. For example, should it become necessary to reduce the maximum center temperature to a figure as low as 1600°F to attain desired fuel burnup, the fuel cycle cost would increase less than 0.1 mill/kwh.

In view of the low capital and operating costs estimated for an ASGR of this size, it is predicted that this reactor will be found to compete favorably with conventional fossil-fueled power plants in most industrialized areas of the world.

F. DEVELOPMENT PROGRAM FOR ASGR

Most components of the ASGR plant are within the capability of current technology. The major exception is the fuel material, uranium monocarbide. Results of preliminary research and development work currently in progress indicate that UC will be an excellent fuel for high-temperature reactors, but temperature and burnup limitations have yet to be firmly established. Additional development work is also required on the steam generators, which are the single-barrier type which will be used in the Enrico Fermi Fast Breeder Reactor plant.

To obtain the additional test information required prior to completing the design of a full-scale ASGR power plant, and Advanced Sodium Reactor Experiment (ASRE) has been suggested. Such a plant would have these major objectives:

1) Establishment of UC fuel elements to obtain firm data for predicting fuel costs.

2) Demonstration of the ease of operation and maintenance of a higher temperature sodium heat transfer system.

3) Operation as a high-temperature, high-flux fuel testing facility.

In addition, an advanced test facility would (1) demonstrate feasibility of a primary cooling system closely coupled to the reactor, (2) operate with the advanced calandria core concept, (3) establish fuel-handling techniques for elements of high power densities, and (4) demonstrate improved thermal efficiency with sustained production of steam at 2400 psi and 1050°F.

Preliminary designs of the proposed ASRE have been made, and construction of this plant is now under consideration.

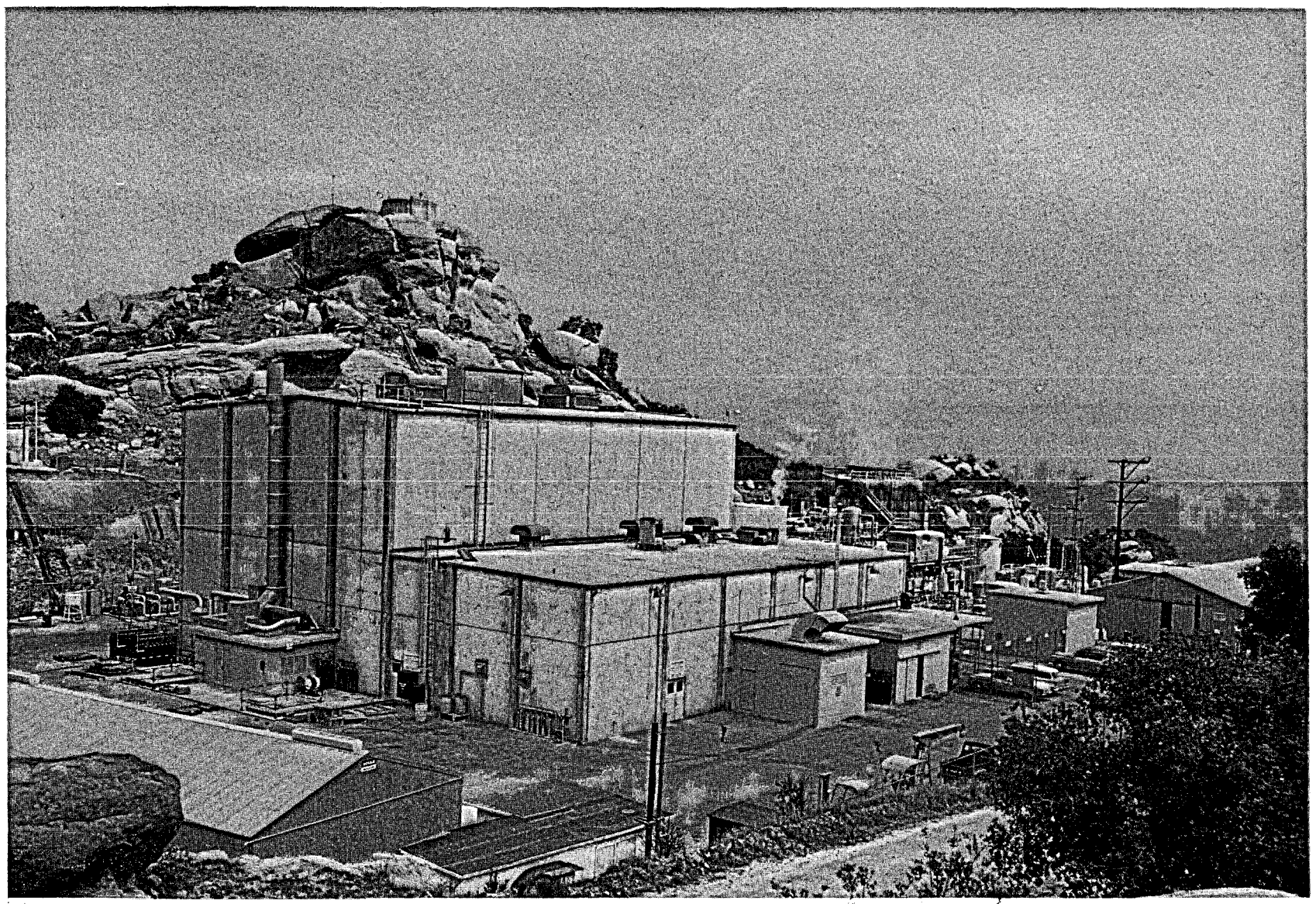

Figure 1. Sodium Reactor Experiment (SRE) in Full-Power Operation

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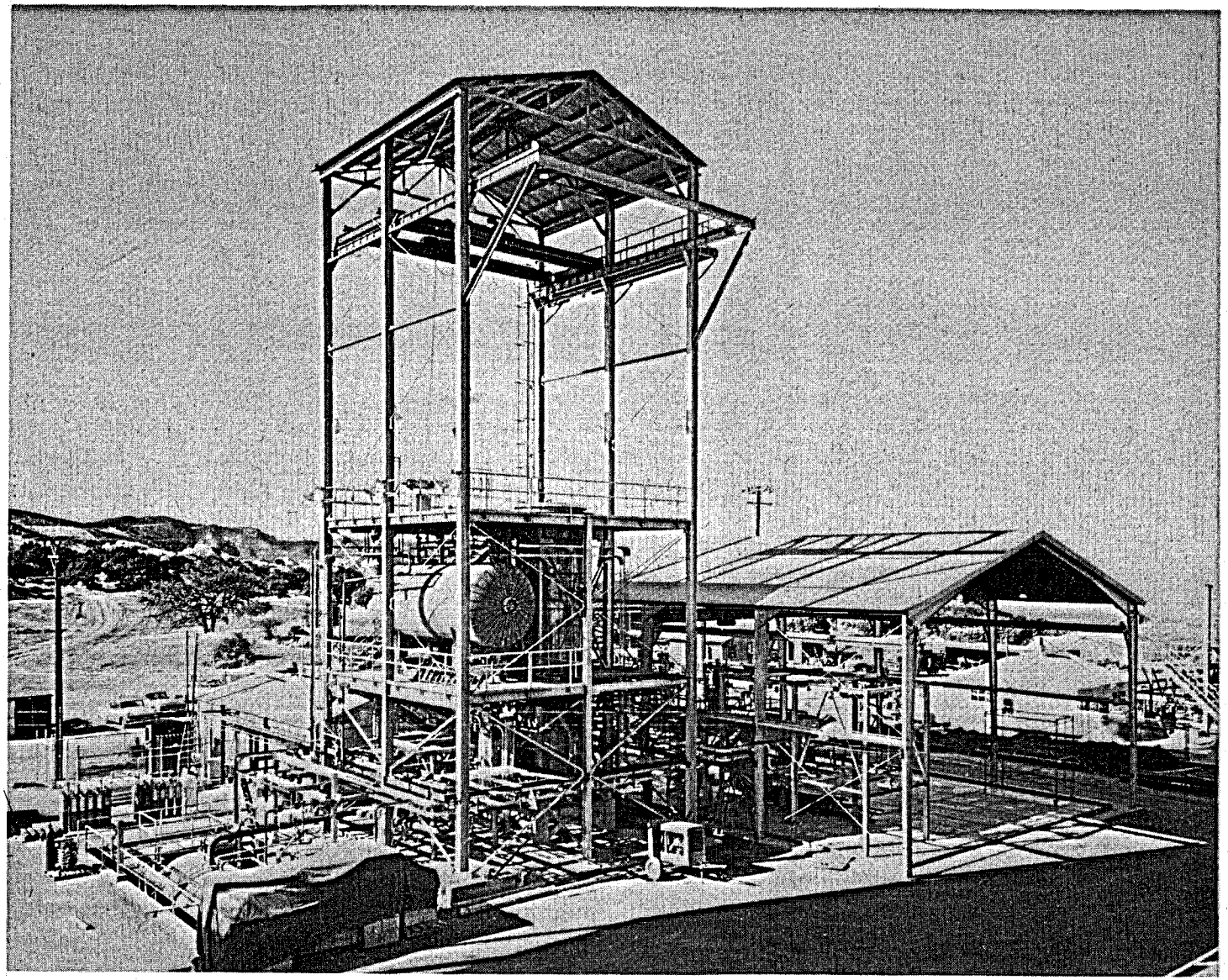

Figure 2. Large Component Test Loop (LCTL) for Environmental Testing of Full-Scale Sodium System Components

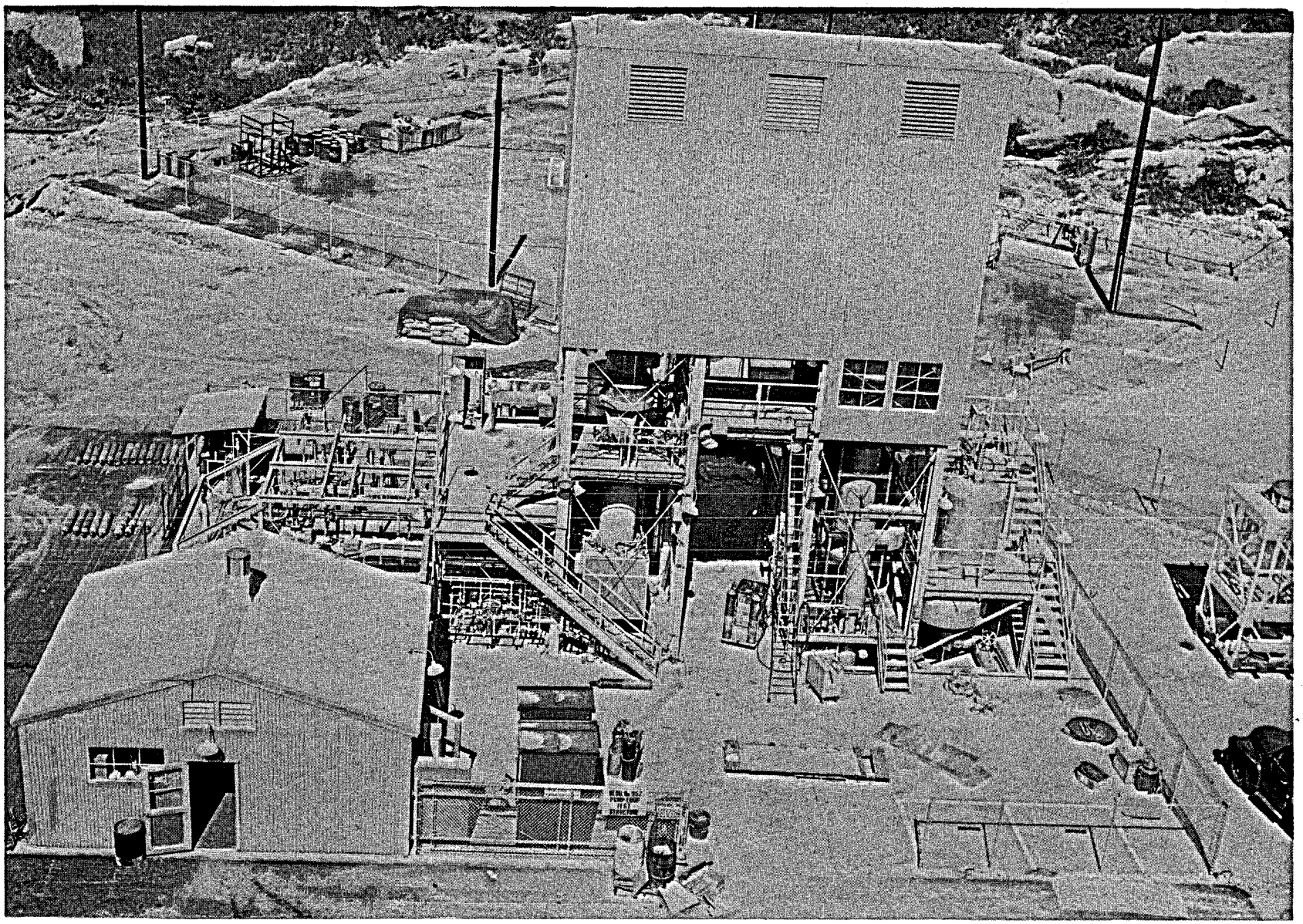

Figure 3. Hallam Pump Test Facility With Two Types of Sodium Pumps Under Test

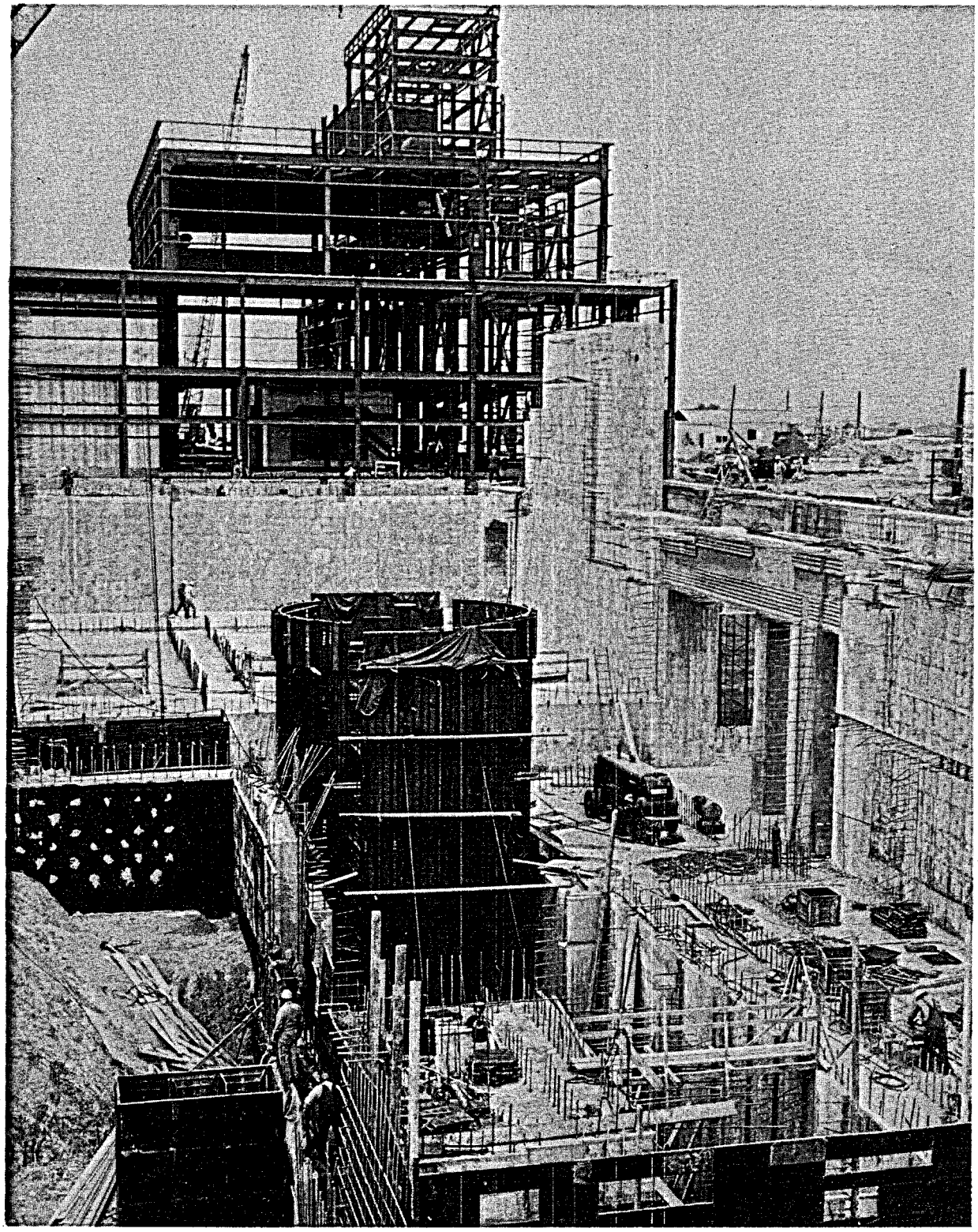

Figure 4A. Construction Progress of the Hallam Nuclear Power Facility (HNPF) - October 1959

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Figure 4B. Construction Progress of the Hallam Nuclear Power Facility (HNPF) - February 1960

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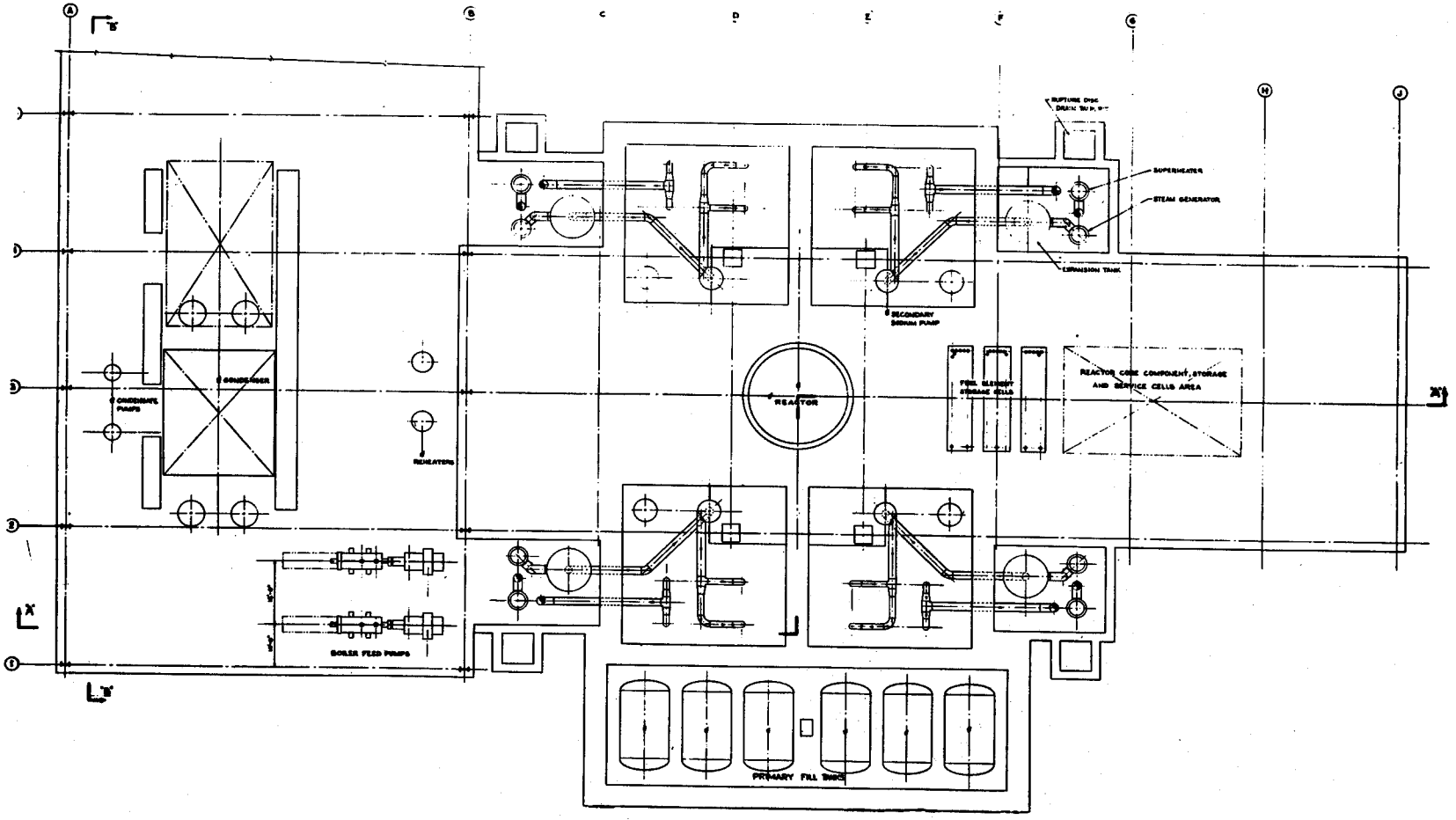
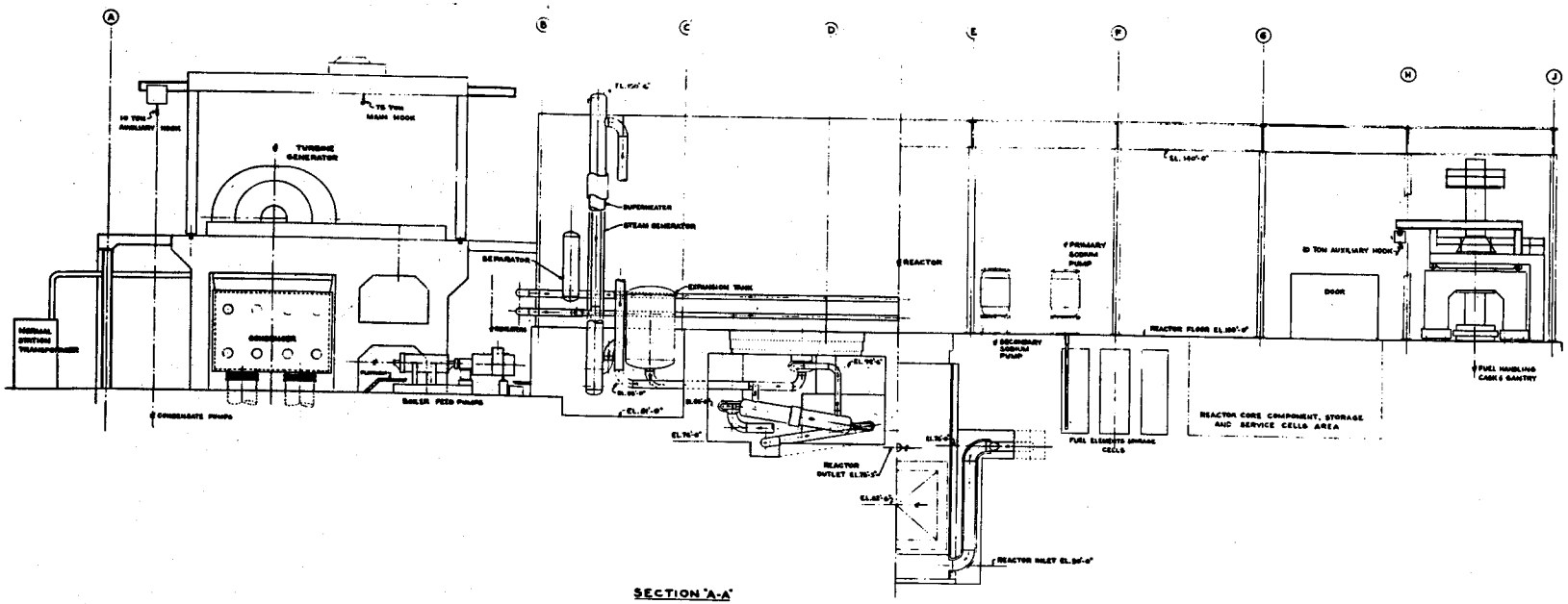
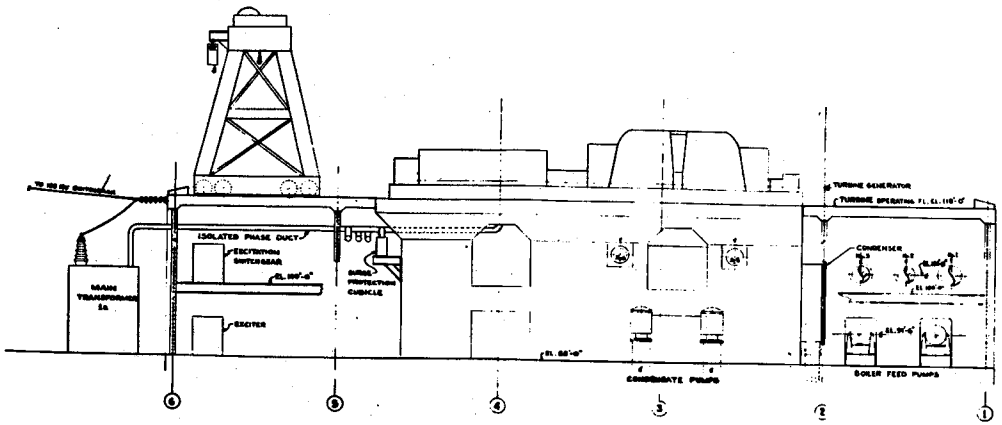


Figure 5. Plant General Arrangement (plan elevation 95'0"



SECTION 'A-A'



SECTION 'B-B'

Figure 6. Plant General Arrangement (sections)

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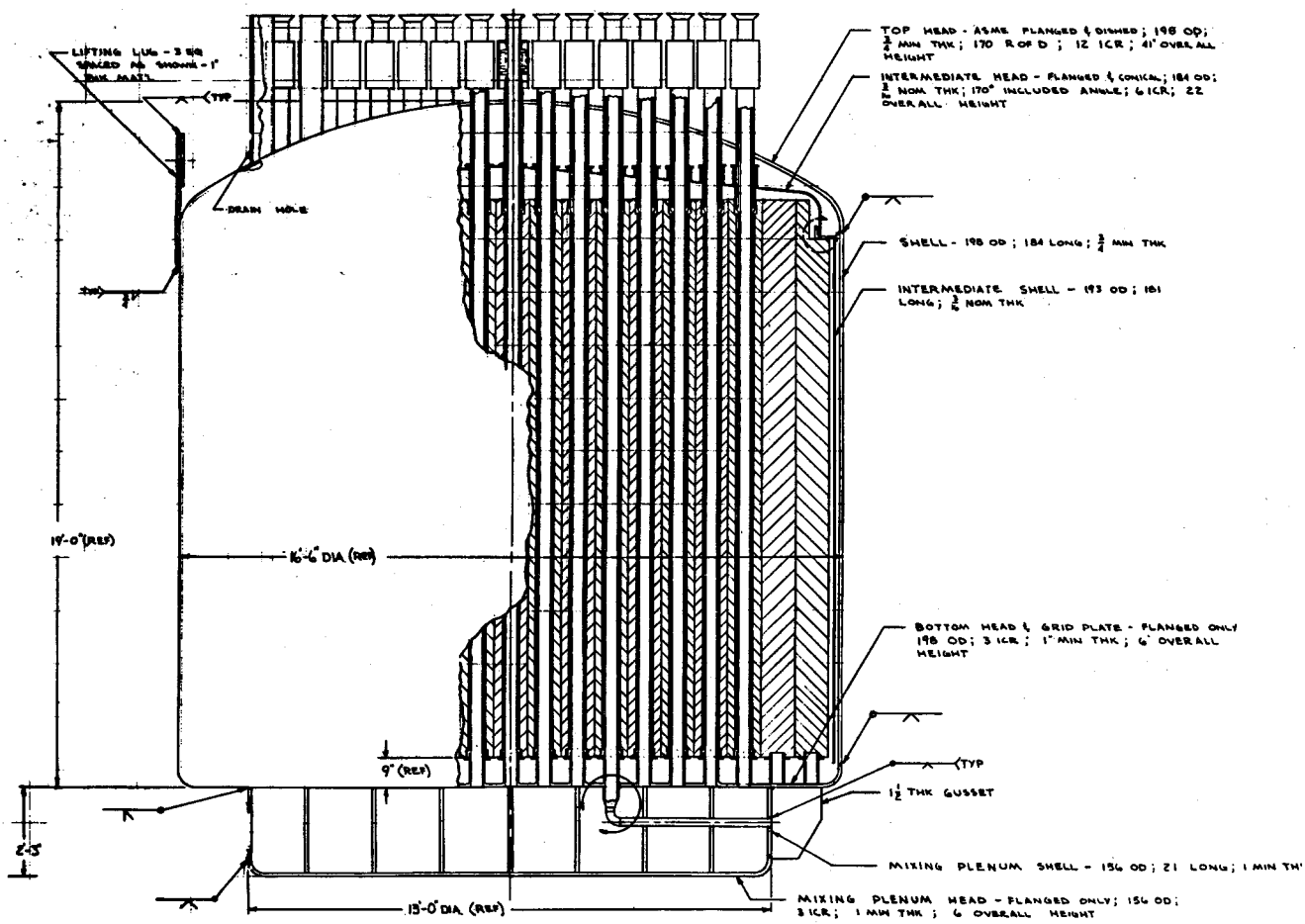

Figure 7. Calandria

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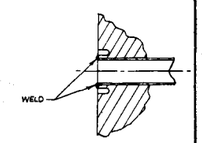
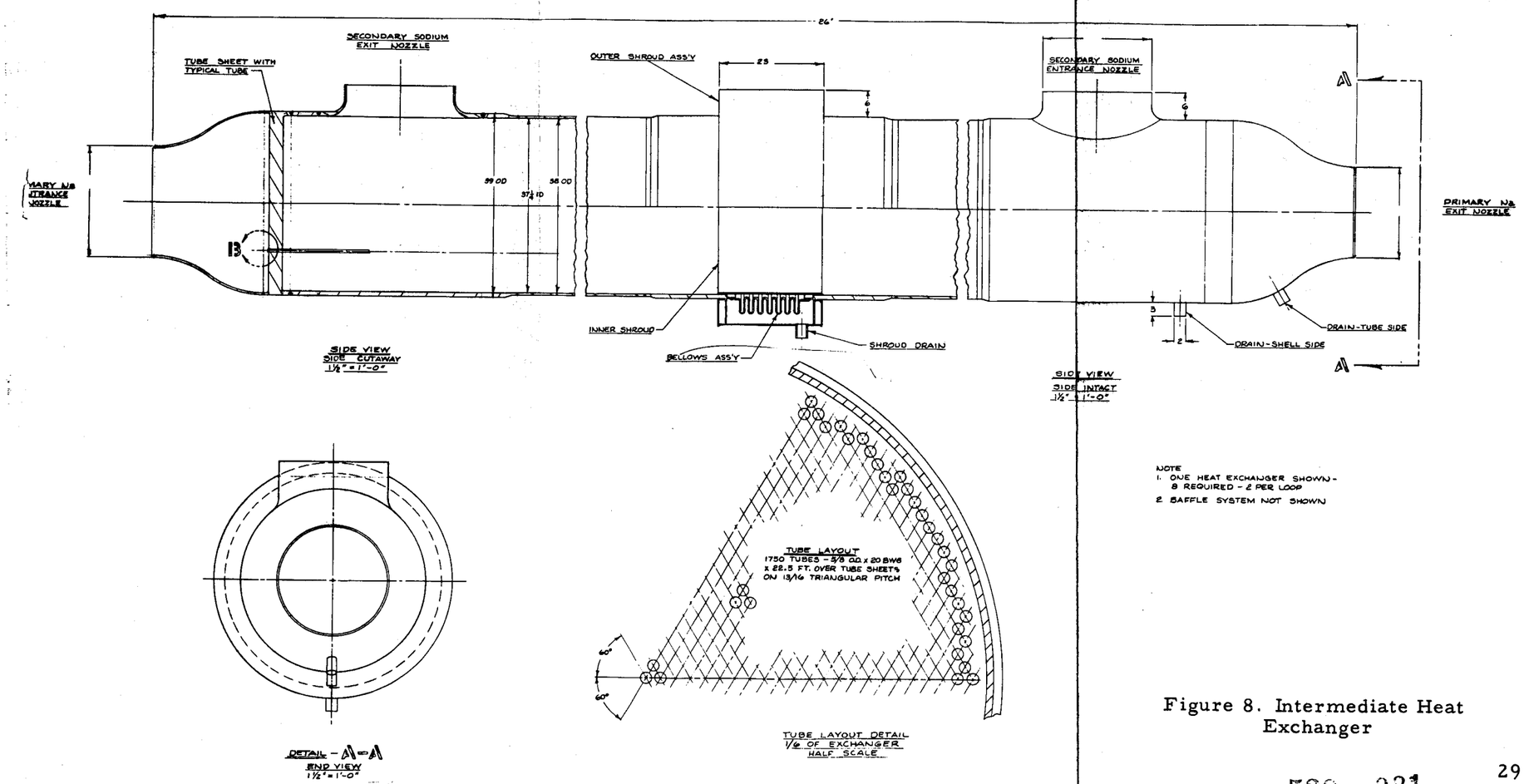
DESIGN & OPERATING DATA FOR ONE OF EIGHT EXCHANGERS			
		OPERATING	DESIGN
FLOW - PRIMARY SODIUM (TUBE SIDE)	LB/HR		
FLOW - SECONDARY Na (SHELL SIDE)	LB/HR		
TEMP. (IN-OUT) PRIMARY SODIUM	°F		
TEMP (IN-OUT) SECONDARY SODIUM	°F		
TUBE SIDE PRESSURE	PSIG	—	150
SHELL SIDE PRESSURE	PSIG	—	150
TUBE SIDE PRESSURE DROP	PSIG	—	10
SHELL SIDE PRESSURE DROP	PSIG	—	10
TRANSFER SURFACE (APPROX)	SQ. FT.	—	
MATERIALS - TUBE SIDE	TYPE	—	304 SS
MATERIALS - SHELL SIDE	TYPE	—	304 SS
CODE (SEE ALSO ADD'L REQ'MTS)	ASME	—	SECT. 8

ADDITIONAL REQUIREMENTS: ATOMICS INTERNATIONAL SPECIFICATIONS ON; CLEANING, HELIUM LEAK DETECTION AND ULTRASONIC INSPECTION OF TUBES.



DETAIL - B
FULL SIZE

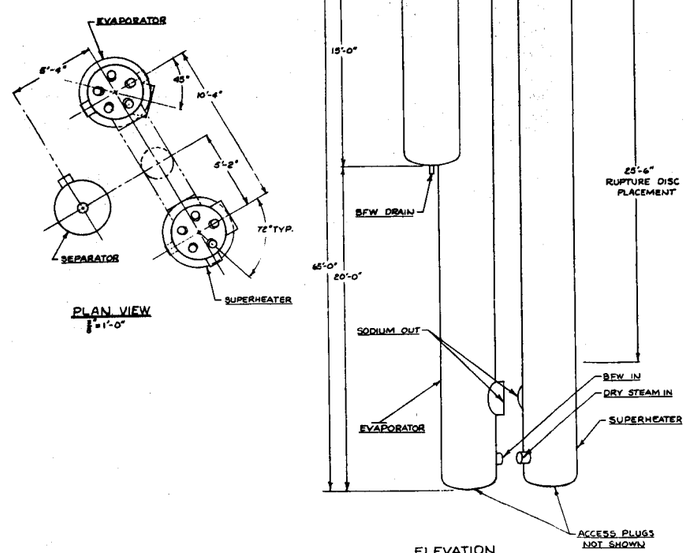


NOTE
1. ONE HEAT EXCHANGER SHOWN - 8 REQUIRED - 2 PER LOOP
2. BAFFLE SYSTEM NOT SHOWN

Figure 8. Intermediate Heat Exchanger

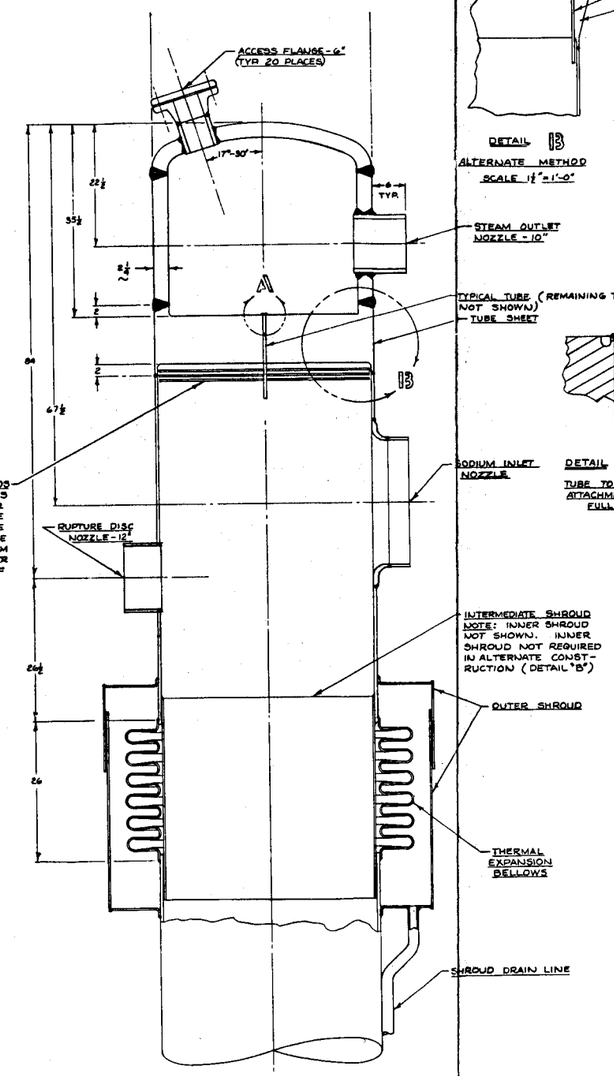
DESIGN & OPERATING DATA (100% FLOW)					
		EVAP. WITH PREHEAT		SUPERHEATER	
		ONE OF FOUR UNITS		ONE OF FOUR UNITS	
		OPERATING	DESIGN	OPERATING	DESIGN
FLOW - SODIUM	LB/HR				
FLOW - WATER-STEAM	LB/HR				
TEMP. - IN & OUT - SODIUM	°F				
TEMP. - IN & OUT - WATER-STEAM	°F				
TUBE SIDE PRESSURE	PSIG				
SHELL SIDE PRESSURE	PSIG		150		150
TUBE SIDE ΔP	PSIG		30 MAX.		40 MAX.
SHELL SIDE ΔP	PSIG		20 MAX.		15 MAX.
TRANSFER SURFACE APP.	SQ.FT.		9500		10 000
MATERIAL - TUBES	Cr-Mo		2¼-1		5-½
MATERIAL - SHELL	Cr-Mo		2½-1		5-½
CODE *	ASME		SECT. 8		SECT. 8

* NOTE: ADDITIONAL REQUIREMENTS ARE TO BE FURNISHED BY ATOMICS INTERNATIONAL IN RESPECT TO ULTRASONIC TUBING INSPECTION & SODIUM SIDE CLEANLINESS.



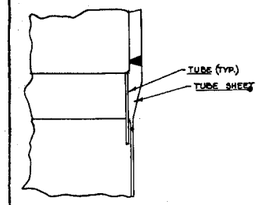
ELEVATION
SCALE: 1/8" = 1'-0"
STEAM GENERATOR

PLAN VIEW
1/8" = 1'-0"

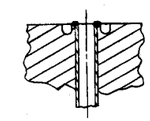


SUPERHEATER
MECHANICAL DETAILS

EVAPORATOR DETAILS SIMILAR. INVERT WITHOUT BELLOWS FOR BOTTOM END OF BOTH UNITS



DETAIL B
ALTERNATE METHOD
SCALE 1½" = 1'-0"



DETAIL A TYPICAL
TUBE TO TUBE SHEET ATTACHMENT DETAIL
FULL SCALE

STEAM GENERATOR DATA

SUPERHEATER:
TUBES - ½" OD x 16 BWG x 60' LONG - 5/8 Δ PITCH.

EVAPORATOR:
TUBES - 5/8" OD x 14 BWG x 60' LONG - 13/16 Δ PITCH.

ONE STEAM GENERATOR SHOWN (NOT INCLUDING REHEATER), FOUR REQUIRED.

Figure 9. 250-Mwe-ASGR Steam Generator

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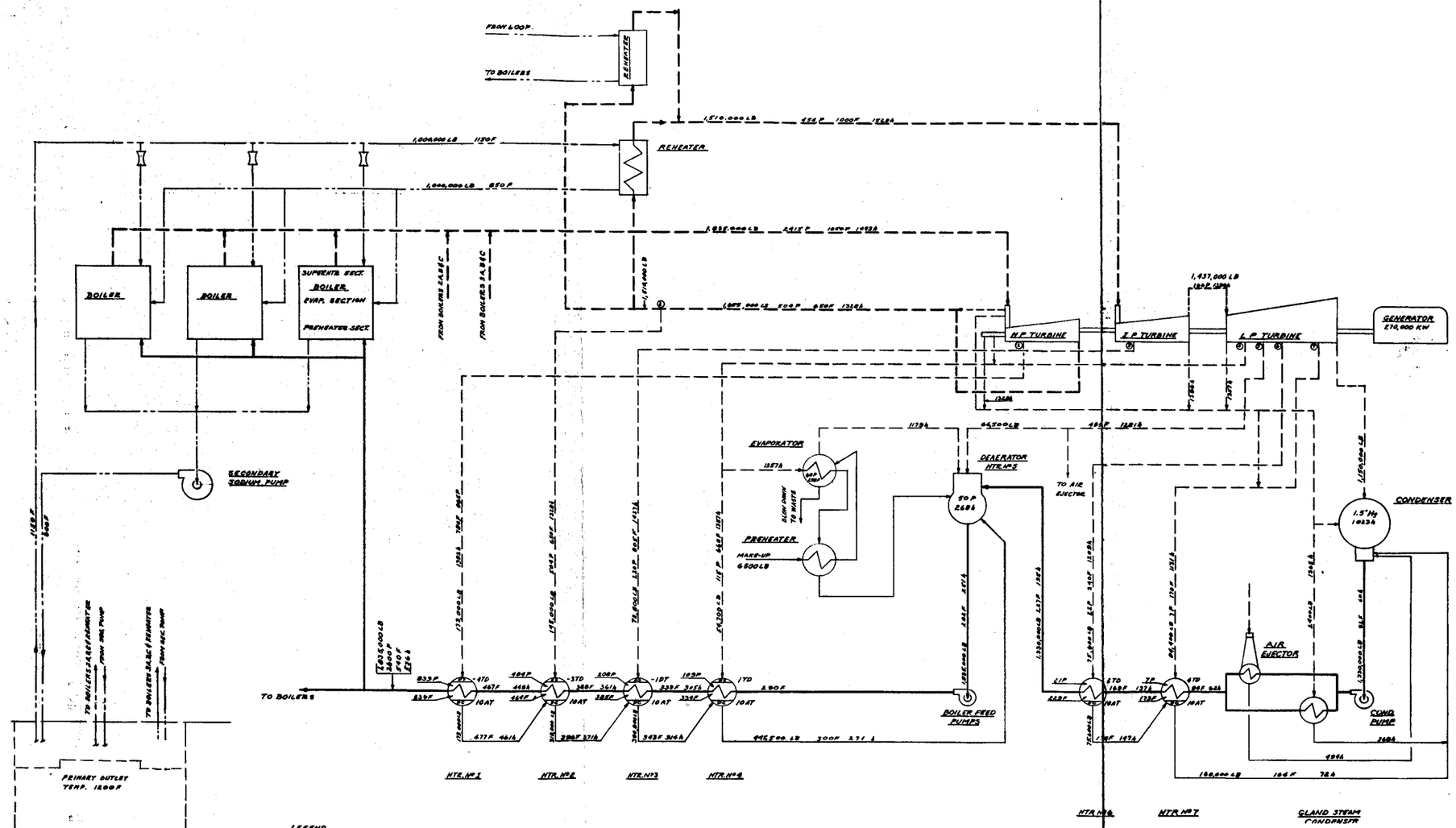

Figure 10. Preliminary Heat-Balance Diagram

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4. "SRE Decay-Heat Problem Solved by Eddy-Current Brake," by W. S. DeBear, Nucleonics, 17, No. 6, 1959, pp. 108-113.

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6. "Operating Experience and Advanced Designs of Sodium-Graphite Reactors," by R. W. Dickinson and L. E. Glasgow, ASME Publication No. 59-A-192.

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