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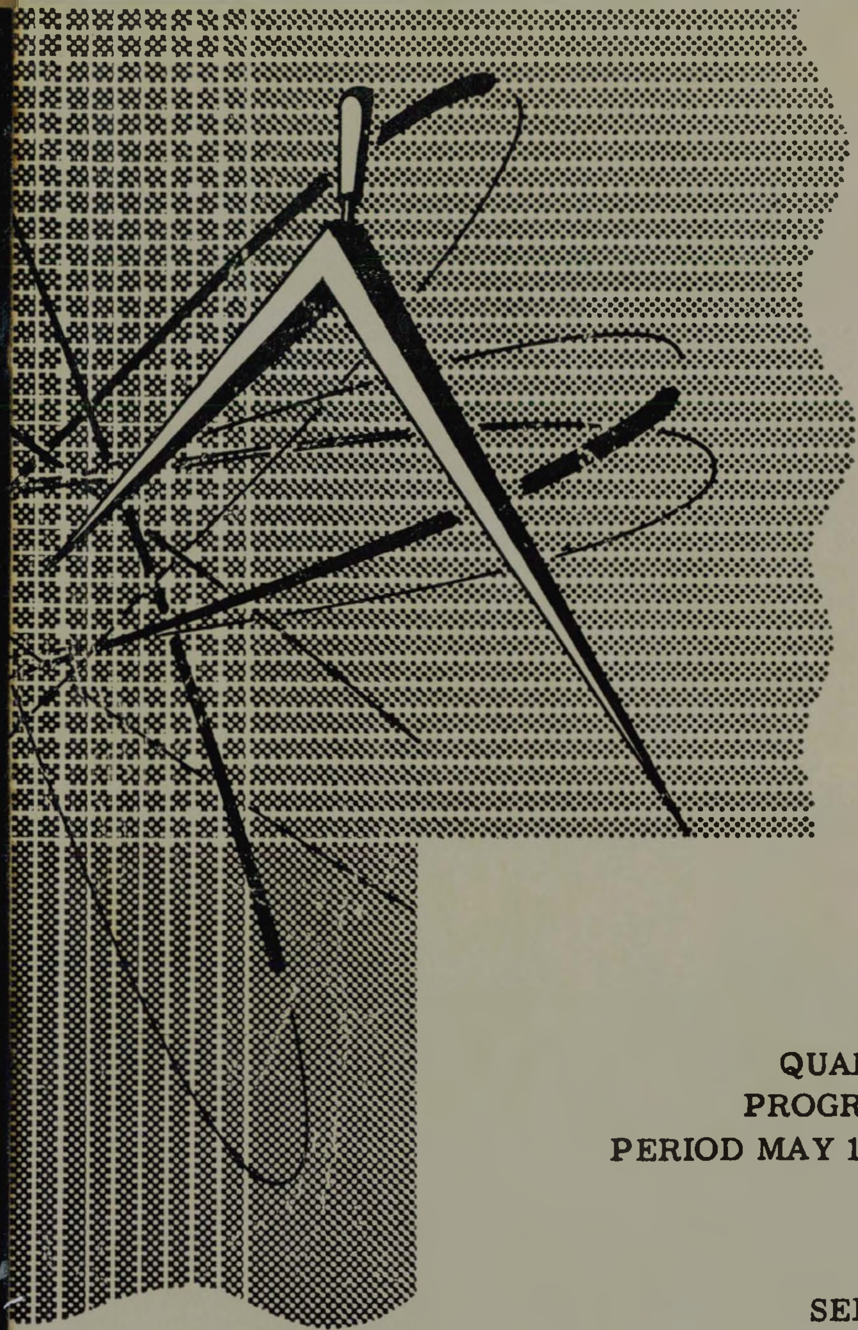

SDR PROJECT QUARTERLY TECHNICAL PROGRESS REPORT FOR THE PERIOD MAY 1, 1957 THROUGH JULY 31, 1957

SEPTEMBER 30, 1957

Work Performed under Contract AT(30-3)-256
for the United States Atomic Energy Commission

NDA -
NUCLEAR DEVELOPMENT CORPORATION OF AMERICA
WHITE PLAINS, NEW YORK

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Justin Kerr[?]

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INTRODUCTION

This report summarizes work performed during the period May 1, 1957 through July 31, 1957 by Nuclear Development Corporation of America for the U. S. Atomic Energy Commission under Contract AT(30-3)-256.

This is the first of a series of quarterly reports covering technical progress on the SDR program, which has been divided into three major areas of effort:

1. Technical Planning and Evaluation
2. Sodium-D_2O Separation
3. Preliminary Design

The latter two areas, in which lie the major technical efforts, have been divided into a number of "Tasks." This report discusses progress on each of the Tasks.

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SUMMARY

SODIUM-D_2O SEPARATION

During this report period, engineering studies have been made and experimental work has been carried out with the objective of demonstrating the feasibility of separation of sodium and D_2O in the SDR reactor.

For the engineering studies, a through-tube reactor design has been selected. Layouts of fuel-coolant tube and header arrangements have been made with a view towards minimizing the probability of mechanical failures of these sodium system components. Preliminary studies on sodium and D_2O system requirements and on the design of the D_2O moderator tank or "calandria" have been started. A survey has been made of possible barrier materials, i.e. materials which may be located between the fuel-coolant tubes and the calandria tubes to minimize the consequences of single or multiple-tube failures.* Mechanical arrangements for mounting and supporting barrier materials and for detecting leaks have been investigated.

Conceptual designs of equipment for testing the mechanical integrity of fuel-coolant tube and header joints have been completed. Major components of a single-failure rig, in which barrier materials can be subjected to sodium streams that simulate sodium system failures, have been designed, constructed, and assembled. Preliminary design of a multiple-failure test apparatus, in which the effects of both sodium and D_2O failures will be investigated, has been completed. Preliminary design of a mockup test apparatus has been finished; this apparatus, which will be a full scale representation of a section of the reactor, will be used to demonstrate the reliability of integrated sodium and water circulating systems under simulated normal and aggravated reactor operating conditions.

PRELIMINARY DESIGN

Several reactor arrangements have been compared on a preliminary basis; one, a through-tube design, has been used for more detailed engineering and nuclear studies. Design data for two variations of the through-tube design are given in this report. The shielding problems associated with radiation limitations on access to header rooms for repairs and maintenance have been evaluated. Temperatures, pressures, and flow rates for the secondary sodium and steam systems have been established.

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*It should be noted that the barrier problem in the SDR is significantly different from the barrier problem in sodium-steam generators since the barrier material for SDR could and should be a good heat insulator while good heat conduction is an essential requirement for the sodium-steam generator barrier.

SODIUM-D_2O SEPARATION

TASK 2-1 SODIUM SYSTEM ENGINEERING

FUEL-COOLANT TUBE AND HEADER DESIGN

A reactor arrangement which has been selected as the basis for preliminary design studies is described under Task 3-1, Reactor Preliminary Design and is shown in Figs. 2-1.1 and 2-1.2.* In this arrangement the D_2O moderator is contained in a calandria, consisting of a cylindrical aluminum tank, penetrated by vertical aluminum tubes, which are welded into the tank top and bottom. A stainless steel fuel-coolant tube containing a cluster of stainless steel clad fuel elements passes through each aluminum tube. The sodium primary coolant flows upward through the fuel-coolant tube. Sodium enters and leaves each coolant tube through individual tubes, or "pigtails," which provide flexibility to accommodate the thermal expansion of the fuel tubes. The pigtails extend laterally beyond the circumference of the calandria, and are then collected into vertical manifolds and horizontal header pipes. In the space between the aluminum tube and the stainless steel fuel coolant tube, a mechanical barrier is provided to minimize the effects of failure of either or both tubes.

A number of other possible tube and header designs has been considered, including bayonet tube arrangements, different header arrangements on the vertical through-tube design, and horizontal tube arrangements. The vertical through-tube design with pigtail connections was chosen for a first detailed study on the basis of its relative mechanical simplicity.

The design shown in Fig. 2-1.1 permits the replacement of individual fuel-coolant tubes, if necessary. Concrete neutron shields separate the header rooms from the reactor core, reducing neutron activation of the header structure to a tolerable value so that access can be had to this area after the fuel has been removed, the sodium drained and flushed, and the residual activity allowed to decay for a brief period (see Task 3-2, Shielding). It is planned to evaluate later the more compact arrangement which would be possible if accessibility to the header rooms were not required. (With such an arrangement, even greater emphasis would be placed on tube reliability, and in the event of a failure of a fuel-coolant tube, the tube would be de-fueled, drained, and sealed off, thereby permitting reactor operation.)

In addition to providing for fuel tube replacement and accommodation of thermal expansions, the selected design meets the following requirements:

1. minimum number of welds (particularly field welds),
2. maximum accessibility of field welds,
3. minimum bending stresses at welds,
4. ease of refueling,
5. gravity draining.

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* The figures for each Task are located at the end of the related Task.

Fuel-Coolant Tube

The fuel-coolant tube, to be made of Type 316 stainless steel, is shown in Fig. 2-1.3. From the top, the tube consists of:

1. A heavy-walled refueling extension, 3.03 in. ID, 0.210 in.-thick wall, 2 ft long, which penetrates the top gamma shield plug. The fuel elements are supported from this section.
2. A thinner walled section, 2.81 in. ID. 0.120 in.-thick wall, 11 ft. 9 in. long, that passes through the top header room (where the pigtail connection is welded to the side of the tube) and penetrates the top neutron shield.
3. The core section of the tube, 2.81 in. ID, 0.030 in.-thick wall, 9 ft-8 in. long. This section is made as thin as possible, consistent with tube reliability, to minimize neutron absorption. The effect of the thickness of this section of the tube on reactivity is discussed under Task 3-1, Reactor Preliminary Design.
4. Below the core section, the tube tapers to 1 in. ID, 0.134 in.-thick wall. This smaller tube penetrates the bottom neutron shield into the bottom header room, where it is welded to a pigtail extension. The length of the lower tube section varies from tube to tube to provide accessibility for welding the pigtail connections. The longest tube is approximately 37 ft long overall. Designs to reduce this length are being studied.

Fuel-Coolant Tube Fabrication

The fuel-coolant tube, consisting as it does of several sections having different wall thicknesses or diameters, presents a fabrication problem. A number of fabricators have been asked to consider methods of fabrication of such a tube. It is generally agreed that this tube can be made by welding the several sections together. One tube-forming company has indicated its ability to fabricate this tube in one piece by a special cold-rolling process using shaped mandrels and grooved rolls. This is an attractive possibility, since it eliminates a great many welds from the sodium system. Thickness, straightness, and roundness tolerances achievable by this process are being explored. Various welds that may be used to join tube sections as alternates to one-piece fabrication will be designed for testing under Task 2-4, Fuel Coolant Tube and Header Tests.

Pigtail Design

Between room temperature and design operating temperature, each fuel tube undergoes a thermal expansion of about 3 in., which must be accommodated by deflection of the horizontal section of its lower pigtail. The lower pigtail is connected to a 4-in. diameter manifold on one of the lower 8-in. diameter header pipes, which in turn is mounted rigidly to the floor of the lower header room. To accommodate thermal expansion of the horizontal pigtail, a vertical length of pigtail is provided below the bottom of the lower neutron shield. A separate inlet pipe is provided for each quadrant of the header to minimize the length of the horizontal pigtails.

Since the fuel-coolant tubes are supported from the top, the required lengths of the horizontal and vertical legs of the upper pigtails are much less than those of the lower pigtails.

The pigtail lengths shown on Fig. 2-1.1 are based on conservative choices for the allowable stress at the pigtail connections.

Alignment Problems

The fuel-coolant tubes must penetrate the top gamma shield plug, the upper neutron shield plug, the calandria, and the lower neutron shield plug. Since the fuel-coolant tubes will be centered with respect to the vertical aluminum calandria tubes, provision must be made for any misalignment by making the shield plug holes oversized. Adjustable alignment plugs may be provided to maintain adequate shield integrity. Various alignment plug designs are under consideration.

PRIMARY SODIUM SYSTEM

A preliminary study has been made of the primary sodium system to define the design requirements and to aid in establishing operating conditions for the test programs. A simplified flow diagram of the system is shown in Fig. 2-1.4. The flow rate of 5400 gpm and the inlet and outlet temperatures of 750F and 950F are based on 40 MW heat input to the primary sodium coolant.

As shown in Fig. 2-1.4, the four inlet and outlet pipes to the reactor are manifolded into headers which feed into an external, two-pipe, parallel system. The two-pipe, parallel circuit was chosen for preliminary analysis on the basis of reliability considerations. The arrangement shown permits independent operation of each leg of the external loop. Additional valving and a crossover between the legs could provide additional operating flexibility if necessary.

To provide for emergency cooling of the reactor in the event of power failure, it is intended that the sodium pumps be operable at low speed by auxiliary diesel generator or battery power. Such an arrangement eliminates the need for a safety (auxiliary) sodium circuit with a separate pump, valves, and auxiliary heat sink. It is not likely that power failure will be accompanied by simultaneous mechanical failure of both heat exchangers or both pumps. A power failure accompanied by a disabling malfunction of one leg of the external, two-pipe parallel system could be handled, since each leg of the cooling circuit would be capable of carrying the entire afterheat.

Pumps

A survey of existing sodium pump technology has been started. The centrifugal pumps produced for EBR-II application and now under test at Argonne National Laboratory appear suitable for the SDR, and are much less expensive and more efficient than electromagnetic pumps of similar capacity. The EBR-II pumps develop about 50 psi at a flow rate of 2700 gpm (design flow rate for each leg of the SDR external system).

The SDR pumps are driven through variable-speed hydraulic or magnetic couplings to provide for sodium flow control. This allows for either part-load operation or power variation without large variations in sodium temperatures and without the use of sodium throttling valves of doubtful reliability.

Heat Exchangers

The two primary heat exchangers are identical units, each handling 2700 gpm of sodium and removing 20 MW of heat from the primary circuit. The primary sodium is cooled from 950 to 750F while the secondary sodium (or NaK) is heated from 600F to 900F. A sodium-to-sodium, tube-and-shell type heat exchanger for the above duty requires approximately 750 ft^2 of surface area.

Cold Trap

A cold trap is provided to control the oxide content of the primary sodium. A design flow rate through the cold trap of 17 gpm, or 0.3% of the total flow, has been established. The cold trap consists of a regenerative heat exchanger which cools the sodium to 275F, and a hold-up tank which provides for a 10-min retention period at that temperature. Dowtherm A is used to maintain the hold-up tank at constant temperature. The heat loss in the cold trap is estimated to be about 12 kw.

Sodium Volume and Expansion

The volume of sodium in components of the primary system is summarized in Table 2-1.1; also shown is the volumetric expansion of the sodium in heating from 250F to operating temperature. (The latter figure determines the size of expansion tank required.)

Table 2-1.1 — Sodium Volume and Volumetric Expansion in Primary System

Component	Sodium Vol, gal	Volumetric Expansion 250F to Operating Temperature, gal
Piping	1100	71
Exchangers	210	15
Cold traps	200	9
Pumps	90	4
Reactor	530	38
Headers	510	37
Total	2640	174

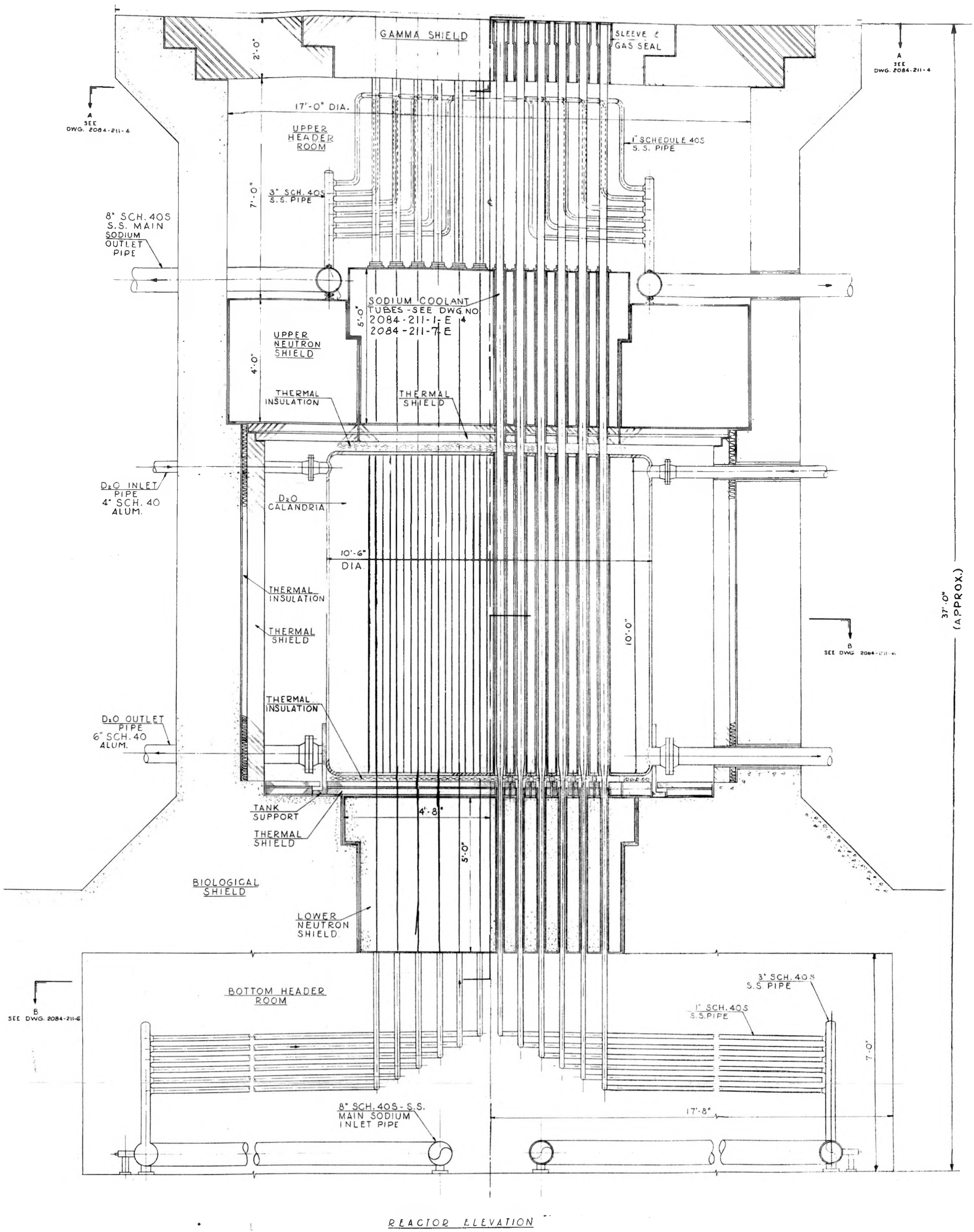

Fig. 2-1.1 — Preliminary reactor arrangement – elevation view

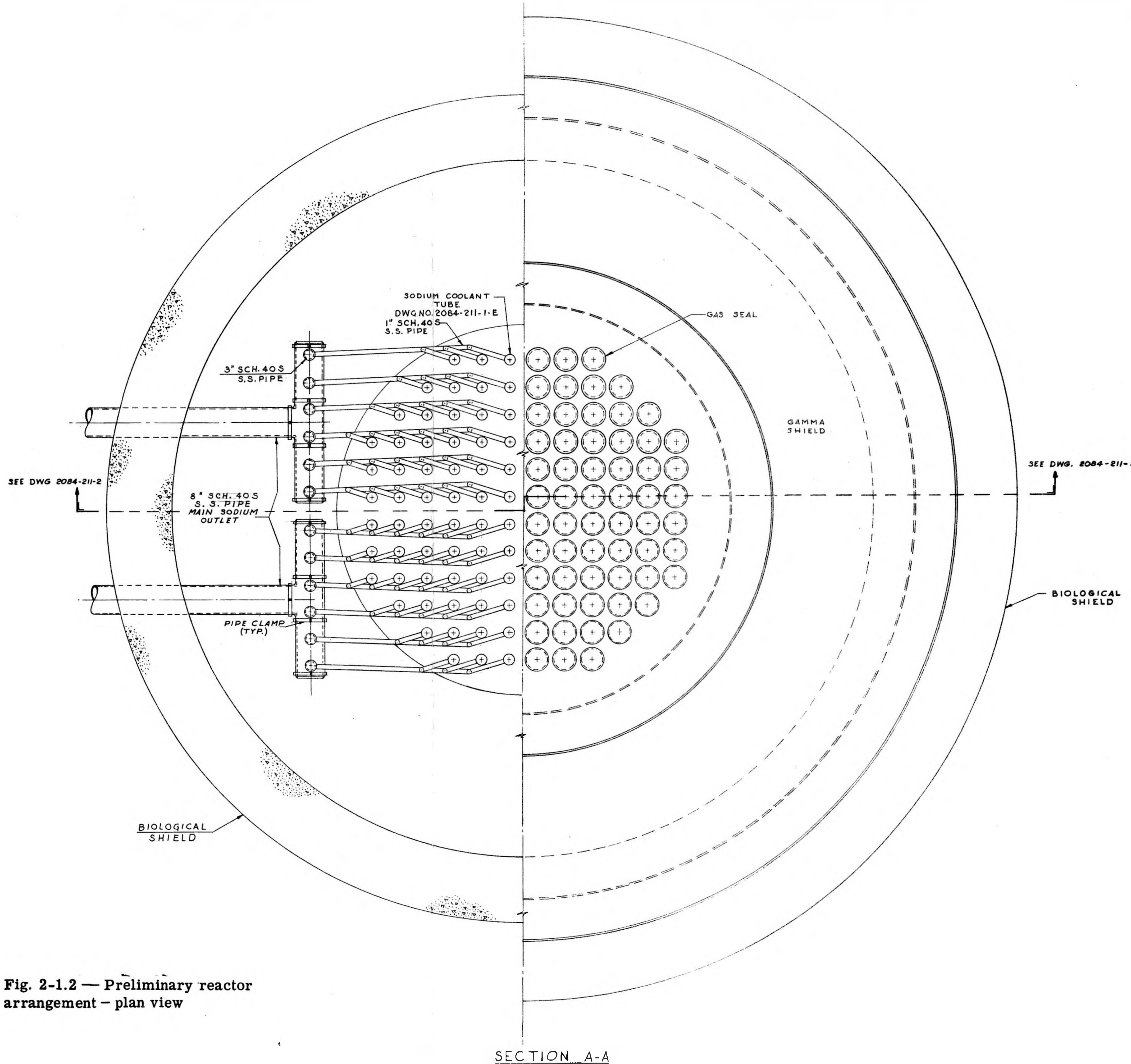

Fig. 2-1.2 — Preliminary reactor arrangement – plan view

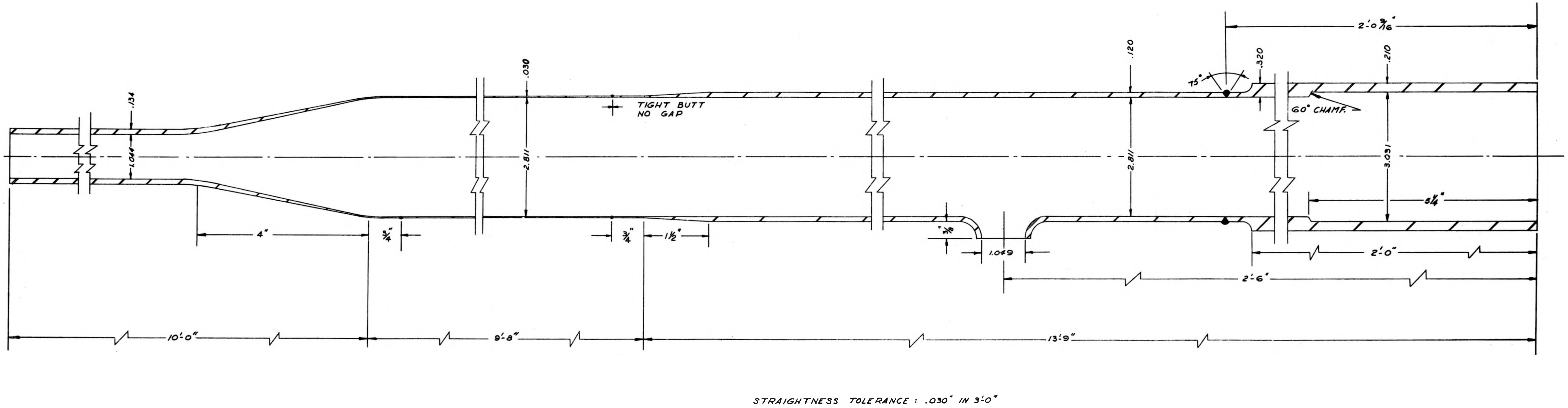

Fig. 2-1.3 — Fuel-coolant tube – section

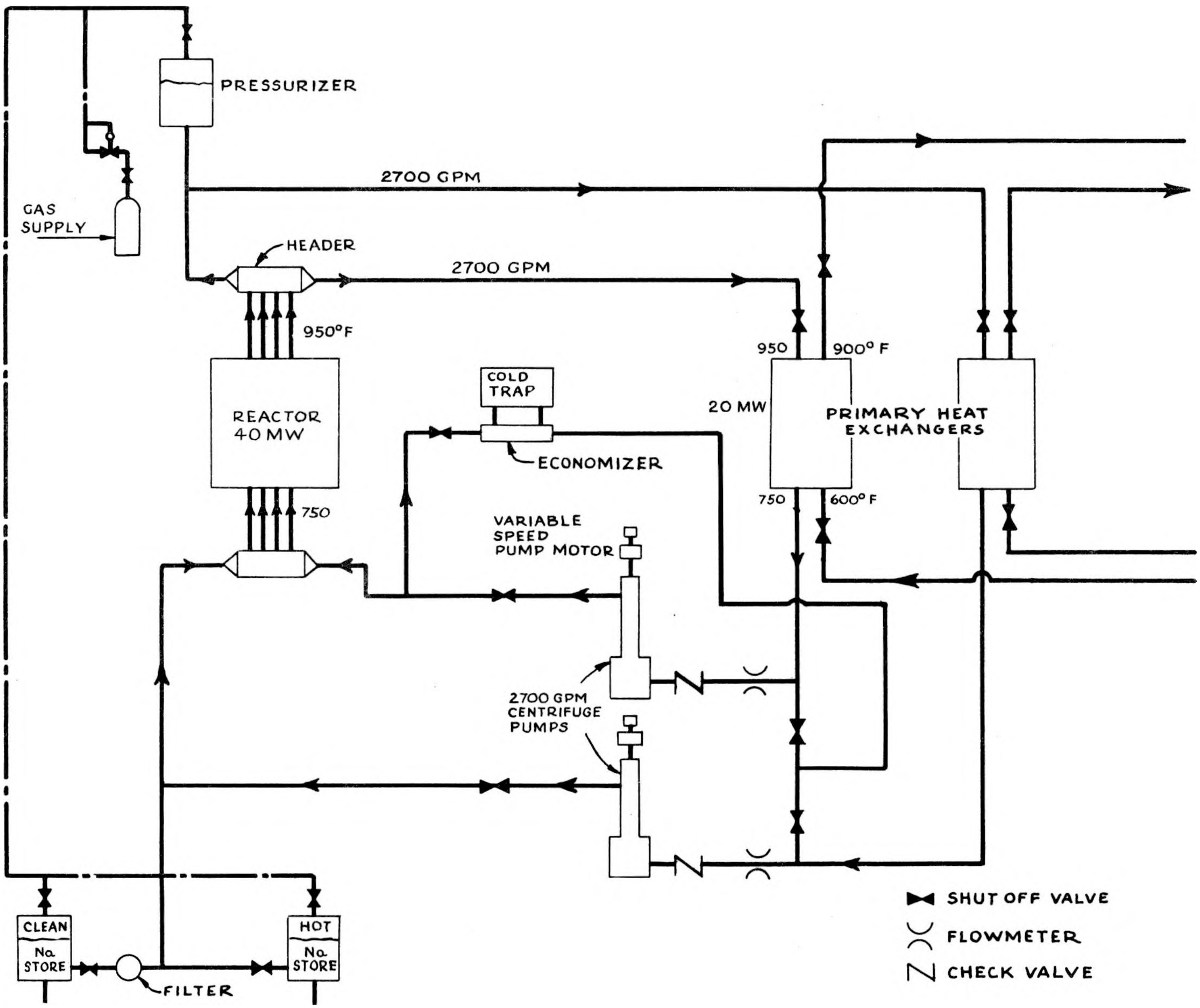

Fig. 2-1.4 — Preliminary flow diagram – primary coolant system

TASK 2-2 D_2O SYSTEM ENGINEERING

A brief study has been made of existing reactors employing D_2O, and some of the system requirements have been established. The D_2O system will require heat removal equipment for about 8% of reactor power (or 3.2 MW), a mixed-bed demineralizer to maintain D_2O purity, and a recombiner to reduce D_2O losses from radiation-induced decomposition. The operating temperature of the D_2O has not yet been established, nor has it been determined whether it will be desirable to use a fast D_2O dump for reactor scram.

TASK 2-3 BARRIER SYSTEM ENGINEERING

Barrier system engineering includes all design features which help to insure D_2O and sodium separation. During the quarter, work was done in the following general areas:

1. survey of possible mechanical barrier materials, i.e., materials which may be located between the fuel-coolant tubes and the calandria tubes to minimize the consequences of single or simultaneous tube failures,
2. consideration of the mechanical arrangements for mounting and supporting the barrier materials,
3. survey of methods of detecting sodium or D_2O leaks in the barrier region of the core.

BARRIER MATERIALS

The following criteria were established for evaluating materials for use as mechanical sodium-D_2O barriers:

1. compatibility (e.g., inertness or desirable reaction) with sodium and D_2O at operating temperatures,
2. low neutron absorption cross section,
3. good radiation and thermal stability,
4. low vapor pressure,
5. availability and fabricability.

Zirconium was chosen as the barrier wall for the design described under Task 3-1 because of its very low thermal neutron cross-section ($\Sigma_a = 0.0069 \text{ cm}^{-1}$) and good compatibility with sodium. Disadvantages of the use of zirconium include its high cost, especially in fabricated form, and its reactivity with such semi-inert gases as nitrogen and carbon dioxide.

Iron, in the form of a mild steel, possesses excellent barrier properties, except for its high capture cross section ($\Sigma_a = 0.195 \text{ cm}^{-1}$). Notwithstanding this disadvantage, a mild steel barrier is used in another design also described under Task 3-1.

Aluminum would be a reasonably suitable barrier material except for its lack of strength at high temperature. A number of high-strength aluminum alloys will be investigated experimentally in the barrier test program, Task 2-5, and the temperature limitations will be more thoroughly delineated.

In the preliminary designs, a graphite sleeve is placed between the fuel coolant tube and the barrier tube to protect the latter from thermal shock in case of a sodium leak. The necessity for such thermal shock protection has not been definitely established, but the incorporation of the graphite does not result in a serious nuclear penalty or design difficulty. Other non-metallic materials, such as refractory oxides, were considered but appear less attractive because of cost and fabrication difficulties.

The nature and functions of the gas filling the barrier region between the various tube walls has not yet been fully defined. The gas spaces serve as thermal insulation, and the gas may be used to preheat the reactor tubes before startup and to provide the possibility of leak detection with an external monitoring system. Helium, nitrogen, and carbon dioxide are under consideration for the barrier gas. The use of carbon dioxide was assumed in the heat transfer calculations for the reference design. The high thermal conductivity of helium points to a heat leakage problem. Nitrogen and carbon dioxide are both acceptable from a heat leakage standpoint but chemical reactivity problems may eliminate them from consideration in conjunction with zirconium barrier materials. These problems will be studied further.

BARRIER MECHANICAL ARRANGEMENT

Fig. 2-3.1 shows a fuel tube assembly, including the graphite sleeve and zirconium tube. Surrounding the coolant tube is a 0.25-in. thick cylinder of graphite, which in turn is surrounded by a 0.020-in. thick zirconium barrier wall. Light spring spacers are provided between the fuel tube and graphite and between the graphite and the zirconium tube to prevent thermal contact. The zirconium barrier wall is sealed to the fuel tube sleeve at the top and bottom. The closure method has not yet been specified, since reliable stainless-to-zirconium welds will require development.

The graphite cylinder is supported on a flange and guide section welded to the bottom of the coolant tube; this section also serves to guide the tube in the lower neutron shield sleeves. The spark plug is shown in a small sump in the bottom barrier extension as a possible means of detecting sodium leaks in the individual tubes. A steel bellows at the top of the outer barrier wall connects the zirconium barrier with the steel coolant channel, and provides for differential thermal expansion between these two tubes. It may be desirable to have the bellows in tension, thus maintaining the outer zirconium barrier wall in tension.

In the event of a leak in the fuel tube, the barrier space will fill with sodium until the barrier pressure balances the sodium system pressure. If this happens, it will be possible to continue reactor operation with a somewhat reduced margin of safety. Alternate mechanical designs for the barrier are under consideration.

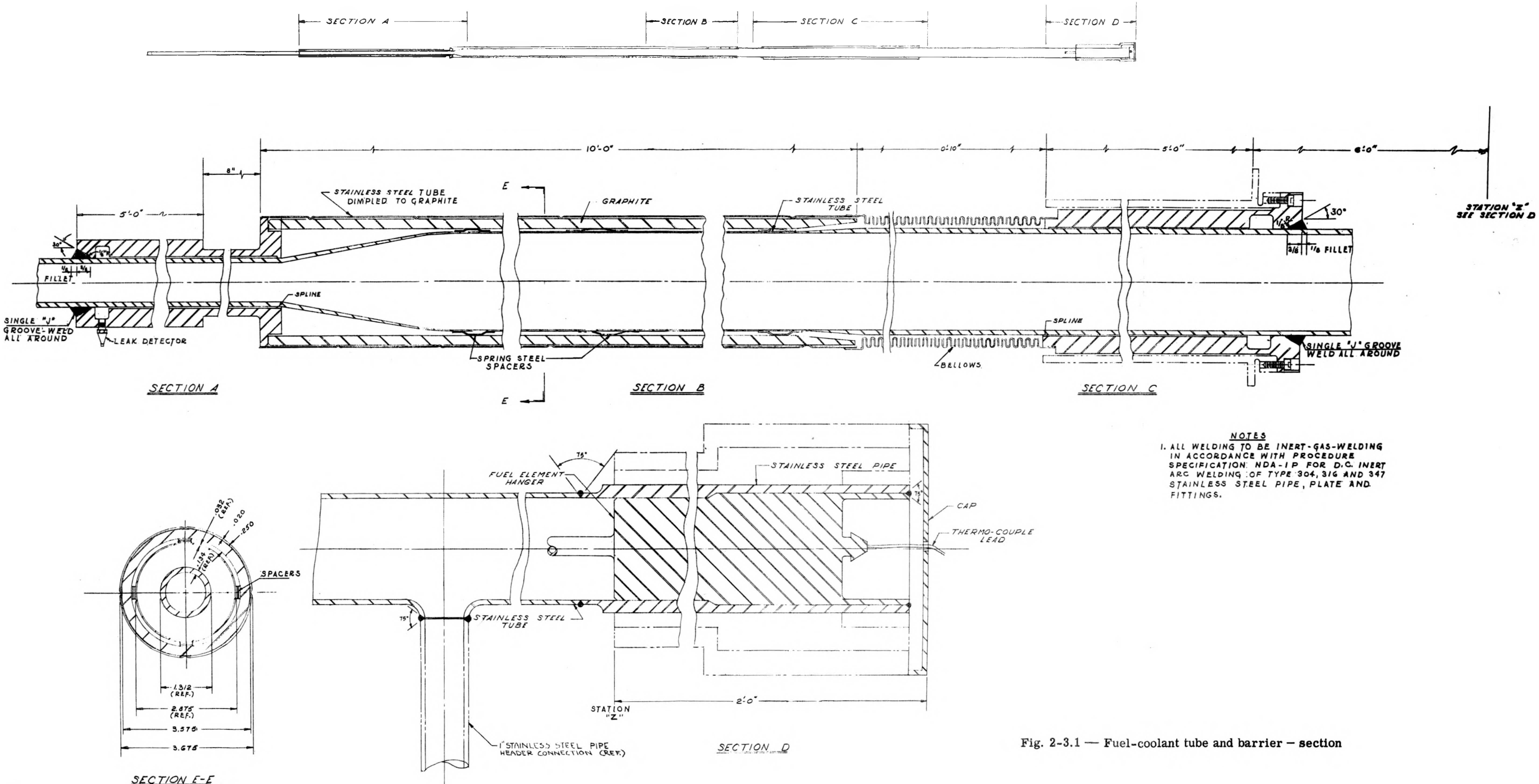

Fig. 2-3.1 — Fuel-coolant tube and barrier – section

TASK 2-4 FUEL COOLANT TUBE AND HEADER TESTS

This task is concerned with the development of suitable joints and designs for the fuel coolant tubes and headers. Critical joints and full-scale sections of tubes and headers are to be tested to provide data for the development of better designs and to demonstrate the mechanical integrity of a final design.

The testing procedure is divided into three main parts: (1) static flexure tests, (2) mechanical cycling tests, (3) thermal cycling tests. Inspection methods will include dye penetrant, x-ray, and metallographic examination.

1. Static flexure tests will be used to screen proposed designs for adequate strength. Samples will be tested in a device similar to that shown in Fig. 2-4.1, and stressed to failure by applying loads by means of a hydraulic jack. Loads and deflections at the point of failure will be noted. Failure is indicated by joint cracking or yielding. A sufficient number of samples will be checked to assure reproducibility.

2. Mechanical cycling tests will be used to check the strength of the joints under alternating flexure after they have been found adequate in the static tests. The specimen, a full-sized subassembly, will be set into a device similar to that shown in Fig. 2-4.2 and subjected to cyclical deflections similar to those expected in the reactor. In general, the test is first conducted cold, and if the joint design is satisfactory for the required number of cycles of deflection, the test is repeated at an elevated temperature. In some cases the cold tests may be bypassed.

3. Thermal cycling tests will be conducted in an assembly similar to that shown in Fig. 2-4.3. Joints will be alternately heated and cooled a sufficient number of times to prove that they are satisfactory.

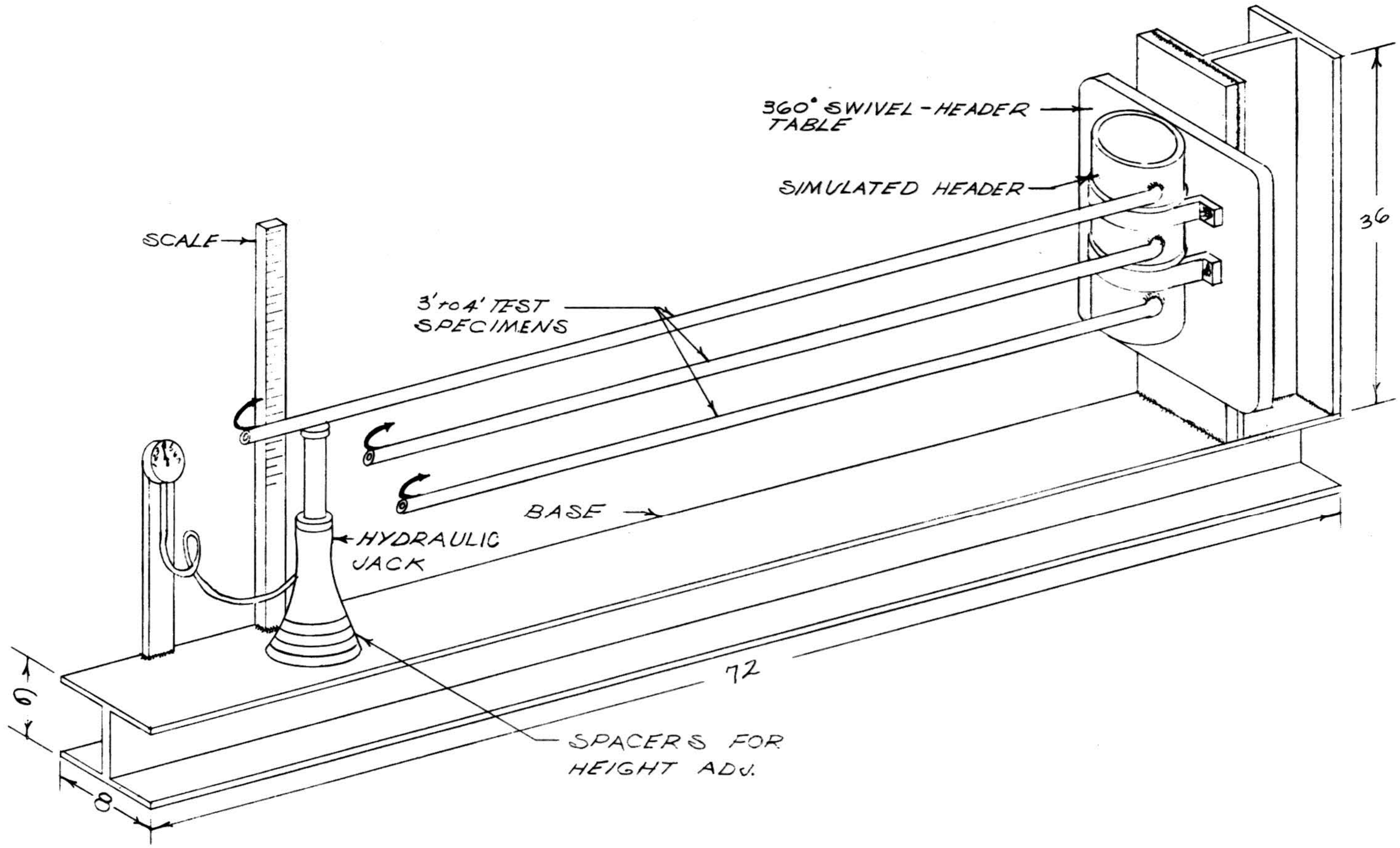

Fig. 2-4.1 — Preliminary concept – tube and header static test rig

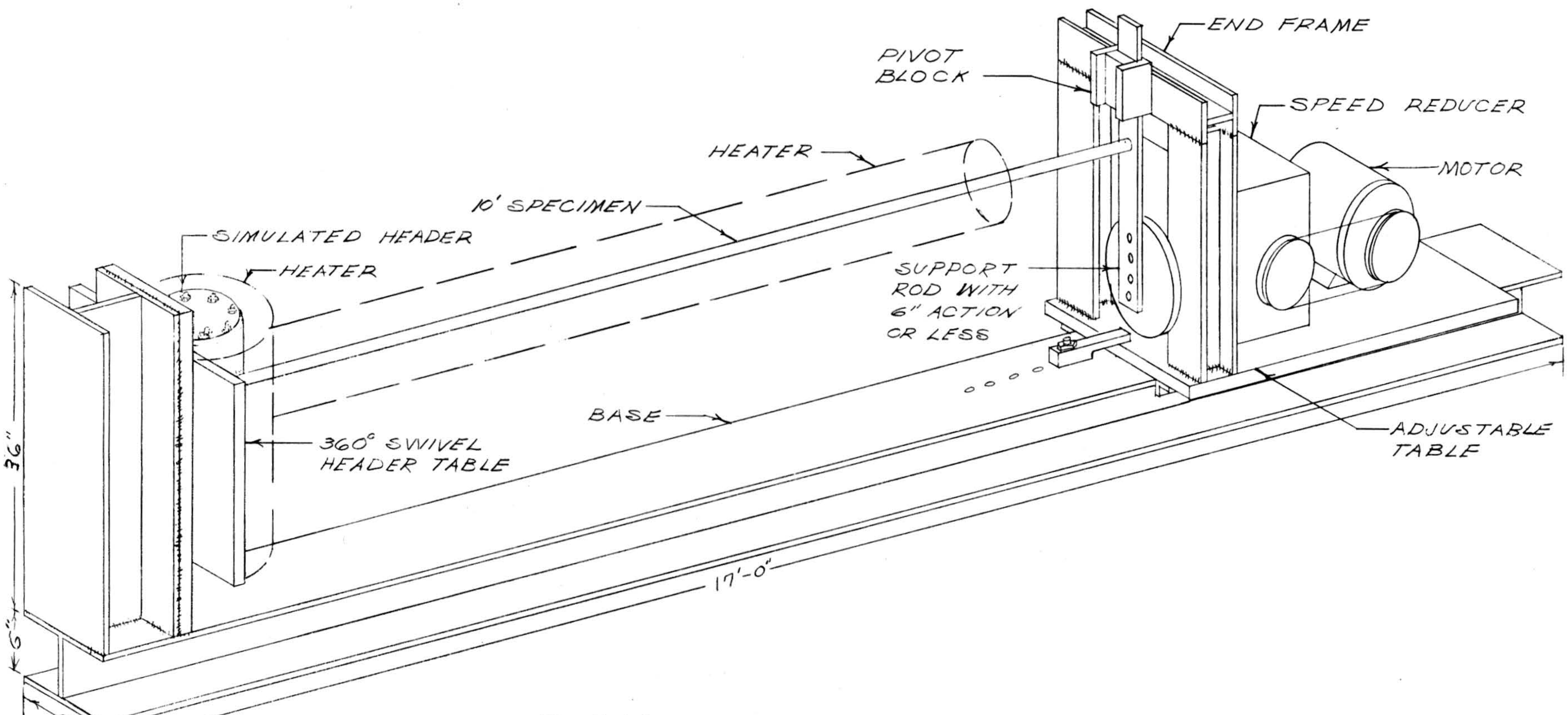

Fig. 2-4.2 — Preliminary concept – tube and header mechanical cycling test rig

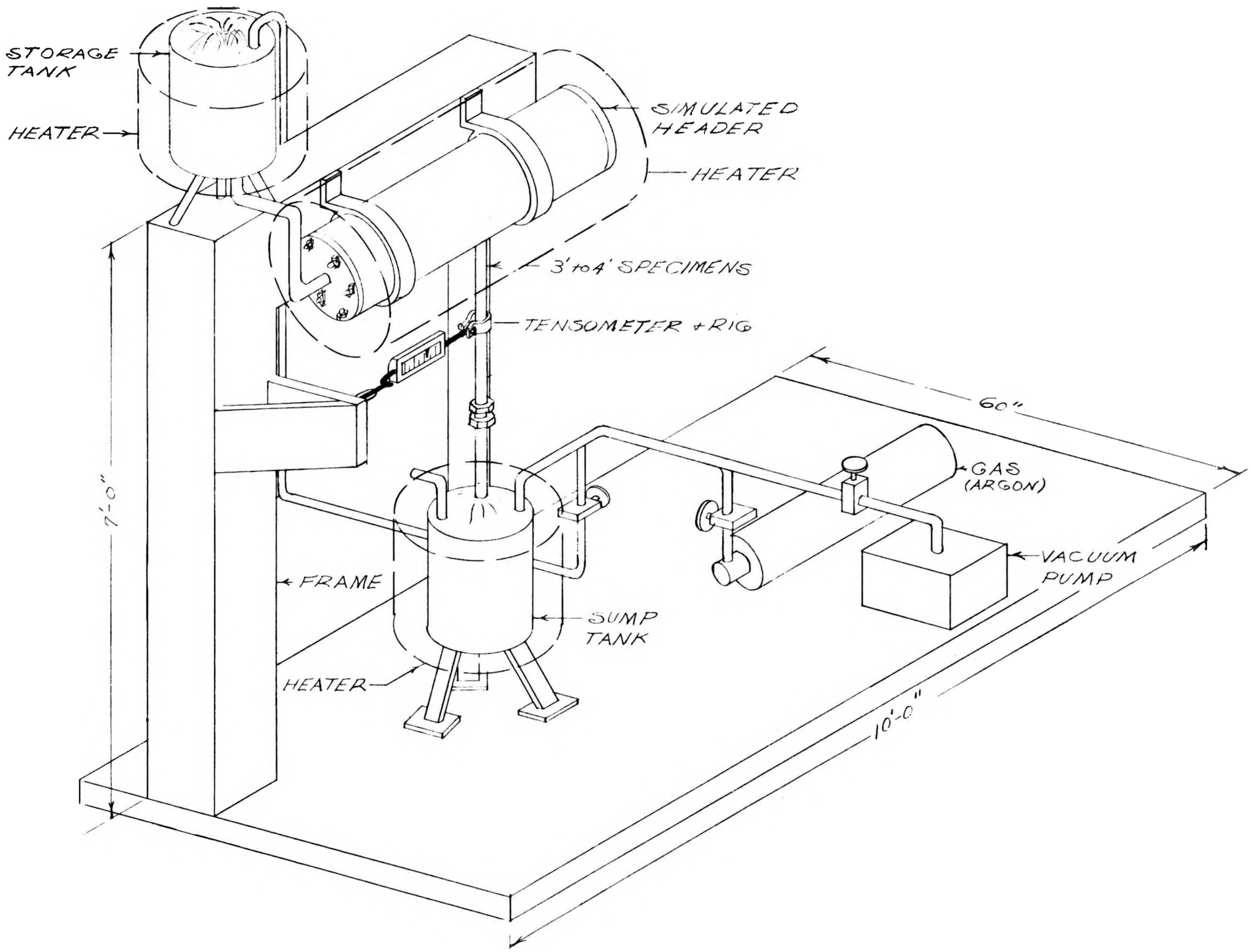

Fig. 2-4.3 — Preliminary concept – tube and header thermal cycling test rig

TASK 2-5 BARRIER TESTS

SINGLE-FAILURE TESTS

Single-failure tests will be conducted to establish the effect of either water or sodium leaks on barrier materials. A flow diagram of a rig to test sodium failures alone is shown in Fig. 2-5.1. Fig. 2-5.2 is a photograph which depicts the state of fabrication and assembly of the rig at the end of the present reporting period.

As shown in these figures, the single-failure barrier test rig is a pumped sodium loop. Its major components — test chamber, surge tank, EM pump, EM flowmeter, and sodium heater — are series-connected by appropriate stainless steel piping. The loop is charged from and drained into an aging (sump) tank, which is connected to the flow system by stainless steel tubing, and separated from it by an isolation valve, filter, and filter bypass.

The test chamber is shown in Fig. 2-5.3. It is a stainless steel tank about 8 1/2 in. OD × 32 in. high, connected to the surge tank by 8 in. of 1-in. pipe. A sight-glass tube and a lighting tube enter the tank near the middle, opposite the nozzle entry port. The chamber head is a bolted, O-ring sealed blind flange, through which pass (via O-ring seals) the turntable shaft and two sodium-shield manipulating rods. The turntable is mounted at the lower end of the shaft, at about the same level as the nozzle. Six specimens of barrier material, 2 in. × 2 in. × 1/4 in. (max), can be mounted vertically on the turntable, and exposed, one at a time, to the sodium jet from the nozzle. A photograph of the turntable assembly is shown in Fig. 2-5.4. With relatively little modification, the rig can be used to test prototype barrier design, full scale in diameter by 1 ft or more in length.

Thermal insulation for the flow system is of two kinds. The permanent insulation is (Johns-Manville) Thermobestos insulating cement; insulation which must be frequently removed is (Johns-Manville) Cerafelt refractory blanket.

The sodium heater is heated by three (Carborundum) Globar elements supported in the refractory brick furnace which surrounds the heater.

The entire rig is supported by an open frame 6 ft long, 4 ft high, and 30 in. wide. The frame is bolted to a steel grating bedplate mounted on six casters for mobility. The top of the test chamber is about 5 1/2 ft above floor level, and stands 15 in. above the roof of the supporting framework. This roof is made of steel and it is intended for use as a floor on which the operator may stand while servicing the test chamber.

Auxiliary equipment consists of a vacuum pump, an inert gas cylinder and regulator, and sheet-metal sodium splash shields placed around the rig frame during operation.

The apparatus is served by three control consoles shown in Fig. 2-5.5. Sodium flow rate is controlled by varying the power input to the EM pump by means of a variable autotransformer. It is measured and recorded by an EM flowmeter and a null-balance potentiometer recorder.

Sodium jet temperature is measured by a thermocouple located in a well in the line about 1 in. upstream from the nozzle. Other sodium temperatures are measured by appropriately located thermocouples.

Barrier specimen temperature is measured by a single-pen (null balance potentiometer) rapid-response recorder and a thermocouple placed against the specimen surface on the side opposite the jet. The thermocouple is stainless steel sheathed, with MgO insulation, and has a time constant rated at 50 msec.

Sodium jet temperature is controlled by a millivoltmeter pyrometer controller. It is expected that simple "on-off" control will prove adequate. However, the instrument can readily be adapted to proportional control if the need arises.

Design and construction of all major components has been completed. The test loop has been assembled, and awaits check-out and installation of auxiliary equipment (thermocouples, auxiliary heaters, insulation). Some minor work remains to be done on the control system, but no delay is anticipated which would prevent completion of the loop and control system at essentially the same time.

Detailed test procedures have been established. At first specimen data will probably be limited to little more than observation of weight and dimension changes, and visual inspection during the initial screening tests of large numbers of samples for short exposure periods. Later, longer and more elaborate tests can be performed on promising materials and configurations. Information obtained during these later tests will include such additional data as metallurgical history of the specimen, surface condition, and hardness data before and after test. In addition to the specimen data, data will be obtained on duration of test, nozzle orifice size, flow rate through the orifice, sodium temperature at the orifice, and nature and pressure of the test chamber atmosphere.

MULTIPLE-FAILURE TESTS

The multiple-failure tests will be conducted to establish the effects of sodium and/or water leaks on container and barrier materials, under simulated reactor operating conditions in which both sodium and water are present. The following ground rules were established for the design of the test apparatus.

1. For reasons of safety, the test should be run at a remote location and it should be controlled from a distance.
2. Provisions should be made for ready change of test section, since a number of barrier types may be tested in a number of ways.
3. Safe access for cleanup should be provided for.
4. It should be possible to duplicate, as far as practical, dimensions, temperatures, and pressures anticipated in the reactor.

On the basis of these ground rules it has been decided that the tests will be conducted within a containment vessel and that sodium, water, and inert gas will be provided by external systems.

The containment vessel is shown in Fig. 2-5.6. It is 121 in. high $\times$ 109 in. wide and is in the shape of an "L" with the test section located in a bell jar provided in the vertical leg. All connections are made through a spool piece just under the bell jar. A preliminary sketch of a test section within the bell jar and connection to it through the spool piece is shown in Fig. 2-5.7. The vessel, with a net volume of 16 ft^3, will be automatically vented in the event of gas generation due to sodium-water reaction. The relief mechanisms will be set to open in the range of 2 to 3 psig pressure. Higher testing pressure levels can be set by adjusting the vents. A second safety feature has been incorporated in the containment vessel — it is designed to withstand an internal pressure of 600 psi. This is the calculated pressure which will be generated in 30 sec if 2 gpm of sodium (the design flow for these tests) reacts with a stoichiometric amount of water, with the vents stuck in the closed position. The vessel is designed to withstand the increased temperature resulting during the pressure buildup.

A preliminary flow diagram of the sodium system is depicted in Fig. 2-5.8. A 50-gal tank, designed to withstand 600 psi pressure and 950F, supplies the heated sodium to the test section. The 950F sodium is forced out of the tank by gas pressure, through a flowmeter, and into the squirter tube. The squirted sodium is conducted from the test section through an annular funnel to a sodium tank. If, in the future, tests of long duration (greater than 15 min) appear desirable, provisions have been made to include an EM pump in the sodium system and to circulate the squirter sodium.

A preliminary flow diagram of the water system is shown in Fig. 2-5.9. The inert gas atmosphere within the container vessel will be provided by means of a standard gas supply system.

Inert gas will blanket the sodium surfaces and all regions where there is a possibility of a sodium-water reaction and the liberation of hydrogen. This is done to eliminate the hazard of a hydrogen-oxygen explosion. Inert gas will also blanket all vent ports to minimize the possibility of air being sucked back through the vents.

General test procedures have been outlined. In one series of tests, sodium is squirted on aluminum walls backed by water (simulating the D_2O tank tubes) for periods up to 15 min. In another series, water is squirted directly on a heated, sodium-containing tube. In still another series of tests, sodium and water squirts are simultaneously directed (from opposite directions) at a barrier under conditions which will simulate a comparable leak in a reactor.

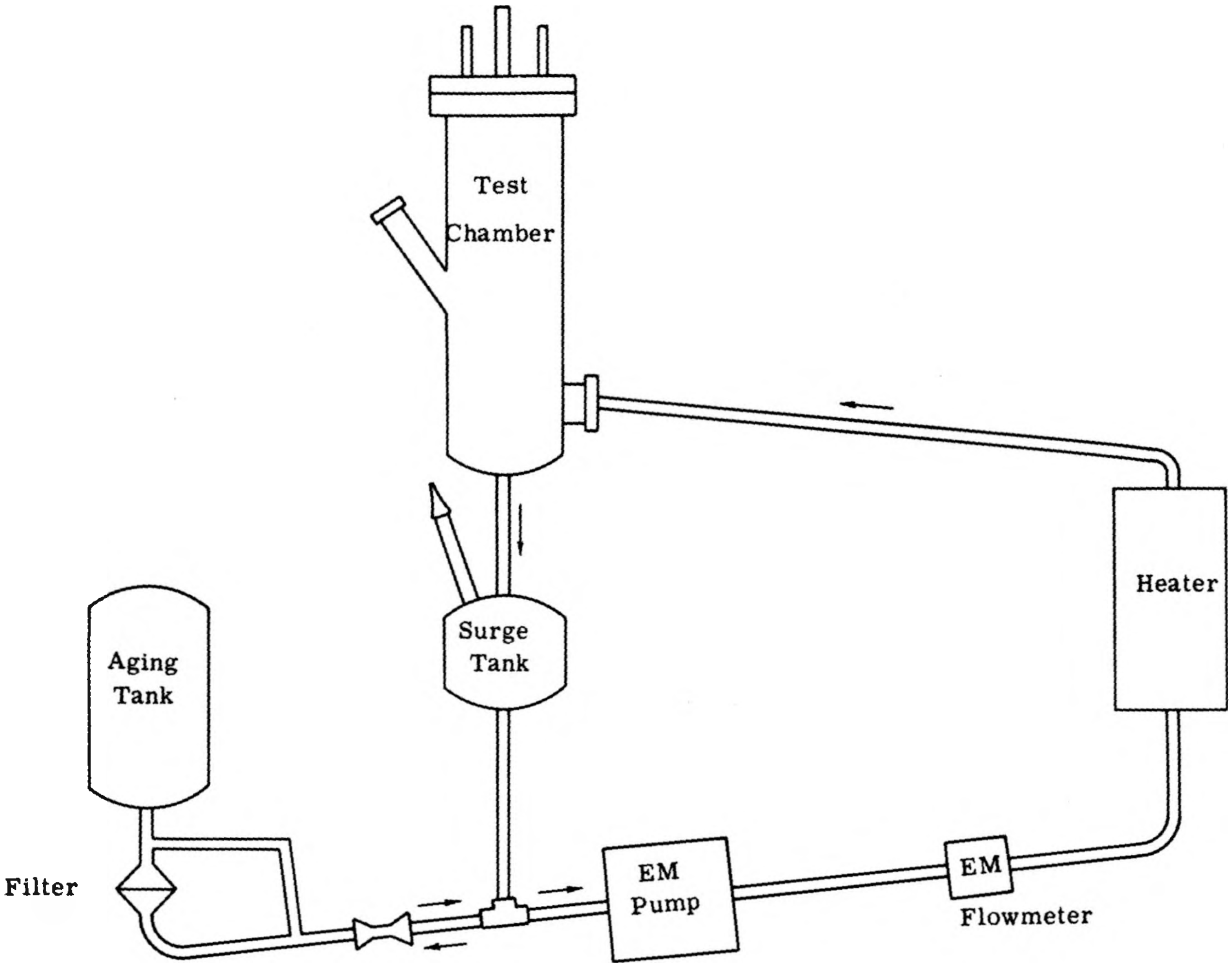

Fig. 2-5.1 — Flow diagram – single-failure test apparatus

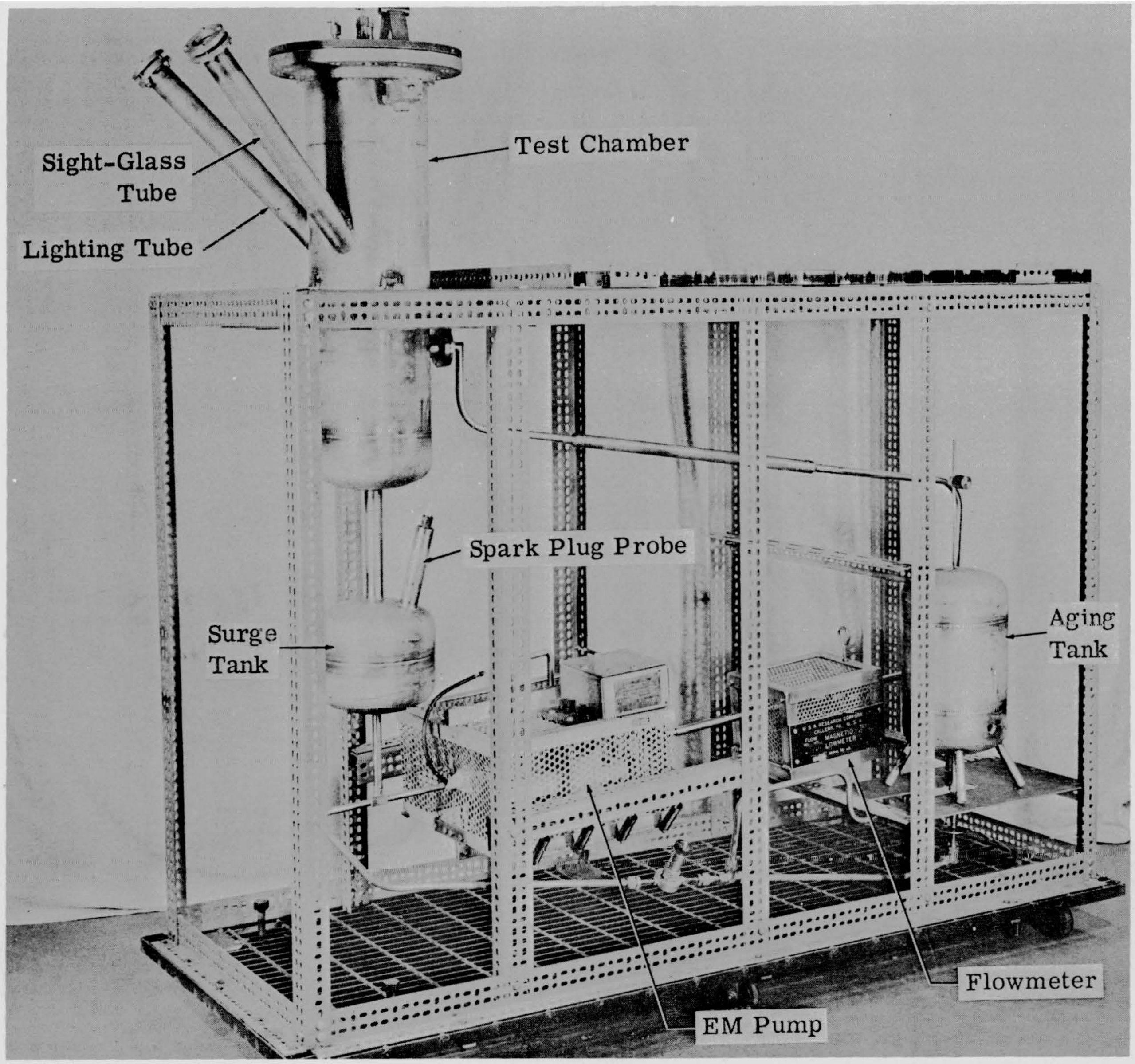

Fig. 2-5.2 — Single-failure test apparatus

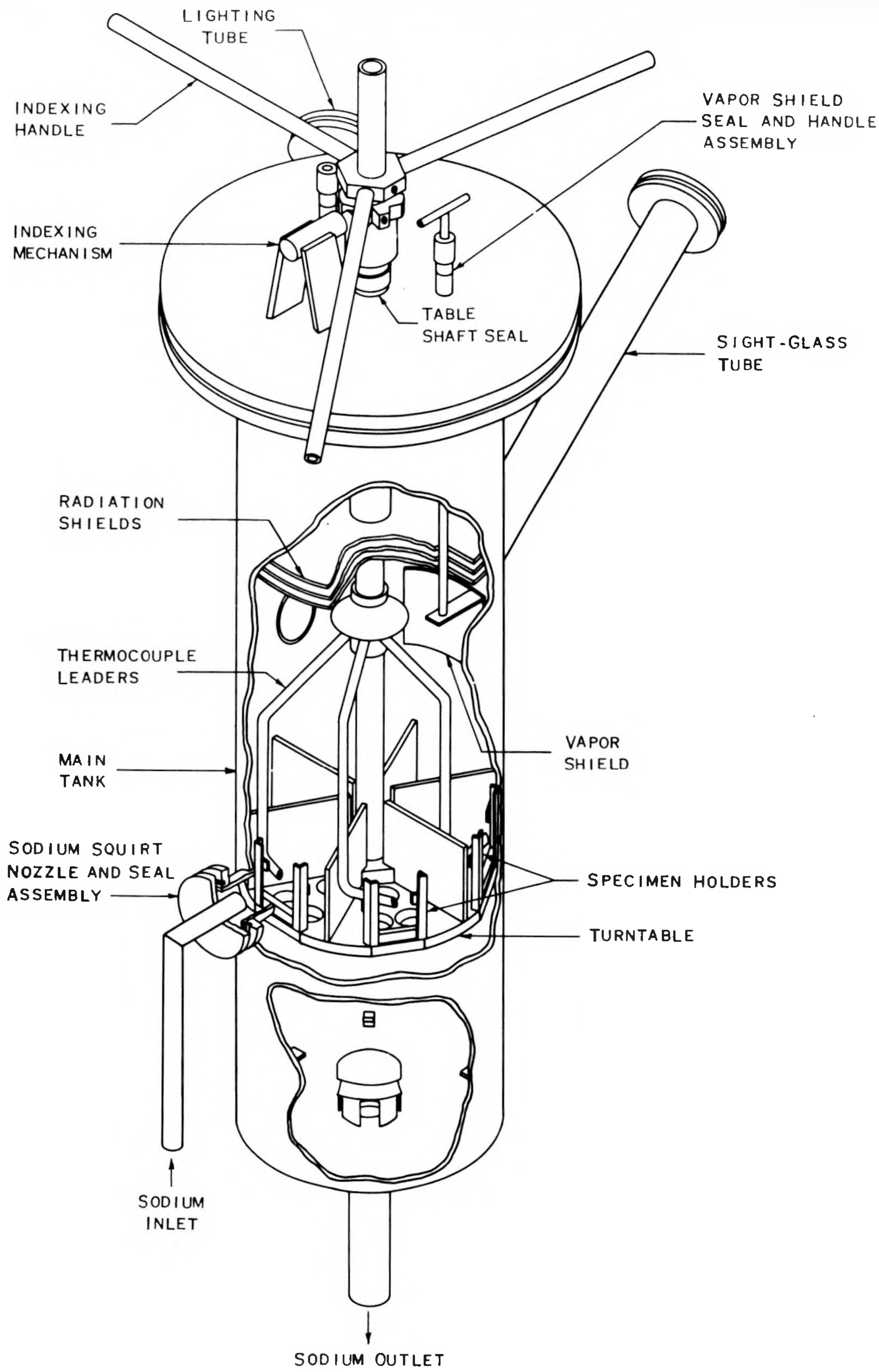

Fig. 2-5.3 — Test chamber assembly – single-failure test

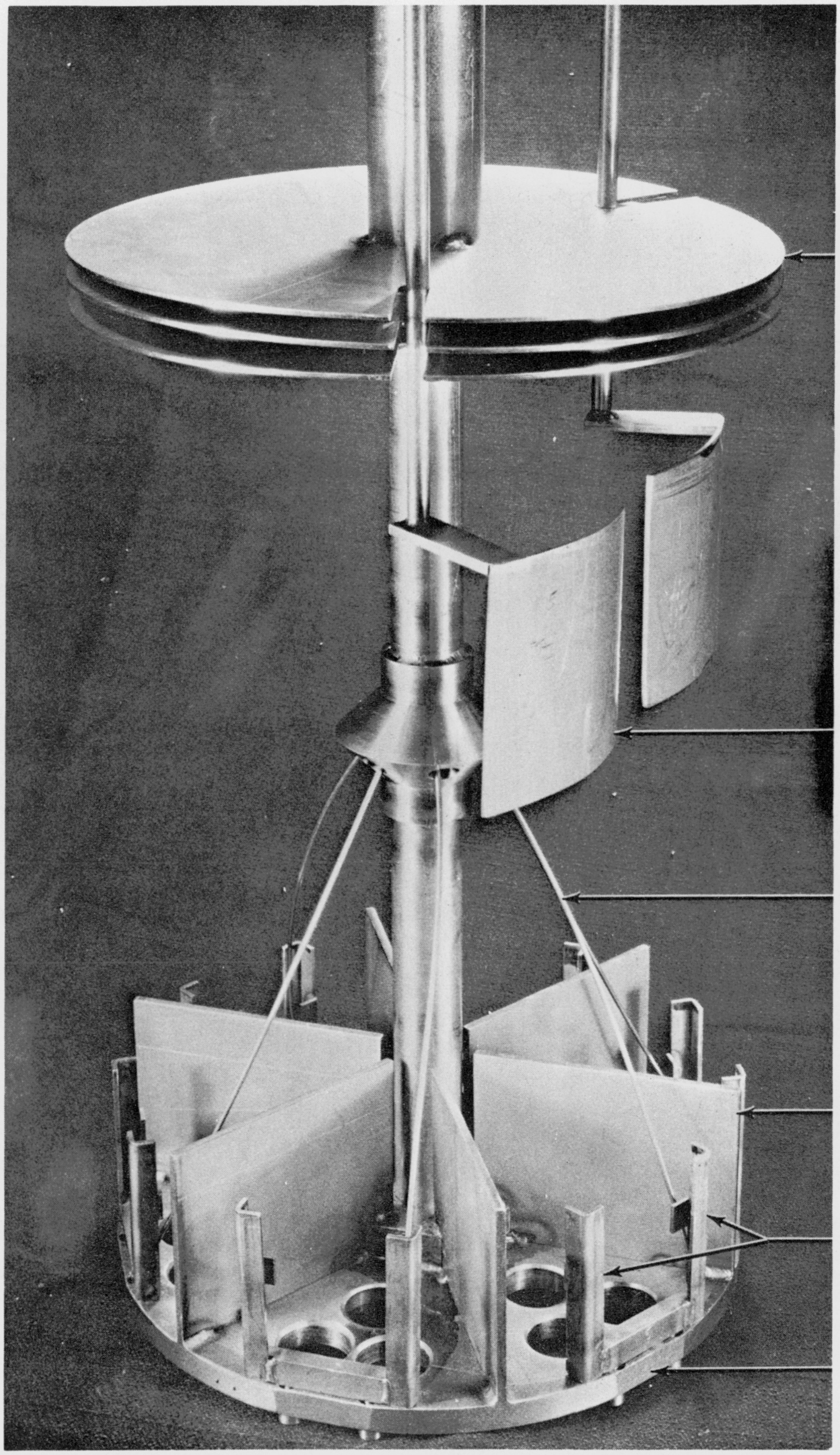

Fig. 2-5.4 — Turntable assembly – single-failure test

Fig. 2-5.5 — Control consoles – single-failure test

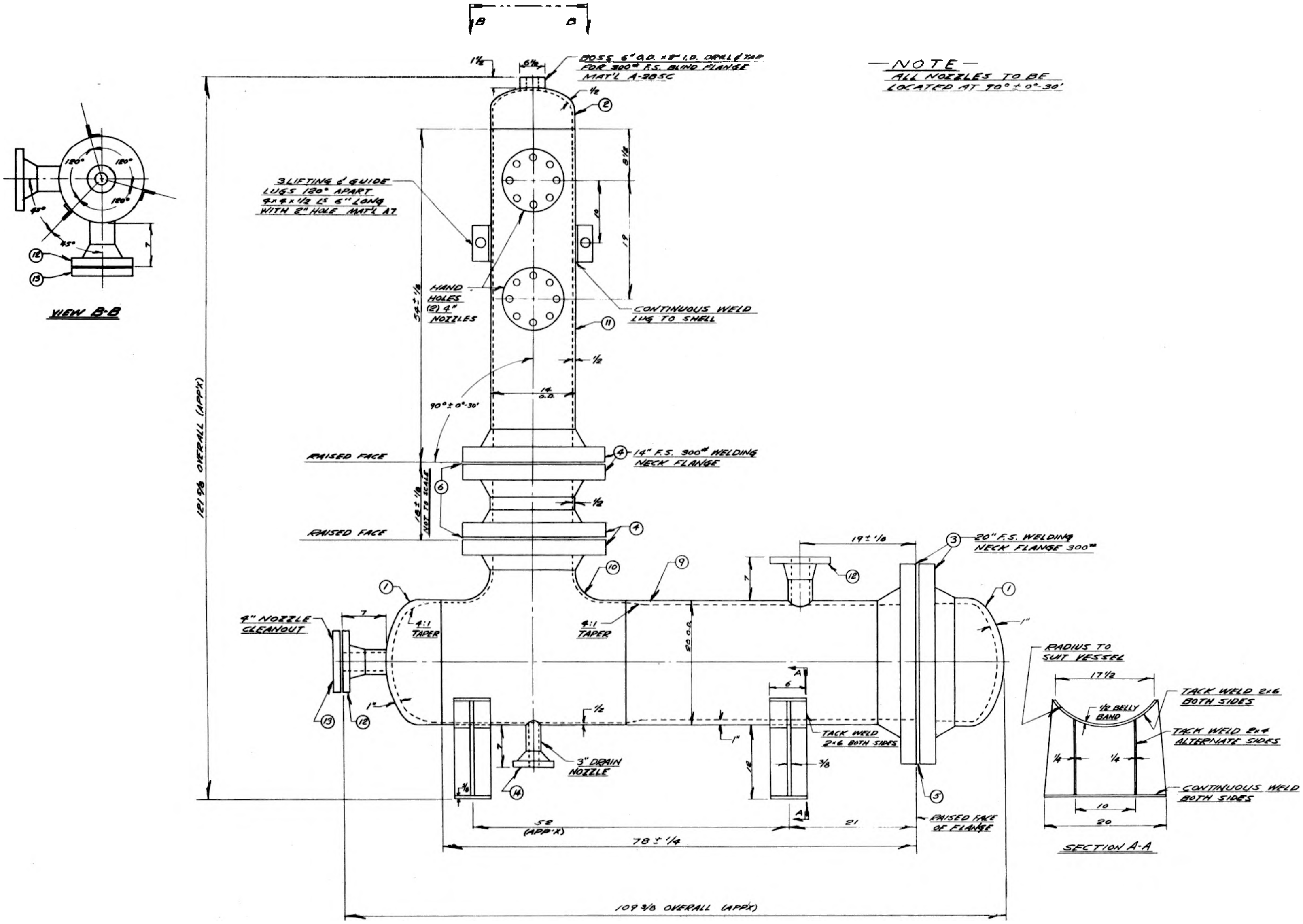

Fig. 2-5.6 — Containment vessel – multiple-failure test

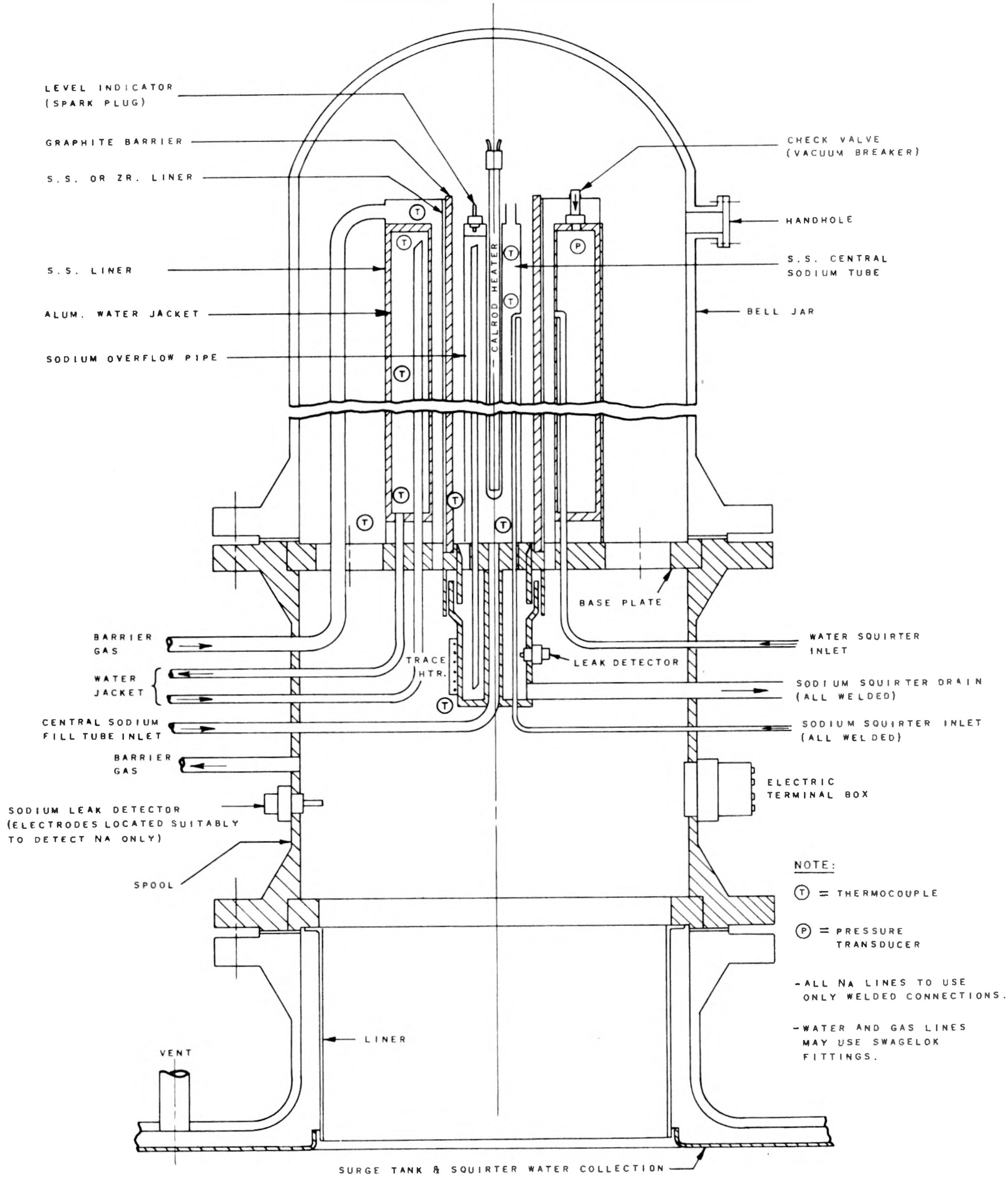

Fig. 2-5.7 — Preliminary test section design – multiple-failure test

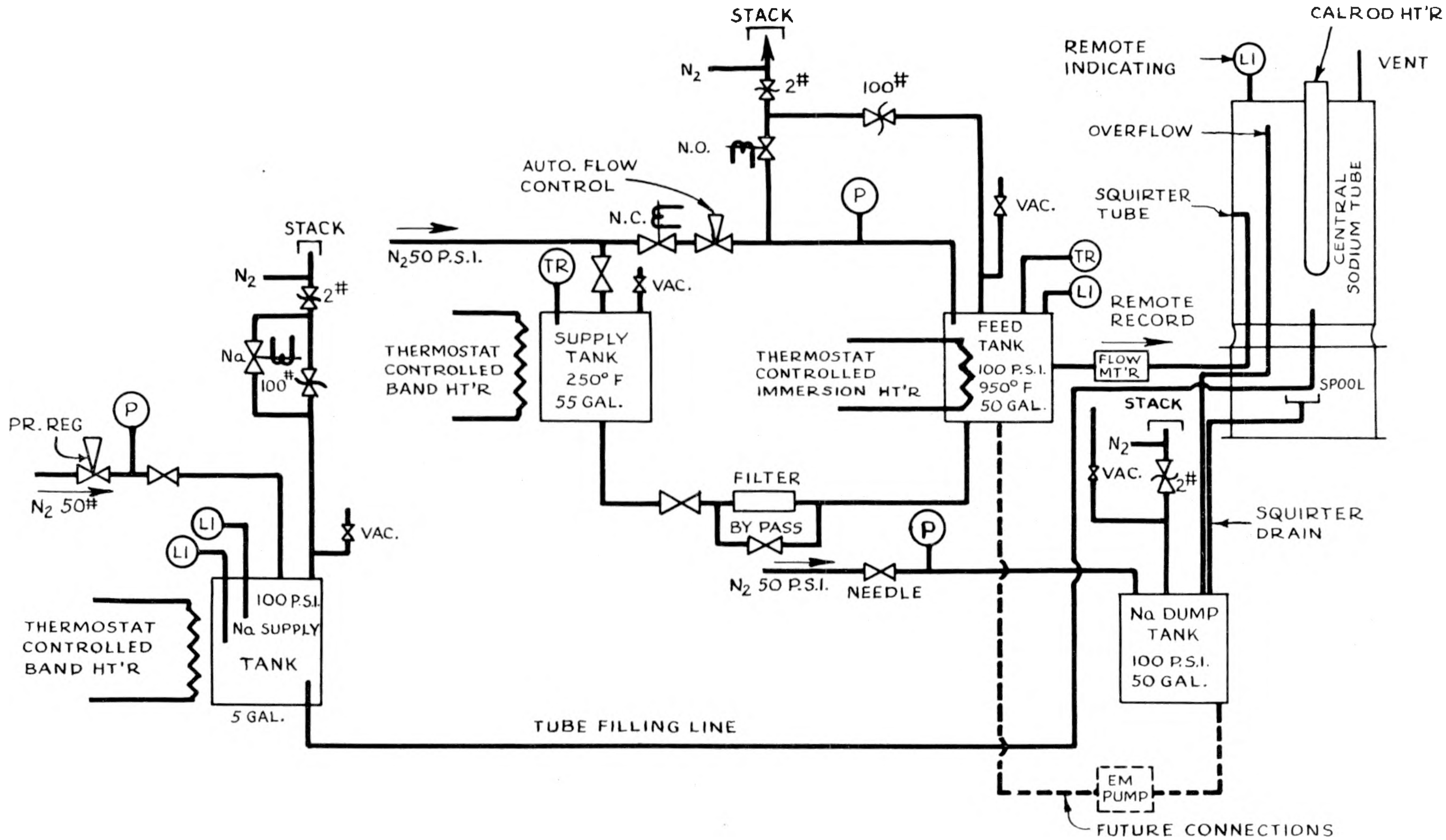

Fig. 2-5.8 — Preliminary flow diagram of sodium system – multiple-failure test

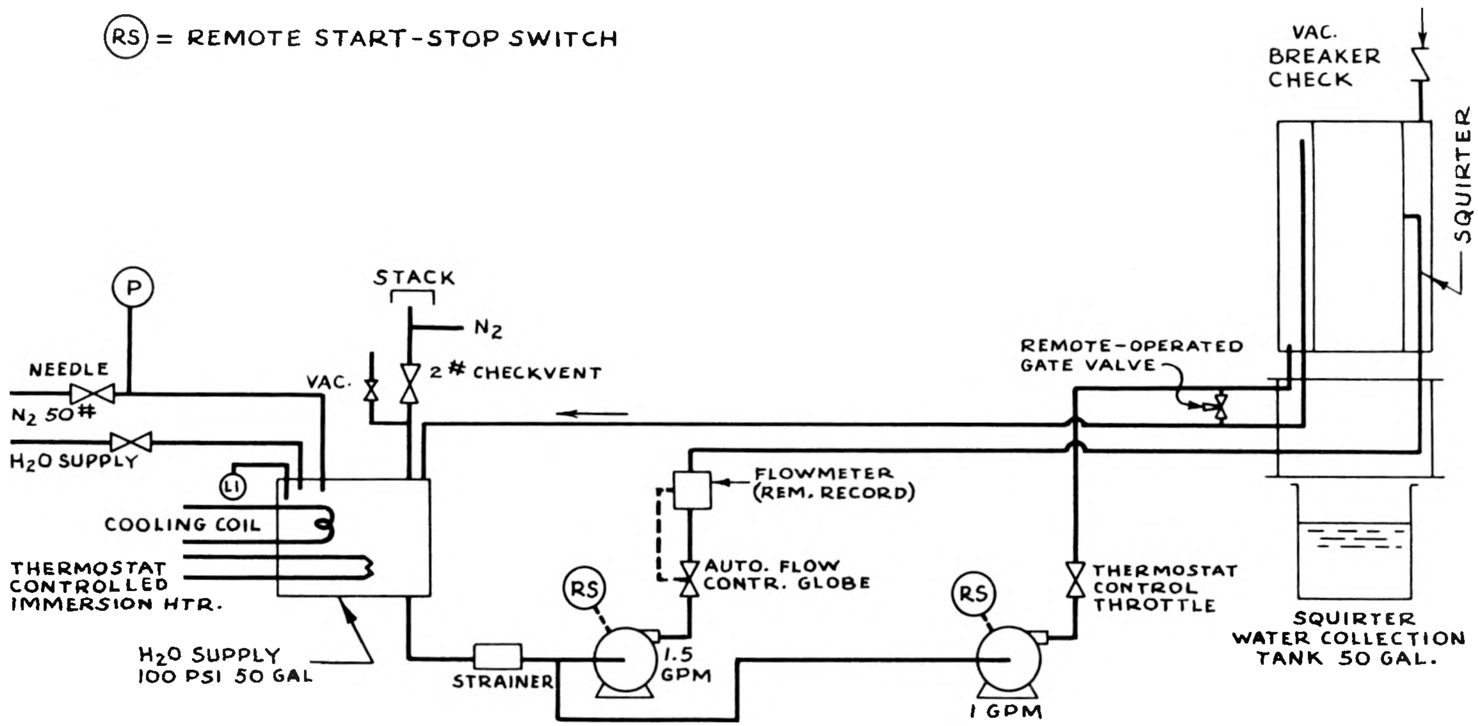

Fig. 2-5.9 — Preliminary flow diagram of water system – multiple-failure test

TASK 2-6 MOCKUP TESTS

Mockup tests will be conducted with flowing sodium and flowing water, separated by a mechanical barrier, to demonstrate the safety of such a system under conditions which will simulate normal and aggravated reactor operating conditions.

The equipment which will be used in these tests is shown schematically in Fig. 2-6.1 and consists of a mockup of the reactor fuel tube and barrier system, called the test tank, and also process systems which will supply the tank with hot sodium, moderator water, and barrier gas. Details of the test tank will mock-up those of an actual reactor as much as is possible.

The nature and dimensions of the materials of construction will be such that a 10,000 kw_e SDR so constructed would require a uranium enrichment of about 2%. For purposes of the test, chemically equivalent materials will be substituted for those possessing desired nuclear characteristics. For example, H_2O will be substituted for D_2O.

As shown in Fig. 2-6.2, the test tank contains three fuel coolant tubes, each 8 ft long, arranged in a triangular pattern and enclosed in a moderator tank 2 ft in diameter. Stresses set up by a difference in temperature between tank wall and aluminum tubes are relieved by an expansion joint. Header boxes are joined to the tank on top and bottom, forming an assembly approximately 12 ft long. The test section is supplied with flowing sodium, moderator water, and barrier gas from individual process systems. In addition, inert gas is used to blanket the sodium.

The sodium process system is comprised of a Globar element furnace capable of heating the liquid metal to 950F, an EM pump and flowmeter, a cold trap that will accept a portion of the sodium flow, and sump and expansion tanks. All external sodium lines are electrically trace-heated.

The water process system includes a storage tank, a circulating pump, and a water cooler (commercial unit heater). The water system contains a latched dump valve capable of dumping the water in 30 sec. Water may be heated by an immersion heater contained in the storage tank.

An external steam supply may be connected to the water system to preheat the test tank to 250F.

The control system for the mockup has two purposes. One is the maintenance of the proper sodium and water temperatures in the test assembly and the other is the protection of the equipment from abnormal temperatures and from the effect of fire.

The sodium system temperatures are controlled by varying the heat input in the furnace and by adjusting the rate of sodium flow. The water system temperatures are controlled by means of electric immersion heaters in the water storage tank and/or by regulating the flow of water through the air-blast water cooler. No control of gas temperature.is attempted.

The control room is located 400 ft from the test site. It contains indicating and recording instruments, push buttons, and similar devices. Located at the test site are all major power devices such as contactors, variacs, power panels, and motor starters. Personnel will not be permitted in the operating area when the test tank contains both sodium and water.

Procedures for startup and shutdown have been worked out in a preliminary fashion. Under steady-state normal operating conditions the sodium temperatures will be held at 950F inlet and 750F outlet and the water temperature will be approximately 180F. The system is to be run under these conditions for the major portion of the operating time. Aggravated operation may include thermal cycling of the sodium inlet temperature between 950F and 500F, raising the sodium inlet temperature to 1050F, reducing purity of the sodium, increasing sodium pressure, and frequent water dumping.

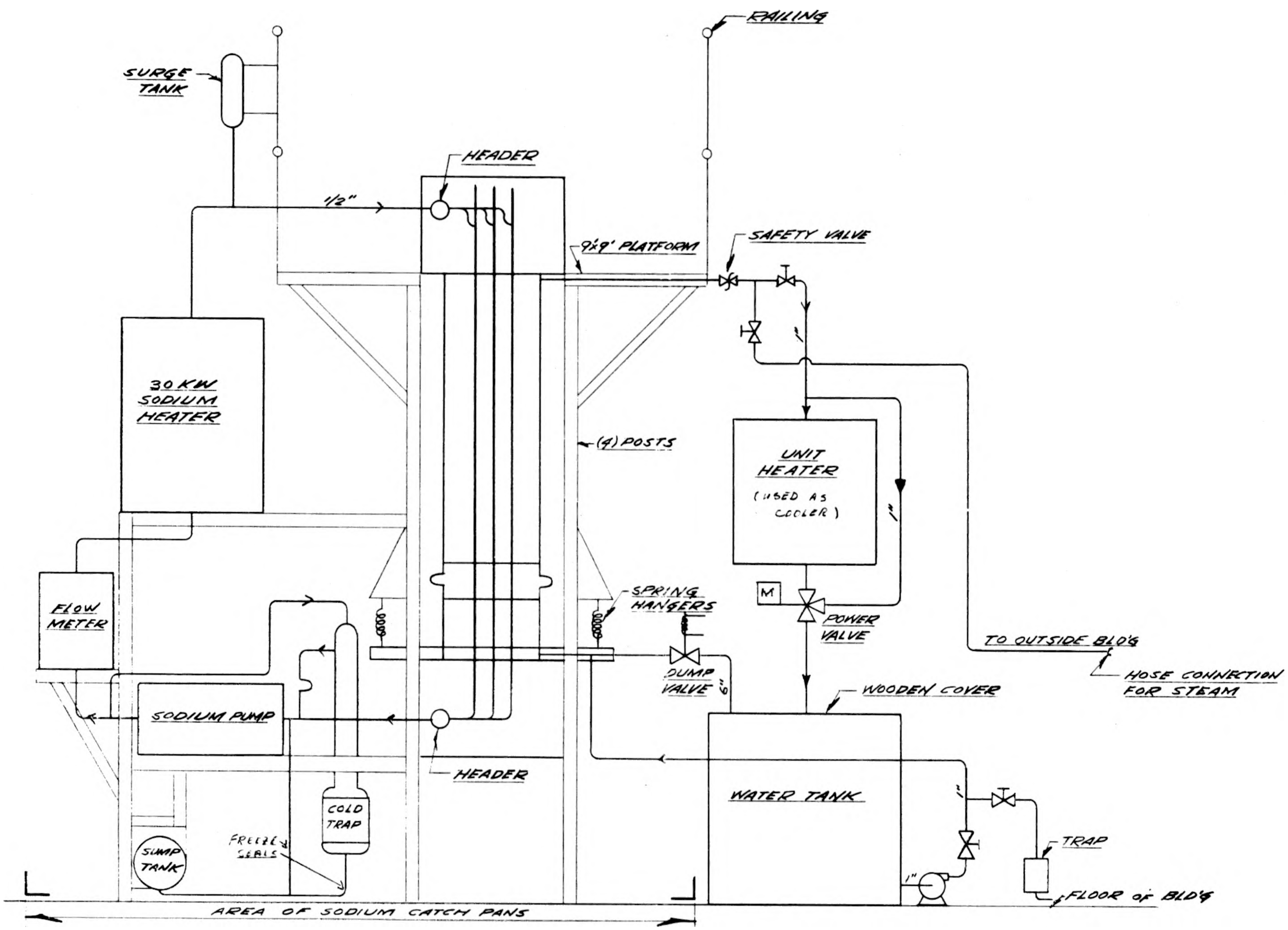

Fig. 2-6.1 — Preliminary flow diagram – mockup test

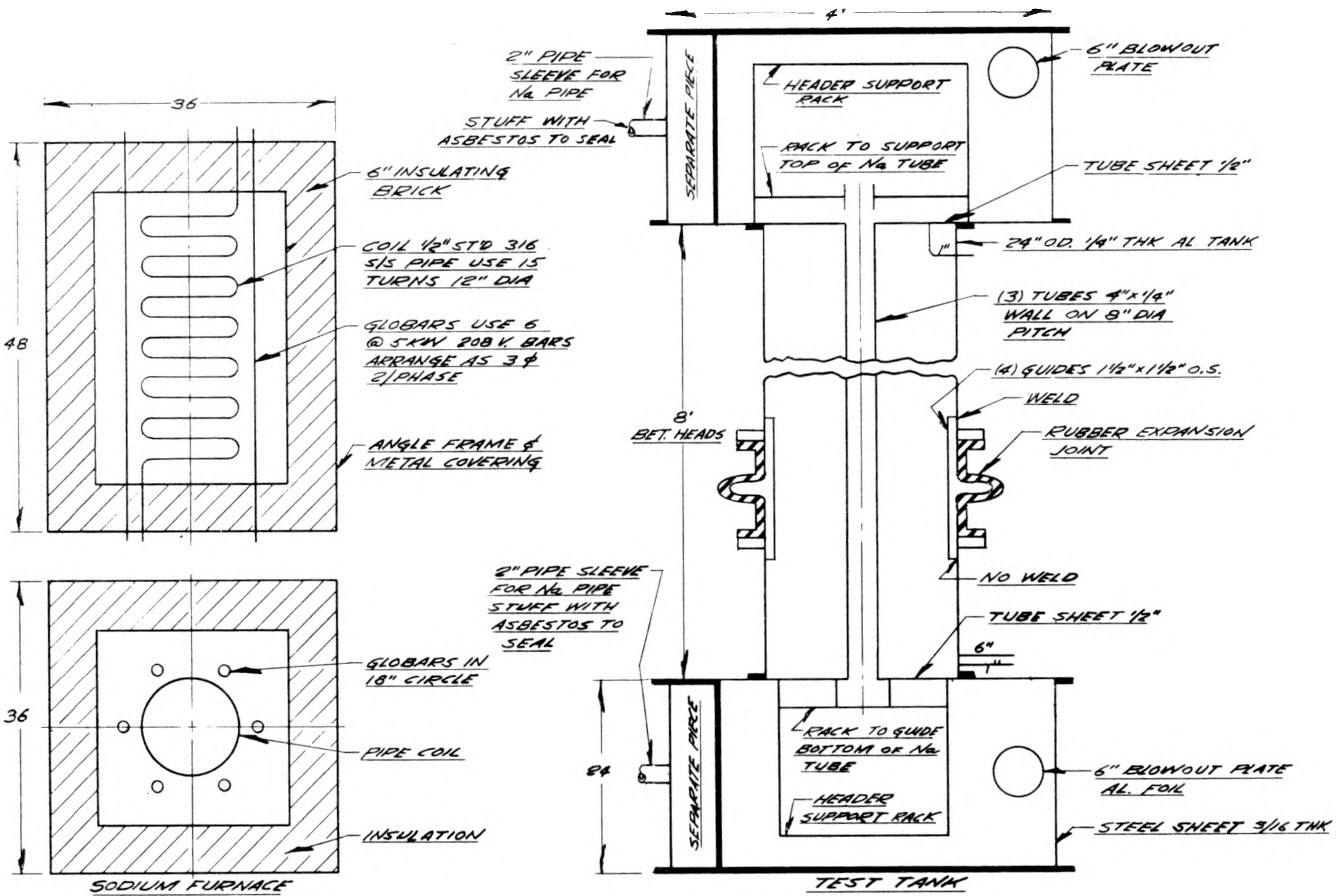

Fig. 2-6.2 — Preliminary sodium furnace and test tank arrangement —mockup test

PRELIMINARY DESIGN

TASK 3-1 REACTOR PRELIMINARY DESIGN

During the quarter, a number of different mechanical arrangements of the SDR were studied, as discussed later in this section, and a particular design configuration (straight-through fuel coolant tube with top and bottom pigtail connections) was chosen for more detailed analysis on the basis of its desirable mechanical features. The mechanical arrangements of the reactor core and of the fuel-coolant tubes and headers for this design are described under Task 2-1, and are shown graphically in Figs. 2-1.1 and 2-1.2.

Parametric studies were performed to outline the area of interest for this reactor configuration. Two designs, designated "Design A" and "Design B" were selected to form the basis for continued evaluation. Data for these designs are summarized in Table 3-1.1. Design A is consistent with the fuel-coolant tube and barrier designs described under Task 2-1. Design B is based on the following more conservative design features:

1. use of 60-mil wall fuel-coolant tube rather than 30-mil wall,
2. use of 20-mil wall stainless steel barrier, rather than 20-mil wall zirconium barrier,
3. use of a uranium-10 weight % molybdenum alloy fuel element, rather than unalloyed uranium.

Each of these features increases the mechanical reliability of the SDR design, or relieves a significant development problem, but imposes a nuclear penalty. As indicated in Table 3-1.1, the combined result is to increase the uranium enrichment from 1.25% to 1.75%, the fuel inventory by 37%, and the D_2O inventory by 9%.

FUEL ELEMENTS

Each fuel tube contains a hexagonal cluster of seven cylindrical fuel elements, separated from each other by a helically wound wire. SRE-type elements have been assumed for both designs. These are long rods of slightly enriched uranium or uranium alloy, 0.75 in. in diameter and contained in a 0.010-in.-thick stainless steel jacket. A NaK heat transfer bond 0.010 in. thick fills the gap between the rod and its jacket.

It is not likely that the unalloyed uranium fuel element will withstand 5000 MW-d/ton burnup. Small amounts of alloying materials, such as are considered for Design B, will considerably improve the metallurgical burnup limits without a severe nuclear penalty.

PARAMETRIC STUDIES

Simplified calculations were made of the nuclear, thermodynamic, and economic characteristics of straight-through designs with various fuel enrichments and lattice spacings. The purpose

Table 3-1.1 — Reactor Design Data

Specification	Design A	Design B
Reactor power, kw	40,000	40,000
Fuel loading, metric tons	7.5	10.25
Fuel enrichment, %	1.25	1.75
D_2O in core tank, metric tons	6.2	6.75
Height of core tank, ft	5.9	6.2
Diameter of core tank, ft	7.4	7.8
Length of fuel elements, ft	5.9	6.2
Number of fuel tubes	113	143
Diameter of fuel rod, in.	0.75	0.75
Number of fuel rods per subassembly	7	7
Fuel alloy	Unalloyed uranium	U-10 w/o Mo
Fuel clad	Stainless steel	Stainless steel
Thickness of clad, in.	0.010	0.010
Sodium-containing tube	Stainless steel	Stainless steel
Thickness of fuel-coolant tube, in.	0.030	0.060
Metallic barrier	Zirconium	Stainless steel
Thickness of metallic barrier, in.	0.020	0.020
Lattice type	Square	Square
Lattice pitch, in.	7.37	6.87
Heat transfer and fluid flow		
Inlet temperature, °F	750	750
Outlet temperature, °F	950	950
Total sodium flow rate, lb/hr	2.24×10^6	2.24×10^6
Avg sodium velocity in fuel tubes, ft/sec	5.9	4.6
Max sodium velocity in fuel tubes, ft/sec	9.4	7.4
Avg sodium velocity in pigtails, ft/sec	19.3	15.1
Max sodium velocity in pigtails, ft/sec	32	25
Avg power per fuel tube, MW	0.35	0.28
Max power per fuel tube, MW	0.57	0.45
Max fuel temperature, °F	< 1200	< 1200
Sodium flow area per fuel tube, ft^2	0.0179	0.0179
Pigtail flow area, ft^2	0.00545	0.00545
Physics		
D_2O-to-fuel volume ratio	14.0	11.2
Na-to-fuel volume ratio	0.22	0.22
Steel-to-fuel volume ratio	0.14	0.30
Equivalent lattice diameter, in.	8.3	7.8
k_∞, clean	1.240	1.210
k_{eff}, clean	1.096	1.088
k_{eff}, equil. Xe and Sm	1.063	1.054
η	1.580	1.695
ϵ	1.025	1.025
p	0.864	0.860
f	0.886	0.810
τ, cm^2	168	179
L^2, cm^2	95	75
M^2, cm^2	263	254
Geometric buckling	5.00×10^{-4}	4.41×10^{-4}
Material buckling	9.13×10^{-4}	8.27×10^{-4}
Initial conversion ratio	0.608	0.528
Assumed average fuel burnup, MW-d/ton	~5000	~5000

of these studies was to indicate range of interest for some of the important design variables, and thus provide a somewhat better basis for current mechanical design studies. Because of the simplified nature of the calculations, the absolute values of some of the results are not significant by themselves, but they nevertheless form a useful basis for comparison of similar reactors.

Approximate methods were used to evaluate criticality and shim reactivity requirements for once-through reactor designs of various fuel enrichments and lattice spacings. The two types of reactor which were studied differed in the amounts of structural and alloying material present in the lattice. Calculations were performed for reactors having fuel enrichments of 1, 1.25, 1.5, 1.75, and 2% U^{235}, and equivalent lattice diameters of 6 to 12 in. (square pitch).

To provide a common basis for comparing the various cases, each design was required to possess sufficient reactivity to accommodate an average fuel burnup of approximately 5000 MW-d/metric ton. In each case, the core height was taken to be 0.8 times the core diameter, and a 1-ft thick graphite reflector was assumed to surround the core.

The variation of fuel inventory and core diameter with lattice spacing and fuel enrichment is shown graphically in Figs. 3-1.1 and 3-1.2 for Design A and in Figs. 3-1.3 and 3-1.4 for Design B.

Although all of the designs included on these plots satisfy criticality and burnup shim requirements, heat transfer considerations in some cases limit the range of designs that can be accepted. The shaded areas indicate designs in which maximum fuel element temperatures in excess of 1200F occur for sodium inlet and outlet temperatures of 750F and 950F. This problem is discussed below under "Engineering Considerations."

ENGINEERING CONSIDERATIONS

Heat Losses from Fuel Tubes

Preliminary calculations have been made of heat losses from the fuel tubes, assuming that stagnant carbon dioxide gas fills the barrier space and the header chambers. It has been determined that about 1.4% of the power generated in each tube is lost by conduction through the tube walls, including 0.4% in the core and 1% in the shield sections. Heat losses in the header chambers can easily be kept down to about 0.1% by providing insulation on the inside shield surface and maintaining the gas temperature near the tube temperatures. The core and shield losses would be much greater if helium were the barrier gas rather than carbon dioxide.

Reactor Pressure Drop

The sodium pressure drop for Design A has been calculated and is summarized in Table 3-1.2.

The pressure drops in Design B are lower because of the larger number of fuel-coolant tubes.

Power Limitation in Design Selection

The thermal power of the high uranium content metallic fuel elements is limited by the requirement that the fuel temperature not exceed the α-β transition temperature (1220F for unalloyed uranium). The imposition of this requirement sets an upper limit upon the maximum fuel element power, and therefore a lower limit on the total fuel element surface area.

In surveying the various designs, some designs were rejected because of this power limitation. Should power limitation become a serious factor, consideration can be given to the use of smaller fuel rods or hollow rods; in either case this would permit a higher heat release per mass of uranium for the limiting maximum uranium temperature.

Table 3-1.2 — Sodium Pressure Drop in Design A

Region	Pressure Drop, psi
Bottom pigtails*	12.8
Duct through lower neutron shield†	2.1
Fuel tube in core	4.4
Duct through upper neutron shield†	2.0
Upper pigtails	16.5
Total reactor	37.8

* The maximum velocity in the 1-in. pigtail tubes is 32 ft/sec.

† A partial plug is placed in the fuel tube where it penetrates the shield to minimize neutron streaming. Maximum velocity in this section is 15 ft/sec.

OTHER CONFIGURATIONS CONSIDERED

A broad survey was made of nine possible reactor configurations. The designs are characterized by different header arrangements or by different sodium flow schemes. Table 3-1.3 presents a rough comparison of the important features of the various proposed designs, each referred to the once through design as a standard.

Once-Through, Free Discharge

The free-discharge reactor was examined as a possible solution to the lower header room problems of fuel channel replacement in the selected design. In the free-discharge design, the coolant tubes are welded to the top gamma shield and extend to the bottom of the reactor tank, where they discharge into a pool of sodium. Above the bottom of the reactor, this design is similar to the once-through design. For safety, the D_2O tank has a double-walled construction because of the presence of the free sodium surface below the tank. Other methods of increasing the overall safety of this design are under study.

Bottom-Connected Bayonet

The bottom-connected bayonet reactor design proposes to eliminate the sodium header system above the moderator tank. In this design, the fuel elements are located in the annular region of the bayonet. Sodium flows upward through the central tube of each bayonet, reverses direction at the top of the fuel channel, and flows downward over the fuel elements to the outlet pigtail. Again, neutron shielding is provided between the core and the lower header room to prevent header activation. The bottom connected bayonet design introduces some severe mechanical problems. Accommodation of thermal expansion at the lower end of the bayonet is very difficult, and fabrication and installation of a long, steel flow divider in the channel are difficult operations. In addition, the incorporation of the steel flow divider and the additional sodium raises the parasitic neutron losses and reduces reactivity.

Top-Connected Bayonet

The top-connected bayonet is a simple inversion of the bottom-connected bayonet. It has the sodium inlet and outlet pigtails connected to the upper end of the fuel tube. By comparison

Table 3-1.3 — Comparison of SDR Design Features

	Once-Through, Top and Bottom Pigtails	Once-Through, Free Discharge	Bottom-Connected Bayonet	Top-Connected Bayonet	Bottom-Connected U-Tube	Top-Connected U-Tube	Slab	Annulus	Unitized
Fuel Enrichment Required	E	E	~1.1 E	~1.25 E	E	> E	> E	> E	> E
Complexity	satisfactory	less complex	complex	complex	satisfactory	problems	good (?)	good (?)	complex
No. of Welds	n	~n-2	> n + 6	> n + 6	n	n	-	-	-
Field Weld Accessibility	satisfactory	satisfactory, fewer field	poor	quite poor	poor	quite poor	-	-	poor, but no activity
Na Velocity	V	V	V	V	~2V	~2V	-	-	V
Na Tube Max Temp, °F	950	950	950	~750	950	950	950	950	950
Pigtails	long	short, fewer pigtails	long for top-supported tube	short	very long, large dia	medium, large dia	-	-	-
Fuel Tube Length, ft	33	25	28	20 (tube) +13 (drain)	30	20 (tube) +13 (drain)	-	-	-
Safety	-	questionable	-	-	-	-	good (?)	good (?)	-

with the bottom-connected bayonet, it has the advantage of eliminating the necessity of access under the reactor for maintenance. In addition, it presents the possibility of eliminating perforations in the bottom of the D_2O tank. This latter advantage disappears if it is found that gravity drain lines are required for each bayonet tube. In this arrangement, the upper header room will be quite congested since the inlet and outlet pigtails must be routed between the refueling extensions. Also, the fuel elements must be located in the central passage, since if they were located in the annular passage, the pigtails leading to the central passage would interfere with fuel element insertion and removal. With the fuel in the central passage, the nuclear penalty is larger than with the fuel in the annular passage.

Bottom-Connected U-Tube

In the bottom-connected, U-tube design, sodium enters the bottom of a fuel tube from an inlet pigtail, flows upward through the reactor, where it receives half of its total heat input, flows through a connecting tube to another fuel tube and then flows back down through the reactor, where it receives the remainder of its heat input. This scheme offers the advantage of locating the inlet and outlet pigtails at one end of the reactor, while retaining the nuclear properties of the once-through configuration. It has the disadvantage that the sodium velocity in the fuel tubes must be twice that of the once-through design. For a given velocity in the pigtails, the pigtail flow area must be doubled and the length required for adequate flexibility must be increased.

Top-Connected U-Tube

The top-connected U-tube eliminates the necessity of access under the reactor for maintenance. If the tubes are supported at the top, the pigtails may be shorter, since the expansion which they must accommodate is smaller. However, some congestion results from routing large pigtails between the refueling extensions. The cross-over tube between paired vertical tubes leads to problems in installation and replacement of tubes. As a result, paired tubes may have to be closely spaced, and a nuclear penalty will result.

Slab Design

In slab configurations the fuel and coolant are arranged in the form of flat slabs extending the full width and height of the reactor. Between the fuel slabs are moderator slabs in the form of flat tanks of D_2O. The reactor may contain a number of fuel and moderator slabs. In the fuel slabs the coolant may be contained between flat steel plates. Alternatively, a fuel slab may consist of a row of closely spaced fuel tubes similar to those of the once-through design. The slab design offers the possibility of simplifying the sodium inlet and outlet piping. In addition, it may be possible to use wider spacing between the fuel and the D_2O, with a resultant increase in safety. The slab design entails a nuclear penalty because the fuel disposition is unfavorable and because it leads to a higher structural poison content.

Annulus Design

The annulus concept is very similar to the slab concept except that the fuel and moderator are disposed in alternating, thin annuli. It may prove to have structural advantages over the slab design, although there may be disadvantages in sodium piping.

Unitized Design

In the unitized design, individual primary coolant pumps and intermediate heat exchangers are provided for each fuel tube. This scheme has the advantage that the unitized assembly is completed at the factory, and the connections for installation and removal are in the secondary coolant circuit, which is not radioactive. The large number of pumps and heat exchangers add considerably to the cost of the reactor.

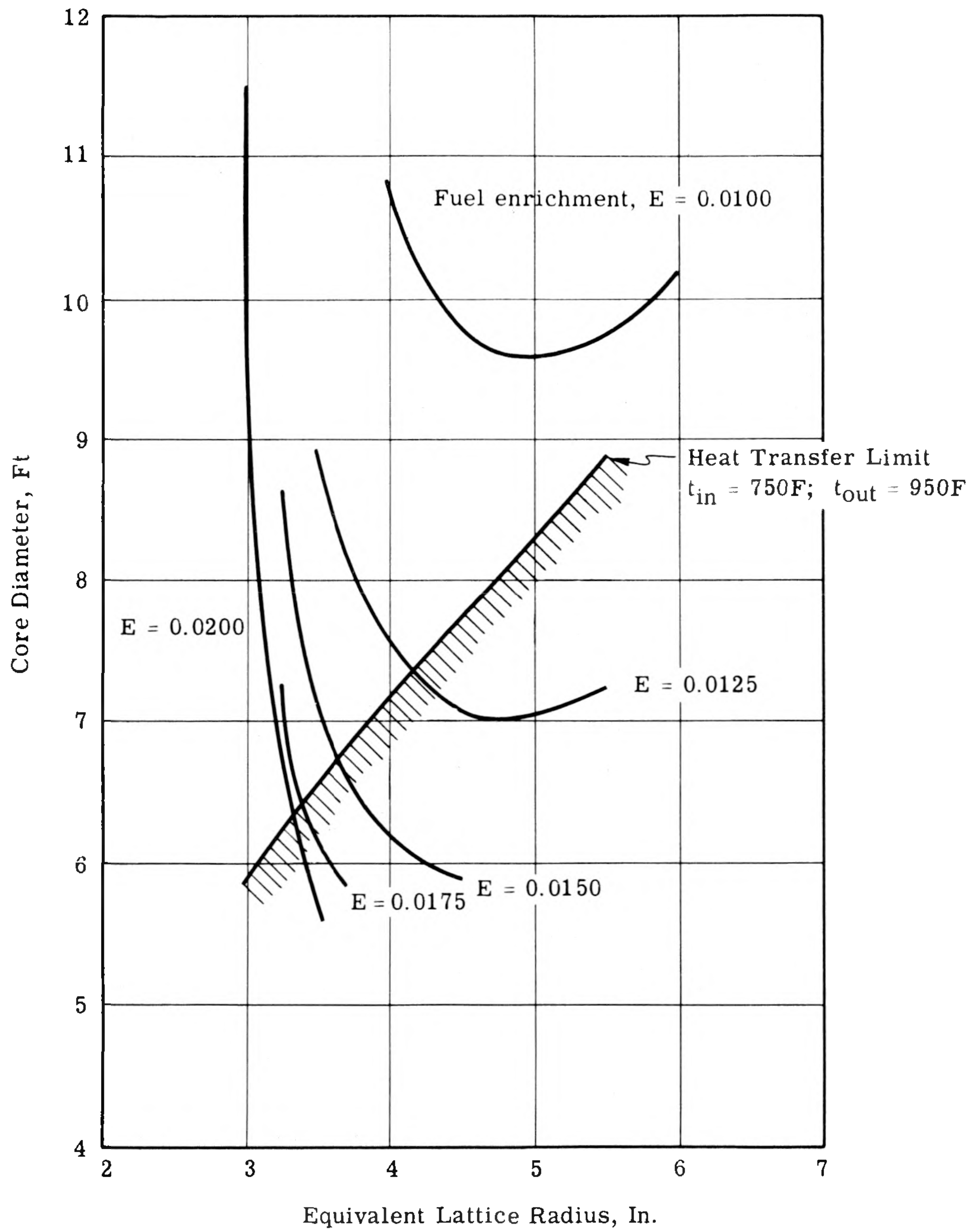

Fig. 3-1.1 — Variation of core diameter with lattice radius at various enrichments. Design A, low structural poison content. Curves on shaded side of heat transfer limit line have inadequate heat transfer properties.

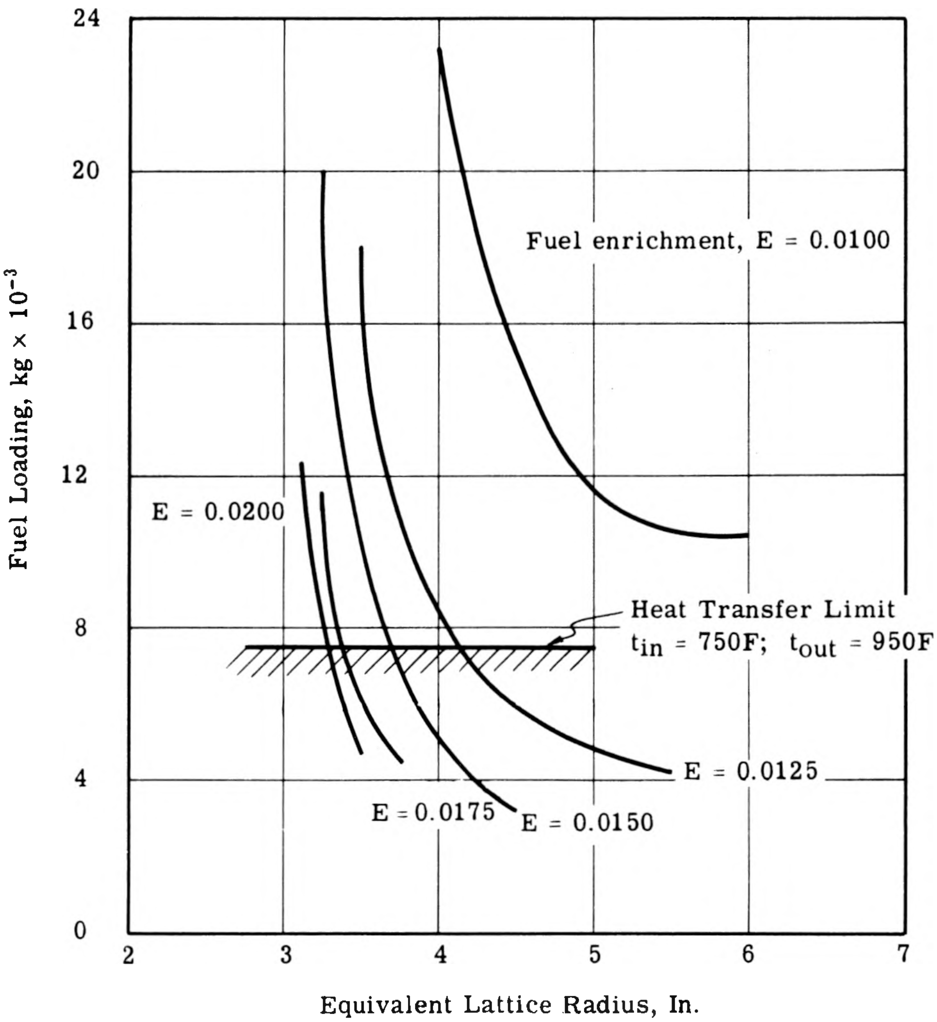

Fig. 3-1.2 — Variation of fuel loading with lattice radius at various enrichments. Design A, low structural poison content. Curves on shaded side of heat transfer limit line have inadequate heat transfer properties.

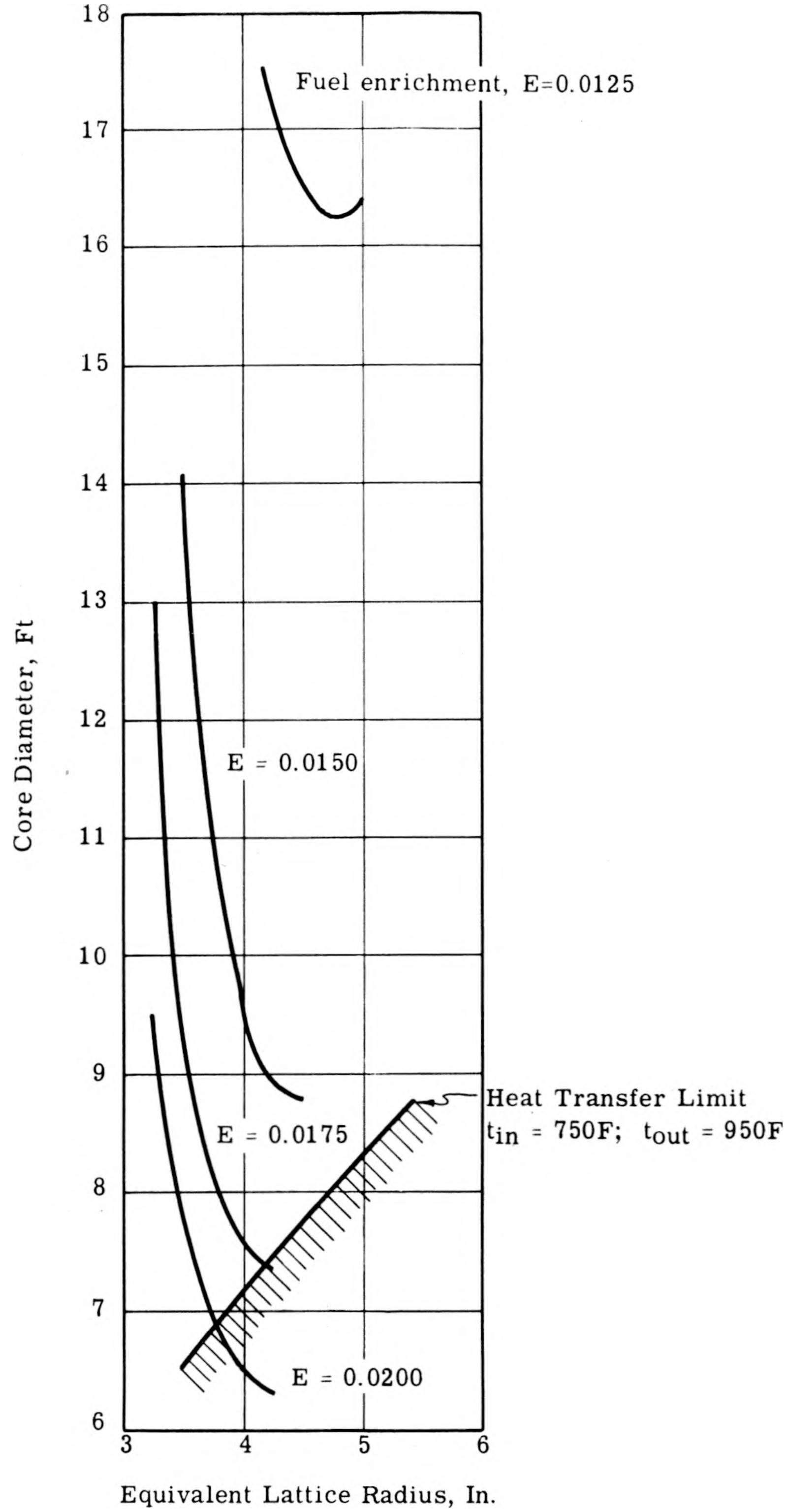

Fig. 3-1.3 — Variation of core diameter with lattice radius at various enrichments. Design B, high structural poison content. Curves on shaded side of heat transfer limit line have inadequate heat transfer properties.

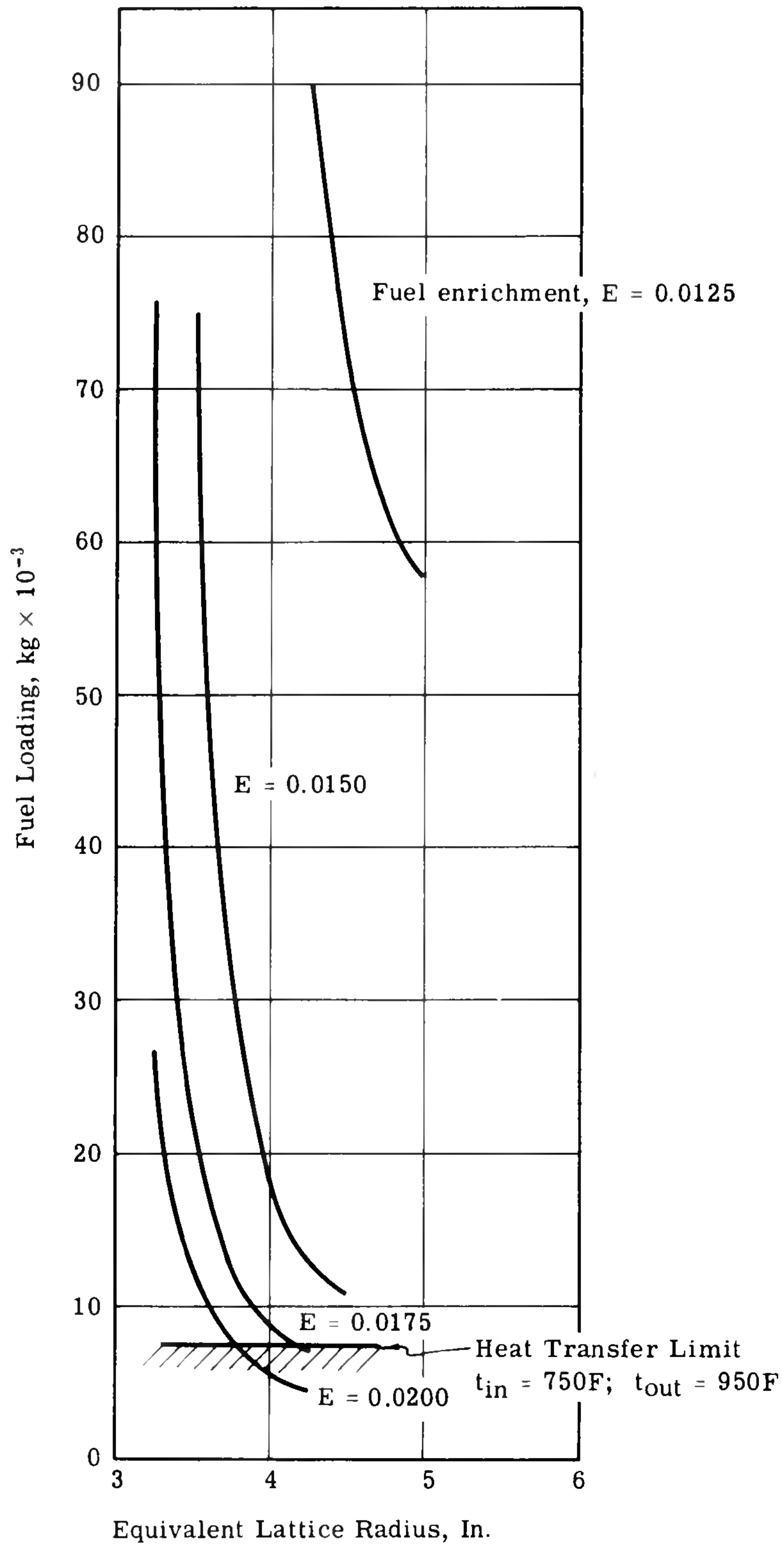

Fig. 3-1.4 — Variation of fuel loading with lattice radius at various enrichments. Design B, high structural poison content. Curves on shaded side of heat transfer limit line have inadequate heat transfer properties.

TASK 3-2 SHIELDING

Shielding studies have been concentrated on evaluating the problems associated with limitations on access to the header rooms due to radioactivity. Design of the basic reactor shield is expected to be straightforward and has not yet been started.

In the event of a fuel tube failure, access to the upper and lower header rooms will be required to remove and replace the defective tube. The radiation level in these header rooms after reactor shutdown and the allowable dose rate will control the waiting time necessary before maintenance can be performed.

The major sources of radiation which will normally be present after reactor shutdown are:

1. activated sodium remaining in the header piping,
2. steel piping activated by neutrons penetrating the shield during reactor operation,
3. steel activated within the reactor, dissolved in the sodium coolant, and deposited on the header-room piping (radioactive mass transport),
4. fission products within the reactor core. (Normally fuel elements would be removed before beginning maintenance, so that this source would not be significant.)

In addition to these "normal" sources, the possibility of fuel element failure must be considered, since some portion of the fission products entering the sodium stream would be deposited in the header room piping.

ALLOWABLE RADIATION LEVELS IN HEADER ROOMS

Although official recommendations for allowable radiation levels have been formulated,* they do not apply directly to the situation of current interest. Examination of these recommendations and supplementary information has led to the following set of assumed limiting exposures for header room maintenance:

1. a maximum hourly dose rate of 0.2 rem/hr, with a total dose received in single or closely spaced exposures of up to 1.5 rem,
2. a maximum yearly exposure of 5 rem for any worker.

The following paragraphs report the results of preliminary calculations of the dose rates to be expected in the top header room of the design described under Tasks 2-1 and 3-1. Similar results are to be expected for the bottom header room.

RADIATION FROM ACTIVATED SODIUM

The piping in each header room contains about 2200 lb of sodium when full. This sodium is distributed approximately uniformly in a cylindrical volume about 6 ft high and 12 ft in diameter. During operation, the sodium has an activity of about 0.2 curie/gram. Even after the headers have been drained, considerable residual sodium will remain not only on the pipe walls (as a film) but also in cracks, pockets, and bends. The non-draining fraction to be expected depends upon the particular configuration. If the piping is carefully designed to eliminate blind corners, small diameter

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* National Committee on Radiation Protection and Measurement, Maximum Permissible Radiation Exposures to Man – a Preliminary Statement, Radiology, 68(2): 260 (Feb. 1957).

pipes, sharp bends, etc., the non-drainage fraction may be as low as 1%.*† On this basis, the calculations summarized in Table 3-2.1 indicate that a waiting period of from 7 to 8 days is necessary to reduce the dose level in the header room from the activated sodium alone to 0.2 rem/hr and that a period of from 11 to 12 days is necessary to reduce it to 2.5 mrem/hr (the AEC laboratory tolerance dose). Some reductions in this waiting period may be achieved by flushing the system with clean sodium. Experimental results indicate that the residual radiation level may be reduced by a factor of about ten by a single flush.* The waiting time will be reduced about 2 days by this reduction in activity. Subsequent flushings are of less value. Further study of the details of the flushing procedure seems advisable.

Table 3-2.1 — Summary of Data on Sodium Activity in Header Room

Total sodium in primary system, lb	19,500
Sodium in each header room, lb	2,200
Equilibrium sodium activity, curies/lb	90
Approximate radiation level in header room due to sodium during operation, rem/hr	10^5
Reduction factor due to drainage	100
Additional reduction factor required	~5,000
Sodium half life, hr	14
Waiting period to reduce radiation level to 0.2 rem/hr, days	7.2

RADIATION FROM ACTIVATED PIPING

There are about 5000 lb of stainless steel piping in each header room, approximately uniformly distributed. The neutron flux in the top header room has been estimated for the reactor design described under Tasks 2-1 and 3-1; the resultant radiation level from activated piping immediately after reactor shutdown is calculated to be in the range of 0.4 to 0.6 rem/hr. This level decreases to about one-fourth this value within 1 day, and thereafter decreases very slowly. Small design changes can greatly reduce these values.

The approximate activity distribution among the constituents of type 316 stainless steel, when all activities are saturated, is indicated in Table 3-2.2. After the decay of Mn^{56}, most of the remaining activity is due to Co^{60}, which accounts for the slow subsequent decay. Should this activity remain a problem, consideration could be given to use of low-cobalt steel for this application.

Additional sources, which have not yet been estimated, are discussed briefly below.

1. The activity of components of the steel deposited by the sodium coolant on the header room piping walls will add to the source strength of the activated steel piping discussed above. Data are being gathered on the expected rate of transfer.

2. Another source, streaming of fission product gammas from the core through the ducts in the concrete shield, is not expected to be a serious problem, since fuel elements will normally be removed during maintenance.

3. The activity introduced by fission products entering the sodium stream and subsequently depositing on the piping walls is potentially more serious. This introduction of activity will occur

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*W. Briggeman, Personal Communication.

†KAPL-824 (Oct. 8-9, 1952). (Classified)

Table 3-2.2 — Relative Activity of SS 316 Constituents at Saturation

Isotope	Half Life	% of Total Activity
Mn^{56}	2.5 hr	73
Cr^{51}	25 d	2
Fe^{59}	45 d	1
Co^{60}	5.3 y	24
		100

only in case of fuel element failure, and to the extent that such failure can be avoided will not contribute to the header room dose. It is not reasonable, however, to assume that such a failure will never occur. Unfortunately, data are meager both on the quantity of fission products which will escape on fuel element failure and on their potential distribution within the primary sodium system. It may be worthwhile to consider the use of detectors in the sodium to discover such fission product leaks as quickly as possible. This is one means of minimizing the resulting activity. This problem is being investigated further.

TASK 3-3 EXTERNAL SYSTEMS

Brief consideration has been given to the secondary sodium system and the steam system to establish the overall system thermodynamics and to determine principal temperatures, pressures, and flow rates.

SECONDARY SODIUM SYSTEM

A simplified flow chart of the secondary sodium system is shown in Fig. 3-3.1. This system, like the primary sodium system described in Task 2-1, is a parallel loop configuration to provide for safety in case of component failure. The secondary system differs from the primary in that there is no mixing of the parallel streams; a cold trap is therefore provided in each loop. Otherwise components are similar to those in the primary system.

The secondary sodium flows at 2200 gpm in each loop, and operates between temperatures of 600F and 900F. The total volume of sodium in the secondary system is expected to be comparable to the 2600 gal estimated for the primary system.

STEAM SYSTEM

The steam system must produce steam at 850 psig and 850F for the main turbo-alternator. This machine must provide 10,000-kw net electrical output. The name plate rating of the generator will be 12,500 kw. About 105,000 lb/hr of steam is required.

The problems of steam generation are being given immediate attention because sodium-heated steam generators have encountered operating difficulties from high thermal stresses and corrosion. The relatively large temperature difference between the saturation temperature of 850 psig steam (525F) and the maximum sodium temperature (900F) requires careful design to avoid excessive local temperature differences and heat fluxes in the steam generator.

The steam generator systems which are presently of interest are: (1) a separate boiler and superheater system similar to that used in SIR, (2) a once-through boiler system similar to that installed at the SRE, and (3) a "Loeffler System."

The SIR system requires great care in design to avoid excessive thermal stresses in the evaporator, because of the large temperature differences which may exist there. However, this design permits easy control of corrosion and scale formation.

The once-through boiler system has the advantage of simplicity, but also requires careful design to avoid high thermal stresses, and in addition requires considerable care in operation to control corrosion and scale formation.

In the Loeffler System, saturated steam is superheated by sodium in a superheater and is then divided into two streams, one of which goes to the turbine and the other to the evaporator, where water is evaporated by direct injection of the superheated steam. This system provides a means for limiting thermal stresses, scale formation, and corrosion but requires the use of a mechanical blower to circulate the steam through the superheater. This blower requires considerable power, and special development may be required to produce a suitable unit.

Provisions will be made to dump steam to the turbine condenser if emergency cooling of the reactor should become necessary.

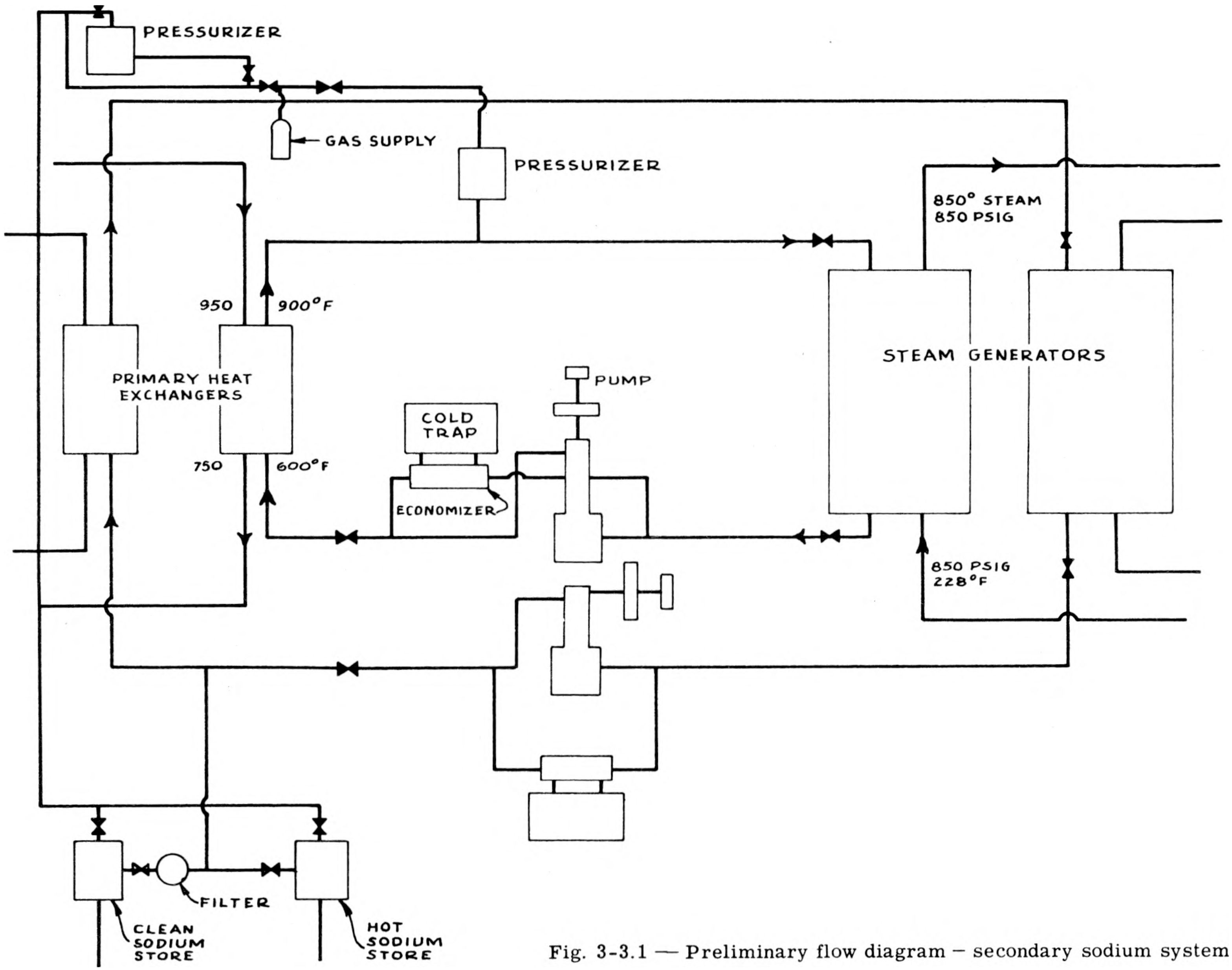

Fig. 3-3.1 — Preliminary flow diagram – secondary sodium system