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AIRCRAFT NUCLEAR PROPULSION PROJECT
QUARTERLY PROGRESS REPORT
for Period Ending December 10, 1950

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SUMMARY

The Aircraft Nuclear Propulsion Project at the Oak Ridge National Laboratory has expanded considerably during the past quarter. There are now 236 technical people engaged in all phases of the research work. Thirteen divisions of the Laboratory are represented on this Project.

Design of the Aircraft Reactor Experiment (ARE) is continuing, with particular emphasis on shielding, control, and fuel material. The NEPA Division, Fairchild Engine and Aircraft Corporation, has loaned 14 men to assist in the design and construction of the ARE.

Three major facilities have been completed: the Shielding Reactor, the ANP Critical Facility, and the 86-in. cyclotron, which will be used part-time for radiation damage studies. Also, work has begun on increasing the power level of the MTR mock-up so that it may serve as a Low Intensity Training Reactor (LITR) and may also be used part-time for radiation damage studies.

ANP CONTROL BOARD

A parallel effort to the ANP Shielding Board was the work on the joint ORNL-NEPA Control Board.(1) This group studied the control properties of the solid-fuel aircraft reactor design developed by Oak Ridge National Laboratory which has been described in previous reports.(2) The Control Board found that although this solid-fuel reactor would have a negative short-time coefficient of reactivity owing to temperature changes of the fuel elements alone, it would have a positive temperature coefficient of reactivity over time intervals of 10 sec or longer owing to temperature changes in the moderator. The Control Board also believed that this reactor would require 10 or more absorber control rods. The mechanical actuation system and the heat-removal system would be somewhat complex. Subsequent reactivity calculations, as described in the Reactor Physics section of this report, have shown that the positive temperature coefficient difficulties can be greatly diminished by a decrease in the fraction of moderator in the core so as to raise the mean energy for fission. However, it is clear that the control of a solid-fuel high-temperature reactor will not be simple.

(1) *Interim Report of the ANP Control Board,* NEPA-ORNL, ANP-54, (Nov. 1, 1950).

(2) *Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending August 31, 1950,* ORNL-858 (Dec. 4, 1950); *Preliminary Feasibility Report for the Aircraft Reactor Experiment,* Oak Ridge National Laboratory, Y-12 Site, Y-FS-15, (July 4, 1950).

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ARE DESIGN

Largely as a result of the above findings of the ANP Control Board, a decision was made to suspend work temporarily on the solid-fuel ARE design, while attempting to develop a suitable liquid-fuel arrangement. It is to be expected that a "stationary" liquid-fuel system can be designed which will be almost self-regulating, and which will require only one or two control rods. The self-regulating feature will arise from the strong negative temperature coefficient of reactivity since increases in reactor temperature will expand the fuel-bearing liquid and force some of the uranium outside the active core, thus leading to a compensating decrease in power. The most promising fuel mixture to date is a solution of UF_4 in NaF, with possible admixtures of other fluorides to lower the melting point to a convenient range. Design is now underway on such a liquid-fuel aircraft reactor and ARE prototype and, as described in this report, research is also in progress on the properties of other possible fuel liquids.

STUDIES ON ALTERNATIVE LINES

In addition to design and research toward a "stationary" liquid-fuel liquid-metal-cooled aircraft reactor and ARE, a certain amount of work is being continued by the Laboratory on various alternative lines which show promise for future aircraft reactors. These studies are described in Part II of this report. They include:

1. Calculations on vapor-cycle systems by North American Aviation, Inc., together with research on the corrosive effects of Na vapor.
2. A complete design study including reactor, engine, and aircraft analysis for a nuclear XB-52 plane powered by a circulating-fuel reactor based on a suspension of uranium-bearing particles in NaOH, carried out by the H. K. Ferguson Company.
3. A detailed analysis of the supercritical water-cooled reactor by Nuclear Development Associates, Inc.

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SUMMARY OF RESEARCH RESULTS

From the body of research now underway on all the problems mentioned above, some of the other tangible results obtained during the last quarter are listed below:

Physics

1. The Reactor Physics Group has performed extensive computations of criticality and thermal xenon coefficients in bare reactors, and similar calculations for reflected reactors are in progress. Graphs of the spectra of several solid-fuel reactors are included. The determination of the kinetic response of some liquid-fuel stationary-moderator reactor designs has been begun.

3. A preliminary analysis of the fuel kinetics of the NaF-UF_4 reactor shows the existence of oscillation in the reactor power following a change in reactivity, which arises from the coupling between fuel displacement and neutron flux. However, this oscillation damps out in a few seconds, and in any event the preliminary result was obtained without taking account of delayed neutrons. In a more exact calculation which is to follow, any power oscillations are expected to be negligible.

4. A resonance at approximately 49 ev was found from the preliminary neutron cross-section measurements on molybdenum at Columbia.

5. Nuclear Development Associates, Inc. have computed, as a function of temperature, both the xenon cross section averaged over a Maxwellian distribution and the temperature derivative of this quantity.

6. Preliminary studies show that it should be possible to prepare a 1000-curie source of Xe^{135} if a measurement of the intermediate xenon cross section proves advisable. The most efficient source for preparing such a quantity of Xe^{135} would probably be the Homogeneous Reactor Experiment.

Heat Transfer

12. Theoretical analyses of heat transfer have been completed on three situations which approximate the entrance conditions involved in current ARE core designs. Heat transfer within a wide range of duct shapes has been analyzed.

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13. Equipment has been designed for the measurement of thermal conductivity, specific heat, and other high-temperature properties of various liquid metals and molten salts. The specific heat of lithium between 550 and 900°C has been measured as 1.0 ± 10%.

14. Both a type 316 and type 347 stainless steel convection harp containing liquid sodium have now been operated almost 800 hr at 1500°F without failure.

Metallurgy

17. Substances so far found to have good resistance to Na at 1800°F include types 316 and 347 stainless steel and Ni. Fair resistance under these conditions is shown by Mo, Ta, Alloy N-155, Inconel, and Inconel X.

Radiation Damage

21. Radiation damage experiments on various metals exposed to a neutron flux of 1.2×10^{12} at 500°F and 1000 psi pressure for as much as four months have shown no significant changes in shape, electrical resistivity, permeability, or hardness of the metals.

22. A creep test is now underway on type 316 stainless steel in the Oak Ridge reactor at 1050°F under a stress of 4000 psi. Preliminary results show at least a transient increase in creep rate under these conditions, although possible spurious effects caused by gamma and neutron heating of the specimen have not yet been excluded.

Liquid Fuels

23. Attempts have been made to find a liquid fuel consisting of a uranium compound dissolved or suspended in NaOH. All uranium materials tested to date have been found to react rapidly with NaOH to form a finely divided compound which is probably sodium uranate. The possibility of developing stable suspensions in this system is being studied.

24. The equilibrium diagram of the molten salt system $LiF\text{-}UF_4$ has been established, and study of such systems as $NaF\text{-}KF\text{-}UF_4$ is underway.

25. Preliminary corrosion studies of numerous metals in $NaF\text{-}UF_4$ at 1300°F for 160 hr show hastelloy C, Inconel, and Mo to be the least attacked.

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Part I

RESEARCH CONTRIBUTING TO THE ARE

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1. DESIGN OF THE AIRCRAFT REACTOR EXPERIMENT

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1. DESIGN OF THE AIRCRAFT REACTOR EXPERIMENT

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A. P. Fraas J. H. Wyld
ANP Division

The reactor designs now being studied for possible use in the Aircraft Reactor Experiment all involve some arrangement of a quiescent liquid fuel and the use of liquid sodium as both the primary and secondary coolant. Five such core designs are presented. The coolant circuit specifications which have been adopted are discussed.

Core Design. Several core designs utilizing a quiescent liquid fuel are being considered for possible use in the Aircraft Reactor Experiment. The liquid which is believed most likely to serve for this purpose is a solution of UF_4 in NaF, with the probable addition of one or more other fluorides to reduce the melting point of the solution. The simplest method of containing the liquid fuel appears to be in tubes. The fuel-container wall thickness should be kept down to the order of 0.010 in. so as to minimize the uranium inventory necessary to overcome neutron absorption in the container metal. The combined effects of changes in thermal gradients and pressure gradients under transient conditions would probably be serious for any container shape other than a round tube.

A number of different types of construction have been proposed. The five designs shown in Figs. 1.1 to 1.8 are representative of the more promising of these proposals. The major features of these may be sketched as follows:

1. Helical coil arrangement, Figs. 1.1 and 1.2. This design contains fuel inside tubing which is wound in helical form between cylindrical shells of moderator. The arrangement permits the use of long fuel containers, thereby minimizing the number of welds required.

2. Hairpin ("U") tube fuel-element design, Figs. 1.3 and 1.4. This configuration features about 12,000 closely spaced U-tube fuel elements, with inlet and outlet headers both located at the top of the reactor for convenience of access and to simplify thermal expansion problems. Any desired degree of compartmentalization may be achieved by dividing the headers.

3. Parallel-tube fuel-element design, Fig. 1.5. This arrangement is essentially the same as the U-tube type except that the tubes

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are interconnected by headers at both top and bottom to facilitate filling and draining.

4. Annular fuel-tube arrangement, Fig. 1.6. This design consists of moderator cylinders surrounded successively by coolant annuli, fuel annuli, and coolant annuli. Approximately three hundred such concentric elements are housed in a moderator matrix.

5. Individual fuel-pin arrangement, Figs. 1.7 and 1.8. In this case the fuel is contained in individual noninterconnected tubes with void volumes provided for fuel expansion and fission product gas accumulation. The design has the disadvantage that the fuel could not be drained from the core and replaced by means of a purely liquid-flow operation.

The first three of these are similar in that they all involve about 30,000 ft of 0.100-in.-O.D., 0.080-in.-I.D. tubing. The fourth type, having the fuel in the annulus between concentric tubes, requires less total footage of tubing but presents a more complex set of fabrication problems.

In comparing these various designs, the fuel element itself is, of course, the item requiring the most study. The nature of the heat transfer through the liquid fuel to the tube wall presents a major question. As many pieces of data as possible have been assembled on the physical properties of substances which might be involved in this problem, and these are shown in Tables 1.1 and 1.2 and Fig. 1.9. When necessary, thermal conductivities have been calculated by Bridgmann's formula for nonmetallic liquids [*Proc. Natl. Acad. Sci.* **9**, 341 (1923)]; specific heats have been calculated by Kopp's rule [Lange, N. A., *Handbook of Chemistry*, 7th ed, Handbook Publishers, Sandusky, Ohio (1949)].

The temperature drop in the liquid-fuel tube from the center to the wall is very high if computed on the basis that thermal conduction is the sole heat transfer mechanism. However, the effects of thermal convection and of any change in the properties of the liquid due to radiation would be expected to reduce this temperature differential. In any case, the coolant tubes will have to be rather small to minimize trouble from this source.

It is expected that the core diameter will fall between 30 and 36 in. and the volume percentage of moderator between 50 and 60%. The core shape will be a right circular cylinder with the corners rounded off to give ellipsoidal ends.

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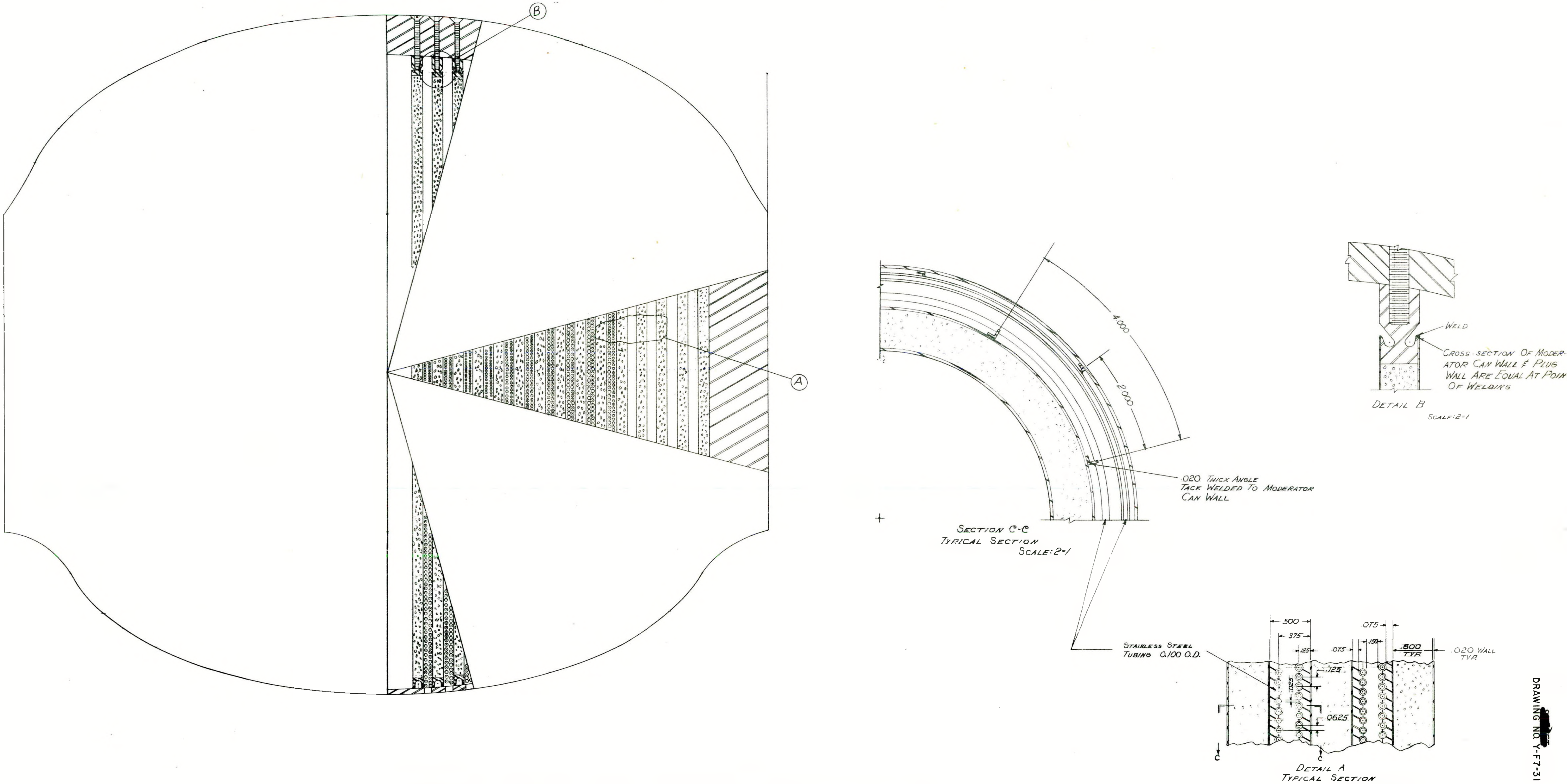

FIGURE 1.1 HELICAL COIL ARRANGEMENT (Reactor Core)

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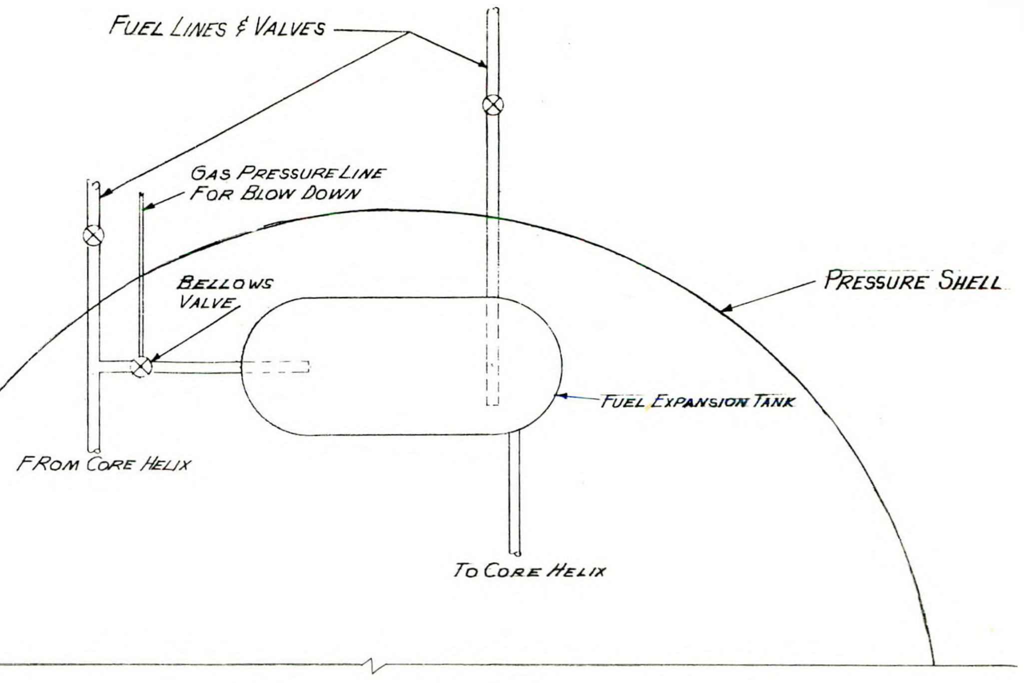
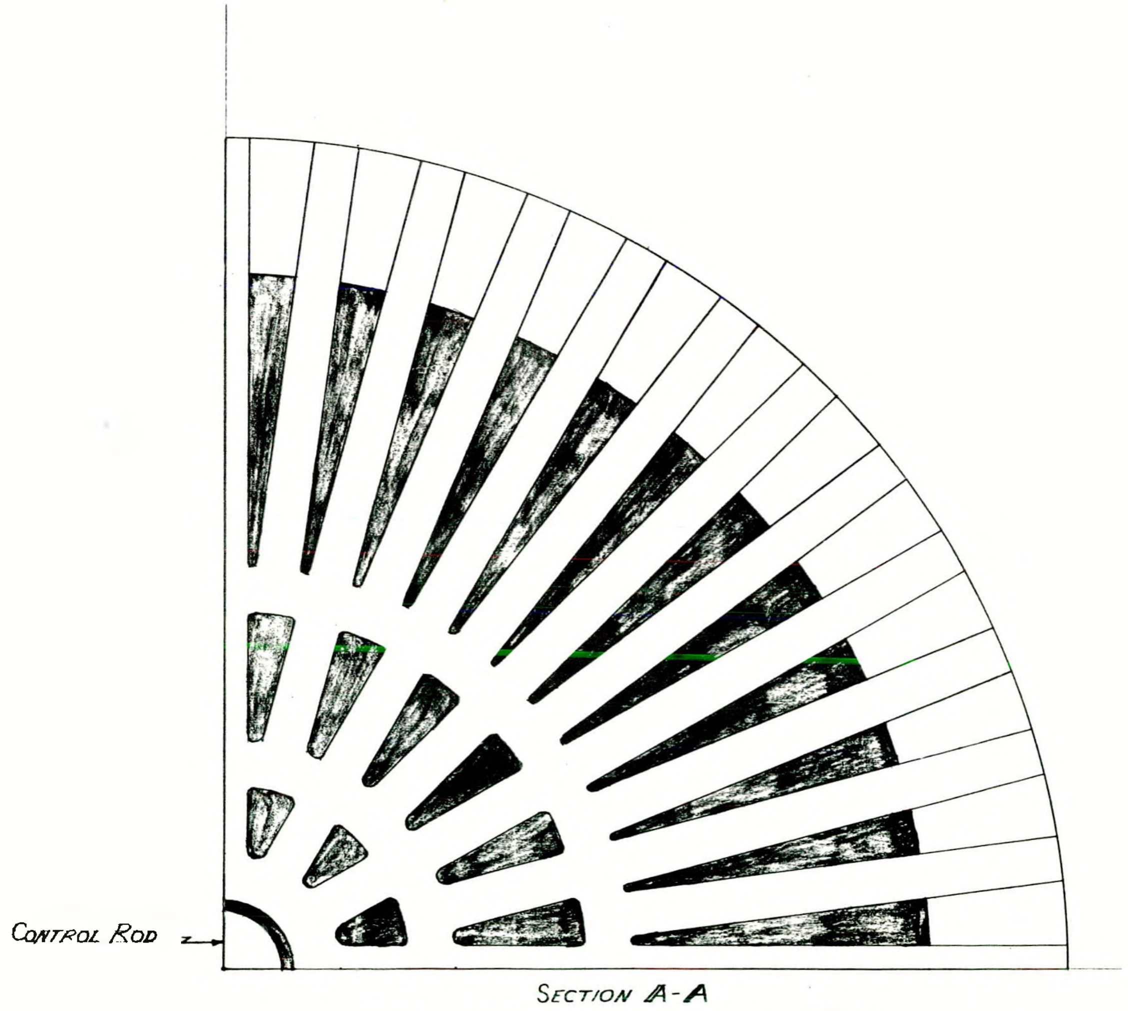


Typical Fuel Flow Diagram



Section A-A

Plan View Of Rib Type Dome

Na Flow Passes In Red

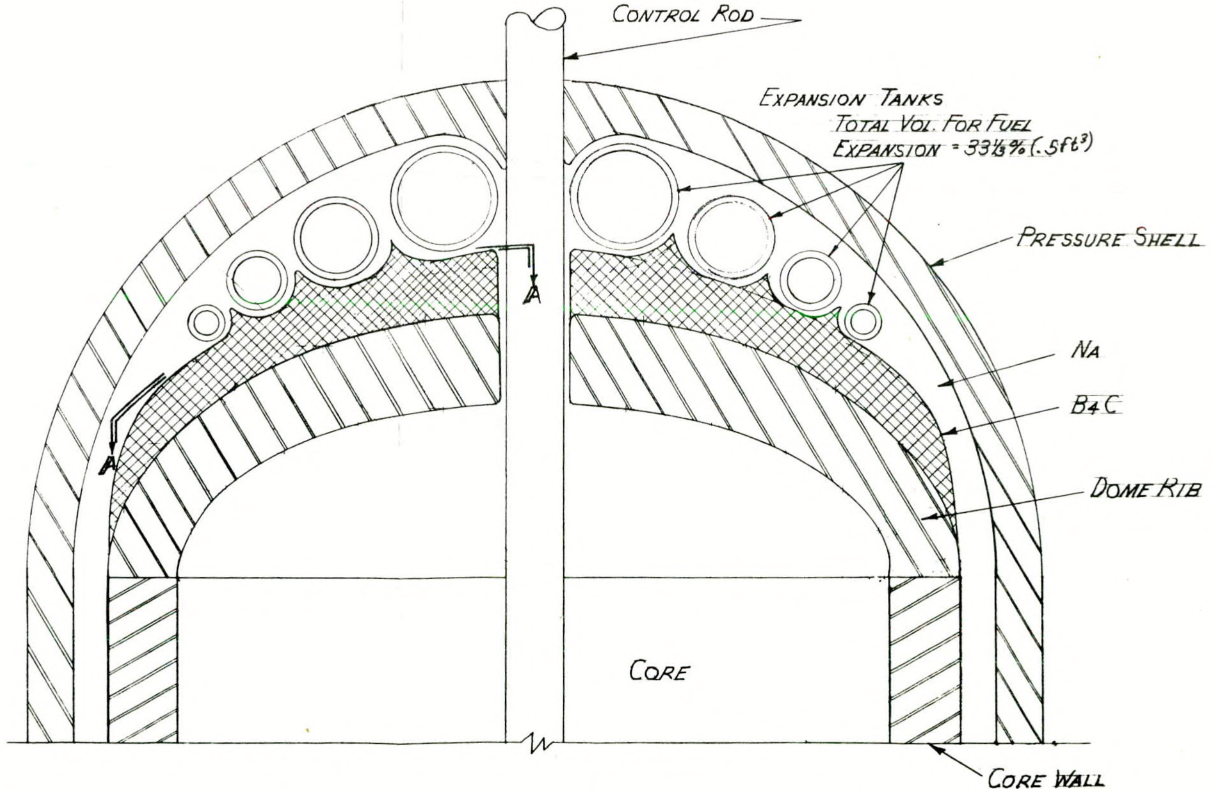


FIGURE 1.2 HELICAL COIL ARRANGEMENT (Dome & Fuel Tanks)

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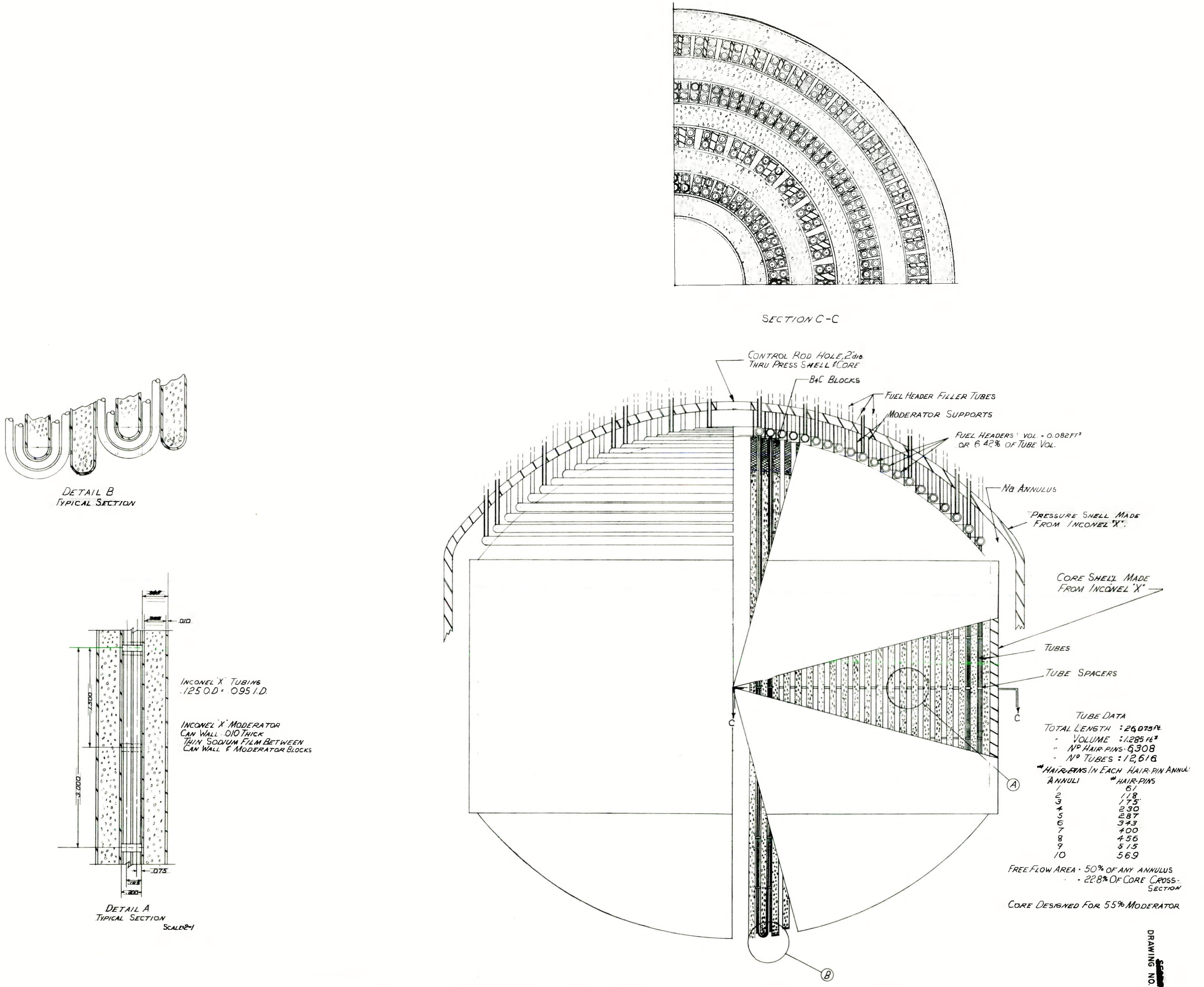

TUBE DATA

TOTAL LENGTH : 26,075 ft.
" VOLUME : 1.285 ft³
" Nº HAIR-PINS · 6,308
" Nº TUBES : 12,616

HAIR-PINS IN EACH HAIR-PIN ANNULI

ANNULI	# HAIR-PINS
1	61
2	118
3	175
4	230
5	287
6	343
7	400
8	456
9	515
10	569

FREE FLOW AREA · 50% OF ANY ANNULUS
" · 22.8% OF CORE CROSS-SECTION

CORE DESIGNED FOR 55% MODERATOR

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DRAWING NO. Y-F7-35

FIGURE 1.3 HAIR-PIN ("U") TUBE FUEL ELEMENT DESIGN (Reactor Core)

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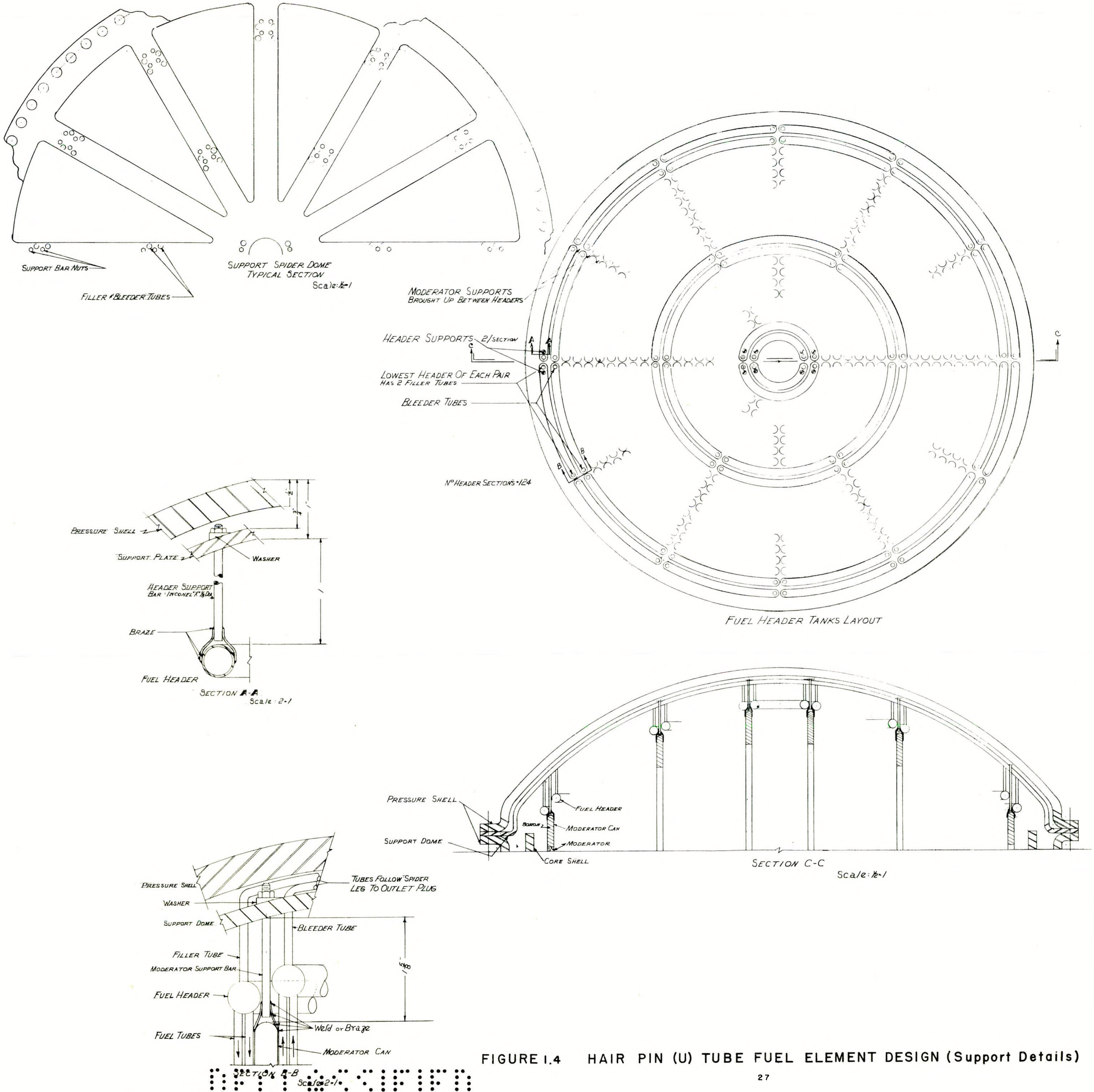

FIGURE 1.4 HAIR PIN (U) TUBE FUEL ELEMENT DESIGN (Support Details)

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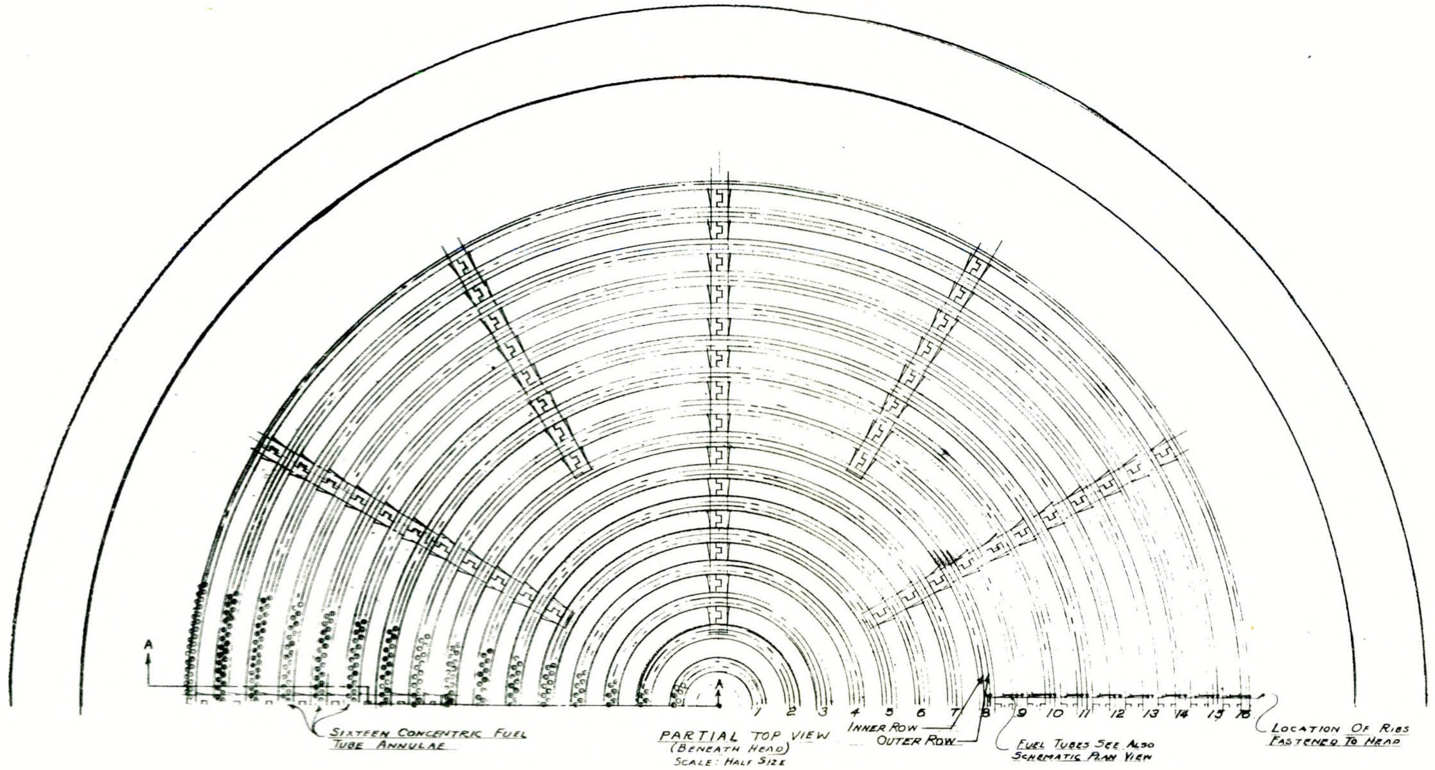
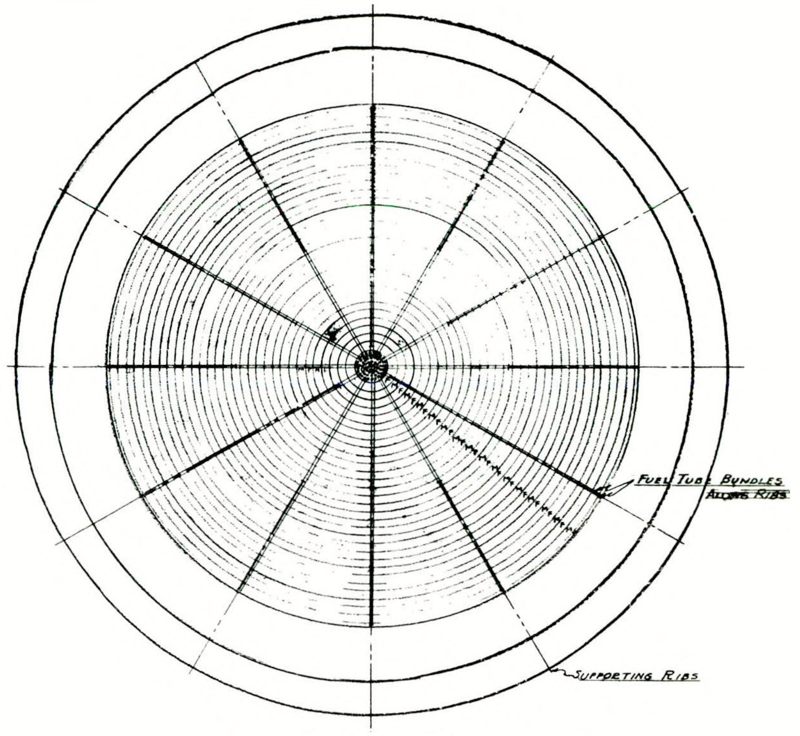
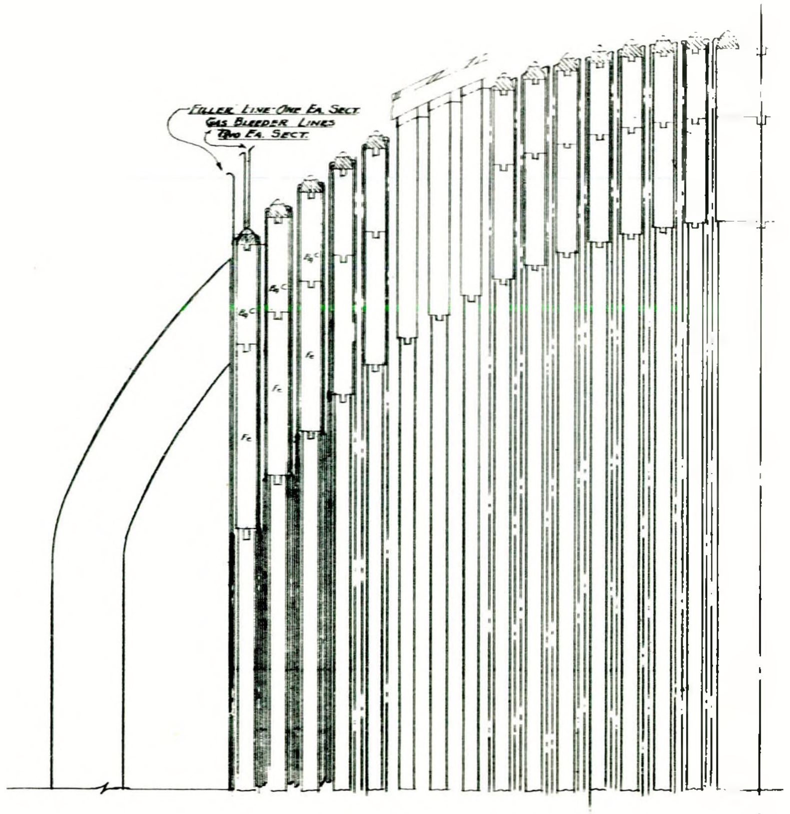


PARTIAL TOP VIEW
(Beneath Head)
Scale: Half Size



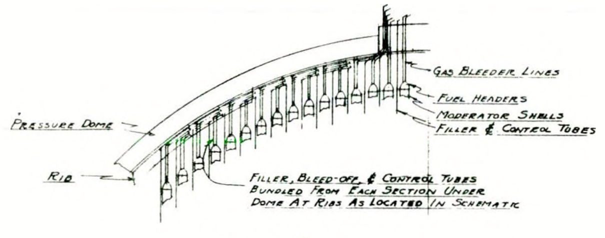
Schematic Plan View
Scale: Quarter Size



Core Arrg't. Same Top & Bottom

SECTION A-A
Scale: Half Size

Sectionalizing Table				
Row		Row		Sections
1	Inner			1
1	Outer .	2	Inner	2
2	"	3	"	3
3	"	4	"	3
4	"	5	"	3
5	"	6	"	6
6	"	7	"	6
7	"	8	"	6
8	"	9	"	12
9	"	10	"	12
10	"	11	"	12
11	"	12	"	12
12	"	13	"	12
13	"	14	"	12
14	"	15	"	12
15	"	16	"	12



Partial Tube Schematic
Scale Quarter Size

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FIGURE 1.5 PARALLEL TUBE FUEL ELEMENT DESIGN (General Arrangement)

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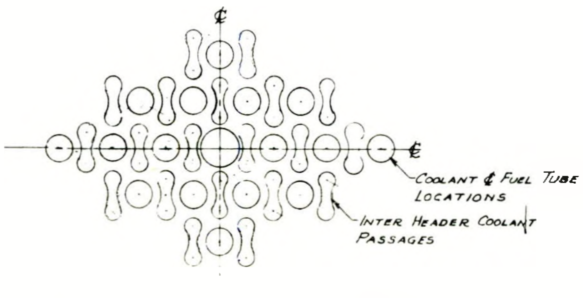
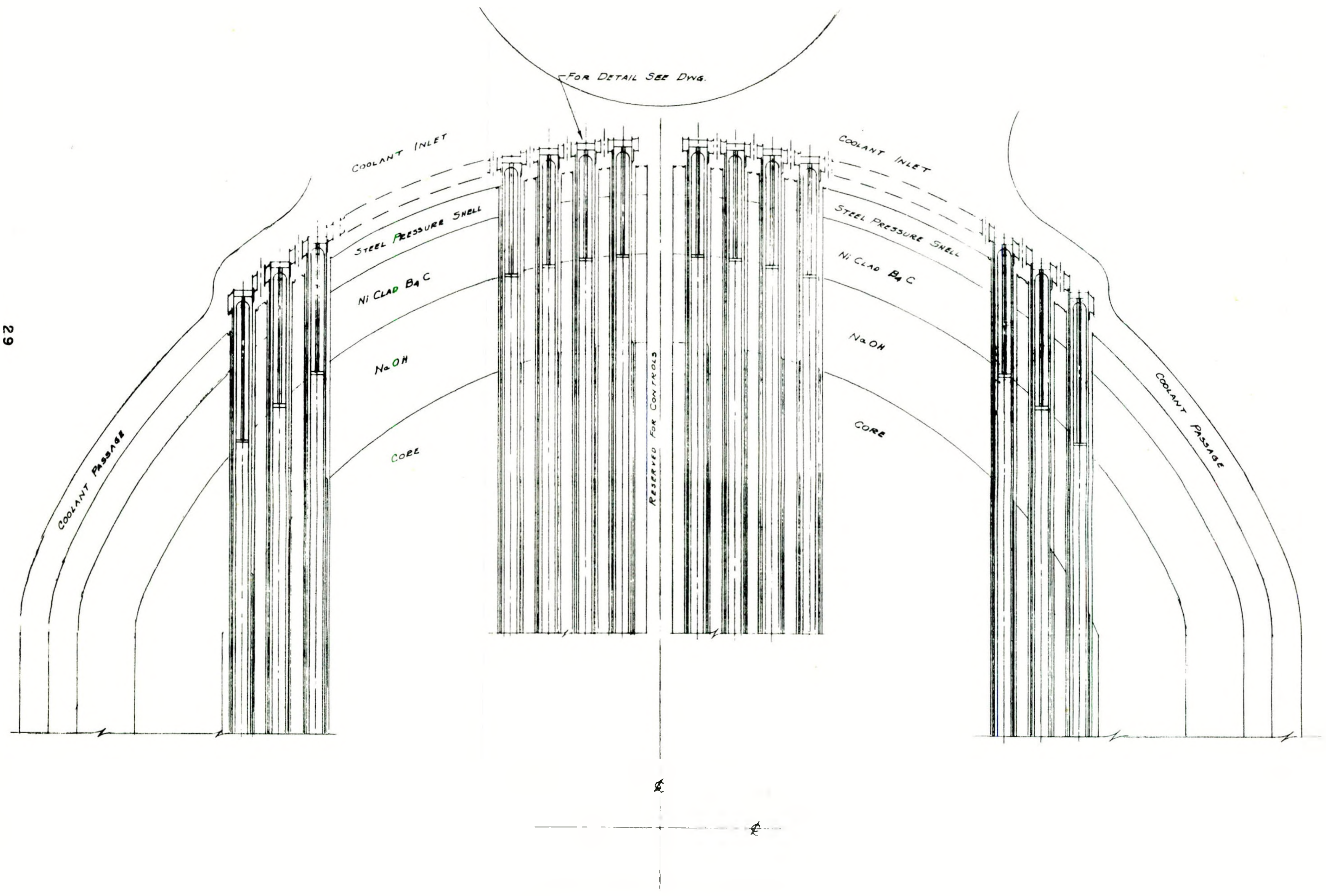


PORTION OF TOP
SHOWING ROD CONFIGURATION



PARTIAL ASSEMBLY SECTION

FIGURE 1.6 ANNULAR FUEL TUBE ARRANGEMENT (General Assembly)

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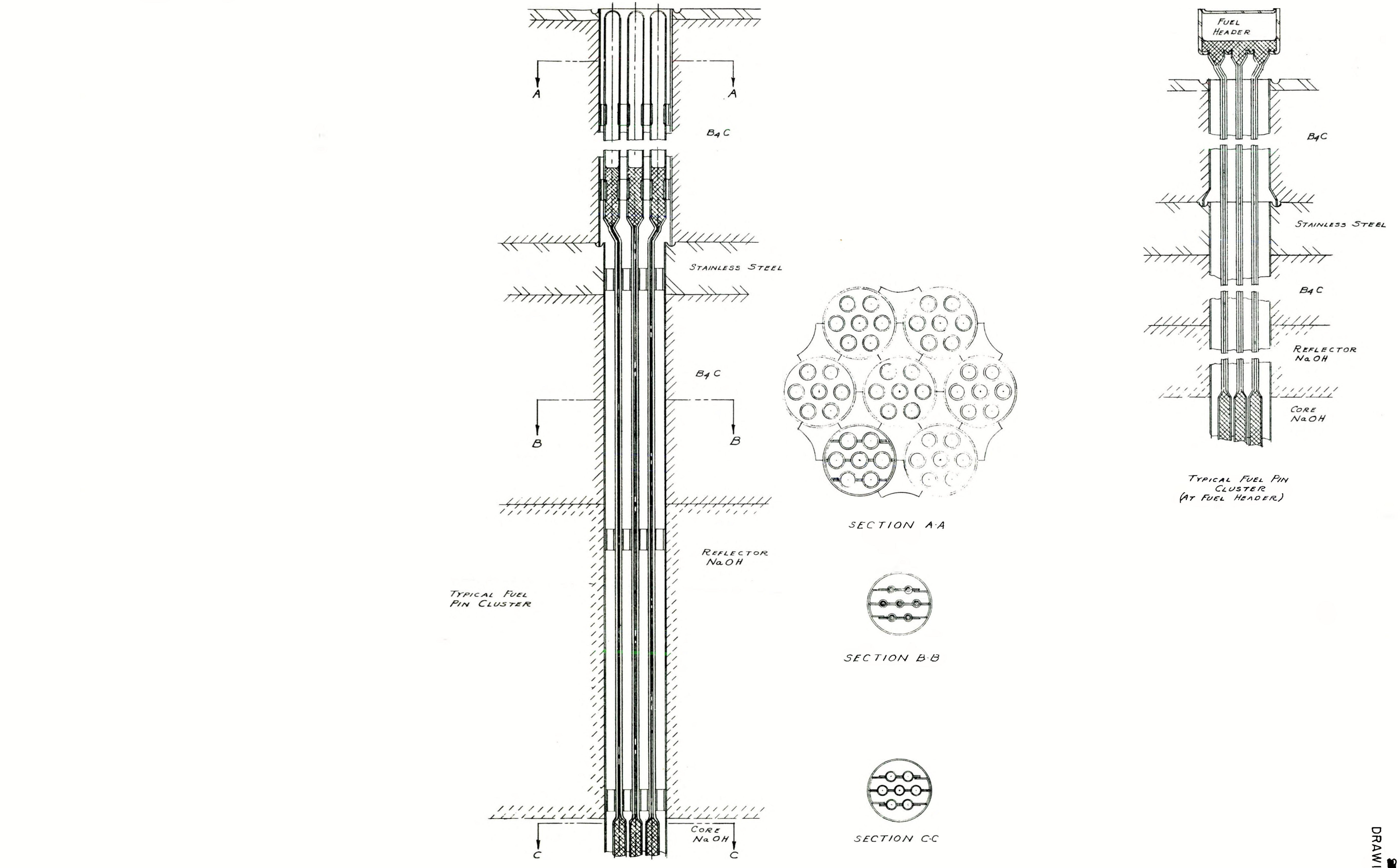

FIGURE 1.7 FUEL PIN ARRANGEMENT
(Typical Pin Assembly)

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[redacted]
DRAWING NO. YF7-29

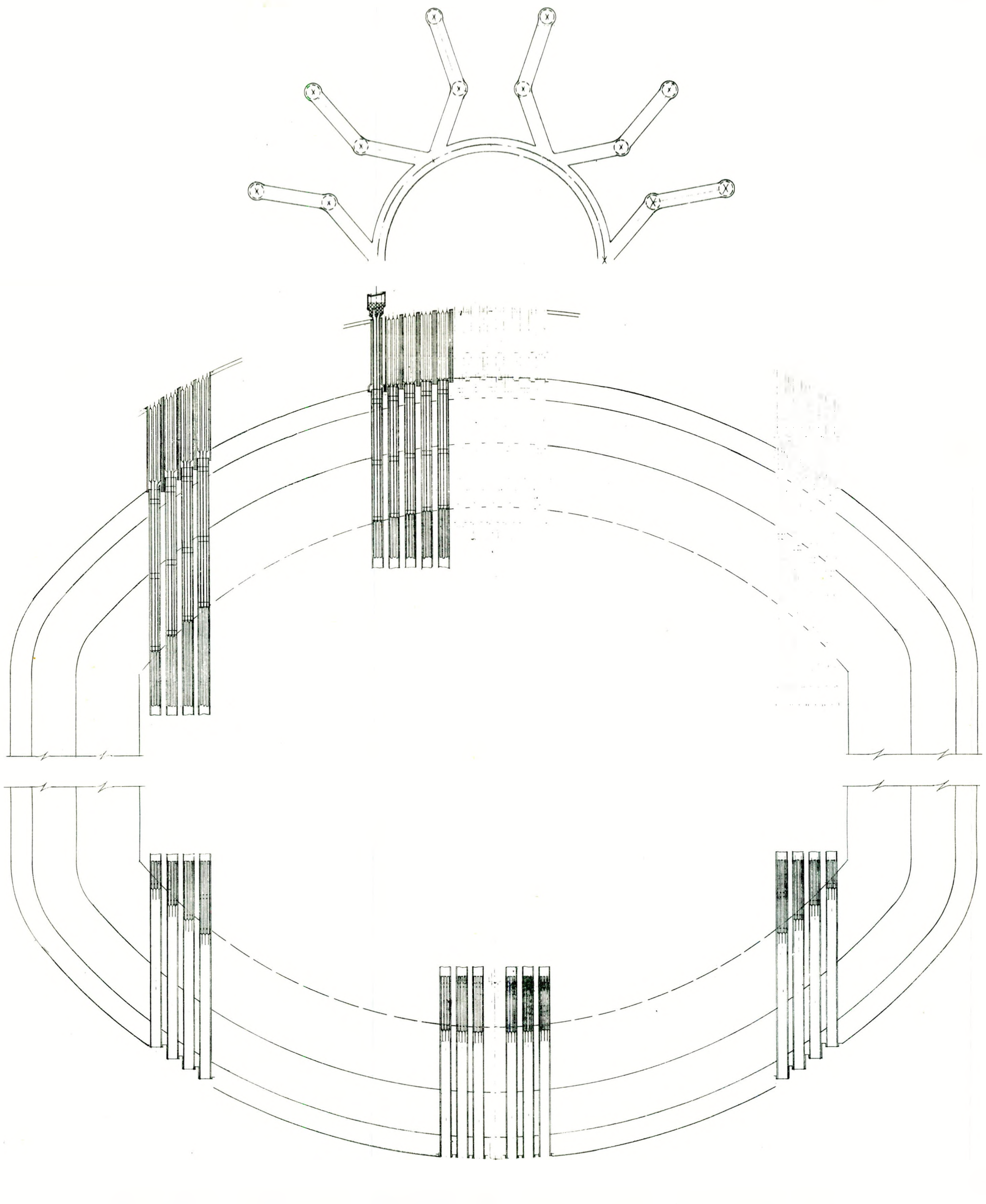

FIGURE 1.8 INDIVIDUAL FUEL PIN ARRANGEMENT (General Assembly)

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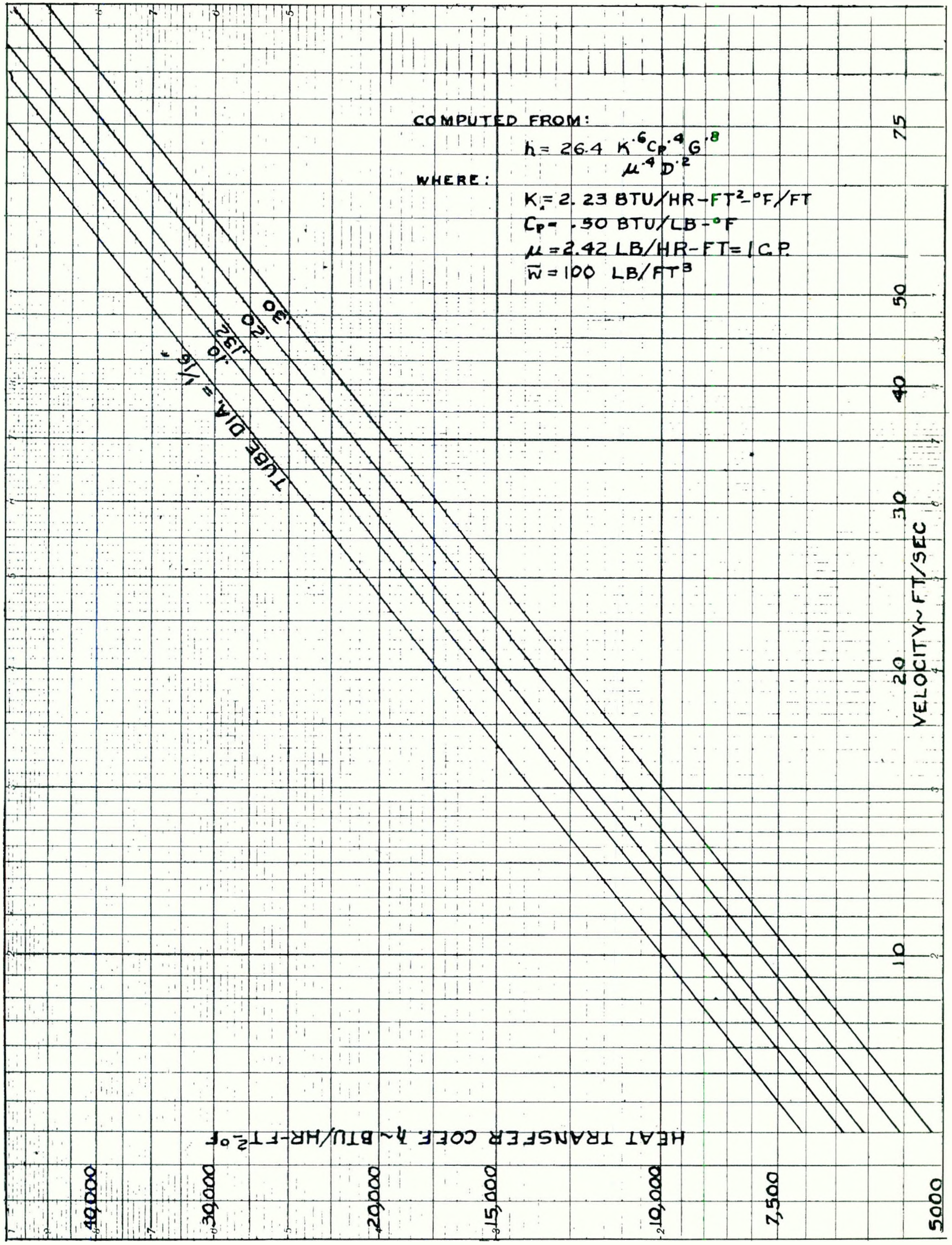

FIGURE 1.9 HEAT TRANSFER COEFFICIENT FOR NaOH AT 1350°

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TABLE 1.

Physical Data of Fluorides in the Fused State

FORMULA	M.P. (°C)	B.P. (°C)	SPECIFIC GRAVITY	SPECIFIC HEAT* (Btu/lb °F)	VISCOSITY (centipoises)	THERMAL CONDUCTIVITY (Btu/hr ft °F)
LiF	$870^{(1)}$	$1676^{(1)}$	$1.798 - 0.000437\ (t - 850)$ at $t\,°C^{(2)}$ 1.789 at 868.5°C$^{(3)}$ 1.753 at 944°C$^{(3)}$ 1.713 at 1029°C$^{(3)}$ 1.672 at 1117°C$^{(3)}$ 1.629 at 1208°C$^{(3)}$ 1.599 at 1270°C$^{(3)}$	0.578		~1.5 (calculated)$^{(5)}$
NaF	$980 - 997^{(1)}$	$1700^{(1)}$	$1.942 - 0.000564\ (t - 1000)$ at $t\,°C^{(2)}$ 1.936 at 1010°C$^{(3)}$ 1.887 at 1097°C$^{(3)}$ 1.859 at 1147°C$^{(3)}$ 1.810 at 1234°C$^{(3)}$ 1.766 at 1313°C$^{(3)}$ 1.714 at 1405°C$^{(3)}$ 1.662 at 1497°C$^{(3)}$ 1.634 at 1546°C$^{(3)}$	0.357	~1 (estimated from NaBr and NaCl)	~1 (estimated from LiF and KF)
KF	$880^{(1)}$	$1500^{(1)}$	$1.878 - 0.000669\ (t - 900)$ at $t\,°C^{(2)}$	0.258		~0.7 (calculated)$^{(5)}$
BeF	$800^{(1)}$	Subl.$^{(4)}$	1.986 for the solid**	0.319		
RbF	$760^{(1)}$	$1410^{(1)}$	$2.873 - 0.000967\ (t - 825)$ at $t\,°C^{(2)}$	0.144		
CsF	$684^{(1)}$	$1250^{(1)}$	$3.611 - 0.001234\ (t - 700)$ at $t\,°C^{(2)}$	0.099		
AlF_3	$1040^{(1)}$	$^{(1,4)}$	$3.07^{(1)}$ for the solid**	0.197		
CaF_2	$1360^{(1)}$	$^{(1,4)}$	$3.18^{(1)}$ for the solid**	0.281		~1 (calculated)$^{(5)}$
MgF_2	$1396^{(1)}$	$2239^{(1)}$	2.9 to $3.2^{(1)}$ for the solid**	0.353		
BaF_2	$1280^{(1)}$	$2137^{(1)}$	$4.83^{(1)}$ for the solid**	0.125		
$AlF_3 \cdot 3NaF$				0.352		

* Specific heat calculated by Kopp's rule for liquids.

** For LiF, NaF, and KF, $(C_p\ liq)/(C_p\ solid) \sim 0.6$ to 0.75; for CsF, $(C_p\ liq)/(C_p\ solid) \sim 1(?)$.

(1) *Handbook of Chemistry and Physics*, Ed. C. D. Hodgman, 31st ed., Chemical Rubber Publishing Co., Cleveland, Ohio, 1949.

(2) Jäger, F. M., *Z. anorg. Chem.*, 101, 1 (1917). (3) International Critical Tables, McGraw-Hill, New York, 1929.

(4) Lange, N. A., *Handbook of Chemistry*, 7th ed., Handbook Publishers, Sandusky, Ohio, 1949.

(5) Manson, S. V., Theoretical Equations for Estimating Thermal Conductivity of Liquids, ORNL, Y-12 Site, Y-F8-6 (Nov. 13, 1950).

TABLE 1.1

Physical Data for Hydroxides In the Fused State

FORMULA	M.P. (°C)	B.P. (°C)	SPECIFIC GRAVITY	SPECIFIC HEAT (Btu/lb °F)	VISCOSITY (centipoises)(3)						THERMAL CONDUCTIVITY (Btu/hr ft °F)
					350°C	400°C	450°C	500°C	550°C	600°C	
NaOH	318.4(4)	1390(4)	2.11 - 0.00063t for t between 340 and 440°C(1) 1.746 $[1 - 2.74 \times 10^{-4}(t - 400)]$(2a) 1.786 at 320°C(2a) 1.771 at 350°C(2a) 1.746 at 400°C(2a) 1.722 at 450°C(2a) 1.90 at 320°C(2b) 1.89 at 340°C(2b) 1.88 at 360°C(2b) 1.87 at 380°C(2b) 1.86 at 400°C(2b) 1.85 at 420°C(2b) 1.84 at 440°C(2b) 110 lb/ft^3 at m.p.(6)	0.50(6) 0.47 by Kopp's rule for liquids(5)	4.0	2.8	2.2	1.8	1.5		2.23(7) 1(6)
KOH	360.4 ± 0.7(4) 380(5)	1320-1324(4) 1320(5)	2.25 0.001t for t from 380 to 440°C(1) 1.717 $[1 - 2.56 \times 10^{-4}(t - 400)]$(2c) 1.717 at 400°C(2c) 1.695 at 450°C(2c) 1.673 at 500°C(2c) 1.651 at 550°C(2c) 1.629 at 600°C(2c)	0.34 by Kopp's rule for liquids(5)		2.3	1.7	1.3	1.0	0.8	If the compressibility of KOH were about 1.5 times that of NaOH, the thermal conductivity of KOH would be about 0.65. (thermal conductivity of NaOH; see Bridgmann's equation and items b and d of Y-F8-6)

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(1) International Critical Tables, McGraw-Hill, New York, 1929.

(2) Landolt-Börnstein Tables, J. Springer, Berlin:
  - (*a*) Eq. II, Part I, Table 81*K*, p. 223.
  - (*b*) Eq. I, Table 81*F*, p. 177.
  - (*c*) Eq. II, Part I, Table 81*F*, p. 219.

(3) Arndt and Ploetz, *Z. phys. Chem.* 121, 439 (1926).

(4) *Handbook of Chemistry and Physics*, Ed. C. D. Hodgman, 31st ed., Chemical Rubber Publishing Co., Cleveland, Ohio, 1949.

(5) Lange, N. A., *Handbook of Chemistry*, 7th ed., Handbook Publishers, Sandusky, Ohio, 1949.

(6) NEPA Report IC-50-4-20 (April, 1950).

(7) Battelle Memorial Institute, telephone conversation.

If the various problems associated with reactors of this type can be solved satisfactorily, designs 1, 2, 3, and 4 would afford great flexibility in loading and unloading the fuel. The entire core could be fabricated and assembled within the shield and the liquid metal coolant put in the system and the various pumps brought up to operating levels and temperatures before the reactor contains any fuel. The fuel could then be added slowly and might have its concentration varied over a fairly wide range. The initial start-up problems thus seem much less difficult than with solid-fuel reactors.

By compartmentalization and the use of various concentrations of uranium in the fuel solution, a considerable variation of fuel distribution across the reactor core could be obtained, if desired for adjustment of the flux pattern. Another potential advantage of the liquid fuel for an aircraft reactor is that most of the radioactivity could be drained from the system at one step, thus greatly simplifying the ground handling problems if a divided shield is used.

Coolant Circuit Design. It is expected, as indicated in previous reports, that sodium will be both the primary and the secondary coolant for the ARE. Coolant circuit design work currently is in progress with particular emphasis on the components which affect the design of the building. Fundamental coolant circuit specifications which have been adopted at the present time are:

1. The fluid circuit will simulate the airplane system as far as practicable within the shield if a unit shield is adopted. The secondary circuit outside the shield is to be designed primarily for safety and convenience. Flow rates are to be as required to remove power at the rate developed by the ARE.
2. Final heat rejection is to be by transfer from the secondary circuit to air.
3. Multiple primary coolant pumps will be used.
4. Multiple intermediate heat exchangers will be used.
5. The ARE is expected to contain two completely independent secondary systems with separate pipes, pumps, radiators, radiator blowers, and accessories.
6. Two independent power sources of auxiliary power will serve the secondary loops and will transmit power to the primary pumps so as to maintain circulation in the event of failure of one power source.

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In addition to the above features, which are similar to those expected in the final aircraft system, several special items are planned for the ARE for protection of personnel and reactor under experimental conditions. These include the following:

7. Dump systems for control of liquid-metal fires will be provided.

8. A pressure shell will be constructed about the reactor-shield assembly, to contain materials that might escape from the shield in the event of an accident.

9. The pump room will be isolated and shielded to protect personnel from radioactive gases in the event of partial failure.

Building Design for the ARE. The Test Facility building design now proposed consists of a steel, concrete, and masonry fire-resistant structure 80 ft wide by 90 ft long. The building is expected to contain a crane bay approximately 42 ft high, so that large pieces of the reactor or of the shield may be lifted out of the assembly. A basement level is expected to house the liquid-metal pumps and the disassembly and decontamination rooms.

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2. REACTOR PHYSICS

N. M. Smith, Jr., Chairman
ANP Physics Group, Physics Division

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2. REACTOR PHYSICS

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A. INTRODUCTION

The principal efforts of the ANP Reactor Physics Group during the last quarter have been directed toward two general types of calculations: computations of criticality and thermal xenon coefficients in both bare and reflected reactors, and calculations on the kinetic response of some liquid-fuel stationary-moderator reactor designs.

These calculations deal with homogeneous reactors, both bare and reflected. The principal effect which must be added to the picture in order to apply the answers in detail to designs of the ARE type is the one brought about by the heterogeneous nature of the core, i.e., the effect of self-shielding caused by lumping. Thus the present calculation of the critical mass of uranium may be in error from 50 to 100%. The effects of perturbations, however, are obtained more accurately. The application of methods of calculating "effective homogeneous" cross sections to replace the atomic cross sections now used should reduce the pre-experimental errors of critical mass to around 30%. Experience with calculating results of critical assemblies should ultimately permit making calculations on a new reactor within 10 to 15%. In spite of these uncertainties, the present refinement of the calculation is valuable in guiding design in a semiquantitative sense and for gaining an understanding of the overall field of intermediate reactors through exploratory calculations.

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It is to be noted that the "bare reactor" calculations described in Section B really represent the first approximation to an actual reflected ARE reactor, since the effect of a reflector was simulated by modification of the core diameter to allow for estimated reflector savings. The so-called "reflected reactor" calculations (Section C) represent a second, and more difficult, approximation in which the core-plus-reflector combination is treated explicitly as a two-region problem.

In addition to the calculations of critical masses and associated core fluxes, neutron lifetimes, and control rod effects, particular attention has been paid to that contribution to the temperature coefficient of reactivity which arises from the shift of the thermal base across the xenon absorption curve. This parameter is vitally important for reactor control. For those ARE designs which involve a large volume percentage of moderator, the positive xenon coefficient is so large that it yields a net positive temperature coefficient for the reactor as a whole. This coefficient is, however, associated with the temperature of the moderator. It is quite clear that the design to be used for the aircraft reactor must be arranged to have either a total temperature coefficient of reactivity which is negative, or else a thermal response time of the moderator which is so long that slow-acting servo controls are adequate and safe.

As described previously in this report, the liquid-fuel designs for the ARE offer the attractive possibility of gaining a rapidly acting negative temperature coefficient of reactivity through expansion of the fuel to points outside the active core. This allows control by means of a relatively sluggish temperature-sensing device in place of a neutron-flux—sensing receptor. However, the problem of the kinetics of a "stationary" liquid-fuel reactor is intricate. It requires study not only of the nuclear response of the system to a change in reactivity, but also an investigation of the time-temperature-density transients of the liquid fuel, the coolant, and the moderator. Studies of maximum stresses in the fuel tubes, etc., under emergency conditions are also required. This large body of analyses is now being undertaken. Some initial perturbation-theory results are presented in Section D on the flux behavior and the kinetics of the liquid fuel for the simplified case where delayed neutrons are neglected. These perturbation theory calculations apply to infinitesimal effects and serve to determine designs having stable kinetics. Numerical integration of more exact equations are required for analysis of the stresses resulting from large perturbations.

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In Section E some of the background problems are listed which contribute to the broader calculations. They include studies on the Doppler effect, on adjoint functions, and on cylindrical multigroup age equations. Further work is in progress, as mentioned in Section F, in preparation for interpreting the results from forthcoming ANP critical experiments.

C. REFLECTED REACTOR CRITICALITY CALCULATIONS

D. K. Holmes and O. A. Schulze

The large amount of time required to make hand calculations for a reflected reactor made desirable the setting up of a machine method of calculations. Accordingly, the G.E. multigroup method was adopted with modifications and set up on IBM machines by the uranium Control Department (F. C. Uffelman) at Y-12. The corresponding step has already been taken by the IBM division at G.E., and they are now able to make calculations for two complete 13-group reflected reactors per day, which is contrasted with 10 computer-days needed for the same calculation by hand.

In conversation and correspondence with the theoretical section of the Physics Division of KAPL at Schenectady the essentials of the KAPL technique for adapting the method for IBM calculation were obtained.(5) However, a slight variation of their present procedure has actually been used here, although the variation is in reality equivalent to a method, called "scheme *b*," which has been discussed in a KAPL report.(6) Whereas KAPL obtains the relation between the average flux over a lethargy group and the values of the flux at the group limits by an iterative, self-consistency method, the present method assumes that the flux is linear over a group. Thus, the present method avoids a certain amount of calculation time which would be devoted to repeating groups. It has been decided that the calculation time on the IBM would

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(5) Letter from Hurwitz and Ehrlich to N. M. Smith, Aug. 14, 1950.

(6) Tonks, L., *Analyses of Errors in Method for Computing Critical Masses of Intermediate Piles Which Arise from a Spatially Discontinuous Source Distribution and Other Factors*, GE-LT-2 (1947).

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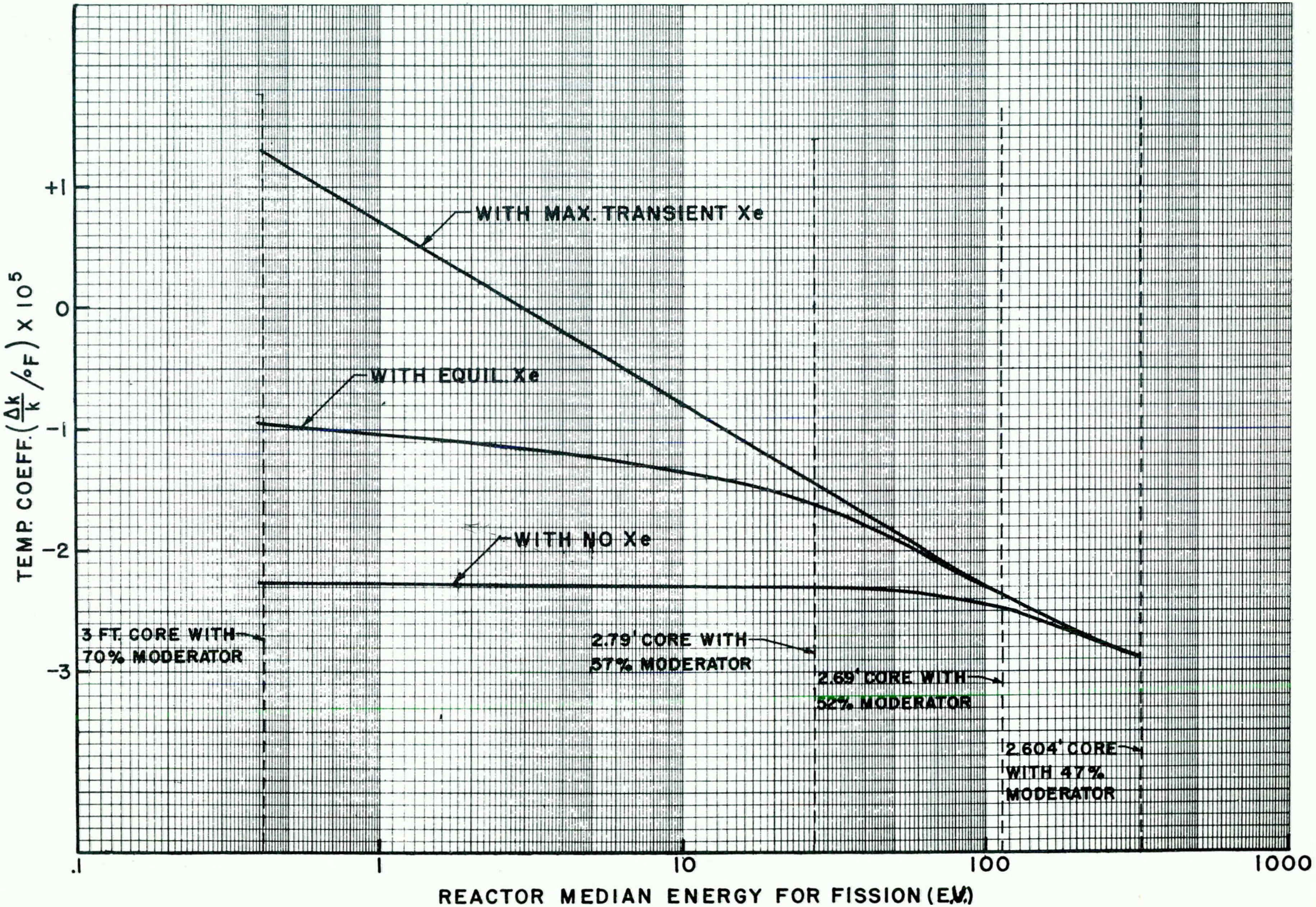

FIGURE 2.2 TEMP. COEFF. $\left(\frac{\Delta k}{k}/°F\right)$ AT MEAN CORE TEMPERATURE = 1286 °F

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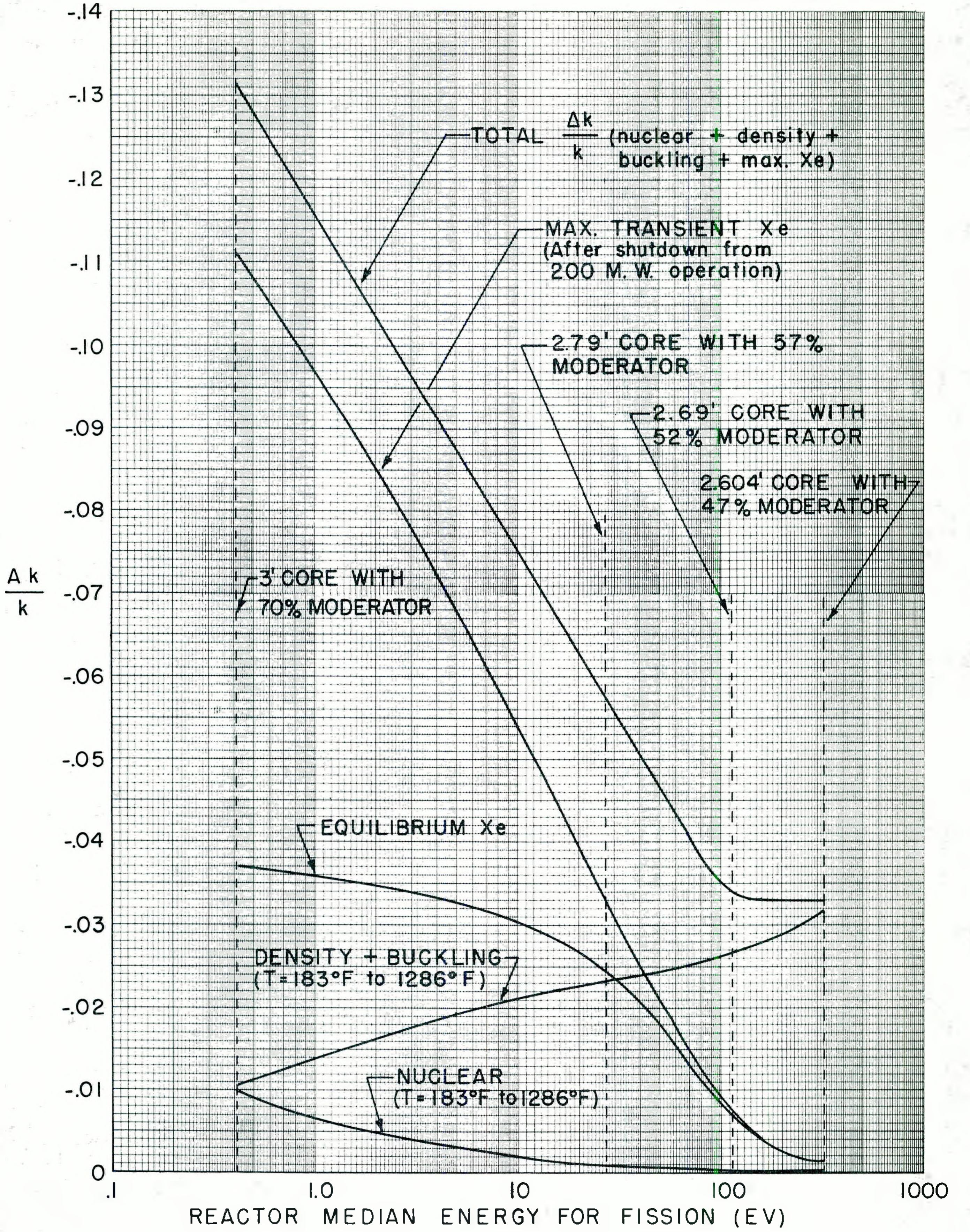

FIGURE 2.3 REACTIVITY EFFECTS AS A FUNCTION OF REACTOR SPECTRUM

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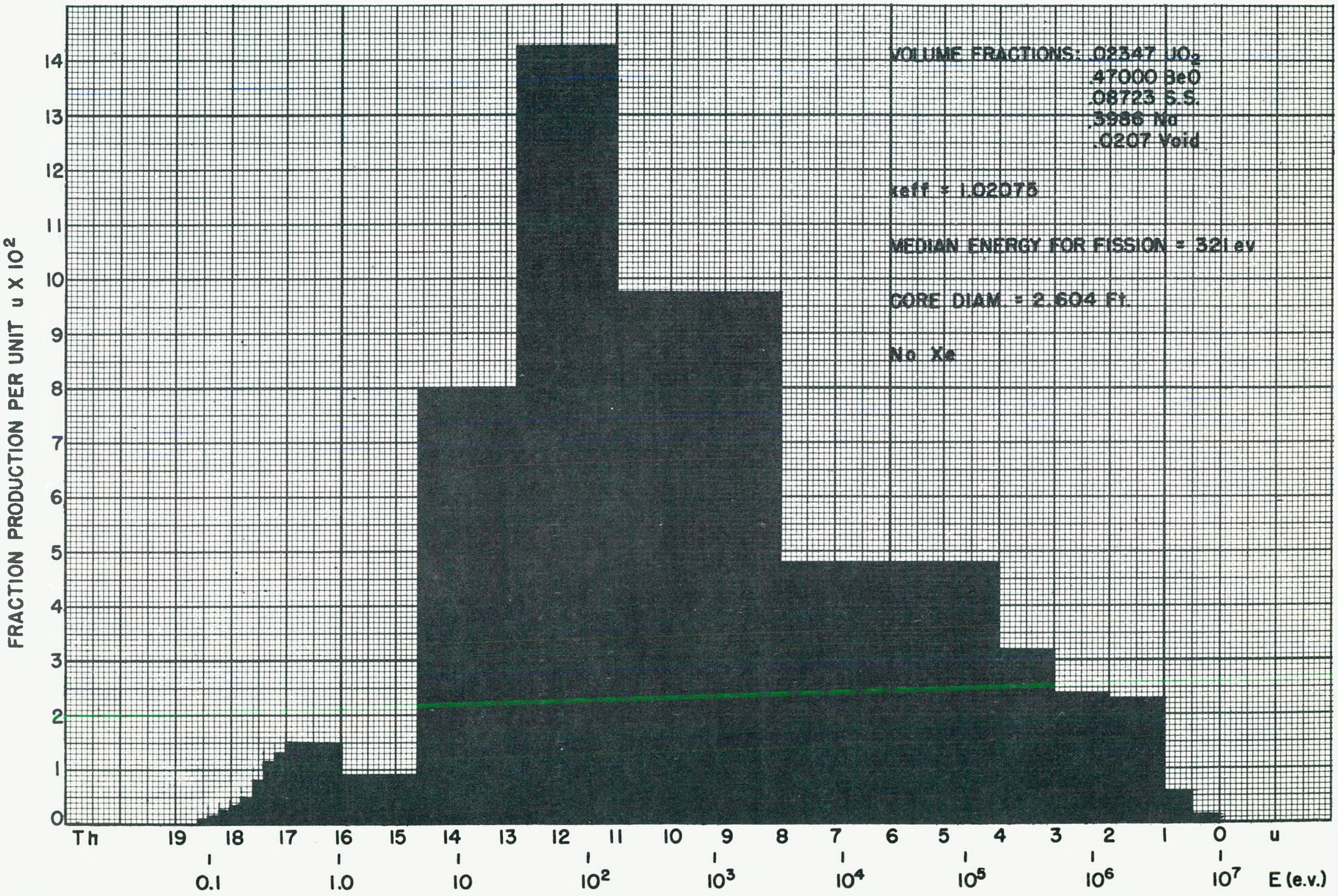

VOLUME FRACTIONS: .02347 UO_2
.47000 BeO
.08723 S.S.
.3986 Na
.0207 Void

keff = 1.02075

MEDIAN ENERGY FOR FISSION = 321 ev

CORE DIAM. = 2.604 Ft.

No Xe

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FIGURE 2.4 PRODUCTION SPECTRUM

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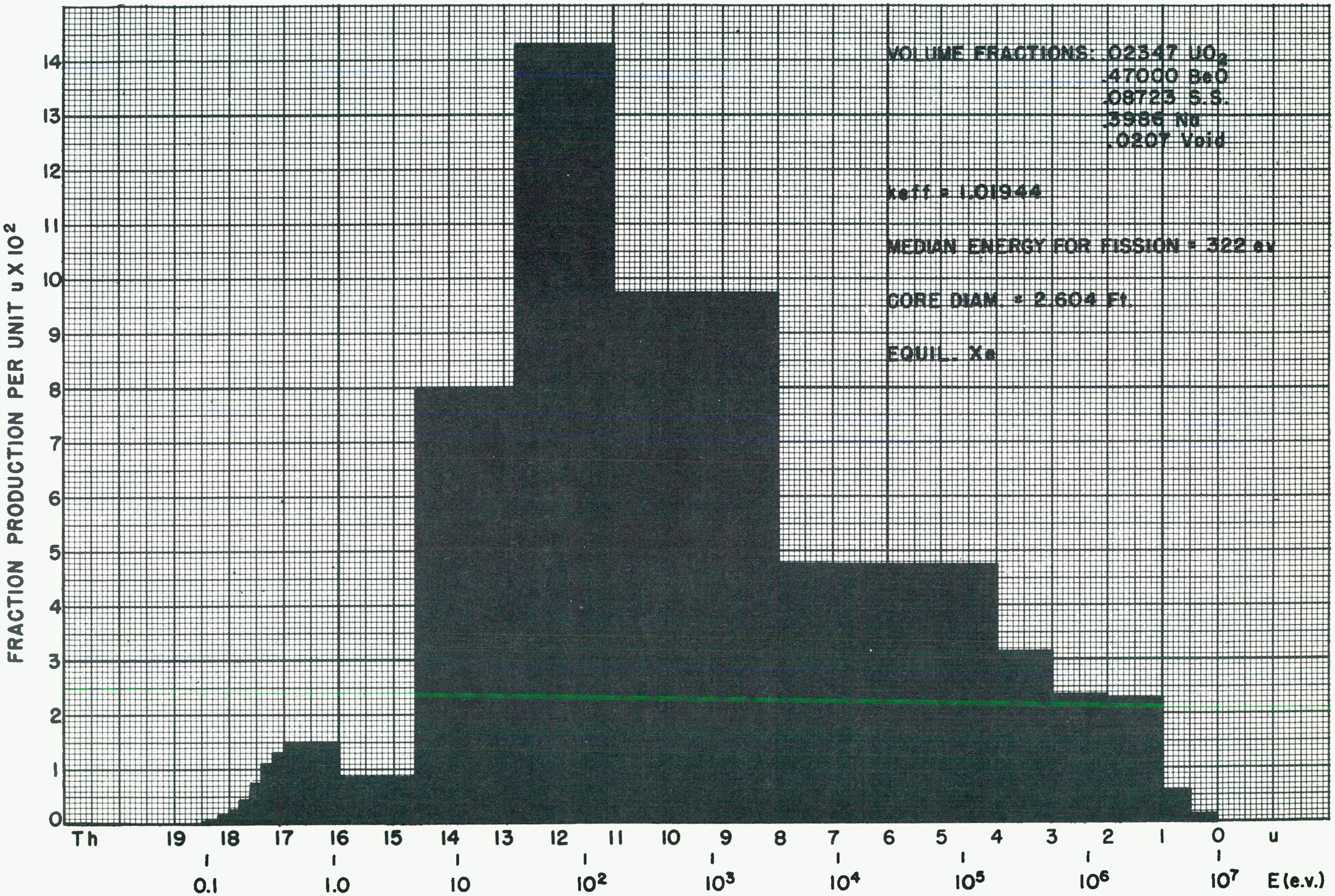

FIGURE 2.5 PRODUCTION SPECTRUM

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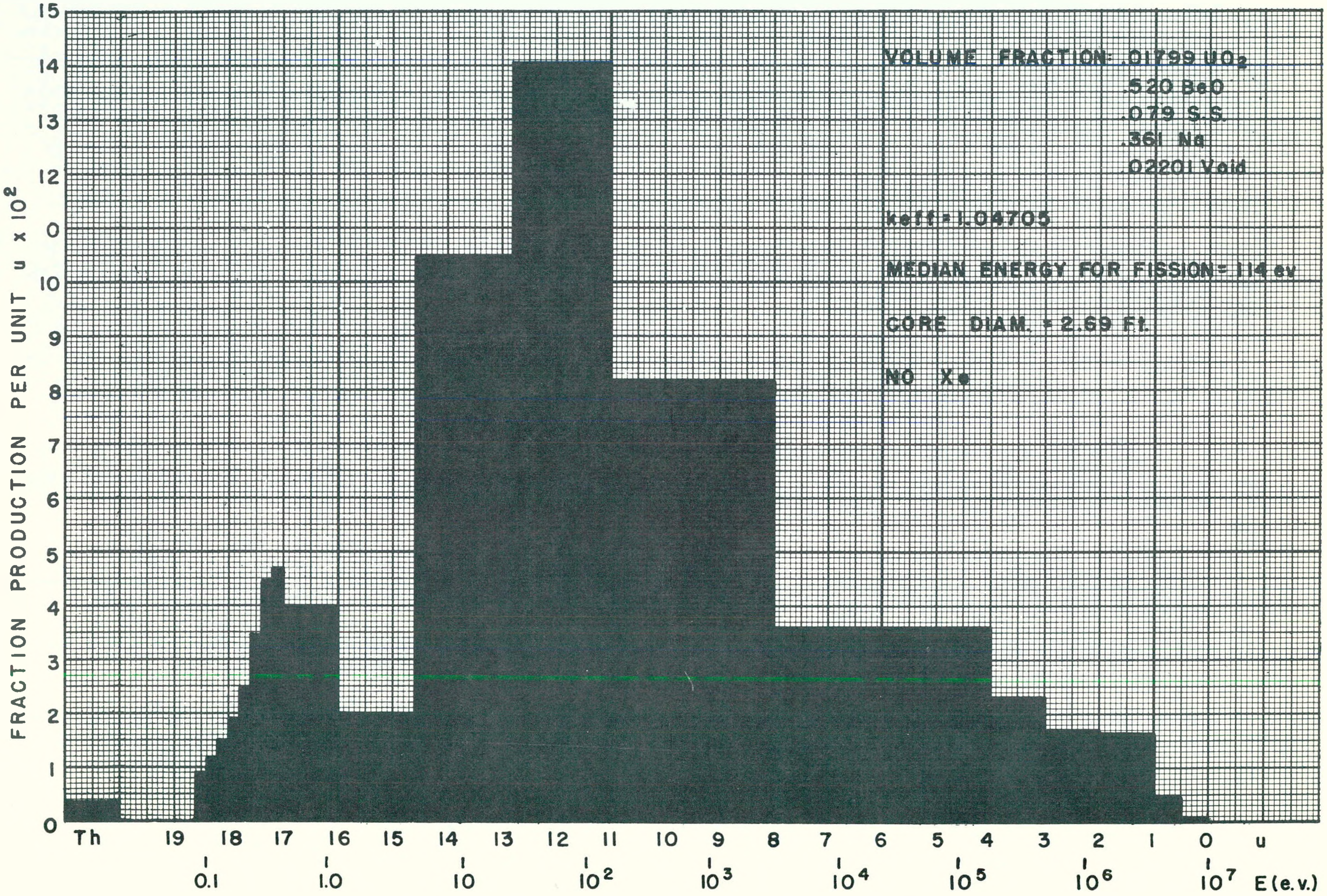

FIGURE 2.6 PRODUCTION SPECTRUM

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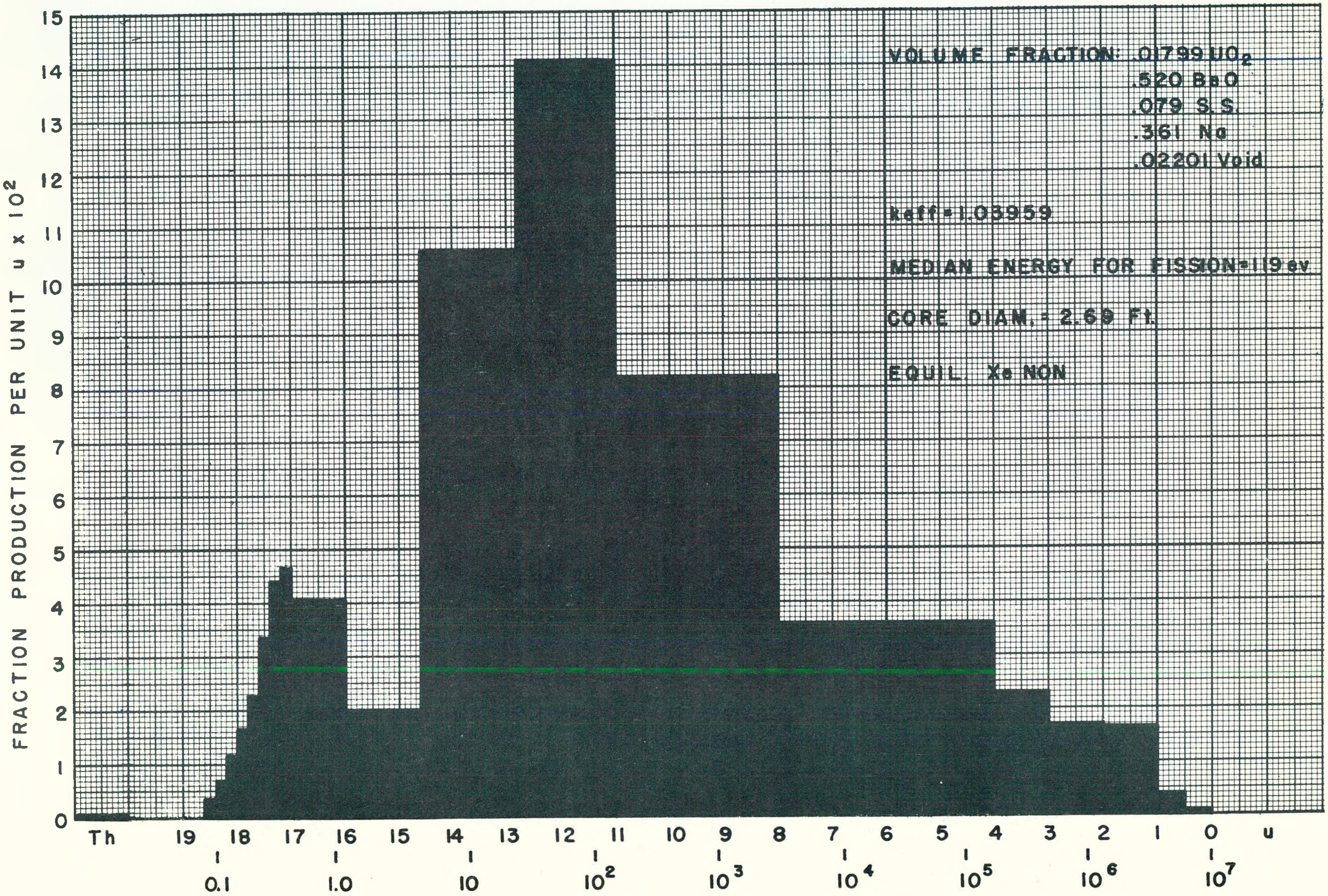

FIGURE 2.7 PRODUCTION SPECTRUM

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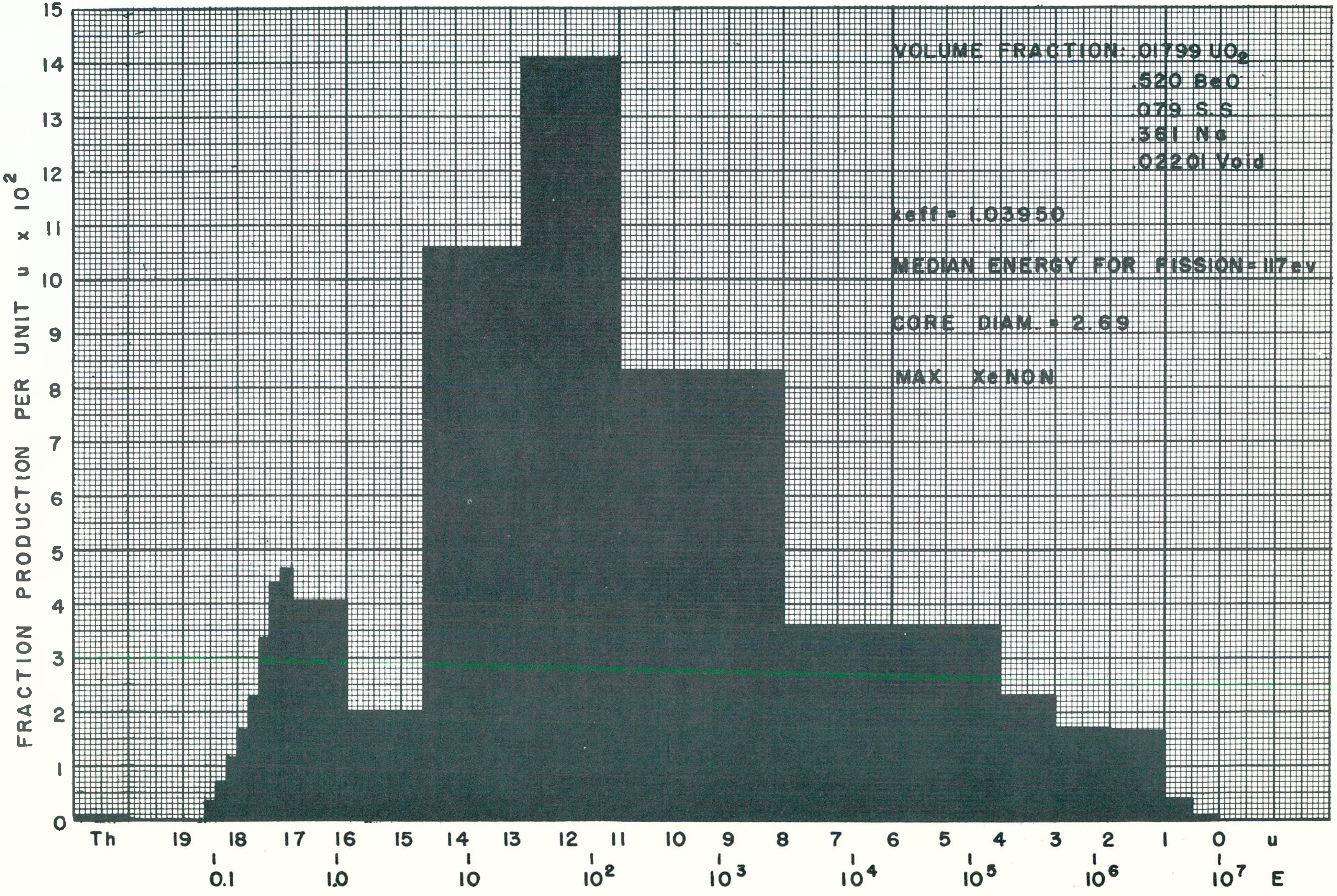

FIGURE 2.8 PRDUCTION SPECTRUM

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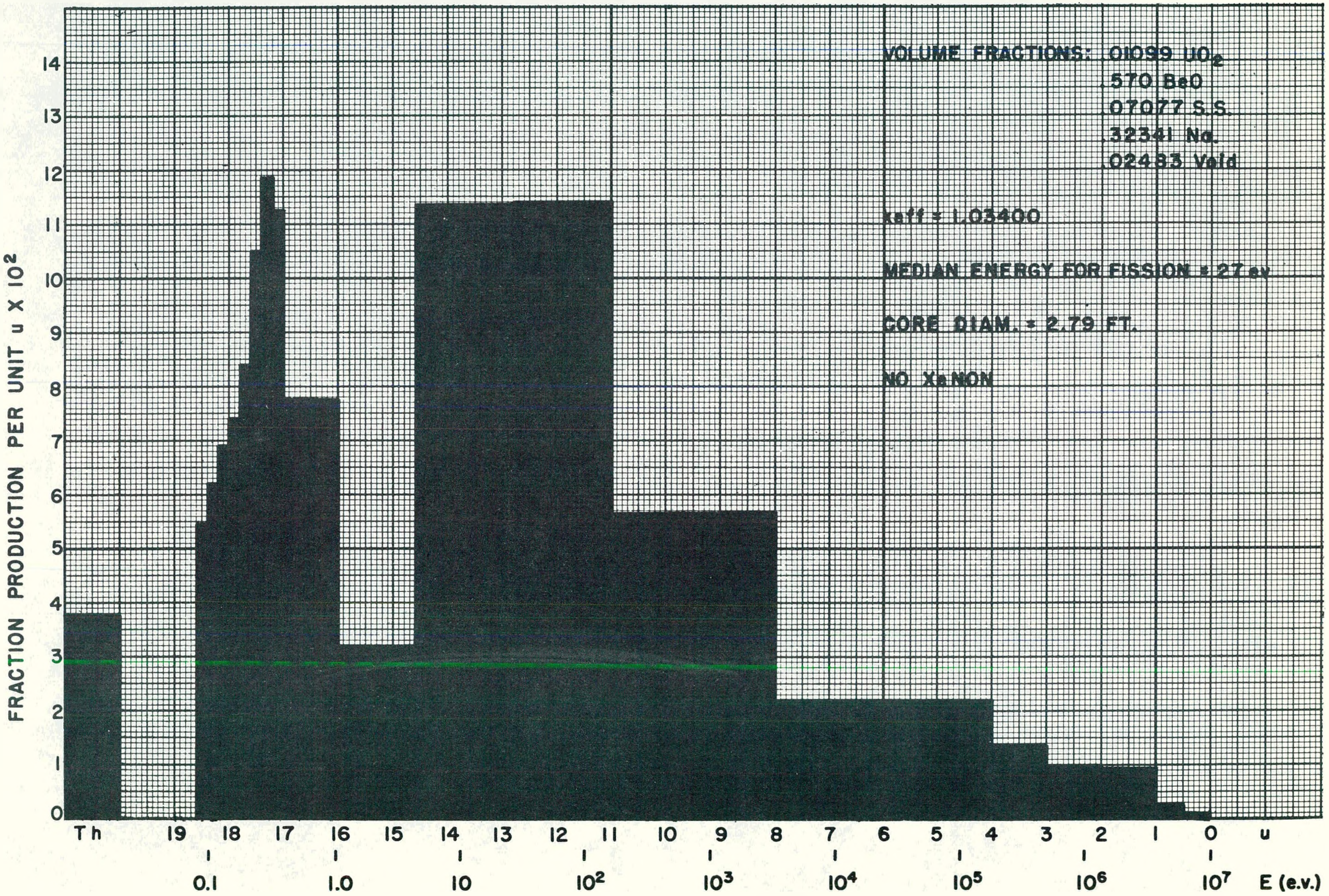

FIGURE 2.9 PRODUCTION SPECTRUM

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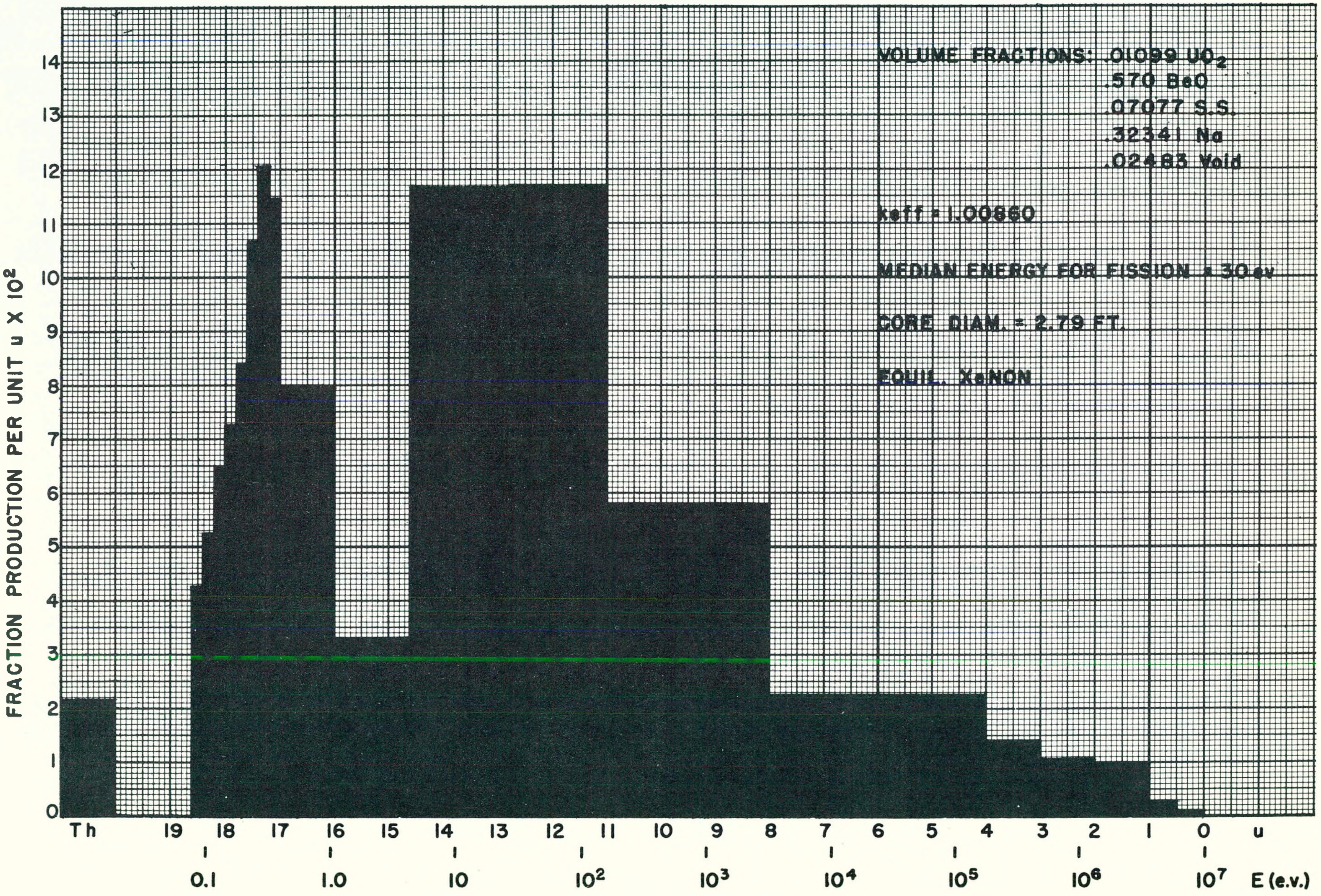

VOLUME FRACTIONS: .01099 UO_2
.570 BeO
.07077 S.S.
.32341 Na
.02483 Void

keff = 1.00860

MEDIAN ENERGY FOR FISSION = 30 ev

CORE DIAM. = 2.79 FT.

EQUIL. XeNON

FIGURE 2.10 PRODUCTION SPECTRUM

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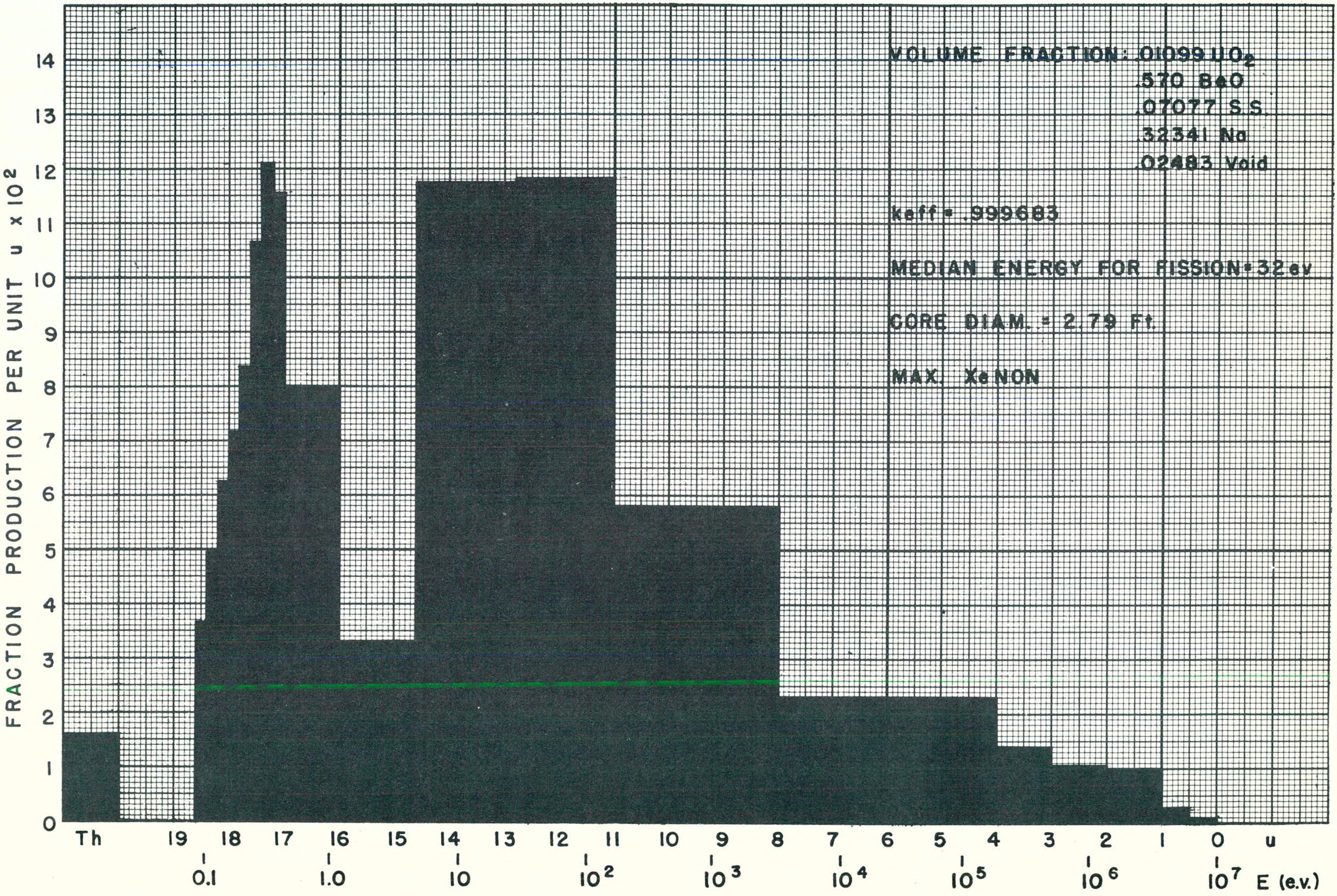

FIGURE 2.11 PRODUCTION SPECTRUM

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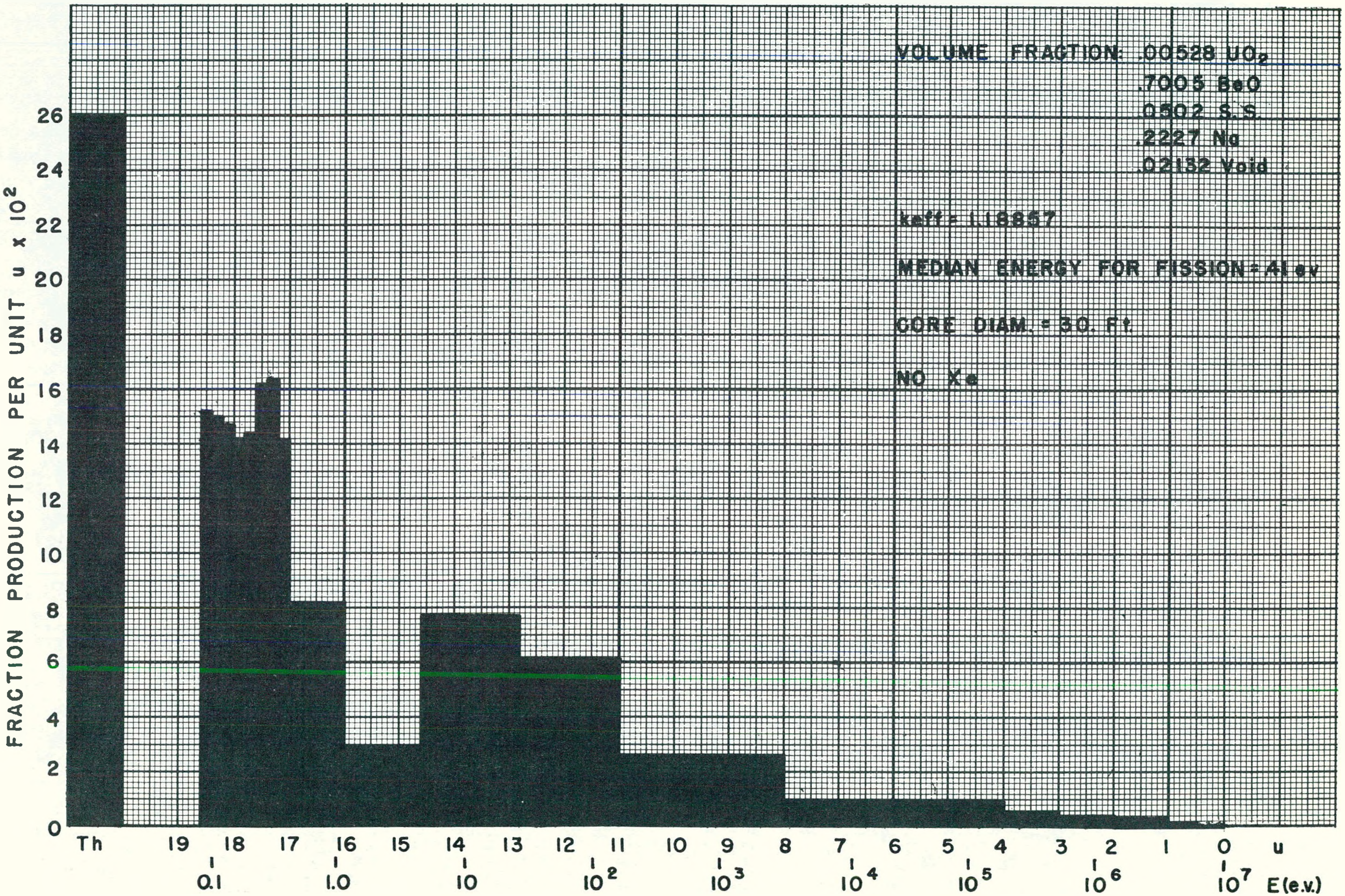

FIGURE 2.12 PRODUCTION SPECTRUM

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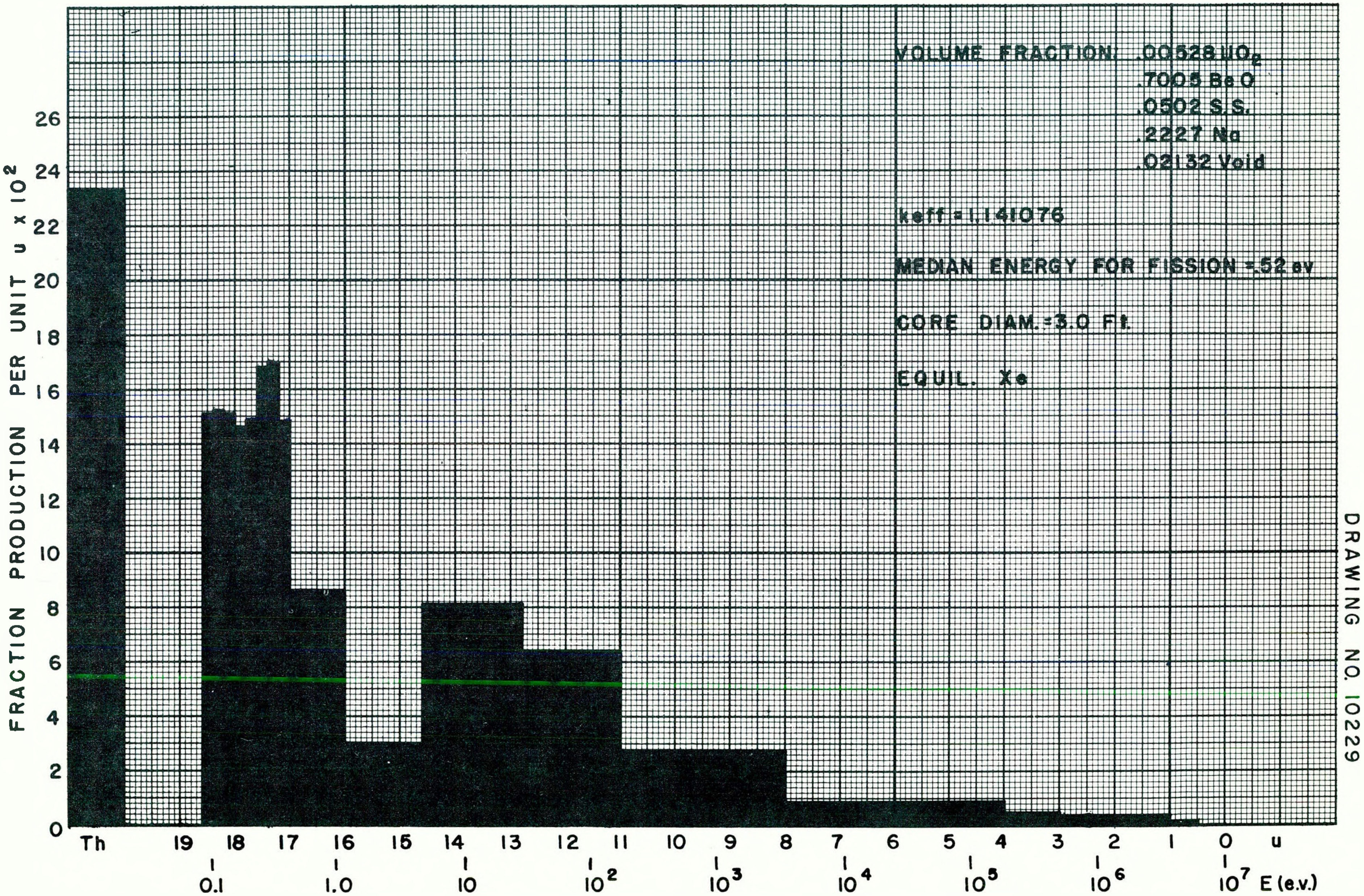

FIGURE 2.13 PRODUCTION SPECTRUM

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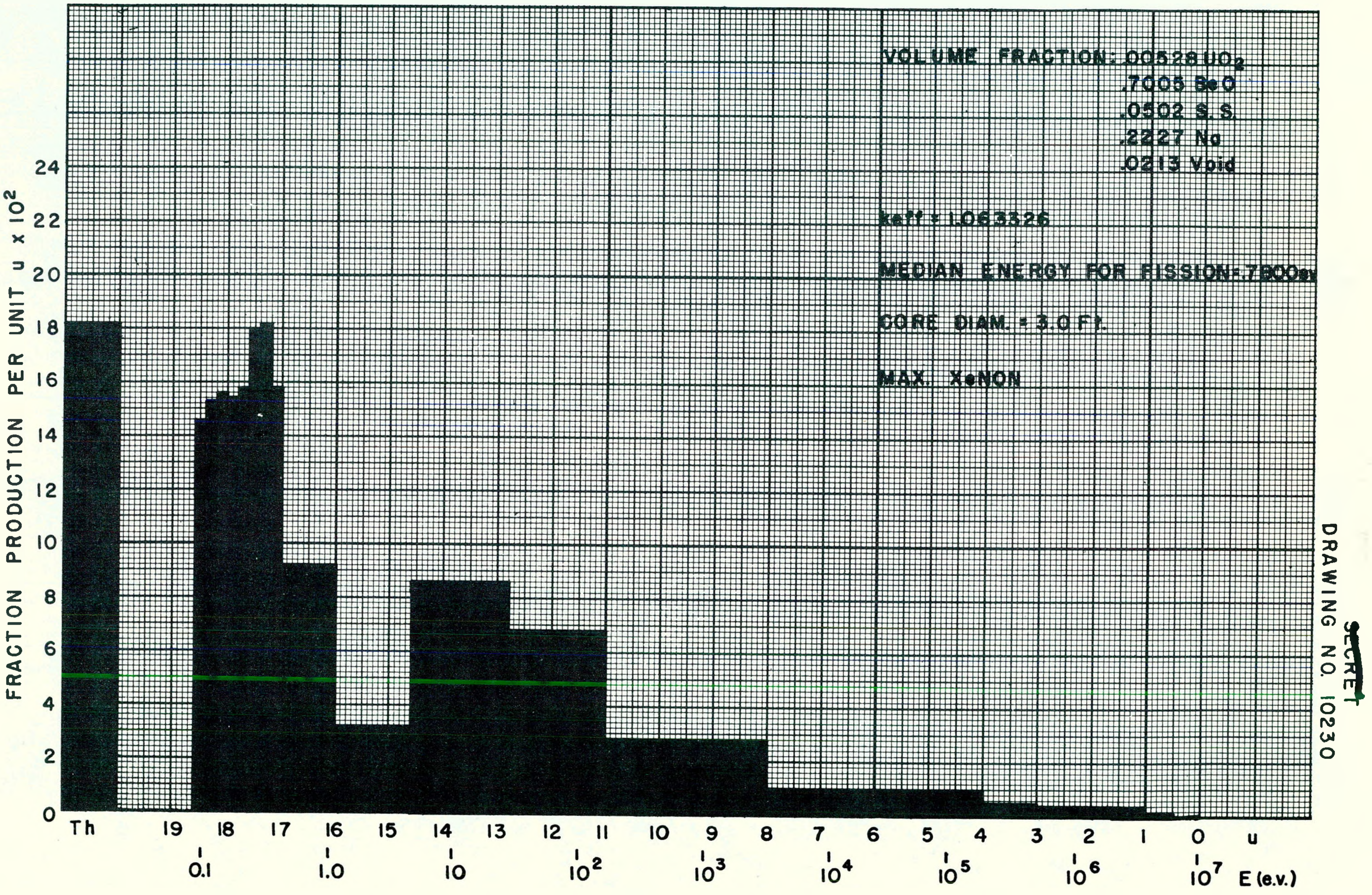

FIGURE 2.14 PRODUCTION SPECTRUM

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not increase linearly with the number of groups, and that, to increase the validity of the general method, it would be acceptable to use 32 groups over the energy range.

Table 2.2 presents the results of a comparison of the two methods for the first six groups of a 13-group reflected reactor. The physical quantities tabulated are the total leakage of neutrons in a group (*E*), the total number of neutrons slowing down from a group (*D*), and the total number of neutrons absorbed in a group (*A*), all normalized to one neutron produced throughout the reactor. A more complete discussion of the linear approximation has been made.(7) A report on the detailed procedure which has now been adopted for multigroup calculations on spherical reflected reactors has been issued.(8)

The complete reactor calculation involves computations to obtain each of the following:

1. Average macroscopic cross sections for the core and reflector from basic cross-section data.
2. Various constants for each group from the average cross sections.
3. Auxiliary functions needed for fitting the boundary conditions in each group.
4. Fluxes and slowing-down densities over the reactor for each group.
5. Net neutron leakage, absorption, and total fissions in each group.
6. Power distribution and the k_{eff} for the reactor.

The setup of all these steps on the IBM machines has been very laborious, requiring, for example, the wiring of 16 electronic control panels and seven supplementary boards.

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TABLE 2.2

SUMMARY OF TEST OF LINEAR METHOD

GROUP NO.		CORE			REFLECTOR	
		G.E. METHOD	LINEAR METHOD		G.E. METHOD	LINEAR METHOD
I	E	0.00484387	0.00484380	E'	0.00293864	0.0029387
	D	0.0180802	0.0180804	D'	0.00190526	0.0019051
	A	0.0000768409	0.00007675	A'	3.81052×10^{-8}	2.858×10^{-8}
II	E	0.0161205	0.0164871	E'	0.00778762	0.00797926
	D	0.103625	0.103246	D'	0.0102378	0.0104133
	A	0.000335745	0.000345174	A'	1.02378×10^{-7}	1.23184×10^{-7}
III	E	0.0924723	0.0926290	E'	0.0535788	0.0533433
	D	0.453414	0.452860	D'	0.0491315	0.0496958
	A	0.00275676	0.00275551	A'	9.82630×10^{-7}	9.01638×10^{-7}
IV	E	0.0483388	0.0480298	E'	0.0237181	0.0243173
	D	0.702250	0.701981	D'	0.0737526	0.0734080
	A	0.00283709	0.00284091	A'	7.37526×10^{-7}	1.23104×10^{-6}
V	E	0.0452667	0.0453560	E'	0.0285398	0.0277470
	D	0.783166	0.782790	D'	0.0904788	0.0910144
	A	0.00382968	0.00382328	A'	9.04788×10^{-7}	8.22112×10^{-7}
VI	E	0.121533	0.122265	E'	0.126826	0.121989
	D	0.636685	0.635922	D'	0.0851816	0.0912816
	A	0.0249708	0.0246217	A'	5.11090×10^{-6}	5.46888×10^{-6}

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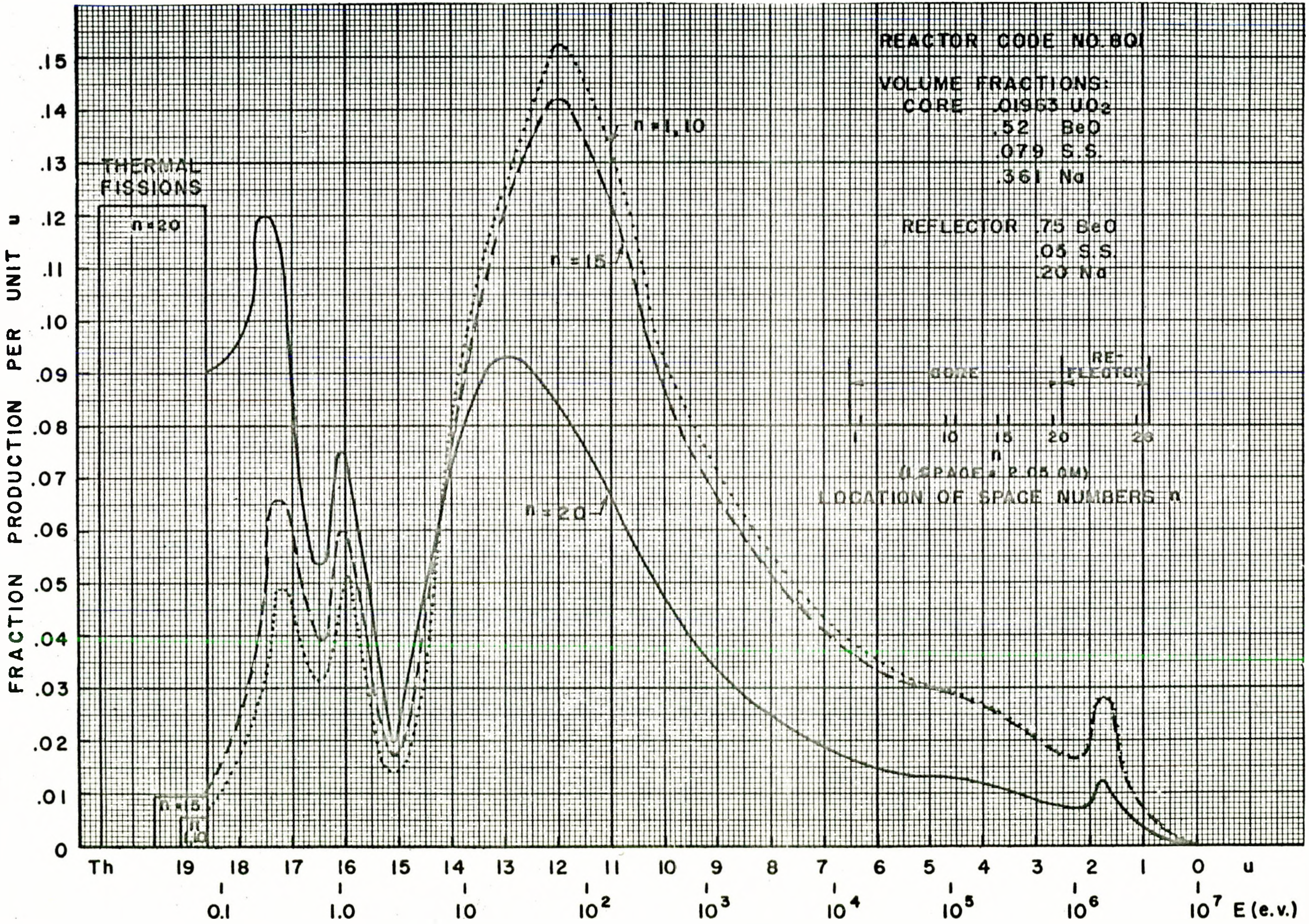

FIGURE 2.17 PRODUCTION SPECTRUM

D. KINETICS OF LIQUID-FUEL REACTORS

Perturbation Calculations(9) (N. M. Smith, Jr.). Perturbation calculations of the kinetic response of a liquid-fuel reactor have been initiated. Such calculations are useful in searching for instabilities, for surveying a broad field for desirable characteristics, and for general self-education into reactor kinetics. Space does not permit the detailed description of the calculations, but these may be obtained from the reference. Furthermore, the physical data on heat transfer, expansion coefficients, and heat capacity of the NaF-BeF$_2$-UF$_4$ systems are not available at the time of the preparation of this report. Hence the results described must be considered largely as illustrative.

The standard method of calculation was followed as used by Nordheim in his first calculations of xenon effects. The present reactor is complicated by having reactivity coefficients associated with the temperature and density of the moderator, the density of the liquid fuel, and, in the high-spectrum reactors, a coefficient associated with the density of the coolant. The

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(9) This describes the beginning of a series of exploratory perturbation calculations to be made on reactor kinetics. The parts will be published separately, with very limited distribution as the calculations proceed and later collected together in one publication with more extensive distribution. The work described here will be included in Y-F10-30 (ANP memorandum at Oak Ridge National Laboratory, Y-12 Site), *Perturbation Equations for the Kinetic Response of a Liquid-fuel Reactor*, by N. M. Smith, Jr., T. Rubin, M. J. Nielsen, and R. R. Coveyou. Copies of any ANP Physics Group report may be obtained by an authorized person by request from the ANP Library.

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thermal relaxation time of the liquid fuel and containing tube is small, with estimates varying, from one design to another, from 0.3 to 0.09 sec; that of the moderator is larger, with estimates from 5 sec to many tens of seconds. The situation is further complicated by the possibility of neutron-flux thermal elastic oscillations of the tube and liquid fuel systems in a single tube, and, since the tubes may be connected by a header at top and bottom of space—modal—neutron flux elastic oscillations. Still further complications arise from the movement of fuel and fission poisons from regions of one importance to another.

Obviously, no attempt is made to consider all these effects simultaneously at first, but, through a series of increasingly realistic approximations, they are added one by one. The reference report contains equations for reactor impedance, including contributions of delayed neutrons, fuel response, moderator response, xenon poisoning, and temperature coefficient and provides for reactor admittance calculations for control rod and for coolant temperature perturbations.

For these constants the temperature increase over the operating temperature of the liquid fuel following a step change in reactivity of 10^{-3} and in the absence of delayed neutrons and xenon effects is shown to be, in degrees centigrade,

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$$\theta_1'(t) = 16.22 + 13.87e^{-0.3597t} - 30.10e^{-0.4422t} \cos (48.54t + 0.002255)$$

This expression is plotted in Fig. 2.20.

The accompanying integrated excess flux is shown to be given by

$$\Phi' = 2.70 \times 10^{13} + 2.02 \times 10^{13}e^{-0.3597t} + 2.59 \times 10^{15}e^{-0.4422t} \sin (48.54t - 0.01818)$$

and is plotted in Fig. 2.21.

Both θ_1' and Φ' are inversely proportional to the neutron lifetime. It is therefore advantageous to have as long a lifetime as possible. The lifetime in a reflected reactor, particularly in a moderating reflector, will be considerably increased over the corresponding value in the "equivalent" bare reactor (i.e., a bare reactor of the same composition as the core of the reflected reactor but increased in size until its reactivity is that of the reflected reactor). The lifetime also is greater in the reactors of lower median fissioning energy. As the reactor is made more nearly thermal, however, the xenon fission poison introduces a positive temperature coefficient of reactivity associated with the moderator.

This positive coefficient is tolerable or not, depending on the thermal relaxation time of the moderator (i.e., the average time required to each equilibrium after a thermal transient). If this time is short compared to the response time of the liquid-fuel-level—temperature-sensing control system, then a positive coefficient of reactivity associated with the moderator is not allowable. However, if this relaxation time is long, a temperature-sensing servo-control system will have adequate time to act. Such a reactor would not be self-regulating in the broad sense but only within a time defined by the moderator thermal relaxation time. Indeed, it is true that there is no meaning to a "completely self-regulating reactor." The so-called "self-regulating features" designed in systems are merely characteristics which allow the requirements of the external servo-control systems to be relaxed to such an extent that control is simple and safe.

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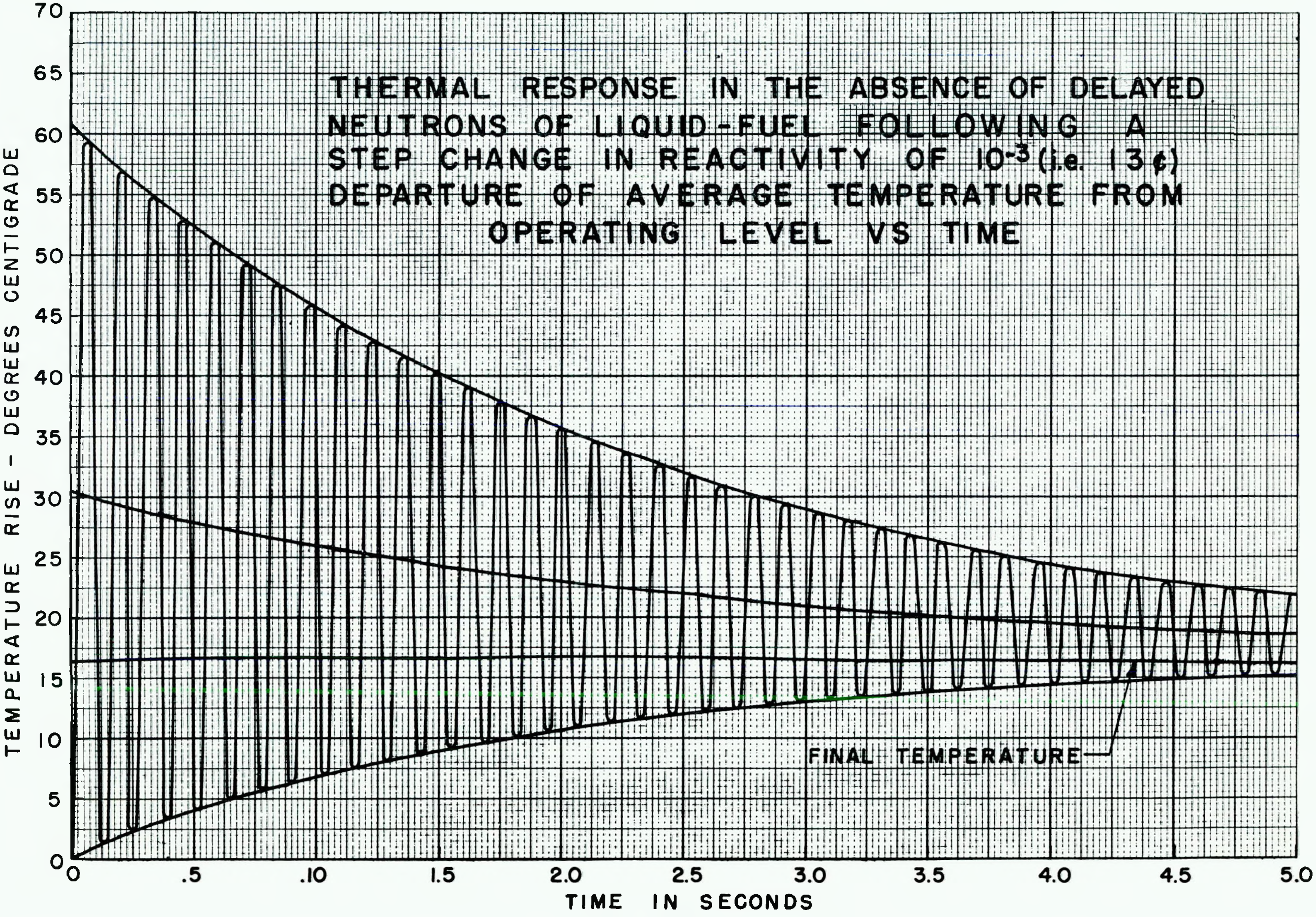

FIGURE 2.20

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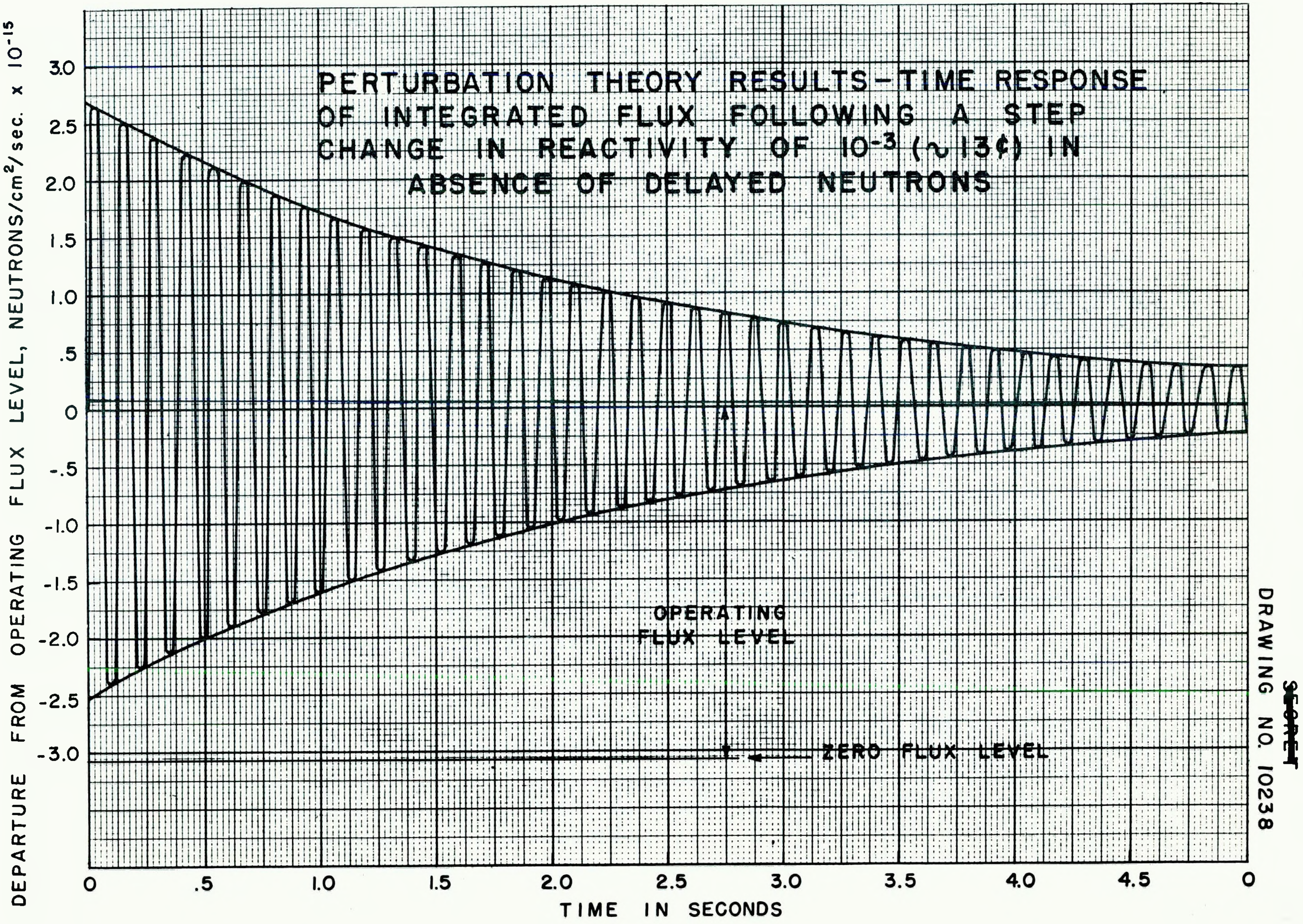

FIGURE 2.21

It is not yet clear what combination of fissioning spectrum, neutron lifetime, xenon effect, etc., will yield an optimum system for control. The discovery of these specifications is the objective of the present course of calculations. It is apparent, however, that the optimum exists and may well be in the direction of the epithermal reactor. Other considerations, such as shield size and weight, may, of course, affect these specifications. The possibility of compensation of a positive xenon coefficient by means of an external mechanical-thermal system exists.

The perturbation calculations are being extended to include:

1. An average group of delayed neutrons.
2. Xenon poisoning.
3. Response to ramp changes of control rod or of entrance coolant.
4. Coolant loop and ramp changes of power demand.
5. Mechanical-elastic response of the liquid-fuel system.

The exact integration of the equations of motion for large perturbations is the subject of a separate investigation.

Thermal Relaxation Time for Fuel Rods (T. Rubin, NEPA). An important quantity in determining the kinetic response of a reactor is the time it takes for a fuel rod to reach temperature equilibrium after having been subjected to a change in flux. This quantity is determined by considering the following heat conduction problem:

A cylinder has finite radius and an initial temperature $v = 0$; heat is produced for time $t > 0$ at the constant rate A_0 per unit volume per unit time; the surface of the cylinder is kept at constant temperature $t = 0$. The solution given(10) is

$$\frac{v}{A_0} = \frac{a^2 - r^2}{4K} - \frac{2}{aK} \sum_{n=1}^{\infty} e^{-k\alpha_n^2 t} \frac{J_0(r\alpha_n)}{\alpha_n^3 J(a\alpha_n)} \tag{1}$$

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(10) Carslaw, H. S., and Jaeger, J. C., *Conduction of Heat in Solids*, p. 277, Oxford, Toronto, 1947.

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where v = temperature of cylinder at radius r and time t

K = thermal conductivity

a = radius of cylinder

k = diffusivity of material; $k = K/\rho c$, where ρ = density and c = specific heat

α_n = roots of the equation $J_0(a\alpha_n) = 0$

The temperature of a cylindrical fuel rod, when subjected to a step increase in power, due to an increase in flux, and whose walls are kept at constant temperature by coolant flow, would respond in the manner given by Eq. (1). Note that changing the boundary condition to $v(t = 0) = a > 0$ and $v(r = a) = b > 0$, where a and b are constants, introduces only slight modifications into the given solution of the problem.

E. BACKGROUND PROBLEMS

Effect on Cross Sections of Atomic Motion* (R. R. Coveyou, Mathematics Panel). In this and succeeding studies methods will be set down for taking into account the effect on various diffusion phenomena of the finite velocity of the atoms of the medium in which neutrons are diffusing. This first study will concern itself with the effect of this motion on values of nuclear cross sections.

Consider a neutron of speed s in collision with an atom of speed t, selected out of a distribution isotropic in direction. Then the relative speed of neutron and atom is given by

$$r^2 = s^2 + 2\lambda st + t^2$$

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* To appear later as part of another report.

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where r is the relative speed and λ is the cosine of the angle between the velocities. Denoting by $P(X)$ the probability of the event described by X, we have

$$
\begin{aligned}
P(r \leqslant x) &= P(r^2 \leqslant x^2) \\
&= P(s^2 + 2\lambda s t + t^2 \leqslant x^2) \\
&= P\left[\lambda \leqslant \frac{x^2 - s^2 - t^2}{2st}\right]
\end{aligned}
$$

By the hypothesis of isotropy, λ is uniformly distributed on $(-1, 1)$. Hence,

$$P_t(r \leqslant x) = \frac{1}{2}\left[1 + \frac{x^2 - s^2 - t^2}{2st}\right], \qquad s - t \leqslant x \leqslant |s + t| \tag{3}$$

or

$$P_t(x \leqslant r \leqslant x + dx) = \frac{x\,dx}{2st}, \qquad |s - t| \leqslant x \leqslant |s + t| \tag{4}$$

Now suppose the distribution of atomic speeds is Maxwellian, i.e.,

$$P(Z \leqslant t \leqslant Z + dZ) = \frac{4\beta^3}{\sqrt{\pi}} Z^2 e^{-\beta^2 Z^2}\, dZ \tag{5}$$

where $\beta^2 = M/2kT$, M is the atomic mass, k is Boltzmann's constant, and T is the absolute temperature.

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Then the distribution of relative speeds "seen" by a neutron of speed S is in this case given by

$$P(x \leqslant r \leqslant x + dx) = \int \frac{x\ dx}{2st} \frac{r\beta^3}{\sqrt{\pi}} t^2 e^{-\beta^2 t^2}\ dt \qquad (6a)$$

where the integration is with respect to t and the limits are given by $|s - t| \leqslant x \leqslant |s + t|$. But this is the condition that x, s, and t form a triangle, hence is symmetric and can be written $|s - x| \leqslant t \leqslant |s + x|$. Thus,

$$P(x \leqslant r \leqslant x + dx) = \frac{\beta x\ dx}{\sqrt{\pi}\ s} \int_{|x-s|}^{|x+s|} (2\beta^2 t\ dt) e^{-\beta^2 t^2}$$

$$= \frac{\beta x\ dx}{\sqrt{\pi}\ s} \left[e^{-\beta^2 (x-s)^2} \quad e^{-\beta^2 (x+s)^2} \right] \qquad (6b)$$

Suppose now that the cross section for a given process is $\sigma(s)$, as a function of relative speed. Let $\sigma^*(s)$ be the effective cross section for this process as a function of neutron speed. Then,

$$\sigma^*(s) = (\beta/\sqrt{\pi}\ s) \int_0^\infty x\ \sigma(x) \left[e^{-\beta^2 (x-s)^2} - e^{-\beta^2 (x+s)^2} \right] dx \qquad (7)$$

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If

$$\sigma(s) = \frac{\sigma_0 s_0}{s} \quad (1/v \text{ absorption})$$

then

$$\sigma^*(s) = \frac{\sigma_0 s_0}{s} \left[\frac{2}{\sqrt{\pi}} \int_0^{\beta s} e^{-u^2}\, du \right]$$

$$= \frac{\sigma_0 s_0}{s} \operatorname{erf}(\beta s)$$

Since erf $(Z) \longrightarrow 1$ as $s \longrightarrow \infty$, this states that the cross section at high neutron velocities is unaffected, as would be expected. If

$$\sigma(s) = \sigma_0 \delta(s - s_0) \quad \{\text{sharp resonance}\}$$

then

$$\sigma^*(s) = \frac{\beta \sigma_0 s_0}{\sqrt{\pi}} \left[\frac{e^{-\beta^2 (s - s_0)^2} - e^{-\beta^2 (s + s_0)^2}}{s} \right]$$

The effective cross-section curves have been calculated for the case of a material of atomic weight 135 and a sharp resonance at 0.0863 ev, at temperatures 300, 1000, and 1500°C (see Fig. 2.24). We note that the maximum of the effective curve always lies below the resonance energy, though the displacement is negligible in the cases considered.

For other cross-section curves $\sigma^*(s)$ can be computed by numerical integration. Existing measurements of cross sections made at room temperature, such as the xenon absorption resonance, can be taken and Eq. (7) can be written as a matrix. Inversion of the matrix by machine computational methods will,

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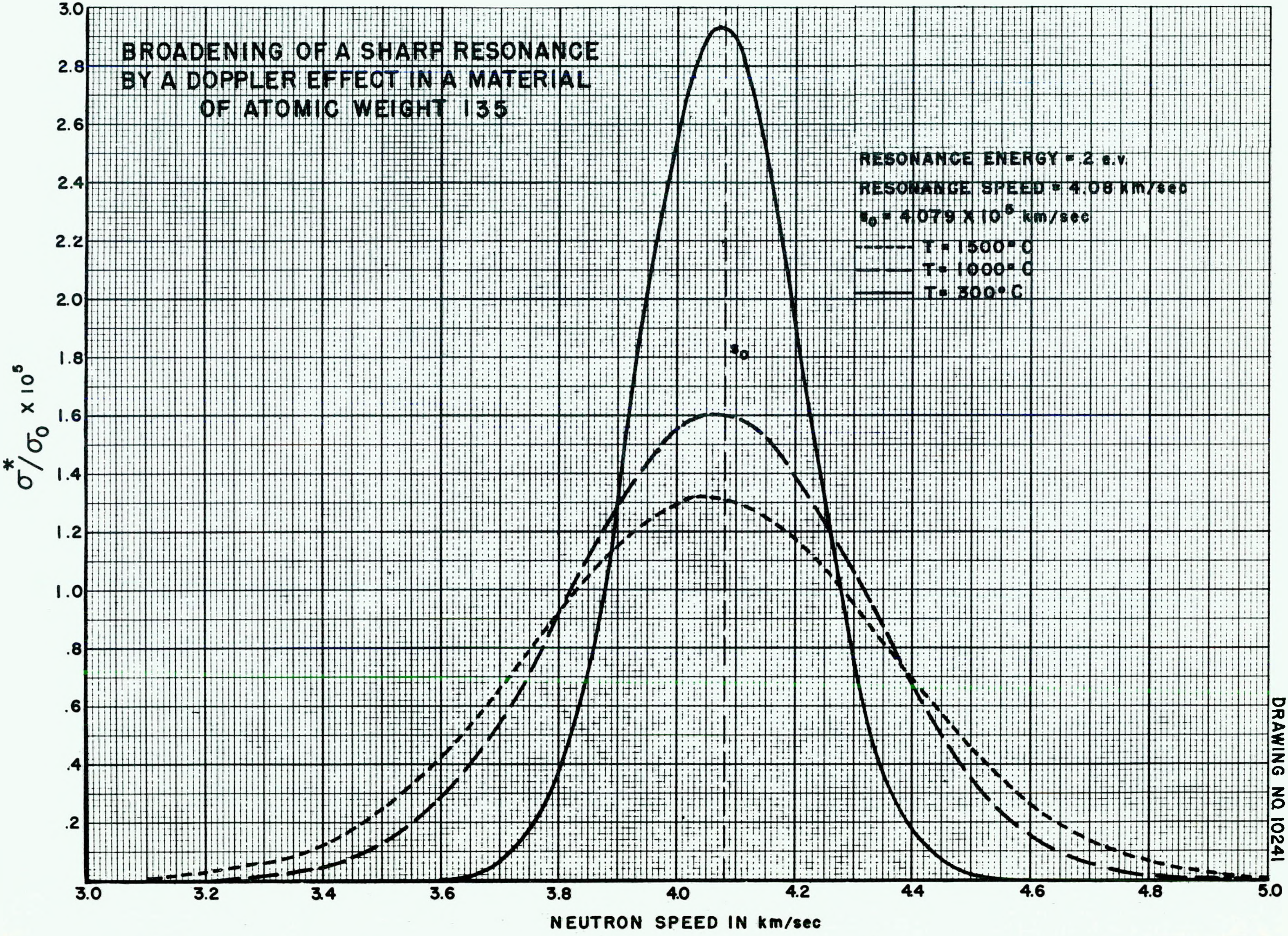

FIGURE 2.24

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in principle, give the true cross section. The xenon resonance, for instance, may be somewhat narrower than that given in the published curves. A narrower xenon resonance would produce a larger positive temperature coefficient of reactivity.

Adjoint Fluxes and Perturbation Theory (M. J. Nielsen, USAF). For each of the family of bare reactors discussed above, the differential equation adjoint to the assumed reactor equation was solved numerically. In solving the adjoint equation, the probability that the ultimate fate of a neutron in the reactor is to cause a fission is computed as a function of energy. This permits the assignment to various neutron groups of an "importance to the reactor" parameter. In addition, using first-order perturbation theory, the change in reactivity resulting from small changes in the pile parameters such as size and fuel content can be computed. (Since most perturbation calculations for the bare reactor involve approximately the same labor as an additional reactor calculation, this use of the adjoint function has so far been very limited.) The adjoint solution also yields a numerical check on the determination of k_{eff}. Calculations of the reactors in the preceding section were checked in this manner.

The lifetimes given above were computed in the standard manner as perturbation calculations by assuming that for $k_{eff} = 1 + \Delta k$, the neutron density as a function of time is given by

$$n(t) = n(0)e^{(\Delta k/l)t}$$

where l is the lifetime. Then

$$\frac{\partial n(t)}{\partial t} = \frac{\Delta k}{l} n(t) = \frac{1}{v}\frac{\Delta k}{l}\phi$$

where $\phi = nv$, the neutron flux. The quantity $\frac{1}{v}\frac{\Delta k}{l}$ is then regarded as a perturbation on the absorption cross section of the pile and the corresponding reactivity change is computed, giving

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$$\Delta k = (\text{constant}) \frac{\Delta k}{l}$$

The constant is in the form of a definite integral which can be evaluated numerically, giving the lifetime directly.

As part of a computational handbook the derivation of the bare-pile adjoint equation and equations for all types of perturbations as integral equations and as difference equations are assembled in report Y-F10-18.(11) Other references are given.(12-14) A compilation of methods planned for multiregion piles is in preparation. The slowing-down density, the importance function, and the statistical weight for the 2.79-ft reactor are shown in Fig. 2.25 as a function of energy. The slowing-down density is the number of neutrons per unit of lethargy slowing down in unit time. The importance function is the net increase in neutron inventory resulting from the introduction of one neutron. The curve labeled "statistical weight" is the product of the other two curves. The ordinate scale is for the importance function.

Cylindrical Multigroup Calculations (M. C. Edlund and T. Rubin). The multigroup age equations have been transformed into difference equations in cylindrical geometry.(15) Hand calculations are now in progress to check the method prior to setting up the problem for the IBM. A satisfactory approximation to the correct boundary conditions for this geometry has not yet been found.

F. CALCULATIONS FOR THE CRITICAL EXPERIMENT

It is planned that the ANP Physics Group will devote considerable time in the future to calculations pertaining to experiments with the critical assemblies. The principal activity will be devoted to interpretation of results from the critical experiments, guiding the experiments so that the results will be useful and interpretable, and using the experiments to check present calculational techniques. Some effort has already been made toward selecting the first few experiments to be of sufficiently simple geometry that a reliable comparison can be made using the IBM calculation.

(11) Nielsen, M. J., *Bare Pile Adjoint Solution*, Oak Ridge National Laboratory, Y-12 Site, Y-F10-18 (Oct. 27, 1950).

(12) Goertzel, G., "Variational Derivation of Adjoint Function and Perturbation Equation," Appendix, p. 13, in *Numerical Integration of Criticality Equations and Use of Perturbation Theory for Bare Intermediate Reactors*, by A. R. Gruber, Oak Ridge National Laboratory, Y-12 Site, TAB-96 (Aug. 15, 1950).

(13) Nordheim, L. W., and Soodak, H., *Application of the Rayleigh-Schroedinger Perturbation Method to the Theory of Piles*, Chicago CP-1638 (June 15, 1944).

(14) Wigner, E. P., *Effect of Small Perturbations on Pile Period*, Chicago CP-G-3048 (June 13, 1945).

(15) Edlund, M. C., *Numerical Integration of the Multigroup Reactor Equations in Cylindrical Geometry*, Oak Ridge National Laboratory, Y-12 Site, Y-F5-22 (Sept. 27, 1950).

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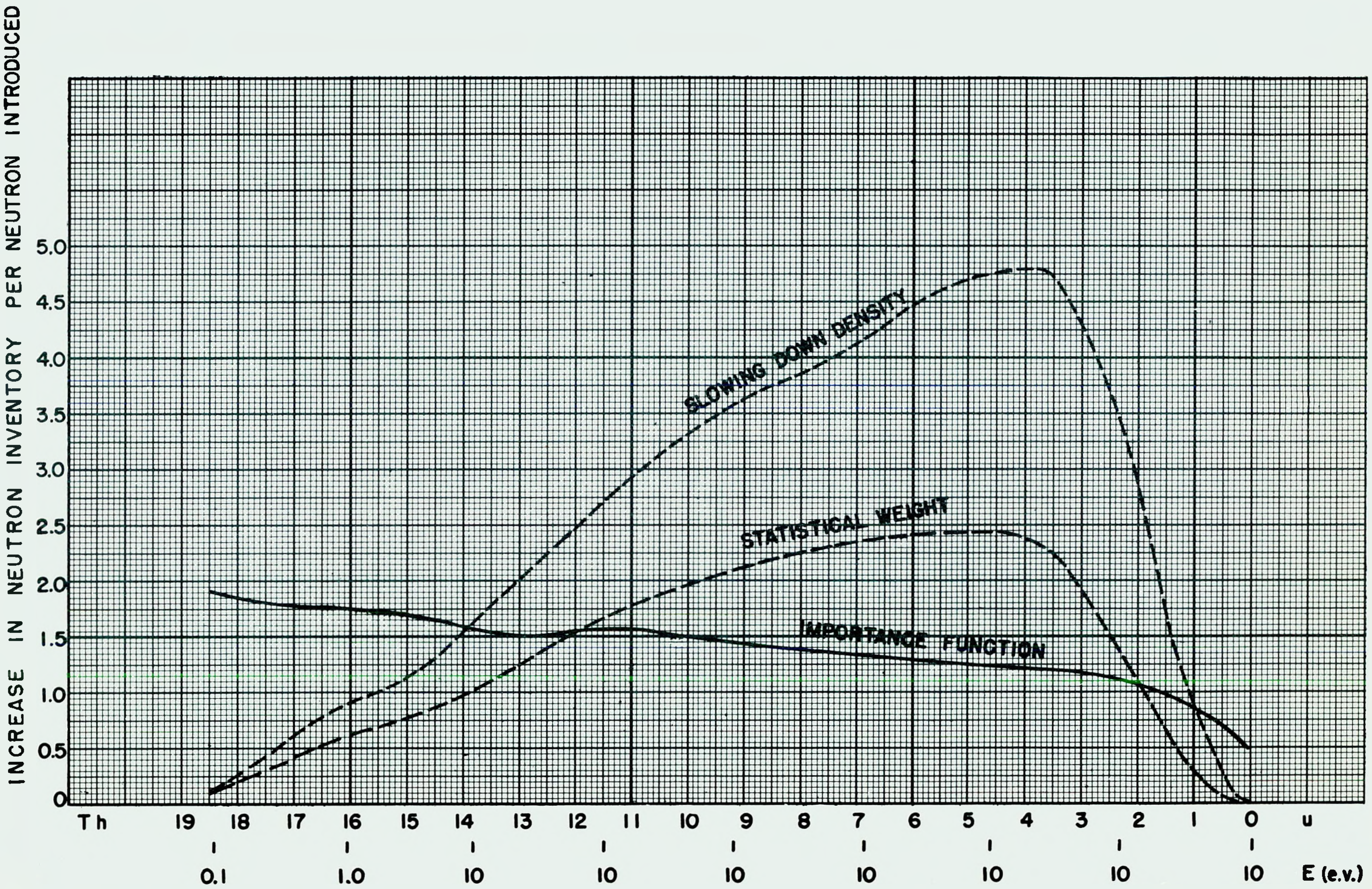

FIGURE 2.25 IMPORTANCE FUNCTION - 2.79 FT. REACTOR

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3. CRITICAL EXPERIMENTS

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3. CRITICAL EXPERIMENTS

A. D. Callihan, Physics Division, and J. F. Coneybear, NEPA

Preparations for joint ORNL-NEPA critical experiments have continued. The installation of equipment in the laboratory is almost complete, the major exceptions being the control and safety rods and the personnel shield. An allotment of uranium has been received and is in the process of fabrication. Some graphite has been received, and the beryllium is being machined. At the time of this writing it was expected that materials and equipment would be ready for the initial experiments about Jan. 1, 1951.

Conferences between ORNL and NEPA personnel have resulted in slight modifications in the initial program described in NEPA-1522. Owing to changes in the ARE design and a desire by the theoretical groups to check the accuracy of calculations, it is likely that the first assembly will be a simple lattice rather than one patterned after a particular ARE reactor. The initial experiments will be done with the moderator and reflector of graphite or beryllium, depending upon the materials on hand at the time the uranium is delivered. The median energy for fissions in the ARE design now considered is between 1 and 100 ev, probably closer to the former. This change has necessitated consideration of Xe^{135} danger-coefficient measurements.

The assembly tables which carry the reactor halves have been installed and are being tested. The aluminum honeycomb structure has been assembled on the tables and fixed into position in an iron framework. The mechanism for inserting a neutron source has been operated. Part of the support for a personnel shield for use during assembly operations has been placed. The control and safety rods have been built, but have not been installed in the assembly because of a delay in obtaining the supporting structure.

The actuating circuits for the table drive and for the control and safety rods have been assembled. Eight radiation-level-measuring circuits, used for safety and for operation, have also been installed. This equipment is being tested.

An allotment of 75 kg of enriched uranium has been made and is now being fabricated into fuel disks 0.01 in. thick. Beryllium is being cast and finish-machined by the Brush Beryllium Corporation. The delivery schedule calls for

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500 lb to be shipped during December, 1950, and the total of approximately 2 tons to be delivered by March 1, 1951. The normal density of AGOT graphite (~1.6 g/cm^3) is being machined by the Y-12 Shop, and partial delivery has been made. Analysis of this graphite indicates acceptable purity.

Samples of the special high-density graphite (~2.0 g/cm^3) have been received but difficulties with the dies have delayed delivery. Analysis after fabrication of test samples indicates acceptable purity.

Since sodium is being considered as the coolant for the ARE it is necessary to investigate its effect in the critical experiments. It may be built into these experiments as NaF or as Na metal. Satisfactory samples of NaF, formed by either hot or warm pressing, have been received from two vendors. One vendor is considering canning sodium in type 304 stainless steel, 0.008 in. thick, and it is believed that the cans may be filled to at least 95% of their volumes.

Stainless steel, type 310, for the heavy-metal reflector studies, is now being fabricated in the Y-12 Shop, and spectrographic analyses have been made of the steel.

Conferences with the NEPA Materials Section concerning danger-coefficient samples have continued. It appears now that samples will be available by the time they are needed.

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4. NUCLEAR MEASUREMENTS

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4. NUCLEAR MEASUREMENTS

The cross-section curve of molybdenum, being determined at Columbia University, indicates a resonance at 49 ev. Consideration is being given to the measurement of the xenon cross section at intermediate energies, and consequently the practicality of the preparation of large xenon sources has been investigated. The chopper time-of-flight velocity selector for operation in this intermediate energy range is still under construction. The 5-Mev Van de Graaff accelerator may be in operation in three or four months.

MECHANICAL VELOCITY SELECTOR

G. Pawlicki and E. C. Smith, Physics Division

The chopper time-of-flight velocity selector for operation in the neutron energy region up to several thousand electron volts is still in the process of design and construction. Design of the rotor is complete, and construction has started in the shops. Construction of the rather elaborate electronic equipment is proceeding satisfactorily in the Instrument Shop. The BF_3-filled ionization chamber to be used for counting the transmitted neutrons is under preliminary test.

MOLYBDENUM CROSS-SECTION MEASUREMENTS

Columbia University*

The cross section of molybdenum is being measured at Columbia University by Prof. W. W. Havens. This measurement is of great interest to reactor engineers since molybdenum has not only exceptional high-temperature strength but also a relatively high resistance to liquid-metal corrosion. The nuclear requirement for aircraft reactors, however, is that the nucleus should have a reasonably low intermediate absorption cross section. A preliminary total cross-section curve, Fig. 4.1, of molybdenum indicates a resonance at 49 ev, with a height of a few barns. Higher resolution work is in progress, and more data are expected shortly.

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* Work performed under contract with the AEC New York Operations Office.

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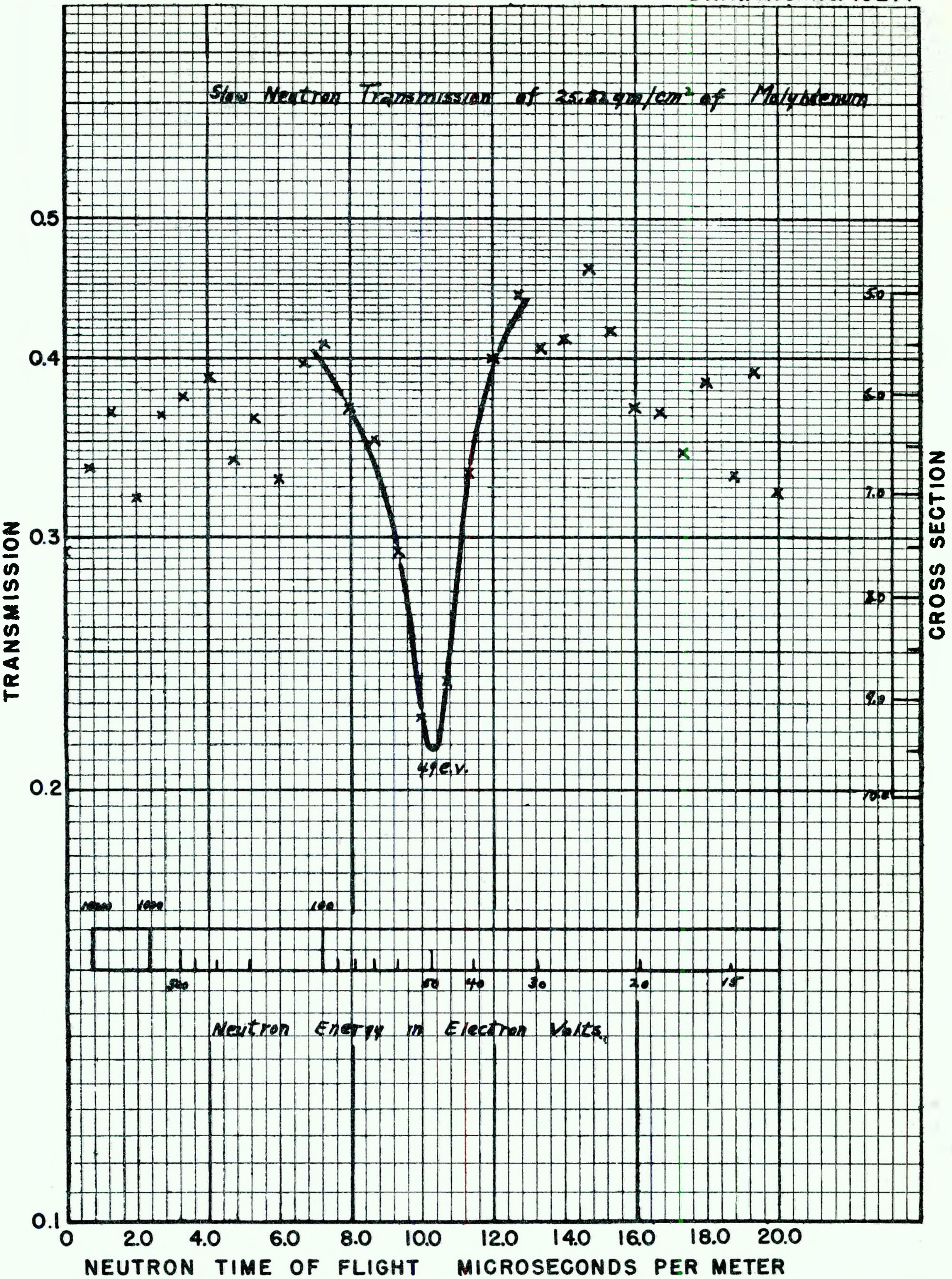

FIG.4.1 PRELIMINARY TOTAL CROSS-SECTION CURVE OF MOLYBDENUM
(Measured at Columbia University by Professor W.W. Havens)

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INTERMEDIATE XENON CROSS-SECTION MEASUREMENTS

During the past quarter some study has been given to the desirability and practicality of measuring the xenon cross section at somewhat higher energies than those reached in the previous ORNL work. Some rough calculations indicate that a danger-coefficient measurement might be possible if 1000 curies of xenon were available.(1) A study was therefore made by the Chemistry Division on the feasibility of preparing a xenon source of this magnitude. As described in the following section, it appears that such an operation would be possible, although it would require work on a rather large scale. However, some theoretical reasons have been advanced(2) for believing that the contribution of possible higher resonances to the resonance integral of xenon will turn out to be negligible in the region of interest for the ARE, compared to the tail of the known xenon absorption band. The matter is, therefore, being considered further before embarking upon such a large experimental program as would be required for complete exploration of this spectrum.

Practicality of Preparing Large Xenon Sources.(3) A xenon source of sufficiently high level may probably be realized from each of the following three reactors:

1. Low Intensity Training Reactor (LITR)
2. Hot Pilot Plant
3. Homogeneous Reactor Experiment (HRE)

The parameters pertinent to the production of xenon are summarized in Table 4.1 for each of these reactors.

Of the three methods for the production of xenon, the minimum yield, in curies, from the Hot Pilot Plant would be lower than desired, whereas that from the LITR should be adequate, and the yield from the HRE should be more than ample. In regard to availability, the production—essentially the collection—of xenon could probably be arranged to fit in with the current program of the Hot Pilot Plant. Neither the LITR nor HRE is now in operation,

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(1) Weinberg, A. M., *Resonance Absorption by Xe*, CF 50-8-116 (Aug. 29, 1950).

(2) Arfken, G. B., Jr., and Welton, T. A., *Estimates of Effects of Higher Levels on the Resonance Integrals of Xe^{135}*, CF 50-12-25 (Dec. 6, 1950).

(3) From a letter of Dec. 15, 1950, from G. W. Parker to C. B. Ellis *re* high-level sources of 9.2-hr xenon, CF 50-12-45.

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TABLE 4.1

High-level Sources of 9.2-hr Xenon

REACTOR	ESTIMATED POWER (kw)	PORTION AVAILABLE	HALFLIVES TO PROCESS	AMOUNT AT SATURATION (curies)	MAXIMUM YIELD (curies)
Hot Pilot Plant (Clinton Pile)	0.2/slug	100 slugs	2*	1000	200
Low-intensity Training Reactor	300	1 of 18 elements	0.5**	1000	750
Homogeneous Reactor Experiment	1000	All	1***	40,000	20,000

* Type of process for Hot Pilot Plant slugs: (1) dissolution in HNO_3; (2) bulk gas fractionation.

** Type of process for LITR element: (1) melting of unit in vacuum; (2) minor gas fractionation.

*** Type of process for HRE off-gas: (1) collection of 24-hr gas sample; (2) minor gas fractionation.

although preparations for placing the LITR in operation are proceeding rapidly. The HRE is expected to operate at low level in mid-1951, and full-level operation may possibly be reached by the end of 1951.

(1) *Low Intensity Training Reactor.* While the operation of the Low Intensity Training Reactor (LITR) has not yet been definitely settled with the Safeguards Committee, preparations for placing the reactor in operation are proceeding rapidly. A shield estimated to weigh at least 8 tons is necessary to safely handle a green fuel assembly of the size and activity required to produce the xenon. Since such a shield would be generally useful for other purposes, it has already been designed and is expected to be built soon.

It is anticipated that the element would be removed from the reactor after a short cooling period and transported quickly to the laboratory working area, where the xenon could be collected and purified by a relatively small charcoal bed and fractionating column. With a flux of 5 × 10^{12} neutrons per second the time after shutdown for maximum xenon growth has been calculated(4) as 4.2 hr, which is adequate for the required operations.

The amount of xenon available would be about 1 kilocurie at 300 kw at whatever flux at which the LITR may be operated. The processing time after the 4-hr growth period could be short, and the yield might therefore be as high as 80 to 90%.

(2) *Hot Pilot Plant.* The equipment (scrubbers, condenser, etc). now being installed in the Hot Pilot Plant for the absorption of the 10-year krypton of mass 85 from the off-gas stream of the dissolver operation will also concentrate xenon, although the latter is cooled to a stable gas in the process. The Pilot Plant may be mentioned as a possible source of 9.2-hr xenon if and when the problem of handling large quantities of oxides of nitrogen and foreign gases can be overcome. Conceivably, a fair portion of a 100-slug charge could be dissolved in 4 to 6 hr, and if the time for fractionation were not excessive, yields of 200 or more curies might be expected.

(3) *Homogeneous Reactor.* The most attractive prospect for future large-scale xenon work is the Homogeneous Reactor Experiment (HRE), from which a continuous flow of hydrogen, oxygen, and water vapor will sweep out kilocurie quantities of xenon and krypton. The off-gas treatment system now proposed includes a large charcoal bed, through which 40,000 curies of xenon will eventually be discharged after around 100 days of holdup for decay. In order to obtain similar quantities of xenon in a portable container, it would be necessary only to pass the off gas through a shielded charcoal cold trap and collect samples of the mixed gases which could then be purified. The equilibrium xenon activity of 40,000 curies will give off heat at the rate of 500 watts, thereby introducing an extra requirement for handling.

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(4) Lane, J. A., *Xenon Poisoning at Varying Power Levels*, CF-49-12-83 (Dec. 14, 1949).

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THE 5-MEV VAN DE GRAAFF ACCELERATOR

W. M. Good, Physics Division, and Conway Snyder, NEPA

A 5-Mev Van de Graaff accelerator, constructed by the High Voltage Engineering Corporation of Cambridge, Mass., has been purchased by NEPA. The final design has been approved, and delivery is expected in the immediate future. Upon receipt the accelerator will be erected in the Y-12 area. It is expected that a period of three months will be required to place it in operation, following which research will begin on shielding, radiation damage, and other problems in nuclear research pertinent to the nuclear propulsion of aircraft.

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6. EXPERIMENTAL ENGINEERING

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6. EXPERIMENTAL ENGINEERING

H. W. Savage, ANP Division

Implementation of the Experimental Engineering Laboratory in Building 9201-3 at Y-12 has continued throughout the past quarter. Operation of some liquid-metal systems has been possible since mid-September with continuous round-the-clock operation after October 23. All experiments so far performed have tested the compatibility of materials with hot liquid sodium, [redacted] and preliminary consideration is being given to systems for sodium hydroxide and for mixed sodium and lithium fluorides. Effort is now shifting toward more complex systems comparable to those anticipated in an ARE installation.

Equipment has been designed and is now being fabricated or installed for inserting test samples of materials in either dynamic or static hot liquid metal for corrosion, erosion, self-welding, and stress-corrosion tests. In certain equipment it will also be possible to move exposed materials with relation to each other, and also to develop adequate pressure, temperature, and flow measuring devices. Pumps, valves, flanges, joints, insulation, instruments, and purification equipment are being developed, and the techniques of pretreating materials and postexperimental examination of used items are being learned.

An adequate power source for electric heating of loops is in process of installation, hoods for adequate ventilation are being fabricated, electromagnetic pumps are on order, tubing for loops is on order, and a small machine shop for experimental fabrication is being installed.

Considerable attention is being given to the hazards involved in handling the coolants proposed, and the Safety Committee, formed during this quarter, is in process of developing adequate protective clothing, fire-fighting equipment, and disposal and cleaning equipment.

Corrosion Tests; Harps (W. C. Tunnell). Corrosion effects of liquid sodium, flowing because of convection, on various metals are being examined in "harps." Each harp is run to failure or for 1000 hr minimum, and the changes

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in the metal and the sodium are determined by metallographic and spectrographic analysis.

Operation of the harps filled with liquid sodium has been performed for the ANP Metallurgy Group under mutually agreed-upon specifications, and has required the development of techniques of cleaning, welding, installation, loading, instrumentation, pressurizing, and analysis of used materials. Much of the equipment was obtained from the Metallurgy Group and revised in Y-12 shops to fit local needs. Ten harps were placed in operation for a 1000-hr continuous test on Oct 23, 1950, and seven others have been placed in operation since. Table 6.1 summarizes the types tested and the characteristics of operation through November 30.

In preparing the first harps for use, it was found that some of the welding had been done without shielding the arc with inert gas and that acid cleaning attached slag inclusions in these welds and resulted in many leaks. Rewelding with heliarc remedied these defects, and cleaning was suspended later since it was suspected to be also attacking the primary metal. It has been necessary to revise the design somewhat, and a procedure was developed to assure consistent treatment, vacuum tightness, and noninclusion of unwanted metals. Degreasing is accomplished with trisodium phosphate; oxide removal appears best accomplished with sodium, since acid pickling often damages the original material although inhibitors may alleviate this situation; and water is removed by a methanol and ether wash and subsequent heating under vacuum.

Samples of the sodium used in each harp are taken at the start and termination of each run and submitted for oxygen and spectrographic analysis. Table 6.2 summarizes the determinations made through November 30. The technique used is patterned after practices at General Electric Co., a sample being drawn into a double-bulbed tube to minimize oxygen contamination. In general the results indicate more oxygen present than can exist in solution in the metal. This may be due to incomplete filtration, consequently the advantages of a 5- instead of a 10-micron filter will be determined.

It appears to be particularly important in preparing any structure for test to have full control of all details of the fabricating process to assure a complete history of the treatment and types of materials included.

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TABLE 6.1

Harp Operations Summary (Liquid Sodium)

HARP NO.	MATERIAL	OPERATING TEMPERATURE (°F)	TOTAL TIME TO 12/1/50 (hr)	REMARKS
1	316 stainless steel	1350	764	
2	316 stainless steel	1350	850	
3	316 stainless steel	1500	767	
4	316 stainless steel	1500	625	Failed in middle of hot leg
5	304 stainless steel	1500	349	Failed in top of hot leg below cup
6	L-605 alloy	1500	688	
7	347 stainless steel	1500	795	
8	Low-carbon iron*	1200	579.3	Failed in middle of hot leg; excessive warpage and oxidation outside
9	Nickel	1500	0	Failed; did not reach operating temperature
10	Low-carbon iron	1200	25.4	Failed in side of bottom cup
11	Nickel	1500	13	Failed in top plate of bottom cup
12	Nickel	1500	164.8	Heater failure, appeared to be leak; has been refilled and reoperated
13	Nickel	1500	223.5	Failure in bottom plate of bottom cup
14	347 stainless steel	1350	725	
15	347 stainless steel	1350	894	
16	304 stainless steel	1350	850	
17	304 stainless steel	1350	890	

* See Fig. 6.1

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TABLE 6.2

Sodium Samples from Harps

HARP NO.	O_2 IN Na (wt %)				SPECTROGRAPHIC ANALYSIS (μg/g of Na)					HARP MATERIAL
	AT FILLING		AT TERMINATION							
	Upper	Lower	Upper	Lower	Mn	Ni	Cr	Fe	Mo	
1	0.092	0.082								316 stainless steel
2	0.56	0.13								316 stainless steel
3	0.067	0.087								316 stainless steel
4	0.086		*0.025 **0.039	0.012 0.032	*<10 **<10	<40 <35	<20 <20	50 35	<4 <5	316 stainless steel
5	0.032									304 stainless steel
6	0.038	0.023								L-605 alloy
7	0.020	0.013								347 stainless steel
8	0.021	0.017								Low-carbon iron
9	0.033	0.018								Nickel
10	0.047	0.025	0.025	0.025						Low-carbon iron
11	0.033	0.063				4100 (vertical leg) 245 (bottom pot)				Nickel
12	0.044	0.012	0.016	0.009						Nickel
13	0.027	0.026				8500 (vertical leg) 3300 (bottom leg)				Nickel
14	0.037	0.090								347 stainless steel
15	0.025	0.019								347 stainless steel
16	0.037	0.051								304 stainless steel
17	0.052	0.016								304 stainless steel

* Harp 4 is still in operation; this sample was taken after 12 hr of operation.

** This sample was taken after 14 hr of operation.

Harp operation has been relatively trouble-free except for the final failures and plugging of gas lines by condensation of the sodium. The latter is serious in that it prevents release of the system pressure when a leak occurs, thereby increasing the hazard. No complete solution to the problem has been found.

Failures appear to have occurred at welds exclusively—some completely unexpected because of invisible tube joints. Complete metallographic examination is being performed in the Metallurgy Laboratory. One of the more extreme failures (harp 8) is illustrated in Fig. 6.1.

In addition to the harps, two other convection loops are being operated. These are different from harps in that the liquid metal can be drained from the loop and the liquid is held in position by inert gas pressure without requiring valves. Testing of the device for holding the liquid level automatically is one of the objectives of operating this loop.

Figure-eight Loop (W. C. Tunnell). The figure-eight loop is designed to receive test sections for dynamic corrosion and erosion tests, self-welding tests, and stress-corrosion tests. One such loop has been delivered and is being installed (Fig. 6.2). Seamless tubing was unavailable, consequently satisfactory operation at very high temperatures is not anticipated.

An a-c electromagnetic pump patterned after a General Electric pump was designed for use with this loop As a part of this problem it was desirable to join copper connectors to the stainless steel pump cell without using soldering alloys and fluxes. A satisfactory joint was made using a type 316 stainless steel rod and building a boss on the stainless steel from the rod to a thickness of about 1/8 in. and heliarc-welding the boss to the copper. The joint was cooled very slowly to prevent cracking.

Test sections for the figure-eight loop are shown in Figs. 6.3 through 6.6. Figure 6.3 shows the method of holding test samples (part 4) in a high-velocity stream for corrosion and erosion studies. Figure 6.4 shows the method of inserting two abutting pieces in the stream with the attachments for applying force to aggravate self-welding by holding the abutting surfaces together under known pressure. Figure 6.5 shows a test piece in the stream subjected to a tensile force and corrosion and erosion effects simultaneously. Figure 6.6 shows details of the foregoing test sections and another modification (item 3)

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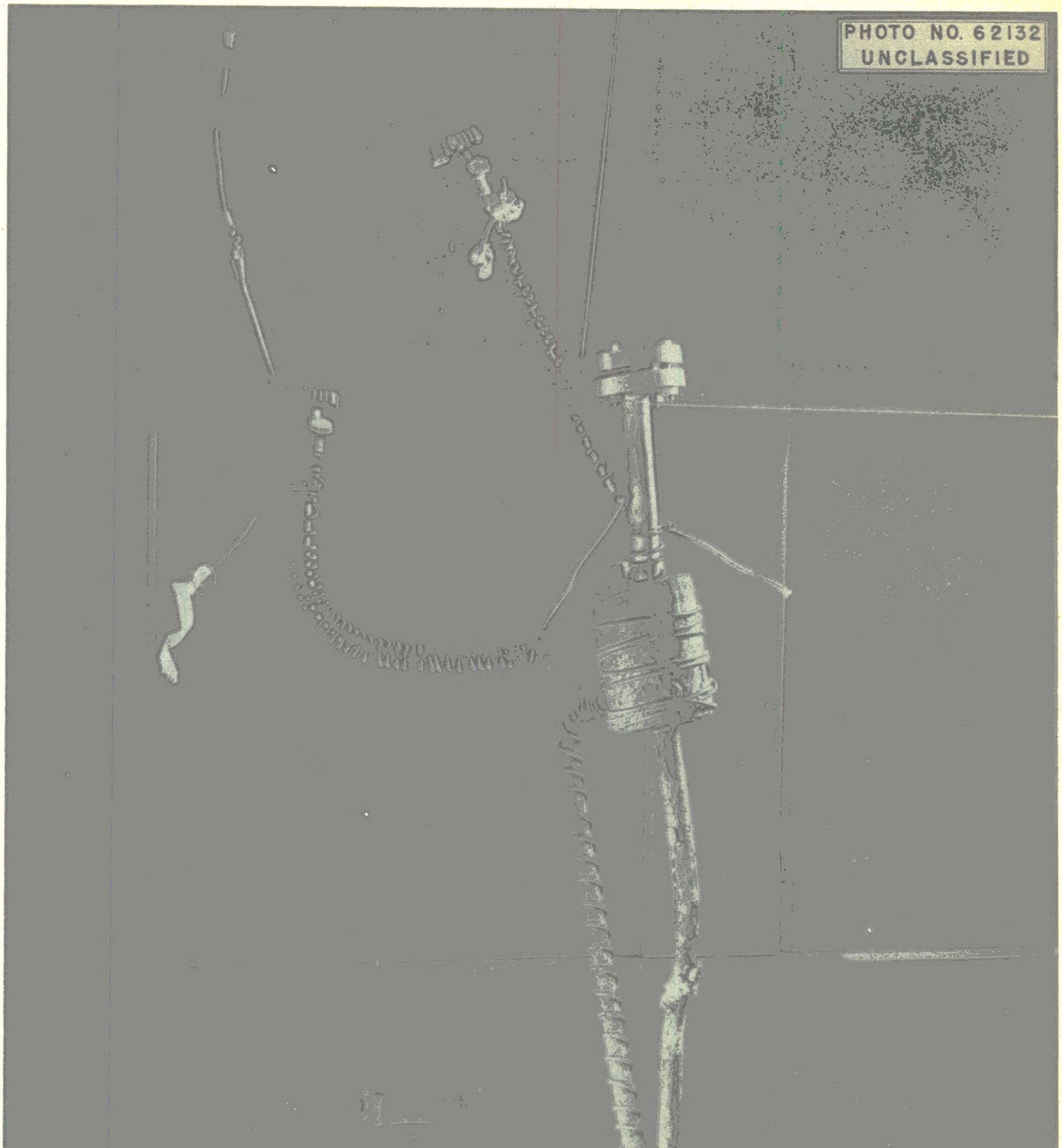

FIGURE 6.1 LOW CARBON IRON HARP AFTER FAILURE

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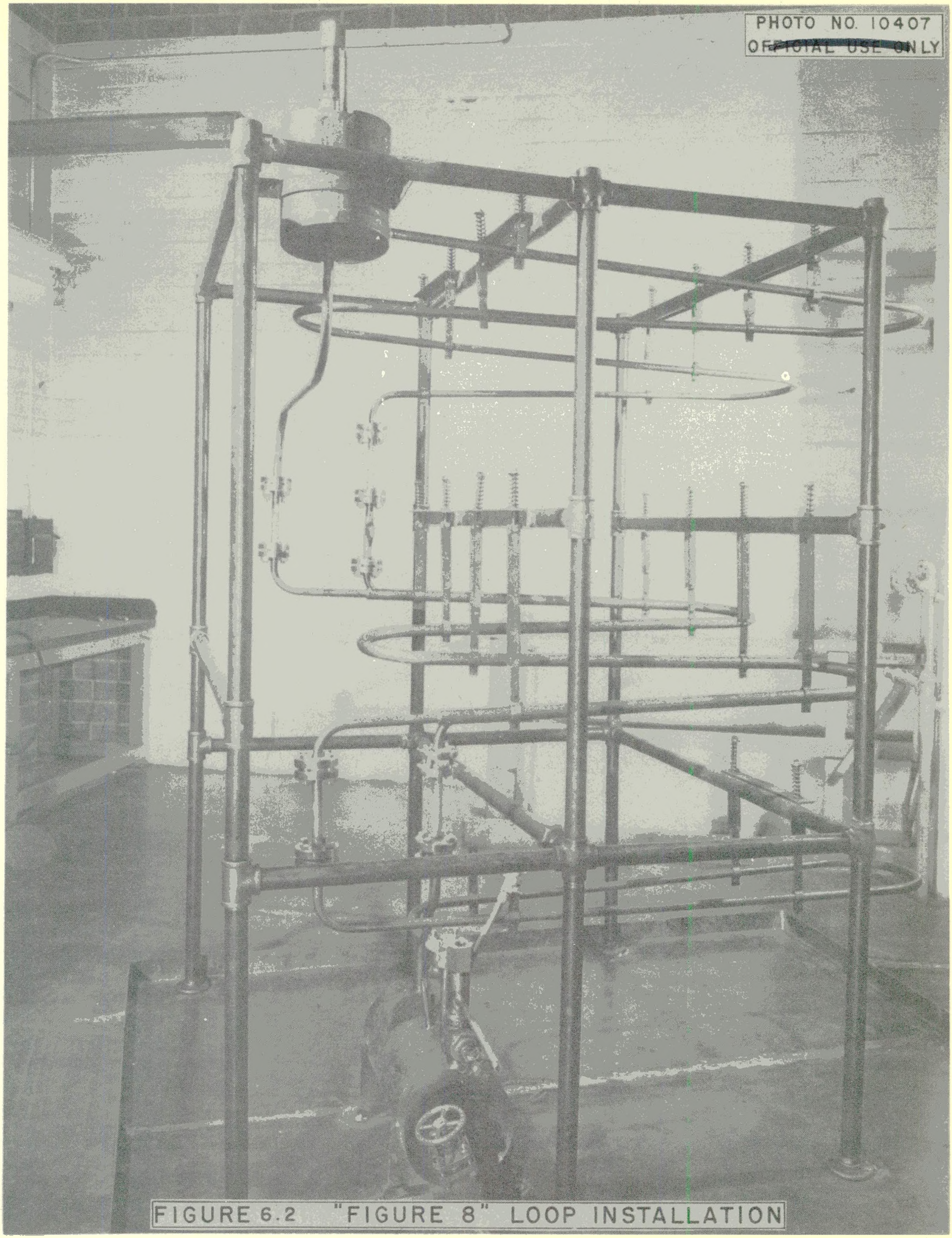

FIGURE 6.2 "FIGURE 8" LOOP INSTALLATION

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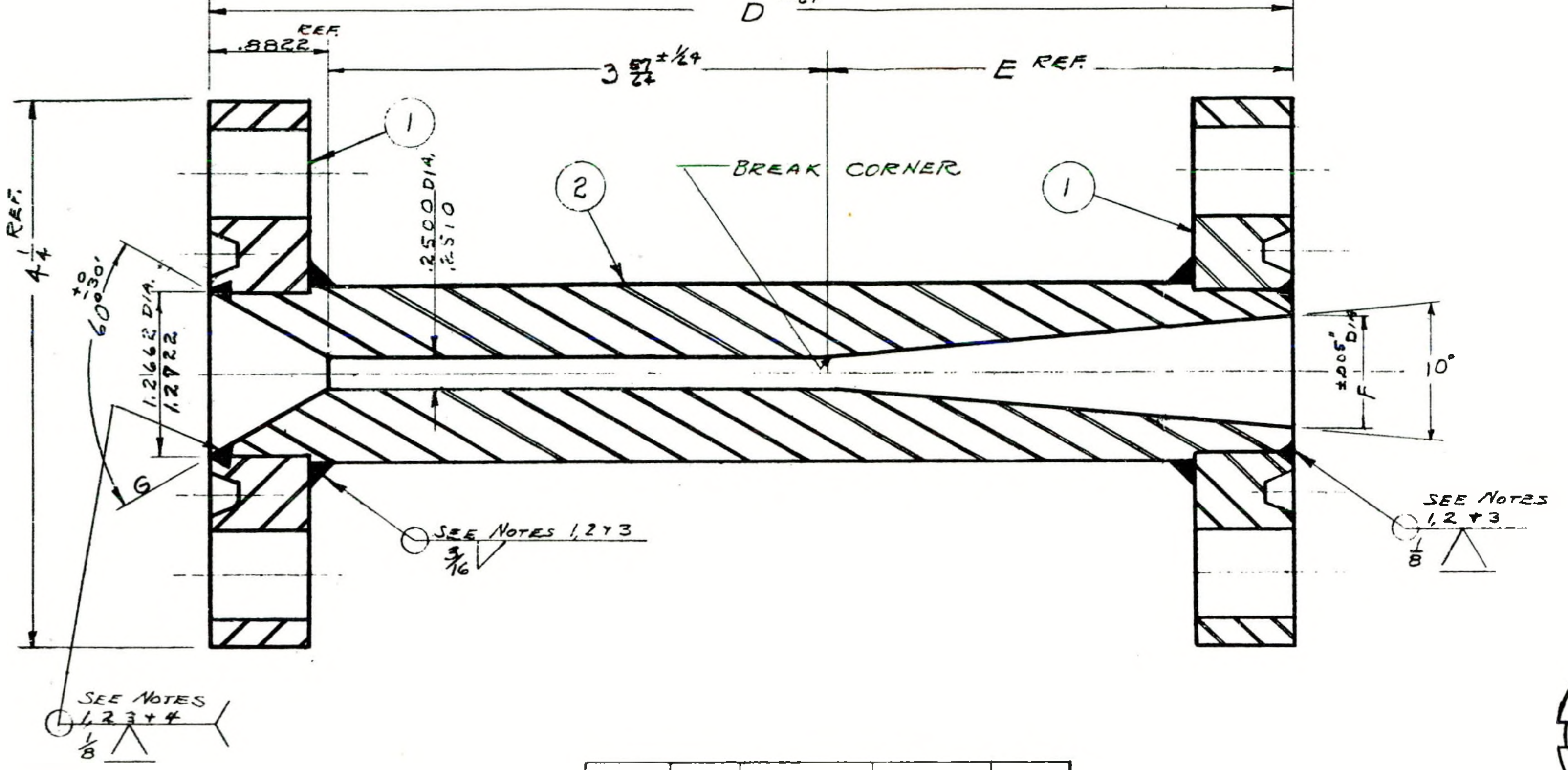
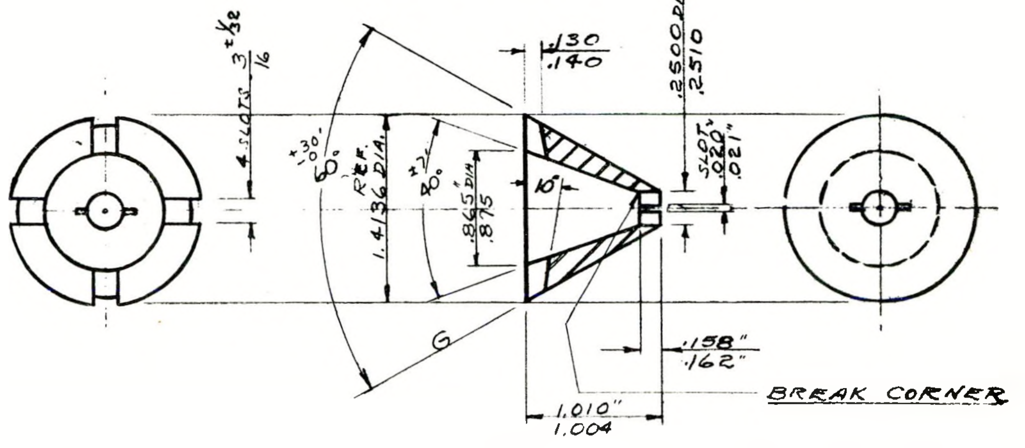


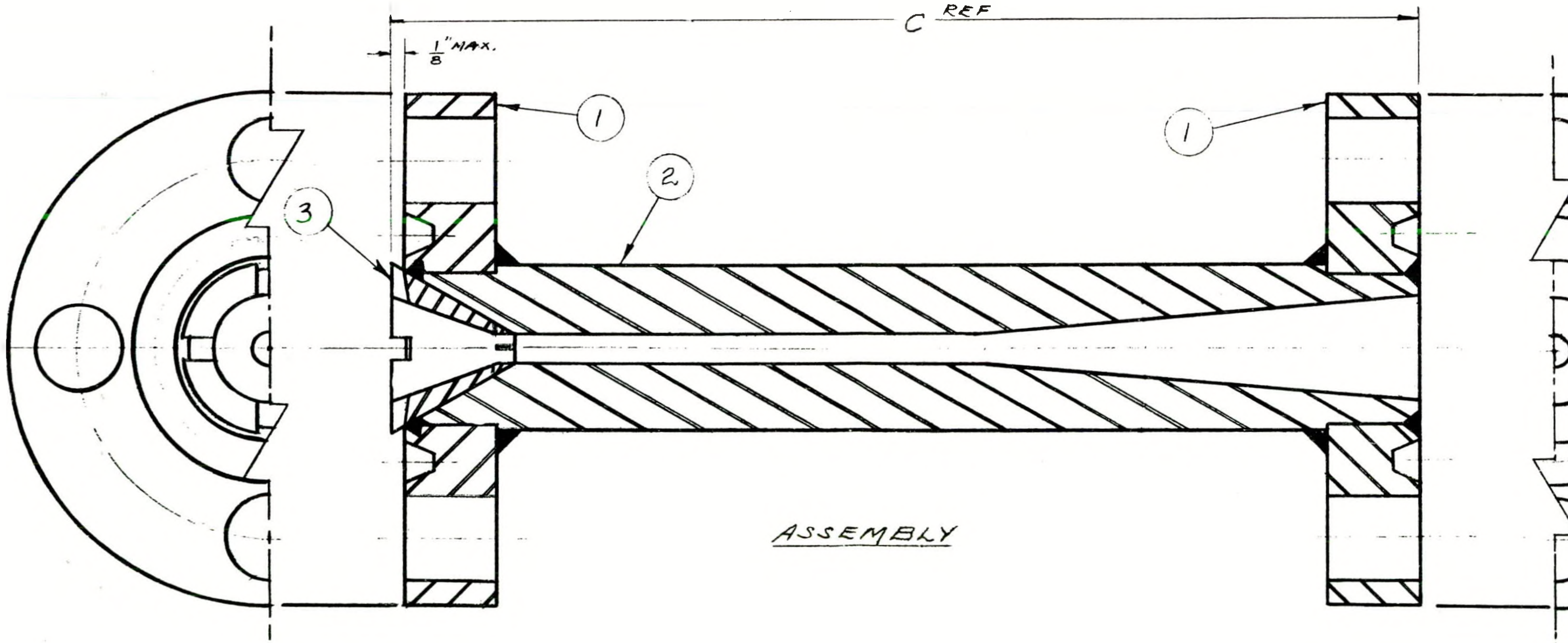
NOTES:

1. ALL MATERIAL MUST BE DEGREASED BEFORE WELDING.
2. ALL WELDING TO BE DONE BY THE HELI-ARC PROCESS WITH INERT GAS IN THE PIPE
3. ALL WELDS MUST BE VACUUM TIGHT AND TESTED.
4. THE ABOVE MATERIAL FOR ONE UNIT ONLY, TYPE A.
5. MATERIAL IS SAME FOR TYPE "B" EXCEPT LENGTH OF ② IS 6 13/16



PART No. ③

TYPE	C	D	E	F
C"	8 7/16"	8 5/16"	3.5404	.870
"D"	6 21/32	6 11/32	1.750	.557



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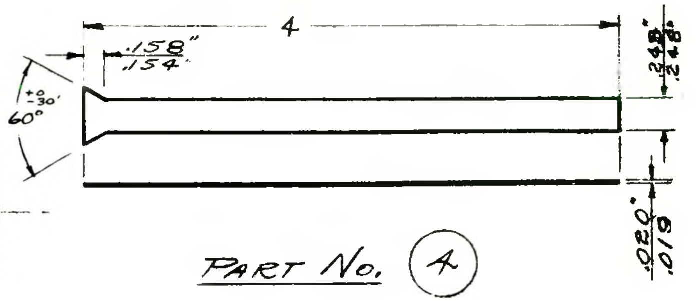

PART No. ④

FIGURE 6.3 TEST SECTION FOR CORROSION TESTS IN "FIGURE 8" LOOP

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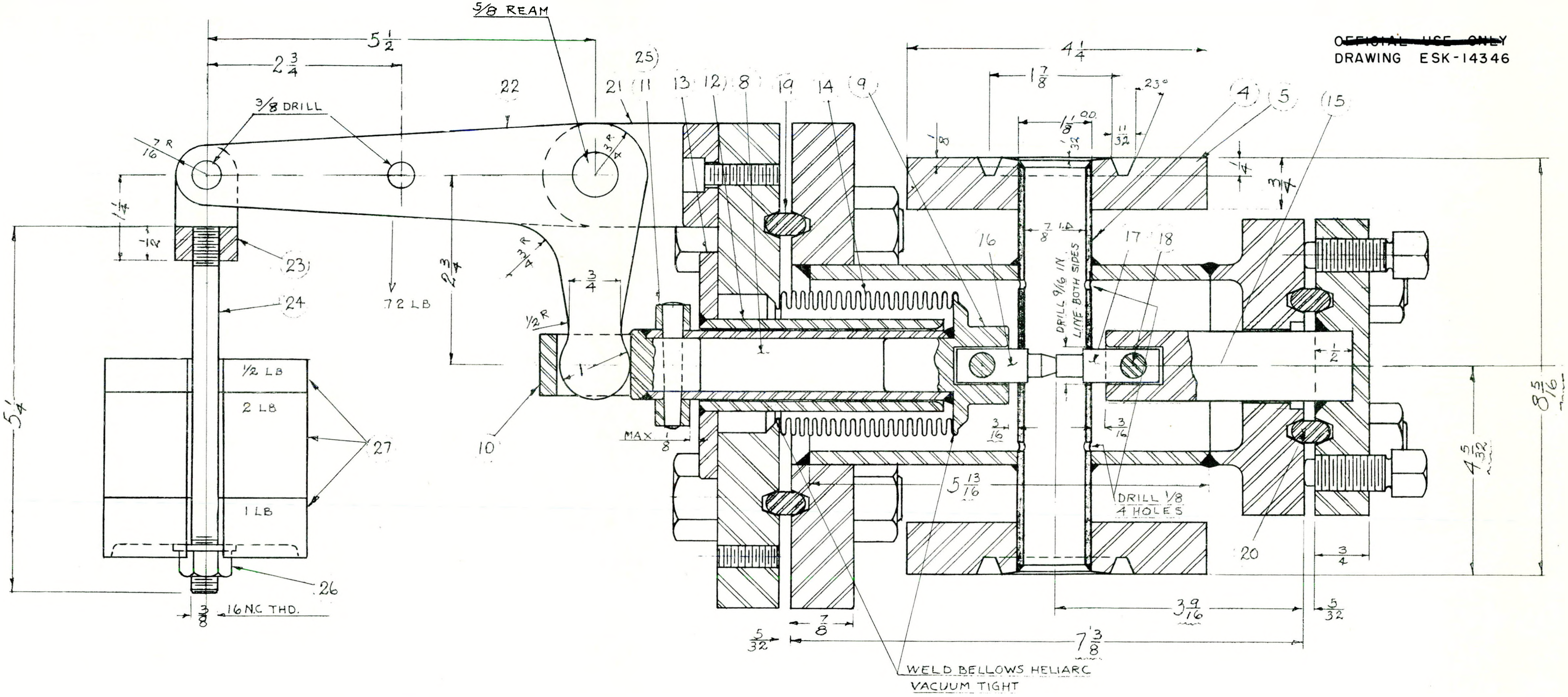

FIGURE 6.4 TEST SECTION FOR SELF-WELDING TESTS IN "FIGURE 8" LOOP

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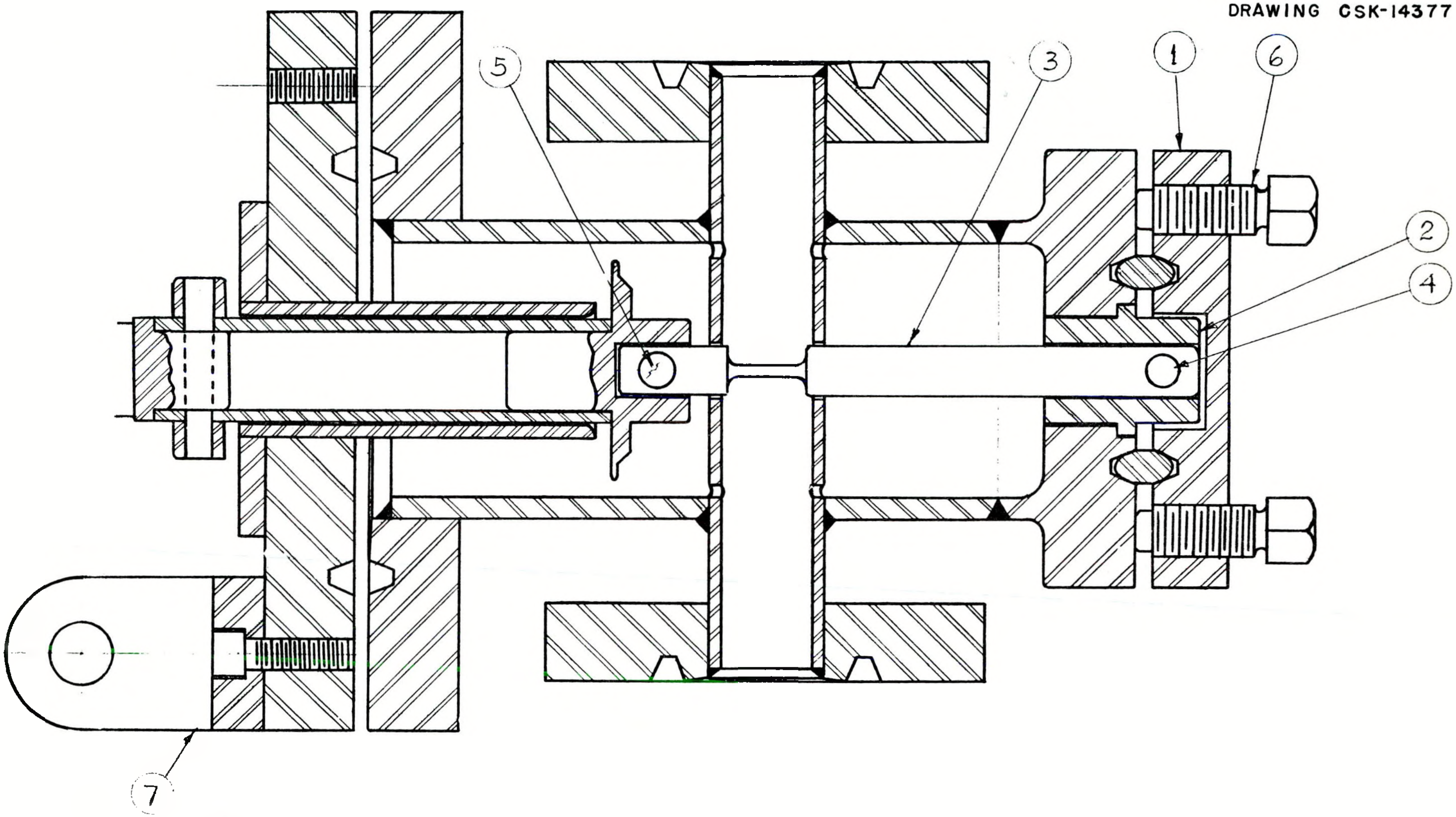

FIGURE 6.5 TEST SECTION FOR STRESS-CORROSION TESTS IN "FIGURE 8" LOOP

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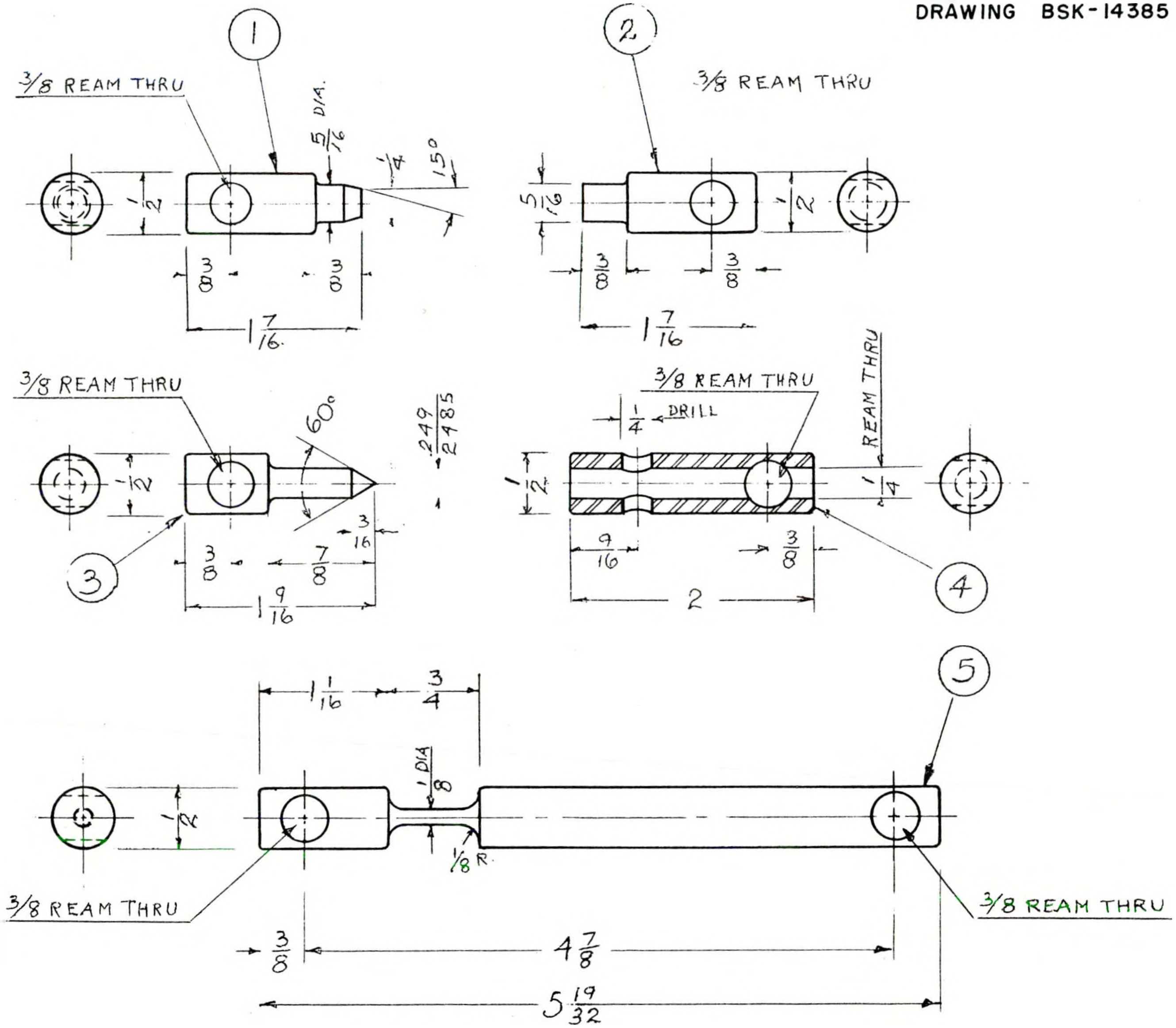

FIGURE 6.6 DETAILS OF PARTS TO BE TESTED IN "FIGURE 8" TEST SECTION

in which two pieces can be moved axially one inside the other, so that while in the stream welding tendencies of oscillating parts can be investigated.

Calibration Loop (W. G. Cobb). An isothermal loop with electromagnetic pump, sump tank, and expansion tank has been designed for the ready insertion of devices for measuring pressures and flow rates (Fig. 6.7). Two venturi meters calibrated on water will be used as standards.

Seal Test Device (J. F. Haines). A device for testing materials to be used as seals is shown in Fig. 6.8. It consists of a sump for molten metal, a vertical overhung shaft, and a holder. One element of the sealing materials is a ring (9) held in the holder and sealed by gaskets to prevent passage of the liquid from outside the holder to the inside. The other element of the seal is a cup (8) attached to the end of the shaft so that it can be rotated; it bears on the stationary ring (9) to form the seal. This type of seal is quite common in other branches of engineering practice, and the effectiveness of the seal in liquid sodium will be determined during operation by visual observation of the well around the shaft through sight glasses.

Bearing Tests (J. F. Haines). A thrust bearing adaptable to the seal test device has been designed, and a radial journal bearing is being designed.

Air Tests (H. Turner). An 800 cfm blower has been set up for the testing of models of the heat exchanger and for testing fuel element configurations for head losses and flow characteristics. Auxiliaries being built for this setup include entrance and exit conduits, entrance orifices for measuring and regulating flow, and a multiple manometer board for pressure determinations. Models for test are being designed, the configurations for which are based on current ARE design concepts.

Insulation Tests (R. T. Schomer). Because of the volume of thermal insulation anticipated for the liquid-metal circulating systems, a number of different types have been subjected to hot liquid sodium to determine the most practical type. Present evaluation indicates the use of a combination insulation composed of an inner layer of either Superex or Hi-Temp No. 19 thick enough to bring the interface temperature down to 1200°F, followed by a layer of lead-mill or steel-mill slag wool thick enough to give the desired heat loss from the pipe. For small pipe (about 2 in.) at 1800°F these thicknesses will

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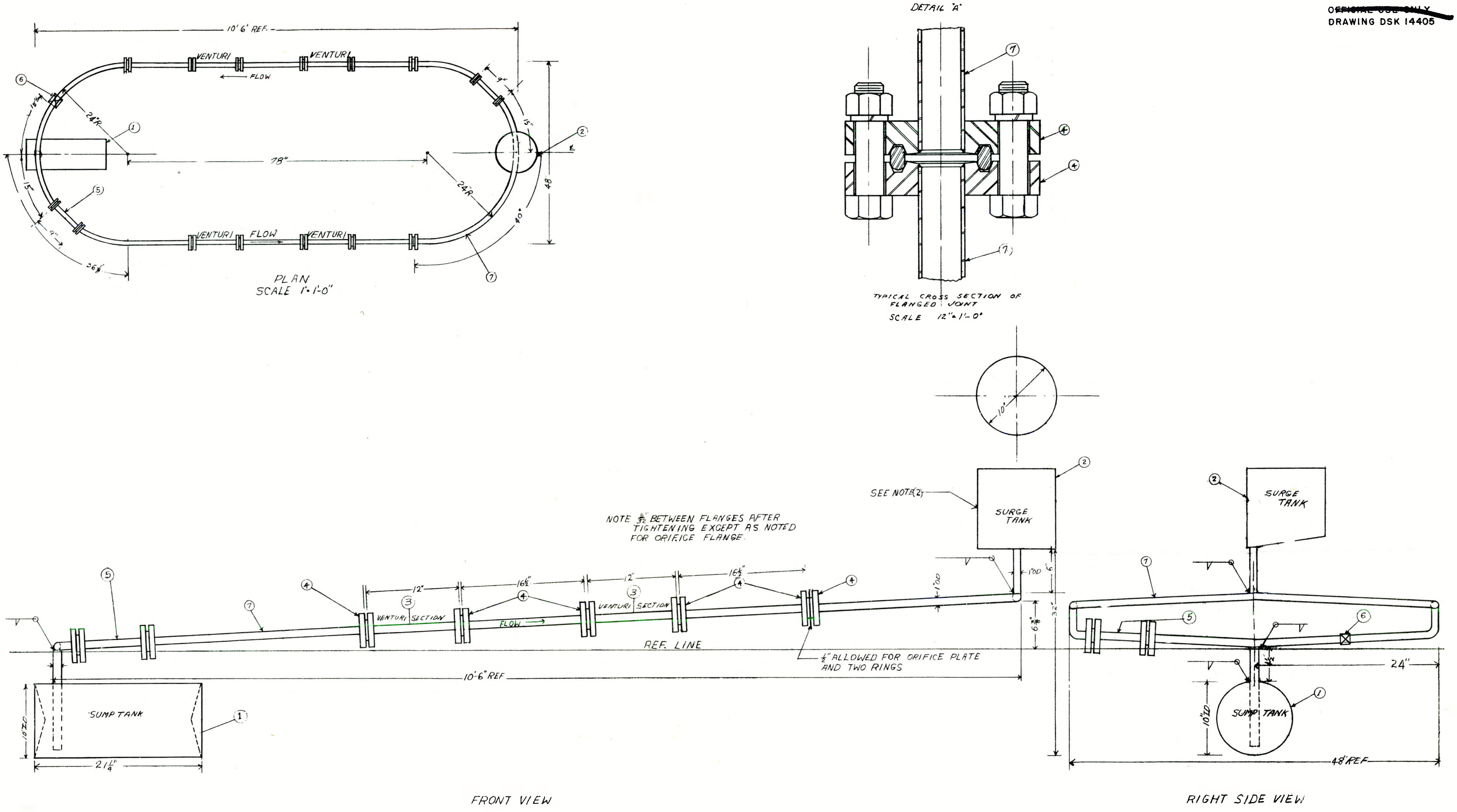

FIGURE 6.7 CALIBRATION LOOP

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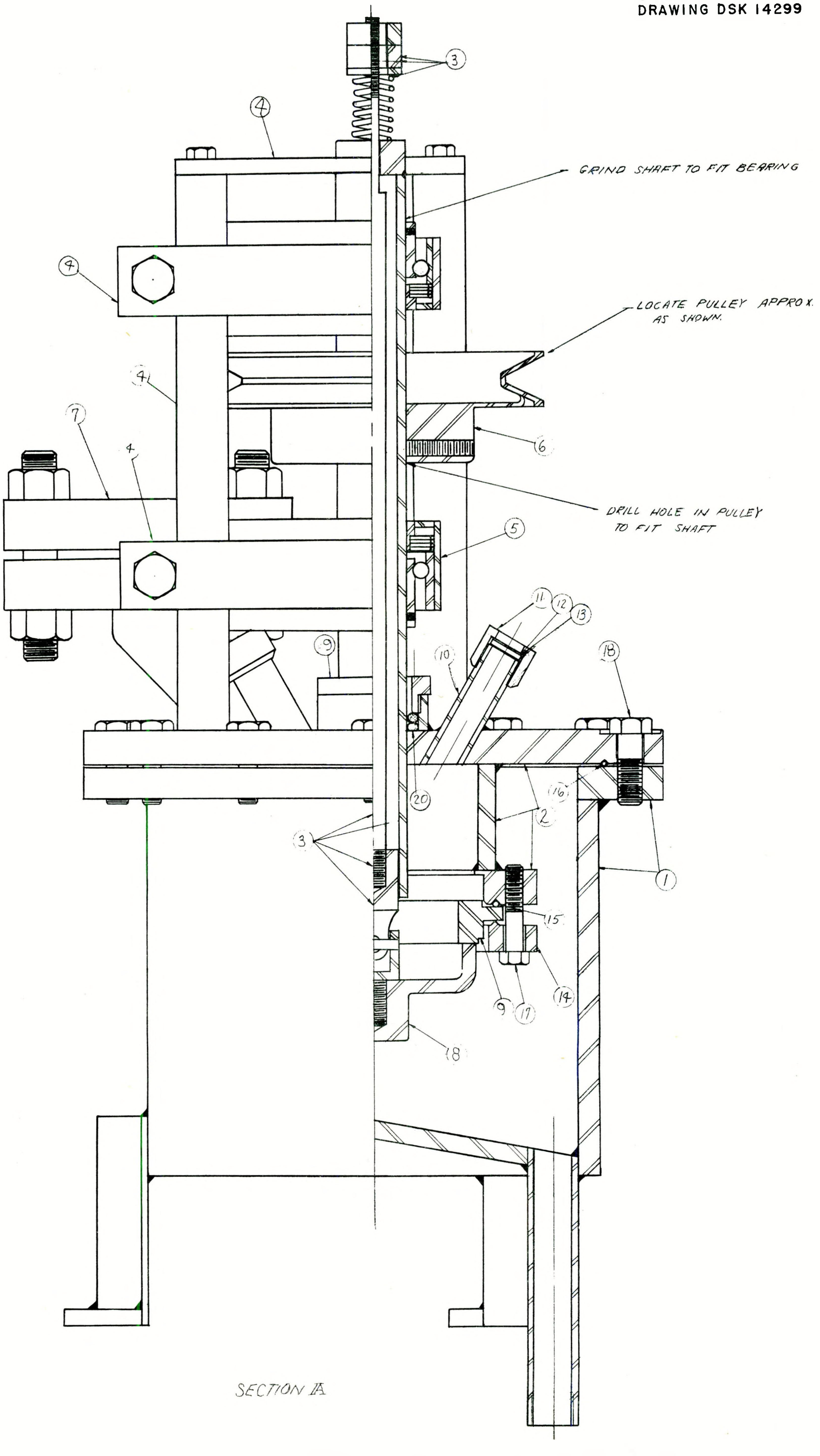

FIGURE 6.8 SEAL TESTING DEVICE

be about 2½ and 3 in., respectively. The basic difficulty with most insulating materials is reaction with the sodium. Table 6.3 summarizes the tests made using the equipment shown in Fig. 6.9.

TABLE 6.3

Insulation Tests

TYPE	PRIMARY CONSTITUENT	PIPE TEMPERATURE (°F)	RESULTS
Johns-Manville RF 300 Thermoflex	Asbestos	1372	Disintegrated
85% Magnesia	MgO	1372	Reacted with Na
Philip-Carey Hi-Temp No. 19	Diatomaceous silica	1100	Gradual hardening; max. penetration ½ in.
Johns-Manville Superex	Diatomaceous silica	1100	Gradual hardening; max. penetration ½ in.
Eagle-Picher DE-85 Block	Diatomaceous earth	1100	Gradual hardening; max. penetration ¾ in.
Baldwin-Hill Neoblock	Lead-mill slag	1100	Caked and crumbly; penetration 7/8 in.
Owens-Selvins Kaylo Block	Lead-silica	1100	Caked and crumbly; penetration 1 in.
Friedrick & Derrick Glass Wool	Glass wool	1100	Fused to steel; penetration ¼ in.
Baldwin-Hill Black Rockwool	Lead-mill slag	Not heated	Penetration less than ¼ in.
Philip-Carey Gray Wool Blanket	Lead-mill slag	Not heated	Penetration less than ¼ in.
Philip-Carey White Wool Blanket Asbestos Board	Blast furnace slag 1 asbestos	Not heated	Violent reaction
Johns-Manville Bauroc Blanket	Steel—mill slag	Not heated	Stuck to metal; penetration less than ¼ in.

Tests of screening of various meshes were also made to determine penetration of a jet of sodium. Single layers of stainless steel screening of 30, 100, and 200 mesh and galvanized screening of 16 mesh did not stop a jet of sodium, whereas a double layer of 200-mesh stainless steel screening was effective.

Pumps (J. F. Haines, W. G. Cobb, and A. P. Fraas). A program of pump development has awaited the definition of types desired as well as the determination of seal and bearing materials. Progress on the latter items has been

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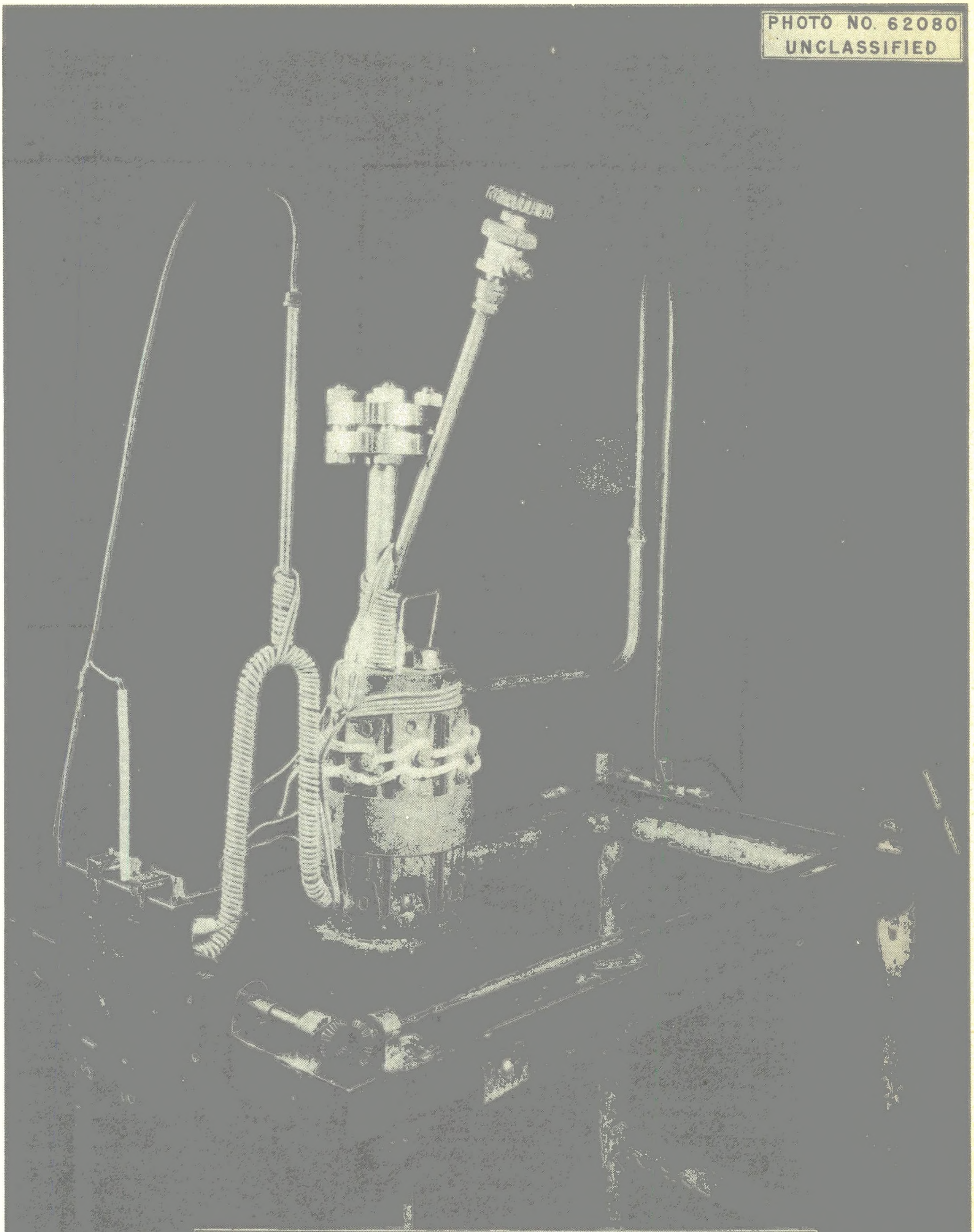

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FIGURE 6.9 INSULATION TESTING DEVICE

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discussed above.

Both centrifugal and axial flow pumps are now being considered in ARE design, and a program of testing is being set up. In this connection a centrifugal pump has been modified, as shown in Fig. 6.10, to have vanes on the back face of the impeller and a heat dam in the shaft. Initial trial of this pump with water failed for mechanical reasons, and the structure is being repaired. The test stand is shown in Fig. 6.11.

Electromagnetic pumps have been considered adequate for the small-scale test loops and one has been fabricated, some are on order, and one new one has been received. Pump development has been extensively surveyed at Allis-Chalmers Corporation, Milwaukee, Wisconsin, and proposals have been submitted by them. The so-called "centrifugal electromagnetic" pump is being considered as a design possibly suitable for ARE use.

It has been agreed with NEPA to set up a loop by July 1, 1951, capable of testing a 1600-gpm pump to be supplied by them.

Purification (P. L. Hill, USAF). The sodium presently being used in the convection harp tests and to be used in future figure-eight tests is received in dry pack form and is freed of oxygen by filtering at about 250 to 300°F in the process of filling the test apparatus. According to the experience of G.E., this low-temperature filtration removes oxygen to a residual concentration of approximately 10 ppm. The filter medium is a sodium oxide cake deposited on a 10-micron pore size Micrometallic 316 stainless steel filter. No attempt is made to remove trace elements from the sodium as received from DuPont, a typical analysis of which is

Na	99.9%	S	15-20 ppm
B	1-2 ppm	Mg	1 ppm
Ca	110-350 ppm	Ni	2-4 ppm
Cl	190 ppm	P	10-50 ppm
Fe	5-10 ppm		

Purification methods for lead, potassium, sodium and potassium hydroxides, and mixed fluorides are being studied with no decision as to the most suitable method having been reached.

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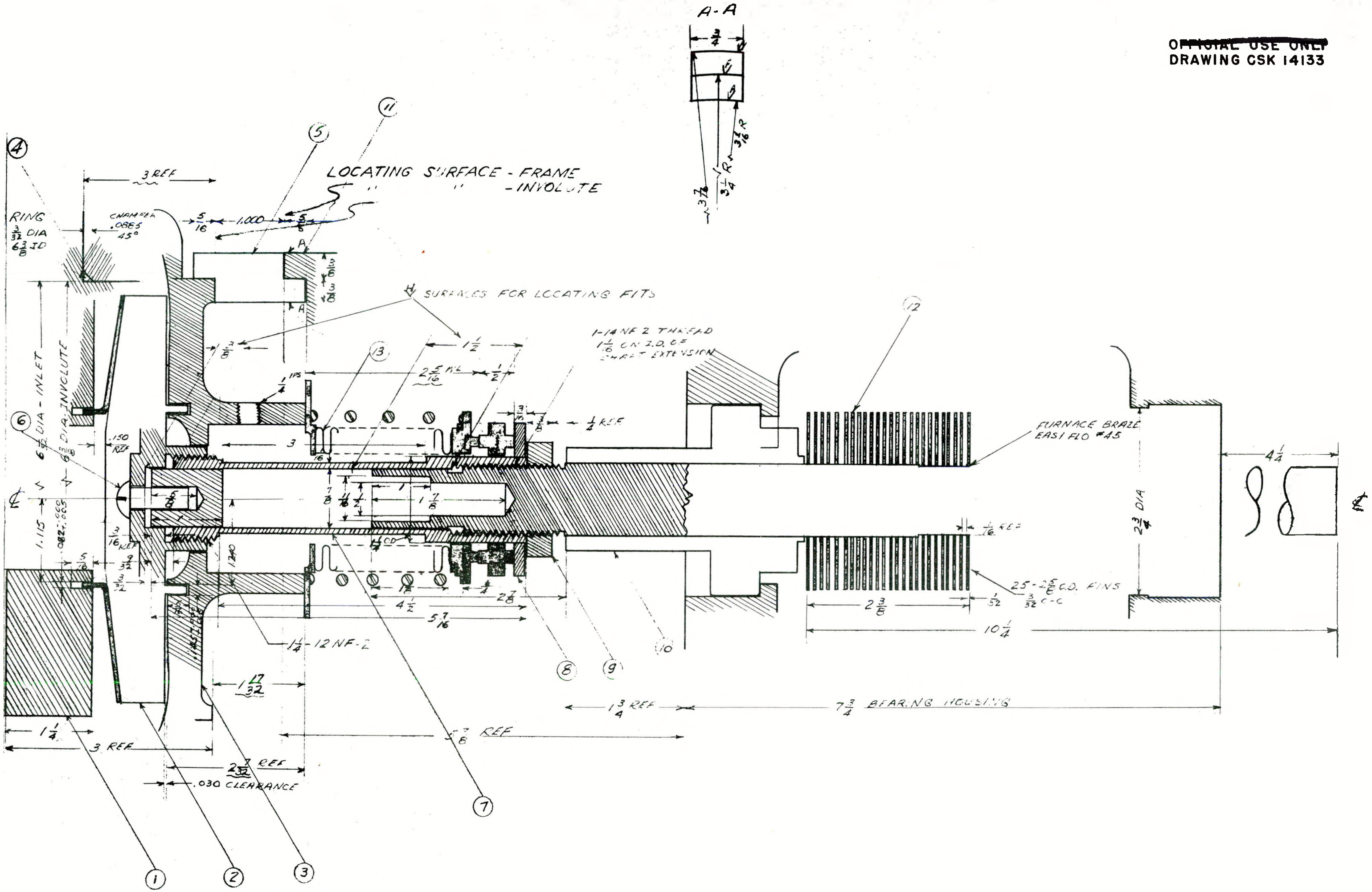

FIGURE 6.10 CENTRIFUGAL PUMP MODIFICATION

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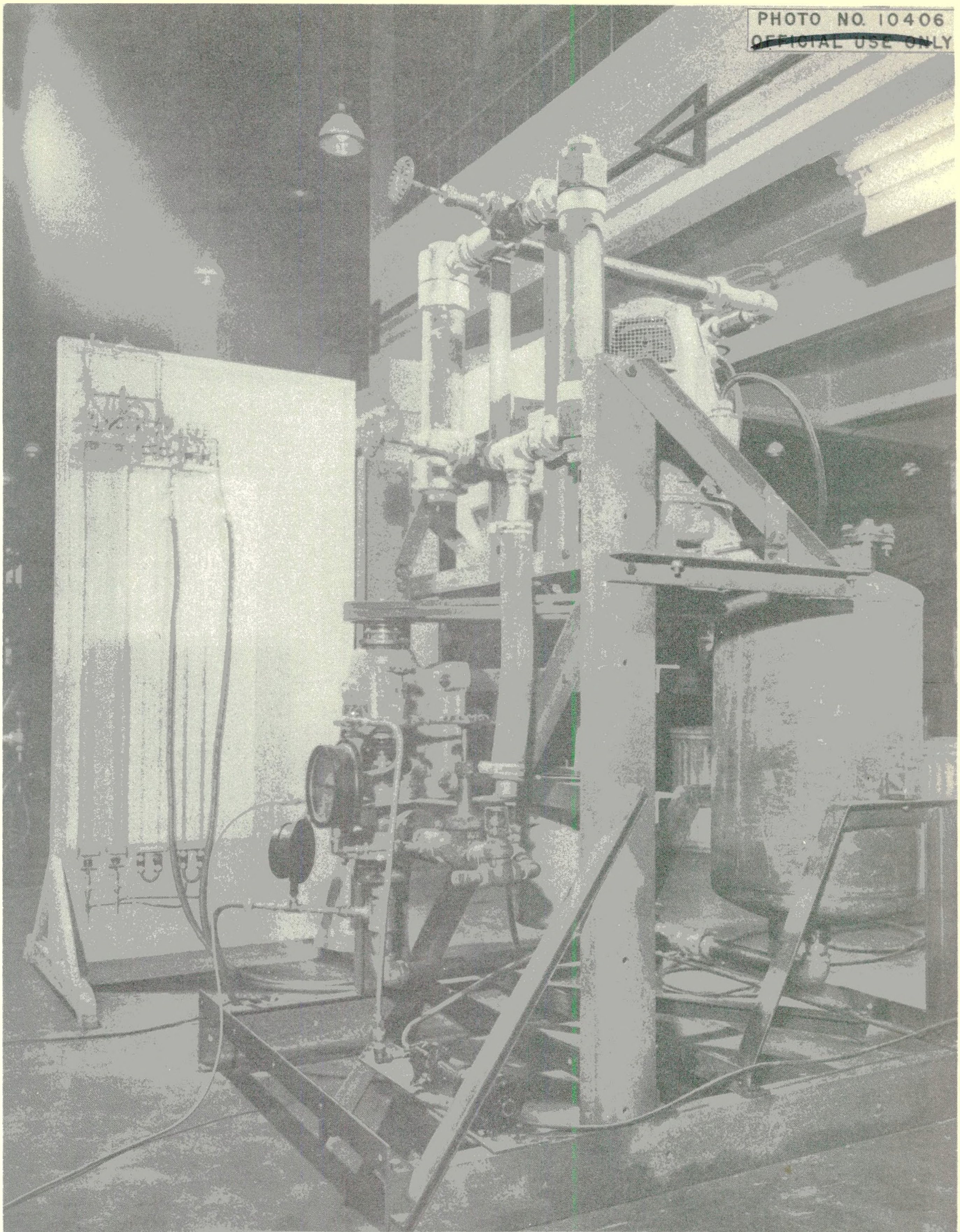

FIGURE 6.11 PUMP TEST STAND

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The inert atmosphere used over the present sodium test units has been lamp-grade argon, which is considered oxygen free. Gas-purification systems will be in operation shortly to ensure the purity of both argon and helium for use as inert atmospheres.

The purification procedure will consist in passing the gas over hot copper turnings to remove the oxygen, through a cold trap to remove water vapor, and then over hot titanium turnings to remove residual oxygen and the nitrogen. In order to prevent impure gas from passing to the test units in case of power failure, a NaK scrubber column is installed after the copper-titanium units, through which the gas will bubble and have the oxygen removed. A second column containing lithium or lithium and NaK will be used to remove the residual nitrogen.

Disposal and Cleaning Facilities (R. Devenish). Small valves and similar pieces of equipment are freed of sodium by immersion in a mixture of butyl and methyl alcohols, followed by water washing to be certain of removing the last traces of sodium. Larger items, such as the convection harps, are freed of sodium by heating above the melting point of the sodium and pouring as much as possible of the metal into a container. The partially cleaned harp is then blown with steam and washed with water or immersed in alcohol and washed with water.

A sodium disposal unit, Fig. 6.12, with a batch capacity of approximately 100 lb of sodium has been designed. The unit is based upon the disposal methods used at Schenectady by General Electric in which molten sodium is jetted under water through a steam nozzle.

The Liquid Metals Safety Committee has recommended a central cleaning and disposal facility for all groups in the Y-12 Area. The facility will be used for removing small quantities of sodium from pieces of equipment and for fumeless disposal of waste sodium.

The cleaning facility will consist of a graveled area upon which will be erected a steel barrier to protect operators and equipment from splashing or from flying pieces in case of explosion. Cleaning will be performed by alcohol immersion, steam blowing, or water washing or a combination of all methods.

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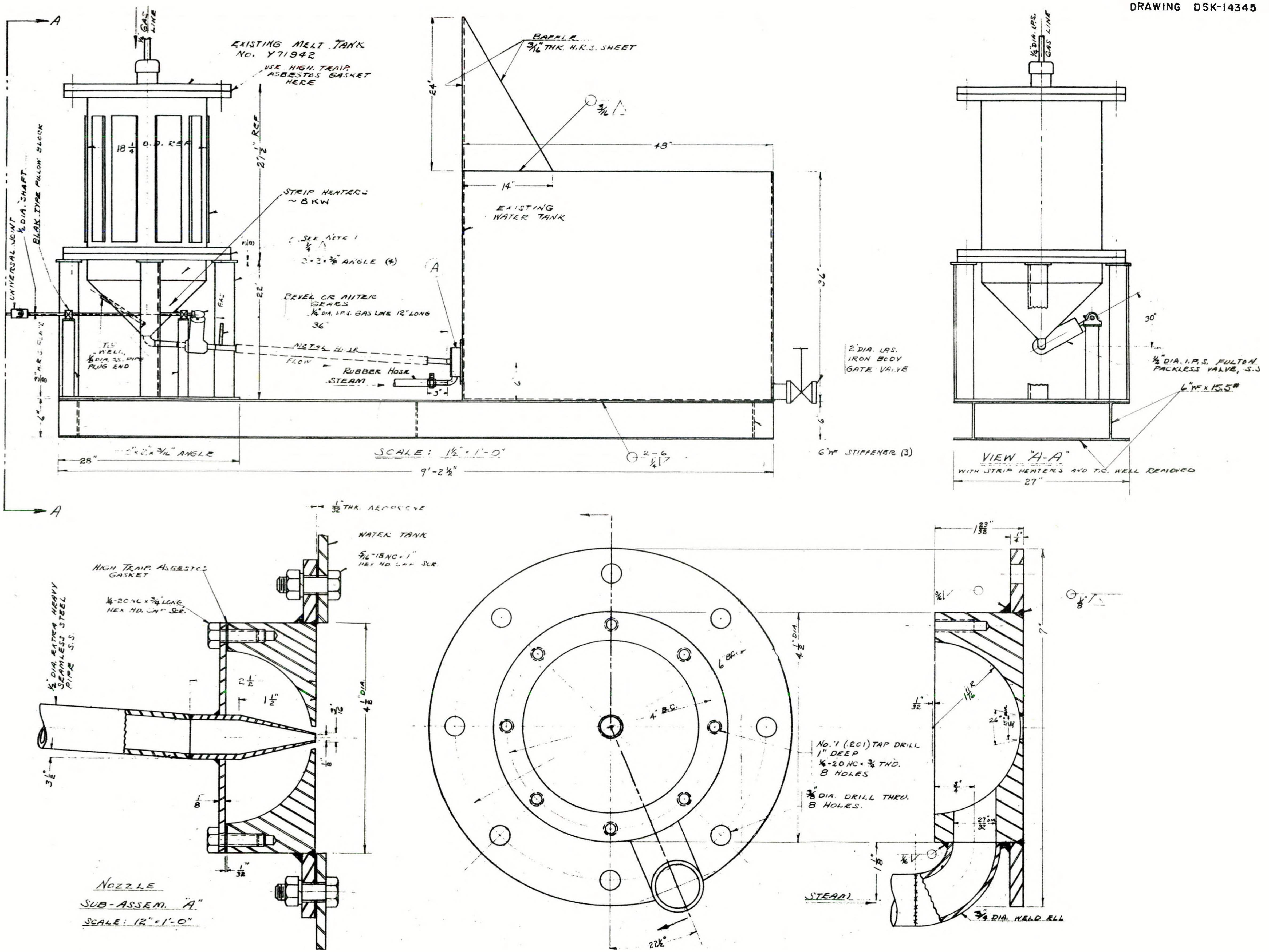

FIGURE 6.12 SODIUM DISPOSAL UNIT

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Waste sodium and sodium-contaminated insulation in small quantities will be disposed of in an abandoned quarry if it proves impractical to use the sodium melt unit.

Safety (P. L. Hill, USAF). Experiments performed by various groups with methods of extinguishing liquid-metal fires have indicated that Pyrene G-1 powder is suitable on sodium, potassium, and calcium fires; Ansul Metal-X is suitable on sodium and potassium fires

rust-free iron filings may be suitable on all metal fires; several other materials warrant further tests.

Tests of hot sodium upon protective clothing are still inconclusive but preliminary results indicate that chrome leather may be suitable. Further tests are being carried out.

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7. LIQUID-METAL AND HEAT-TRANSFER RESEARCH

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7. LIQUID-METAL AND HEAT-TRANSFER RESEARCH

R. N. Lyon and H. F. Poppendiek
Reactor Technology Division

During the past quarter notable progress has been made in heat-transfer research, in the determination of the physical properties of liquids and solids at high temperatures, and in the development of a loop for circulating hot liquid metals through the ORNL reactor.

In the heat-transfer work, final stages have been reached in the high conductance—entrance conductance sodium system. Design of a boiling-liquid-metal heat-transfer system is near enough to completion to start construction, and preliminary designs of a sodium hydroxide heat-transfer system and a system to determine the natural convection in liquid-fuel elements have been completed. Theoretical analyses of heat transfer have been completed on three postulated systems which approximate some of the entrance conditions involved in current ARE core proposals, and a wide range of duct shapes have been analyzed with regard to velocity distribution and heat transfer with liquid metals.

A new physical-properties laboratory has been equipped with benches, lights, and other services. The specific heat of lithium has been found to be 1.0 ± 10% between 550 and 900°C, and redesign of the specific heat equipment to give improved accuracy has been completed. Detailed design of equipment for the thermal conductivity of liquids is complete, as is the overall design of a high-temperature viscosimeter. Consideration has been given to the problem of measuring the densities of high-temperature liquids. Liaison has been established with the Office of Air Research and the Materials Laboratory of the Air Materiel Command, USAF, who have agreed to begin physical-property work in their own laboratories and to contract with other research laboratories for determination of physical properties of materials of interest to the ANP.

As a result of a decision at the last meeting of the Committee on Basic Properties of Liquid Metals, arrangements were made for the rapid dissemination of reportable liquid-metal data. Also, organization of a second edition of the "Liquid Metals Handbook" has started. As with the first edition, the second edition is to be a cooperative effort with other sites, but with the overall editorial responsibility at ORNL.

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HEAT-TRANSFER COEFFICIENTS

W. B. Harrison

Construction is about 95% complete on a system designed for two primary purposes: (1) Determination of the highest values of heat-transfer coefficient which are practically attainable with liquid sodium, and (2) exploration of the change in local coefficients resulting from thermal effects in an entrance region. It is expected that heat-transfer coefficients between 200,000 and 500,000 Btu/hr per square foot per degree Fahrenheit will be achieved with reasonable accuracy.

An assembly of the test section is indicated in Fig. 7.1. Liquid sodium at 250°F will be pumped through the 1/32-in.-diameter hole in the center of a copper disk. In order to promote radial heat flow, the disk will be insulated on each side and cooled by water in the tube around its periphery. Observations of disk temperature at various radial locations will be made after steady operation is achieved. From these data and the thermal conductivity of the copper, heat flux and surface temperature at the copper-sodium interface may be computed. Sodium temperature will be measured so as to permit computation of the heat-transfer coefficient. The coefficients computed in this way will be average values over the disk thickness. Disk thickness will be varied from 1/16 to 1/4 in. in order to explore the entrance effects.

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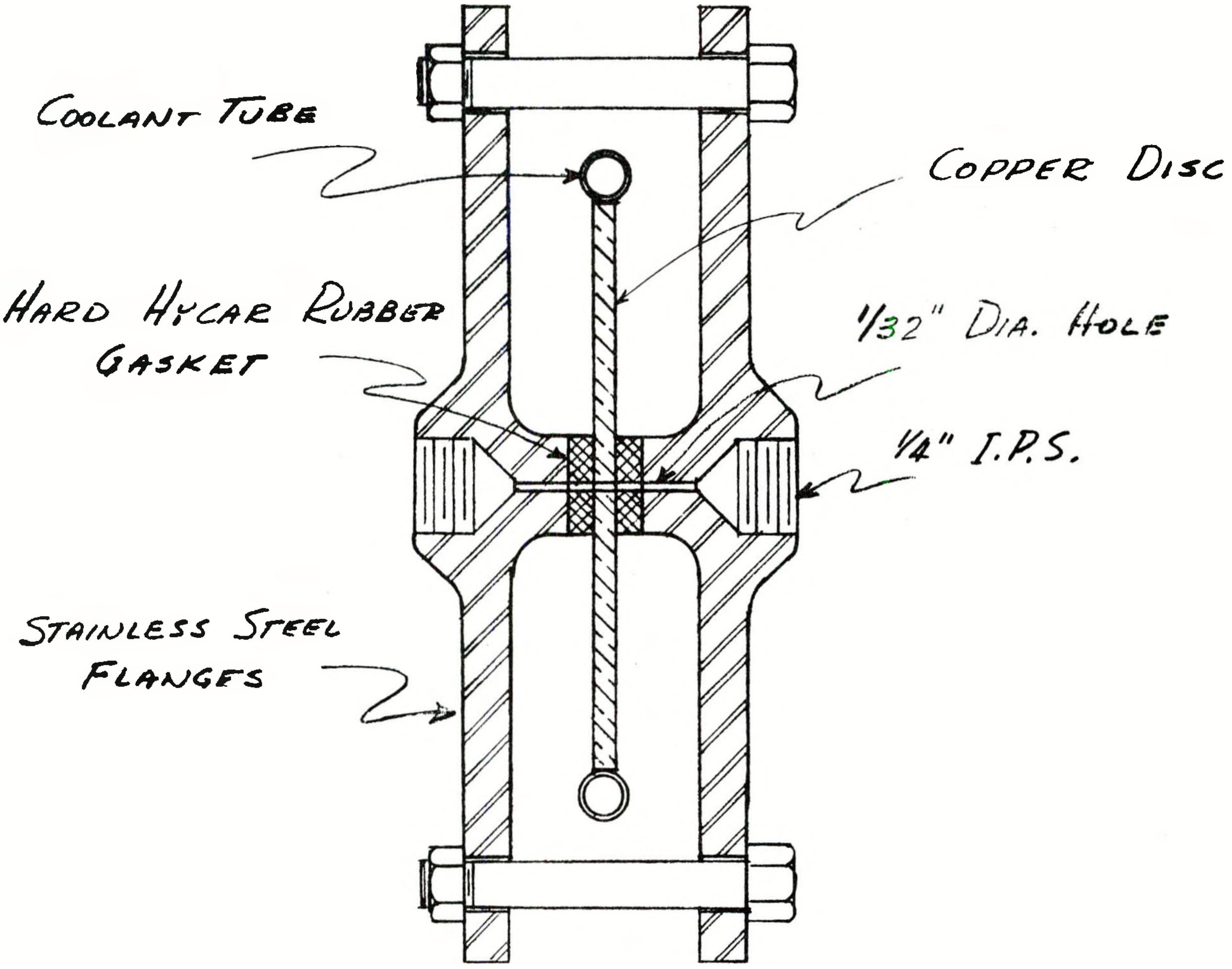

FIGURE 7.1 SECTIONAL VIEW OF PROPOSED TEST SECTION FOR DETERMINING HEAT TRANSFER COEFFICIENT OF LIQUID SODIUM

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MEAN CONDUCTANCE DATA USING NaOH

H. W. Hoffman

A tentative design has been made for an apparatus for the determination of the heat-transfer coefficient for a molten sodium hydroxide system under forced convection at temperatures from the melting point (604°F) to 1000°F and possibly higher. The test section will consist of an electrically heated tube, 0.25 in. I.D., 0.375 in. O.D., 25 in. long, with appropriately positioned thermocouples for the measurement of surface temperature. It has been decided to construct the system of "L" nickel. The components of the system and their relative positions are indicated in Fig. 7.2.

The design of a system for the measurement of heat-transfer coefficients can be divided into five major categories, namely, materials of construction, methods of handling and moving the fluid through the system, the design of the test section, methods of introducing and removing heat from the system, and measuring and control instruments.

In this particular investigation both corrosive and high-temperature conditions exist. The handling of materials at high temperatures does not pose too difficult a problem since numerous metals and metal alloys exist which will withstand temperatures well in excess of those expected in this investigation (604 to 1000°F). However, the number of materials able to withstand the corrosive action of molten sodium hydroxide is very limited. In a preliminary test at Battelle Memorial Institute regarding the feasibility of using various materials for containing molten sodium hydroxide, it was found that silver was best with a corrosion rate of approximately 0.1 mg/cm^2/month, with nickel and graphite next at approximately 1.0 mg/cm^2/month. These three are the only materials recommended.

Silver will soften at temperatures in the vicinity of 1000°F and must be used in the form of silver-clad steel tubes. Graphite has the serious disadvantage of having very poor mechanical properties, thus necessitating heavy awkward sections. Nickel possesses none of the disadvantages of the above two, and it has been decided to use "L" nickel for at least the test section of the apparatus.

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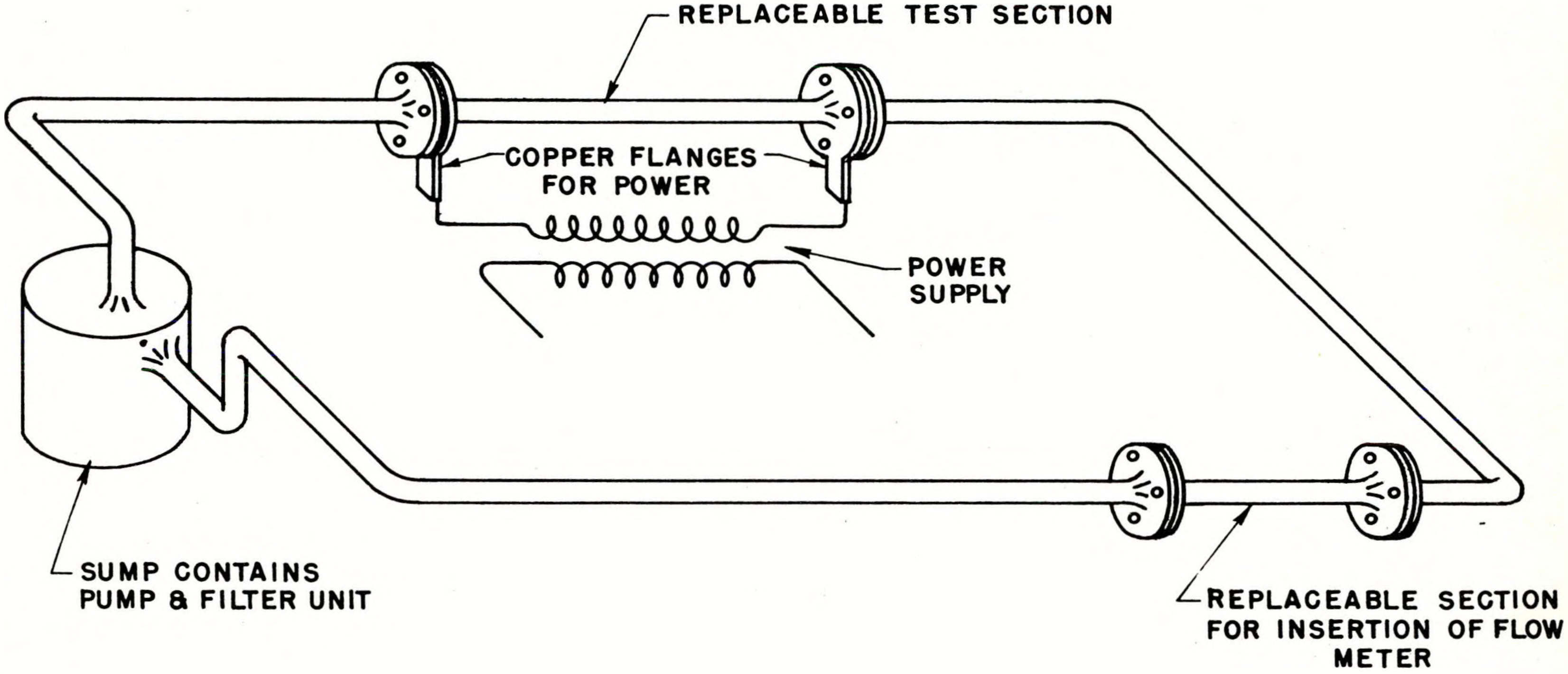

APPARATUS TO BE CONSTRUCTED FROM "L" NICKEL

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FIGURE 7.2 PROPOSED APPARATUS FOR DETERMINATION OF HEAT TRANSFER COEFFICIENT FOR MOLTEN NaOH

The molten sodium hydroxide will be circulated through the system by a centrifugal sump pump. Since the pump will be in the molten sodium hydroxide, it is necessary that the entire casing, in addition to the impeller, be constructed of nickel. It is planned to use a pump capable of handling 10 gpm at a discharge pressure of 15 psi.

The sodium hydroxide will be melted with the nickel sump tank and maintained at a temperature of about 800°F. Since it is suspected that Na_2CO_3 contributes greatly to the corrosive action of NaOH, it will be necessary to purify the commercial caustic. To prevent formation of Na_2O in the melt, a positive pressure of "forming" gas (10% H_2, 90% N_2) will be maintained over the melt in the sump.

Three tentative designs for the measurement of the flow rate are being considered. Two involve the use of strain gauges in association with an orifice plate. Both of these require calibration because of the variable temperature in the system. The third method uses a volumetrically calibrated catch tank with a magnetic (or radioactive) float.

Several alternatives present themselves in the design of the test section. The following three were considered:

1. Parallel channels of rectangular cross section
2. Concentric channels of circular cross section
3. Electrically heated tube of circular cross section

While all three systems will give information of value, stress has been placed on the third system, primarily because of the simplicity of its construction, the ease of calculation of results, and the elimination of the need for a supplementary system for adding heat. Preliminary calculations indicate that a 15-kw electrical system will be necessary.

Currently, consideration is being given to removing heat by radiation from the system as a whole.

BOILING LIQUID METALS

W. S. Farmer

During the past quarter it has been possible to select an arrangement of apparatus for determining heat-transfer coefficients to boiling liquid

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metals. The overall design is 75% complete while detail drawings are 30% complete at this time. A work order for construction will have been issued by the first of December.

The apparatus arrangement finally selected consists of a 12-in. vertical single-tube boiler. Heat is supplied by radiation from a graphite tube, which will be inside and concentric with the molybdenum boiler tube. The metallic vapors produced in the boiler will be condensed in a vertical tubular header consisting of several banks of finned tubes, across which air will be drawn. The design is shown schematically in Fig. 7.3.

Initial experiments will be conducted on natural convection boiling outside a 1-in.-diameter 6-in.-long tube. The flux will be variable up to 10^6 Btu/hr per square foot with existing variable-voltage power equipment. Graphite temperatures up to 5000°F will be required.

NATURAL CONVECTION IN LIQUID-FUEL ELEMENTS

P. C. Zmola

Natural convection will be required in the liquid-fuel elements now under consideration if excessive temperatures are to be avoided at the center of the liquid. Preliminary design has been completed for equipment to determine the extent and the effectiveness of natural convection within the fuel elements. Electric current will be passed through the liquid in a simulated fuel element to generate heat within the liquid itself, and heat will be removed from the surface of the fuel element at fluxes which will exceed 10^6 Btu/hr per square foot.

THEORETICAL THERMAL ENTRANCE ANALYSES

H. F. Poppendiek

Three new sets of analytical solutions have been developed for inclusion in a collection of analytical solutions which is now being compiled. The collection is to consist of solutions for postulated systems which approximate actual heat-transfer systems now under consideration in the ANP reactor development.

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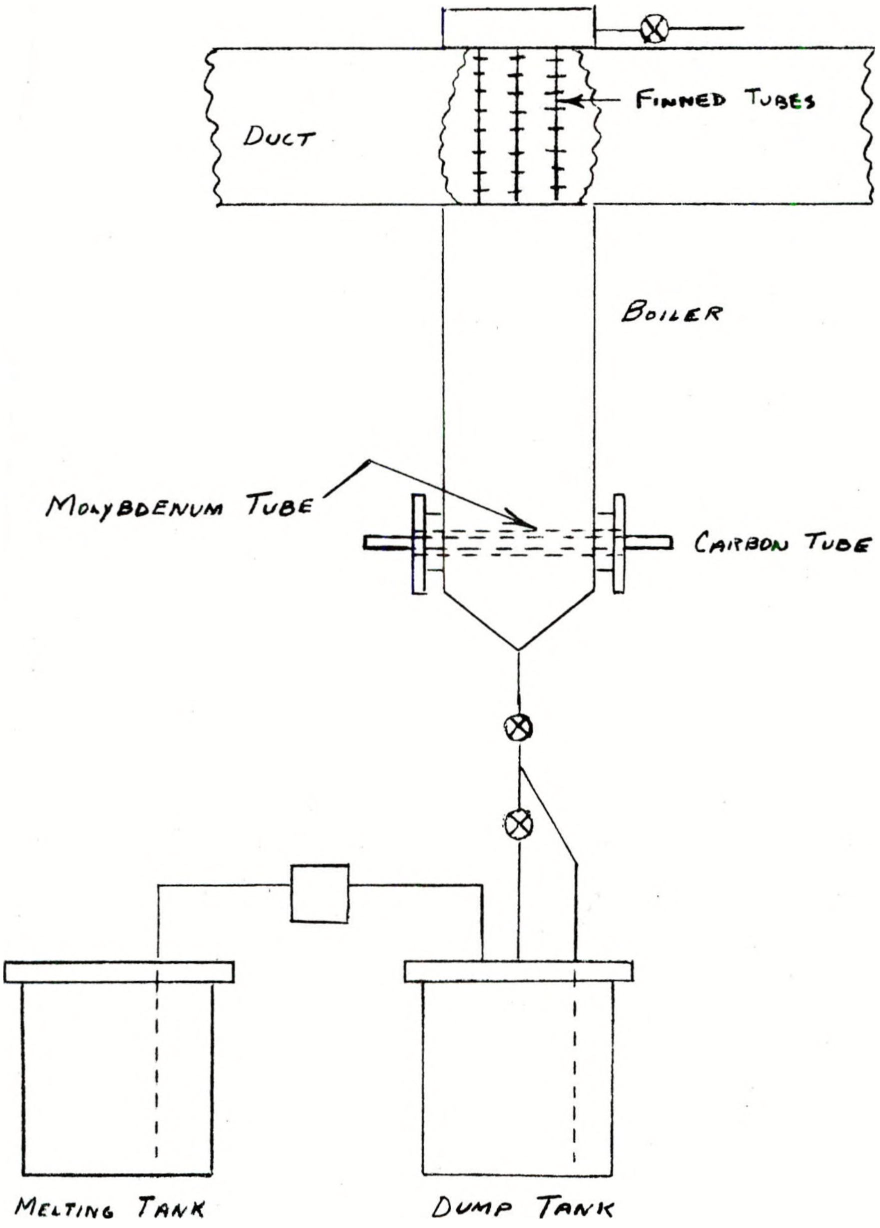

FIGURE 7.3 APPARATUS ARRANGEMENT FOR BOILING LIQUID METALS

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An analytical turbulent flow solution has been developed for a system in which (1) a fluid is flowing uniformly (slug flow) between two parallel plates, (2) the initial fluid and wall temperatures are uniform, (3) the fluid suddenly flows over surfaces which are emitting a uniform heat flux to the fluid, and (4) the eddy diffusivity varies linearly with distance from the duct wall (the total thermal diffusivity becomes the thermal molecular diffusivity at the wall). This analysis is in the process of being prepared as a technical report.

Analytical solutions for temperature distributions have been derived for parallel-plate and pipe-duct systems in which (1) the fluid is characterized by a turbulent velocity profile (a power-loss relation) and a high thermal molecular diffusivity compared to the thermal eddy diffusivity, (2) the initial fluid and wall temperatures are uniform, and (3) the fluid suddenly flows over surfaces which are at some new uniform temperature. Local heat flow and convective conductance solutions have also been developed. These analyses are being prepared as a technical report.(1)

An analytical thermal entrance solution has been obtained for a system in which (1) a fluid flowing uniformly between parallel plates is characterized by a high thermal diffusivity compared to the thermal eddy diffusivity and by a symmetrical volume heat source whose strength is a function of distance between the parallel plates, (2) the initial fluid and wall temperatures are uniform, (3) the fluid suddenly flows over surfaces through which heat is lost to a heat-transfer fluid on the other side (the heat-transfer fluid is at a uniform temperature and the duct-wall resistance is not the controlling resistance in the thermal circuit). It is felt that solutions of this type may be useful in describing heat transfer in reactor liquid-fuel elements.

FLUID FLOW AND HEAT TRANSFER IN NONCIRCULAR DUCTS

H. C. Claiborne

Theoretical investigations of fluid flow and heat transfer in noncircular ducts has progressed to a point where a status report(2) is being prepared.

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(1) Poppendiek, H. F., *Heat Transfer*, ORNL-913, to be issued.

(2) *Ibid.*

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The report will be released before the end of the coming quarter and will contain the following topics: (1) analytical solutions for velocity distribution for laminar or streamline flow in rectangular, equilateral triangular, right isosceles triangular, elliptical, and any circle sector ducts; (2) analytical solutions for the resulting temperature distribution when heat is transferred to a fluid flowing with a square velocity wave (slug flow) through the same type of ducts mentioned above with the duct walls at a constant temperature; (3) analytical solutions for the resulting temperature distribution when heat is transferred to a fluid flowing with a square velocity wave through rectangular, equilateral triangular, any right triangular, elliptical, and 60° circle sector (a general solution has not been obtainable) ducts for the case of constant heat flux from the duct walls.

Average Nusselt's moduli for the case of constant flux were found to be 6, 4, 3, 2, and 0.000386 for rectangular, equilateral triangular, 45°, 30°, and 1° right triangular ducts, respectively, as compared with the value of 8 for a circular pipe.

It can be shown that the slug flow solutions will be reasonable approximations for the case of turbulent flow of liquid metals for Reynolds' moduli below about 10,000, and that heat transfer will be underestimated by these solutions at higher Reynolds' moduli.

Other more complex problems involving noncircular ducts under consideration are entrance solution, effects of finite wall thickness for the case of constant flux to the outer side of the duct wall, heat transfer to a fluid flowing with an established laminar velocity distribution, and some possible approximations for the case of turbulent flow.

PHYSICAL PROPERTIES

A. R. Frithsen, USAF

During the past quarter a laboratory was established for the measurement of physical properties of liquid metals, liquid salts, liquid caustics, structural materials, insulators, and other materials that may be utilized in the design of an aircraft reactor. The design of apparatus for the measurement of specific heats, thermal conductivities, viscosities, and densities

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has been initiated. At the time of this writing it was anticipated that the specific heat apparatus would be completed and ready for routine measurements with an accuracy of the order of ±5% by Jan. 1, 1951. It is expected that the thermal conductivity apparatus and the viscosity apparatus will be capable of accuracies of about ±10%, and will be completed by Apr. 1, 1951; and that the density apparatus will be capable of accuracies of about ±2%, and will be completed about Apr. 15, 1951. An approximate value for the specific heat of lithium has been obtained between 550 and 900°C.

In order to avoid unnecessary duplication of effort, personnel of the physical properties laboratory are coordinating their efforts with other agencies also involved in this type of work. As a result of a conference with personnel from the Office of Air Research and the Materials Laboratory of the Air Materiel Command, USAF, these agencies of the USAF will in the near future begin work on the measurement of liquid metals, liquid salts, and other materials at their own laboratories at Wright-Patterson Air Force Base, Dayton, Ohio, and also by contracting with various universities and research organizations. In addition these agencies will contract directly for the development of physical-property-measurement techniques and apparatus in an attempt to develop methods with extremely high accuracies.

A survey shows that the following data already have been obtained by other agencies and are considered acceptable for ARE design studies:

MEASUREMENT	AGENCY	TEMPERATURE (°C)
Specific heat	Bureau of Standards	
Sodium		100 to 800
Potassium		100 to 700
Sodium-potassium eutectic		100 to 700
Beryllium		0 to 900
Viscosity	Naval Research Laboratory	
Sodium		100 to 700
Potassium		100 to 700
Sodium-potassium alloys		100 to 700
Viscosity	Critical Tables	
[redacted]		441 to 844
Density	NEPA	
Bismuth		m.p. to 1000
[redacted]		m.p. to 1000
[redacted]		m.p. to 1000
Lead-bismuth eutectic		m.p. to 1000

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(Continued)

MEASUREMENT	AGENCY	TEMPERATURE (°C)
Density	Naval Research Laboratory	
Sodium		100 to 700
Potassium		100 to 700
Sodium-potassium alloys		100 to 700

In addition to the quantities listed above, it is deemed necessary to determine the specific heats, thermal conductivities, viscosities, and densities of all materials of interest in the design of an aircraft reactor, and, in addition, it will be necessary eventually to extend the above information to 1000°C in each case.

Specific Heat (R. F. Redmond, L. F. Basel, J. Lones, A. Bates,* D. Smith,* W. H. Bowman,* D. James,* and J. Roarty*). The specific heat project has progressed to the stage where the method of determination has been selected and the apparatus has been designed, constructed, and used to determine the specific heat [redacted] over the temperature range 550 to 900°C. At present the apparatus, a Bunsen ice calorimeter (Figs. 7.4 and 7.5) is being redesigned.

The enthalpy values [redacted] as obtained experimentally in the temperature range 500 to 1000°C, are shown in [redacted] A careful statistical analysis of these points in this figure indicates that the best curve through them is a straight line, and, inasmuch as the specific heat is defined as

$$C_{p\ \mathrm{absolute}} = \left(\frac{\partial H}{\partial T}\right)_P$$

then the slope of this straight line, [redacted] is the specific heat [redacted] [redacted] over the temperature range. However, the calorimeter was not calibrated, and subsequent experiments with metals of known specific heats showed the calorimeter to be 5 to 8% low. Calibration of this calorimeter by electrical means evidenced excessive heat leakage. Since minor alterations did not materially decrease this leakage, the calorimeter has been redesigned with emphasis given to the improvement of the rate of heat transfer from the heat capsule to the container. The new calorimeter together with refinements in technique will, it is hoped, substantially eliminate the previously encountered experimental errors.

*From MIT Practice School.

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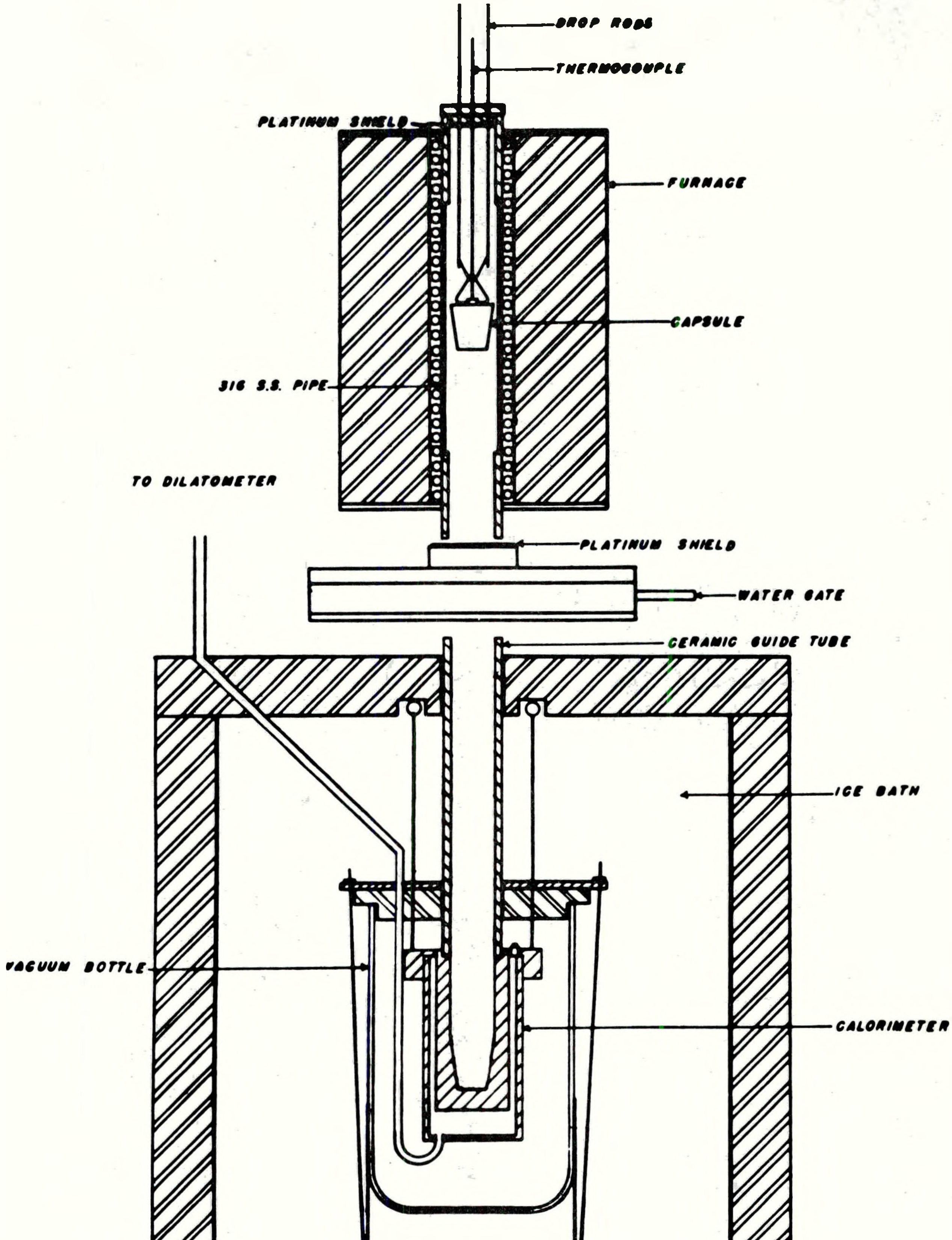

FIGURE 7.4 CALORIMETER ASSEMBLY

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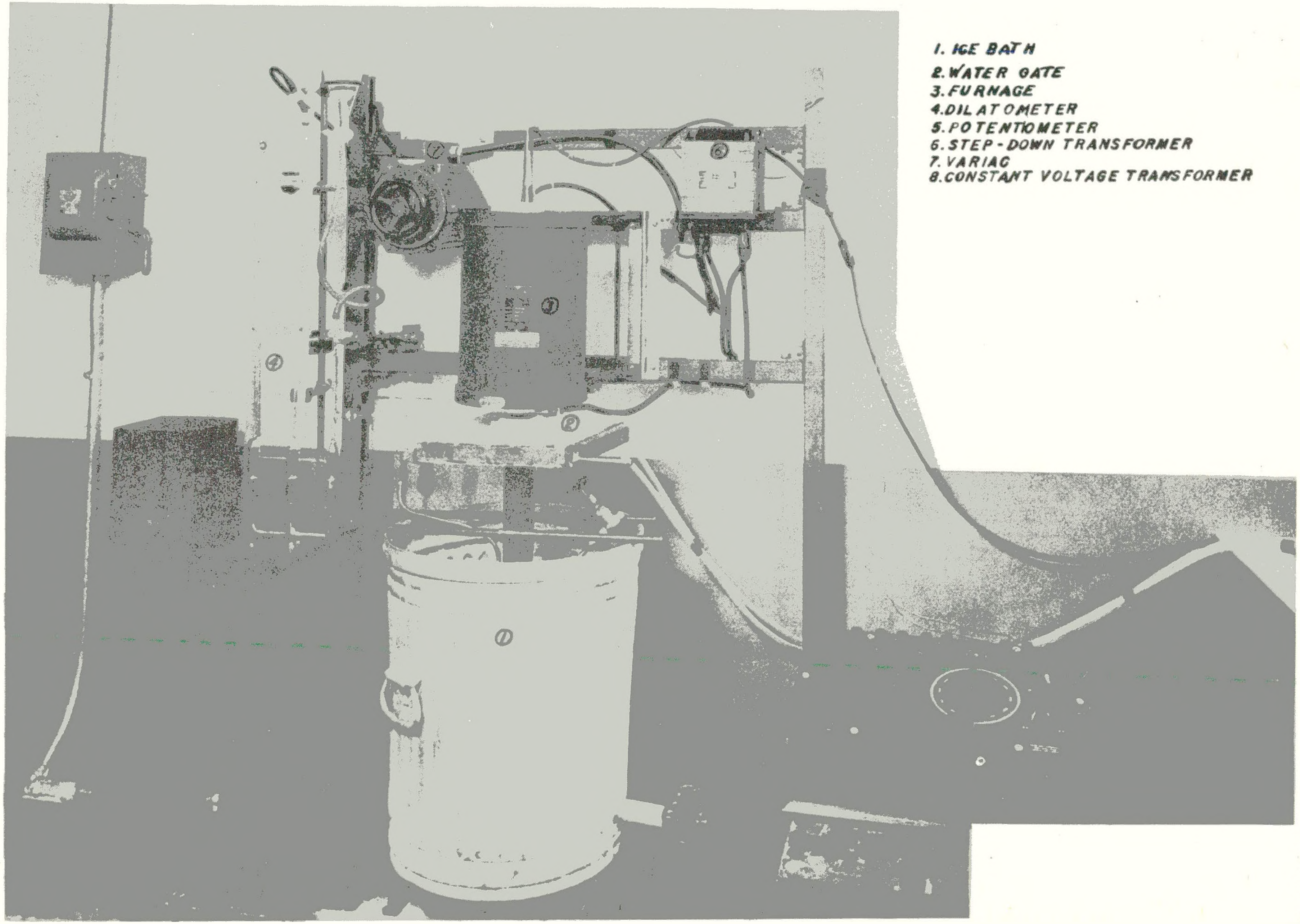

1. ICE BATH
2. WATER GATE
3. FURNACE
4. DILATOMETER
5. POTENTIOMETER
6. STEP-DOWN TRANSFORMER
7. VARIAC
8. CONSTANT VOLTAGE TRANSFORMER

FIGURE 7.5 CALORIMETER (Closed)

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Thermal Conductivity (L. F. Basel and M. Tobias). As a result of a survey of possible methods for the measurement of thermal conductivities of liquid metals from their melting point to 1000°C, a steady-state linear flow system, similar to one used by Dr. Deem of the Battelle Memorial Institute, has been selected. The design of this equipment is now nearing completion.

A schematic diagram of the Deem apparatus is shown in Fig. 7.7*a*. Heat flows from the upper heating plate down through the liquid, through the Armco-iron standard, to the copper cooling chamber. The heating plate is movable so that the liquid-specimen thickness can be varied. Thermocouples are located in the Armco-iron standard and in the metal surfaces above and below the liquid specimen. The resistance to heat flow between the two specimen thermocouples can be considered to be the sum of the specimen resistance, x/kA, and of the fixed resistance of the two liquid-solid interfaces, F_f.

It is clear that

$$\Delta T_s/Q = R_f + x/kA$$

where

ΔT_s = temperature difference between specimen thermocouples

Q = total heat flow through specimen

R_f = fixed resistances at liquid-metal interfaces

x = specimen thickness

A = specimen cross-sectional area

k = specimen thermal conductivity

By making several runs at different specimen thicknesses but at the same mean temperature, a plot of $A\Delta T_s/Q$ vs. x can be constructed; the slope of the line through the points is the reciprocal of the thermal conductivity.

Tests of an unsteady-state method were made at a temperature of approximately 250°C on Armco ingot iron using the apparatus shown schematically in Fig. 7.7*b*. The block was heated initially to *c* constant temperature and,

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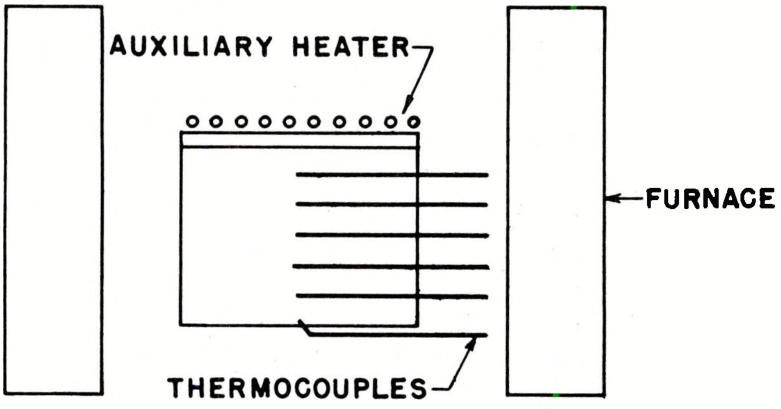
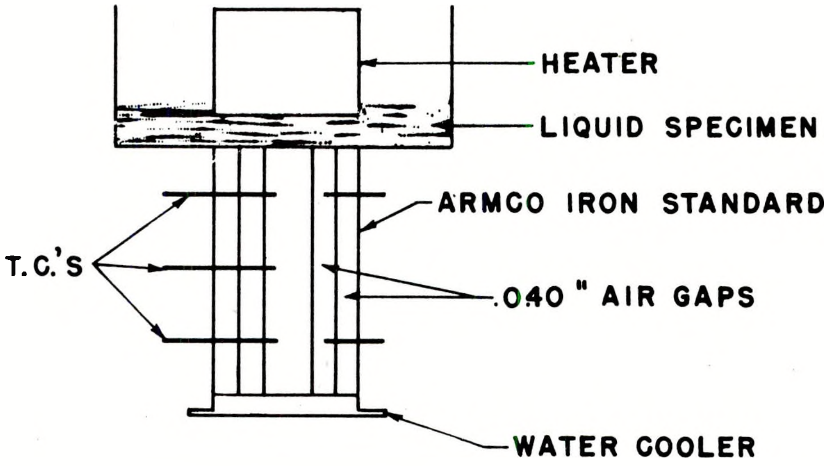


B. UNSTEADY STATE APPARATUS



A. DEEM APPARATUS
STEADY STATE

FIGURE 7.7 APPARATUS FOR THE MEASUREMENT OF THE THERMAL CONDUCTIVITY OF LIQUID METALS at a given time, the auxiliary heater was turned on. Temperatures were recorded at each of the thermocouple positions every few seconds. Ideally, α, the thermal diffusivity, can be calculated from the temperature, time, and distance measurements, since it is given by $\alpha = \frac{\partial T}{\partial \theta} \frac{\partial^2 t}{\partial x^2}$ for unidirectional flow. However, the results were inconclusive because much accuracy was lost in taking first and second derivatives. It is believed that this source of error can be reduced by improved design to give greater precision.

Several other types of unsteady-state systems have been considered, but in most of these systems it is difficult to maintain the proper boundary conditions. Radial heat flow methods cannot be used with liquids because of heat transfer by convection. To maintain a stable system, heat must be conducted unidirectionally from top to bottom of the liquid specimen. A steady state system, similar to that used by Van Dusen of the National Bureau of Standards for metals, has been adapted for use with liquid metals at KAPL. In this apparatus, heat is first conducted through the sample and then through a metal of known thermal conductivity. In the liquid-metal system a thin-walled container holds the sample and a correction is applied for the heat conduction down the wall. This apparatus cannot be used for liquids of low thermal conductivity since the wall correction would be large and the final determination would be inaccurate. Since there is some interest in materials of low conductivity, and since this apparatus appears to be complicated, the method was rejected.

Viscosity (S. I. Kaplan). Several methods of measuring viscosity were considered, including capillary-flow, falling-ball, and rotational viscosimeters. A literature search was made and numerous manufactures were polled on the availability of suitable commercial units, but none were found which would fulfill, without extensive modifications, the requirements peculiar to this project, namely, corrosion resistance, easy cleaning, ready adaptability to various liquids, and the ability to yield results of useful accuracy after brief preparation.

The initial method finally decided upon will employ a falling-ball viscosimeter. Preliminary design of the unit has been completed and fabrication is now in progress. The apparatus consists essentially of a stainless steel tube mounted vertically in a furnace and filled with the liquid being tested. A radioactive ball, of a diameter small compared with that of the tube, is dropped into the liquid. The time the ball takes to fall a measured distance through the liquid is determined by externally mounted detectors. The viscosity can then be calculated in absolute units from the known relationships given by Stokes' law. The dimensions of the instrument are ample to ensure precise velocity measurements (Fig. 7.8).

As a possible secondary means of measurement, a modification of a commercial viscosimeter employing concentric rotating cylinders is being considered, and correspondence with the manufacturers on adaptation problems is being conducted. A third method, involving measured flow through a capillary, has been used by the Naval Research Laboratory for liquid metals. However, the apparatus is more cumbersome than those proposed, and is more difficult to change over from one liquid to another.

Density. Methods for the determination of the density of high-temperature systems are currently being investigated. Possibilities include the weighing of a suspended bob in the liquid, and determination of the pressure required to force a gas bubble from the mouth of a submerged tube against a known depth of liquid.

DEVELOPMENT OF COMPONENTS FOR EXPERIMENTAL HEAT-TRANSFER SYSTEMS

A. R. Frithsen and M. Richardson

A completely sealed rotary pump has been developed for use with experimental heat-transfer systems. Although the previously reported hydraulic bearings have proved adequate, new bearings which will permit increased flow are being fabricated. Four electromagnetic pumps have been received from G.E. and three Flowrators from Fischer and Porter Co.

Pumps. The main efforts during this quarter have been devoted to continuing the development of a completely sealed rotary type pump since this type appears to show the greatest flexibility for utilization in experimental systems. The hydraulic bearings discussed in detail in the previous quarterly report(3) were tested in the bearing test system and operation proved to be

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(3) Frithsen, A. R., "Pump Development for Experimental Systems," *The Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending August 31, 1950*, ORNL-858, 56 (Dec. 4, 1950).

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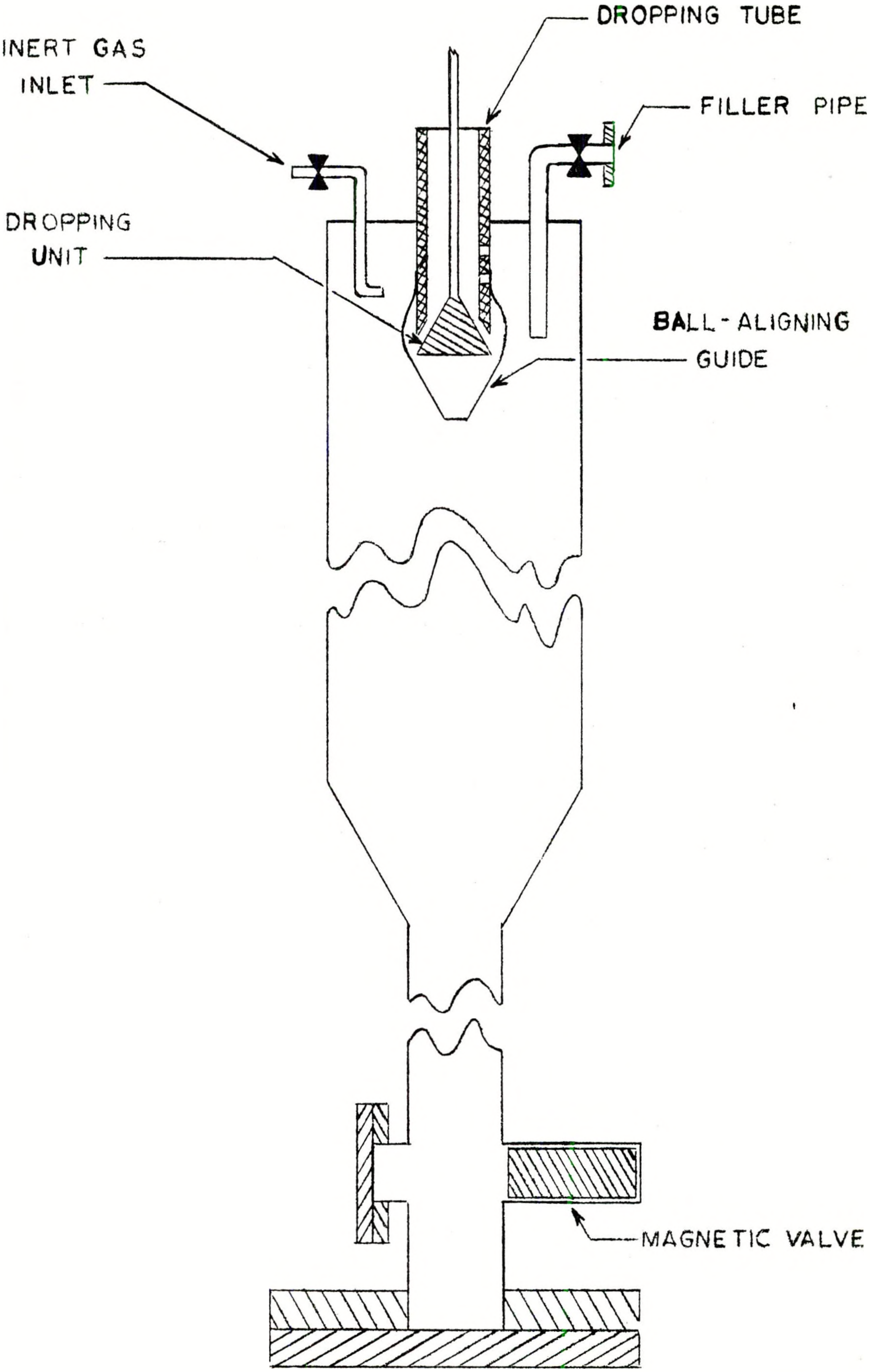

FIGURE 7.8 VISCOSITY MEASURING TUBE

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satisfactory. Several runs of ½ to 3 hr duration each were made; however, testing had to be terminated before long-duration runs could be made because of misalignment in the bearing test system.

A new hydraulic bearing has been designed and fabricated, which is similar to those previously described except that two rows of longitudinal pressure slots are used and flow of liquid from each end of these slots is now possible. This latter feature was incorporated owing to the fact that work at Allis-Chalmers Manufacturing Co. on similar type bearings indicates that performance is greatly improved if the flow through the bearing is increased.

The bearing test system has been modified so as to test these new bearings, and initial tests indicate that, although more flow is required in the new type bearings, they will operate at a much lower pressure than the original bearings (i.e., 10 psi as compared with 45 psi). An adapter is now being fabricated which will make it possible to use the same bearing test system to test both types of bearings. Thus it will be possible to obtain a quick comparison of performance of the two types so as to determine the best compromise between efficient operation and simple construction.

The design of a pump using hydraulic bearings and having a capacity of 12 gpm at 230 ft is almost complete. The system for testing this pump up to a temperature of 400°F is in the process of being designed.

Four G.E. electromagnetic pumps have been received and one has been transferred to the Experimental Engineering Section of the ANP Division for use in a liquid-metal corrosion testing system. The other three pumps will be used as the need arises.

Flow Measuring Devices. Three Flowrators (rotameters) have been received from the Fischer and Porter Co. which were designed to meet the following conditions:

Fluid	[redacted] sodium
Operating pressure	100 psi
Operating temperature	100°F
Flow	2 Flowrators for 3.3 to 33.0 gpm 1 Flowrator for 7.5 to 75.0 gpm

One of the 3.3- to 33.0-gpm Flowrators will be incorporated in the pump test system to test its operation in sodium and sodium-potassium alloys.

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8. METALLURGY

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8. METALLURGY

E. C. Miller, Metallurgy Division

One of the most important requirements of any structure is that the various members be mutually compatible. Very little information is available on the high-temperature compatibility of the various materials now being considered for use in the ARE. Consequently, emphasis has been placed on both static and dynamic corrosion tests of such materials. The static corrosion of metals, particularly the stainless steels, by sodium is being extensively studied;

Dynamic corrosion tests in thermal convection loops are underway but no results are yet available. The laboratory for the fabrication of fuel elements will not be completed before February; nevertheless, it is in operation and some fuel elements have been pressed and extruded. Static corrosion tests were performed on combinations of materials pertinent to this program. The Welding Laboratory and the Creep-Rupture Laboratory are still being set up, although preliminary experiments have been undertaken by both groups.

STATIC CORROSION TESTING

J. E. Cunningham

This initial phase of the liquid-metal corrosion testing program is designed to sort out potential structural materials for use in contact with various coolants. Sodium is now being considered for use in the proposed ARE Testing of materials in sodium and lithium is being continued, but at a reduced effort.

Two important observations made concerning liquid-metal corrosion work in the past six months are: (1) The depth of attack or penetration in liquid-metal media as measured by metallographic examination is a more reliable criterion for evaluating the corrosion behavior of materials (particularly alloys) than are weight change data and (2) in many cases, mass transfer effects prevent use of an inert or third material in the test system, and tests must be conducted in tubing containing only the test coolant. On the other hand, information obtained in this manner may be of little value in an application requiring the presence of more than one solid metal.

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Metallic Elements in Sodium. Tests were initiated on several elements and alloys to determine their corrosion resistance to sodium at high temperatures. The sodium used was of commercial purity (99.95%) from Merck and Company. The first tests were made with the specimens in Armco-iron capsules, examination of which showed that Armco would not contain sodium as it had apparently leaked through the capsule bottom, sometime during the test period. Metallographic examination showed that these leaks possibly occurred as the result of reaction of sodium with the manganese sulfide stringers. Because of the uncertainty of the length of time the specimens had been exposed to the molten sodium, these tests were discarded.

A trial test was made using nickel for the capsule and it was found that this material would contain sodium barring any imperfect welds. Since this time, the tests in sodium have been in evacuated nickel capsules. Preliminary inspection, on the basis of weight change data only, indicates that nickel has good resistance and that cobalt, molybdenum, tantalum, Alloy N-155, inconel, and inconel X have fair resistance to sodium at 1000°C for 400 hr.

The stainless steels are also being studied extensively in the form of both flat stock and tubing exposed to sodium at high temperatures. The metallographic work on the stainless steel alloys has not been completed although data have been compiled on several special tests that have been run. Two samples of 304 stainless steel (18-8) flat stock, with an extra low carbon content (0.006%) were tested for 40 hr in sodium at 1000°C. Both had a high weight loss. A heavy precipitate was seen at the grain boundaries throughout the specimens and a new phase (possibly sigma) appeared in crystallographic planes as needles. The specimens seemed to be fairly heavily corroded although the attack did not appear selective. A dark, nonadherent, unidentified film formed on the specimens. The corrosion depth was 0.006 to 0.008 cm.

Two ferritic type 446 stainless steel samples with low carbon content (0.006%), flat stock, were also tested in nickel capsules for the same period and temperature. These samples also showed a precipitate both in the grain boundaries and within the grains, but in these it appeared very angular in shape; it also could be sigma. Some irregular corrosion which was also seen was confirmed by the small weight loss. The surface exposed to the sodium was rough and had a blue-gray compound adhering to it. The corrosion depth for type 446 was 0.002 to 0.005 cm.

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Two tests were performed on stainless steel tubing filled with sodium. These were heated for 400 hr at 1000°C (1832°F). The sample of type 310 showed an extensive intergranular attack with large voids in the grain boundaries at the exposed surface. The matrix appeared to have complex carbides precipitated both in grain boundaries and along crystallographic planes. The sample of type 420 showed essentially no attack and the structure was that of a typical sample of 420 stainless steel.

In general the tests of stainless steels in sodium indicate a mild attack.

DYNAMIC CORROSION TESTING

Anton Brasiman

The current dynamic corrosion testing has been limited to experimentation on thermal convection loops. The work is being carried out by the ANP Experimental Engineering Group with metallurgical control and examination to be done by the ANP Materials Group in the X-10 Metallurgy Division. No results are yet available although 19 loops are being, or have been, operated with sodium at Y-12. So far, no loops have been examined metallographically. More loops are on order.

Thermal Convection Loops. The loops are filled with filtered sodium and operated under a helium atmosphere at temperatures up to 1500°F for 1,000 hr or until failure occurs. At present, temperature differentials between 60 and 100°C are being obtained around the loops.

Immediately after the loops are filled a sample of sodium is taken for determination of oxygen. After the loop has been operated, a sample of sodium is taken from the top cup for determination of oxygen and samples are taken from the top and bottom cups for spectroscopic analyses for carbon and metallic corrosion products. From both the hot and cold zones of the loop metallographic specimens will be taken from the pipe and welds. These samples will be examined for corrosion and any evidence of a local build-up of corrosion products or film formation.

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WELDING LABORATORY

P. Patriarca

Welding research to date has been confined to preliminary experiments, using the welders and existing facilities of the X-10 Research Shops pending the completion of a welding laboratory in the Metallurgy Division. A satisfactory technique for the welding of molybdenum has not yet been developed. Preliminary experiments on welding niobium are more promising. In addition, technical assistance is being supplied to the Dynamic Corrosion Testing group in the welded fabrication of convection loops.

Welding of Molybdenum. All welds were made manually and found to be brittle regardless of the welding process or welding conditions used. A massive copper jig was constructed which permitted a steep thermal gradient during welding in order to minimize recrystallization, this being one of the factors believed to be responsible for the brittleness of molybdenum welds. Ample inert-gas shielding was also provided by the jig by supplementing inert gas flow from the welding torch with a separate inert gas cover of the work within the copper welding jig. Both direct-current straight polarity and alternating current were used with the tungsten-arc process under commercial helium and argon as a shielding gas. Direct-current straight polarity was found to provide better control of the weld bead. Atomic-hydrogen welding was also attempted. All welds were edge welds using Fansteel molybdenum sheet 0.020 in. thick. Recent research indicates that present grades of molybdenum must be improved by further purification or by minor alloy additions to render them weldable. Future experiments are contemplated using Climax arc-cast molybdenum and available grades of molybdenum alloyed with minor additions. It is hoped that near-ideal welding conditions can be achieved by conducting future experiments within the confines of an inert-gas-purged dry box.

Fabrication of Thermal Convection Loop. Technical assistance is being supplied to the Liquid Metals Dynamic Corrosion Group in the form of supervision of welded fabrication of all future loops. Written welding specifications and schedules will be set forth on each thermal convection loop and will include specific instructions on edge preparation of component parts, welding sequence, and welding conditions to be used. It is believed that this quality control will minimize premature convection loop failures.

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CREEP-RUPTURE LABORATORY

R. B. Oliver

Eight Baldwin lever-arm machines and two Baldwin screw type stress-rupture machines have been received and erected. Six additional lever-arm machines have been designed and are in the process of construction. Sixteen Leeds and Northrup Speedomax-DAT recorder-controller units have been received and installed, and one 56-point precision indicator and one 12-point recorder have been received and installed. Delivery of the power distribution equipment is expected by the middle of December. A 20-kw gasoline driven emergency power plant has been ordered and should be installed by February. The chambers for testing in vacuum or in inert atmosphere have been designed, and production of these chambers and their accompanying furnaces will be started in the near future.

Since most of the metal going into the reactor and its auxiliaries will be in the form of pipe and tubing, attempts are being made to devise a creep-rupture test for such shapes. Tests will be performed over the range of temperatures expected in service and in the several environments chosen for the conventional creep-rupture tests. It is hoped that a method can be devised to measure the tubing creep, both radially and longitudinally

The overall ANP Creep-Rupture program will test pipes and tubing in the temperature range from about 600 to 1000°C in air, in liquid metals, and in inert atmospheres. These data will not only indicate possible design stresses but will also give information on stress corrosion in liquid metals and oxidizing atmospheres. The initial effort will be with the austenitic stainless steels with subsequent efforts being indicated by the interim ANP materials studies.

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9. RADIATION DAMAGE

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9. RADIATION DAMAGE

D. S. Billington, Institute for Solid Studies, Metallurgy Division

Radiation damage experiments are underway at a number of installations. The Y-12 cyclotron is being adapted for such work, and targets are now being constructed. Both stationary and rotating targets are being developed, and initial measurements of creep and resistivity under bombardment will be made. The lithium-iron corrosion experiments in the Berkeley cyclotron will substitute Globe iron for Armco. The Purdue University cyclotron group is preparing to run creep experiments.

Initial experiments on various metals have indicated no significant changes in shape, electrical resistivity, permeability, or hardness after an exposure to a neutron flux of 1.2×10^{12} and while at 400 to 500°F and 1000 psi. In-pile creep tests will be performed on a cantilever type apparatus because of its reliability. Operational techniques with this apparatus are now being perfected.

Y-12 CYCLOTRON EXPERIMENTS

R. S. Livingston, Electromagnetic Research Laboratory

The construction of targets and the adaptation of the 86-in. cyclotron at Y-12 for radiation damage work is well underway. Modifications consist, in the main, of a vacuum lock and handling equipment for the targets. Consideration has also been given to changing the resonance frequency and magnetic field so that low-energy protons of 10 to 15 Mev can be obtained at a larger radius. Initial plans, however, call for using the cyclotron as it is presently operating.

The basic design for the targets consists essentially of two concentric tubes through which a cooling (or heating) fluid is passed (Fig. 9.1). With a large flow of liquid through the system, small variation in beam intensity

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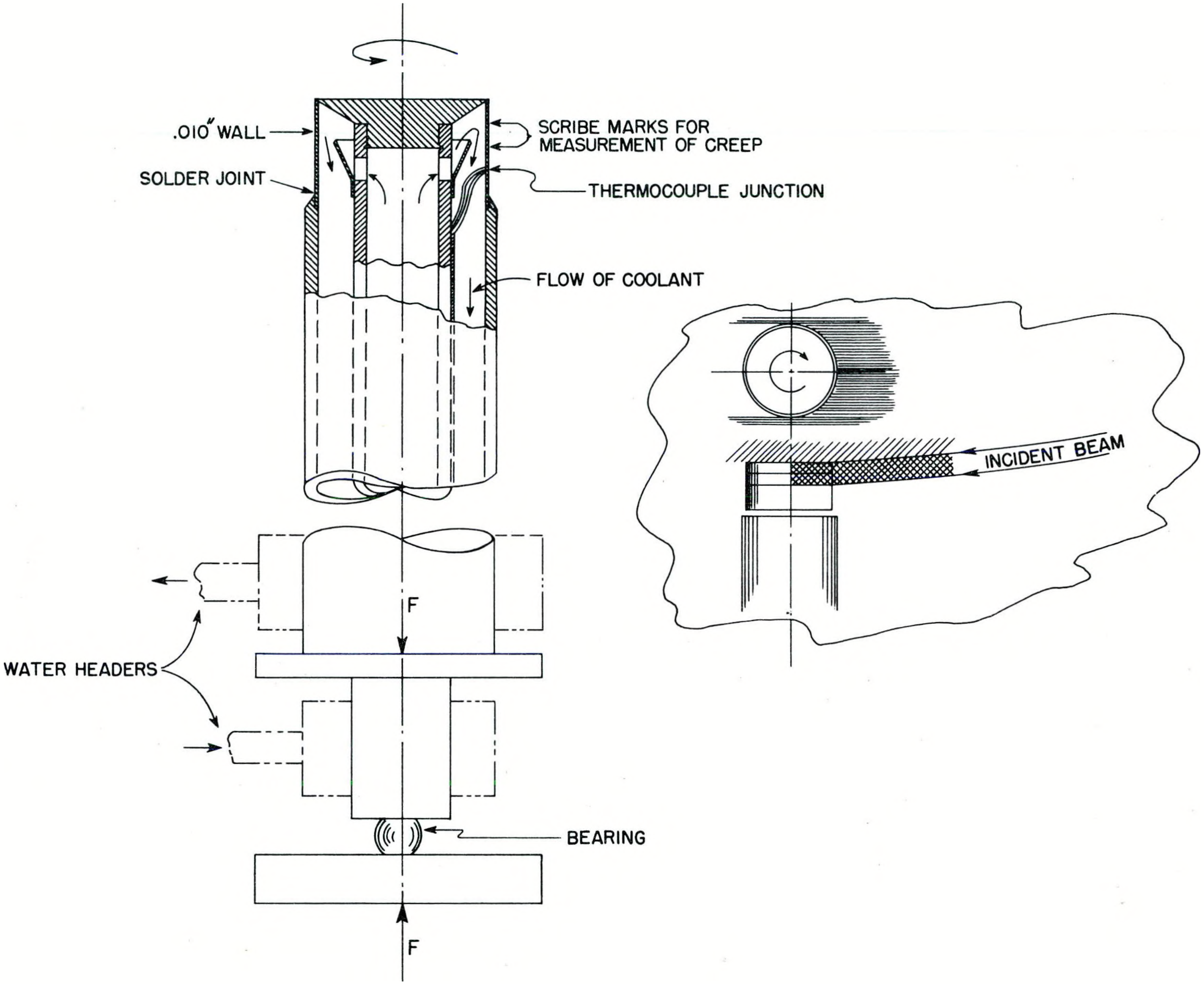

FIGURE 9.1 ROTATING TARGET

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is not expected to affect the target temperature seriously. When running at high temperatures and using liquid metals to heat the target, the power input of the beam is small compared to the power used in the fluid Rotation of the target is achieved by turning the entire assembly through a vacuum seal. The rotation ensures uniform exposure of the target material to the beam.

Controlled stresses must be applied to the target so that creep of the material under bombardment can be measured. The inner "squirt" tube provides not only a path for coolant flow, but also transmits a force which keeps the target area under tension. Creep which may occur under these conditions is determined by marking the target in two places around the circumference. Pictures will be taken of these targets and shown on a calibrated screen. Thermocouples will probably be used for temperature measurements.

In cases in which water-cooled targets are used, the assembly will look essentially the same as that in Fig. 9.1. The water-cooled type of target is under construction; the liquid-metal target is in the design stage. Figure 9.2 illustrates a stationary target to be used for exploration work in the initial stages.

IN-PILE CREEP

Data obtained from the creep of a cantilever beam of type 316 stainless steel at 1050°F under a stress of 4000 psi show at least a transient increase in creep rate upon subjecting a creeping specimen to neutron bombardment in the ORNL reactor.

The tensile creep apparatus described earlier(1) proved erratic, mainly because of pulley friction difficulties, and, furthermore, its removal from the pile was hazardous. The cantilever creep test is not commonly used, but

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(1) *Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending May 31, 1950,* ORNL-768, p. 96 (Aug. 14, 1950).

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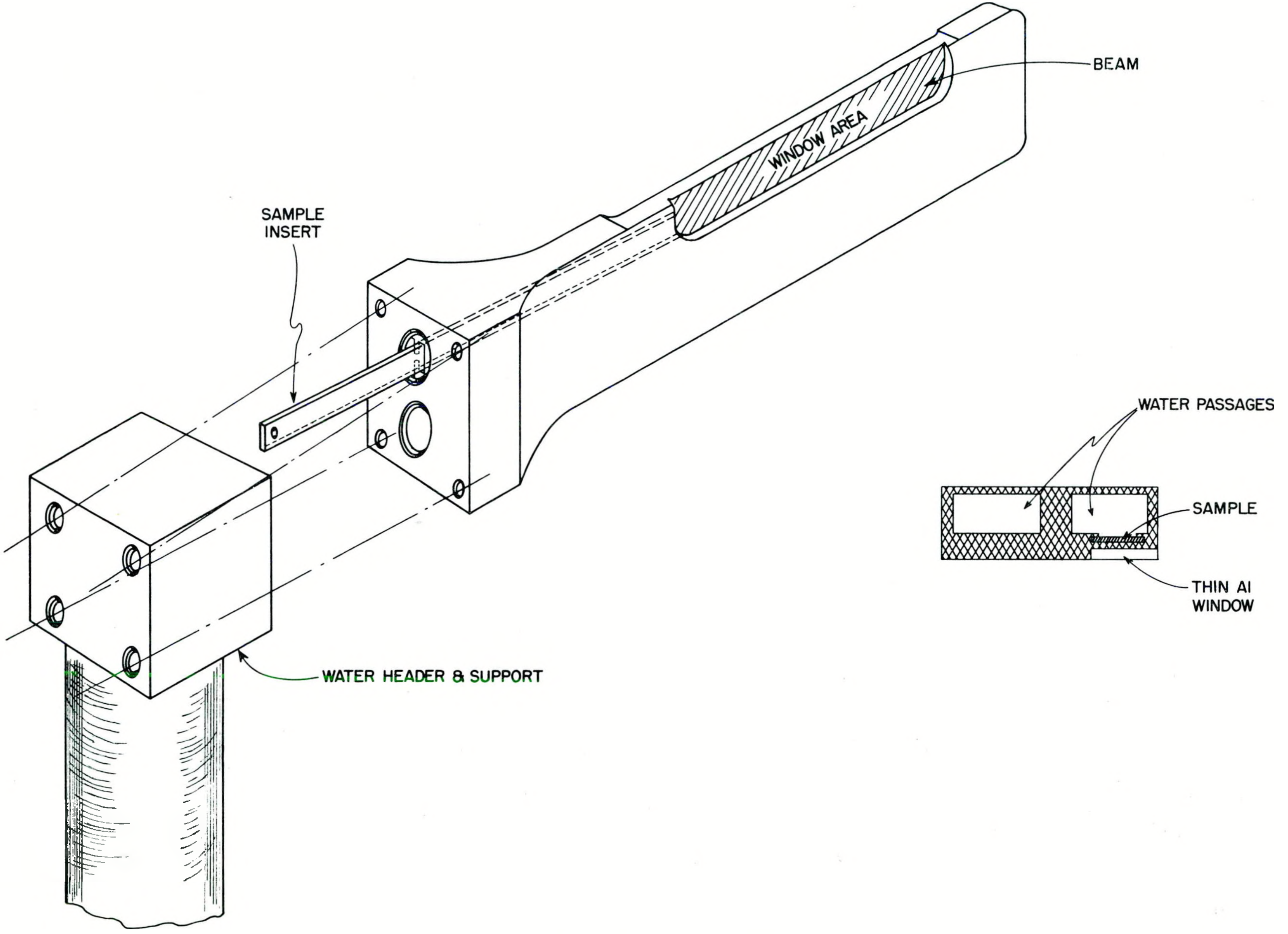

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FIGURE 9.2 STATIONARY TARGET

since some correlation between tensile and cantilever creep is possible,(2) it may be used for the determination of tensile creep. From the standpoint of ease of construction and operation the cantilever has much to recommend it for in-pile operation. A very small weight is all that is required for stressing to any reasonable stress level so that only electrical leads need be run to the apparatus in the pile. The strains are magnified mechanically by the long lever arm carrying the weight; furthermore, the effect of temperature on the apparatus (other than the effect on intrinsic creep behavior) is much smaller than for a conventional tensile creep test.

Figure 9.3 shows a partially assembled and unwired apparatus of the type used in these experiments. Supported on a long aluminum base plate are the test specimen with the furnace windings in place, the weight on the end of the beam, and the microformer for the measurement of displacements of the end of the beam. In the foreground are an unmounted specimen, two lavite furnace supports, and the concentric nickel-foil radiation shields which surround the specimen upon assembly. The specimen, its support on the base plate, and the loading beam are integral. The gauge length (approximately ¾ in.) is the reduced portion extending between the loading beam and the support. The furnace windings are nichrome wound around lavite spacers and through ceramic tubes. The furnace is composed of three separate windings to facilitate removal of temperature gradients across the gauge length. Thermocouples are welded directly to sides of the test bar at three points. Figure 9.4 shows the apparatus itself and an outer furnace tube (controlled temperature) to keep the microformer at constant temperature. Control of this temperature also aids in the control of the test bar temperature by its separate controller. This outer furnace is wrapped with aluminum foil and inserted in the rectangular can for insertion in one of the stringer holes in the ORNL reactor.

The apparatus was charged during an extended pile shutdown. Difficulties in removing the temperature gradient across the bar and burnout of one of the furnace sections prevented operation at the anticipated temperature (1500°F) and caused large temperature gradients across one end of the bar. A deflection (strain) time curve plotted from the data showed an increase in strain rate immediately (within 30 sec) upon starting up the pile (to 4400 kw, fast flux approximately 4×10^{10}). The deflection rate for the half hour after the

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(2) Harris, G. T., and Child, H. C., "Creep Testing by the Cantilever-bending Method," *J. Iron Steel Inst.* 165, 139 (1950).

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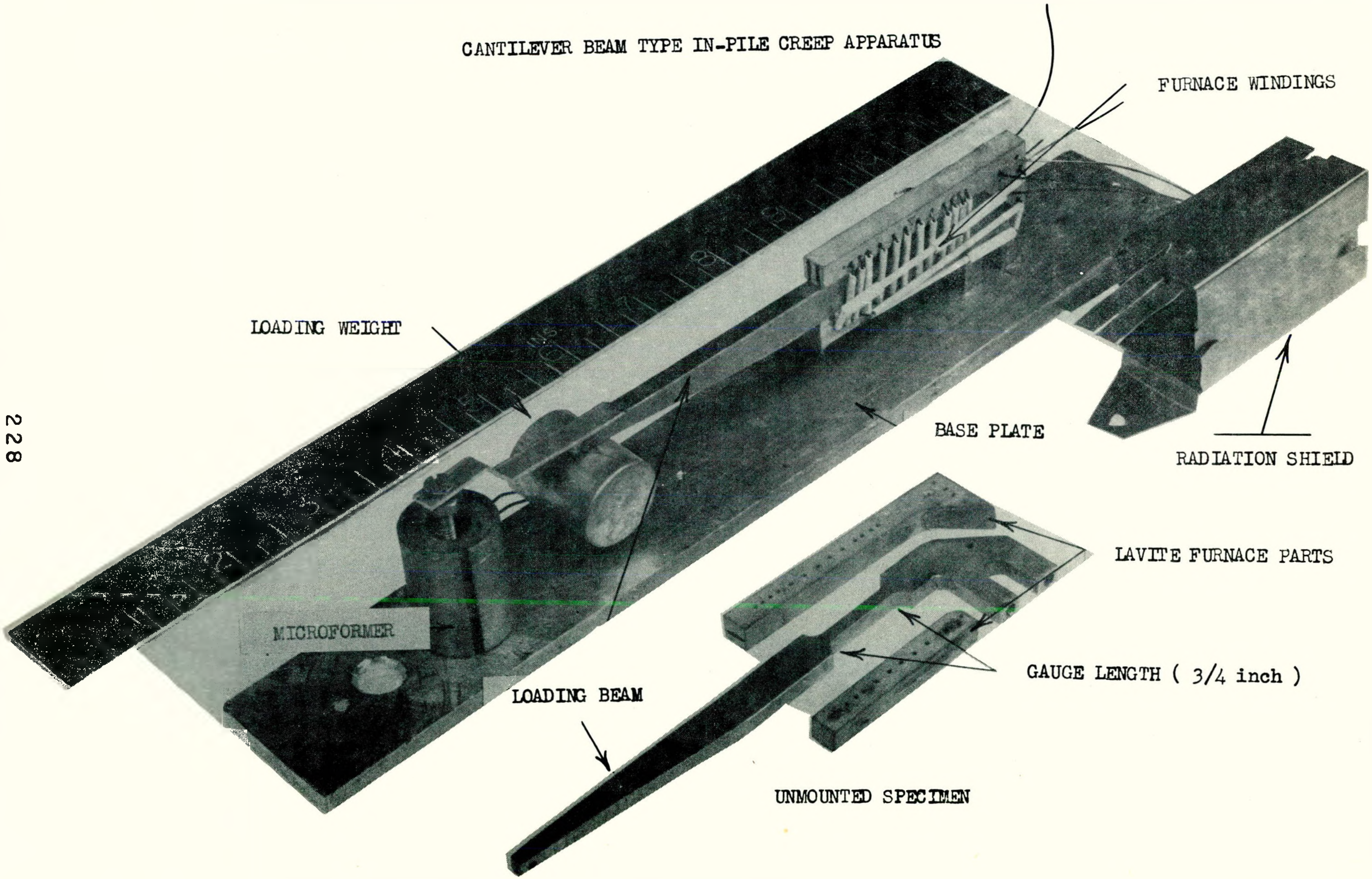

FIGURE 9.3

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CANTILEVER BEAM TYPE IN-PILE CREEP APPARATUS

(WITH FURNACE)

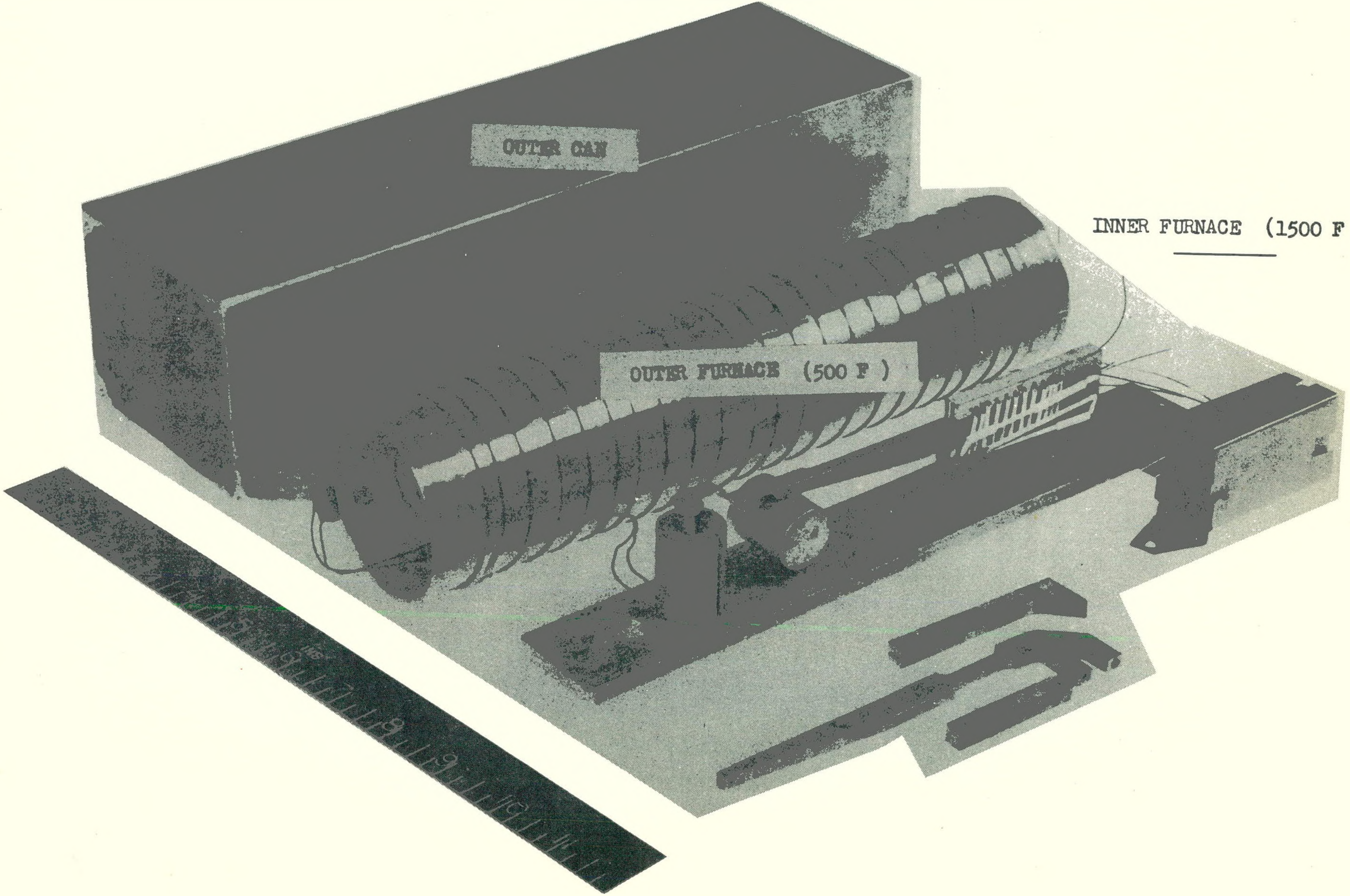

FIGURE 9.4

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pile was started was approximately five times the rate in the half hour preceding (pile off). The temperature gradient across the bar and the present lack of correlation between tensile and cantilever creep preclude any quantitative conclusions.

Although the deflection rate fell off with time, subsequent transient increases in deflection rate were again observed upon applying the flux after each pile shutdown during the following week.

Inasmuch as the stress was applied to the specimen all during the heating cycle from pile ambient temperature to operating temperature, and because the amount of time at constant temperature before the pile was operating was short, the state of the metal (whether in so-called "first" or "second" stage of creep) cannot be described with any accuracy.

There is a possibility that the changes in the deflection rate when the pile is started are being caused by gamma or neutron heating of the specimen. However, the behavior of the controllers and the chart record of the temperature indicate no appreciable heating here. The controller on the outer furnace (although unrecorded) and the controller on the inner furnace showed no abnormal variations at the time of pile start-up. The chart on the controller for the inside furnace was unaffected by pile start-up. Separate temperature measurements at three points on the bar showed variations of less than 2°F during the period covered by the measurements described above. The Duration Adjusting type control unit used is very sensitive to temperature variations, and rates of variation, if any existed, would in all probability show up on the chart as changes in the on-off cycle, provided their magnitude was greater than 2°F.

The data obtained are only qualitative to date, and such variations in rate might not be nearly so marked in tensile creep tests. Some theoretical calculations by F. R. N. Nabarro(3) predict that the greatest effect of neutron bombardment on creep rate will occur when there is a stress gradient in the specimen. Of course, the stress gradients in the cantilever test are greater than would be expected in pure tension of a polycrystalline metal. However, most engineering structures have many points of complex stress distribution where gradients exist, so the observations reported above are not

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(3) Nabarro, F. R. N., *Report on a Conference on Strength of Solids Held at the H. H. Wills Physical Laboratory, University of Bristol, on 7-9 July 1947*, p. 85, Physical Society, 1948.

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without value.

At present a number of bench test rigs are being built to obtain comparative data under conditions free from radiation. The apparatus is being improved to make possible loading of the beam at any time while in the pile and to permit extended operation at 1500°F and higher. Also being investigated are various forms of strain measuring devices, since microformers would be difficult to cool under the conditions of high radiation heating that might be encountered in the MTR

CREEP EXPERIMENTS OF PURDUE UNIVERSITY

The Purdue University group is setting up creep experiments to be run on the Purdue cyclotron. The first materials to be studied will be stainless steel types 316 and 347 and molybdenum.

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(4) *Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending August 31, 1950*, ORNL-858, p. 86 (Dec. 4, 1950).

* See also Section 12 of this report, "Vapor-cycle Reactors," where other corrosion experimentation of North American Aviation is discussed.

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Much of the preliminary work involving the metallography of molybdenum and the construction of furnaces has been completed. In addition an a-c Hall-coefficient apparatus has been designed.

PROPERTIES OF METALS

The Radiation Damage Group has completed some measurements of samples of stainless steel types 304, 309, 316, and 347, nickel "A," and hastelloy C done in conjunction with the ANL—Naval Reactor Program. The samples were exposed at ORNL for a nvt of 1.2×10^{19} (flux 1.2×10^{12}) at 400 to 500°F and a pressure of 1000 psi. Samples were above 400°F for four months. *Before and after measurements of the electrical resistivity, dimensional stability, magnetic permeability, and hardness showed no significant changes.*

OTHER ACTIVITIES

The Radiation Damage Group is actively studying the problem of conducting experiments in the MTR. The gamma-ray flux is so high that considerable heating will apparently take place in all uncooled samples. The problem of conducting suitable experiments under this condition is not a simple one. Plans have been made for activation of the Thermal Conductivity Group by Jan. 1, 1951.

The samples being exposed at Hanford in the ANL—Naval Reactor test loop will be discharged in January, according to the latest estimate.

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10. CHEMISTRY OF LIQUID FUELS

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10. CHEMISTRY OF LIQUID FUELS

W. R. Grimes, Materials Chemistry Division

A previous report(1) has discussed some of the general characteristics of liquid-fuel systems and has described the preliminary phases of the research program designed to produce satisfactory liquids. During the past quarter, interest in such liquid fuels has increased materially. Two types of liquid fuels, suspensions of uranium compounds in sodium hydroxide and solutions of UF_4 in alkali and alkaline earth fluorides, have been under intensive study. It is possible at this time to view with optimism the prospect of such a fuel in the reasonably near future; it is true, however, that it is not possible at this time to state the composition of a liquid which will prove suitable.

It has been demonstrated that all uranium materials tested to date react rapidly with sodium hydroxide to form a finely divided material which is almost certainly sodium uranate. Late in the period a satisfactory method for evaluation of the suspensions in terms of settling rate has been established. Studies designed to improve the stability of these suspensions are currently underway.

The equilibrium diagram of the system $LiF-UF_4$ has been established by thermal analysis techniques, and study of the ternary systems $NaF-BeF_2-UF_4$, $NaF-KF-UF_4$, and $NaF-LiF-UF_4$ is well underway. Of these the $NaF-BeF_2-UF_4$ system appears the most promising in so far as low melting point and reasonable uranium content are concerned. It must be emphasized, however, that a great deal of effort is still required before these systems are established sufficiently to be recommended for use.

Concurrently, the corrosion of metal immersed in $NaF-UF_4$ and $LiF-UF_4$ eutectics has been examined. There is considerable correlation in the corrosive properties of these eutectics, molybdenum, inconel, and hastalloy C (in that order) being the least corroded in the lithium eutectic, and hastelloy C, inconel, and molybdenum the least corroded in the sodium. All other of the many metals under study were far inferior. A test was performed to determine which, and to what extent, fission products react with the proposed eutectic.

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(1) "Chemistry of Liquid Fuel Systems," *The Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending August 31, 1950*, ORNL-858, p. 104 (Dec. 4, 1950).

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SUSPENSIONS OF URANIUM COMPOUNDS IN SODIUM HYDROXIDE

J. D. Redman D. E. Nicholson
L. G. Overholser

Uranium compounds are not appreciably soluble in sodium hydroxide.(2) It has, however, been established that uranium compounds react with sodium hydroxide to produce finely divided materials. The prime objective of this study is to establish those conditions which will produce a suspension of maximum stability in the temperature range 600 to 900°C. The work to date has been concerned primarily with attempts to determine the stability of these suspensions and the identity of the compound or compounds present.

Preliminary Observation of the Suspensions. Suspensions containing 4 to 5% uranium were prepared from various uranium compounds by adding the uranium compound to sodium hydroxide contained in a silver crucible at 700°C. The suspensions were held at 700°C for 30 min, poured into platinum dishes, and allowed to solidify. The button was removed and portions of the material were examined microscopically. The following compounds were used: ammonium uranyl pentafluoride, uranyl nitrate, uranyl acetate, uranyl formate, uranyl tartrate, uranyl pyrophosphate, sodium peruranate, peruranic acid, ammonium diuranate, potassium diuranate, sodium diuranate, and uranium trioxide (commercial product and specially prepared samples from ammonium diuranate and peruranic acid). Suspensions were also prepared from uranium metal by heating with sodium hydroxide at 500°C in an atmosphere of nitrogen. Uranous chloride and uranous fluoride were similarly treated at 500°C.

An evaluation of these different suspensions, based on very rough settling rate observations and approximate particle size determinations by observation with the microscope, indicated that the suspensions prepared from uranium trioxide and those from uranium metal were probably superior to any of the others; the average particle size in these preparations was approximately 2 microns. Actually, none of the suspensions examined had an average particle size much greater than 10 microns and the majority of them appeared to have about equal particle size. The specially prepared uranium trioxides yielded suspensions comparable to those obtained from the commercial uranium trioxide, indicating that the original particle size is not the most important factor involved in preparing these suspensions.

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(2) ORNL-858, *op. cit.*, p. 107.

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By the same technique it was demonstrated that use of carbonate-free NaOH, mixtures of Na_2CO_3 and NaOH, and KOH did not yield suspensions which were superior to those described. Mixtures of Na_2O_2 and NaOH yielded suspensions which had definitely larger particles.

A few experiments were performed in which the suspensions were held at 700°C for 24 hr or longer to ascertain whether such treatment promotes growth of the particles. Evidence was found suggesting that such growth does occur on prolonged heating, but the microscopic examination of these materials was complicated by the silver oxide present.

Settling Rate Measurements. Although the microscopic examinations proved useful in the preliminary work, the evaluation of the suspensions by measuring the settling rate appeared to be of more practical value than the approximate particle size determination based upon microscopic examination. Consequently, studies were made using various techniques for measuring the settling rate of the suspensions.

The first method that afforded any degree of success made use of a stainless steel nipple (1 in. I.D. and 3 to 4 in. in length) provided with a cap having a poured silver disk in the bottom to afford a seal against sodium hydroxide. The nipple, with the cap in place, was loaded with 20 g of sodium hydroxide and heated to 700°C before addition of approximately 1 g of the uranium compound. The suspension was heated at 700°C with frequent stirring for 30 min, removed from the furnace, and air cooled (about 10 min was required for the sample to freeze). The cap was removed from the nipple, and the sodium hydroxide plug was driven out. Samples were cut off the plug by use of a hot copper blade. Analyses of sections of the plug for uranium yielded data, of which those shown in Table 10.1 are representative.

These results confirm the earlier observation that the uranium trioxide suspensions are the most stable of any studied and also show rather conclusively that growth of the particles occurs upon aging for 24 hr at 700°C.

Although this method was capable of measuring relative settling rates, it was not so flexible as desired and might be subject to serious error arising from convection currents. Further studies by this method were, therefore, abandoned upon delivery of a silver reactor. The apparatus being used at

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TABLE 10.1

Settling Rates of Uranium Suspension Measured in Stainless Steel Nipples

COMPOUND ADDED TO NaOH	PERCENT OF COMPOUND FOUND	
	TOP	BOTTOM
UO_3	4.1	4.7
UO_3	4.2	4.5
UO_3	1.3*	6.2*
UO_3	2.8*	6.0*
U_3O_8	1.7	6.8
U_3O_8	3.9	7.1
U_3O_8	2.9	6.7
U_3O_8	3.2	6.1
$Na_2U_2O_7$	3.0	5.3
$Na_2U_2O_7$	2.0	6.5
$(UO_2)_2P_2O_7$	2.0	6.7
$(UO_2)_2P_2O_7$	3.4	4.5
$(UO_2)_2P_2O_7$	3.5	4.0
$(UO_2)_2P_2O_7$	3.3	4.1
$(UO_2)_2P_2O_7$	1.6	4.1

* Digested for 24 hr at 700°C (no agitation).

present for determining relative settling rates is shown in Fig. 10.1. In using this apparatus the temperature is raised to 700°C, 180 g of sodium hydroxide is added, and, after all water is removed, the uranium compound is carefully added and the suspension is agitated by stirring for the desired length of time. After this time has elapsed, the stirrer is cut off and samples are removed after various intervals of time by inserting a silver rod through a hole in the cover. The silver rods are collared to sample the upper one-fourth of the molten sodium hydroxide. After the silver rods are cooled

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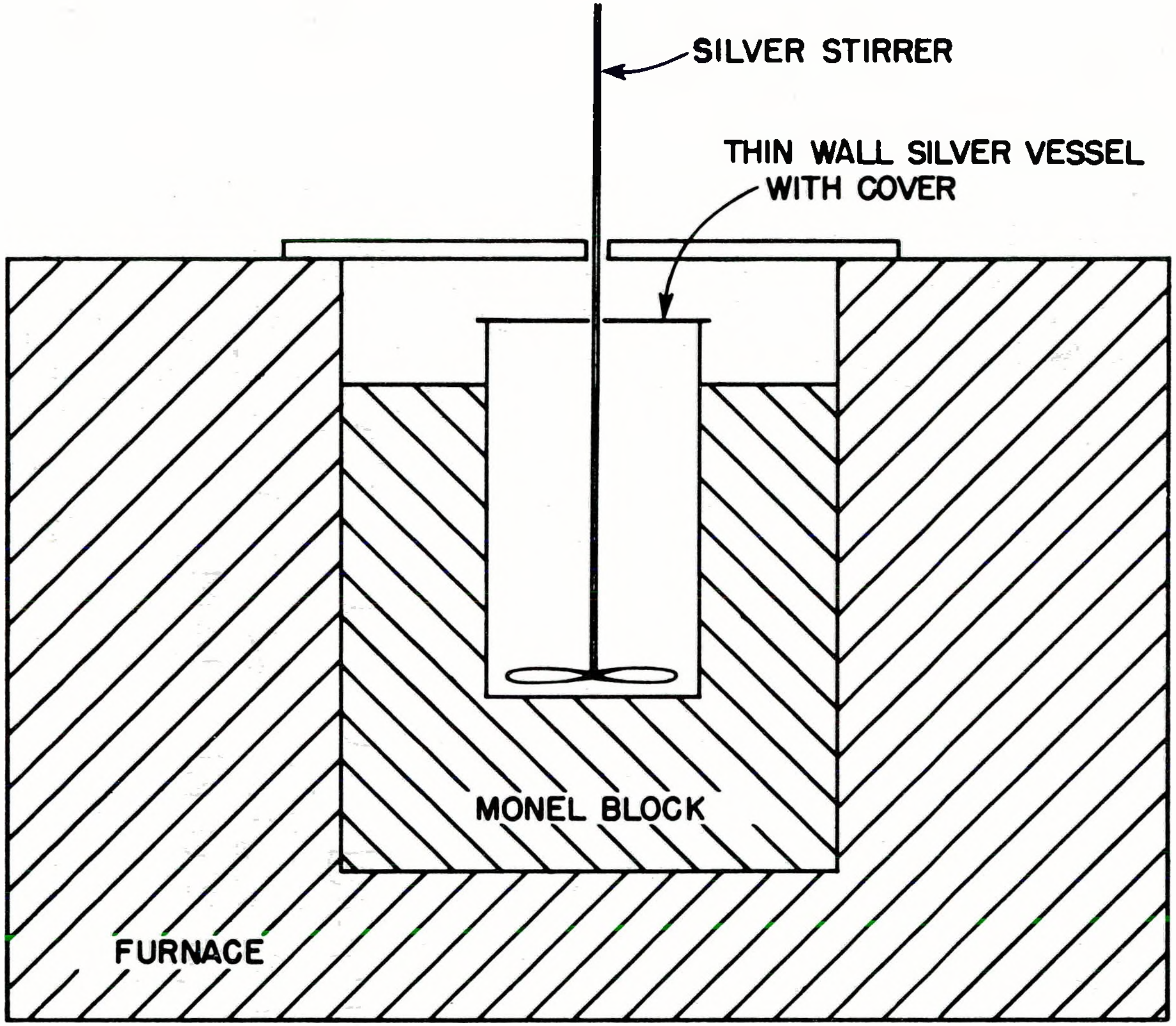

APPARATUS FOR STUDYING URANIUM
SUSPENDED IN MOLTEN SODIUM HYDROXIDE
FIGURE 10.1

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and weighed, the uranium is dissolved and the solution is analyzed for uranium. By this method the system may be digested with agitation for any desired period and may be sampled with a minimum of error due to convection currents in the liquid (Table 10.2).

TABLE 10.2

Settling Rates of Uranium Suspensions Measured in the Silver Reactor

SETTLING TIME (min)	DIGESTION TIME, 700°C (hr)	PERCENT OF COMPOUND FOUND		
		UO_3	$Na_2U_2O_7$	$(UO_2)_2P_2O_7$
1	¼	2.4	0.2	0.3
2	¼	1.7		0.1
3	¼	1.6	0.09	0.2
4	¼	0.5	0.04	0.2
5	¼	0.4	0.09	0.2
10	¼	0.2	0.04	
1	1½	3.1	0.08	0.4
2	1½	2.2	0.06	0.3
3	1½	1.3	0.04	0.2
4	1½	0.4	0.04	0.4
5	1½	0.3	0.07	0.4
10	1½	0.2	0.03	
0	4	3.6	2.6	
1	4	2.3		0.2
2	4	1.2		
3	4	1.0		
4	4	0.5		
5	4	0.4		

The results agree with those obtained by use of the steel nipples inasmuch as they show that the uranium trioxide suspensions settle less rapidly than those prepared from the diuranate or pyrophosphate. However, it will be

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noted that the settling rate for the uranium trioxide suspensions measured in the silver reactor is greater than that observed by the nipple method. For the best suspension studied in the silver reactor, the upper one-fourth of the molten sodium hydroxide is virtually free of uranium after 3 to 4 min as contrasted to the relatively uniform distribution found in the nipples after 10 min. This suggests that convection currents played an important role in prevention of the settling in the earlier studies.

It is significant to note, however, that only very slight agitation is required to keep the particles well suspended.

Identity of the Uranium Compound. For simplicity, repeated reference has been made in the preceding discussion to uranium trioxide suspensions although the uranium is certainly not present as the trioxide. A program for identification of the uranium compound is still incomplete.

The X-ray diffraction patterns from caustic buttons prepared at 700°C with uranium trioxide or sodium diuranate are identical, but they differ considerably from the patterns for uranium trioxide or sodium diuranate. Caustic buttons (prepared with uranium trioxide), when extracted with anhydrous ethyl alcohol, yield a residue of sodium diuranate; the alcohol extraction alters the X-ray diffraction pattern, indicating that the uranium compound originally present in the caustic cannot be isolated by this technique. Samples of sodium diuranate have also been prepared by precipitation of uranyl nitrate from methyl or ethyl alcohol by sodium hydroxide. Precipitation from ether failed to give the diuranate.

It is believed that sodium monouranate is present in the caustic buttons, but the X-ray diffraction pattern for the monouranate is not available for comparison. Vacuum distillation of the excess sodium hydroxide from the suspensions is being tried, using the apparatus shown in Fig. 10.2. A suspension prepared from uranium trioxide was heated at 800 to 850°C [vapor pressure of $NaOH(l)$, 4.0 mm Hg at 800°C] for approximately 50 hr. The sodium hydroxide distilled off leaving an orange residue in the silver crucible, but this residue was contaminated by particles from the stainless steel pipe fittings and by sodium hydroxide which had fallen back into the crucible. Another run is planned using a cover over the silver crucible to prevent this contamination. Results of X-ray analysis on the first sample are not available, and identification of this orange residue has not as yet been accomplished.

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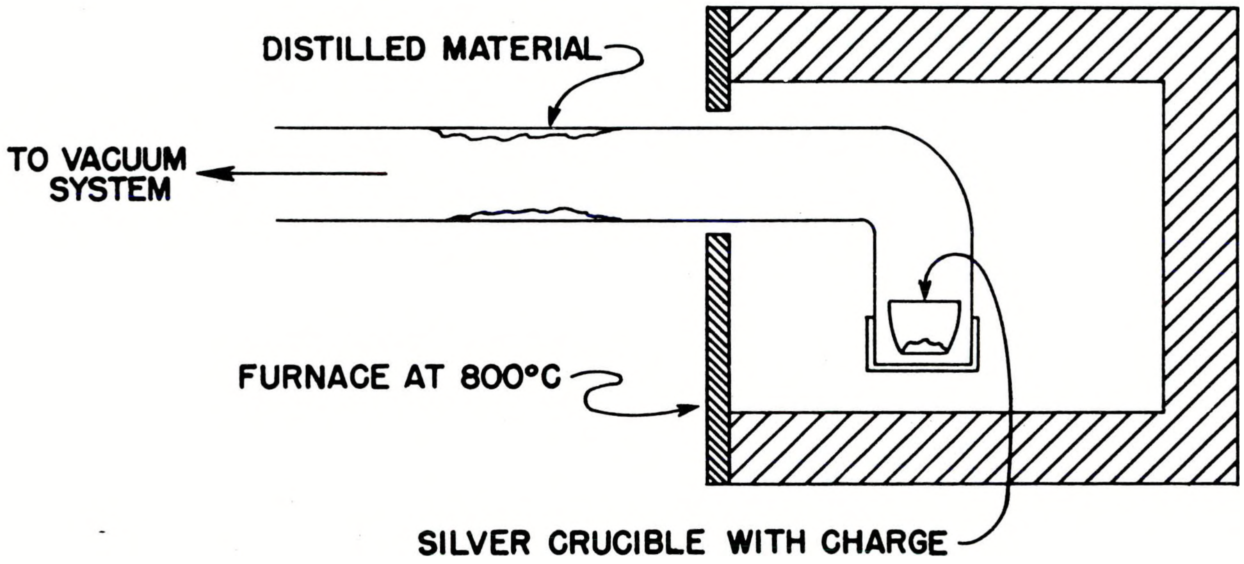

VACUUM DISTILLATION OF SODIUM HYDROXIDE

FIGURE 10.2

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LOW-MELTING FLUORIDE SYSTEMS

R. E. Moore J. P. Blakely
G. J. Nessle C. J. Barton

Studies of molten fluoride systems which might serve as fuels have been continued in the directions indicated in the last quarterly report.(3) During the past quarter the equilibrium diagram for the UF_4-LiF system has been established, and considerable progress has been made on the ternary systems UF_4-NaF-BeF_2, UF_4-NaF-KF, and UF_4-NaF-LiF. The equilibrium diagrams for these systems are, however, still far from complete, and it is not as yet possible to state the precise composition of a fuel which would be completely satisfactory from a phase stability point of view. Although the UF_4-NaF-BeF_2 system has shown most promise in preliminary studies, it is not yet certain that this system is best for the purpose. The general characteristics of the systems studied are discussed briefly under individual headings below.

Experimental Methods. The conventional techniques of thermal analysis have been used in virtually all the experiments to date. The procedures and materials are essentially as described earlier(4) although the actual apparatus used has undergone considerable modification since that report was issued.

The melts are still contained in high-density graphite crucibles, but the graphite stirrer—thermocouple well combinations have been replaced by motor-driven stirrers of monel. These stirrers are drilled to serve as a thermocouple sleeve from which the couple is insulated by a ceramic tube. Lubrication of the ceramic tube with graphite permits the stirrer to turn freely without twisting of the thermocouple wire. The stirrer is rotated at about 150 rpm by means of a belt drive which slips without injury to the equipment when the melt solidifies. A graphite sleeve bearing supports the stirrer shaft to facilitate smooth operation of the stirrer.

Cooling curves have usually been recorded by automatic pen-type recorders of the Brown or Leeds and Northrup variety. Manual recording of the curves using type K potentiometers is used to verify points of special importance.

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(3) ORNL-858, *op. cit.*, p. 104.

(4) *Ibid.*, p. 110.

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Efforts have been made to separate by filtration the phases present under equilibrium conditions. While this has been accomplished at temperatures below 420°C in the NaF-BeF_2-UF_4 system, apparatus difficulties have so far precluded general application of the method.

At low temperatures the melts are essentially without effect on glass equipment, provided the samples are well dried and outgassed. Systems which contain sufficient amounts of liquid near or below 400°C can, therefore, be separated easily by use of the apparatus shown in Fig. 10.3. The materials, which have previously been melted under an inert atmosphere, are placed in the apparatus on the sintered glass disk, and the entire assembly is carefully evacuated. The sample is then heated to the desired temperature while the pumping is continued. After 1 to 2 hr at the desired temperature helium is admitted to the apparatus to accomplish the filtration. The liquid phase that accumulates in the lower tube is removed for analysis.

Apparatus of similar design but with sintered stainless steel filters and of all-metal construction should permit general application of this direct procedure.

Properties of the Fluoride Systems. *Lithium Fluoride—Uranium Fluoride.* This system has been investigated in considerable detail by thermal analysis, and some information has been obtained from X-ray diffraction photographs. While some further study would be necessary to establish with certainty the identity of the compound with incongruent melting point, the main features of the diagram (Fig. 10.4) are well established. Only one eutectic exists in this system; this material, containing 26 mole % UF_4 and melting at 490°C, would be quite interesting as a possible fuel mixture if separated lithium isotopes were available. The compound of incongruent melting point is probably LiF-$3UF_4$ although LiF-$2UF_4$ is a possibility. The compound does undergo a phase transition at about 600°C.

NaF-BeF_2-UF_4 System. The binary NaF-BeF_2 system has been reported as showing a eutectic at 45 mole % BeF_2 which melts at about 360°C. Such a mixture prepared in this laboratory has been shown to melt at 342°C. The purity of the only beryllium fluoride available here is unknown; it is, therefore, quite possible that impurities are responsible for this discrepancy. The extremely low melting point available has encouraged studies of the

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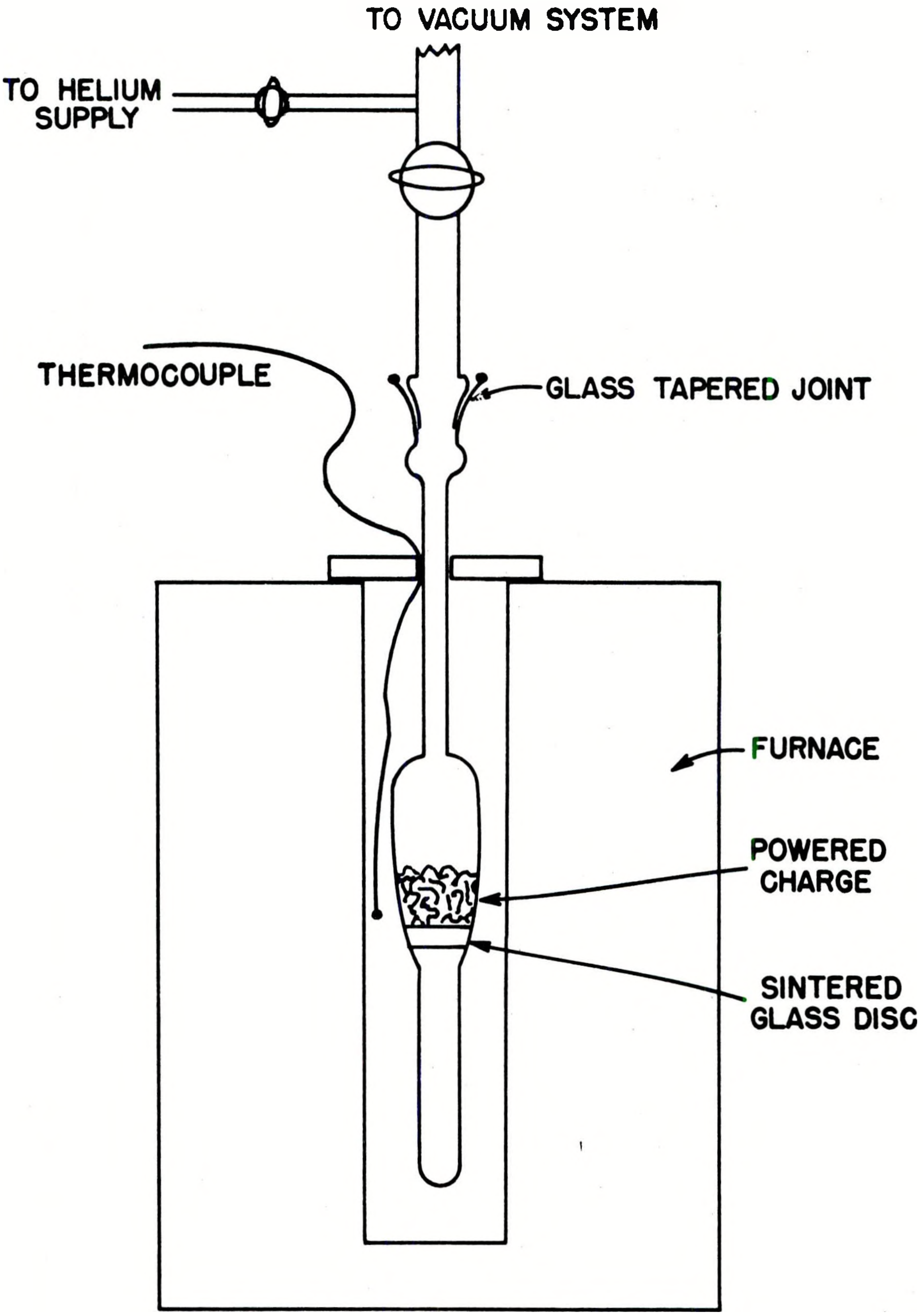

FILTRATION OF LOW-MELTING SYSTEMS

FIGURE 10.3

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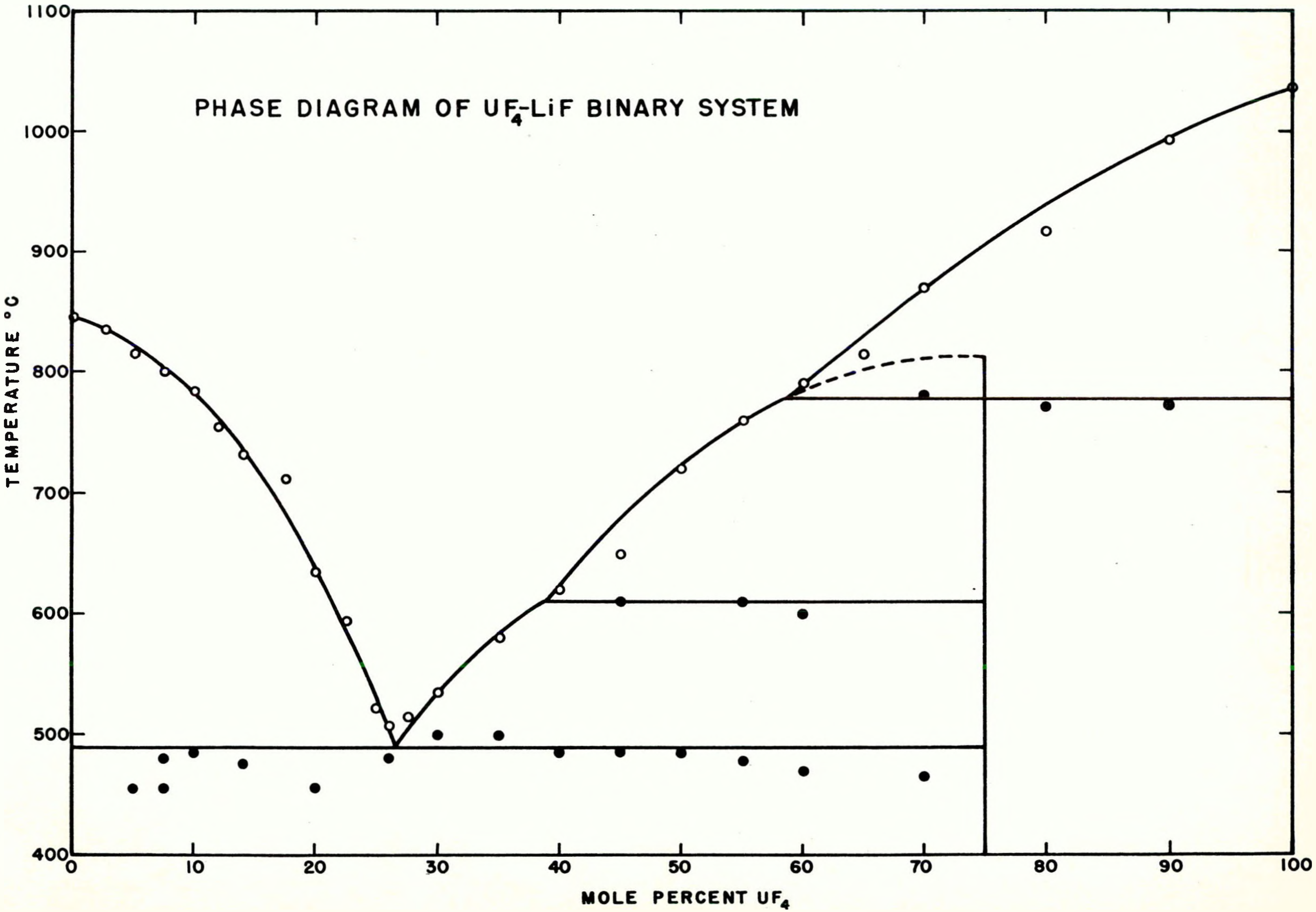

FIGURE 10.4

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solubility of UF_4 in this material. It must, however, be recognized that the vapor pressure of BeF_2 at high temperatures is not known and may be relatively high (perhaps as high as a few atmospheres at 1000°C). The well-recognized toxicity of beryllium compounds, especially in the form of air-borne dusts, is a factor which necessarily makes the study a time-consuming one.

Study of this ternary system has been attempted by thermal analysis and by the filtration techniques described above. It has been demonstrated that uranium fluoride dissolves in mixtures of NaF and BeF_2 even at temperatures below 340°C. Unfortunately, however, the uranium content of this liquid phase is less than 5% by weight. Since the volume available for an unmoderated fuel in reactors of current design is limited to about 2 cu ft, such dilute solutions of uranium are of little interest. It is likely that, using reasonable estimates for the density of such liquids, liquids of about 50% UF_4 by weight are required.

Thermal analyses of such mixtures, in the range 10 to 20 mole % UF_4, are far from complete. Such studies as have been completed, however, indicate that mixtures containing 15 to 18 mole % UF_4 and 35 to 45 mole % BeF_2 are completely liquid below 550°C. It is extremely likely that the optimum composition has not yet been reached and that the goal of critical amounts of uranium in 2 cu ft with melting point below 500°C may be realized.

NaF-LiF-UF_4 System. The relatively low temperature (490°C) obtained in the LiF-UF_4 binary system and the known binary eutectics of NaF-KF (640°C) and of NaF-UF_4 (600°C) have encouraged the study of this ternary system. A large number of compositions within this system have been checked by thermal analysis methods. Not all regions of this diagram have been explored as yet. It would appear that the lowest melting point in the system is at a ternary eutectic at 26 mole % UF_4, 48 mole % LiF, and 26 mole % NaF. Material of this composition shows only the eutectic halt at 450°C. It appears that in regions considerably removed from this composition, i.e., 25 mole % UF_4, 45 mole % NaF, and 40 mole % LiF, a new ternary eutectic of higher melting point is observed. A large number of compositions must be investigated before this could be definitely stated.

The uranium concentration of the 450°C ternary eutectic is sufficiently high and the melting point sufficiently low to make it of considerable interest, provided the separated lithium isotopes are available.

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NaF-KF-UF_4 System. Studies of this ternary system have been initiated quite recently, and, to date, the few cooling curves available are insufficient to characterize the system. It seems likely, however, that the lowest melting points available are higher than those indicated for the two ternary systems discussed previously. In spite of this fact, there is evidence of a ternary eutectic at 505°C, and samples containing up to 25 mole % UF_4 have remained liquid to 550°C. Determination of the eutectic composition is the object of present efforts.

CORROSION TEST OF METALS

P. J. Hagelston, Isotope Research and Production

The first corrosion test was made to determine the effects of NaF-UF_4 eutectic on various metals and to select the least susceptible for the next test. The samples were cut from sheet materials approximately 1½ by 2 in., weighed, and placed in a reactor containing the molten eutectic at 700°C for approximately 160 hr. The following samples were tested:

Durimet
International Ni Special
Illium G
Hastelloy A
Hastelloy B
Hastelloy C
Hastelloy D
Illium R
Tantalum
Inconel X
Inconel
Stainless steel 316
Nickel
Tungsten
Molybdenum
Monel
Stainless steel 304
Platinum
Stainless steel 347
Zirconium

It was found that tantalum and zirconium disintegrated. Some of the samples lost weight from erosion while others gained weight from penetration by the salts.

The test with LiF (74 mole %) — UF_4 (26 mole %) eutectic was made using samples from the following materials:

Hastelloy A
Hastelloy B
Hastelloy C
Nickel
Inconel
Molybdenum
Illium R
Stainless steel 316
Stainless steel 321
Stainless steel 347
Ticonium

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The samples were placed in a graphite box filled with molten eutectic which was maintained at 700°C. The graphite box was surrounded by a tightly lidded mild-steel "shoe box" through which argon was circulated. The samples were removed and weighed after 24, 68, 134, and 204 hr run time. It was necessary to replenish the eutectic each time the run was interrupted, owing to the creeping of the mixture over the sides of the container. During the final run (134- to 204-hr period), the mild-steel shoe box oxidized and the samples were exposed to the atmosphere. The test was therefore terminated.

It was found that some materials showed an initial weight gain which gradually diminished with time. Scale formations were not removed. In the order of initial weight gain, starting with the least weight gain, these materials were:

Molybdenum
Inconel
Illium R and hastelloy C
Ticonium
Hastelloy A
Nickel
Hastelloy B

Stainless steels 316, 321, and 347 all showed a loss in weight which increased with time. Listed in the order of weight loss, least weight loss first, they are as follows:

Stainless steel 316
Stainless steel 321
Stainless steel 347

The attack on 347 stainless steel represented the extreme case among those tested. Figures 10.5, 10.6, and 10.7 picture this attack.

In order to compare the corrosion effects of $LiF-UF_4$ and $NaF-UF_4$ on identical samples, two tests were started on November 30 using the following materials in interior containers made of hastelloy C instead of graphite:

Stainless steel 304 (extra low carbon)
Stainless steel 316 (extra low carbon)
Stainless steel 304
Stainless steel 316
Stainless steel 321
Stainless steel 347
Stainless steel 321
Stainless steel 405
Stainless steel 410
Stainless steel 430
Stainless steel 446
Allegheny 590
Molybdenum
Inconel
Hastelloy C

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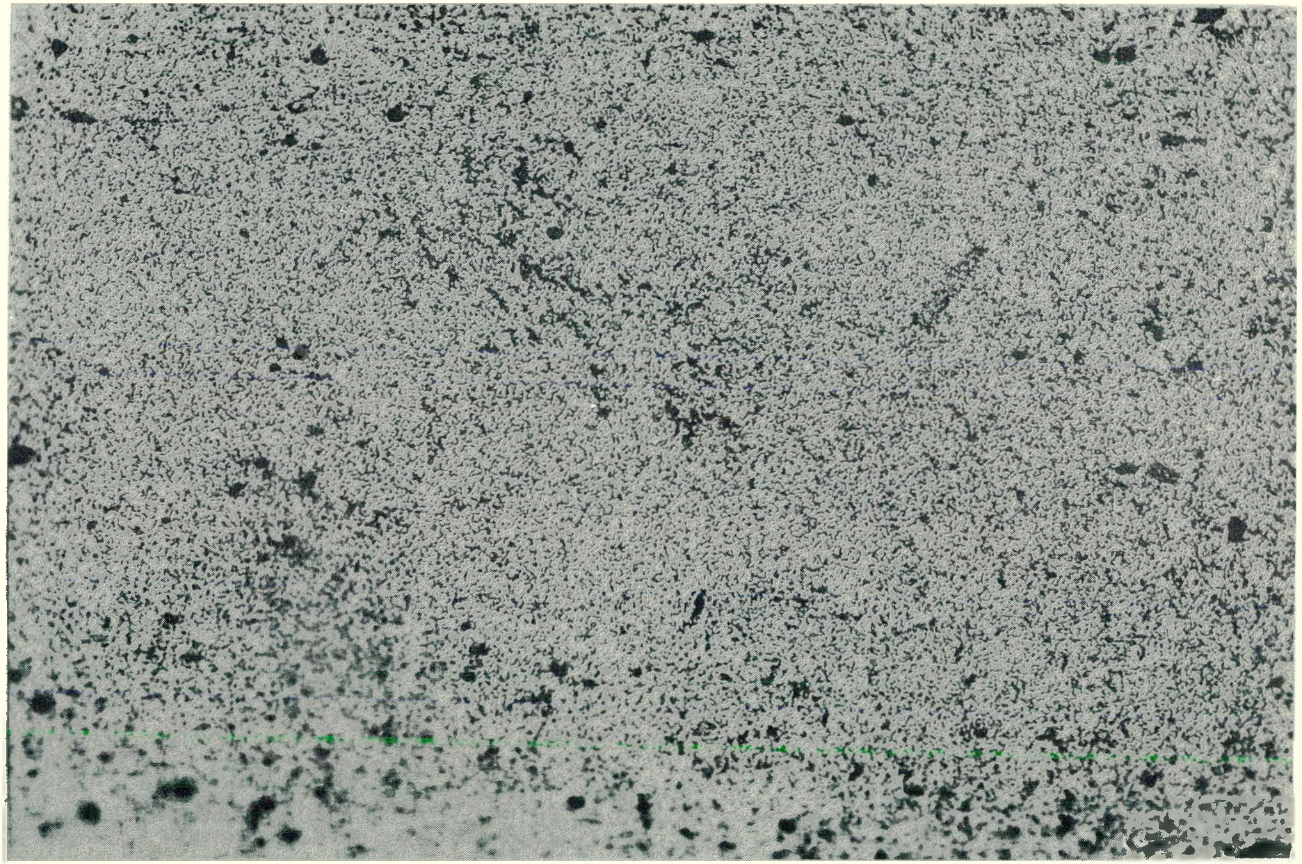

347 STAINLESS STEEL PRIOR TO TEST

FIGURE 10.5 347 STAINLESS STEEL (UNEXPOSED)
(mag. 100X)

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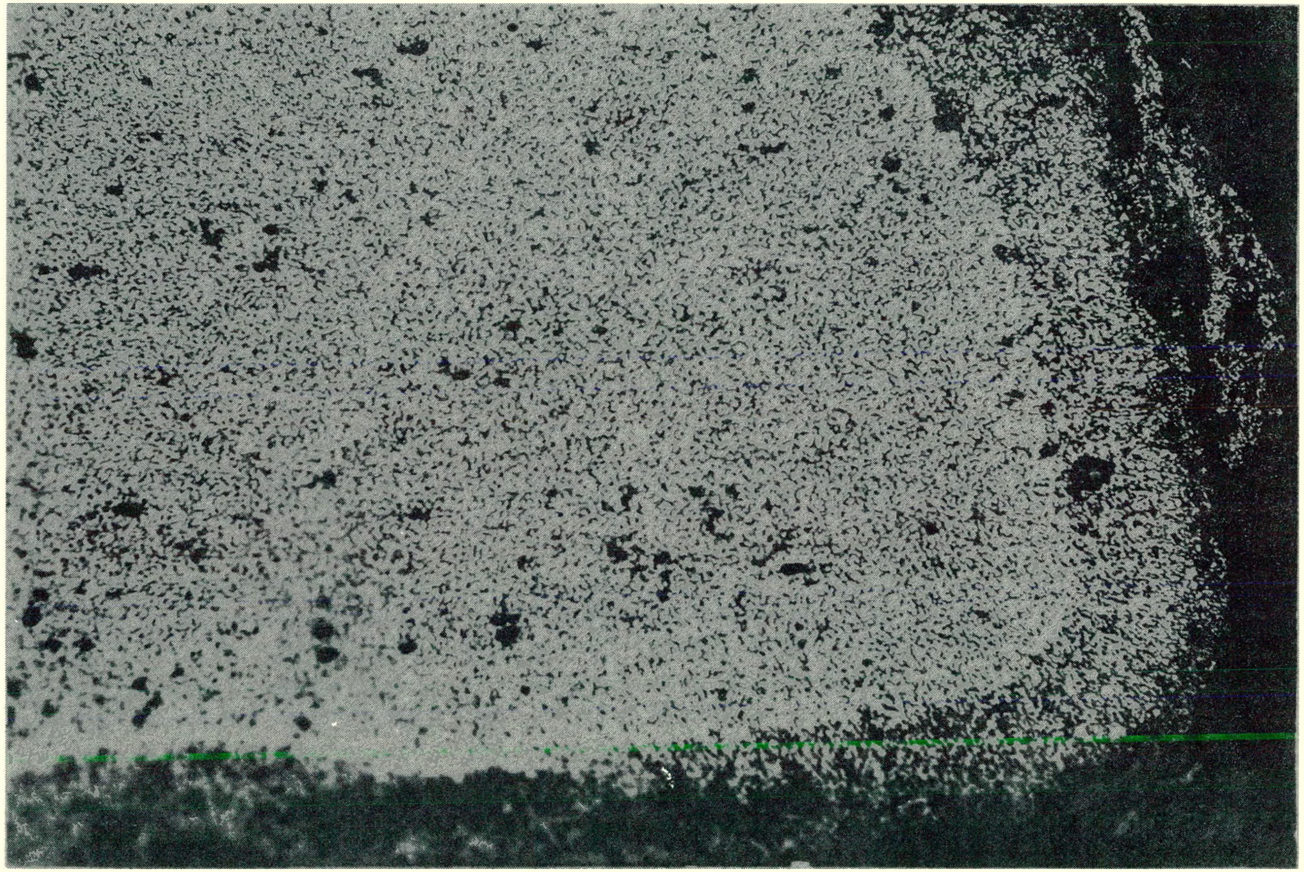

347 STAILESS STEEL, SHOWING ATTACK (DARK OUTER EDGES) AFTER 24 HOURS EXPOSURE TO $LiF-UF_4$ EUTECTIC

FIGURE 10.6 347 STAINLESS STEEL (24 HR. EXPOSURE) (mag. 100X)

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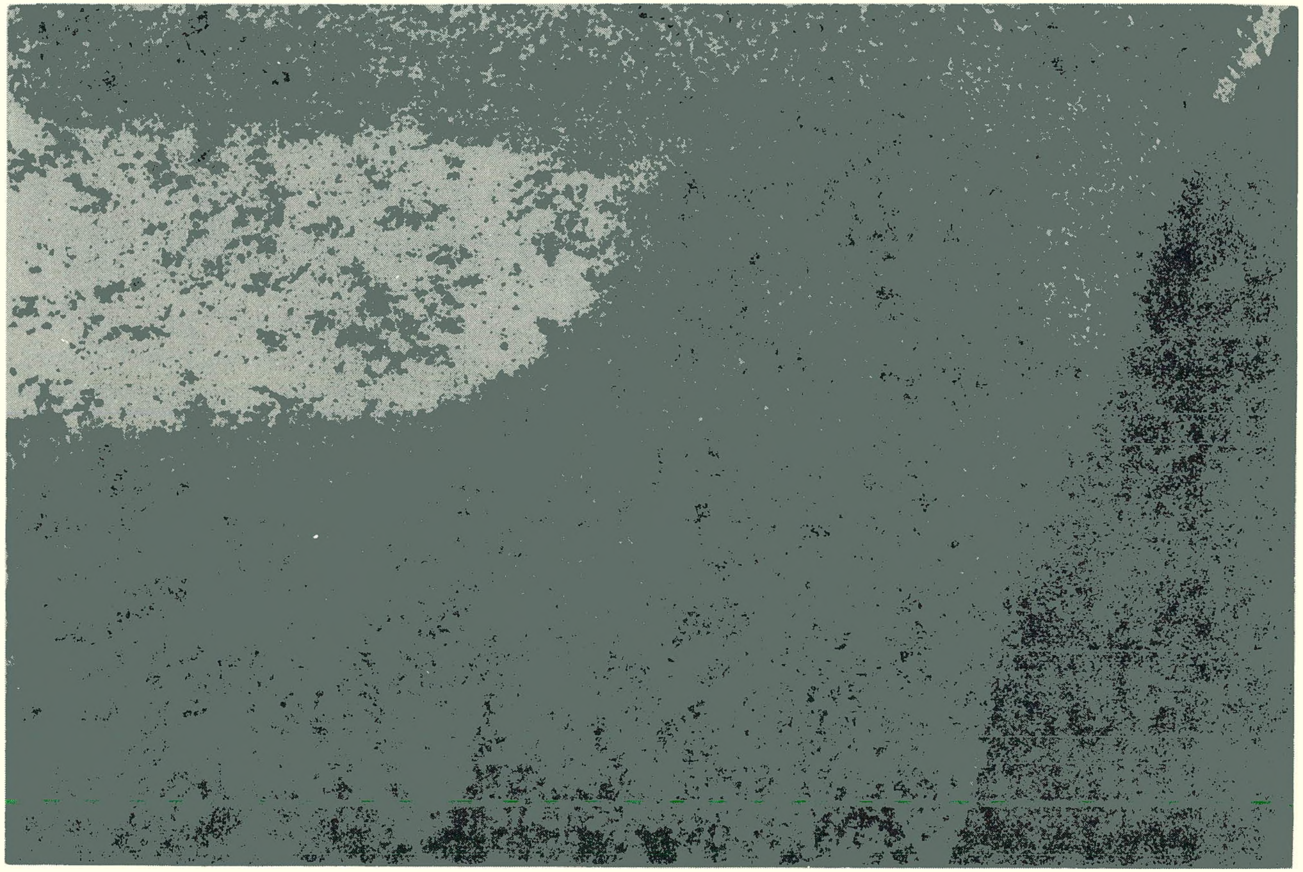

347 STAINLESS STEEL, SHOWING ATTACK AFTER 135 HOURS EXPOSURE TO $LiF-UF_4$ EUTECTIC

LIGHT SECTION IS UNATTACKED METAL

FIGURE 10.7 347 STAINLESS STEEL (135 HR. EXPOSURE) (mag. 100 X)

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Both of these runs were terminated because of oxidization of the mild-steel shoe boxes when the argon lines were plugged by condensation of the vaporized eutectics. The test with LiF-UF_4 eutectic failed after 113 hr, and the one using NaF-UF_4 eutectic failed after 71.25 hr. The samples were removed and cleaned by first dipping into a molten mixture of NaCl and Na_2CO_3 and then into cold water. After washing in water, the samples were free from scales and a fairly accurate weight loss was obtained. Results are summarized in Table 10.3.

TABLE 10.3

Comparison of Material and Eutectic

MATERIAL	WEIGHT LOSS (mg/dm^2/day)	
	LiF-UF_4	NaF-UF_4
Mo	26.1	64.4
Inc	68.5	−44.7*
HaC	198.5	19.67
Al-590	607.0	514.0
304 SS (ELC)	1698.0	876.0
316 SS (ELC)	1732.0	1138.0
347 SS	1850.0	856.0
321 SS	2475.0	1482.0
304 SS	3270.0	2050.0
316 SS	3490.0	2070.0
410 SS	5000.0	3360.0
430 SS	6510.0	3670.0
405 SS	7175.0	4830.0
446 SS	7280.0	5470.0

* Weight gained.

CORROSION TEST OF CERAMICS AND FISSION PRODUCTS

R. O. Hutchinson

A test was performed to determine which, and to what extent, of the better than 1% yield of fission products reacted with the liquid fuel. When the

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working fluid is decided upon for the ARE, experiments may be conducted to determine what effect the fission products present in the system will have on the various materials of construction under consideration. Some ceramics were tested to determine which would be least attacked by the liquid fuel at 700°C.

The better than 1% yield of fission products desired as samples were rubidium, strontium, yttrium, zirconium, columbium, molybdenum, technetium, ruthenium, rhodium, palladium, cesium, barium, lanthanum, cerium, praseodymium, neodymium and samarium. Of these metals, only six were found in a slug form. The others were either not available at all or were very expensive. Zirconium, neodymium, palladium, hafnium, barium, and strontium were used. The last two, because of their known activity, were first tested by dropping into a boat containing molten UF_4-LiF eutectic. A spontaneous reaction occurred with both metals.

The remaining four metals were immersed in a boat of molten eutectic and heated up to 1300°F. They were then suspended on molybdenum wire and lowered into the salts. Beryllium metal, BeO, Zircon, lava, and graphite were treated in the same manner. The beryllium metal began immediately to change the salts to a red-looking slag. The slag formed a coating over the zirconium and hafnium, but the palladium and neodymium were gone. The zirconium and hafnium gained about 1 g owing to the slag formation.

The second run was the same as the first except for the ommission of beryllium metal. The four metals were removed at the end of 24 hr. The neodymium and palladium were completely dissolved, and the hafnium metal was still in original shape but completely carbonized when the sample was crushed. The zirconium was also completely dissolved.

Information regarding the weights of samples after 24 hr of immersion is given in Table 10.4.

TABLE 10.4

Corrosion Tests of Ceramics and Fission Products

SAMPLE	WEIGHT (g)		OBSERVATION
	IN	OUT	
FIRST RUN			
Hf	25.4592	26.7331	Slag formed over
Nd	7.8377		90% or better dissolved
Pd	2.4264		90% or better dissolved
Zr	13.8474	12.9677	Slag formed, easily removed
SECOND RUN			
Hf	21.1092		Carbonized 100%
Nd	7.7303		90% dissolved
Pd	1.1102		90% dissolved
Zr	7.7169		Completely dissolved

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The molybdenum wire used in the suspension of samples was noted to be much more ductile after subjection to extreme heat and immersion in the bath. After repeated bending, it still remained soft.

The following ceramics were also tested in LiF-UF_4 eutectic at 700°C: .

Graphite
CaF_2
Lava A
Zircon
ZrO_2
Al_2O_3
MgO

After 22 hr CaF_2, ZrO_2, Al_2O_3, and MgO showed signs of complete attack and were eliminated from further test.

A new test was started using graphite, lava, and Zircon, At the end of 90 hr the zircon was completely dissolved and the lava badly eroded. The graphite showed no signs of attack.

After 190 hr the test was terminated; it was found that graphite had lost 1.15% of its original weight,
and the lava was about 75% dissolved.

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11. ANALYTICAL CHEMISTRY

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11. ANALYTICAL CHEMISTRY*

C. D. Susano, Analytical Chemistry Division

The Analytical Chemistry Division (Y-12 Section) has assisted in the ANP program by performing service analytical work and by carrying out development, as required, of methods for use, principally, in analyzing materials of construction, liquid-metal coolants, and materials for use in fuel elements. Summaries of the development and analytical activities of this Division which are concerned with the ANP project are here presented.

DETERMINATION OF OXYGEN IN SODIUM

J. C. White and W. J. Ross

Although the method of Pepkowitz and Judd(1) has been adopted for the routine determination of oxygen in sodium, it has not proved entirely satisfactory with regard to precision and it does not permit a spectrographic analysis to be made on the same sample. A search is being made for a reagent other than mercury which will take sodium into solution but will not react with or dissolve sodium monoxide. Of 25 organic liquids tested for this purpose, all except two (ethyl acetoacetic ether and acetonyl acetone) have been eliminated on the grounds of unreactivity with sodium, lack of selectivity of action, high viscosity, or production of insoluble products with sodium. Ethyl acetoacetic ether and acetonyl acetone, together with other as yet untried reagents, will be further investigated.

An additional assembly of the Pepkowitz and Judd type for the determination of oxygen in sodium has been set up and tested. The equipment consists of a purification train and two reactors.

Preparation of Standard Samples (J. C. White and C. M. Boyd). Difficulty was encountered in preparing suitable samples to check the accuracy attained in the determination of sodium monoxide in sodium by the method of Pepkowitz and Judd. Samples were prepared by drawing molten sodium into an evacuated glass vial containing a weighed amount of mercuric oxide. The reaction between sodium and mercuric oxide to produce sodium monoxide was so vigorous that it shattered the glass vial.

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* This section, except for minor editorial changes for conformity with other sections, is the same as Report Y-B31-221.

(1) Pepkowitz, L. P., and Judd, W. C., "Determination of Sodium Monoxide in Sodium," *Anal. Chem.* 22, 1283 (1950).

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Samples were next prepared by adding sodium and mercuric oxide in separate glass vials to the reactor of the apparatus. After the addition of mercury, the vials were broken and the reaction was allowed to proceed inside the reactor. Test results for samples formed in this manner indicate that the reduction of mercuric oxide is complete and that recovery of the added oxygen is possible. Representative results were as follows:

OXYGEN ADDED (%)	OXYGEN FOUND (%)
0.13	0.14
0.057	0.065
0.17	0.14

Work is continuing on the improvement of the accuracy and precision attainable by the Pepkowitz and Judd method.

Time Study of the Pepkowitz and Judd Method (J. C. White). A detailed study was made of the Pepkowitz and Judd method with regard to the actual time spent in the various operations when the method was performed on a routine basis. It was possible, on the basis of the study, to make several changes designed to speed up the operation. It was concluded that one man could be expected to do eight determinations per day and two men sixteen per day with apparatus consisting of two purification trains and four reactors. A report(2) was written in which the results of the time study, along with a detailed procedure, were given.

MICROANALYSIS OF SILICON CARBIDE

J. C. White and W. J. Ross

Some fourteen samples of irradiated silicon carbide have been submitted for the determination of free silicon and silicon dioxide. Since the average weight of these samples is of the order of 50 mg, a microanalytical method must be used.

A method described in the literature, based on the removal of SiO_2 by treatment with HF and H_2SO_4 and the subsequent removal of silicon by treatment with HF, H_2SO_4, and HNO_3, was shown to be unsuitable. Silicon is not inert to

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(2) White, J. C., *Suggested Procedure for the Determination of Oxygen in Sodium by the Pepkowitz and Judd Method*, Memorandum to Dr. R. Rowen, Analytical Chemistry Division, Oak Ridge National Laboratory, Y-12 Site (Nov. 17, 1950).

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HF and H_2SO_4, as was supposed by the authors of the method, but undergoes continuous loss in weight

A method has been developed based on the removal of silicon as $SiCl_4$ by reaction with dry chlorine at 350°C and the subsequent removal of SiO_2 by treatment with HF and H_2SO_4. Repeated tests have established that silicon can be quantitatively removed from known synthetic mixtures of the metal and the carbide by this procedure.

Test results representative of those obtained in the analysis of approximately 5-mg portions of the NEPA samples are listed in Table 11.1

TABLE 11.1

Analyses of Silicon Carbide Samples for Silicon and Silicon Dioxide

SAMPLE	SILICON (%)	SILICON DIOXIDE (%)
Bureau of Standards	1.74	1.63
	1.72	1.90
	1.71	1.35
1	0.95	3.23
	1.09	3.12
2	0.1	2.38
	0.1	2.56
3	0.59	2.61
	0.57	2.45
4	0.12	1.63
	0.17	1.93
6	0.41	0.69
	0.22	0.66

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STABILITY OF A SILICONE OIL IN CONTACT WITH SODIUM

J. C. White and W. J. Ross

At the request of the ANP Experimental Engineering Group, an investigation was made of the stability of Dow Corning silicone fluid DC-550 in the presence of sodium at elevated temperatures. This material is a clear, slightly yellowish liquid of exceptional heat stability, high resistance to oxidation, low volatility, and high flash point (600°F, minimum). Because of its favorable properties, it is being considered as a pressure transmitting agent in equipment in which the fluid must be in contact with sodium metal at temperatures of 300 to 350°F. In the event of certain types of failure of the apparatus, however, the oil could come in contact with sodium at considerably higher temperatures.

In order to simulate conditions approaching those which might be encountered in the case of an apparatus breakdown, tests were carried out in

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which samples of oil and sodium were heated together at temperatures up to 1050°F. In these experiments the sample was heated under an atmosphere of helium in a test tube which was placed in a lead bath. Temperatures were measured by means of a thermocouple which was inserted in the oil. Two tests were made in which the oil was heated with sodium for 30 min at 1050°F. No violent reaction took place but the oil darkened in color. Upon cooling, the material solidified into a hard, oily, crystal-like mass. A qualitative test showed the presence of sodium

In a test in which the oil was heated without sodium at 1000°F for 15 min, the only effect was an increase in viscosity and refractive index, apparently due to the loss of some of the lower boiling constituents of the oil.

When the oil was heated for 6 hr with sodium at 400 to 450°F there was no visible change, but the basic reaction shown by a water extract of the oil was evidence that some reaction between oil and sodium may have taken place.

The results of these experiments indicate that silicone oil DC-550 is stable in the presence of sodium at 400 to 450°F for several hours and probably stable at 600 to 700°F for short periods.

SERVICE ANALYSES

J. W. Robinson and L. J. Brady

During this period the analytical work carried out in furtherance of the ANP program consisted chiefly of the determination of minor impurities in liquid metals (coolants) and of the analysis of materials of construction and of uranium compounds and mixtures.

Sodium Analysis (J. M. Peele and L. H. Jenkins). Corrosion-test samples of sodium after exposure to various materials of construction at elevated temperatures were analyzed spectrographically for minor components using the porous cup method, and for oxygen content by the method of Pepkowitz and Judd.

Solutions of test samples and of standard samples were prepared in the chemical laboratories for spectrographic analysis; these solutions were then examined spectrographically in the Physics and Spectrographic Department of the Isotope Research and Production Division.

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Analysis of Boron Carbide (J. R. Lund). Acceptable methods have been established for the determination of boron oxide, total boron, carbon, and moisture in boron carbide. The boron oxide is determined by titration of a hot-water leach of the sample. In determining total boron complete dissolution of the sample is required. It was found that the product of a sodium carbonate fusion would dissolve completely only if the material was ground to pass a No. 120 sieve prior to fusion. The resulting solution was titrated for total boron in the usual manner. Carbon was determined by combustion at 1400°C, using copper as an accelerator. Moisture was calculated from loss in weight at 110°C.

Analysis of Ferrous and Nonferrous Alloys (C. K. Talbott). In general, standard methods were used in the analysis of stainless steels and nickel alloys. A number of special alloys such as Co-Ni, Zr-Ni, and Mn-Cu, required less common methods of analysis.

Analysis of Uranium Compounds (E. C. Lynn and A. F. Roemer, Jr.) Sodium and potassium uranate, oxides of uranium, and fusion products containing small amounts of uranium were analyzed for uranium, alkali metal, hydroxide, and carbonates. Potentiometric, colorimetric, or fluorimetric methods—depending on the concentration—were used for the determination of uranium. The alkali metals were determined by means of the flame photometer, carbonates by gas evolution, and hydroxides by titration with acid after precipitation of the carbonate as barium carbonate.

SUMMARY OF SERVICE ANALYSES

A brief summary of the analytical work for the ANP program during the period is tabulated below.

	NO. OF SAMPLES		
	NEPA	Y-12	TOTAL
Backlog of samples 9-16-50	50	1	51
Samples received 9-16-50 to 11-30-50 inclusive	157	71	228
Total	207	72	279
Samples reported 9-16-50 to 11-30-50 inclusive	128	48	176
Backlog as of 12-1-50	79	24	103

	NO. OF DETERMINATIONS
Samples from NEPA	581
Samples from Y-12	128
Total	709

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