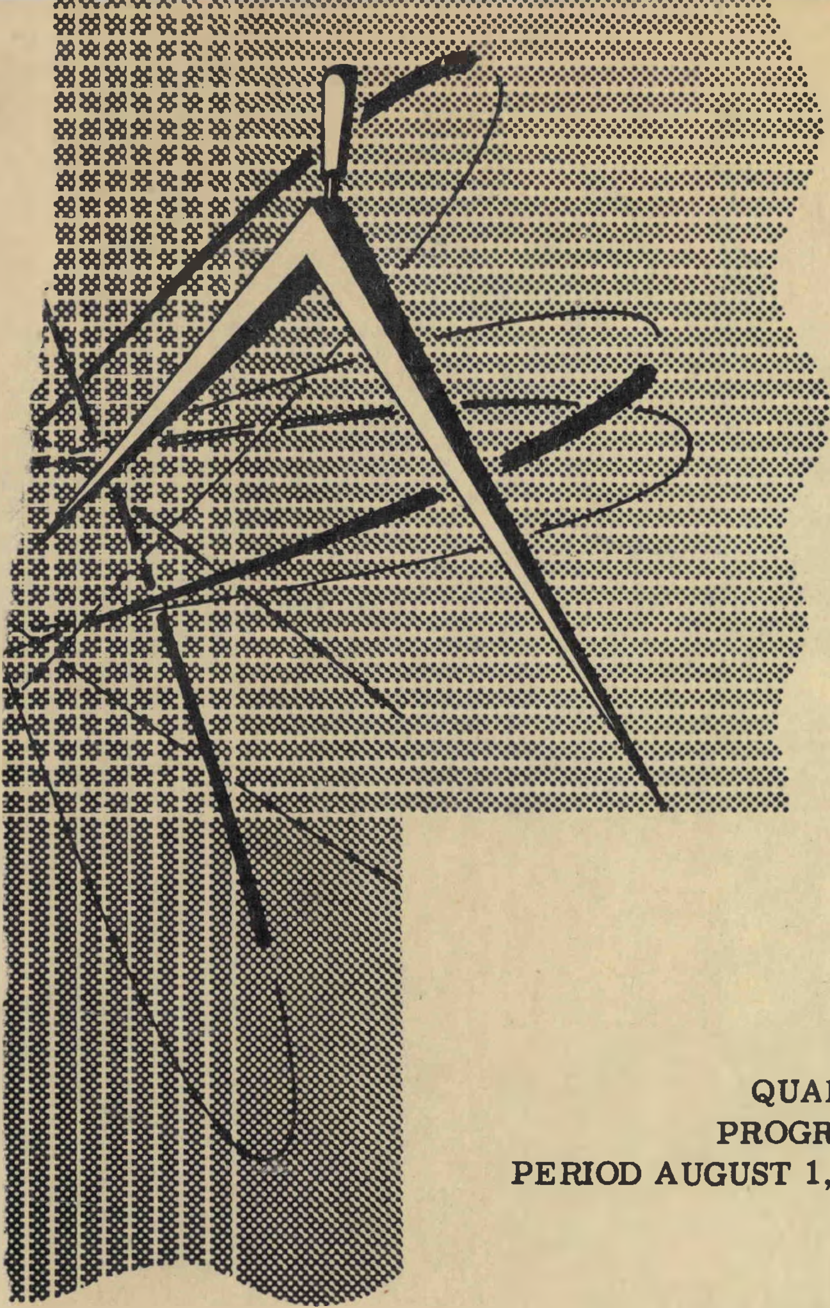
NDA 084-3



SDR PROJECT QUARTERLY TECHNICAL PROGRESS REPORT FOR THE PERIOD AUGUST 1, 1957 THROUGH OCTOBER 31, 1957

DECEMBER 31, 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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INTRODUCTION

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

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

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

The latter two areas, which include substantially all of the technical effort, have been divided into a number of "Tasks." This report discusses progress on each of the Tasks.

A summary list of installations visited during this quarter is appended.

The preceding report in this series, NDA 84-2, covered the period May 1, 1957 through July 31, 1957.

SUMMARY

SODIUM-D_2O SEPARATION

During this report period, engineering design and experimental studies have been continued, with the objective of demonstrating the feasibility of separating sodium and D_2O in the Sodium Deuterium Reactor (SDR).

Fabrication methods for manufacturing SDR fuel-coolant tubes were examined; there is every indication that the design is feasible, and a good possibility that the tube can be formed in one piece, without welds. D_2O calandria design was backed up by visits to Chalk River and Savannah River. A modified barrier arrangement, simpler than that discussed in NDA 84-2, has been studied and will be evaluated further. A sodium leak detection system has been outlined.

Detailed design is proceeding on both static and fatigue rigs for testing the mechanical integrity of fuel-coolant tube and header joints.

Screening tests on barrier materials were essentially completed, using the single-failure test apparatus at the NDA Pawling Laboratory. Results on aluminum were especially encouraging; two samples of 0.060-in. thick aluminum were exposed to a jet of 950F sodium for more than 16 hr, and both survived. Design work has been completed, and fabrication is now proceeding, on the multiple-failure test apparatus, in which both water and sodium system failures can be simulated at the same time. Detailed design is progressing on the mock-up test, which consists of three fuel-coolant tubes and associated barriers in a tank of water; this test will be used to demonstrate the reliability of integrated sodium and water circulating systems under simulated reactor operating conditions.

PRELIMINARY DESIGN

Additional analysis was done on the straight-through configuration, for two types of fuel element — a U–10 weight % Mo alloy, and UO_2. An alternative design concept, based on a slab configuration, was also studied; the slab design has some potential maintenance and safety advantages as compared with the calandria (or lattice) configuration. Shielding studies continued to concentrate on after-shutdown radiation levels in the header rooms.

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SODIUM-D_2O SEPARATION

TASK 2-1 SODIUM SYSTEM ENGINEERING

FUEL-COOLANT TUBE AND HEADER DESIGN

The through-tube reactor arrangement discussed in NDA 84-2* continues as the basis for design and experimental studies. Many aspects of the conceptual design have been further detailed, and modifications have been made in some design areas.

A basic aim in the design of the reactor has been to make the sodium-containing fuel-coolant tubes easily replaceable. Also, the tubes of the D_2O calandria should be replaceable, but not necessarily as easily. The repairability must not, of course, impair the reliability of the tubes.

The current reactor arrangement is shown schematically in Fig. 2-1.1.† The fuel-coolant tubes have been changed from the square arrangement shown in NDA 84-2 to a triangular arrangement which has the same ratio of moderator volume to fuel tube volume (Fig. 2-1.2). In addition, revisions have been made in the design of the fuel tube and barrier supports, and in the calandria design.

The distance between fuel tube centerlines is 8.6 in. and the distance between tube rows is 7.45 in. The triangular arrangement results in a slight saving in over-all reactor diameter and an increase in the minimum ligament between holes in tube sheets and shield. This arrangement provides for 102 fuel elements and 19 control rods in lattice positions.

The fuel tube removal system (by which the outlet pigtail is rotated to clear the pigtails above) is still feasible with the triangular arrangement, but is made somewhat more difficult by the smaller distance between rows.

The dimensions and numbers indicated above are preliminary design estimates, and are subject to change as the physics and engineering studies proceed.

Operational Aspects of the System

Thermal Stress at Pigtail Weld

An estimate was made of the transient thermal stresses that might occur in the weld between the fuel-coolant tube and the pigtail. Such stresses would be most severe during a startup or during a reactor scram from high power. It was found that the maximum thermal stress that would be developed is about 6,000 psi, considerably less than the high temperature yield stress of the

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* SDR Project, Quarterly Technical Progress Report for the Period May 1, 1957 through July 31, 1957 (Sept. 30, 1957).

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

stainless steel. The existence of a thermal shock problem at the pigtail welds is therefore not very likely.

Thawing of Sodium in Fuel Tubes

In the event that sodium becomes completely frozen in the coolant channels, it would be advantageous to provide a method for melting it within the reactor core. This could be done in a relatively convenient manner by circulating a hot gas between the header rooms through the inner gas space of the fuel tube barriers. The circulation of 2000 ft^3/min of 300F CO_2 would melt the sodium in about 4 hr. This would require about 1/2 psi pressure difference between header rooms, and a blower rated at about 12 HP.

Fuel-Coolant Tube Fabrication

The basic design of the fuel-coolant tube as discussed in NDA 84-2 has not been changed, but additional fabrication techniques have been examined. Proposals have been received from several manufacturers to fabricate sample fuel-coolant tubes for (1) demonstrating feasibility, (2) establishing inspection procedures, and (3) mechanical testing. The proposals indicate that the tube can be made either in one piece or by welding several sections together.

The proposal to form the tube in one piece is being reviewed to determine whether the straightness, thickness, and roundness tolerances achievable by this process (the Rockrite process) will be adequate for the reactor. This process is a standard commercial process which cold-sizes tubing. NDA has received samples of tubes with gradual steps in wall-thicknesses from Tube Reducing Corporation, Wallington, N.J. These tubes, which approximate SDR diameter and thickness requirements, were made with already existing tooling.

For the tube fabricated by welding together several sections, it is proposed that the very thin-walled section be prepared from a thicker tube by a centerless grinding (or possibly a turning) process, to permit the welds to be made at a heavier wall thickness at the ends of the section. Proposals for fabricating tubes by such a process are also being reviewed. Various welds which may be used to join tube sections have been designed and preliminary welding specifications have been prepared for the testing program under Task 2-4.

EXTERNAL PRIMARY SODIUM SYSTEM

During the quarter, studies were made on some operational aspects of the sodium system relating to startup, scram, and emergency cooling. The survey of available components was continued, and the merits of various system subdivision schemes were examined.

System Subdivision

After a scram occurs, a means for removal of the fuel afterheat must be provided. The provision of parallel multiple coolant circuits assures afterheat cooling despite failure of a major piece of equipment in the coolant system; this would not be possible with a single sodium or primary circuit. For reasons of economy of procurement and maintenance, provision of circuits of identical capacity and size of equipment appears preferable at this time.

Whether the primary circuit duty should be shared between two, three, or more identical parallel loops will depend on a detailed economic analysis of actual equipment and construction costs, but it seems probable that capital costs would be least if only two equal primary systems were provided.

Emergency Cooling Studies

Analytical studies are in progress to determine the emergency cooling necessary to prevent a fuel meltdown in the event of a rupture of either the upper or lower pigtail connection to a fuel tube. These cooling requirements will be incorporated into the process system design to insure, insofar as possible, that adequate emergency cooling is available. The work to date has shown that if detection of a major rupture (such as a broken pigtail) and consequent initiation of a scram take place within 10 sec or less, there will be an additional period of at least 5 min available during which emergency cooling provisions must be initiated to prevent serious damage and fuel meltdown. (There is the possibility that overheating and accompanying solid-state phase change may occur within some fuel elements if the emergency cooling cannot be started as early as one minute after a major rupture, but this should not lead to a serious hazard.) It is felt that the time requirements for prevention of serious damage can be met.

Pumps

The survey of existing sodium pump technology, mentioned in NDA 84-2, has continued.

A visit was made to Atomic Power Development Associates to discuss the sodium pumps which APDA has ordered for its fast breeder reactor. The pumps are vertical shaft, sump-type, centrifugals; they are being built for APDA by the Byron Jackson Company. These pumps have substantially larger capacity than is required for the SDR system.

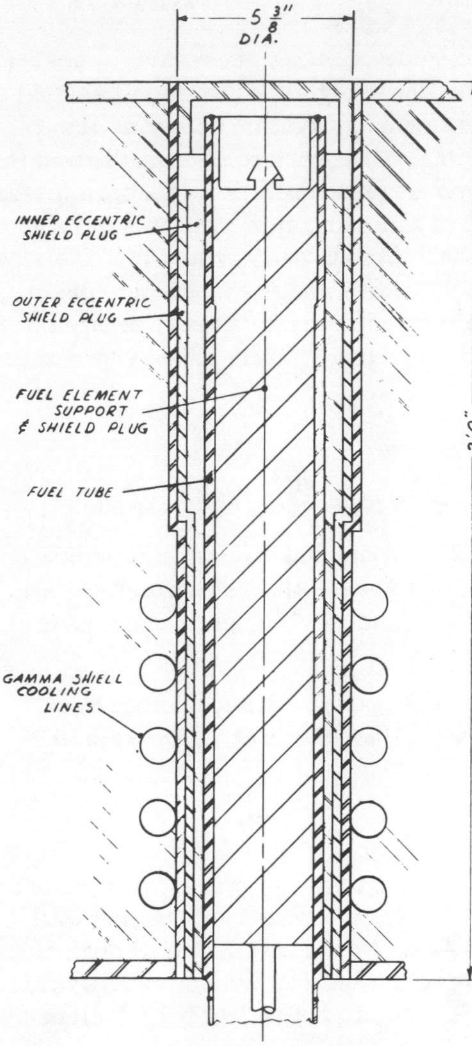
The EBR-II centrifugal pumps, which are now under test at Argonne National Laboratory, appear more suitable for SDR application. These pumps develop about 50 psi at a flow rate of 2700 gpm.

Steam Generator

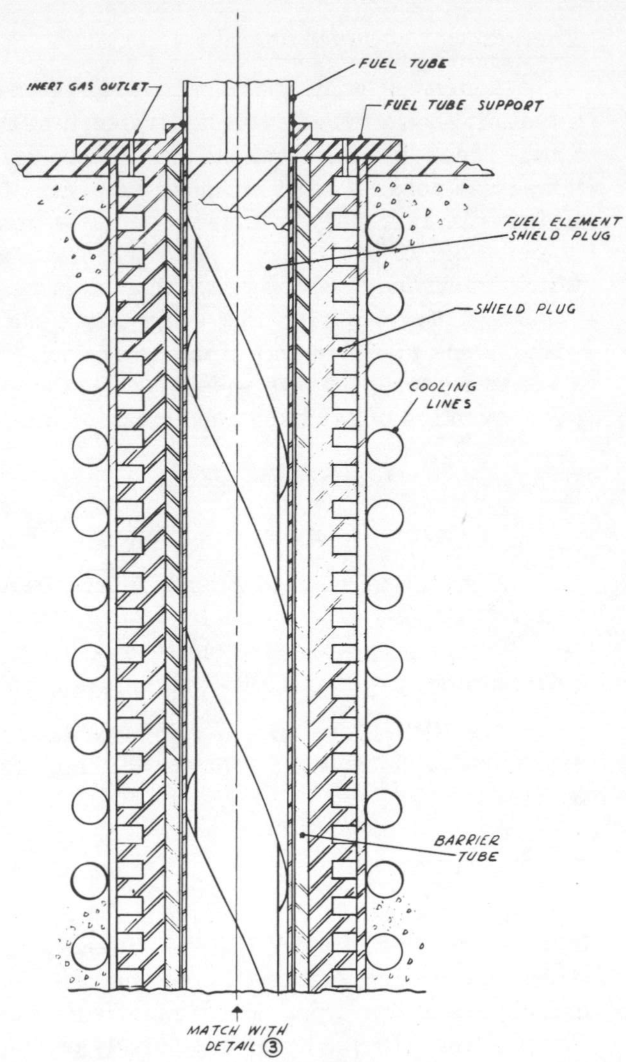
APDA is sponsoring the design of a sodium-heated, once-through steam generator in which there is no "third fluid," the sodium and water being separated only be a single metal tube wall. APDA is obtaining test results on sodium-water reactions resulting from leaks in small heat exchangers of this type. Their experience to date is that the reaction effects are not catastrophic. The pressure does not rise explosively during the reaction, and a rupture disk has been included to prevent damage to the equipment.

Heat Exchangers

An intermediate heat exchanger being built for APDA by Alco Products, Incorporated will transfer about 490,000,000 BTU/hr between sodium streams of about 12,500 gpm each, at about 35F terminal temperature difference. While this heat exchanger is considerably larger than the 2700 gpm exchanger mentioned for SDR in NDA 84-2, the same design approach used in the Alco exchanger is applicable to SDR needs.

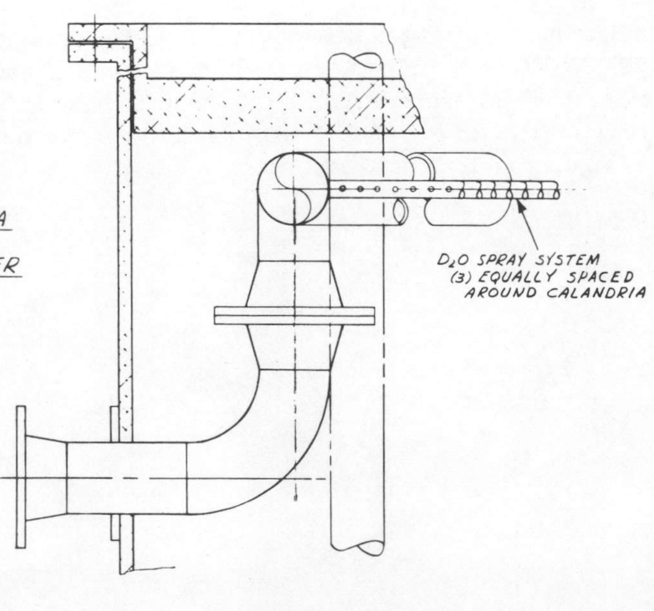


(1) TOP OF FUEL TUBE



(2) TOP OF UPPER NEUTRON SHIELD

(6) TOP CALANDRIA TUBE SHEET & D_2O INLET HEADER



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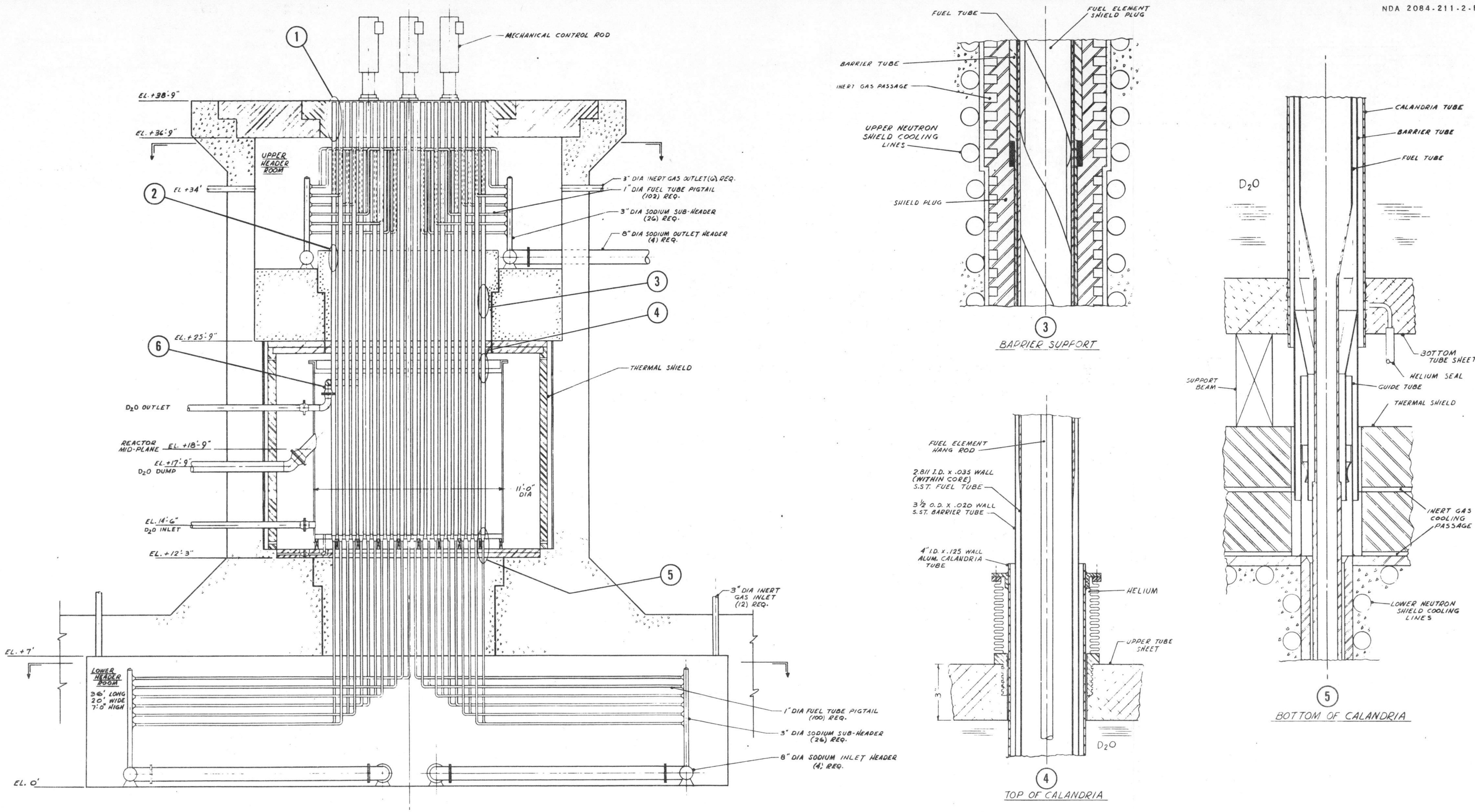

Fig. 2-1.1 — SDR arrangement – elevation and details

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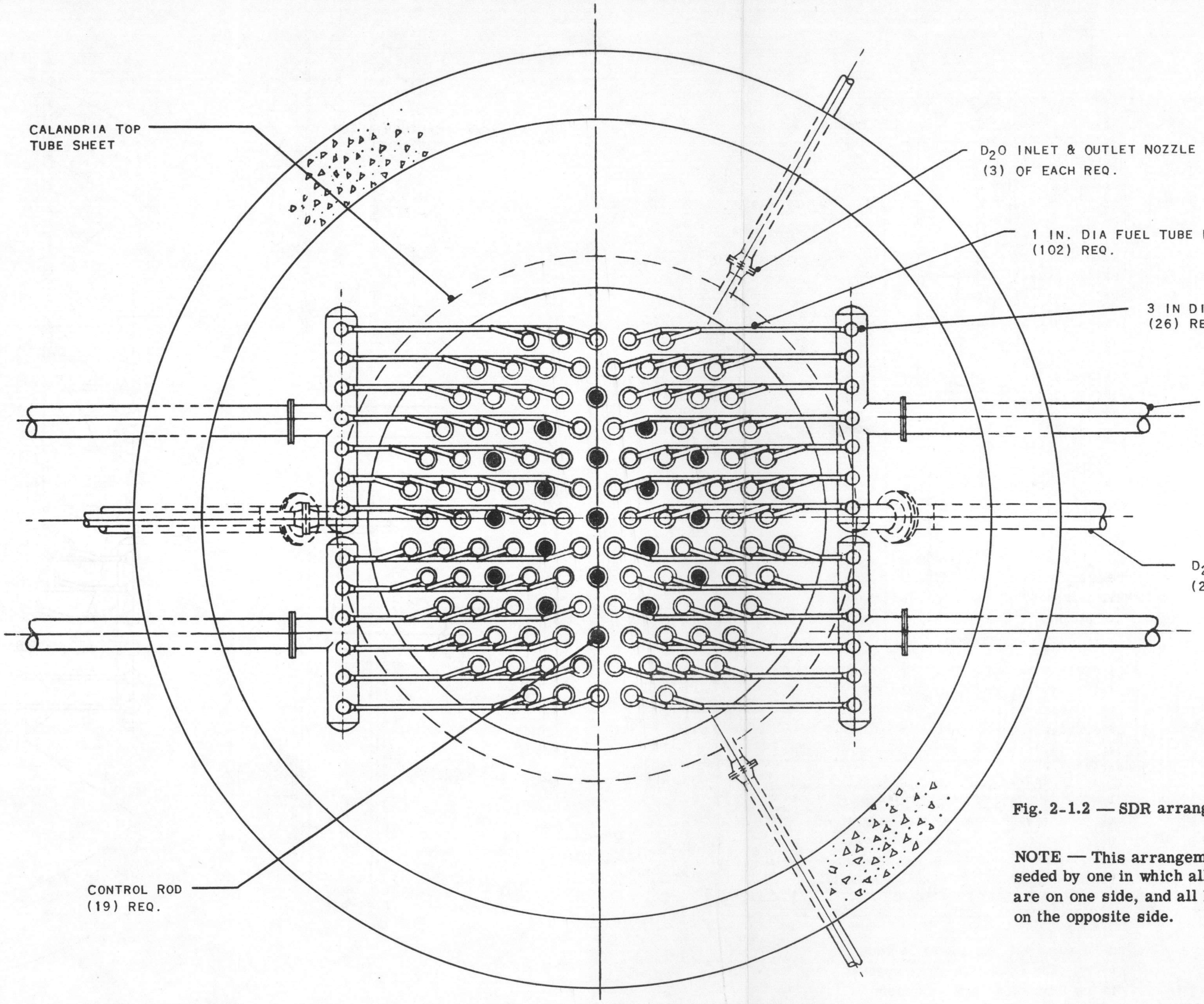

Fig. 2-1.2 — SDR arrangement – plan

NOTE — This arrangement may be superseded by one in which all sodium connections are on one side, and all D_2O connections are on the opposite side.

TASK 2-2 D_2O SYSTEM ENGINEERING

In conjunction with the preparation of a more detailed layout of the straight-through reactor, work was started during this quarter on the design of the calandria. A description of the calandria as currently conceived is given below and its design is shown in Fig. 2-1.1. During this quarter work was also begun on the external D_2O system. Information obtained through visits to Atomic Energy of Canada, Limited (Chalk River) and Savannah River helped in substantiating the design effort on the D_2O system.

CALANDRIA DESIGN

The D_2O calandria is a cylindrical aluminum vessel about 11 ft in diameter and 10 ft high. The outer shell is $^3/_4$ in. thick, and the bottom and top tube sheets are each 3 in. thick.

One hundred and twenty-one aluminum tubes, $4^1/_4$ in. OD $\times$ $^1/_8$ in. wall thickness are located on a triangular pitch with a center-to-center spacing of 8.6 in.

Rolling-in the tubes in the lower tube sheet is being considered, based primarily on AECL (Chalk River) experience with the NRX calandria, in which helium-leak-tight joints were obtained by rolling.

An alternate design is being considered using pipe stubs added to the lower tube sheet, with a lip weld made between the calandria tube and the stub after rolling. Since one of the requirements is that the tubes must be replaceable from above the reactor, such a weld could not be made on the replacement tube, which would have to be rolled in. An annular groove around the tube at the middle of the lower tube sheet might be provided so that gas pressure could be used to back up either method of fabrication and prevent leaks.

The method of attachment of the tubes to the top tube sheet has not yet been definitely selected. Fig. 2-1.1 shows the top tube sheet bolted to the calandria shell and seal-welded. This design is intended to simplify initial construction and major maintenance. Further consideration may indicate that a welded head is preferable and that the removable feature is unnecessary.

Both the top and the bottom tube sheets may be cooled by circulating water through small tubes brazed to the outside surface. The inner surfaces may be cooled by the D_2O within the calandria. Inlet pipes with spray nozzles are located in the gas space above the D_2O. These sprays will cool the top tube sheet and the upper half of the tubes if the D_2O has been dumped to the core mid-plane during an emergency. The spray system will be fed from three separate headers, each of which sprays the entire vessel.

During an emergency dumping of the D_2O, stresses resulting from constraining the thermal expansion of the relatively-uncooled calandria tubes might be significant. If about 180 lb/hr per tube of 100F gas is circulated between the barrier and the calandria tube after the moderator has been dumped, it will hold the tubes below 300F, and the thermal stresses in the tubes and the tank walls will not be damaging. There would probably be no need to include expansion bellows on each tube, although further analysis of possible bowing of the tubes is necessary before a decision on providing bellows can be made.

Two large dump lines have been provided in the calandria to allow the reactor to be shut down rapidly if the mechanical safety rods do not function properly. The D_2O inlet spray system would continue to operate after such a dump in order to cool the top tube sheet and the upper part of the tubes. Lines for complete draining of the calandria are located in the bottom tube sheet in the reflector space. Nozzles in the top tube sheet above the reflector space are provided for the cover gas system and for sampling and instrument lines.

The cover gas over the D_2O in the calandria will probably be helium, which has been used successfully at Chalk River and Savannah River. Nitrogen cannot be used since it could result in formation of nitric acid, which would corrode the aluminum.

EXTERNAL D_2O SYSTEM

The requirements of reactor external D_2O system have been outlined, based to a large extent on successful practice on D_2O systems at Chalk River and Savannah River. Fig. 2-2.1 presents a preliminary flowsheet for the SDR external D_2O system. Some of the features are:

1. The D_2O storage tanks and fill pump to hold the original charge of D_2O and to supply make-up as needed.
2. The dump tank and valve for emergency removal of the heavy water from the calandria.
3. The D_2O heat exchangers and the cooling water recycle pump, which keep the mean temperature of the moderator in the calandria at the prescribed value, regardless of variations in cooling water temperature.
4. The ion-exchange beds which maintain chemical purity and an electrolytic cell which maintains isotopic purity of the D_2O. (The beds are located in a bypass loop carrying ½% of the main D_2O circulating flow. The ion-exchange loop contains a heat exchanger which will reduce the D_2O temperature to a level which will not damage the ion exchange resins, and which will also probably contain the pH control equipment.)
5. The helium cover gas system which maintains a blanket of helium (pressure of 3 to 5 in. of water) over the free surface of the D_2O in order to exclude air and atmospheric moisture. (Also shown is an expansion chamber connected with the helium space above the main body of D_2O, for cushioning the system. The supply cylinders of compressed helium, the pressure reducers, and the helium purification equipment are not shown.)
6. A recombination catalyst system which recombines the small amount of heavy water normally dissociated into deuterium and oxygen in the reactor. (To avoid drowning the catalyst, this system also has a small condenser which removes D_2O vapor and entrained mist.)
7. The drainage system (not shown in detail) which will consist of an array of pipe lines and tanks to receive D_2O which may have to be drained from plant components temporarily disconnected for repair or replacement. (Several tanks may be needed to isolate contaminated portions of D_2O.)

The D_2O leak detection system has not been included on the D_2O system flowsheet.

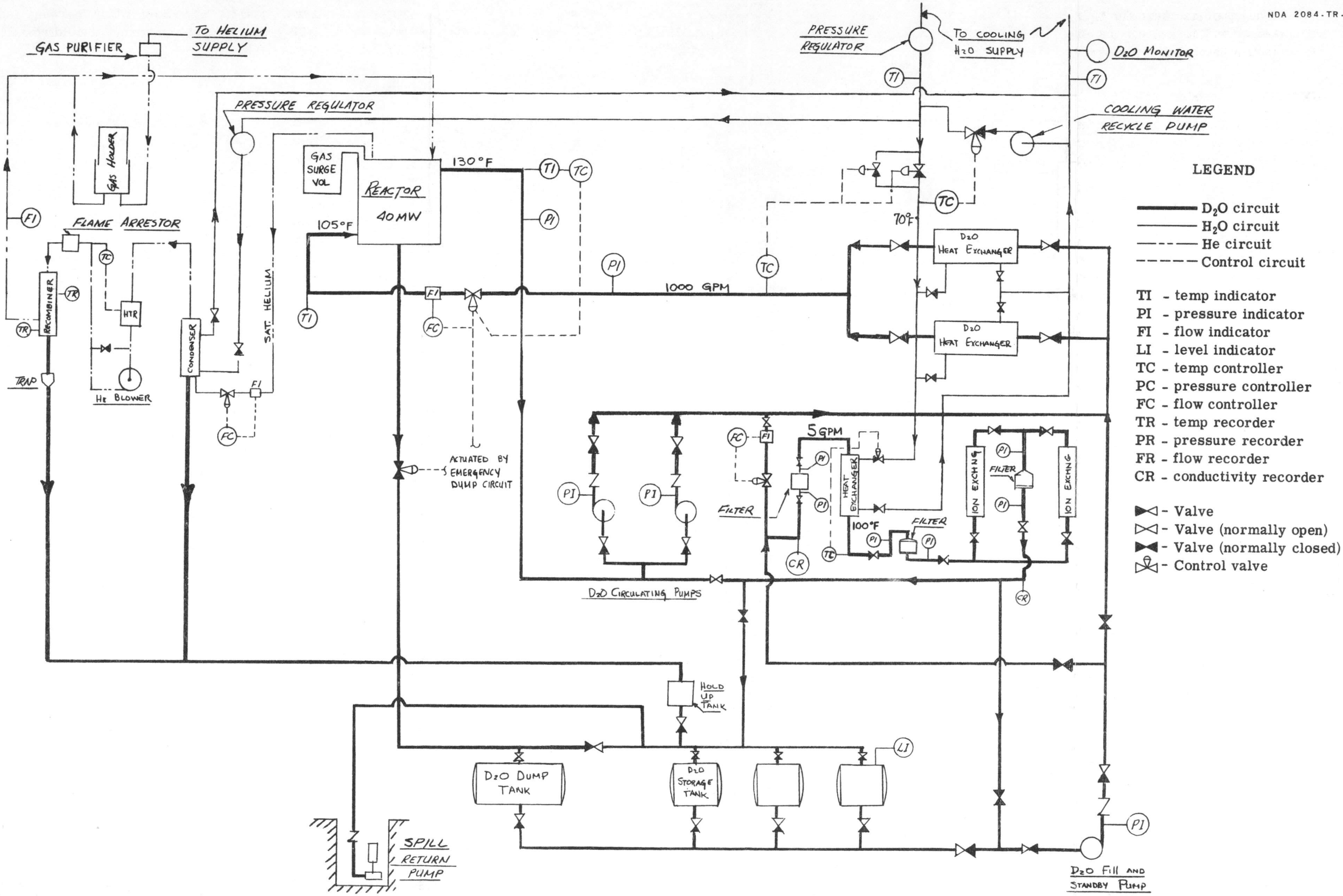

Fig. 2-2.1 — SDR — preliminary D_2O flowsheet

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TASK 2-3 BARRIER SYSTEM ENGINEERING

During the quarter, barrier design work was concentrated on mechanical design of the means of supporting the barrier and on sodium leak detection in the reactor.

MODIFIED BARRIER ARRANGEMENT

The barrier design described in NDA 84-2 consists of a graphite sleeve in a sealed region between the fuel-coolant tube and an outer barrier wall (of zirconium or stainless steel). An alternate design has been developed, in which the graphite sleeve is eliminated and the barrier tube is open at the bottom end. While a final choice has not been made, the new, simpler design has been given much impetus by the results of the barrier test program (Task 2-5). The tests have demonstrated that a single aluminum barrier stands up very well against a jet of 950 F sodium.

The alternate barrier design consists of an open-end tube, suspended from the top biological shield, hanging through the calandria tube, and laterally positioned at the lower biological shield. The fuel-coolant tubes slip inside these hanging barrier tubes. The arrangement at the bottom is shown in Fig. 2-1.1, item 5.

The barrier tube is 3.50 in. OD, with a 0.060 in. wall of aluminum or zircaloy, or a 0.020 in. wall of stainless steel. The barrier tube is supported at the upper concrete shield by a retainer ring which is welded to the tube and which sits on a step in the shield. A shield plug inserted between the barrier tube and fuel tube prevents the barrier from moving, and provides shielding in the space between the fuel tube and barrier tube. At the bottom shield, the barrier tube is prevented from vibrating or moving laterally by means of leaf springs.

In case of a fuel-coolant tube failure, sodium leakage is channeled down through the lower biological shield into the lower header room, where it is collected in catch pans. Liquid sodium detectors are placed in the outlet pipes from the lower sodium catch pans. In order to monitor individual fuel channels, gas sampling tubes may be introduced at the top of the barrier space. These tubes would be connected to a detection system similar to that described in this section. Some of the D_2O leakage striking the barrier wall would flash to steam, and remaining liquid water leakage would be collected in a catch pan placed on the upper surface of the lower shield and then piped out of the reactor compartment.

This alternate, open-end barrier design is considerably simpler than the totally sealed barrier. The barrier tube is a completely unrestrained member requiring no welds to any structural piece, and therefore can be fabricated from any ductile metal without regard for its weldability to stainless steel. This change gives a more simply fabricated fuel-coolant tube. The open-end barrier design also provides for removal of sodium from the reactor in the event of a fuel tube failure during operation, reducing the hazard of a D_2O-water reaction occurring at some later time. The primary disadvantage of the open-end design is that a large failure would drain a large quantity of sodium from the system, instead of just filling one sealed barrier space. However, a constriction may be provided at the opening to limit the rate of sodium flow.

LEAK DETECTION SYSTEM

A survey of sodium leak detection methods indicates that:

1. the most suitable internal detectors (where detection occurs at or near the leak site inside the reactor) are of the electrical conduction type that would be shorted electrically when contacted by sodium, and

2. the most suitable external detectors (where detection occurs at a distance from the leak and external to the core) are based on detection of sodium vapor carried by an inert gas stream.

Because of the importance of sodium leak detection, a system has been designed using both types of detector, to insure reliability and to provide a method for verifying the signal and for checking one detector against the other.

Description of System

The sodium leak detection system (Fig. 2-3.1) consists of a liquid detection path and a vapor detection path interconnected to reduce spurious leak signals. A liquid sodium detection path with a spark plug-type detector at the bottom of the barrier space detects leakage channeled down the barrier wall of each individual fuel tube. The external circuitry for this detector is simple, requiring only a power supply and a current detecting device. The term "spark plug" is intended to describe a class of possible detectors including shortable graphite sleeves, shortable condensers, etc.

The vapor detection system uses a gas sampling scheme, developed by Atomics International.* Gas samples are cyclically drawn from each barrier space by a pump to a gas manifold through individual gas lines. The samples then pass through sodium vapor detectors and sodium traps, and then are either vented or returned to the reactor gas system.

Operation of System

Under normal operating conditions, a routine leak check of each barrier space is performed at least once during each operating shift, using the vapor detector. The presence of sodium will be indicated by an alarm actuated through an alarm integrator which accepts and interprets all signals from both detector systems.

If a leak should develop between the routine check cycles, the presence of liquid sodium will be indicated by the spark plug-type detector. This signal will actuate the vapor sampling system, which will check the suspected barrier space by removing a gas sample which will check for sodium in the barrier space. This method of leak verification is needed since the spark plug detector may be subject to spurious shorting of the electrodes or leads.

To monitor a channel with a shorted spark plug, the vapor detector will continuously sample a barrier space until the leak is verified. A problem expected with the vapor detector is occasional false alarms occurring from trapped or condensed sodium in the carrier gas line after it has been exposed to sodium vapor. Replacement of the sampling tube at the time the leak is repaired should eliminate this difficulty; this replacement will be relatively easy to accomplish in the present design. Other expedients such as heating or flushing the sampling tube and manifold can be considered.

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*Sodium Graphite Reactor — Quarterly Progress Report — October–December, 1954, NAA-SR-1292 (May 15, 1955).

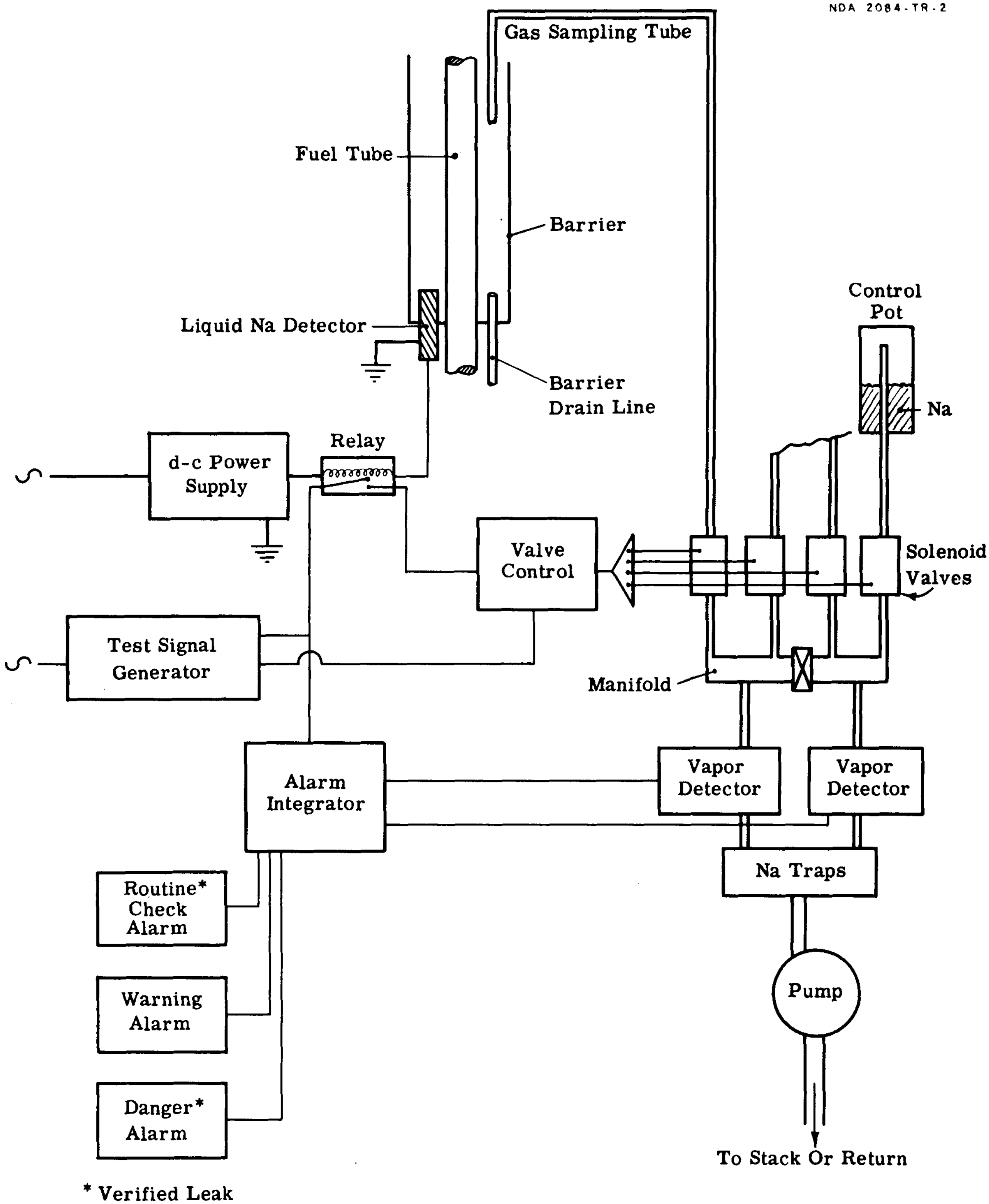

Fig. 2-3.1 — SDR – sodium leak detector system – schematic

TASK 2-4 FUEL-COOLANT TUBE AND HEADER TESTS

The SDR fuel-coolant tube and header test program, as previously reported in NDA 84-2, had been divided into three main parts: (1) static flexure tests, (2) mechanical cycling tests, and (3) thermal cycling tests. Parts (1) and (2) of the program remain unchanged. However, an analysis (see Task 2-1) of the temperature variations to be anticipated in the event of a reactor scram has shown that there will not be severe thermal shock conditions at the pigtail joints, and consequently the thermal cycling test phase of the program has been deleted.

In the static flexure and mechanical cycling tests, the major interest is to determine the ability of the austenitic stainless steel piping and weldments of various designs to withstand strain cycling at elevated temperatures under conditions simulating reactor operation.

In preparation for the testing phases of the program, a weld development effort has been initiated, with several different types of weld design under study.

The fatigue and static testing apparatus for this program is being designed around a modified shaper mechanism. A structural steel frame supports the shaper and holds the test specimen by means of a suitable grip-type fixture. The free end of the specimen is deflected by the ram of the shaper through an SR-4 load-measuring device and a load-transmitting end piece. Modification of the drive system by means of a jack shaft and compound step-down pulleys (or other suitable means) is necessary to reduce the shaper speeds to those of interest in the test program.

TASK 2-5 BARRIER TESTS

SINGLE-FAILURE TESTS

Single-failure tests are being conducted in the test apparatus described in NDA 84-2 to determine the effect of sodium leaks on possible barrier materials. The apparatus is a pumped sodium loop, which includes a test chamber, surge tank, EM pump, EM flowmeter, sodium heater, and stainless steel piping. As shown in Fig. 2-5.1, the entire apparatus is supported in an open frame. The loop has been in operation at the NDA Pawling Laboratory, and screening tests on barrier materials have been essentially completed.

Nine runs have been made, with a total of 50 specimens tested. All of the samples tested were in an argon atmosphere and were impinged upon by a jet of 950F sodium coming from a $^{3}/_{32}$-in. diameter nozzle with a jet velocity of about 50 fps. In all but two of the tests, the sodium jet duration was 15 min.

A list of materials and test results is given in Table 2-5.1. The most significant finding is that none of the scheduled aluminum samples failed at 950 F, including two samples which were exposed to the sodium jet for 16 hr or more. (Fig. 2-5.2.) The only failures of aluminum samples were found (Fig. 2-5.3) in two added tests using 0.005 in. aluminum foil (instead of the standard 0.060 in. barrier thickness).

Reactor grade graphites, in general, stood up well. Failures, where they occurred, may have been due to mechanical restraints caused by swelling of the graphite when exposed to a sodium atmosphere. (Fig. 2-5.4.)

Zirconium and stainless steel (Fig. 2-5.5) stood up well. There was some surface staining of the zirconium and stainless steel as a result of cleaning the specimens following the tests. These materials are available as alternates to aluminum, but they are less desirable because of cost and neutron absorption, respectively.

AGOT (reactor grade) graphite, a dense material, stands up well. A photograph of AGOT graphite exposed for 15 min is shown in Fig. 2-5.6, with an unexposed control specimen shown for comparison.

A second series of tests is scheduled to study the effect of exposure of aluminum to a jet of higher temperature sodium (up to 1150 F).

With repeated removal of the head, the test chamber walls have become coated with oxide. A steam cleaner has been set up for cleaning the loop before proceeding with the second series of tests.

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 design of the auxiliary systems and components for the multiple-failure test apparatus described in NDA 84-2 has been completed and the equipment is now being fabricated. Figs. 2-5.7 and 2-5.8 are, respectively, isometric drawings of the sodium and water systems which supply the test apparatus housed in the containment vessel. The vessel, together with the bell jar, is shown in Fig. 2-5.9. (The bell jar will be set above the containment vessel before test operations are started.)

Detail design of the first test section to be operated in the apparatus has been started. This test section contains a barrier consisting of concentric tubes of stainless steel and aluminum. (See Fig. 2-5.10.) Aluminum was chosen as part of the first barrier because of the success of aluminum in the single-failure tests. This test apparatus is capable of squirting a stream of water at about 200F and a stream of sodium at temperatures up to 1150F against opposite sides of the barrier structure.

The design is such that additional tests can be made by removing one or both of the initial barrier tubes. Provision has been made for testing the effect of impingement of a hot sodium stream upon an aluminum can containing water, and to test the effect of impingement of a water stream upon a sodium-containing tube.

The control and recording systems for these tests have been designed, and fabrication and assembly have been started. The control system is designed to allow a remote operator to initiate, control, and stop simulated leaks in the sodium and water containers. An automatic shutoff circuit is provided which will stop both sodium and water squirts in case there is a high pressure in the containment vessel.

The structure to house the multiple failure apparatus has been erected, and electrical wiring has been completed. (See Figs. 2-5.11 and 2-5.12.) Raceways and cables for instrument and control lines have been placed between the control room and the experiment structure, a distance of several hundred feet.

Table 2-5.1 — Single-Failure Barrier Test Results

(all samples tested for 15 min unless otherwise noted)

Material	Thickness, in.	Specimen No.	Run No.	Remarks* (no failure unless noted)
SS(316)	0.037	10B	1	
SS(316)	0.037	10C	1	
Al 2S(1100)	0.010	11C	1	
Al 2S(1100)	0.010	11D	1	Slightly dented at impact point
Al 2S(1100)	0.010	11E	4	
Al 2S(1100)	0.010	11F	4	
Al 2S(1100)	0.060	13A	1	
Al 2S(1100)	0.060	13B	1	
Al 2S(1100)	0.060	13C	3	Ran for 16 hr
Al 2S(1100)	0.060	13D	4	
Al 2S(1100)	0.060	13G	5	Ran for 18 hr. Tiny pits visible with naked eye at point of impact.
Al 2S(1100)	0.125	14A	2	
Al 2S(1100)	0.125	14B	2	
Al 2S(1100)	0.125	14C	4	
Al 2S(1100)	0.125	14D	4	
Al 2S(1100)	0.250	15A	2	
Al 2S(1100)	0.250	15B	2	
Al 2S(1100)	0.250	15C	3	
Al 2S(1100)	0.250	15D	3	
Karbate (graphite)	0.250	16A	2	Swelled, stuck in holder, broke when extracted from holder.
Karbate (graphite)	0.250	16B	2	Swelled, stuck in holder, broke when extracted from holder.
Karbate (graphite)	0.250	16C	6	Badly cracked, but still in one piece. Did not stick in holder because holder was enlarged.
Graphite (AGOT, perpendicular to grain)	0.250	17A	5	
Graphite (AGOT, perpendicular to grain)	0.250	17B	5	
Graphite (AGOT, parallel to grain)	0.250	18A	5	
Graphite (AGOT, parallel to grain)	0.250	18B	6	Whole, crack on vertical axis
Zirconium	0.010	22A	4	
Zirconium	0.010	22B	5	

Table 2-5.1 — (Continued)

Material	Thickness, in.	Specimen No.	Run No.	Remarks* (no failure unless noted)
Graphite(Nat. Carbon Code 82, perpendicular to grain)	0.250	24A	6	Whole, many cracks
Graphite(Nat. Carbon Code 82, perpendicular to grain)	0.250	24B	6	Failed, fell apart in post-test handling
Graphite(Nat. Carbon Code 82, parallel to grain)	0.250	25A		Failed, fell apart in post-test handling
Graphite(Nat. Carbon Code 82, parallel to grain)	0.250	25B		Whole, many cracks
Al(6061-0)	0.125	29A	7	
Al(6061-0)	0.125	29B	7	
Al foil	0.005	—	7	Punctured at point of impact after several minutes' operation
Al foil	0.005	—	7	Punctured at point of impact after several minutes' operation
Cu	0.010	—	7	Flow pattern of jet etched on specimen at point of impact
Cu	0.010	—	7	Flow pattern of jet etched on specimen at point of impact
Al 2S, chrome plated	0.060	33A	8	Chrome surface very clean
Al 2S, chrome plated	0.060	33B	8	
Al, sintered	0.060	34A	8	
Al, sintered	0.060	34B	8	
Al(6061-0)	0.060	35A	8	
Al(6061-0)	0.060	35B	8	
Al foam	0.250	36A	9	Some signs of penetration
Al sprayed with 4-mil 18-8 SS		31A	9	Warped, blistered
Al sprayed with 4-mil 18-8 SS		31B	9	Warped, blistered
Al sprayed with 4-mil Fe		32A	9	Warped
Al sprayed with 4-mil Fe		32B	9	Warped
Graphite, porous	0.250	21A	9	Disintegrated.

* In all cases except copper there were varying degrees of general surface fouling.

NEG. No. 784

Fig. 2-5.1 — Single-failure test apparatus in operation

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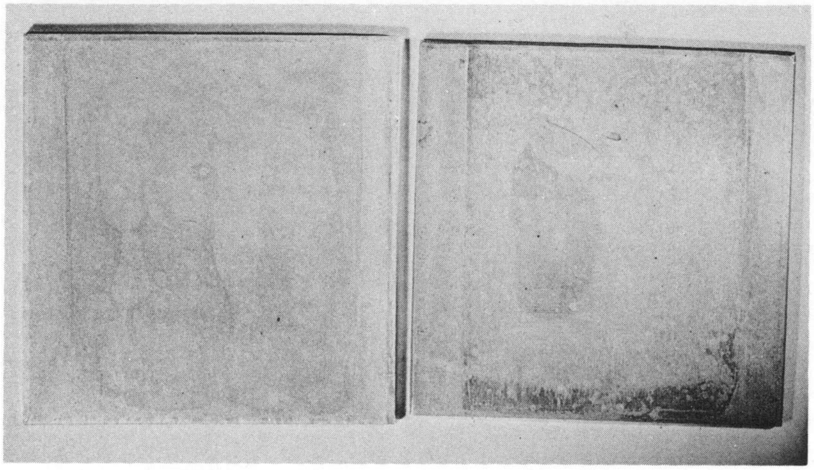

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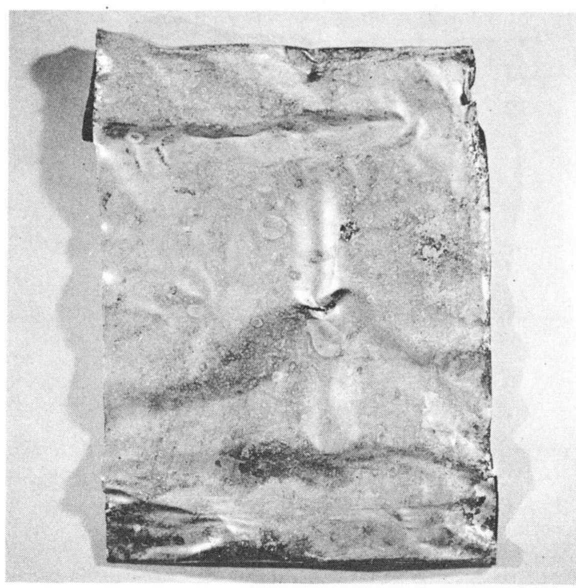
Specimen 13C Specimen 13G

Fig. 2-5.2 — Aluminum, Specimens 13C and 13G, exposed to sodium jet for 16 hr and 18 hr, respectively

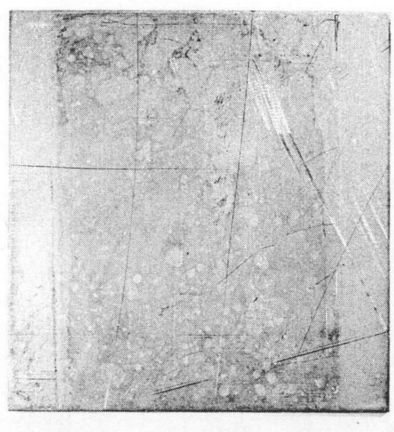


NEG. NO. 932C

Fig. 2-5.3 — Aluminum foil (0.005 in. thick) punctured after exposure to sodium jet for a few minutes

NEG. NO. 914A

Fig. 2-5.4 — Graphite, Specimen 24A, exposed to sodium jet for 15 min



NEG. NO. 900B

Fig. 2-5.5 — Type 316 stainless steel, Specimen 10C, exposed to sodium jet for 15 min

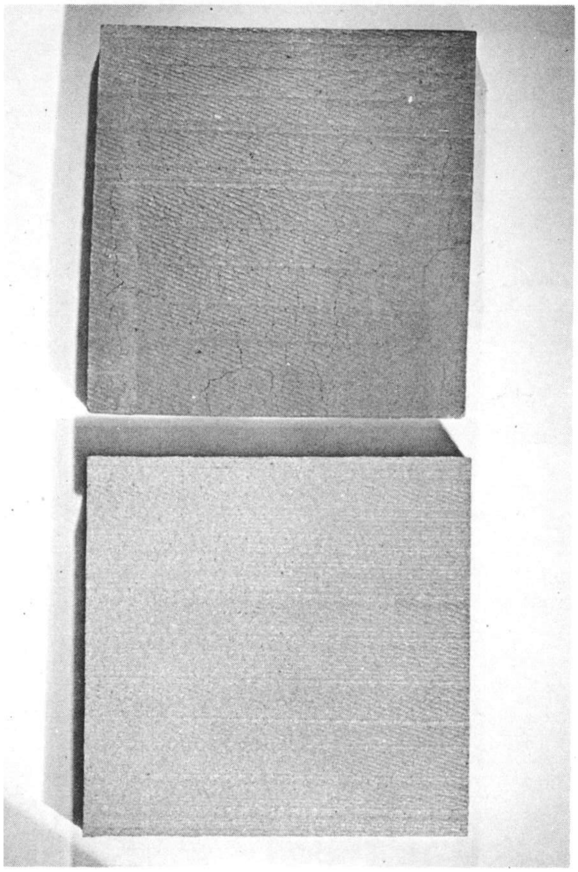


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Fig. 2-5.6 — AGOT graphite, Specimen 17A, exposed to sodium jet for 15 min; unexposed control specimen shown below for comparison

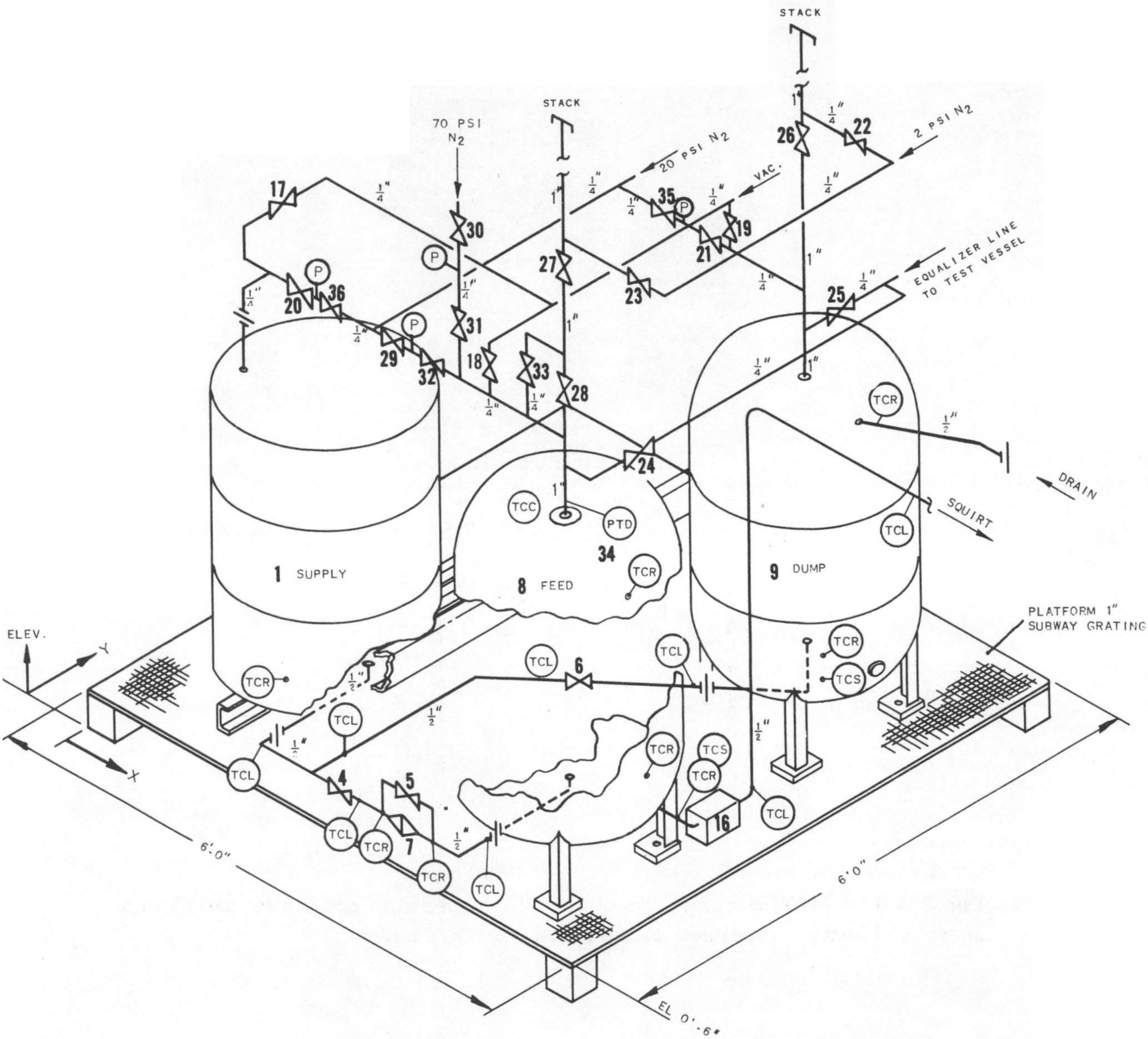
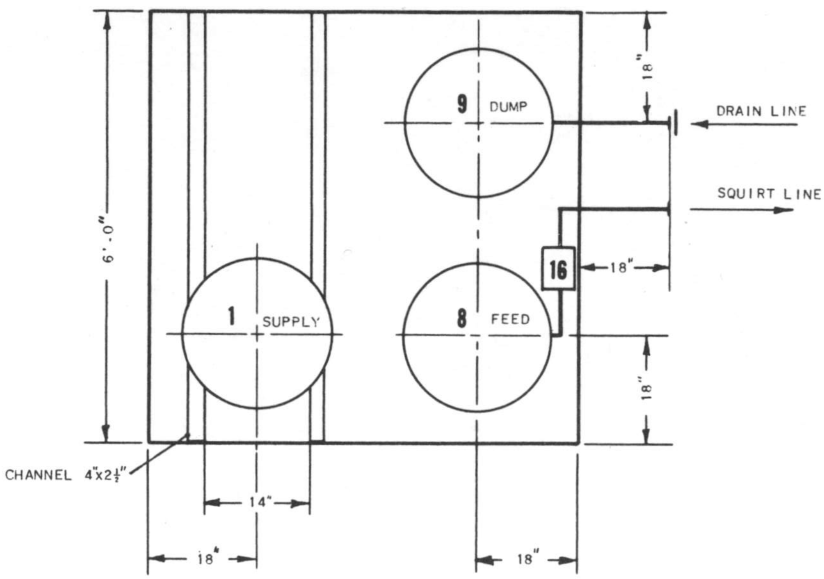


Fig. 2-5.7 — SDR – sodium system for multiple failure test apparatus – isometric



PLAN VIEW

CODE

Key	Item
P	pressure gage
TCC	control thermocouple
TCR	recording thermocouple
TCS	spare thermocouple
TCL	local thermocouple
PTD	pressure transducer
1	55-gal sodium supply tank
4	$\frac{1}{4}$ in. bellows seal valve
5	$\frac{1}{4}$ in. bellows seal valve
6	$\frac{1}{4}$ in. bellows seal valve
7	sodium filter
8	feed tank
9	dump tank
16	EM flowmeter
17	$\frac{1}{4}$ in. globe valve
18	$\frac{1}{4}$ in. globe valve
19	$\frac{1}{4}$ in. globe valve
20	$\frac{1}{4}$ in. needle valve
21	$\frac{1}{4}$ in. needle valve
22	$\frac{1}{4}$ in. needle valve
23	$\frac{1}{4}$ in. needle valve
24	$\frac{1}{4}$ in. gate valve
25	$\frac{1}{4}$ in. gate valve
26	1 in., $2\frac{1}{2}$ psi relief valve
27	1 in., 2 psi relief valve
28	1 in., 75 psi relief valve
29	$\frac{1}{4}$ in., 4 psi regulator, with gage
30	$\frac{1}{4}$ in., 55 psi regulator, with gage
31	$\frac{1}{4}$ in. solenoid valve
32	$\frac{1}{4}$ in. solenoid valve
33	$\frac{1}{4}$ in. solenoid valve
34	pressure transducer
35	$\frac{1}{4}$ in., 0-15 psi regulator with gage
36	$\frac{1}{4}$ in., 0-15 psi regulator with gage

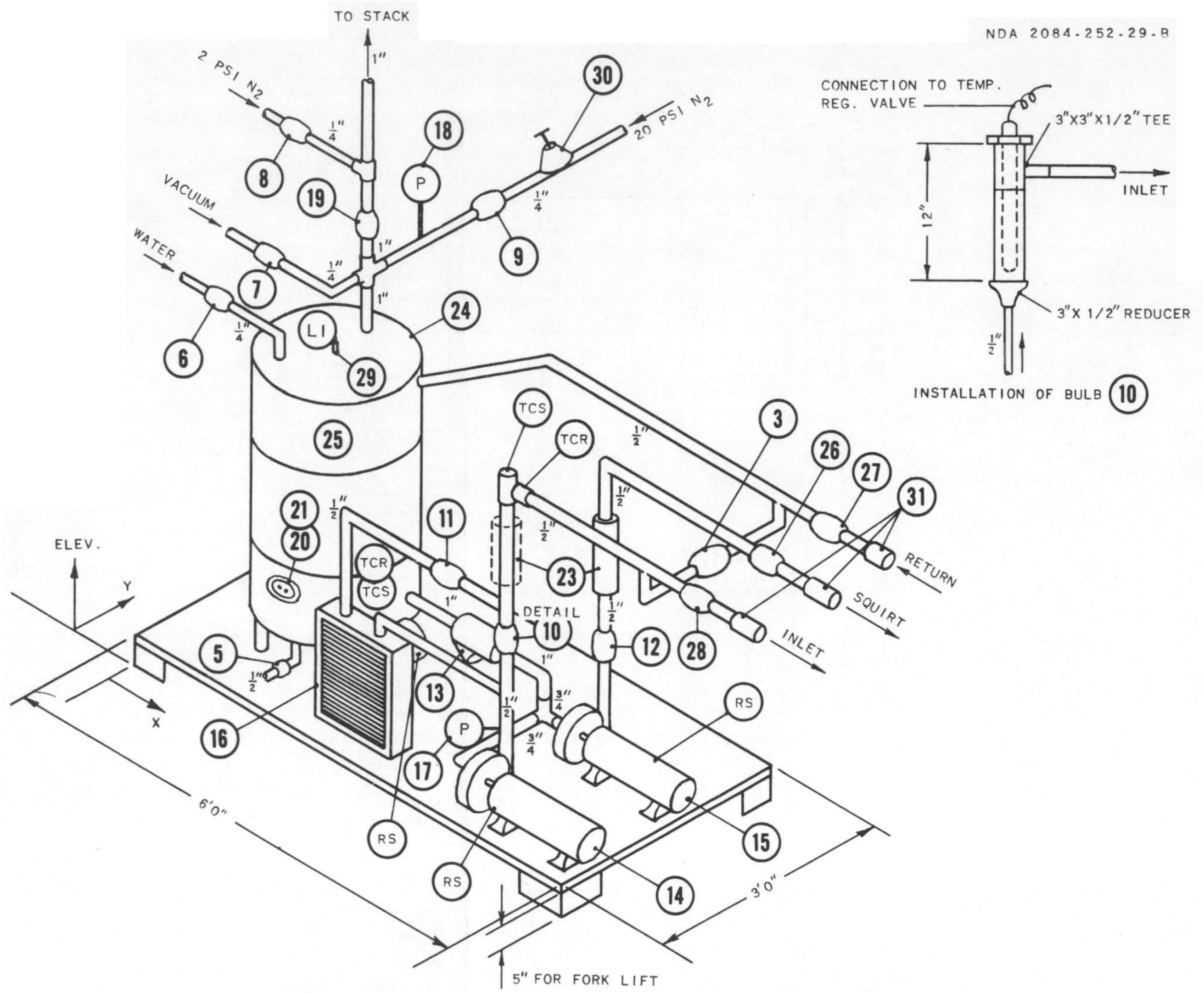
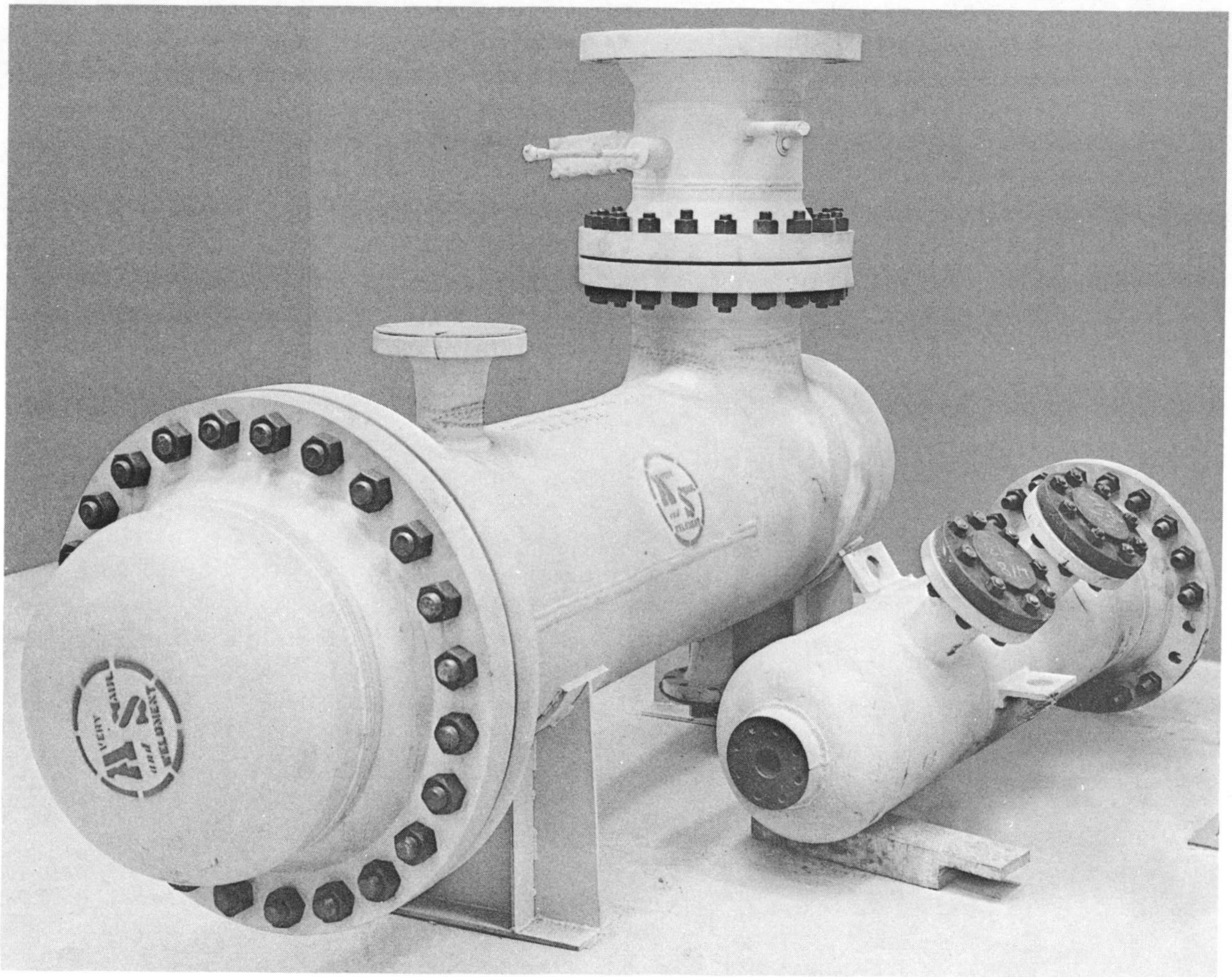


Fig. 2-5.8 — SDR – water system for multiple failure test apparatus – isometric

CODE

Key	Item
TCR	recording thermocouple
TCS	spare thermocouple
RS	remote start-stop switch
P	pressure gage
LI	level indicator
3	½ in. gate valve
5	½ in. gate valve
6	¼ in. globe valve
7	¼ in. globe valve
8	¼ in. needle valve
9	¼ in. needle valve
10	½ in. temperature regulator valve
11	½ in. globe valve
12	½ in. globe valve
13	1 in. pipe line strainer
14	1 gpm, 20 psi centrifugal pump
15	1.5 gpm, 35 psi centrifugal pump
16	3 kw watercooler and fan
17	0-50 psi pressure gage
18	0-10 psi pressure gage
19	1 in., 2 psi relief valve
20	5 kw immersion heater
21	heat control thermostat
23	150 psi, 3 gpm rotameter
24	insulation, 2 in. Fiberglas blanket
25	water supply tank
26	½ in. full flow, vacuum sealing valve
27	½ in. full flow, vacuum sealing valve
28	½ in. full flow, vacuum sealing valve
29	level indicator, 2 levels
30	¼ in., 2 psi pressure regulator
31	½ in. npt to ⅜ in. swagelok adaptors



NEG. NO. 815

Fig. 2-5.9 — Multiple-failure containment vessel, with bell jar shown on right. Vessel and bell jar are, respectively, 20 in. and 14 in. in diameter.

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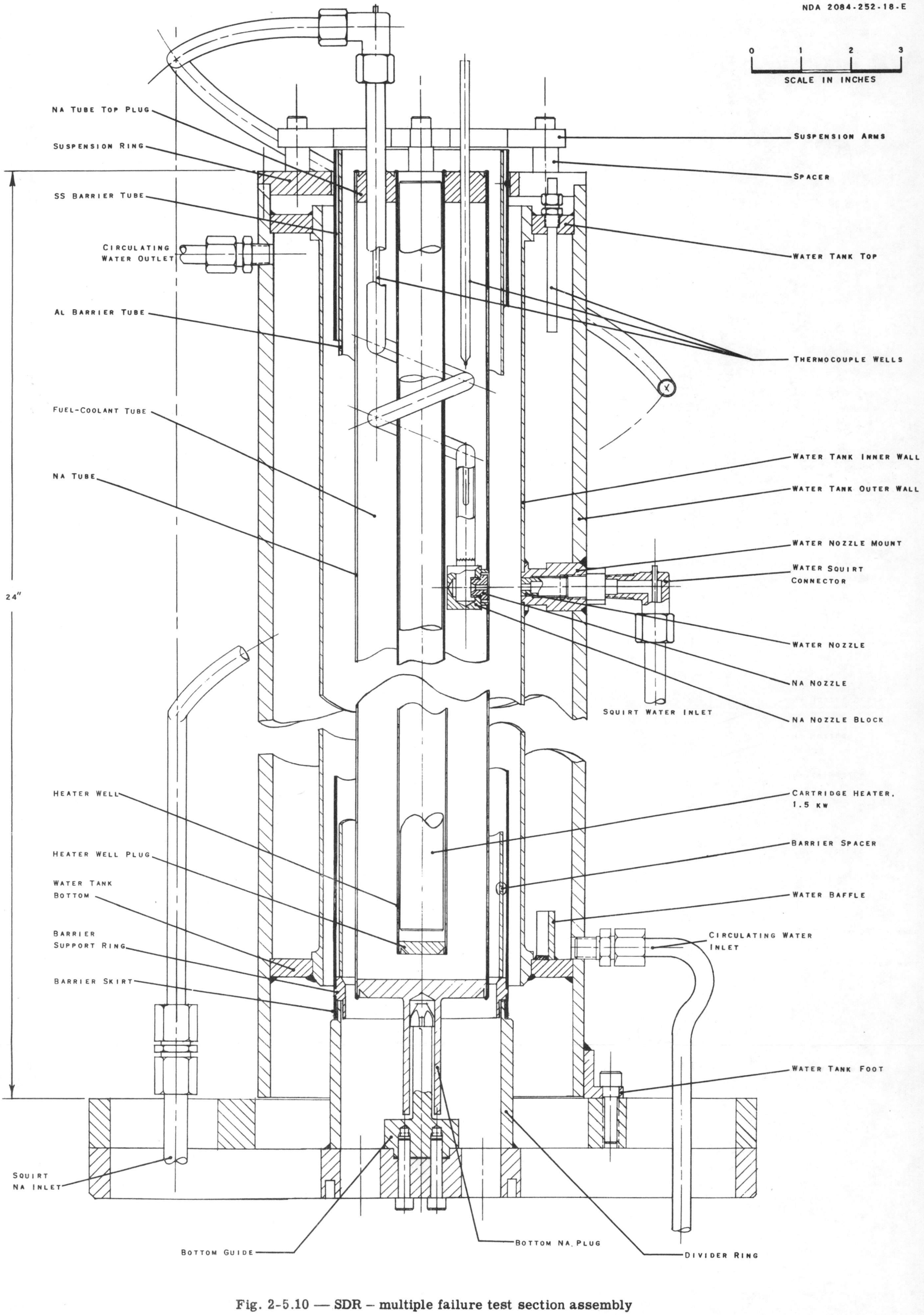

Fig. 2-5.10 — SDR – multiple failure test section assembly

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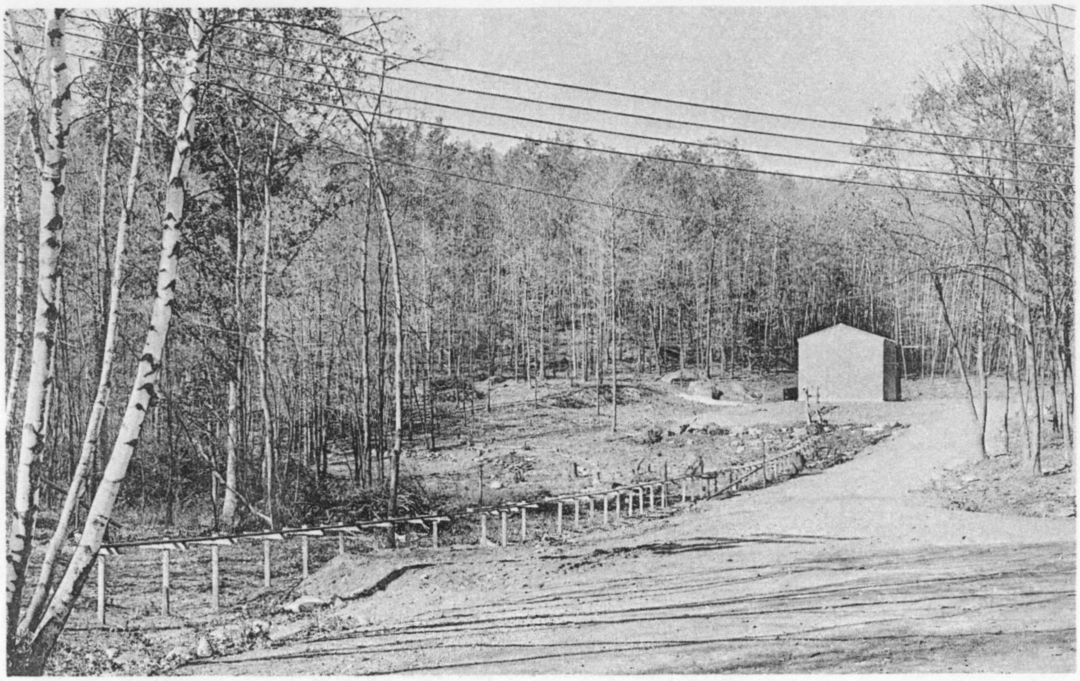

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Fig. 2-5.11 — Multiple-failure structure and raceways for instrument and control lines

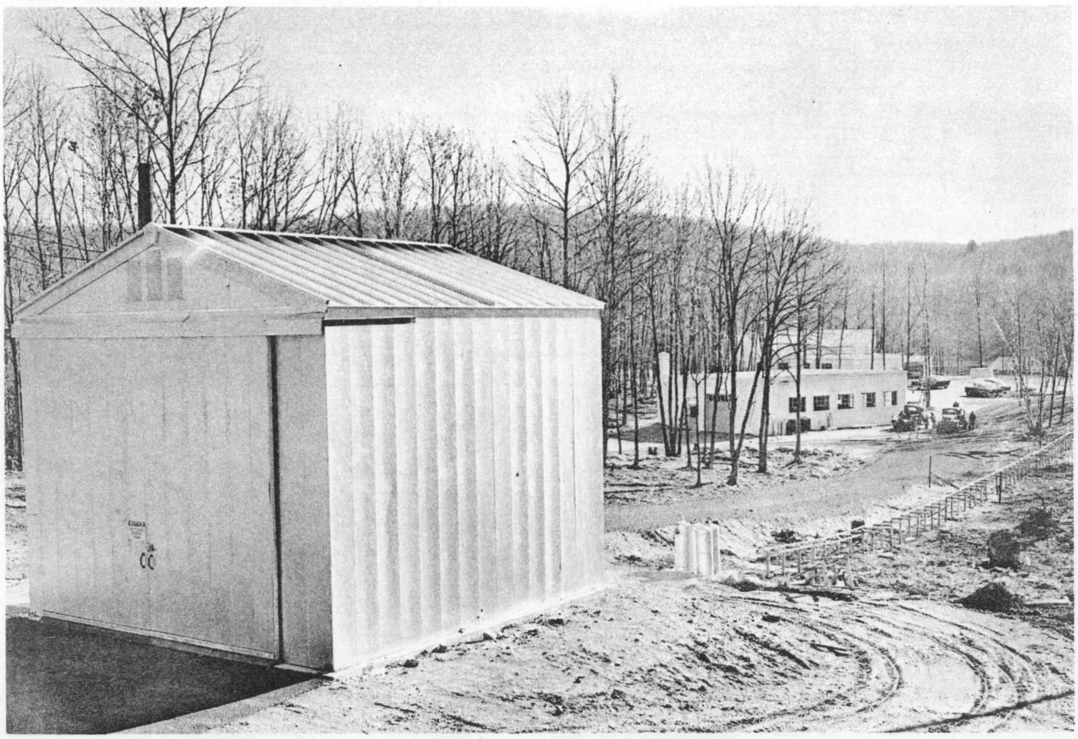

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Fig. 2-5.12 — Multiple-failure structure – view looking south towards Engineering Building

TASK 2-6 MOCKUP TESTS

Detailed design is progressing on the mockup test apparatus described in NDA 84-2. The apparatus consists of a mockup of three reactor fuel-coolant tubes and their barrier systems in a tank of water; provisions are made for supplying sodium, moderator water, and barrier gas.

GENERAL ARRANGEMENT

The test apparatus for the mockup tests will be located in a prefabricated structure at the NDA Pawling Laboratory. Instrument leads are connected to the Engineering Building, 500 ft away, where the controls for the test apparatus are located.

In the general layout of test apparatus, the test section is located to permit removal of barrier tubes and coolant tubes within the sheltered area. Also, the sodium and water systems are placed on opposite sides of the test section and are separated by partitions to reduce the possibility of leaks or splashes from one system getting into the other.

Gas, power, control, and instrument service panels are located at the end nearest to the Engineering Building, to minimize transmission distance.

TEST SECTION

A drawing of the assembled test section is shown in Fig. 2-6.2. The aluminum water tank, which represents a section of the calandria, is approximately 8 ft high and 2 ft in diameter. Three surge tubes in the calandria contain low pressure inert gas in close communication with the water in the calandria to reduce the initial surge pressures which might accompany a sodium-water reaction inside the calandria.

The barrier gas will be stagnant and will be enclosed in two steel headers flange-mounted to the ends of the calandria. The headers are about 3 ft long and are 2 ft in diameter. They have removable 2 ft diameter covers to permit access to the interior of the assembled test section. To insure that pressures do not become excessive, these headers will be provided with blowout disks, rated at 10 psi, with suitable vents and stacks.

Three stainless steel fuel-coolant tubes pass through the calandria. Both ends of the fuel-coolant tubes are attached to centrally located manifolds through helical pigtails of $^3/_8$ in. Schedule 40 stainless steel pipe. The tubes are supported at their bottom ends by a plate attached to the calandria bottom tube sheet. They are guided at the top by a light plate attached to the upper header. Each manifold is rigidly mounted to its header by three support arms. The line leading from each manifold out through the side of the header is almost identical to the pigtails. All pigtails, manifolds and other sodium containing lines are trace-heated, insulated, and supplied with thermocouples.

The barrier design for the first test includes an inner aluminum tube and an outer stainless steel tube in the space between the coolant tube and the calandria tube. The spaces between the above materials will be filled with inert gas. Above the calandria a sodium drip pan protects the top of the calandria from leaks in the upper pigtails, manifold, etc. Beneath the calandria is a water drip pan. In the bottom of the lower header is a sodium catch pan to collect any leaks from the coolant tubes, lower pigtails, etc. All of these pans are equipped with spark plug-type leak detectors.

The assembled test section is mounted in a support frame on trunnions to permit easier handling and maintenance. The frame supports a water surge tank on one side and ties into the framing of the sodium system on the other side.

SODIUM SYSTEM

A schematic of the sodium system is shown in Fig. 2-6.1. An EM sodium pump rated at 30 psi at 5 gpm circulates the sodium through the loop, which is composed of a flowmeter, a heater, and a test section. A 25-gal sump tank will be used to fill the sodium system through a filter, after which the filter section will be frozen off. A filter by-pass with oversized heaters can be melted in about 2 min to drain the sodium system. A regenerative cold trap is mounted across the pump as a loop by-pass, flow being controlled by orifices in the cold trap line.

A 3-gal surge tank at the top of the loop allows for fluid expansion and for pressurizing the system. High and low level probes are installed in the surge tank.

All components in the system are rigidly mounted either on a skid or on a structural column. Suitable pipe bends and pigtails between components serve to accommodate thermal expansion at low stress levels.

OTHER SYSTEMS

Water System

A 350-gal supply tank will be used to fill the calandria with water. (See Fig. 2-6.3) Electric heaters in the water supply tank and an air cooler will control water temperature as it is circulated through the system at 10 gpm by a centrifugal pump. Provision is made, by means of a 4 in. electrically tripped valve, to dump the water from the calandria into the supply tank in less than 45 sec. Water is pumped in at the bottom of the calandria, rises into the water surge tank, and returns through a float valve to the supply tank.

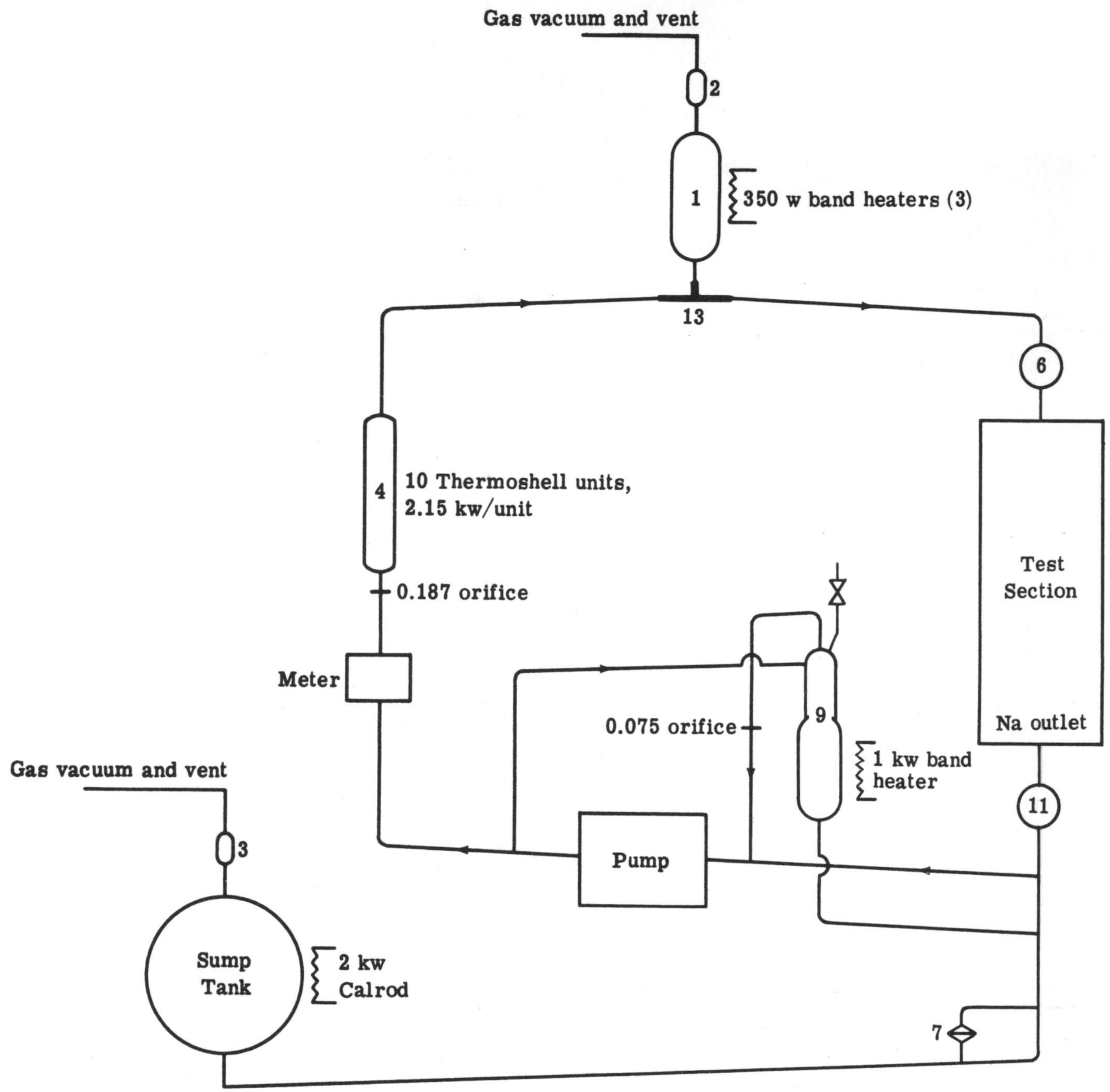
Gas, Vent, Steam, and Vacuum Systems

Nitrogen at about 2 psig is used as the barrier gas. (See Fig. 2-6.3.) Since the gas will not be circulated through the barrier space, the only flow required will be that necessitated by leakage. Nitrogen will also be used as the cover gas in the water surge tank, and for dumping and purging the calandria.

Argon will be used as an inert cover gas, and for pressurizing and transferring the sodium system.

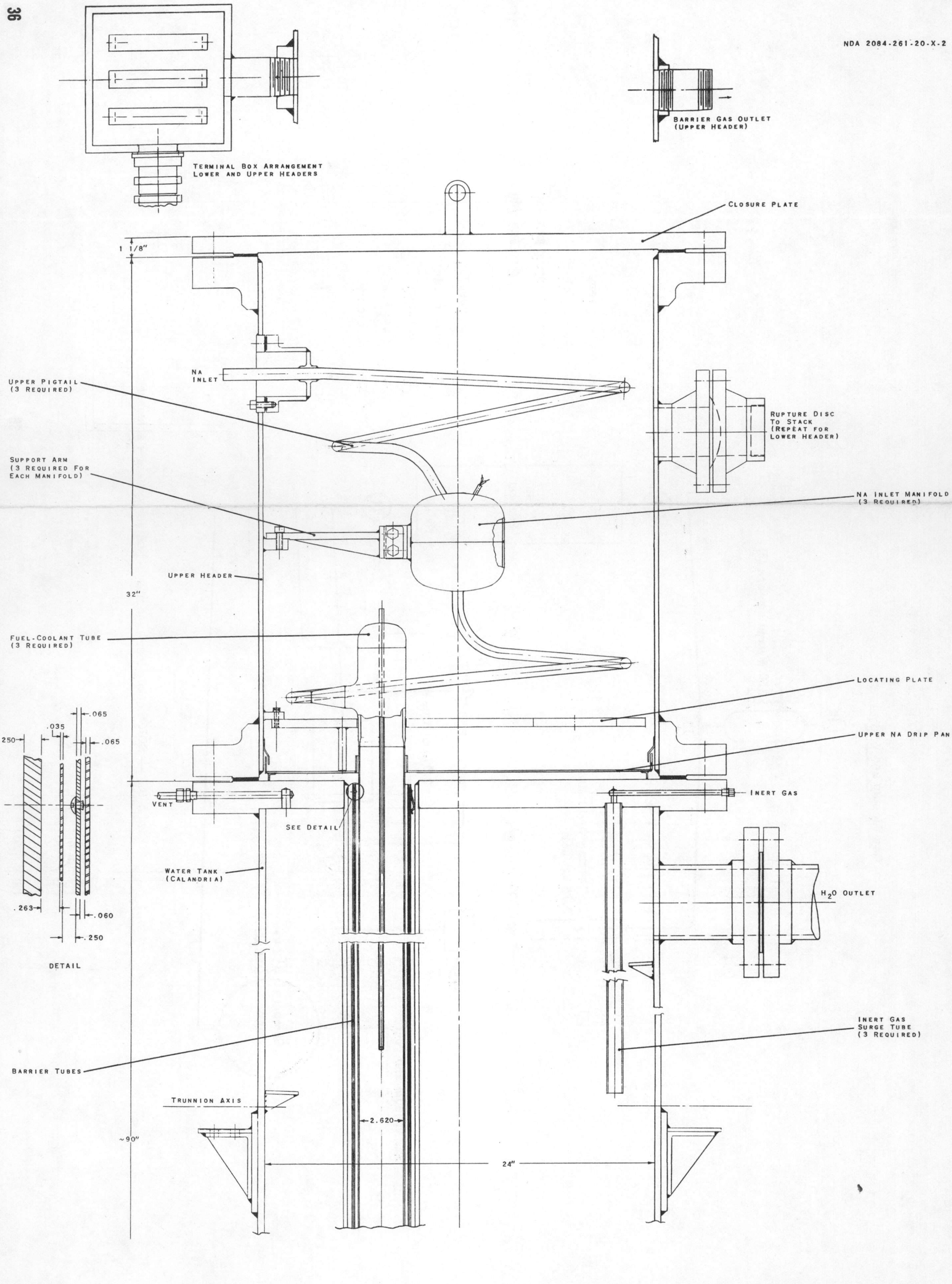
Steam for preheating the calandria prior to filling the test section with sodium will be delivered at 250F and about 15 psig from a portable generator next to the test section.

Vacuum connections are provided on the sodium system, calandria, and barrier gas system for the removal of air, steam, etc. A portable 5-cfm vacuum pump will be placed near the apparatus for this purpose and connected as required.



Key	Item
1	surge tank
2	sodium trap on gas line
3	sodium trap on gas line
4	sodium heater
6	upper header
7	sodium filter
9	cold trap
11	lower header
13	bubble trap

Fig. 2-6.1 — SDR mockup test assembly – sodium system – schematic



Terminal Box Arrangement
Lower and Upper Headers
Barrier Gas Outlet
(Upper Header)
Closure Plate
1 1/8"
Na
Inlet
Upper Pigtail
(3 Required)
Rupture Disc
To Stack
(Repeat for
Lower Header)
Support Arm
(3 Required For
Each Manifold)
Na Inlet Manifold
(3 Required)
Upper Header
32"
Fuel-Coolant Tube
(3 Required)
Locating Plate
.065
.035
.250
.065
Upper Na Drip Pan
Inert Gas
Vent
See Detail
Water Tank
(Calandria)
H_2O Outlet
.263
.060
.250
Detail
Inert Gas
Surge Tube
(3 Required)
Barrier Tubes
Trunnion Axis
2.620
~90"
24"

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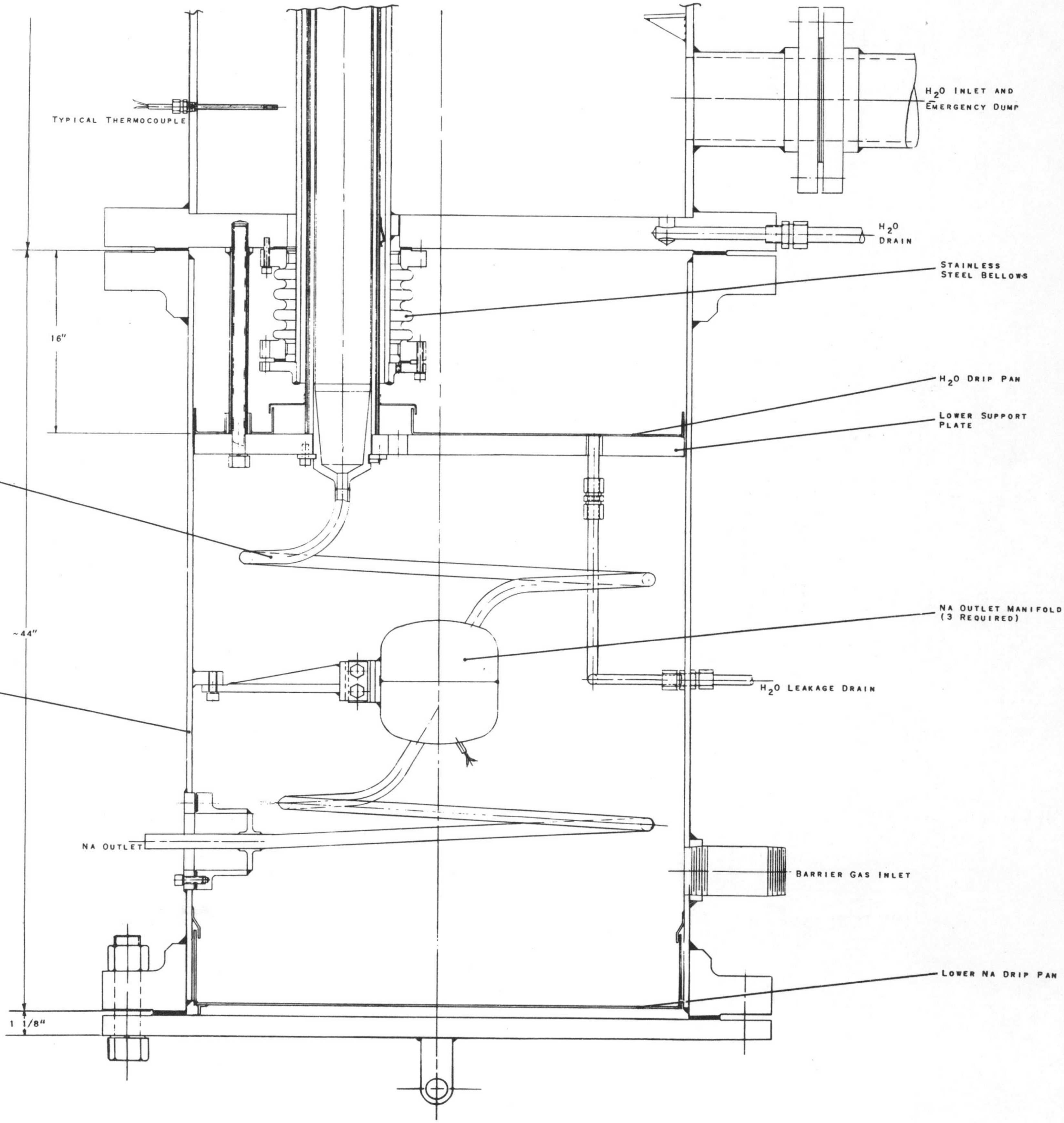

Fig. 2-6.2 — SDR mockup test section assembly – section

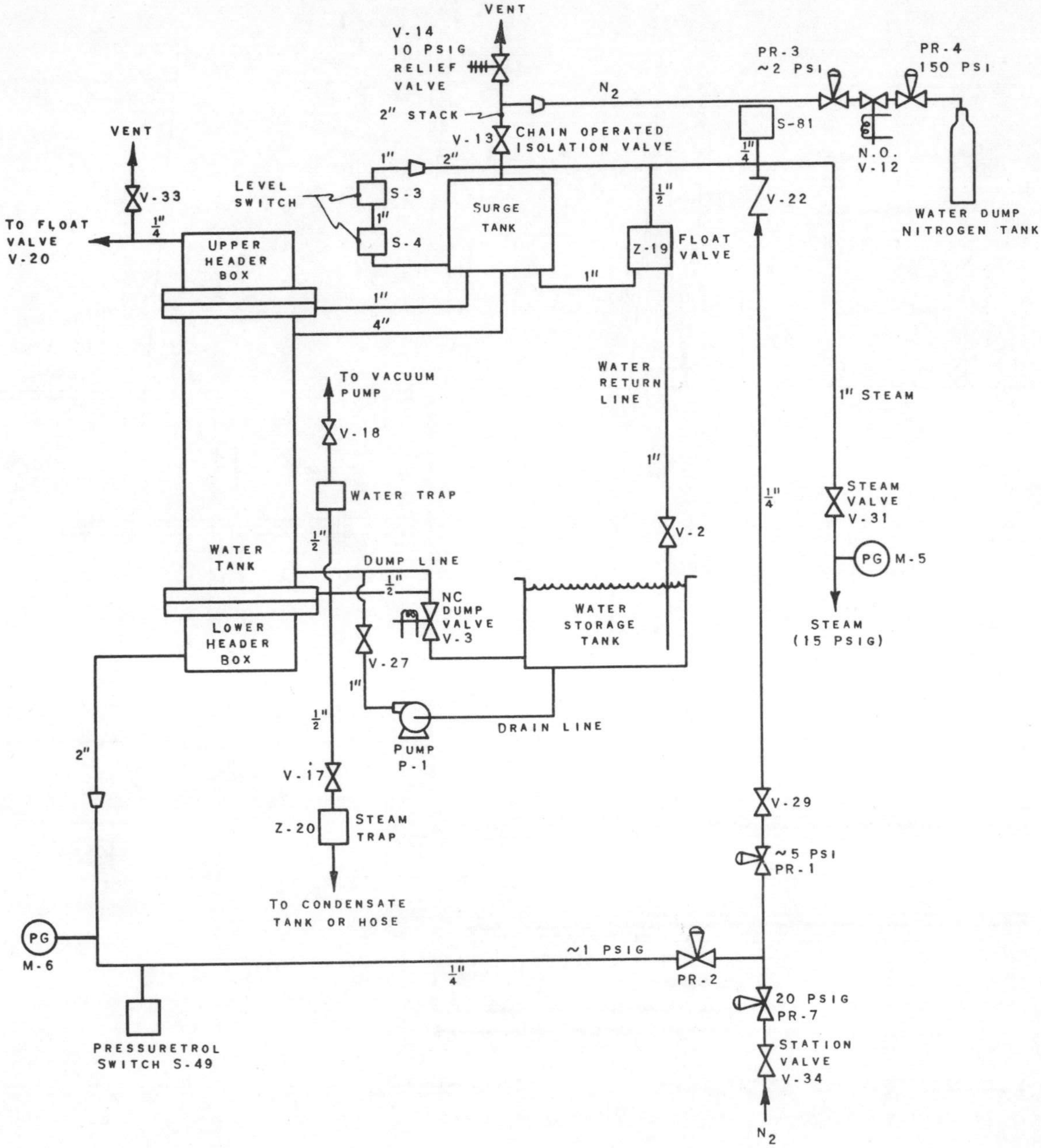

Fig. 2-6.3 — SDR mockup test assembly – water, nitrogen, steam supply systems – schematic

PRELIMINARY DESIGN

TASK 3-1 REACTOR PRELIMINARY DESIGN

Additional design work has been done on the straight-through configuration. Results of the mechanical design work are discussed under Tasks 2-1, 2-2, and 2-3, and calculations on nuclear and thermodynamic performance of the reactor are presented below for two types of fuel element.

An alternative design concept based on a slab configuration has also been investigated.

PERFORMANCE CHARACTERISTICS FOR THE STRAIGHT-THROUGH REACTOR

During the quarter, work was centered on reactor designs using U–10 weight % Mo alloy fuel elements and UO_2 fuel elements. The U–10 weight % Mo design presented in the last quarterly report has been revised and a brief parametric study was made for reactors using UO_2 fuel elements.

Data for typical, but not necessarily optimum, designs of reactors using U–10 weight % Mo alloy and UO_2 fuel elements are given in Table 3-1.1. Both reactors have sufficient initial reactivity for an average fuel burnup of 5,000 MW-d/metric ton, and both are heat-transfer limited (see "Power Limitation in Design Selection," below). Each design has a core height 0.8 times the core diameter and a 1-ft thick D_2O reflector at the ends and sides (instead of the graphite reflector assumed in NDA 84-2).

It should be noted that these designs were prepared on different bases and were not intended to provide a direct indication of the relative merits of U-Mo and UO_2 fuels. It should be noted, further, that information obtained in the course of these studies has not yet been incorporated in work on Task 2-1.

POWER LIMITATION IN DESIGN SELECTION

As was mentioned above, the reactor designs compared in Table 3-1.1 are thermal power (heat-transfer) limited; that is, they are both operating at the upper limit of fuel element power. This limit is different for the U–10 weight % Mo and UO_2 fuel elements, since it is primarily dependent upon the maximum allowable fuel temperature, the thermal conductivity of the fuel, and the fuel rod dimensions.

For the U–10 weight % Mo fuel, 1300 F was assumed as the maximum fuel element temperature, and the thermal conductivity was taken as 12 BTU/hr-ft-F. For the UO_2 fuel, a maximum temperature of 4000 F was set, and a value of 1.0 BTU/hr-ft-F was assumed for UO_2 thermal conductivity. These assumptions are considered to be conservative.

Table 3-1.1 — Reactor Design Data

Specification	Design Using U–10 Weight % Mo Fuel	Design Using UO_2 Fuel
Reactor power, kw (heat)	40,000	40,000
Fuel loading, metric tons uranium	7.7	11.7
Fuel enrichment, %	2.0	2.0
D_2O in calandria, metric tons	17.9	27.1
Height of calandria, ft	8.4	9.8
Diameter of calandria, ft	10.0	11.8
Diameter of core region, ft	8.0	9.8
Length of fuel elements, ft	6.4	7.8
Number of fuel tubes	128	155
Diameter of fuel rod, in.	0.75	1.00
Number of fuel rods per subassembly	7	7
Fuel material	U–10 weight % Mo	UO_2
Fuel clad	stainless steel	stainless steel
Thickness of clad, in.	0.010	0.030
Sodium-containing tube	stainless steel	stainless steel
Thickness of fuel-coolant tube, in.	0.060	0.030
Metallic barrier	stainless steel	stainless steel
Thickness of metallic barrier, in.	0.020	0.020
Lattice type	square	square
Lattice pitch, in.	7.57	8.35
Heat Transfer and Fluid Flow		
Inlet temperature, °F	750	750
Mixed outlet temperature, °F	950	950
Total sodium flow, lb/hr	2.24×10^6	2.24×10^6
Avg sodium velocity in fuel tubes, ft/sec	5.2	2.4
Max sodium velocity in fuel tubes, ft/sec	8.4	3.9
Avg sodium velocity in pigtails, ft/sec	17.0	14.0
Avg power per fuel tube, MW	0.31	0.26
Max power per fuel tube, MW	0.50	0.41
Max fuel temperature, °F	1300	4000
Sodium flow area per fuel tube, ft^2	0.018	0.033
Pigtail flow area, ft^2	0.0054	0.0054
Thermal conductivity of fuel, BTU/hr-ft-°F	12	1.0
Physics		
D_2O-to-fuel volume ratio	14.4	9.11
Na-to-fuel volume ratio	0.832	0.842
Steel-to-fuel volume ratio	0.300	0.237
Equivalent lattice diameter, in.	8.54	9.42
k_∞, clean	1.202	1.155
k_{eff}, clean	1.083	1.070
k_{eff}, equil. Xe and Sm	1.053	1.043
η, clean	1.734	1.734
ϵ, clean	1.025	1.020
p, clean	0.869	0.842
f, clean	0.778	0.776
τ, clean, cm^2	165	177
L^2, clean, cm^2	90	101
M^2, clean, cm^2	255	278
Geometric buckling, cm^{-2}	4.31×10^{-4}	2.98×10^{-4}
Material buckling, cm^{-2}	7.96×10^{-4}	5.58×10^{-4}
Initial conversion ratio	0.479	0.525
Assumed average fuel burnup, MW-d/metric ton	5000	5000

SLAB DESIGN STUDY

A more detailed examination has been made of the feasibility, and possible advantages and disadvantages, of a reactor configuration which consists of alternating vertical slabs of moderator and fuel-coolant assemblies. The study was motivated by considerations which indicated potential maintenance and safety advantages of the slab configuration as compared with the calandria (or lattice) configuration.

Fig. 3-1.1 and 3-1.2 are elevation and plan views of a slab reactor. There are eight identical fuel-coolant assemblies, alternating with nine D_2O assemblies.

The fuel-coolant slab is an integral assembly containing 16 fuel-coolant tubes, each 3 in. OD, on 6 in. centers. Each fuel-coolant tube contains seven 0.75 in.-diameter uranium metal (SRE-type) fuel rods, 8 ft long. Within the fuel slab, each fuel-coolant tube is surrounded by graphite. Figs. 3-1.3 and 3-1.4 show details of the fuel slab and core cross section.

The fuel-coolant tubes of each fuel slab are manifolded together at top and bottom. The coolant lines and all other process lines penetrate one vertical end-face of the fuel slab. Each slab is independently connected to the external system. Both fuel and moderator slabs are arranged in this manner. With this arrangement, it is possible to remove any slab by making simple pipe cuts (or opening disconnects) in areas not highly radioactive and then sliding the slab out of the core into a storage area. Each fuel slab is mounted on rollers riding on tracks located in the floor, and each moderator slab is suspended from an overhead rail system.

The moderator tanks are constructed of $\frac{1}{4}$ in. aluminum plate and contain vertical thimbles (two per slab) which house the reactor control rods. Each tank is 12 in. wide.

Detailed nuclear calculations are not yet complete. A first calculation was made assuming a fuel enrichment of 1.65% and an initial excess reactivity of 10% (to compensate for fuel burnup and for overriding poisons), and it was found that six fuel slabs are needed. For eight fuel slabs, as required by heat transfer considerations, it would be possible to achieve the specified reactivity with a somewhat lower fuel enrichment (or the same enrichment could be used with somewhat thinner D_2O slabs).

This slab design contains about 28 metric tons of D_2O in the core and reflector regions, which is substantially more than the 18 metric tons indicated in Table 3-1.1 for the calandria design using U–10 weight % Mo fuel rods.

A 6-ft thick concrete neutron shield surrounds the reactor core. The top shield is not removable, since access to the fuel elements can be obtained through small removable plugs in the side shield adjacent to the slab storage area; here a sliding shield allows access to the slab requiring replacement.

The design provides for relatively simple and rapid removal of any fuel-coolant assembly or moderator assembly in the event of a mechanical failure. At the same time, it permits normal fuel element replacement with relative ease. In addition, the design provides a substantial barrier consisting of a minimum of $1\frac{1}{2}$ in. of graphite and a 0.280-in. thick aluminum plate between moderator and coolant. The barriers permit the isolation of a failure to a relatively small space.

The preliminary study has led to the following conclusions:

1. The slab reactor process design is not significantly different from that of the calandria design.
2. From a basic nuclear standpoint, the slab design is similar to the calandria design.
3. The slab design may be easier to maintain than the calandria design, particularly when repairs to the compartmentalized D_2O systems are contemplated.

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4. The elimination of several hundred field welds (in pigtail connections) in favor of shop welds should reduce the probability of sodium leaks.
5. The slab design permits repairs to be made in locations which are not highly radioactive.

CHOICE OF PRINCIPAL STRUCTURAL ALLOY

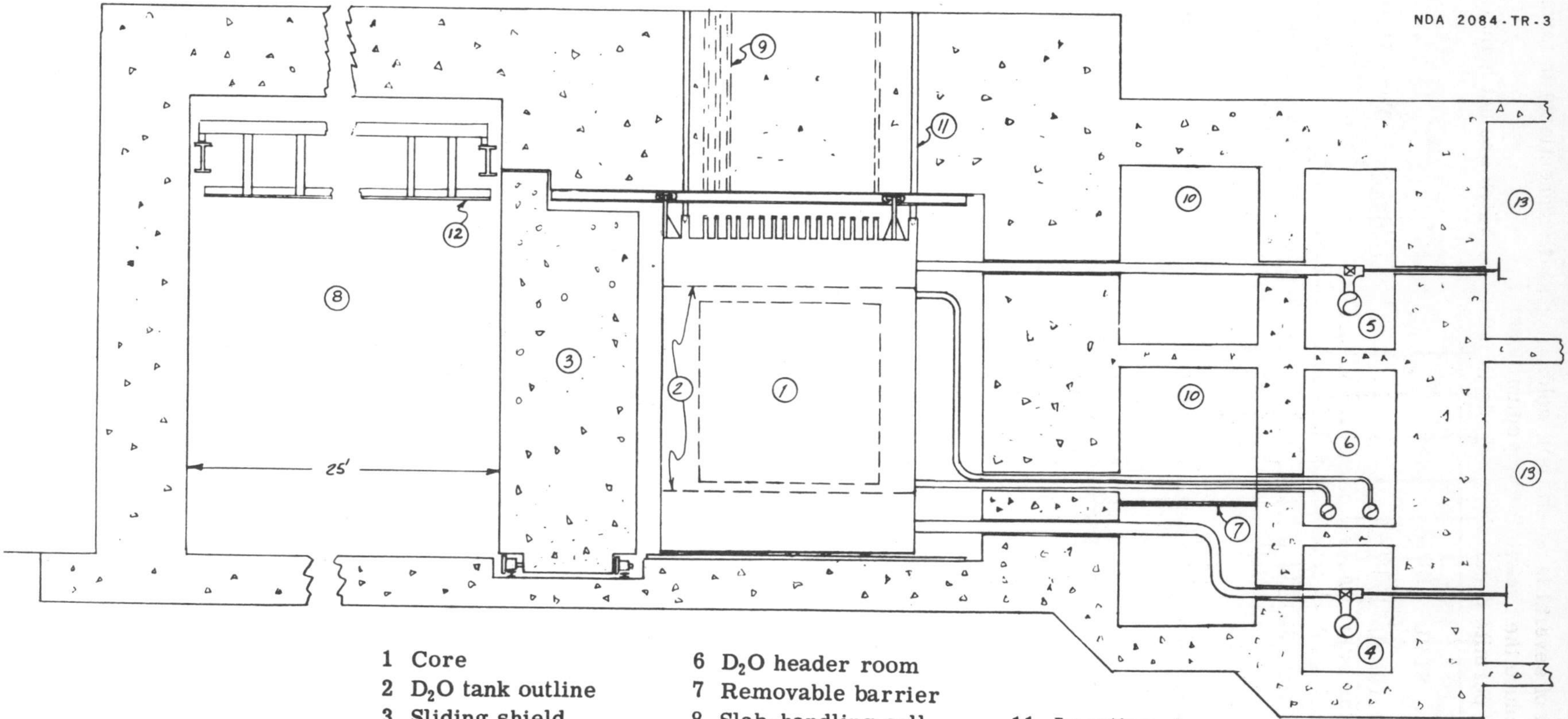
The selected structural alloy for the fuel-coolant tubes and pigtails is Type 316 stainless steel, with Type 304 as an alternate. The choice of Type 316 was arrived at by elimination. It has its own disadvantages, such as higher neutron absorption cross section than other materials, high coefficient of expansion, poor thermal conductivity, and possible sigma formation at operating temperatures. However, other materials have disadvantages and more unknowns as indicated below:

Low alloy Cr-Mo steels and 400-series stainless steels were not chosen because of

1. insufficient data on corrosion resistance,
2. insufficient data on effects of radiation, (it being probable that the radiation-induced increase in transition temperature of carbon steels applies similarly to Cr-Mo steels),
3. the necessity of pre and post heat of welds.

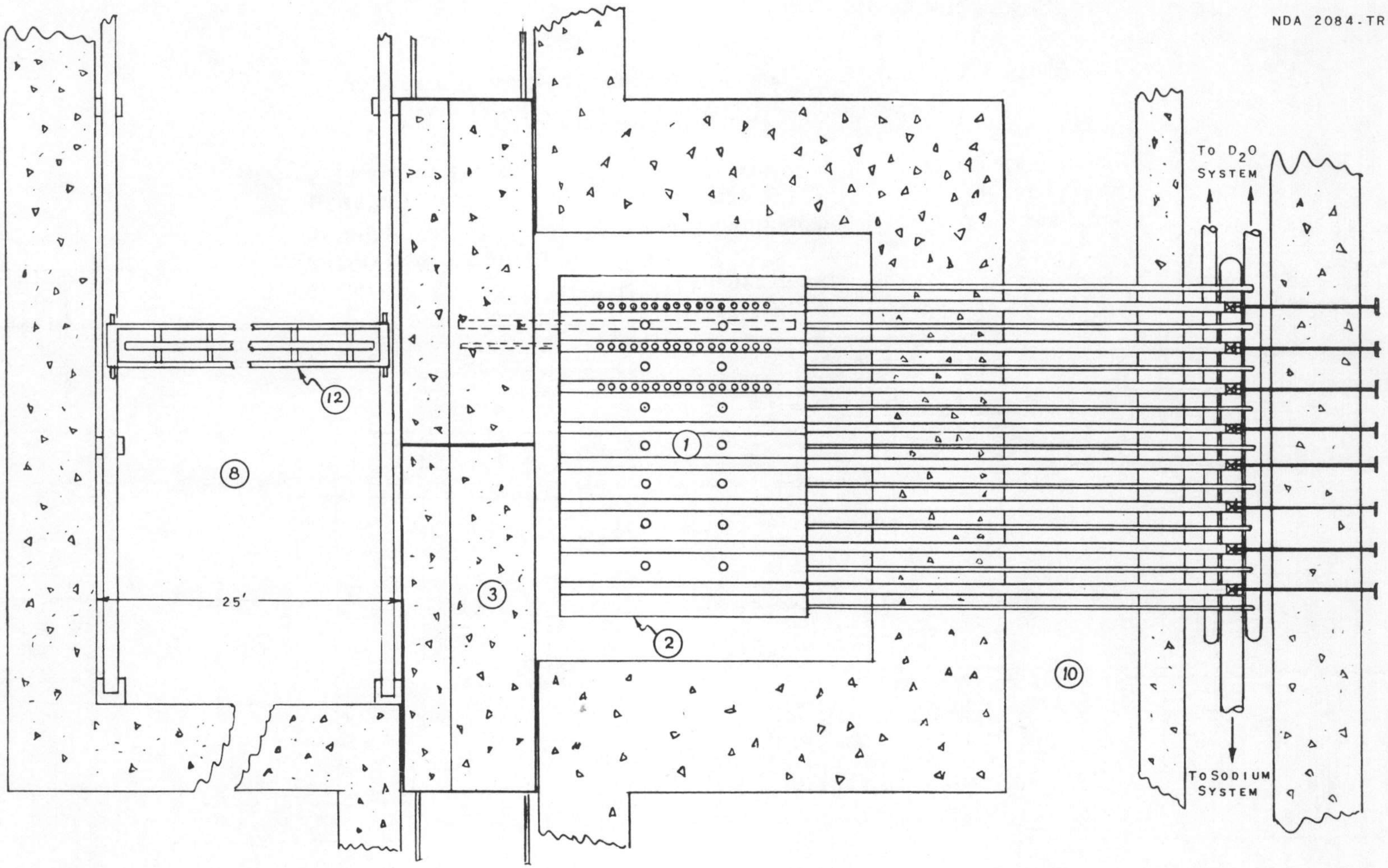
Type 347 stainless steel was not chosen because of weld cracking that has been experienced in several applications. Although an expensive test to determine crack-sensitive heats of Type 347 has been developed, the causes and cures of this cracking are not known.

Type 304 stainless steel would be a suitable substitute for Type 316, if the operating stresses will permit its use. As mentioned, it has been chosen as the alternate to Type 316.



1 Core
2 D_2O tank outline
3 Sliding shield
4 Na inlet header
5 Na outlet header
6 D_2O header room
7 Removable barrier
8 Slab-handling cell
9 Removable plug
10 Pipe gallery
11 Locating pins
12 Retractable rail
13 Control room

Fig. 3-1.1 — SDR – slab design – general arrangement – elevation



1 Core
2 D_2O tank outline
3 Sliding shield
8 Slab-handling cell
10 Pipe gallery
12 Retractable rail
13 Control room

Fig. 3-1.2 — SDR – slab design – general arrangement – plan. Note — barrier between sodium and D_2O pipe gallery and header trench not shown; see Fig. 3-1.1.

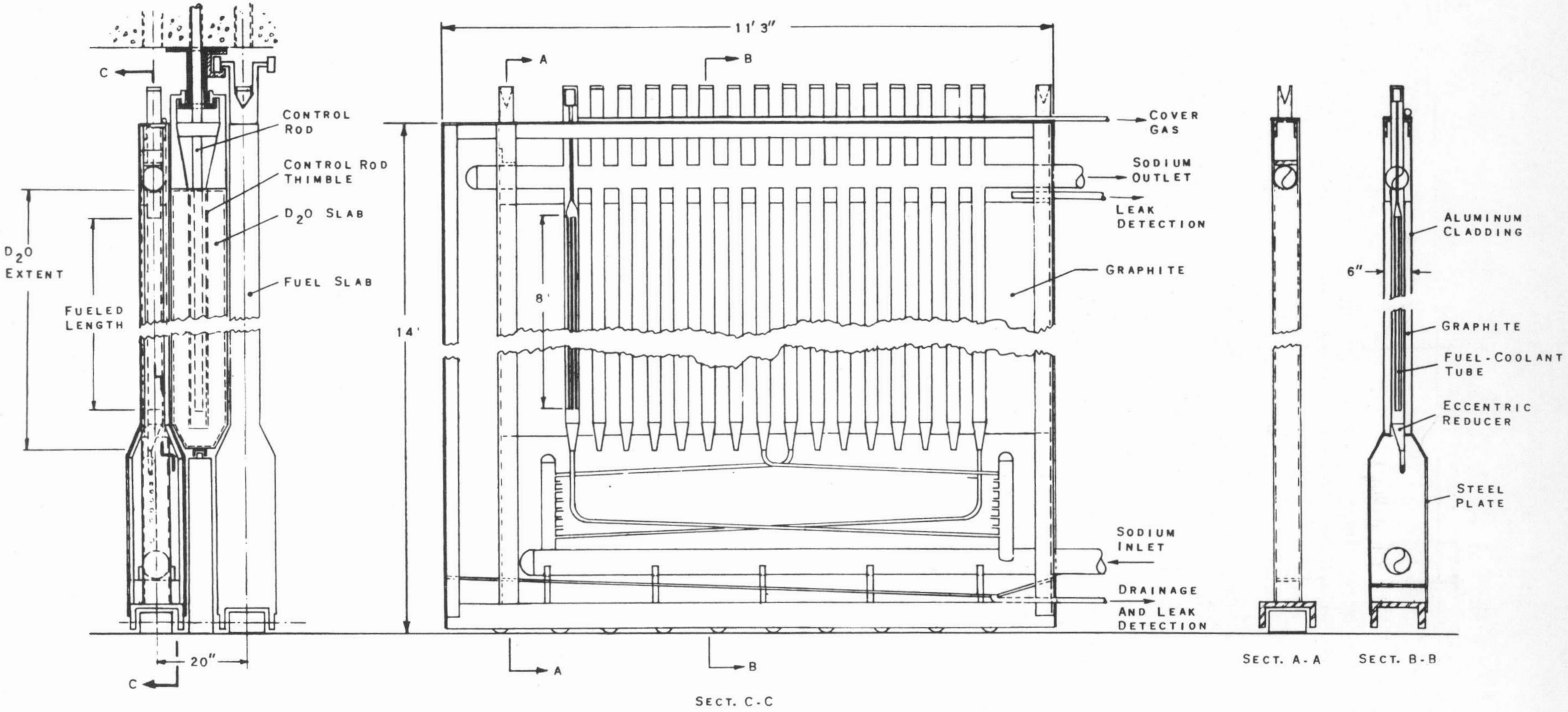


Fig. 3-1.3 — SDR – slab design – typical fuel slab

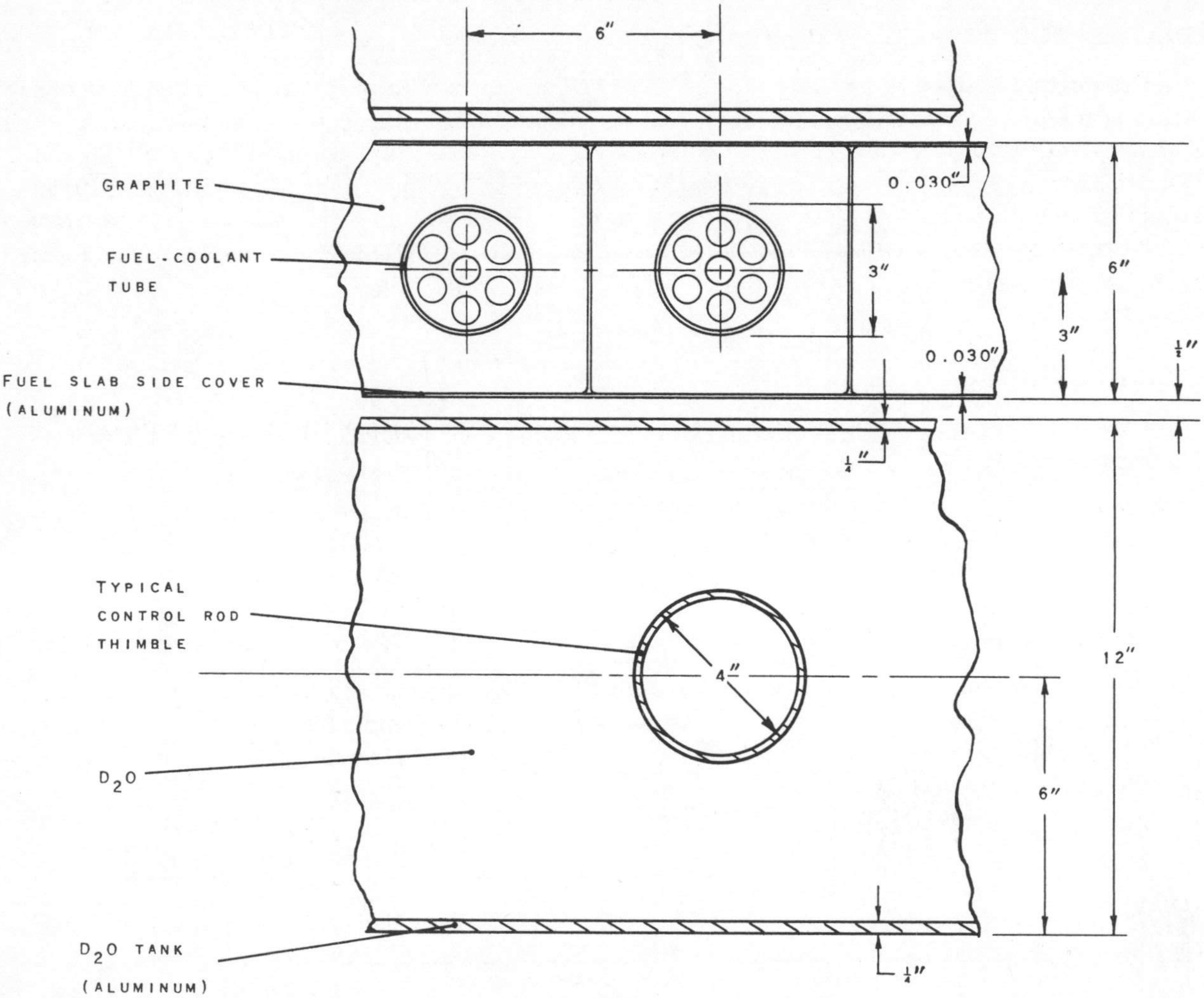


Fig. 3-1.4 — SDR – slab design – typical core cross section

TASK 3-2 SHIELDING

Shielding studies have continued to concentrate on the problem of accessibility to the header rooms after shutdown. The sources of header room radiation which have been considered during the quarter are those due to mass transfer and fission product escape.

RADIATION DUE TO MASS TRANSFER

An important source of radiation in the SDR header room after reactor shutdown is that due to mass transfer of radioactive stainless steel constituents from the core. The dose rates within the SDR header room due to this source were calculated, using mass transfer rates measured by Knolls Atomic Power Laboratory in a 3000-hr in-pile loop experiment run at the MTR.* The results of these calculations indicate that after one month of full power operation the dose rate would be between 4 and 7 mr/hr. Assuming the mass transfer rates to be linear with time, after 2 years' operation the dose rate would be 0.5 to 0.7 r/hr, while after 20 years it would rise to 4 to 6 r/hr.

RADIATION FROM A FUEL ELEMENT FAILURE

A calculation of the radiation in the header room due to a single fuel element failure was made assuming that:

(1) the fuel element that fails is in the region of maximum flux and the fission product concentration has reached equilibrium,
(2) 1% of the fission products within the fuel element are released into the sodium,
(3) the fission products are uniformly deposited throughout the system,
(4) a decontamination factor is obtained equivalent to that obtained by draining the sodium.

Results indicated that the dose rates 7 days after shutdown due to this source were about 0.2 r/hr, the same order of magnitude as the dose rates due to residual sodium. (NDA 84-2, page 48.) Further work on this problem will be aimed at determining the validity of the assumptions made in this calculation.

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* F. G. Haag, Activity Transport in Sodium-Cooled Systems, Nucleonics, 15(2): 58 (Feb. 1957).

APPENDIX

INSTALLATIONS VISITED

A list of installations visited during this quarter, together with topics discussed, is presented below.

Installation	Topics Discussed
Atomic Power Development Associates, Inc.	Sodium system design; NaK-water reactions
Atomics International (North American Aviation, Inc.)	Sodium technology; fuel elements
Atomic Energy of Canada, Ltd., Chalk River	Calandria design; D_2O system design; reactor control; fuel elements
E.I. du Pont de Nemours & Co., Wilmington	D_2O system design
Hanford Atomic Products Operation	Through-tube reactor designs
Oak Ridge National Laboratory	Sodium technology
E.I. du Pont de Nemours & Co., Savannah River	D_2O technology