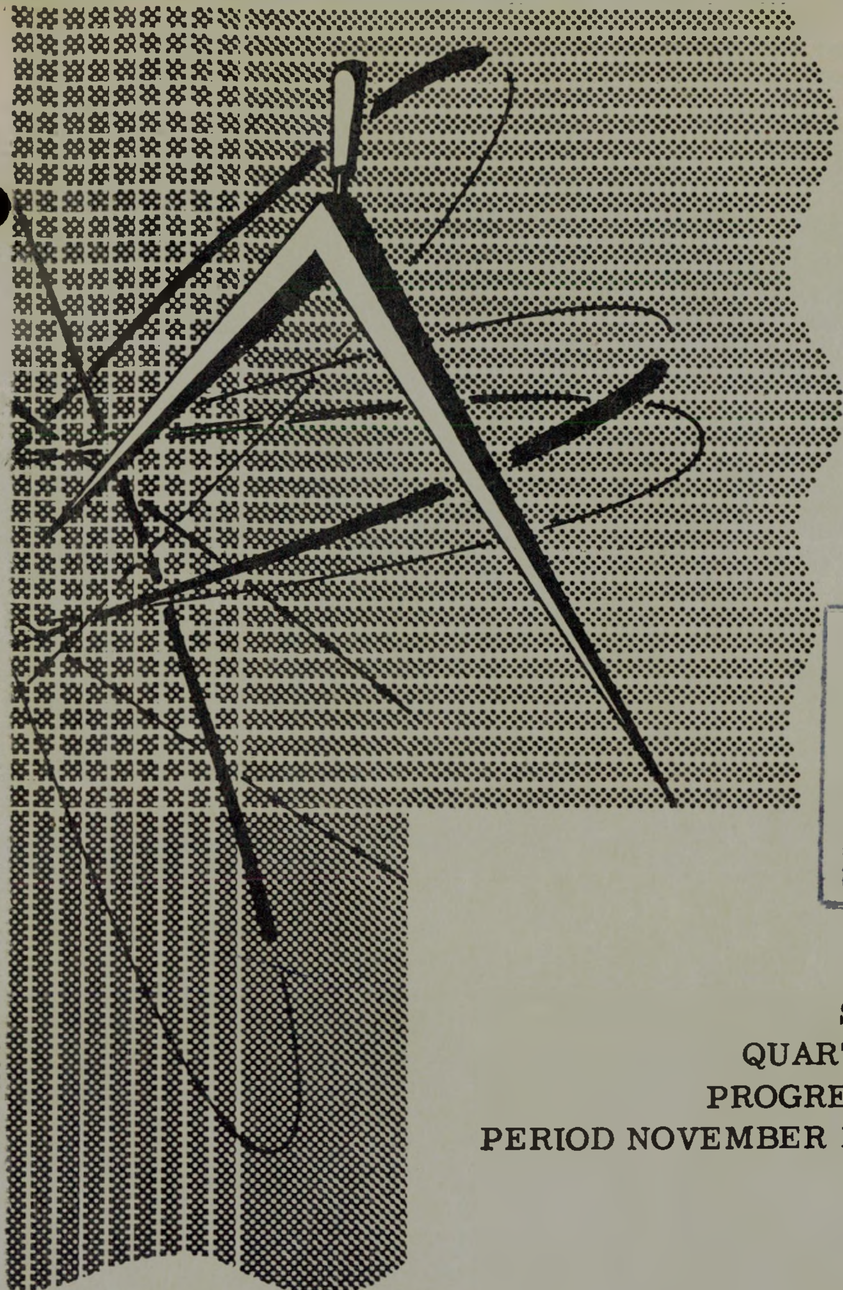
NDA 2084-5



SDR PROJECT
QUARTERLY TECHNICAL PROGRESS REPORT FOR THE PERIOD NOVEMBER 1, 1957 THROUGH JANUARY 31, 1958

MARCH 31, 1958

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

NDA 2084-5

SDR PROJECT
QUARTERLY TECHNICAL
PROGRESS REPORT FOR THE
PERIOD NOVEMBER 1, 1957 THROUGH JANUARY 31, 1958

MARCH 31, 1958

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

NUCLEAR DEVELOPMENT CORPORATION OF AMERICA
White Plains, New York

Information Category: UNCLASSIFIED

Morton S. Silberstein

Authorized Classifier

March 18, 1958

Date

CONTENTS

TABLES

FIGURES

INTRODUCTION

This is the third* Quarterly Technical Progress Report submitted by Nuclear Development Corporation of America summarizing work performed for the U. S. Atomic Energy Commission under Contract AT(30-3)-256. The period covered in this report is November 1, 1957 through January 31, 1958.

In assaying these accomplishments, it should be noted that the objectives of Phase I of the SDR program, as stated in the Contract, are: "to accomplish, to the extent necessary to permit a decision on whether to proceed with Phase II, the research, development and preliminary design work required to (a) demonstrate the feasibility of separation of sodium and heavy water in the SDR reactor and (b) establish potential of the SDR concept as an economic power producing system."

The Phase I Program has been divided into three major areas of effort whose objectives have been defined as follows:

1. Technical Planning and Evaluation

Initiate, organize and assure the proper execution of an appropriate program to achieve the overall project objectives. Evaluate the results of the program as a whole. Make available these results and evaluations in report and other forms as appropriate.

2. Sodium-D_2O Separation

Design a reliable sodium-D_2O system for this reactor and demonstrate experimentally

(a) that the reactor can be built with a low probability of mechanical failures of sodium and D_2O containers and barriers,

(b) that the design incorporate features which will avoid a single failure inducing multiple failures,

(c) that if multiple failures do occur in the D_2O and sodium systems and if the barrier between these two systems also fails simultaneously bringing sodium into contact with the D_2O, the accident from such contact can be contained.

---

* "A Summary Report of Accomplishments during the period January 25, 1957 to April 30, 1957," was submitted on May 29, 1957 (NDA 84-1), in addition to the regular Quarterly Technical Progress Reports (NDA 84-2, dated September 30, 1957, and NDA 084-3, dated December 31, 1957).

3. Preliminary Design

Generate a reactor preliminary design based on the SDR concept. Assure the compatibility of the reactor and plant components with the requirements of the SDR concept. Outline the general plant requirements, taking into account their interaction with the sodium primary and D_2O systems.

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 these tasks.

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

SUMMARY

SODIUM-D_2O SEPARATION

Engineering studies have continued on the through-tube reactor design discussed in NDA 084-3. Full-scale drawings showing the units in a complete lattice position have been prepared. A bellows assembly has been added to each calandria tube to accommodate possible thermal stresses.

Design of the barrier system now includes a second barrier tube between the stainless steel barrier and the calandria tube. The heat loss from the sodium in the fuel tubes to the D_2O in the calandria is calculated to be less than 0.3 MW.

Studies completed during the quarter indicate that sodium-water reactions do not inherently limit the feasibility of the SDR. A careful review of the literature has shown that sodium-water systems can be designed for safety against a reaction between the two fluids. Adherence to basic safety design rules prevents shock wave formation and high temperature peaks. Adequate surge volume, low system pressures, inert gas blanketing, and oxygen exclusion are the most important of these rules.

Fabrication and assembly of the apparatus for the fuel-coolant tube and header test program is nearing completion.

Single-failure barrier tests similar to those discussed in NDA 084-3 were performed, using higher sodium temperatures, on six specimens of type 1100 aluminum (2S). The feasibility of using aluminum as a barrier material against hot sodium has been demonstrated. Since the necessary information was obtained, the tests were concluded. Results of the single-failure test program have been summarized in NDA 084-4.

Apparatus for the multiple-failure barrier tests has been installed, and a test program has been established.

Detail design of the mockup test apparatus has been completed. Fabrication and assembly of the apparatus at the test site is nearing completion. A test program has been established which is directed at demonstrating the ability to handle sodium and water under simulated reactor temperature conditions.

PRELIMINARY DESIGN

The preliminary design is basically the same as that described in NDA 084-3. General studies during the quarter included an investigation of reactor maintenance techniques, fuel element studies, and reactor control and core physics calculations. It was tentatively concluded that a U–10 weight % Mo alloy is satisfactory for a slightly enriched SDR and that UO_2 is satisfactory for both a natural uranium and a slightly enriched SDR. Results of a series of two-group reactor criticality calculations indicate that a 6/1 fuel channel to control rod arrangement is satisfactory for the present design (~120 fuel tubes).

A brief study was made of the advantages and disadvantages of incorporating some of the desirable features of the slab design described in NDA 084-3 into the present calandria design.

Shielding studies were concentrated on the design requirements for the thermal shield and the bulk shield between the reactor and header rooms.

Preliminary design flow sheets for the D_2O cover gas, sodium cover gas, and barrier gas systems were prepared.

A preliminary study of the layout of the reactor building was started.

SODIUM-D_2O SEPARATION

TASK 2-1 SODIUM SYSTEM ENGINEERING

Design work on the sodium system has continued, using the arrangement described in NDA 084-3. The work reported below covers fuel-coolant tube and header design, fabrication of tubes and headers, sizing of pigtails, and the problem of heat loss to the neutron shields from sodium pipes penetrating the shields.

FUEL-COOLANT TUBE AND HEADER DESIGN

The design of the fuel tube and the associated pigtail and header arrangement remains substantially unchanged. A set of full-scale drawings showing a complete lattice position has been prepared. The spacing between tube centers has been tentatively increased from 8.6 in. triangular spacing to 10.0 in. triangular spacing. In addition to easing design problems within the shield, this change substantially increases the calandria tube sheet ligament and allows more space for swinging the upper pigtail during fuel tube replacement. A concurrent effort is being made to reduce the diameter of the reactor (and the number of fuel tubes and control rods), at the expense of a small increase in reactor height.

INSULATION AROUND SODIUM PIPES PENETRATING SHIELDS

With the present through-tube design, the fuel-coolant tubes must pass through thermal and neutron shields both above and below the calandria. Since the neutron shields probably will be composed mainly of concrete, the shield temperature should not be much above 200F. Thermal insulation between neutron shields and fuel-coolant tubes will be required. Consideration of neutron streaming, together with that of the allowable heat loss from the sodium to the neutron shields limit the choice of insulation and the space which it occupies. With 5-ft thick neutron shields, two 0.1-in. gas spaces between the fuel tube and concrete, each stepped half way along their lengths, will satisfy neutron streaming limitations. The resulting insulation provided by these spaces will limit heat losses to about 0.6 MW for the upper neutron shield and about 0.4 MW for the lower shield while the reactor is at full power. Design efforts are currently under way in an attempt to lower the heat loss.

The problem of insulating the thermal shield should not be difficult since the allowable operating temperature can be relatively high. It is not necessary to maintain the small gas gaps to reduce neutron streaming as in the neutron shields. In addition the thermal shields will probably be made of iron and will not be required to support significantly more than their own weight.

TASK 2-2 D_2O SYSTEM ENGINEERING

CALANDRIA DESIGN

It was indicated in the previous quarterly report that expansion bellows for the attachment of the calandria tubes to the upper calandria tube sheet might not be required. Subsequent analyses have shown that bellows should probably be included to insure safe operation. Consideration was given to two possible conditions where the presence of expansion bellows might be required: first, where an appreciable temperature difference exists between the calandria tubes and the calandria shell (here, the calandria tubes are all assumed to be at the same temperature, which is higher than that of the calandria shell), and second, where such a temperature difference exists between one isolated calandria tube and all the surrounding tubes. The first condition would reach an extreme during an emergency dumping of the D_2O and the second condition would be significant if coolant ceases to flow through a single fuel channel. On the basis of limiting values of stress for the tubes and the tube sheets, the analyses showed that for the first condition, the maximum temperature difference that could be withstood by the calandria tube without a bellows attachment is 80F. For the second, a maximum temperature difference of 70F would be permissible. Present information indicates that the values given above would probably be exceeded. Therefore, bellows appear desirable to safeguard against a failure of a tube, tube sheet, or a tube joint.

The design of the bellows and its end connections is receiving detailed attention at this time since the size of the bellows and means of attachment are a controlling factor in the determination of the lattice spacing. The present arrangement of bellows and end fittings allows for remote replacement of either the bellows or the calandria tube. Vendors have indicated the feasibility of the present design. More detailed information is being developed.

TASK 2-3 BARRIER SYSTEM ENGINEERING

BARRIER DESIGN

The design of the barrier system within the reactor remains substantially unchanged. Provision has been made in the design for inserting a second barrier tube between the 0.020 in.-thick stainless steel barrier and the 0.125 in.-thick calandria tube. This secondary barrier would probably be a tube of 0.065 in.-thick aluminum. Besides providing an additional obstruction between sodium and D_2O, this tube would help to further reduce heat losses to the D_2O system and would improve the distribution of the blanket gas which flows through the reactor under certain conditions. This tube can be supported by the inner stainless steel barrier tube or by the shield above the reactor. This general barrier configuration, along with others, is being tested in the SDR experimental program. (See Task 2-5, Multiple-Failure Tests.)

BARRIER HEAT TRANSFER STUDIES

The rates at which heat will be lost from the fuel-coolant tube through various barrier configurations were estimated. The results showed that with the present two-tube barrier, heat losses from the fuel tubes to the D_2O will be less than 2.8 kw per tube (or 0.33 MW for 120 tubes) at full reactor power. Reversing the present materials of the barrier tubes (inner tube – SS, outer tube – Al) will reduce heat losses by about 20% but results in aluminum tube operating temperatures which are higher than desirable. A single stainless steel tube barrier located in the center of the space between the fuel tube and the calandria tube will result in only about 10% larger heat losses than the present two-tube design.

SODIUM-WATER REACTION STUDIES

Work on the sodium-water reaction problem continued. Additional literature sources were studied and trips were taken to the Oak Ridge National Laboratory, Atomic Power Development Associates, Mine Safety Appliances and the Knolls Atomic Power Laboratory.

A topical report is currently being written which summarizes the chemical problems in a sodium cooled, heavy-water-moderated reactor. The major conclusion of the study is that sodium-water reactions do not inherently limit the feasibility of t'.e SDR. A careful review of the literature has shown that sodium and water systems can be designed for safety against the reaction and that adherence to basic safety design rules will prevent shock wave formation and high temperature peaks. Adequate surge volume, low system pressures, inert gas blanketing, and oxygen exclusion are the most important of these rules.

TASK 2-4 FUEL-COOLANT TUBE AND HEADER TESTS

The apparatus for the SDR fuel-coolant tube and header test program has been designed. Fabrication and assembly are now nearing completion.

The major objective of the static and mechanical cycling tests is to determine the reliability of the austenitic stainless steel piping and weldments of various designs, when subjected to strain cycling at elevated temperatures under simulated reactor operating conditions.

GENERAL ARRANGEMENT

The tube and header tests will be conducted in the Engineering Building, which is the central control and recording station for the SDR experimental facility at the NDA Pawling Laboratory.

The majority of the fuel-coolant tube and header tests will be performed on a shaper fatigue testing machine (Fig. 2-4.1). This apparatus, which consists of a modified shaper mechanism mounted on an L-shaped structural steel frame, is used to stress pigtail-to-header and pigtail-to-coolant tube weld specimens through appropriate load transmitters and load-measuring devices. Specimens are rigidly held at the header or coolant tube end by means of an adapter fastened to a wide-flange beam. Deflections are measured by means of a long-stroke dial indicator. Static testing will be performed manually at room temperatures using a crank to attain desired deflections. In the fatigue tests "clam-shell" molded ceramic heating elements will be used to maintain the desired temperature in the region of the pigtail-to-header connection.

Fig. 2-4.2 is a photograph of the test apparatus, load cell, and loading device.

DESIGN FEATURES

Two load cells, with 120-lb and 600-lb capacity, are used for testing over the required fatigue test range. They will also serve for static tests and full-scale tests which simulate more closely typical pigtail designs. Loads can be continuously recorded for test control and subsequent analysis of the data obtained.

Deflection data will be obtained, using the long-stroke dial indicator, which is adjustably mounted to permit readings over a 6 in. range. During fatigue tests the dial indicator will be disconnected.

A multipoint recorder will be used to monitor and record test specimen temperatures. The temperature distribution relative to the control temperature will be adjusted manually by means of three variable auto-transformers operating on each of the three separate circuits built into the "clam-shell" heaters.

Using the present equipment, four loading frequencies are available. These frequencies will be obtained by using the cone pulleys supplied with the shaper mechanism and a ratio motor. For normal tests the ratio motor will be used to reduce the cyclic loading frequency to approximately

5 cycles per min. Three other loading frequencies are expected to be between approximately 10 and 2 cycles per min.

An auxiliary vacuum system has been constructed for pre- and post-test inspection of the test specimens. The vacuum system consists essentially of a mechanical "roughing" pump, diffusion pump, and two vacuum gages, together with associated manifolding. The entire vacuum system is mounted in a rolling frame approximately 5 ft long × $3\frac{1}{2}$ ft high × $2\frac{1}{2}$ ft deep (Fig. 2-4.3). It is intended to test the specimens by evacuating the interior to about 10^{-5} mm of mercury absolute, measured by means of a cold-discharge ionization gage. A thermocouple gage is used for measuring higher pressures. Microfissures which might develop in the strain-cycled specimens may be determined by failure of the vacuum system to reduce the interior pressure to the pretest value.

FABRICATION STATUS

Fabrication of the equipment for testing simulated pigtails is nearing completion. Shakedown of the apparatus at the Engineering Building at Pawling is scheduled to be completed during February 1958.

TESTING PROGRAM

The testing program consisted of weld development work, a static test program, and a fatigue test program.

Weld Development

The results of this portion of the program are as follows. (See Figs. 2-4.4 through 2-4.7.)

1. As was expected, the weld in View 1 was the easiest to fabricate by both field and shop fabrication. Preparation was simple, and fit-up was easily accomplished.

2. The design shown in View 2 was easier to fit up and had greater accessibility than the design shown in View 3. In addition, penetration was easier to control, and less buildup was required to fill out the weld.

3. The design shown in View 3 was the least desirable because of excessive and difficult preparation and fit-up time. Weld buildup inside the pipe was a problem, and the gap was difficult to maintain in tacking.

4. Welds for thin wall tubing were made as shown in View 4, and both joints gave good welds. Alternate 1 (with a flared joint) proved easier to make and control. However, unless the ends could be prepared in the shop, it would be difficult to make such a joint in the field. The elimination of the use of filler rods makes this a desirable joint. Simplicity of preparation, however, favors the straight butt joint shown in Alternate 2.

5. For the thick-walled fuel tube (View 5), Alternate 1 was easier to fabricate than Alternate 2. The latter is difficult to control, and the high heat required to penetrate the butt joint tended to make a non-uniform penetration on the inside of the pipe. In addition, the arc tended to blow through in any misfit crack or opening.

Static Test Program

The purpose of the static test program is to determine the type of ultimate failure that is most likely to occur, to get a representative value of the load required to cause failure, and to compare, evaluate, and screen the several weld types proposed for each joint.

The static tests will consist of flexure tests for pigtail-to-header and pigtail-to-fuel tube joints, and tensile tests for light- and heavy-walled fuel tube joints. All tests will be run at room temperature.

Fatigue Test Program

The purpose of the fatigue tests is to determine the kind of failure which will occur after repeated mechanical cycling, to get a representative value of the number of cycles required to cause failure at elevated temperatures, and to select the best weld from among those proposed for each joint.

The tests consist of mechanical cycling flexure tests at constant deflection of top pigtail-to-header joints, bottom pigtail-to-header joints, and top fuel tube-to-pigtail joints. The joints themselves will be kept at a constant elevated temperature for the duration of the test. Top pigtail-to-header and top fuel tube-to-pigtail joints will be tested at 1050F. The remainder of the test sections (bottom pigtail-to-header joints) will be tested at 750F.

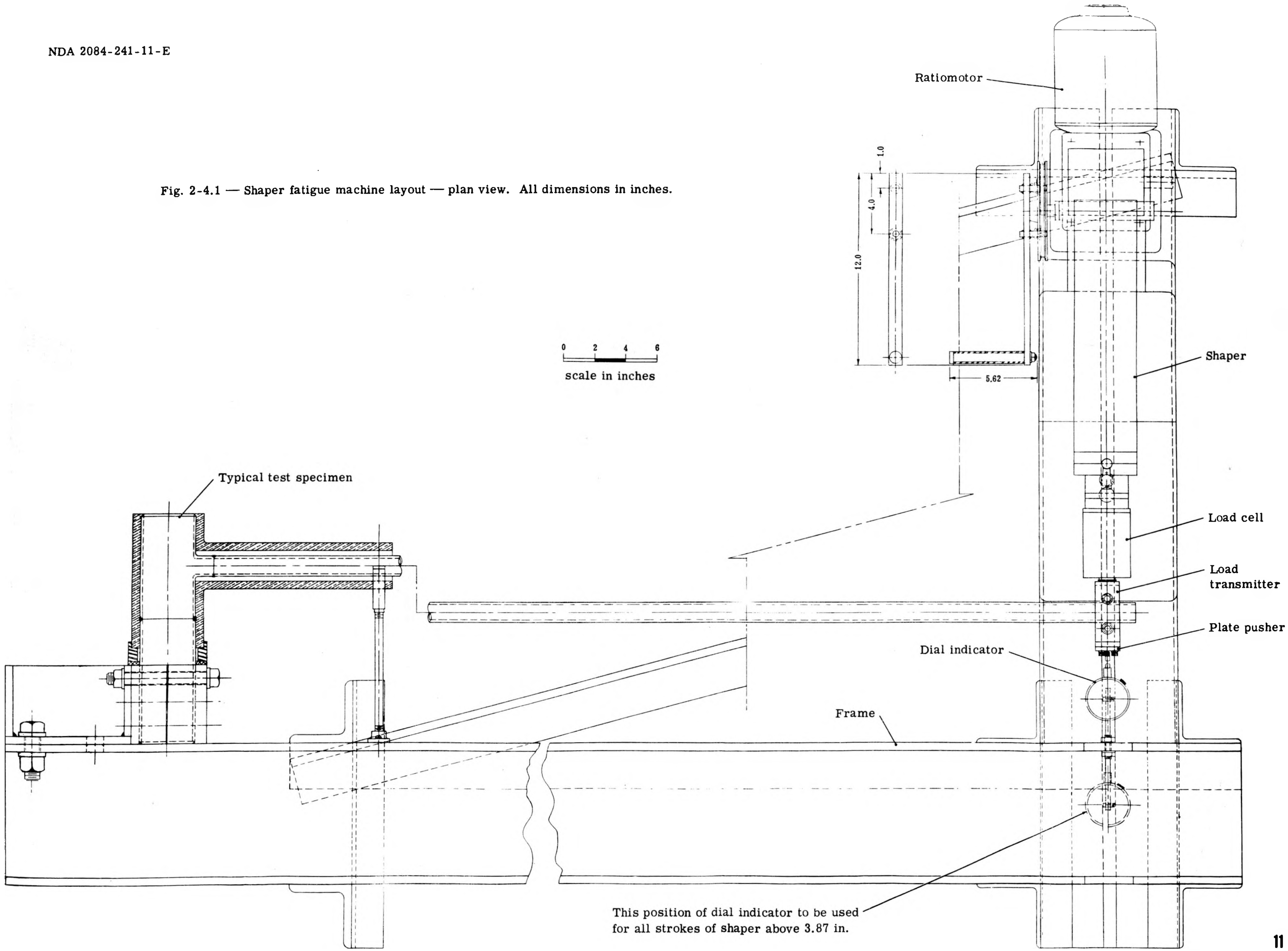
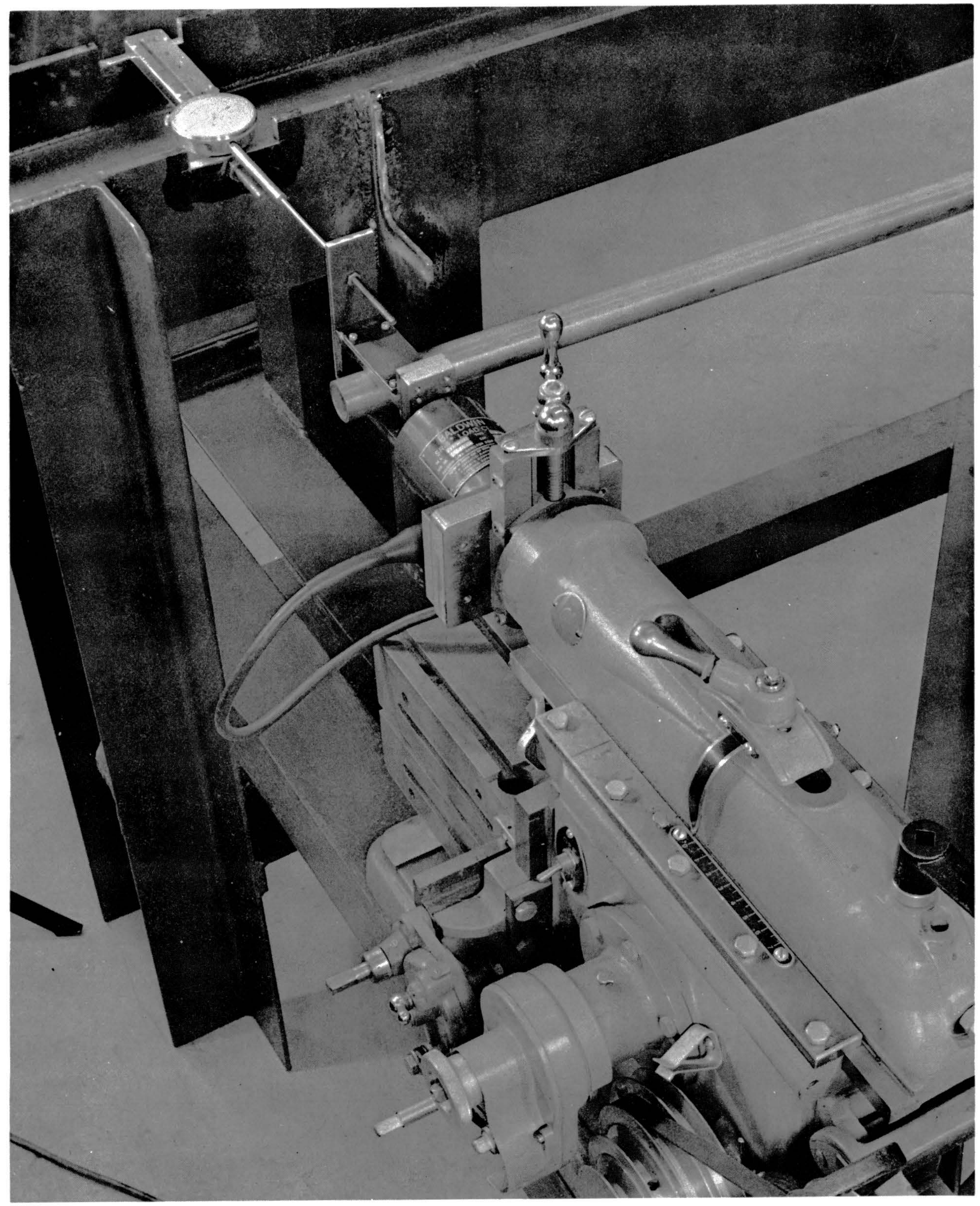


Fig. 2-4.1 — Shaper fatigue machine layout — plan view. All dimensions in inches.



NEG. NO. 996

Fig. 2-4.2 — Test apparatus, showing shaper mechanism, load cell, test sample, and dial indicator

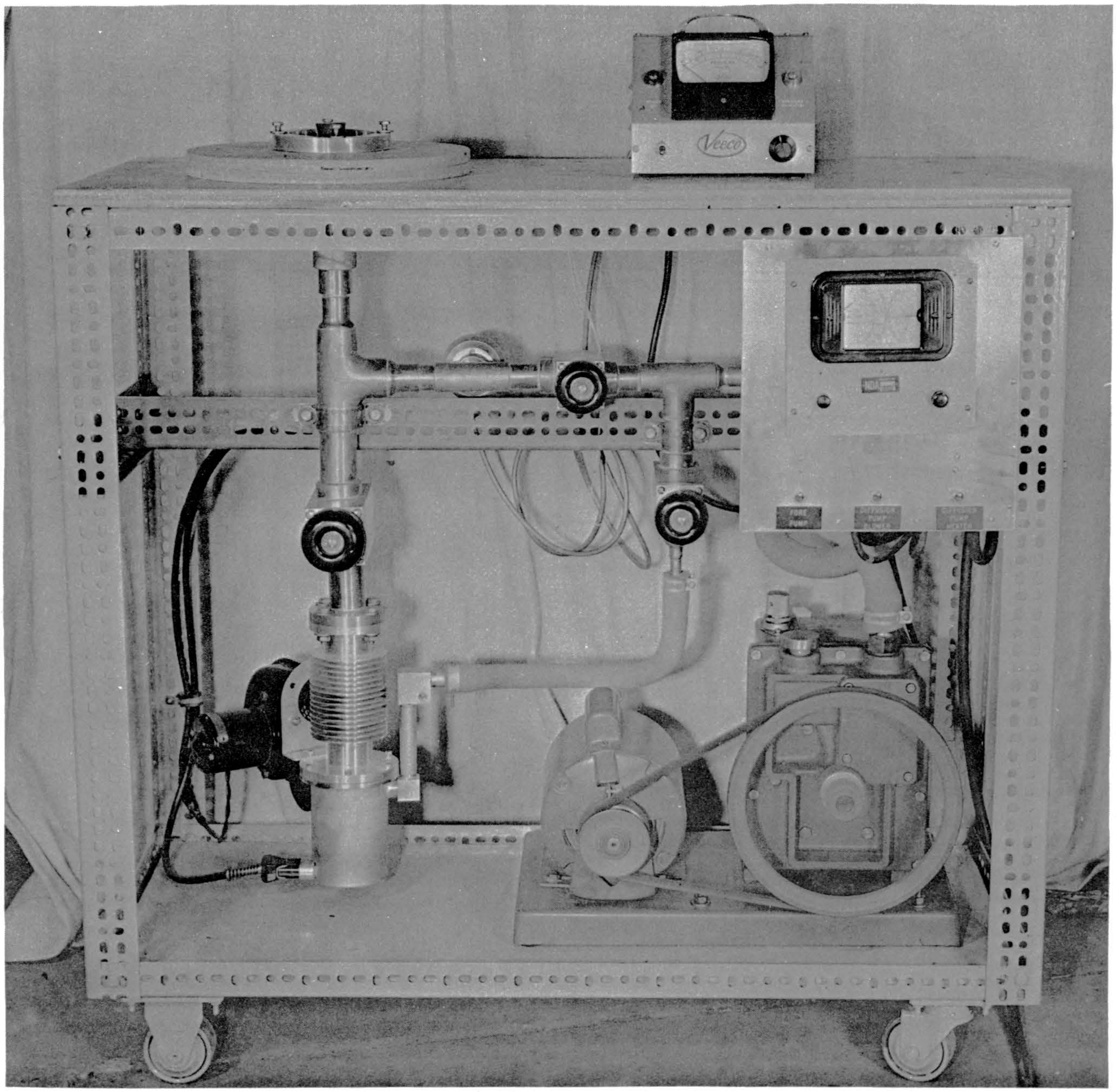


Fig. 2-4.3 — Vacuum cart, showing diffusion pumps and gages

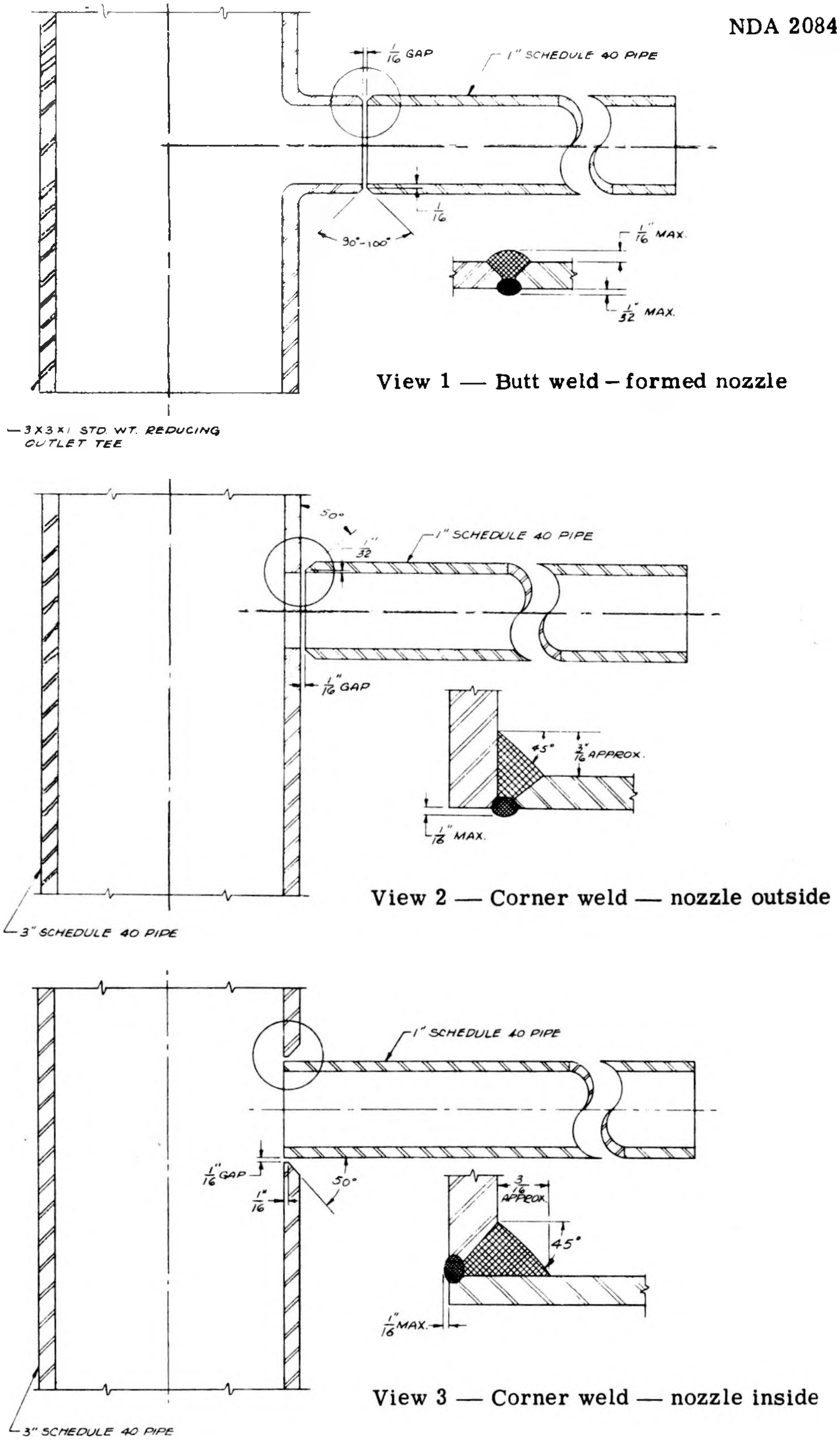
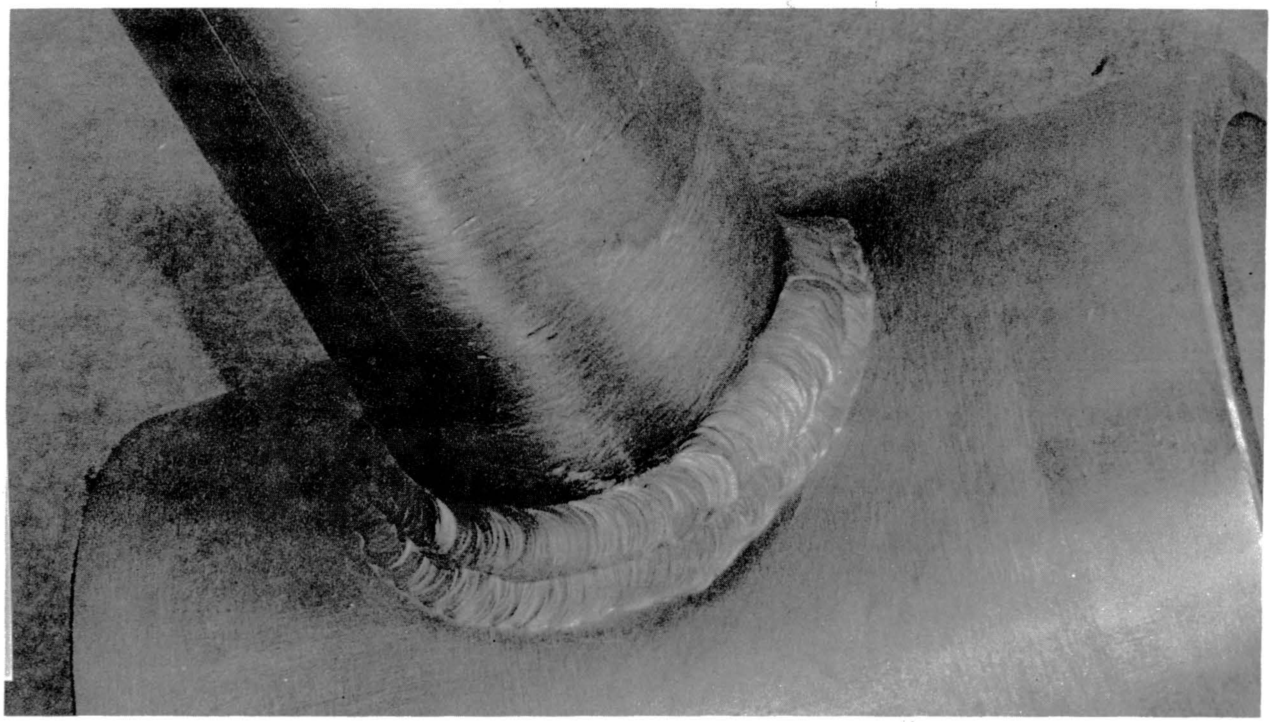
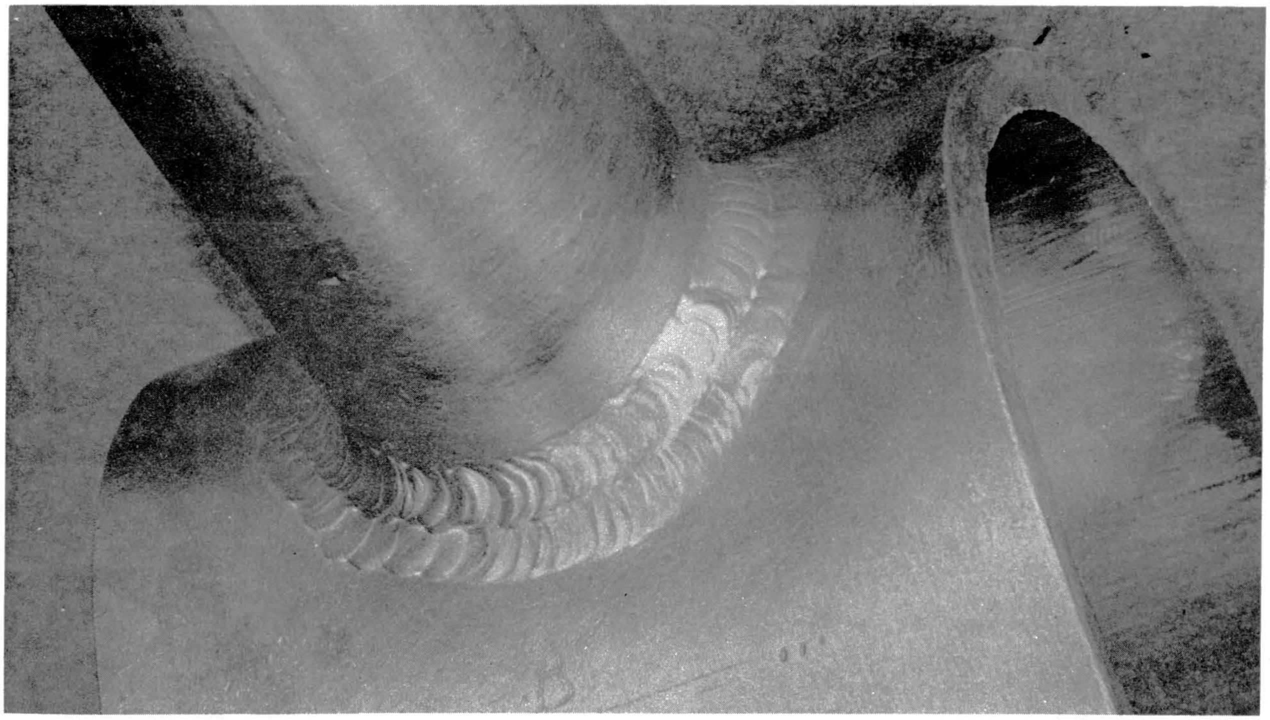


Fig. 2-4.4 — Welds for pigtail-fuel tube and pigtail-header connections. All dimensions in inches. All material 316 SS.

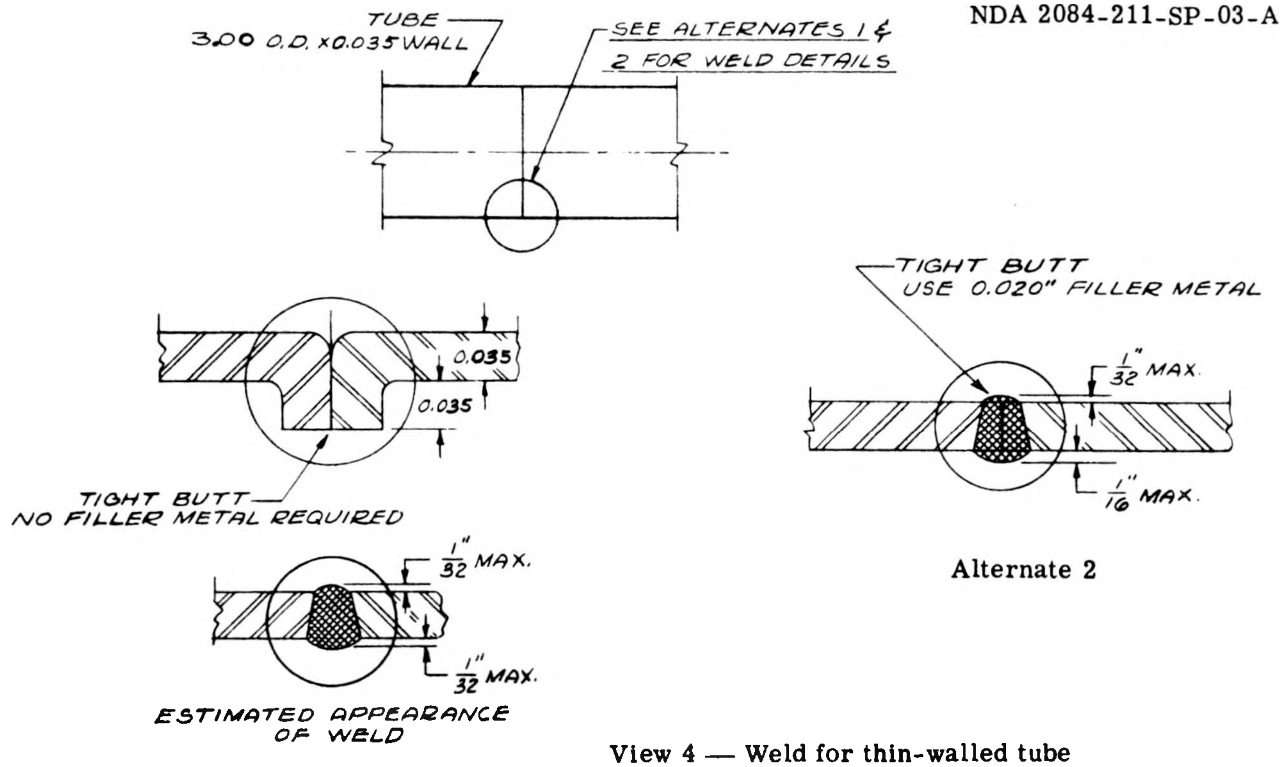


NEG. NO. 933



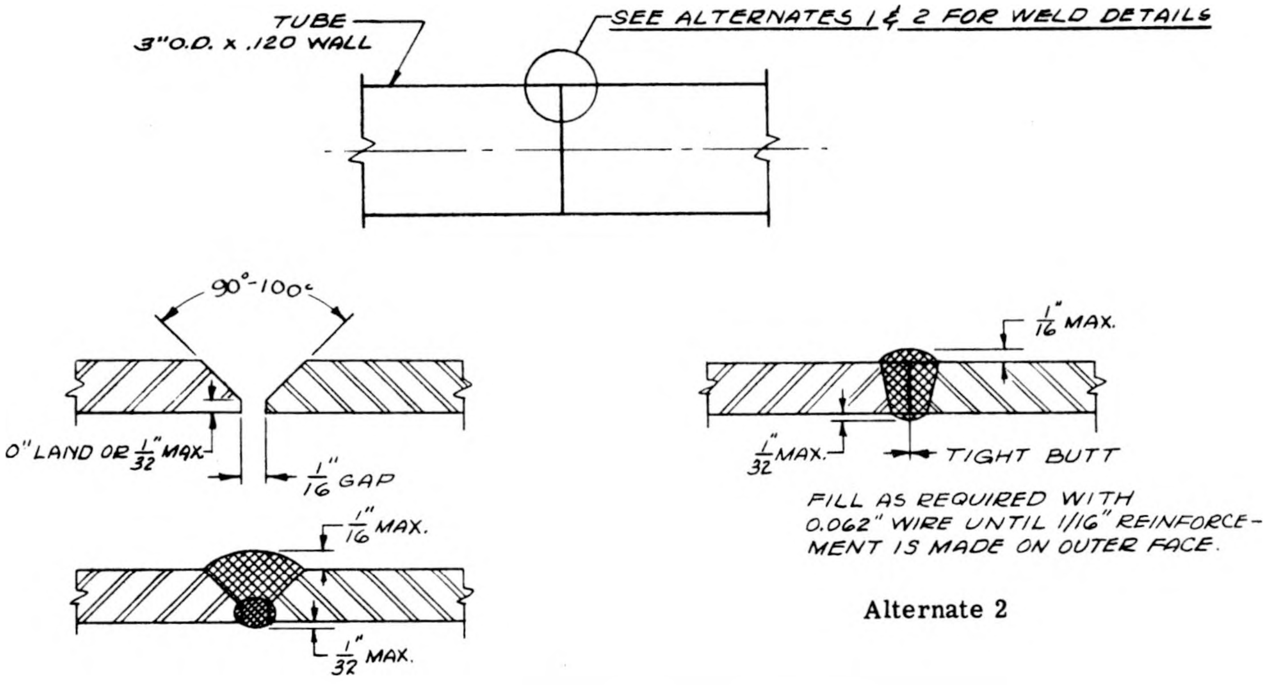
NEG. NO. 940

Fig. 2-4.5 — Corner welds. Top — nozzle outside. Bottom — nozzle inside.



Alternate 1

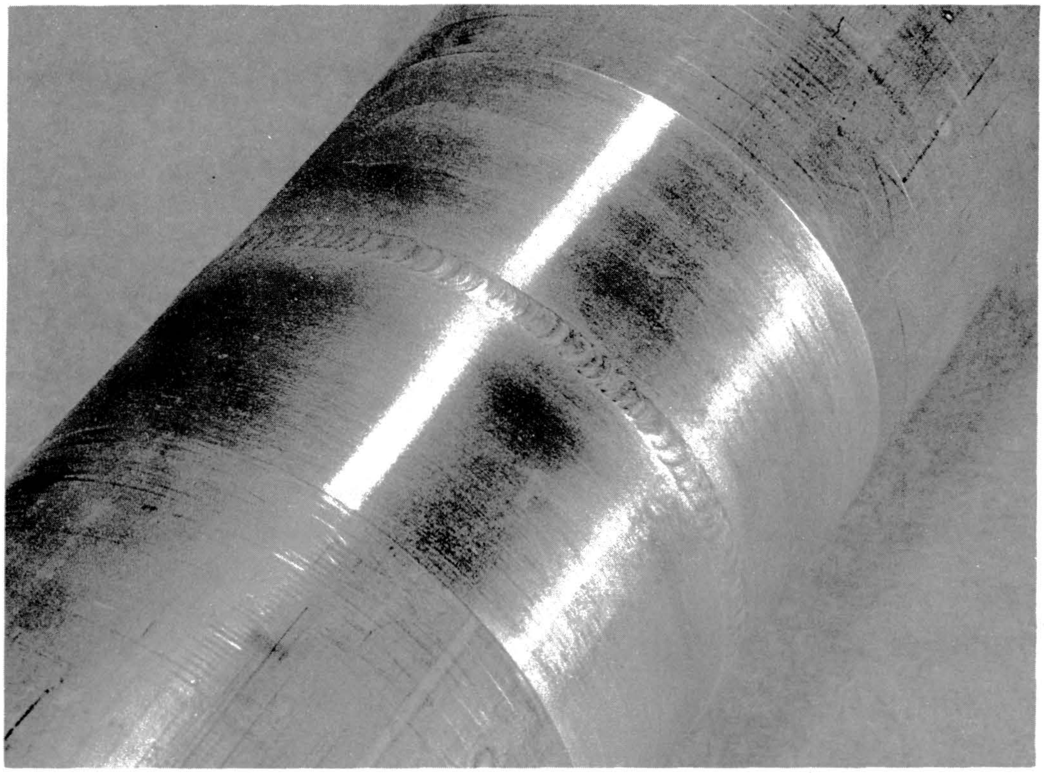
View 4 — Weld for thin-walled tube



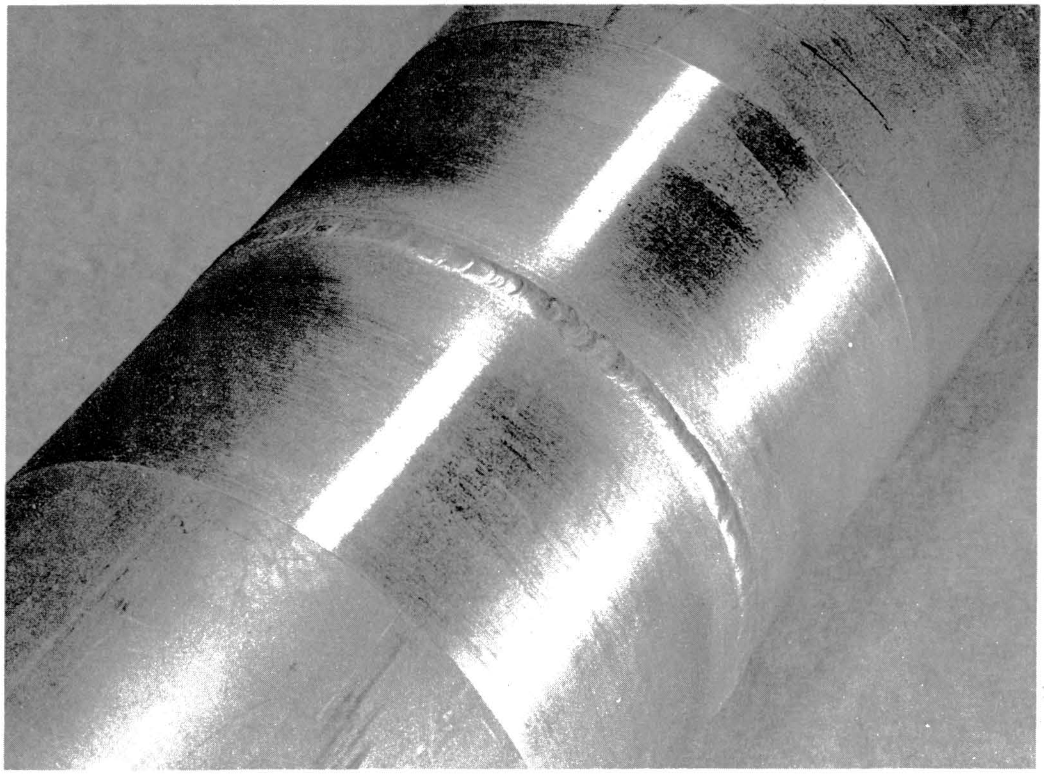
Alternate 1

View 5 — Weld for thick-walled tube

Fig. 2-4.6 — Welds for thin- and thick-walled fuel tube. All dimensions in inches. All material 316 SS.



NEG. No. 335



NEG. No. 934

Fig. 2-4.7 — Welds for thin-walled fuel tube. Top — alternate 1. Bottom — alternate 2.

TASK 2-5 BARRIER TESTS

SINGLE-FAILURE TESTS

The preceding quarterly report (NDA 084-3) discussed results of single-failure barrier tests using 950F sodium. During this quarter a run was made using higher temperatures (1000F to 1140F) on six specimens of type 1100 aluminum (2S).

High Temperature Runs

The test conditions of the high temperature run are listed in Table 2-5.1. The samples were 2 × 2 in. square and 0.060 in. thick.

Table 2-5.1 — Single-Failure Barrier Test Results at High Temperatures

Material: 1100 aluminum

Specimen No.	Sodium Jet Temperature, °F	Time	Sodium Jet Velocity, fps
13H	1000	15 min	45
13J	1050	15 min	40
13K	1050	3 hr, 40 min	40
	1140	20 min	40
13L	1050	$3\frac{1}{2}$ hr	50
13M	1100	$3\frac{1}{2}$ hr	50
13N	control	0	0

Figs. 2-5.1 through 2-5.5 show some of the results of these tests. Figs. 2-5.1 through 2-5.3 show that there was no subsurface attack in any of these high temperature runs. In addition there is little, if any, difference between the condition of Specimens 13L and 13N, the former having been run at 1050F for $3\frac{1}{2}$ hr and the latter being an as-received control. (The blurred edge in 13L results from rounding occurring in the polishing prior to photographing.) Specimens 13H and 13J were not affected by the jet. There is no discernible damage to Specimen 13L where the jet hit. Pitting of unknown origin can be seen in Specimens 13K, 13L, and 13M.

Cross sections of pits in Specimens 13K and 13M are shown in Figs. 2-5.2 and 2-5.3. The pit shown on the photograph of Specimen 13M is one of the "small" peripheral pits. No microstructural changes are visible in the region of attack.

Specimen 13M was eroded by the sodium. When a few mils of material remained, the jet tore through (after $2\frac{1}{2}$ hr). The piece is visible in the lower picture on Fig. 2-5.5.

Summary of Results of Single-Failure Barrier Test Program

It was found that aluminum performed well in these tests. No failure of samples having thicknesses of practical interest occurred at 950F in tests which ranged up to 18 hr. Failure was finally achieved at 950F with a 5-mil aluminum sample. It was found that 0.060 in. aluminum plate also performed well at 1050F; the maximum test time at this temperature was $3\frac{1}{2}$ hr. One sample was tested at 1140F and stood up for the 15 min duration. One sample tested at 1100F failed after $2\frac{1}{2}$ hr.

Zirconium and steel performed well, as was expected. While graphites in general performed poorly, AGOT-type (reactor grade) performed satisfactorily.

The feasibility of using aluminum as a barrier against sodium has been experimentally demonstrated. The tests were concluded and a topical report on the single failure barrier tests was prepared.*

MULTIPLE-FAILURE TESTS

General Arrangement

The multiple-failure apparatus described in NDA 084-3 was assembled and installed at the NDA Pawling Laboratory (see Fig. 2-5.6). The recording and control system is located in the Engineering Building to allow remote operation of the experiment. This is accomplished using cable raceways which connect the Engineering Building and Multiple-Failure Structure (see NDA 084-3, p. 32).

Testing Program

A program has been established to test the effects of

1. water squirt on barrier,
2. sodium squirt on barrier,
3. simultaneous sodium and water squirts on barrier,
4. water squirt on sodium filled tube (barrier removed),
5. sodium squirt on aluminum tube containing water.

To accomplish these tests, three different types of test sections have been designed and are currently being fabricated.

Design and Fabrication of Test Sections

The first test section contains a barrier consisting of concentric tubes of stainless steel and aluminum. This barrier arrangement has been described previously in NDA 084-3 (see Fig. 2-5.10). A water tank simulates the aluminum calandria tube, while a stainless tube containing stagnant sodium simulates the fuel coolant tube. With this test section, a stream of water at about 200F and/or a stream of sodium up to 1150F can be squirted against opposite sides of the barrier structure. (See Figs. 2-5.7 through 2-5.9.)

A second set of barriers has been designed and fabricated. In this case the steel tube is welded to the sodium funnel and drain line, forming a continuous sealed path for squirted sodium to flow out of the test section.

---

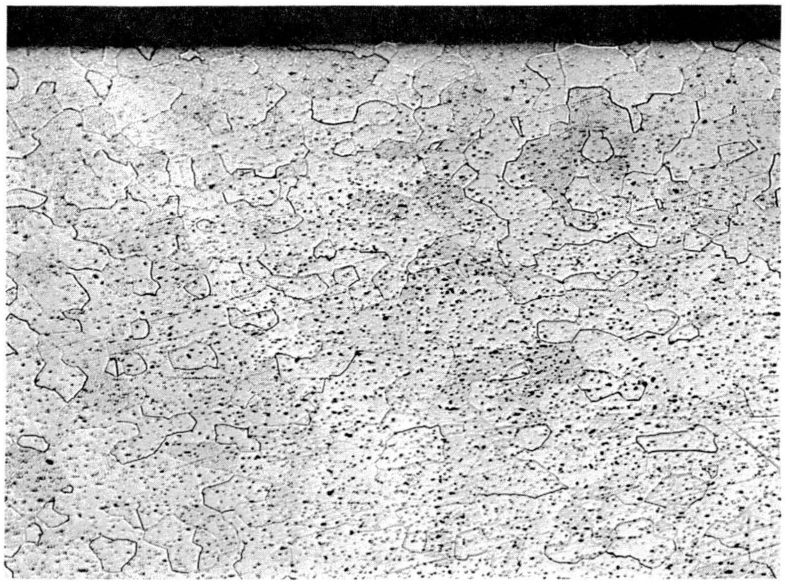
* Resistance of Barrier Materials to Sodium Jet Impingement, NDA 084-4 (February 26, 1958).

A second test section will test the effects of sodium impingement upon an aluminum tank containing water. Water will flow past the opposite point of impact of the sodium stream at velocities up to 5 ft/sec. (See Fig. 2-5.10.)

A third test section is designed to test the effects of water streams on a steel tube containing hot sodium. This test section is made from assembling parts that were designated for the other test sections.

Fabrication of all test sections has been started, and assembly of the first is completed.

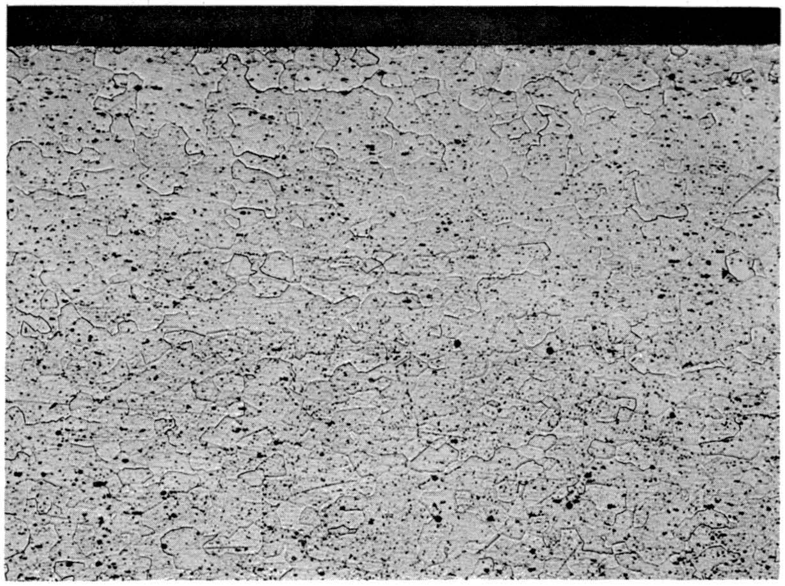
Shakedown of the electrical system, water system, and gas system has been completed. Shakedown of the sodium system is in progress.



NEG. No. 903H1

Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Jet Velocity: 50 fps
Test Duration: 3 hr, 30 min

Temperature: 1050 F
Specimen No.: 13 L
Magnification: 170 ×

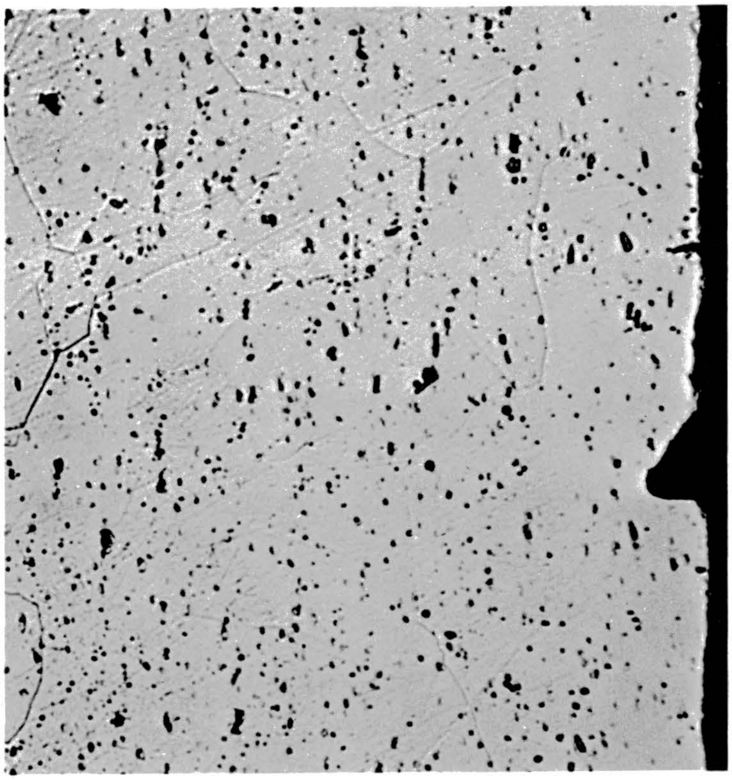


NEG. No. 903A1

Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Specimen No.: 13 E

Magnification: 170 ×
Remarks: as received (annealed)

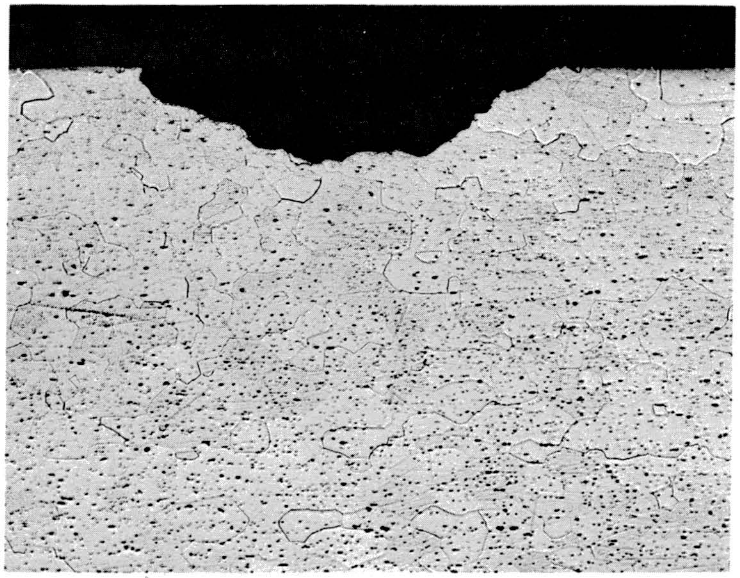
Fig. 2-5.1



NEG. NO. 903K

Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Jet Velocity: 40 fps
Test Duration: 3 hr, 40 min at 1055 F,
then 20 min at 1140 F
Specimen No.: 13 K
Magnification: 500 ×

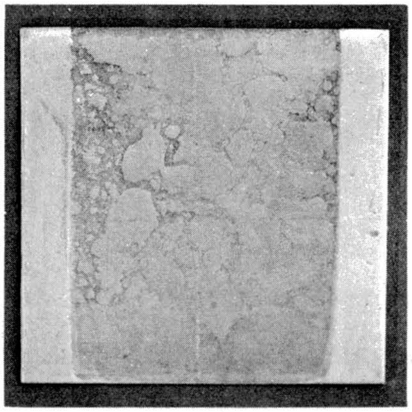
Fig. 2-5.2



NEG. NO. 903E4

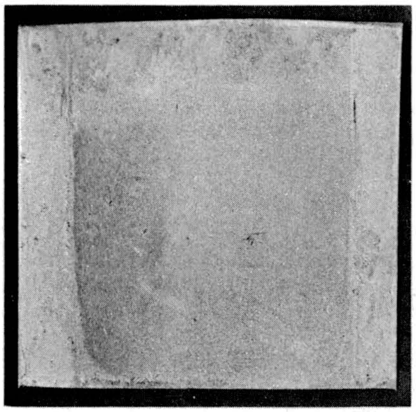
Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Jet Velocity: 50 fps
Test Duration: 3 hr, 30 min
Temperature: 1100 F
Specimen No.: 13 M
Magnification: 170 ×

Fig. 2-5.3

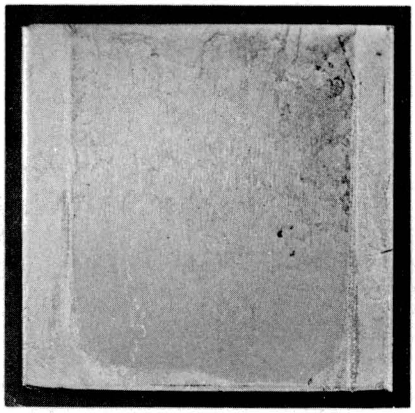


NEG. NO. 903F

Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Specimen No.: 13 N
Remarks: exposed to sodium atmosphere at high temperature for about 12 hr; not tested with jet, used as control



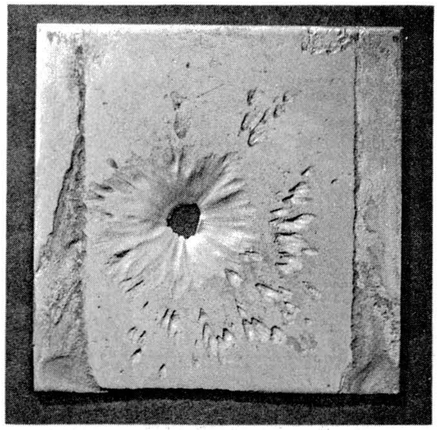
Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Jet Velocity: 40 fps
Test Duration: 3 hr, 40 min at 1055 F, then 20 min at 1140 F
Specimen No.: 13 K



NEG. NO. 903H

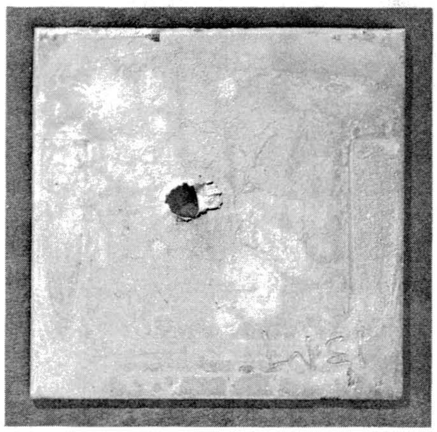
Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Jet Velocity: 50 fps
Test Duration: 3 hr, 30 min
Temperature: 1050 F
Specimen No.: 13 L

Fig. 2-5.4



NEG. NO. 903E1

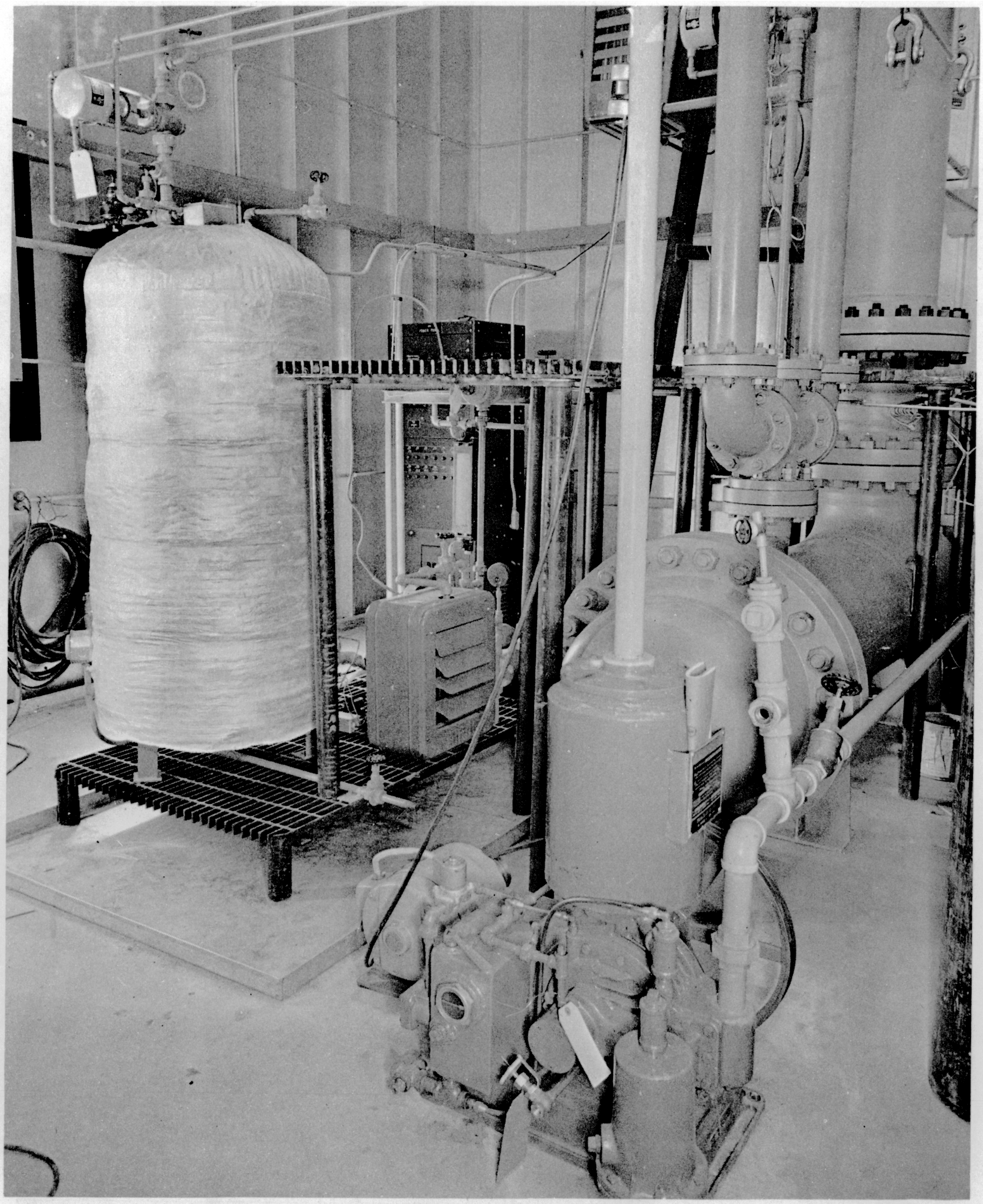
Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Jet Velocity: 50 fps
Test Duration: 3 hr, 30 min
Temperature: 1100 F
Specimen No.: 13 M
Remarks: front (side impinged on)



NEG. NO. 903E3

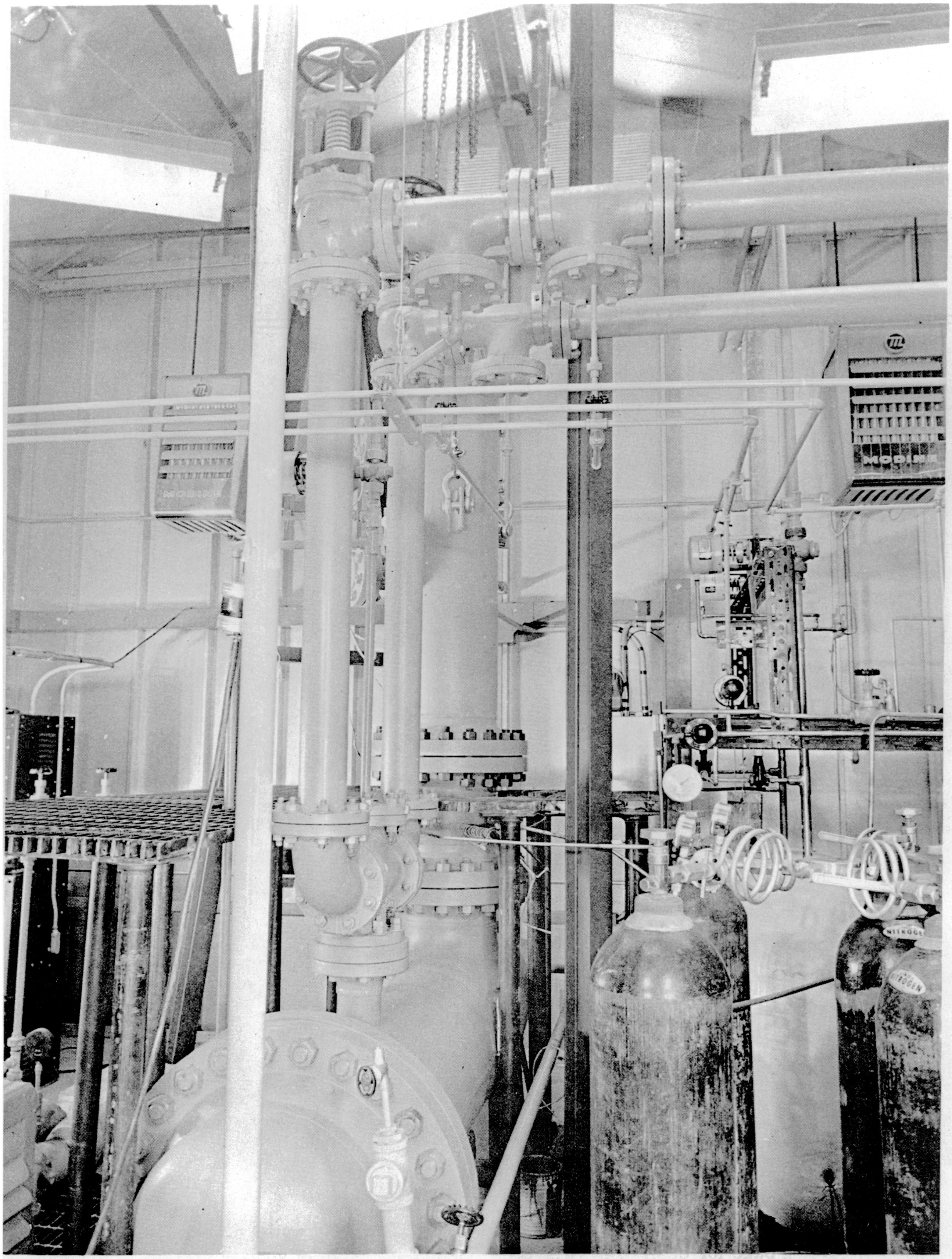
Material: aluminum, 2S (1100)
Thickness: 0.060 in.
Jet Velocity: 50 fps
Test Duration: 3 hr, 30 min
Temperature: 1100 F
Specimen No.: 13 M
Remarks: rear

Fig. 2-5.5

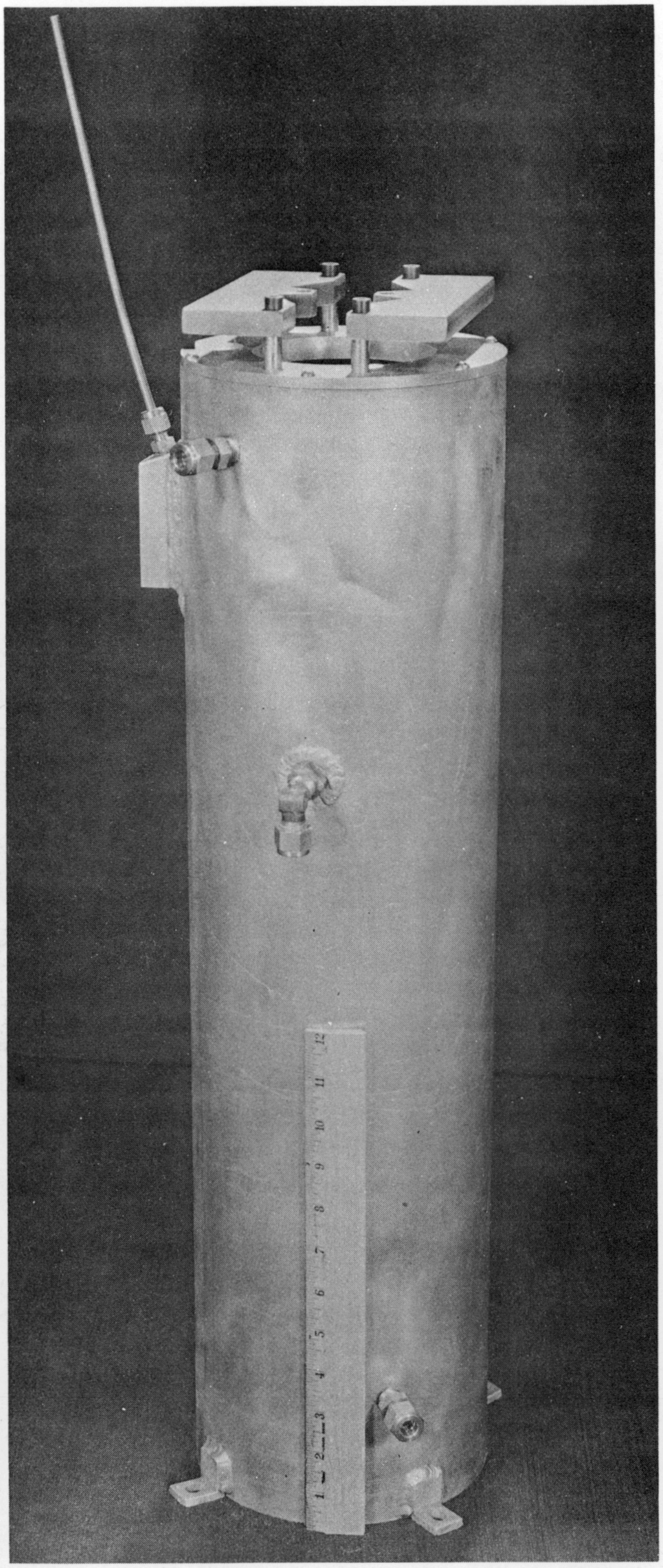


NEG. NO. 1002

Fig. 2-5.6 — Multiple-failure test apparatus, including water system. See facing page for other view, which includes components of sodium system.

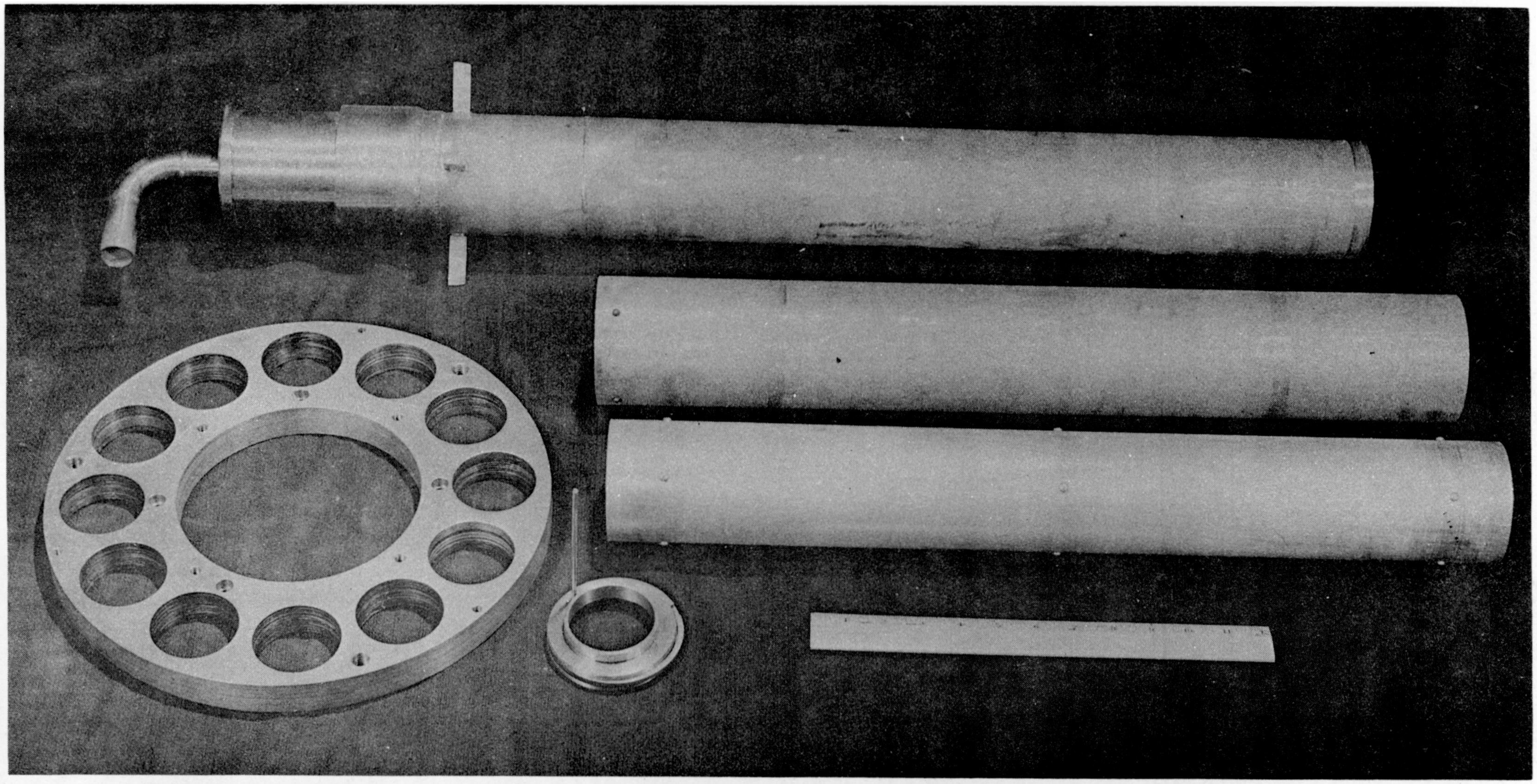


NEG. NO. 1006



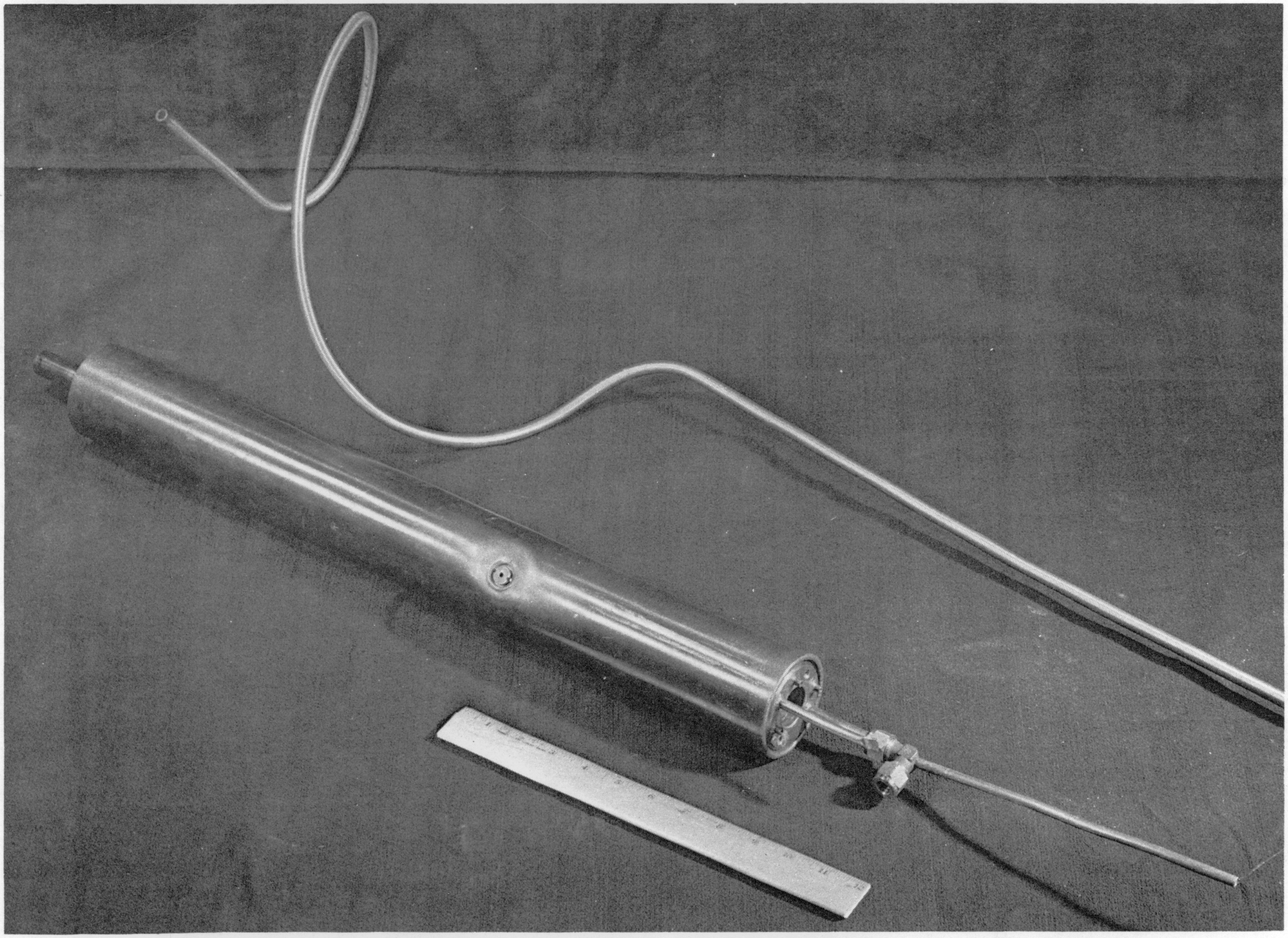
NEG. NO. 960

Fig. 2-5.7 — Test section part — aluminum water tank



NEG. No. 966

Fig. 2-5.8 — Sodium tube, inner and outer barrier tubes, and base plate



NEG. NO. 967

Fig. 2-5.9 — Sodium squirt tube

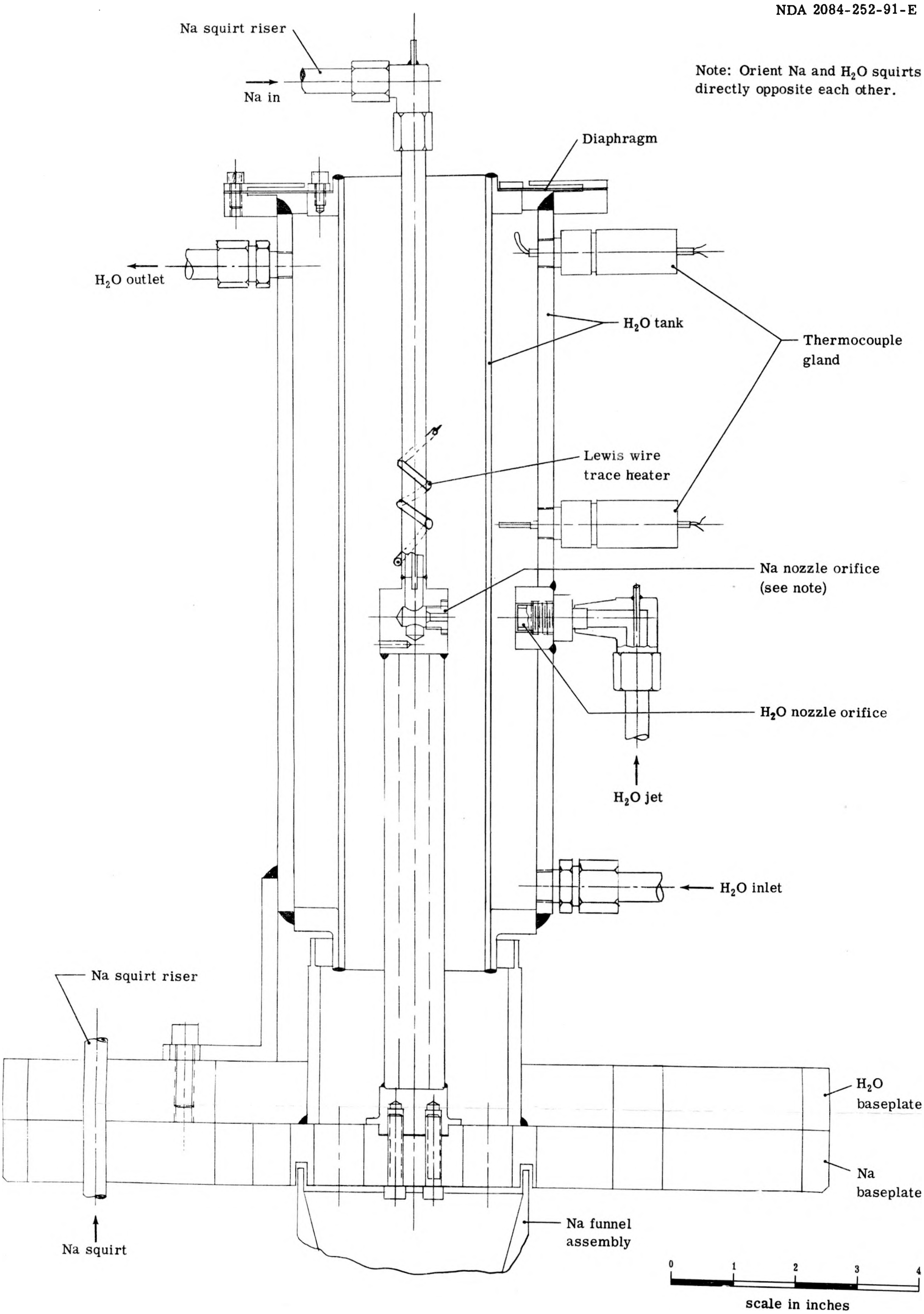


Fig. 2-5.10 — SDR – multiple failure test section – sodium squirt on water – layout

Blank Page

TASK 2-6 MOCKUP TESTS

DETAILED DESIGN

Detailed design of the test apparatus described in NDA 084-3 has been completed. The apparatus consists of a mockup of three reactor fuel-coolant tubes and their barrier systems in a water-filled calandria-type tank; provisions have been made for supplying sodium, moderator water, and barrier gas.

LEAK DETECTION SYSTEM

A leak detection system has been designed to indicate the presence of liquid sodium, water, water vapor, or oxygen between the water and sodium containers of the mockup. It is shown schematically in Fig. 2-6.1.

Detection of liquid water and liquid sodium is accomplished by the installation of stainless steel shorting wires in the water catchpan and in the upper and lower sodium catchpans. (The catchpans are designed to catch and contain any liquid escaping from either the water tank or any part of the sodium system which is inside the mockup, and to maintain separation of liquid water and sodium in the event of simultaneous leaks in both the water system and the sodium system.)

In order to monitor the barrier gas (nitrogen) for water vapor and oxygen, a continuous gas sample (approximatley 200 cc/min) will be drawn from the upper header box. This nitrogen sample will pass through remote-indicating electronic gas analyzers which will transmit dewpoint and oxygen content readings to recorders located at the Engineering Building.

FABRICATION STATUS

Test Section

Fabrication and assembly of the components is progressing rapidly. The assembled pigtails and header is shown in Fig. 2-6.2. The calandria (test section) is shown in Fig. 2-6.3.

Water System

The water system is completely fabricated and assembled. It has been installed in the Mockup Structure at the NDA Pawling Laboratory.

Sodium System

Fabrication of all major components has been completed, and assembly of the system is progressing rapidly (see Fig. 2-6.4).

Auxiliary Systems

Materials and equipment for the auxiliary gas, steam, and vacuum systems have been procured. Some components have been prefabricated; the remainder will be assembled after installation of the test section in the structure.

Installation

An arrangement drawing of the entire assembly is shown in Fig. 2-6.5.

TEST PROGRAM

A test program has been established which will demonstrate experimentally the ability of the mockup to contain sodium and water reliably under simulated reactor temperature conditions. The program consists of three categories: shakedown, normal operations, and aggravated operations. Shakedown will consist of a careful, detailed manipulation of the system to establish heating and cooling rates, temperature distributions, time responses, and operating procedures for emergency situations. Normal operations will consist of an extended run at reactor operation conditions. During aggravated operations the system will be subjected to thermal cycling of the sodium system, increased sodium temperatures and pressures, and rapid water dumping.

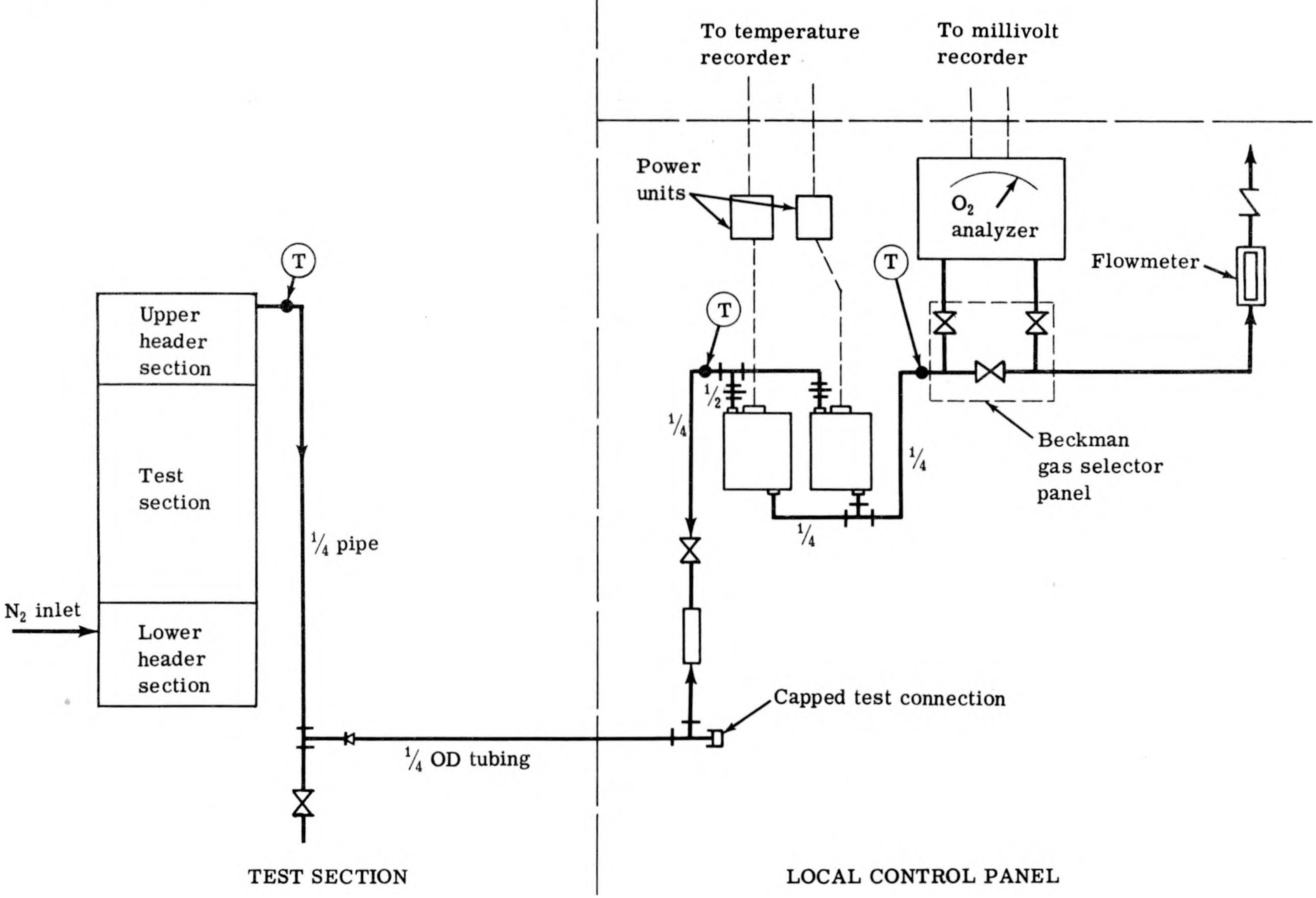
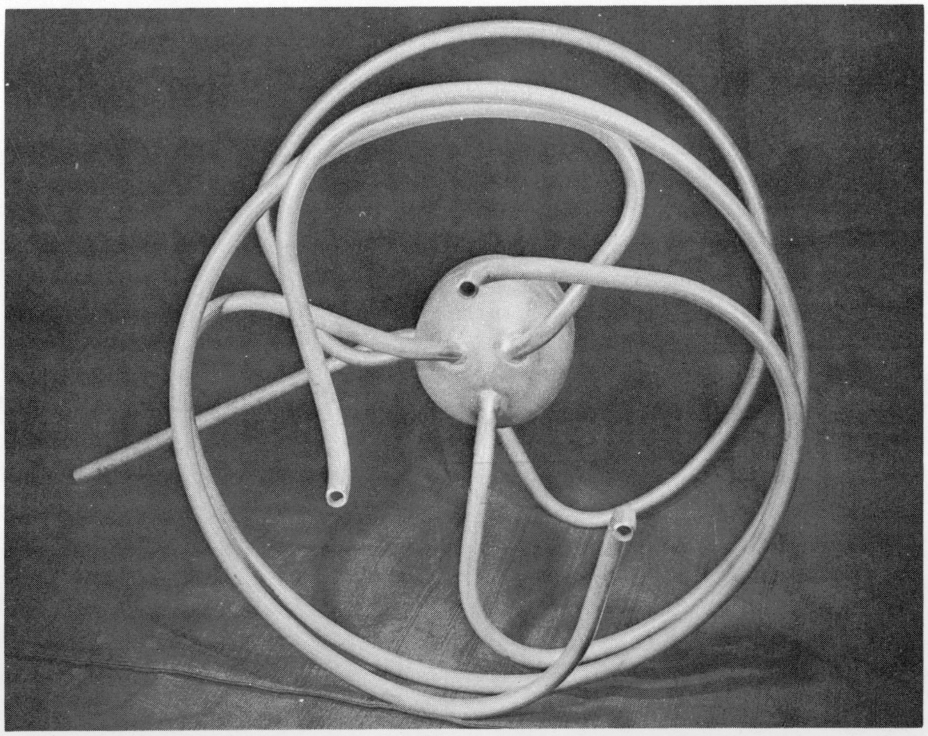
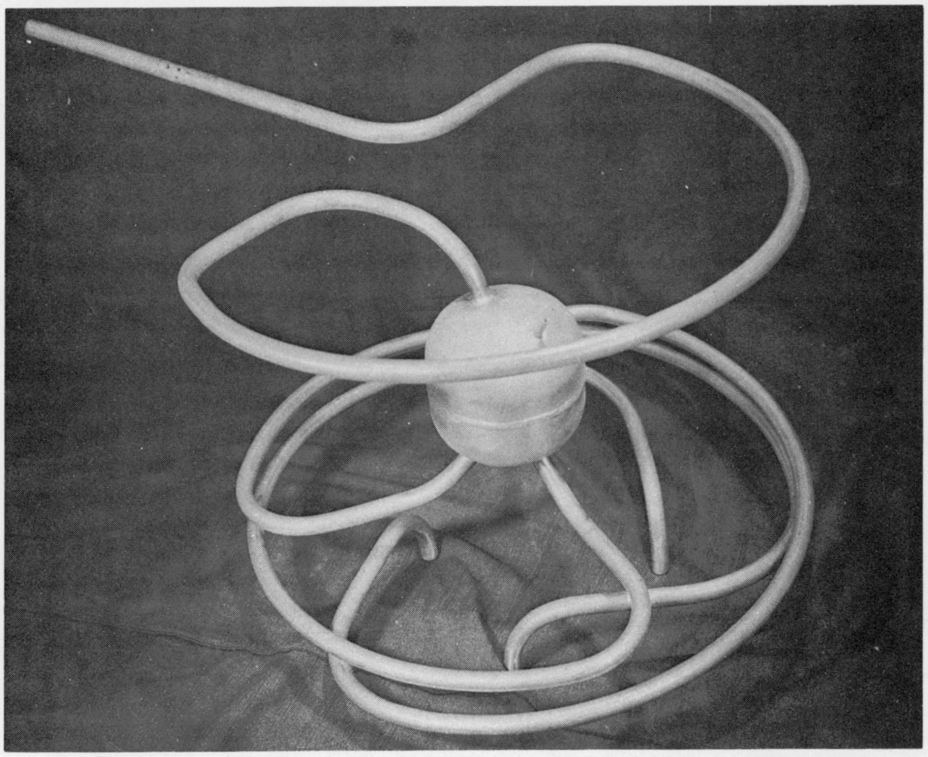


Fig. 2-6.1 — SDR mockup – test section – piping schematic. All dimensions in inches. (T) = thermocouple location.

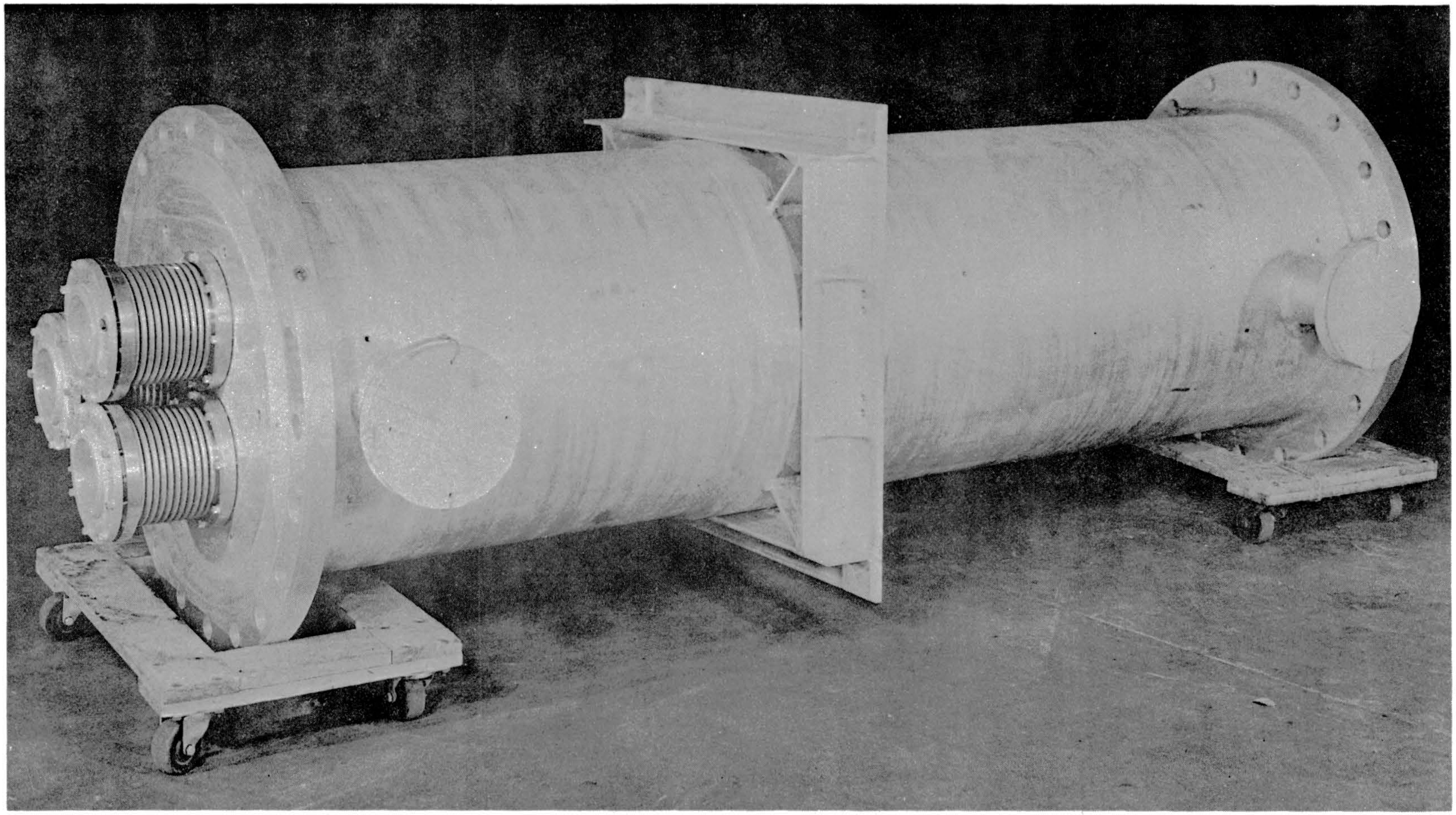


NEG. NO. 972



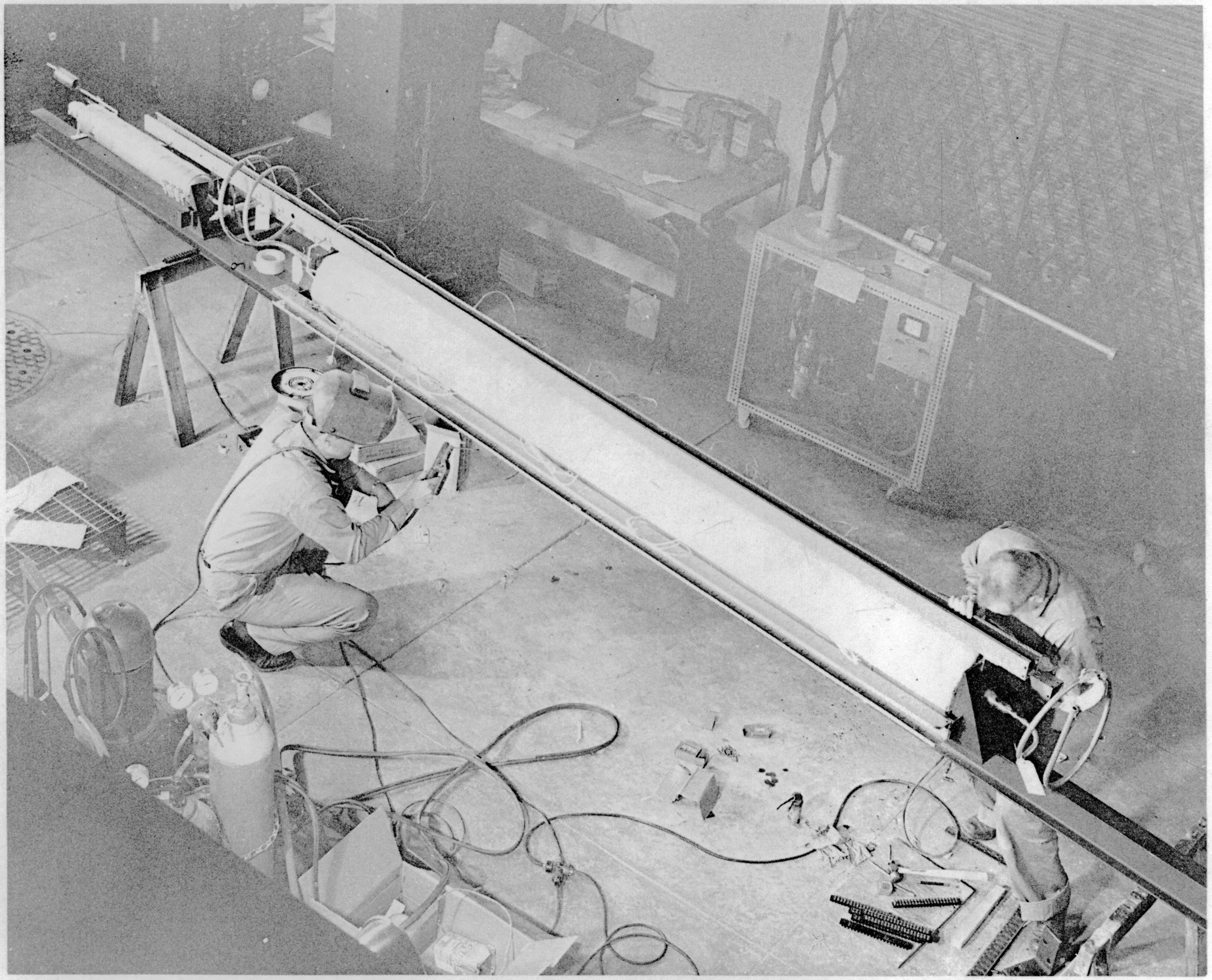
NEG. NO. 971

Fig. 2-6.2 — SDR mockup – assembled pigtails and manifold. Note thermocouple well in lower view. Manifold diameter ~6 inches.



NEG. NO. 949

Fig. 2-6.3 — SDR mockup – aluminum calandria, showing stainless steel expansion bellows



NEG. NO. 956

Fig. 2-6.4 — SDR mockup – assembling of sodium heater and heater support

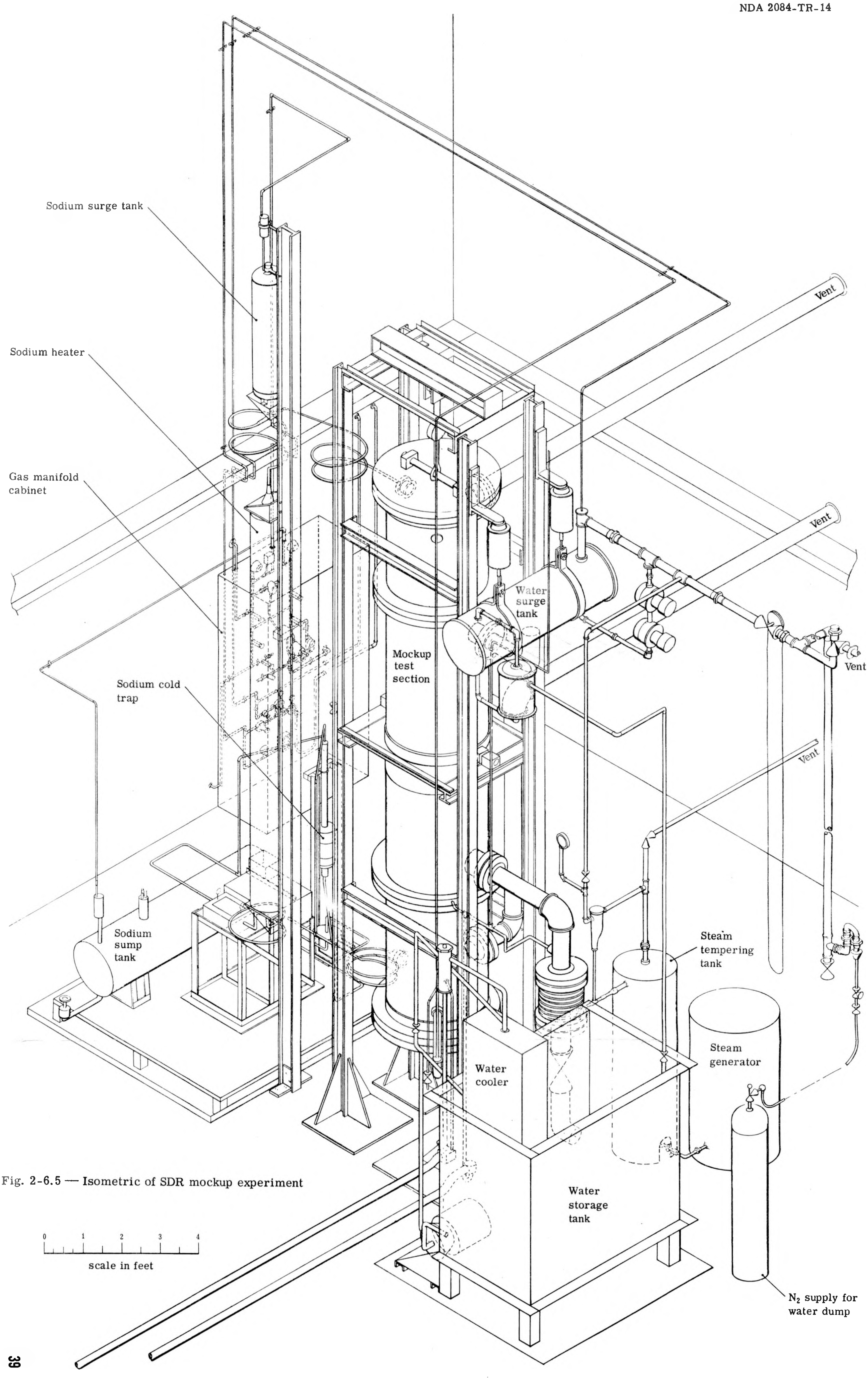


Fig. 2-6.5 — Isometric of SDR mockup experiment

PRELIMINARY DESIGN

TASK 3-1 REACTOR PRELIMINARY DESIGN

The preliminary design is basically the same as that described in the previous quarterly progress report. Design work on specific systems has been covered in previous sections of this report (Tasks 2-1, 2-2, and 2-3) while more general studies pertaining to the reactor proper are discussed below. These include reactor maintenance, fuel element studies, and reactor control and core physics.

In addition an alternate version of the straight-through reactor concept (the sliding tube design) was investigated briefly and the main features of this design are presented.

REACTOR MAINTENANCE

Questions of reactor maintenance were considered further during the quarter. Some alternate approaches are discussed below.

It is currently planned that extra fuel tubes will be provided in excess of the number required for operation at rated power. In the event of a failure in a fuel tube or pigtail, the reactor can be shut down and the fuel element removed from the faulty tube. The faulty tube may then be closed off, fuel element(s) inserted in one or more of the extra tubes, and normal operation resumed. If the failure is in the fuel tube itself, the tube may be plugged where it passes through the upper and lower neutron shields. Since these shields will probably be held at a temperature below 200F, the sealing of the plugs could be supplemented by freezing of the sodium at the plugs. The plugs can be installed without removing any shielding and without problems of access to activated areas.

If the failure is in the pigtail, the pigtail may be pinched off at both sides of the failure and a plug rolled into the opposite end of the fuel tube. The faulty tube may be replaced at the next scheduled routine maintenance shutdown.

In order to perform maintenance operations in the header rooms, such as pinching off pigtails, cutting and welding pigtails, etc., some header room accessibility will be necessary.

Calculations have shown that direct activation of header room piping can be reduced to a tolerable level by the neutron shields and that sodium activity decays to an acceptable level within about one week. A pessimistic interpretation of KAPL tests of mass transport activity* indicates that the radiation level may be too high for routine maintenance after several years of operation.

Should a more accurate determination show the activity to be somewhat higher than permissible for normal maintenance, local shielding will be required to protect the worker. If the activity proves to be considerably higher than permissible, alternate pigtail and header room designs must be de-

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

vised. One such design has been considered in which the horizontal legs of the upper pigtails were made long enough to penetrate the side shields of the upper header room. This scheme provides more space between pigtails and thus permits more efficient local shield for the welder. Some schemes are under consideration for providing local shielding in the lower header room.

SLIDING TUBE DESIGN STUDY

A brief study was made of the possibility of incorporating into the present calandria straight-through design some of the desirable features of the slab design described in NDA 084-3. The main purpose of this study was to investigate the feasibility of an SDR design without upper and lower header rooms, with short fuel tubes, and with simpler fuel tube and calandria repair procedures.

The study resulted in the conceptual reactor layout shown in Figs. 3-1.1 and 3-1.2. The calandria shown differs from the present design only in that a square rather than a triangular tube array is used. Each row of fuel tubes above the calandria is welded into a common manifold. Below the calandria, the same row of fuel tubes enters a common lower manifold through sliding joints. Since these sliding joints allow unrestrained axial thermal expansion of the fuel tubes, and since clearance between fuel, barrier and calandria tubes can accommodate horizontal expansion of the manifolds, the upper and lower pigtails are eliminated. The upper and lower sets of manifolds lead out from two sides of the reactor into shielded pipe access rooms and then to headers. A fuel tube and manifold assembly can be removed from the calandria by cutting the manifold in the pipe access room, removing the appropriate reinforced concrete beams that make up the top biological shield, and then lifting of the entire assembly out of the calandria through the open spaces in the shield. The piping and shielding are arranged to permit convenient and safe access to welds, and as a result there is no need for shielded upper and lower header rooms. The free surface of the sodium is maintained at a safe level in the lower manifolds well below the sliding joints so that these joints need serve only as a partial seal against sodium vapor. Details of a typical sliding joint are shown in Fig. 3-1.3.

The details of maintaining the level of the free surface during operation, filling, draining, and certain accidents have been studied and appear to be feasible.

The advantages of the sliding tube design over the present straight-through design are as follows:

1. Fuel tubes will be approximately 20 ft shorter.

2. Fuel tube replacement is considerably easier, due mainly to the sliding connections at the bottom and the elimination of pigtails. No welding is required on radioactive pigtails.

3. The number of field welds, aside from the welds for the refueling closures, is reduced from about 240 to about 48.

4. Calandria tube replacement will be easier because the elimination of the upper header room and its shield lowers the working level from which remote operations are required by about 7 ft.

5. The spiral neutron shield plugs in the fuel tubes are elminated.

6. Pressure drop through the sodium system is considerably less due to the elimination of the pigtail connections, shortening of the fuel tubes, and elimination of the spiral neutron shield plugs in the fuel tubes.

7. The lattice spacing can be appreciably smaller than the 10 in. required in the present calandria design, with resultant advantages in D_2O inventory.

8. Heat losses from the primary sodium system will be significantly less since fuel tubes need not pass through the concrete neutron shields, and the large heat transfer area of the many pigtails in upper and lower header rooms is eliminated.

Disadvantages of the sliding tube design compared with the present calandria design are as follows:

1. It is not a completely sealed sodium system. Therefore, it depends on a partial mechanical seal and a liquid level overflow provision to prevent sodium from escaping through the fuel tube slip joint in the event of some mishap.

2. A more complicated external gas system is required to control the free surface during filling and draining.

3. Thermal expansion in the horizontal headers may require more careful design of tolerances with respect to the "cold" location of the fuel tubes in relation to the calandria tubes.

4. Several fuel tubes must be aligned simultaneously during replacement, compared with only one in the present calandria design.

5. A major leak in the upper portion of the sodium piping may cause the sodium to drain from the fuel tubes and expose fuel elements. This will be different from the present calandria design only after the primary pumps are off and may not be serious if adequate gas cooling can be made available in time to prevent a fuel element meltdown.

6. Downflow of the sodium is against thermal convection. This may be important during accidents in which pumping pressure is lost.

FUEL ELEMENT STUDY

Fuel Element Requirements

SDR requires a fuel element which operates up to 5000 MW-d/ton maximum burnup, and produces a mixed sodium outlet temperature of 950F. The cladding temperature range is 955-1050F, and the heat flux may vary from 150,000 to 300,000 Btu/hr-ft^2. Enrichments of less than about 2.0% are being considered at present.

Available Fuel Materials

A survey has been made of available high fuel density materials. The fuel materials were grouped in the following categories: unalloyed uranium, uranium alloyed to refine grain size, uranium alloyed to retain γ structure at room temperature, and UO_2. Uranium compounds other than UO_2 were not considered because their technology is not developed sufficiently for application to the SDR. Findings of the survey are summarized below.

Unalloyed Uranium

Burnups and temperatures lower than that required cause dimensional changes induced by radiation which are not tolerable in SDR.

Uranium Alloyed to Refine Grain Size

The dimensional instability of uranium can be limited by small amounts of alloying additions which produce a fine-grained, randomly oriented crystal structure, and a stronger alloy. At their present stage of development, none of the alloys can be recommended for application in SDR.

Uranium Alloyed to Stabilize the γ Phase at Room Temperature

The fully retained γ phase is the most stable to deformation induced by thermal cycling and radiation. Of the materials in this category developed to date only the U – 10 weight % Mo alloy shows requisite stability for SDR operating conditions. Ternary alloys being developed show promise, but insufficient data are available at present.

U – 10 weight % Mo has a maximum 4% volume change per atom percent burnup (equivalent to about 8000 MW-d/ton at 2% enrichment) at a central fuel temperature of about 1200F. (The data are based on 0.100 in. to 0.150 in. diameter pins irradiated for the APDA program.* The curve of percent volume change per percent atom burnup vs central fuel temperature starts to rise steeply between 1100F and 1300F. A 1200F value appears to be a feasible one based on present data. An irradiation program on small diameter pins presently sponsored by APDA is attempting to define the central temperature more closely.)

UO_2

UO_2 has more than requisite dimensional stability for SDR burnups and temperatures. Bettis Field has achieved successful burnups to 20,000 MW-d/ton and predicts the feasibility of 50,000 MW-d/ton at PWR temperatures.† Tests have shown that even central melting does not necessarily mean failure of the fuel element.

The considerable advantage of dimensional stability is balanced by the following disadvantages:

1. $\frac{1}{3}$ lower fuel density,
2. low thermal conductivity of the fuel and fuel-to-clad "bond" limits heat generation rates and raises fuel temperatures,
3. fission gas release rates from UO_2 may require thicker cladding thickness than required by metallic fuel.

Tentative Conclusion

The tentative conclusion reached is that a U–10 weight % Mo alloy is satisfactory for a slightly enriched SDR and that UO_2 is satisfactory for both natural uranium and a slightly enriched SDR. The U – 10 weight % Mo alloy appears to be limited to a central temperature of about 1200F; additional development work may show that the limit is higher. The UO_2 central temperature should be limited to give reasonable fission gas release rates to make low cladding thickness possible, probably from 3000 to 4000F.

CONTROL ROD CALCULATIONS

A series of two-group reactor criticality calculations were run to determine the adequacy of a 6/1 fuel channel to control rod arrangement. The primary requirement for the control system was that it must be capable of reducing the k_{eff} of the clean reactor to 0.97 or less for emergency shutdown. The study was concluded when it was ascertained that the 6/1 ratio met the above requirement, with a fuel enrichment for this reactor which gave sufficient excess reactivity for a maximum burnup of 5,000 MW-d/ton in some portion of the fuel.

---

* Second High Temperature Fuel Element Meeting, BMI, Dec. 12-13, 1957.
Effect of Heat Treatment and Burnup on the Radiation Stability of Uranium – 10 percent Molybdenum Fuel Alloys, G. D. Calkins et al., 6th Annual ASTM Meeting, Atlantic City, N.J., June 16-21, 1956.

† J. D. Eichenberg et al., Effects of Irradiation on Bulk UO_2, WAPD-183, Oct. 1957.

The control rods occupy lattice positions and are assumed to be uniformly distributed throughout the core; thus the fuel channel to control rod ratio is constant in all regions of the core. The 6/1 ratio gives a simple lattice array in which six fuel clusters situated at the apexes of a hexagon are controlled by one control rod located at the center of the hexagon.

The 6/1 fuel channel to control rod ratio gave adequate control at 1.75% enrichment (k_{eff} clean with control rods fully inserted = 0.893), and the excess reactivity at this enrichment (after taking Xe and Sm override into account) was sufficient to give a maximum burnup of 5,120 MW-d/ton. Thus, a 6/1 ratio at 1.75% enrichment meets both requirements discussed above. Adequate control with the 6/1 ratio was also available at 2% enrichment (k_{eff} clean with control rods fully inserted = 0.946). However, for 2% enrichment, the excess reactivity was sufficient to give about twice the required maximum burnup (9,750 MW-d/ton). Table 3-1.1 summarizes the results at 1.75 and 2.00%.

Table 3-1.1 — Control Calculation Results for Enrichments of 2.00 and 1.75%

Enrichment:	2.00%	1.75%
Maximum point burnup MW-d/ton	9,750	5,120
k_∞ (control rods removed)	1.250	1.205
k_∞ (control rods completely inserted)	1.041	1.004
k_{eff} (control rods removed)	1.115	1.072
k_{eff} (control rods completely inserted)	0.946	0.893

FUEL AND SODIUM TEMPERATURE COEFFICIENTS OF REACTIVITY

The reactivity effects of changes in fuel and sodium temperature were estimated. These effects are important because the temperatures of these materials change quickly with changes in power and therefore determine the magnitude and sign of the prompt temperature coefficient.

An increase in sodium temperature decreases sodium density, thereby reducing parasitic absorption by expelling sodium from the reactor. The fractional increase in reactivity is proportional to the fractional increase in the thermal utilization of the uranium fuel. The temperature coefficient of reactivity resulting from sodium expansion was calculated to be $+0.87 \times 10^{-5}/°C$.

An increase in uranium temperature decreases reactivity because Doppler broadening of the U^{238} resonances decreases the resonance escape probability. The fractional decrease in reactivity is proportional to the fractional decrease in resonance escape probability. The fuel temperature coefficient of reactivity was calculated as $-1.67 \times 10^{-5}/°C$.

Thus for a uniform temperature increase in the fuel and coolant, the net reactivity effect will be negative, with a value of $-0.8 \times 10^{-5}/°C$. Since, in a power excursion, the fuel temperature changes more rapidly than the sodium temperature, the coefficient will be more strongly negative. This negative coefficient will tend to limit power excursions.

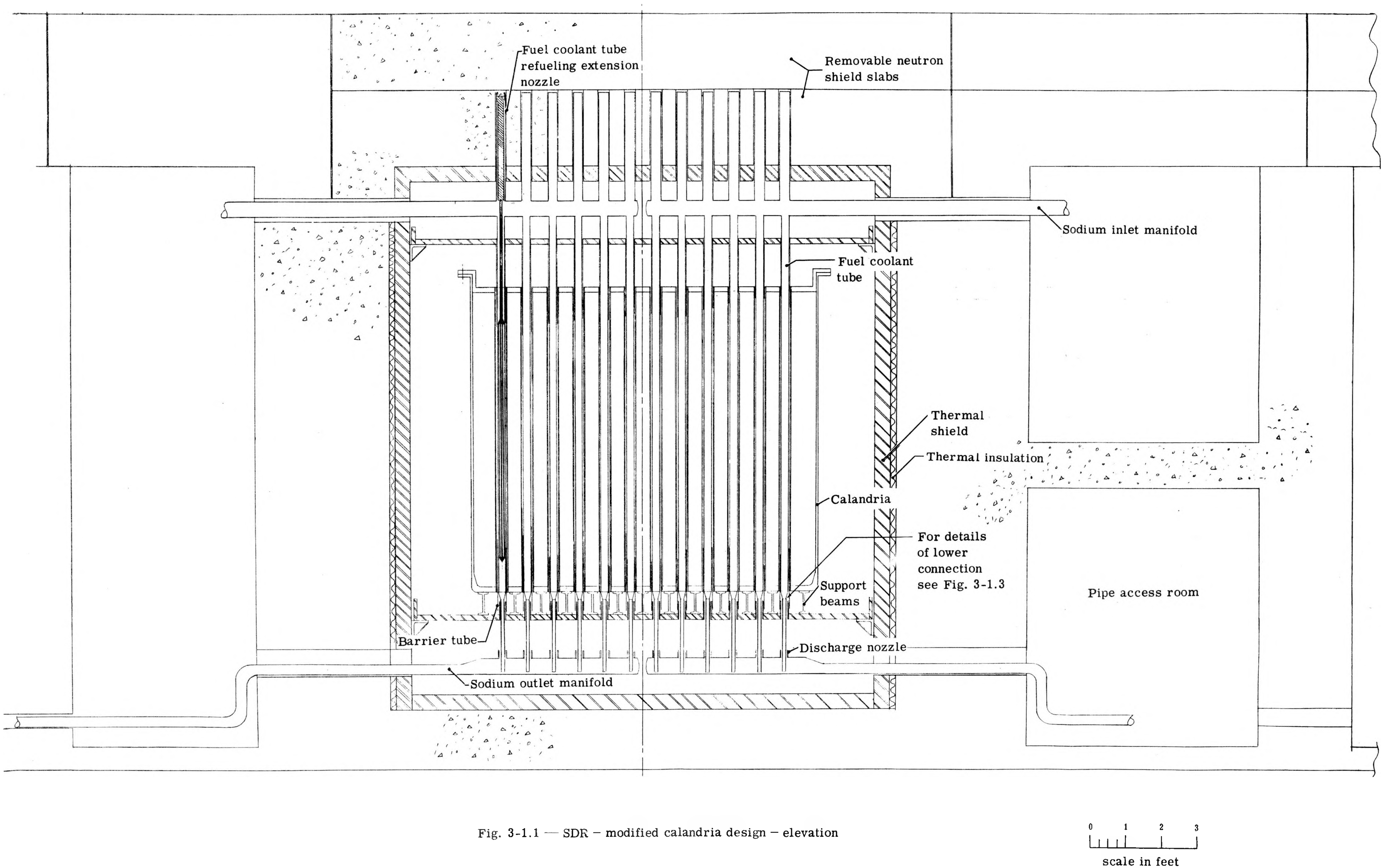


Fig. 3-1.1 — SDR – modified calandria design – elevation

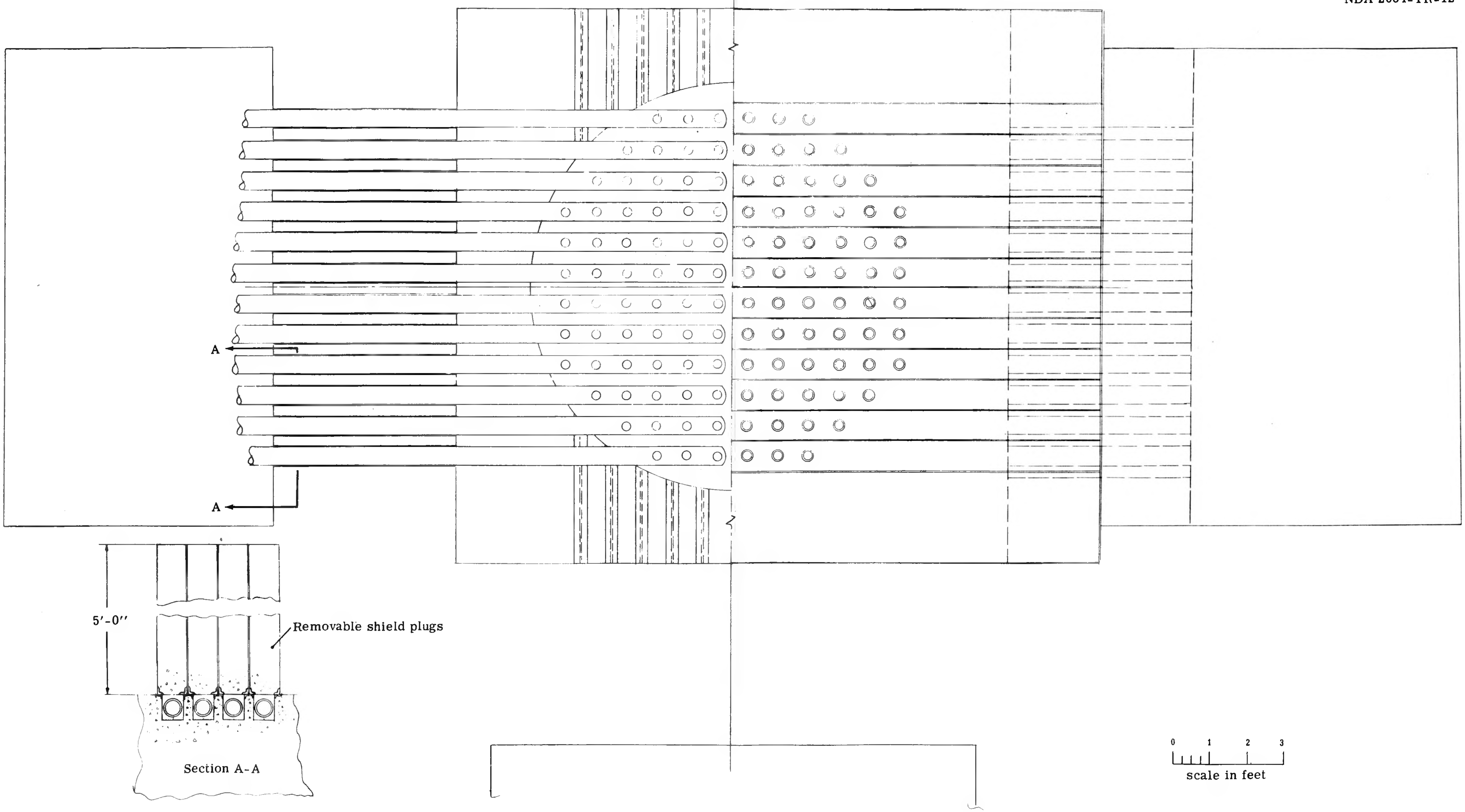


Fig. 3-1.2 — SDR – modified calandria design – plan

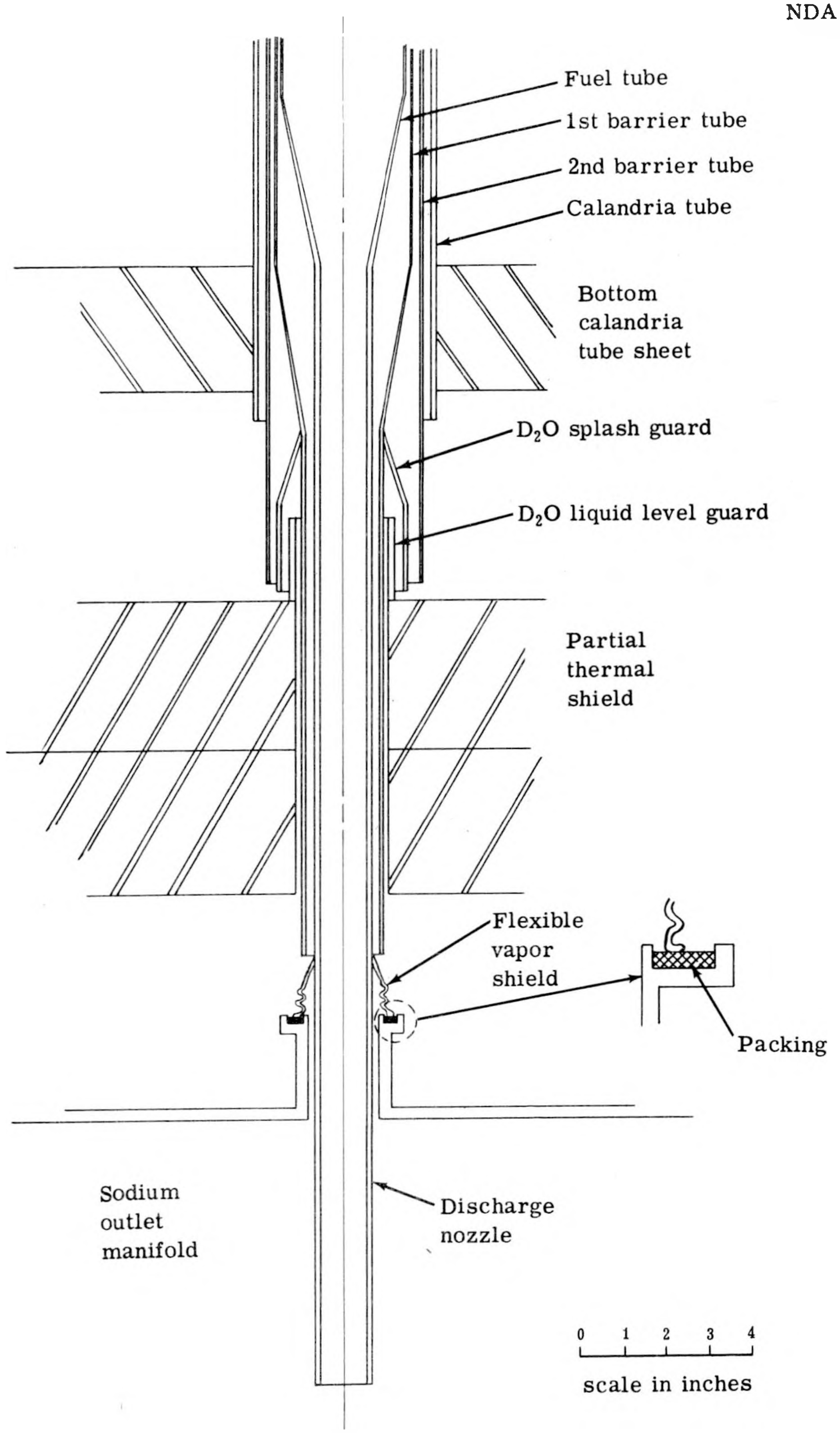


Fig. 3-1.3 — SDR – modified calandria design – detail of lower tube connection

TASK 3-2 SHIELDING

Shielding work as reported in previous SDR quarterly reports concentrated on the problem of accessibility to the header rooms after shutdown. In this quarter attention was directed toward determining the design requirements for the thermal shield and the bulk shield between the reactor and the header rooms.

THERMAL SHIELD DESIGN

The purpose of the thermal shield is to reduce the gamma and neutron fluxes on the concrete biological shield sufficiently so that heat generation in the concrete will not be excessive. Detailed calculations were made to determine the thickness of iron required to give a maximum heat flux on the inside surface of the concrete of 100 Btu/hr-ft^2. It was found that $6\frac{1}{2}$ in. of iron is required to provide sufficient attenuation for direct fission gammas, gammas from fission product decay during operation, capture gammas from core materials, capture gammas from the thermal shield itself, and fast and thermal neutrons leaking from the core.

A plot was obtained of the total gamma heat generation rate in the thermal shield vs distance into the shield. At an approximate average volumetric heat generation rate of about 8000 Btu/hr-ft^3, (0.08 watts/cm^3), the total gamma heat generated in the entire thermal shield was calculated to be about 3×10^6 Btu/hr (0.9 MW). The neutron heating was found to be negligible (~10^3 Btu/hr) compared to the gamma heating.

BULK SHIELD DESIGN

The present design shows a 5-ft thick barytes concrete shield between the reactor and the upper and lower header rooms. This is the approximate shielding thickness required to attenuate gammas from fuel element fission product decay to tolerance levels in the header rooms one week after reactor shutdown. In addition to protecting personnel entering the header rooms from after-shutdown core gammas, this shielding also protects the header room piping from excessive activation by neutrons during reactor operation.

The possibility of drastically reducing the thickness of the bulk shield was investigated. To go to a 2-ft thick shield would require that the fuel elements would have to be removed from the reactor before personnel could enter the header room and that the two annular gaps in the bulk shield plug would have to be reduced to a thickness of about 0.04 in. each. It was concluded that the problems involved in the 2-ft thick shield and the advantage of not requiring fuel element removal, were sufficient to justify continuation of the present 5-ft thick shield design.

TASK 3-3 EXTERNAL SYSTEMS

Preliminary design flow sheets of several of the auxiliary gas systems (D_2O cover gas, sodium cover gas, and barrier gas systems) were prepared. The survey of major components of the external sodium system was continued.

D_2O COVER GAS SYSTEM

During the quarter a preliminary D_2O cover gas system flow sheet was studied. Helium has been selected as the cover gas because of its low neutron capture cross section, stability to heat and radiation, and chemical inertness. Helium at a few inches water gage pressure will be supplied to all D_2O components where a free liquid surface exists, i.e., the calandria, the dump storage, and the holdup tanks. Gas connections will be provided to the D_2O ion-exchange resin beds and filters. To facilitate moderator dumping, a large diameter helium return vent will be placed between the calandria and the dump tank.

The cover gas will also serve the function of sweeping away gases resulting from D_2O decomposition from the space above the moderator in the calandria, and carrying them to a catalytic recombiner where D_2O will be reformed and returned to the system.

SODIUM COVER GAS SYSTEM

A flow sheet of the sodium cover gas system has been developed. Helium has been selected as the cover gas for the sodium system, for the reasons discussed above. Gas lines will be provided to both primary and secondary sodium components where a sodium – gas interface may exist. Such components include: sodium storage and expansion tanks, centrifugal pumps, cold traps, fuel storage pits, and fuel handling equipment. The primary and secondary sodium cover gas systems are separated to prevent potentially radioactive gas in the primary circuit from contaminating "clean" gas in the unshielded secondary circuit. Equipment has been included for maintaining gas purity, both radiological and chemical.

BARRIER GAS SYSTEM

The barrier gas will fill the calandria room, and will probably also fill the header rooms. The barrier gas may also serve an auxiliary function of heating frozen sodium headers, pigtails, and fuel tubes during the startup, and it may be used to cool the calandria tubes should a moderator dump be necessary. Based on these and other requirements, nitrogen has been chosen as the barrier gas, although CO_2 is considered as an alternate. Helium was ruled out, primarily because of its high thermal conductivity since good thermal insulation in the barrier is necessary during normal reactor operation.

The pressure of the barrier gas will be lower than that of the D_2O and sodium cover gases, in order to insure purity of these other gas systems. A leak will result in contamination of only the barrier gas, the purity of which is not as critical as that of the other two.

Calculations have shown that similar gas flow rates are required for the emergency cooling and heating functions. This suggests that a single set of barrier gas piping and related equipment could be used for both operations.

TASK 3-4 OVERALL PLANT

REACTOR BUILDING LAYOUT

A preliminary study of the building layout has been started. The objective of this study is to select and arrange the major items of equipment which will be housed in the reactor building, so as to provide safety, low cost, and operating convenience.

For safety, the sodium and D_2O systems outside of the reactor will be separated by physical barriers such as shielding walls, in addition to the separating barriers provided within the reactor itself. All systems which contain or may contain substantial amounts of radioactive fluids will be kept entirely within the reactor building. The spent fuel storage facility may be located outside of the reactor building to further conserve space and to separate the fuel handling function from the reactor operation.

The included height from the floor of the lower header room to the top of the upper shield plug will be about 40 ft. In addition, a lower level must be provided for a sodium sump tank, a header room spill return pump, a sodium system drain pump, etc. Hence the building internal height will be about 50 ft from the lowest point up to the top of the reactor shield. Crane rails will need to be about 35 to 40 ft above the reactor shield. A total building height of somewhat over 90 ft may be needed, and a diameter of the order of 80 to 100 ft.

APPENDIX

INSTALLATIONS VISITED

Installation	Topics Discussed
Argonne National Laboratory	Sodium technology; sodium-water reactions; design and operation of sodium systems and components, including intermediate heat exchangers, steam generators, and sodium pumps; high density, high burnup fuel elements; fuel element fabrication
Atomic Power Development Associates, Inc.	Liquid metal loop design; steam generator operation; liquid sodium-water systems and operating experience
Battelle Memorial Institute	High density, high burnup fuel elements (primarily the effect of radiation on dimensional stability)
Knolls Atomic Power Laboratory	Liquid metal system operation and safety procedures; sodium-water reactions; reactor containment calculations
Mine Safety Appliances Research Corporation	Liquid metal technology; sodium-water and sodium-steam reactions; liquid metal-steam generator design
Oak Ridge National Laboratory	Sodium-water and sodium-oxygen reactions; sodium alloy flammability studies; liquid metal instrumentation; sodium leak detection system; liquid metal system operation and fabrication
Sylvania-Corning Nuclear Corporation	Fuel element fabrication, design, and performance

DISTRIBUTION

	No. of Copies
Atomic Energy Commission	
New York Operations Office, Patent Group	1
Schenectady Operations Office	4
Technical Information Service Extension, Oak Ridge	20
Washington, D. C.	2