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REACTORS—RESEARCH AND POWER

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G. M. INMAN

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ABSTRACT

Engineering pertinent to the development of the sodium-cooled, graphite-moderated type of reactor was continued. This included work on problems related to the zirconium canned moderator, low enrichment uranium fuel, sodium piping, secondary coolant system, shielding, and the control and safety elements. A large fraction of the work was devoted specifically to problems of the proposed Sodium Reactor Experiment (SRE) configuration. In this connection, an integrated effort was initiated to prepare a complete preliminary design of the SRE by an early date.

In addition, two alternate sodium-graphite reactor configurations were studied. One was an intermediate size, 145 thermal megawatt, unit optimized for the production of low cost plutonium. The second was a low power, 10 thermal megawatt unit intended for power production, but in which sodium circulation through the core was entirely dependent upon thermal convection.

This report is based on studies conducted for the Atomic Energy Commission under Contract AT-11-1-GEN-8.

Previous SGR Progress Reports

NAA-SR-227 SGR Quarterly Progress Report September-November, 1952
NAA-SR-260 SGR Quarterly Progress Report December, 1952 - February, 1953
NAA-SR-274 SGR Quarterly Progress Report March - May, 1953
NAA-SR-878 SGR Quarterly Progress Report June - August, 1953
NAA-SR-956 SGR Quarterly Progress Report September - November, 1953

I. INTRODUCTION

Much of the work conducted during this quarter was related to problems of general importance to the sodium-graphite type of reactor. Experimental data were obtained regarding the strength of zirconium welds, and work was devoted to the development of methods and equipment suitable for the fabrication of the zirconium canned graphite moderator assemblies. For the reactor fuel, additional measurements were made on the thermal conductivity of uranium, as well as the effect of thermal cycling on the dimensional stability of uranium and uranium alloy specimens. A preliminary examination was made of a modified fuel slug design which should permit higher specific power and better dimensional stability. A study was made of the possibility of using freeze seal pipe connections in the sodium system, where the frozen seal is maintained by convection cooling to the ambient atmosphere. Work was completed on the development of an apparatus for performing welds to join stainless steel tubing in remote locations. A comparison was made of the heating effects in hafnium and in boron-steel control rods. Another study was devoted to novel types of the secondary heat transfer system wherein mercury fog and an organic vapor are used to produce high temperature steam.

Other work was specifically devoted to problems of interest in the Sodium Reactor Experiment (SRE). An analysis was made of a possible start-up incident in the SRE and the results used to establish recommended speeds of control rod withdrawal. An over-all safety evaluation of the SRE was made and formed the basis for a separate report and a presentation to the Advisory Committee on Reactor Safeguards at their fourth meeting. This study was relative to 20 megawatt operation of the SRE at the proposed site near Santa Susana, California.

The possibility of using an organic fluid as the secondary coolant in the SRE was explored, but not concluded to be a desirable alternate to the use of sodium at this time. Progress was made on the assembly of an experiment to determine the heat transfer characteristics of the SRE boron-steel control element assembly. Experiments were completed on the pneumatic recovery of steel balls at the bottom of the ball safety element thimble. Hydraulic tests were conducted to determine the coolant flow pattern and pressure drops characteristic of the seven-rod fuel element. Analytical studies were performed to estimate the heating and temperature effects in the SRE graphite moderator, as well as in the thermal and biological shields.

An integrated effort was initiated to carry through the preliminary design of the SRE. It is planned that a complete set of preliminary design drawings will be available by April 1, 1954. This effort is divided into six categories covering the SRE installation as follows:

a. Moderator and tank assembly.
b. Top shield and core components.
c. Sodium systems.
d. Auxiliary systems and reactor facilities.
e. Instrumentation.
f. Site and buildings.

The results of this preliminary design work are to be included in the next quarterly report.

Studies were conducted on two alternate sodium-graphite reactor configurations. As a result of renewed interest in plutonium production, the conceptual design of an intermediate size reactor with the primary objective of low cost plutonium was investigated. This unit employs a tank type design, produces 145 thermal megawatts, and uses slightly enriched uranium fuel in the form of 19-rod clusters. A second alternate design intended for the production of small quantities of electrical power was given preliminary consideration. The objective was to determine if appreciable savings in power cost could be made in a unit of this type if the sodium cooling of the fuel elements was dependent entirely upon thermal convection of the sodium. The general performance was outlined for a unit capable of production of 10 thermal megawatts.

II. REACTOR PHYSICS ANALYSIS

A. Start-up Incident in the SRE (F. Fillmore, J. Garvey)

The pile kinetic equations have been solved for the SRE using an initial $k_{eff} = 0.999$, an extraneous source of 10^7 neutrons/second present in the reactor, and with a constant rate of increase of reactivity. The calculations were made using a prompt neutron generation time of 0.455×10^{-3} second which is characteristic of the SRE.

The relative power level N and the inverse period $1/\tau$ vs time, with reactivity increasing at the rate of 0.0100 dollar/second, are plotted for the SRE in Fig. 1. The response of the SRE to a rate of reactivity increase of 0.0200 dollar/second is shown in Fig. 2. Figure 3 shows the inverse period vs dr/dt.

An investigation was made of possible start-up incidents in sodium-graphite power reactors with particular emphasis on the 20-megawatt SRE. The purpose of the study was to find the responses of such systems under varying conditions in order to establish a safe start-up procedure. A start-up incident was assumed to be caused by the unmanaged withdrawal of a control rod after the system has been brought to a multiplication factor of 0.999. The resultant power surge was assumed unchecked by any of the standard safety devices-the sole restraint was the negative metal temperature coefficient of reactivity. One objective of the study was to determine a rod withdrawal speed so that the operator would have an interval of several minutes after the attainment of normal full power in which to manually shut down the reactor before the hottest fuel rod enters the gamma-melt phase.

The criterion of a "safe" incident is that no fuel rod melting occurs. This is a stringent requirement and, in general, it is necessary that melting be forestalled not only with the aid of the metal coefficient but also by means of secondary methods such as the manual operation of a safety device.

Calculations performed for a number of control rod speeds and for various initial cooling conditions indicate that a speed of withdrawal of 0.003 dollar/second provides an adequately long period before fuel melting during a start-up incident, and yet does not lead to inconveniently long start-up procedures. Figure 4 shows the time in seconds after attainment of normal full power until the hottest fuel

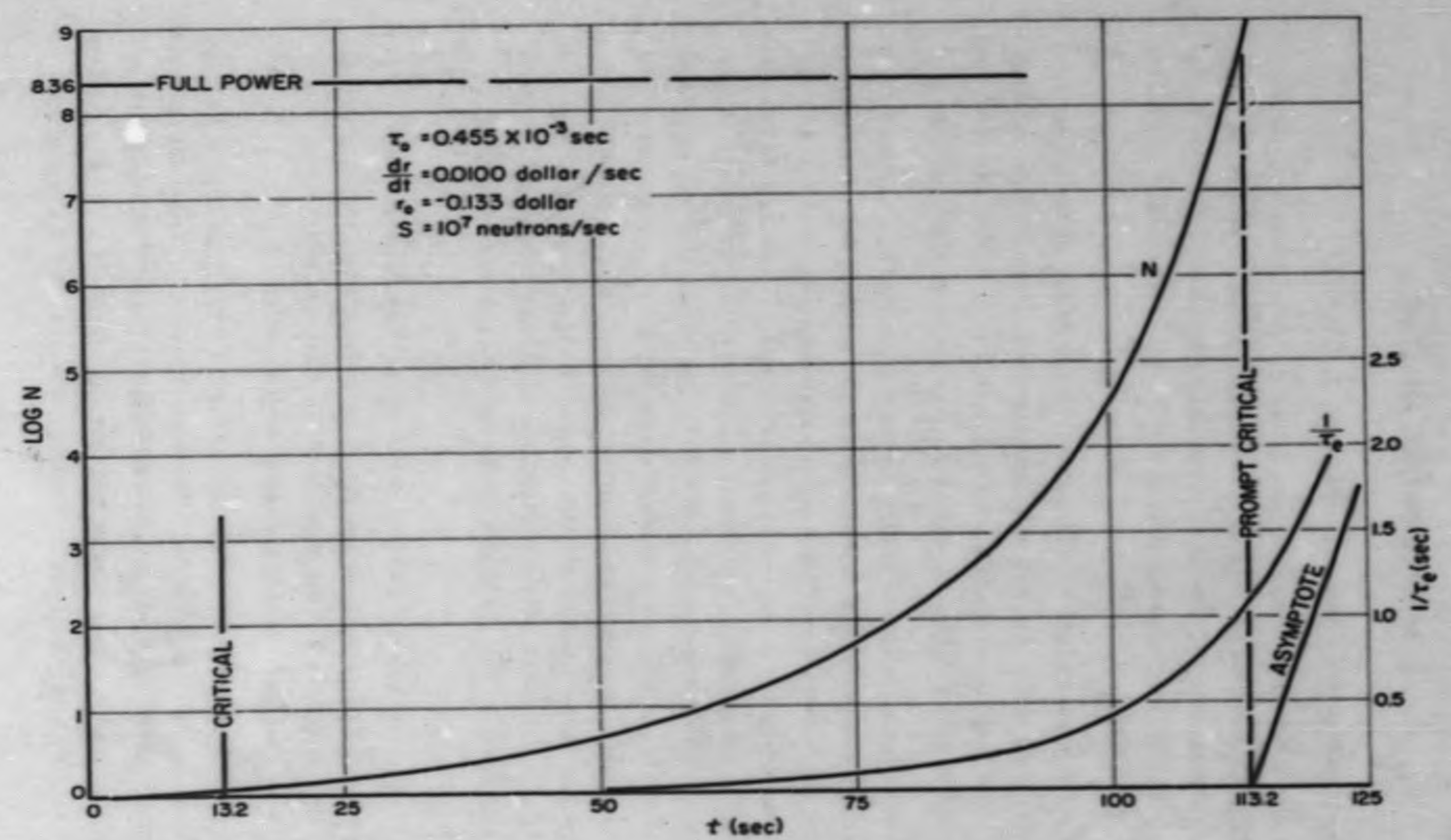

Fig. 1. SRE Start-up Incident - Relative Power Level and Inverse Period
vs Time for dr/dt = 0.0100 dollar/second

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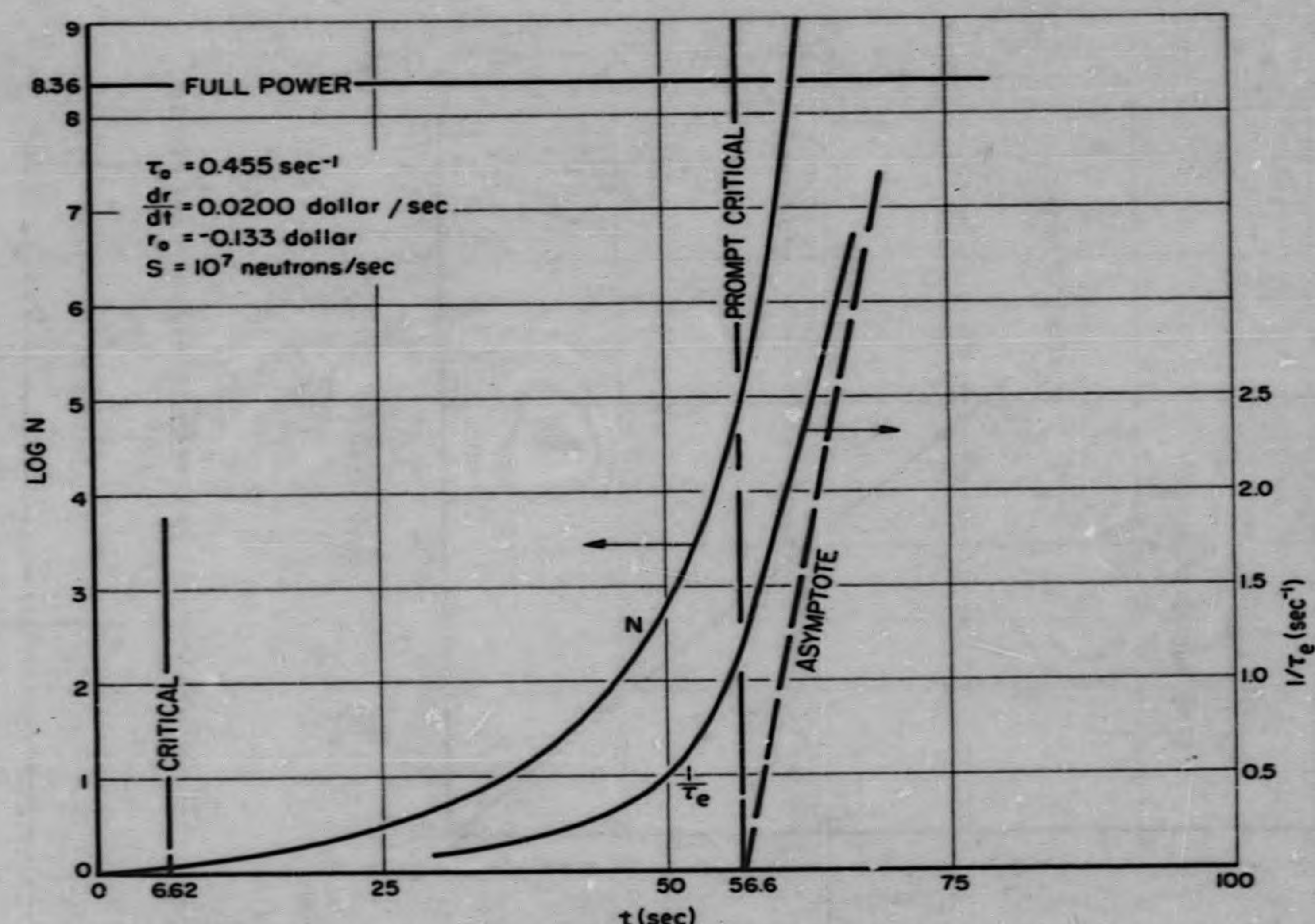

Fig. 2. SRE Start-up Incident -- Relative Power Level and Inverse Period vs Time for dr/dt = 0.0200 dollar/second

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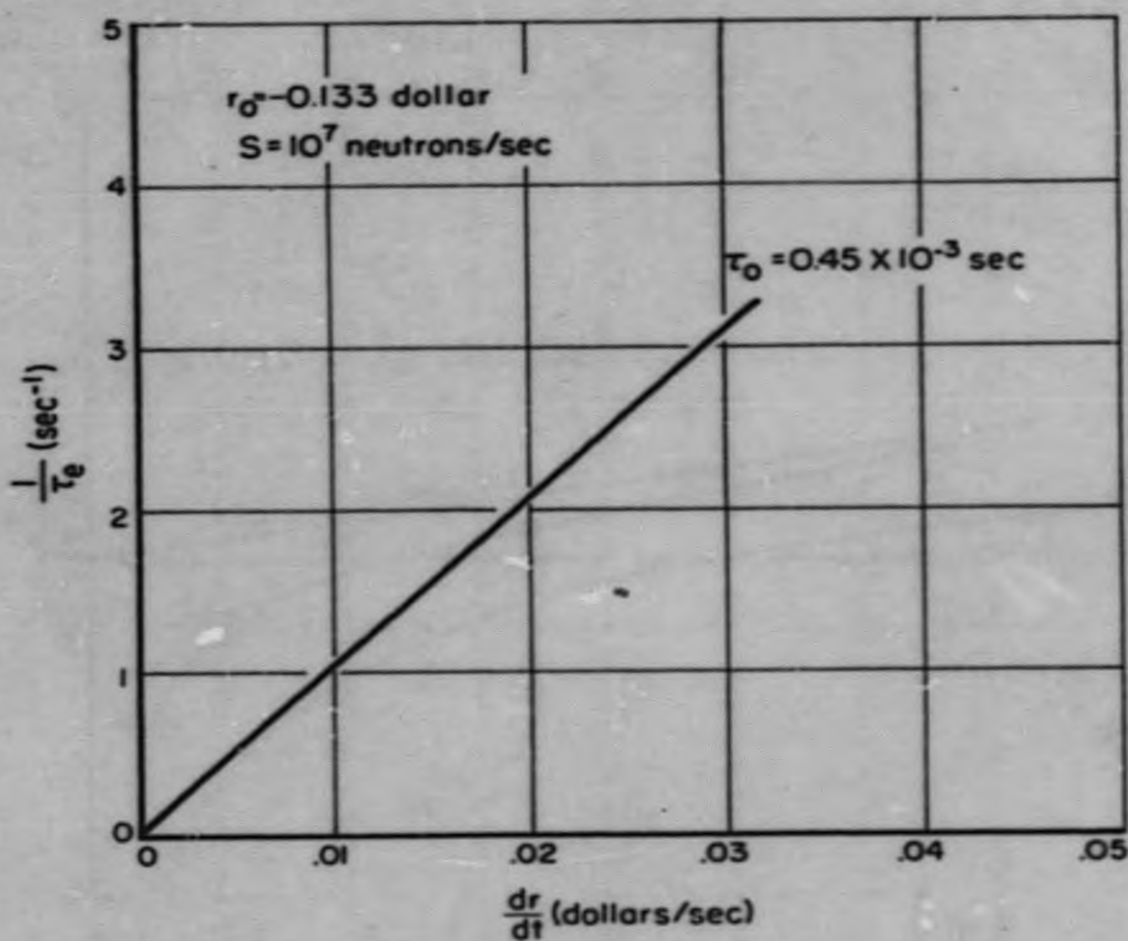


Fig. 3. SRE Start-up Incident - Inverse Period at Full Power vs dr/dt

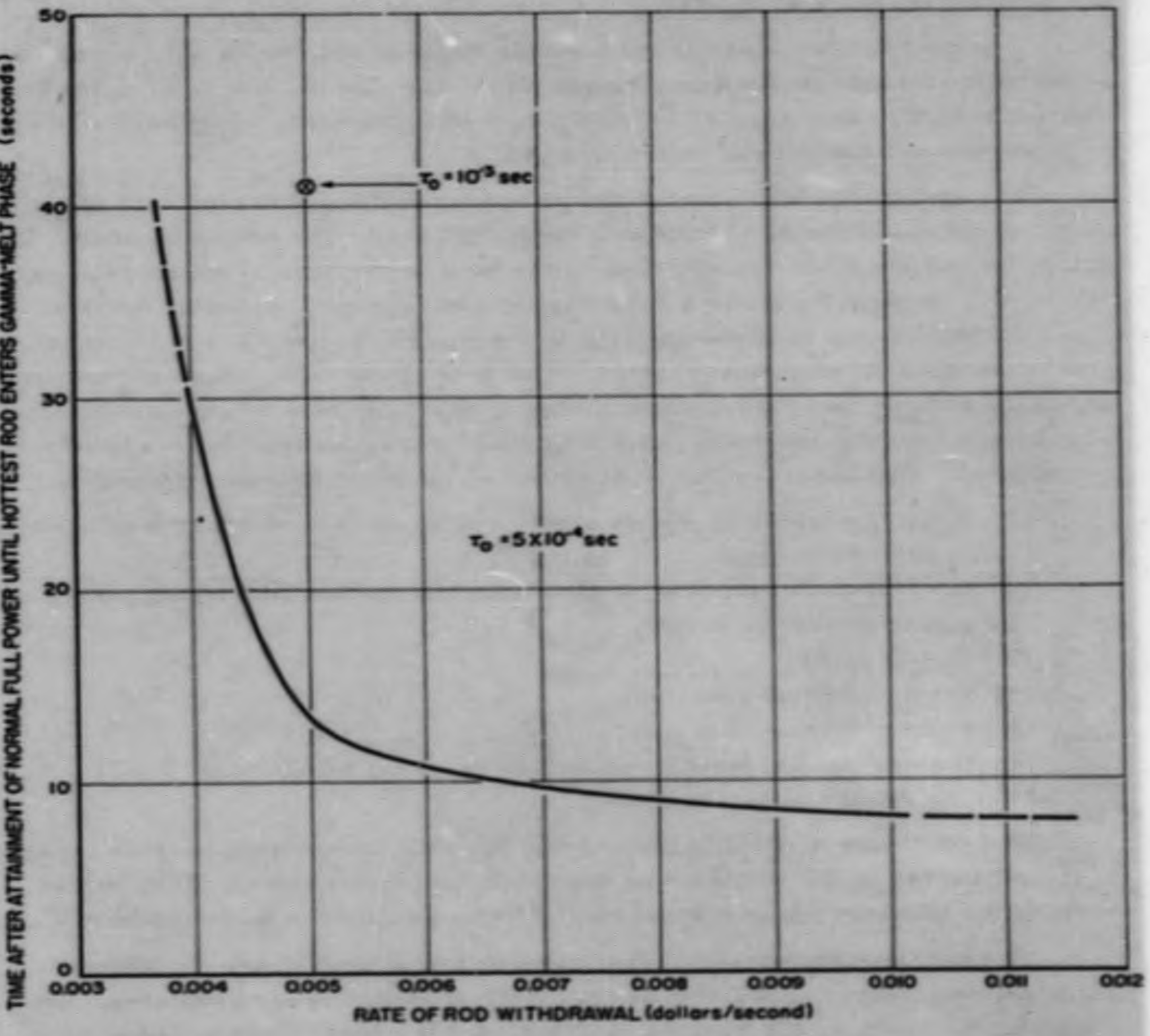


Fig. 4. SRE Start-up Incident - Time After Attainment of Normal Full Power Until Hottest Fuel Rod Enters the Gamma-Melt Phase

rod enters the gamma-melt phase as a function of a controlled rate of control rod withdrawal in dollars/second.

B. Safety Evaluation of the SRE

A study has been made of the possible hazards associated with the operation of the SRE at the proposed Santa Susana site. The site is located 30 miles northwest of downtown Los Angeles, 6 miles west of Chatsworth, and 3 miles south of Santa Susana, California, in the Simi Hills.

The elevation of the reactor site is approximately 1850 feet, and the maximum elevations of the Simi Hills are about 2400 feet. The elevation of the San Fernando and Simi Valley floors at the base of the hills is approximately 900 feet. The Simi Hills are a very rugged outcropping of sandstone strata. The hilly and barren surrounding area provides the required 1.4 mile radial clearance for a 20-megawatt reactor. The consequences of equipment failures and malfunctions, the events which follow operating errors by personnel, acts of sabotage, and the incidents which might arise from natural causes have been considered. Particular emphasis has been given to the following points:

(1) Reactor nuclear runaways (uncontrolled release of excessive amounts of nuclear energy)
(2) Interruption to the flow of coolant in the reactor (failure of the normal means of heat removal)
(3) Sodium fires
(4) Other chemical reactions
(5) Release of stored energy
(6) Detonation of chemical explosives as an act of sabotage
(7) The effects of earthquakes

The conclusions of the study are that the SRE presents no serious hazard to the public nor to the surrounding area under the most severe conditions of accident or sabotage which may be realistically assumed to be possible.

The sodium-graphite-low enrichment uranium system has an inherent negative temperature coefficient of reactivity at operating temperatures.[1] In addition, the reactor, the reactor controls, and the cooling system have been so designed as to prevent accidents insofar as this is possible and practical and to limit the consequences of any accidents. The reactor core and primary cooling system are located below ground level as shown in Fig. 5. This arrangement permits complete containment of any radioactivity which would result from the melting of the fuel or a leak in the radioactive sodium system.

There has also been calculated the "ultimate hazard"-the purely hypothetical and not at all realistic case-wherein all the radioactivity in the reactor is released to the atmosphere. Calculations indicate that even in this case the situation is not catastrophic, and there is still a factor of safety over conditions assumed acceptable on an emergency basis.

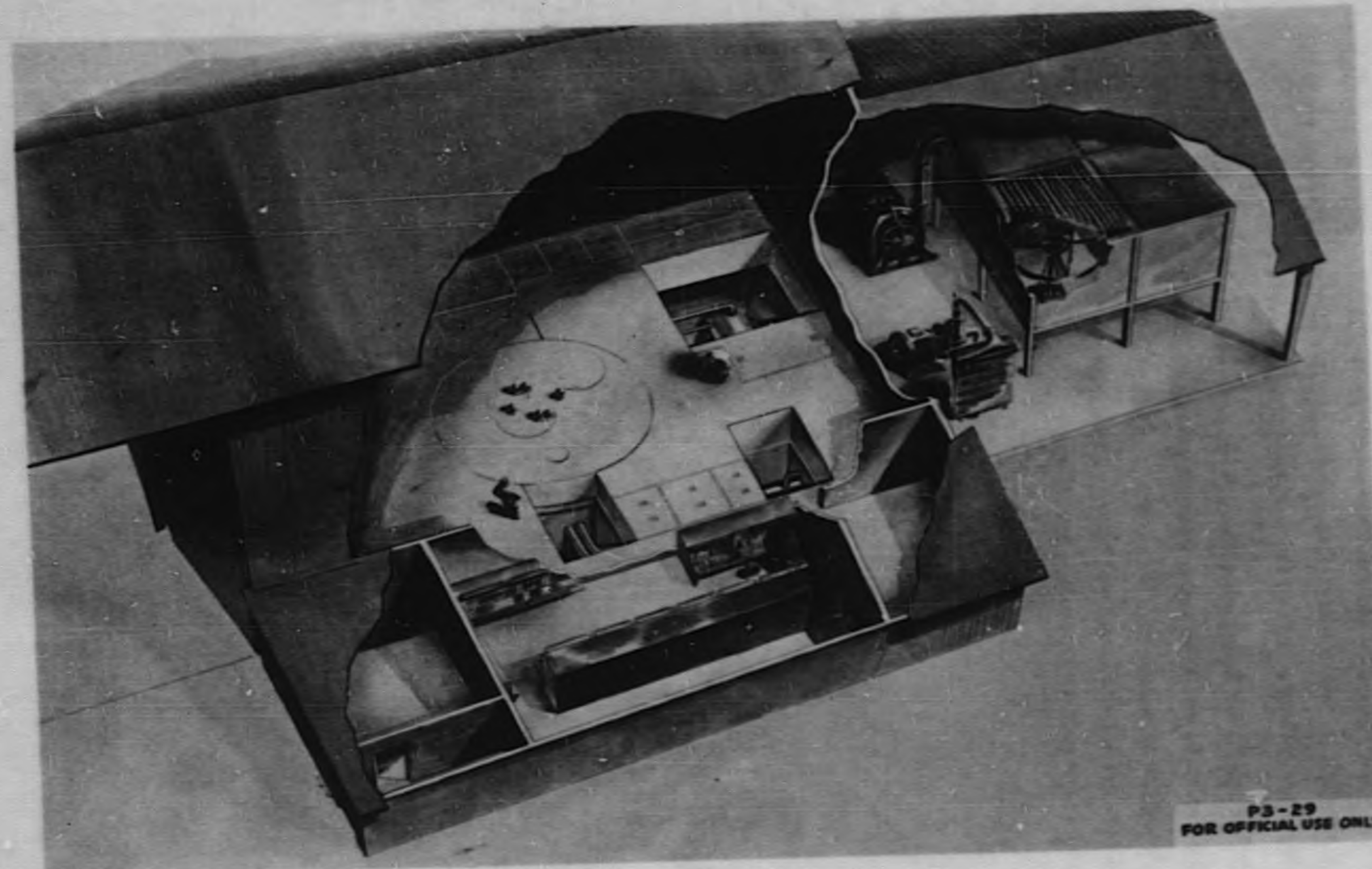


Fig. 5. Sodium Reactor Experiment Arrangement

III. REACTOR ENGINEERING

A. Experimental Studies Related to the Canned Moderator

1. Fabrication of Zirconium Canned Moderator Cells (W. Cockrell) - Six hexagonal zirconium canned graphite moderator cells are being made to develop fabrication methods. Two formed halves of the hexagonal skin for a 10-foot long can were welded together using a welding machine with a 6-foot horn. Each of the two seams were started at one end of the can and run for slightly over 5 feet; the parts were then reversed and the seams run from the other end with a slight overlap. The seams produced appear satisfactory and only relatively minor modifications of usual welding technique are required. Steps are being taken to procure a welding machine more specifically designed for this application.

Zirconium ingots required for the SRE moderator cans are now in production. Negotations are under way for the fabrication of the ingots into the required plate, strip, and sheet. It is expected that the fabrication will be completed by June, 1954.

An order has been placed for six 10-foot long graphite logs. They will be machined to the required hexagonal cross section and used in the assembly of prototype zirconium canned moderator cells.

2. Tensile Strength of Zirconium Welds (F. Bowman) - Automatic heliarc welded tensile test coupons of 0.050-inch thick annealed sponge zirconium sheet have been made. The specimens were cut in the directions both parallel and perpendicular to the rolling direction. The results of the tests are given in Table I.

B. Fuel Element Development

1. Dimensional Stability of Unrestrained Unalloyed Alpha Rolled Uranium (B. Hayward) - Studies to determine the effect of thermally cycling unalloyed uranium specimens 500 times between 200 and 700° C were continued (See Ref. 1, page 64). Specimens were made of powder compacted uranium, alpha rolled uranium, and high alpha rolled uranium. The appearance of the specimens after 500 cycles is shown in Fig. 6.

(All Specimens Are 1 1/2 x)

POWDER COMPACTED

Maximum change in Diameter	+12%
Maximum change in Length	+34%

POWDER COMPACTED

Maximum change in Diameter	-10%
Maximum change in Length	+23%

ALPHA ROLLED

Maximum change in Diameter	+16%
Maximum change in Length	+6.4%

HIGH ALPHA ROLLED

Maximum change in Diameter	+34%
Maximum change in Length	+8.2%

Fig. 6. Unalloyed Uranium Specimens After 500 Thermal Cycles Between 200 and 700° C

TABLE I

TENSILE STRENGTH OF ZIRCONIUM WELDS

Temperature °F	Direction	Yield Stress, psi 0.2% Offset	Ultimate Stress, psi	% Elongation in 2.0 in.
70	Longitudinal	41,800	57,800	14
	Transverse	47,100	63,300	19
400	Longitudinal	22,200	35,300	25
	Transverse	26,400	35,700	sample failed
750	Longitudinal	12,300	18,200	36.5
	Transverse	11,700	17,200	35
1000	Longitudinal	8,500	13,000	43
	Transverse	9,000	13,800	30

In no case did failure take place in or near the weld. Tests at 750 and 1000° F were made in an argon atmosphere; the tests at lower temperatures were made in air.

2. Dimensional Stability of Unrestrained Powder Compacted Uranium Alloys (B. Hayward) - The thermal cycling of uranium alloys between 200 and 700° C was continued (see Ref. 1, page 60). The specimens were powder compacted and were made by Sylvania Electric Products, Inc. The specimens were cycled 500 times with the results as shown in Fig. 7. Metallographic examination of the specimens indicate the following:

a. The 0.42 w/o Si-U alloy retained a finely divided second phase with a smaller grain size, as a result of the thermal cycling.

b. The cycled 1.0 w/o Cr-U alloy appears to have a small decrease in grain size together with some agglomeration of a widely distributed second phase or precipitate.

c. The 1.6 w/o Nb-U alloy, as received, had a martensitic type structure and medium to fine grains with the grain boundaries outlined by the second phase. The cycled specimen showed an increase in grain size with the homogeneous second phase tending to spherodize.

d. Another U-Nb alloy, 3.0 w/o Nb, made from prealloyed powder, in the as-received condition had medium to large grains and a homogeneous two phase structure. There was no great change in grain size or structure from cycling.

All completed thermal cycling experiments to date indicate that the powder-compacted uranium alloys are considerably more dimensionally stable than any of the unalloyed uranium specimens which have been cycled in either high or low temperature ranges.

3. Dimensional Stability of Restrained Unalloyed Uranium (B. Hayward) - Unrestrained alpha rolled unalloyed uranium specimens were cycled 200 times between 630 and 680° C at a rate of four cycles per hour. The result of these low amplitude cycles through the alpha-beta transition temperature (660° C) is quite similar to the results shown in Fig. 7. The slugs shrink about 10 per cent in length, increase about 10 per cent in diameter, and develop a bumpy and badly cracked surface.

In order to estimate the strength of material required to restrain the dimensional changes which result from alpha-beta cycling, Type 321 stainless steel jackets were shrunk onto alpha rolled uranium slugs. The wall thicknesses of the jackets were then reduced by machining to approximately 10, 20, 40, and 60 mils. All of the specimens were 5/8 inch by 2 inches alpha rolled rods, and the ends of the jackets were machined flush with the uranium. All of the jacketed specimens and the control specimens (bare uranium) had the same original length within ± .005 inch.

The experiment consisted of cycling the specimens 200 times (or until failure) through the alpha-beta transition temperature (630 - 680° C); each test series consisted of an unjacketed control specimen and one or two jacketed specimens. The results of the tests are shown in Fig. 8. The number of cycles prior to the rupture of the 0.006-inch jacket was not determined.

The results, although inconsistent, indicate that all of the jackets afford some degree of restraint to the expansion of the uranium. The only definitie conclusion from these preliminary tests is that the dimensional changes increase up to 200 cycles, the termination point of the tests. Further tests are now on progress, and the specimens will be examined after 10, 100, 200, and 400 cycles.

4. Thermal Conductivity of Uranium (J. Droher, K. L. Johnson) - Thermal conductivity data for uranium above 400° C are required for the design of fuel rods required for high temperature reactors. Such data are very meager. Data have been obtained in this laboratory utilizing radial heat flow measurements for various temperature ranges between 70 and 800° C. Particular emphasis has

(All Specimens Are 1 1/2 x)

U-Si Alloy 0.42 w/o Si

Change in Diameter	+4.4%
Change in Length	+5.5%

EFFECT OF CYCLING: Slightly rough surface plus two small cracks

U-Cr Alloy 1.0 w/o Cr

Change in Diameter	+2.0%
Change in Length	+1.2%

EFFECT OF CYCLING: Slightly rough surface plus one tear

U-Nb Alloy 1.6 w/o Nb

Change in Diameter	+0.3%
Change in Length	+3.9%

EFFECT OF CYCLING: Slightly rough surface

U-Nb Alloy 3.0 w/o Nb

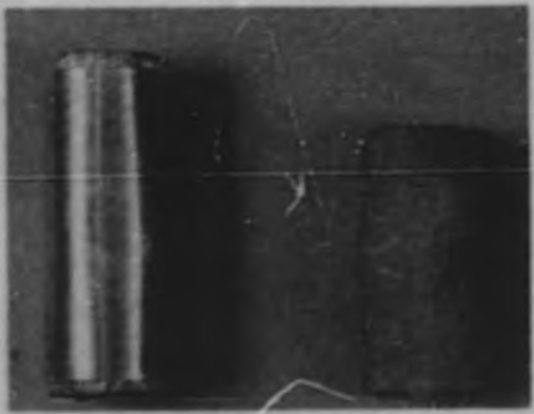
Change in Diameter	-4.3%
Change in Length	+13.0%

EFFECT OF CYCLING: Slightly rough surface

Fig. 7. Uranium Alloy Specimens After 500 Thermal Cycles Between 200 and 700° C

0.006-inch WALL JACKET — CONTROL

200 cycles



0.020-inch WALL JACKET — CONTROL

102 cycles

0.040-inch WALL JACKET — CONTROL

200 cycles

0.060-inch WALL JACKET — CONTROL

200 cycles

(All Specimens Are Full Size)

Fig. 8. Alpha Rolled Uranium After Thermally Cycling Between 630 and 680° C

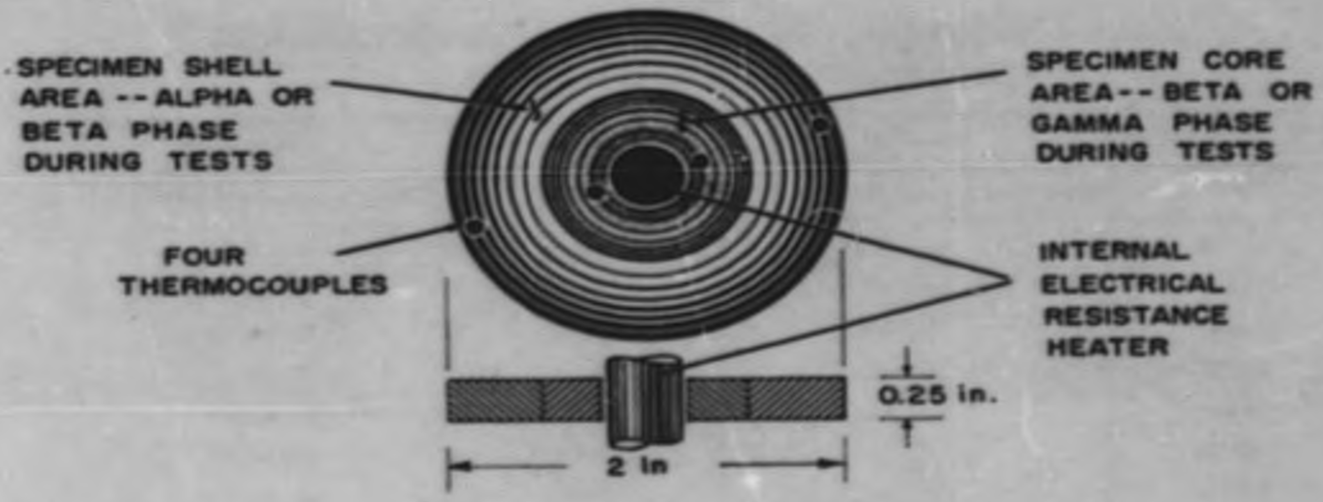


Fig. 9. Typical Uranium Disc Specimen Used for Radial Bi-phase Thermal Conductivity Measurements (full size)

been given to the region from 400 to 660° C, the high alpha region. The possible effects of heat treatment, biphase operation, and anisotropy have also been reported (Ref. 1, page 71).

The radial conductivity values for unalloyed alpha rolled beta-treated uranium which are based upon the curve obtained from all measurements made to date are given in Table II. The conductivity values, based on the sample geometry and experimental accuracy, are believed accurate to ± 10 per cent.

Previous data had indicated that the conductivity of alpha rolled beta-treated uranium is essentially unchanged at any given temperature, both <u>before</u> and <u>after</u> transition into the beta-phase. Current measurements indicate the conductivity is also unchanged for transitions into gamma-phase.

During the course of the previous temperature cycling measurements, data were taken with the specimen core in the beta-phase and the specimen outer shell in the alpha-phase (Fig. 9). During this quarter, measurements were made with the specimen core in the gamma-phase and the shell in the beta-phase.

TABLE II

RADIAL THERMAL CONDUCTIVITY OF ALPHA ROLLED BETA-TREATED URANIUM

Mean Specimen Temperature °F	Radial Conductivity Btu/(hr) (sq ft) (°F/ft)	Mean Specimen Temperature °C	Radial Conductivity Cal/(sec) (sq cm) (°c/cm)
70	13.8	20	0.0571
200	14.4	100	0.0598
400	15.4	200	0.0633
600	16.4	300	0.0671
800	17.5	400	0.0711
1000	18.7	500	0.0757
1200	20.3	600	0.0804
1400	22.5	700	0.0875
		770	0.0938

Such biphase operation did not cause any deviations in the temperature drop between the core and the shell, thus indicating the conductivity was not sharply changed at the phase-transition temperature.

Additional measurements were made during the past quarter using very thin strip specimens in the pulse-annealing apparatus. Previously reported measurements at -40° C had revealed no significant differences in the conductivity of specimens oriented parallel and perpendicular to the rolling direction. It was also found that at this temperature the conductivity was not significantly altered by prior annealing of the specimens at various temperatures up to 1000° C. These anneals were of 5 minutes duration and it was believed that this time was adequate to permit any reorientation of the grains of the 0.030-inch thick specimens. To make certain that a 5-minute anneal was adequate, two specimens previously tested were sealed under vacuum in a Vycor capsule and annealed for 100 hours at 800° C. No increase in thermal or electrical conductivity was observed when compared with a control specimen. It was noted, however, that the thermoelectric power was decreased as a result of a 100-hour anneal.

C. A Fuel Slug Design for Higher Total Reactor Power (B. Hayward)

The present SRE fuel slug is assumed to be limited to a peak maximum operating temperature of 1200° F, the alpha-beta phase transformation temperature in uranium. Although there is currently considerable interest in this problem, there is insufficient knowledge upon which to recommend an increase in this temperature. Thermal cycling tests have shown the extent of dimensional change that may occur within fuel slugs subjected to biphase temperature conditions. Some experimental effort is directed to determining the stress required to restrain these dimensional changes. This restraint may be thought of as being supplied by an outer layer of alpha phase uranium or alpha phase uranium plus cladding material. Methods of reducing the restraint required can be either through materials or design. A preliminary study indicates some small change in design may minimize the dimensional instability problems associated with two phase temperature conditions in uranium metal fuel slugs. It may also permit an increase in the power density by permitting more rapid heat removal under certain conditions.

Examples of two relatively minor modifications of the present fuel slug design are (1) a small coaxial hole in each uranium slug and (2) a radial slot along the full length of each uranium slug. The only change in the seven-rod fuel element is in each individual fuel slug. The configuration remains the same as the present design with the reactor coolant only in contact with the jacket of each fuel rod; these hollow slugs are not internally cooled with the reactor coolant. In addition, the ends of either of the above slugs may be radially tapered and grooved to reduce the appearance of beta phase material at a free surface by allowing the bond sodium to contact each slug end and permit flow of sodium from the center area. These designs have both advantages and disadvantages of varying importance as outlined in Table III. The design of the modified slugs is shown in Fig. 10.

D. A Sodium Freeze-Seal Pipe Connection (H. Sletten)

Piping systems which carry liquid sodium must be leak tight and in the past have usually been of all welded construction. In order to increase the flexibility and ease of replacement of system components, mechanical joints and connections have been tested. These have generally been unsatisfactory because of leaks. Success in sealing sodium in pumps and valves by freezing sodium between the

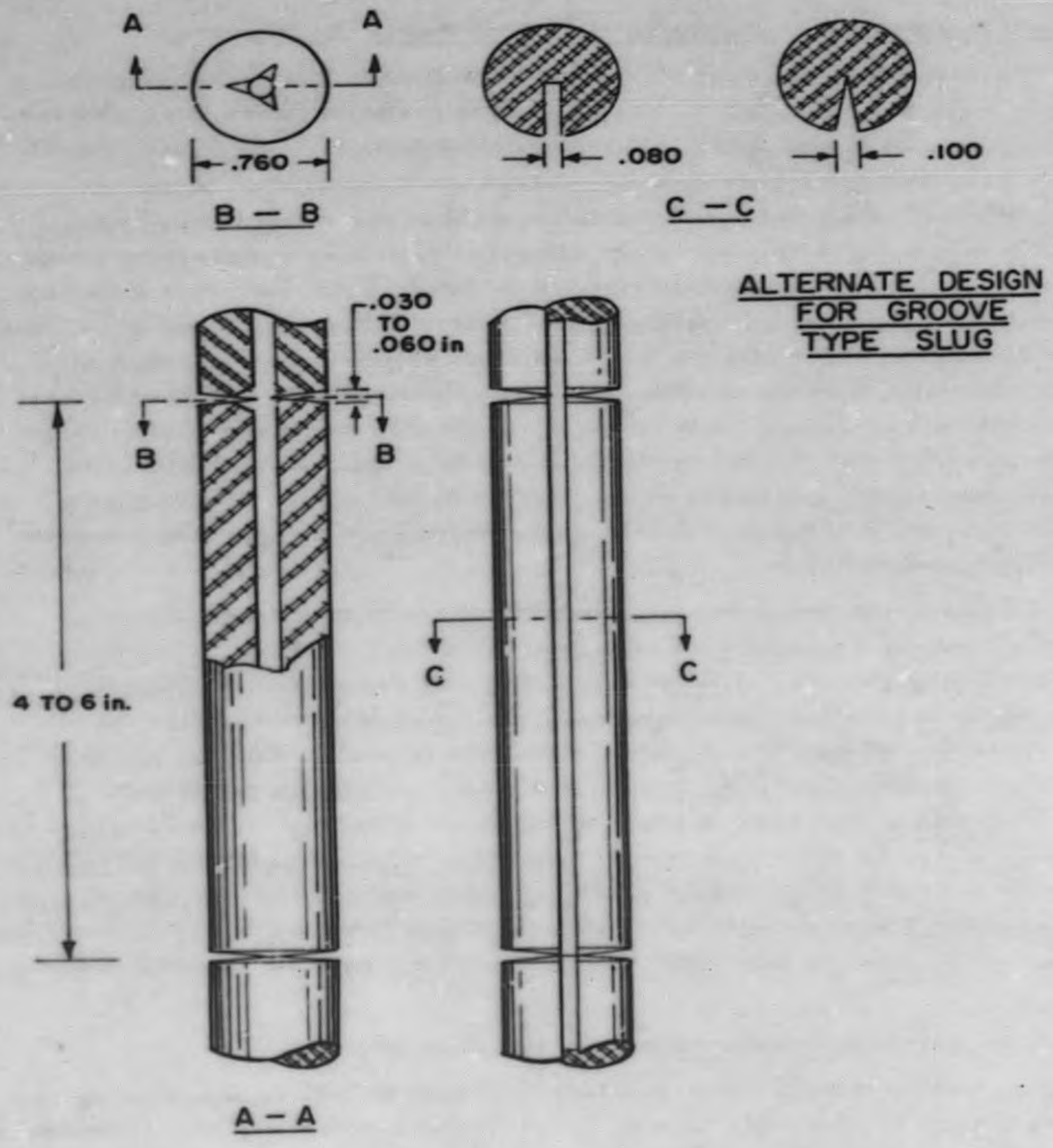


Fig. 10. High Temperature Fuel Slug Design

TABLE III

COMPARISON OF URANIUM FUEL SLUGS WITH COAXIAL HOLE AND LONGITUDINAL GROOVE

Slug Design	Advantages	Disadvantages
Axial Hole:	1. Potential increase in power density with peak temperature <1200° F. 2. Potential increase in slug stability from biphase temperature conditions. 3. Potential further increase in power density with biphase peak temperature conditions, (>1200° F) 4. Reduces end instability effect of having beta phase at a free surface. 5. Can be adapted to current fuel materials 6. Uniform stress and temperature pattern.	1. May have some flow within the Na bond; however, this would be an advantage in heat transfer. 2. Increases quantity of fuel element Na. 3. Increases slug OD or total fuel length to keep reactor uranium volume constant. 4. Limits practical slug length if hole is machined. 5. Unknown dimensional stability (some Hanford data). 6. Unknown fabrication procedure. (some adaptable Sylvania data).
Slotted:	1. Potential increase in power density with peak temperature <1200° F. 2. Possible increase in dimensional stability under biphase temperature conditions. 3. Potential further increase in power density with biphase peak temperature conditions, (>1200° F). 4. Reduces end effect of having beta phase at a free surface. 5. Can be adapted to current fuel materials.	1. Unknown and non-uniform radial temperature gradient and stress pattern, with possible failure through fatigue. 2. Unknown dimensional stability, may warp (no data). 3. Increased machining costs per slug. 4. Increases slug OD or total fuel length for same volume of U. 5. Increases quantity of fuel element sodium. 6. Probably limited to minimum of 3/4-inch diameter slug.

mating parts has indicated the possibility of applying the freeze-seal principle to a pipe connection to make a leak tight, readily separable, mechanical connection. A thermal analysis and design study of such a connection for a 2-inch diameter pipe has been made.

Despite the low freezing point of sodium (208° F) and the high operating temperatures of the proposed sodium systems (~1000° F), it is believed that the temperature of the mating surfaces of a mechanical connection can be reduced sufficiently to cause freezing of the sodium by some suitable scheme of free convection cooling to the ambient atmosphere (~100° F). Figure 11 shows a connection design in which some sodium from the main stream is allowed to flow radially outward from the pipe into a thin space. A finned section of copper or other good heat conducting material is bonded to the outer periphery of the space to provide a low resistance heat path to the ambient atmosphere. The freeze-seal surfaces are not required to sustain the mechanical loads that the connection may be subjected to; these loads are absorbed by a clamp which is applied to a heavy flange as shown. In order for this connection to be leak tight, the sodium must freeze at a sufficient distance from the outer end of the seal space to prevent the internal pressure from extruding the frozen sodium. A metallic ring between the mating parts will limit the flow if extrusion should occur or in the event that the sodium should melt completely. A thermal analysis has been made based upon the assumption that there is a perfect insulator on the pipe such that the only heat path to the finned section is through the sodium freeze-seal space and enclosing discs. Actually about 10 per cent of the heat transferred to the fins may be through the insulation. This improves the efficiency of the seal.

As shown in Fig. 11 a finned cylindrical "sleeve" slides over two halves of the coupling, and clamps hold the sleeve in place. The clamps on the sleeve also draw the two halves together as closely as possible at the outer periphery.

Freezing of the sodium can be assured by increasing the length of the sleeve until the temperature drop from the sodium to the air (assumed to be at 100° F) at the end of the seal is less than 100° F. In the design of Fig. 11, a 5.4-inch length of cylinder is required to freeze the sodium. A 2-inch standard pipe will require an 11-inch diameter connection.

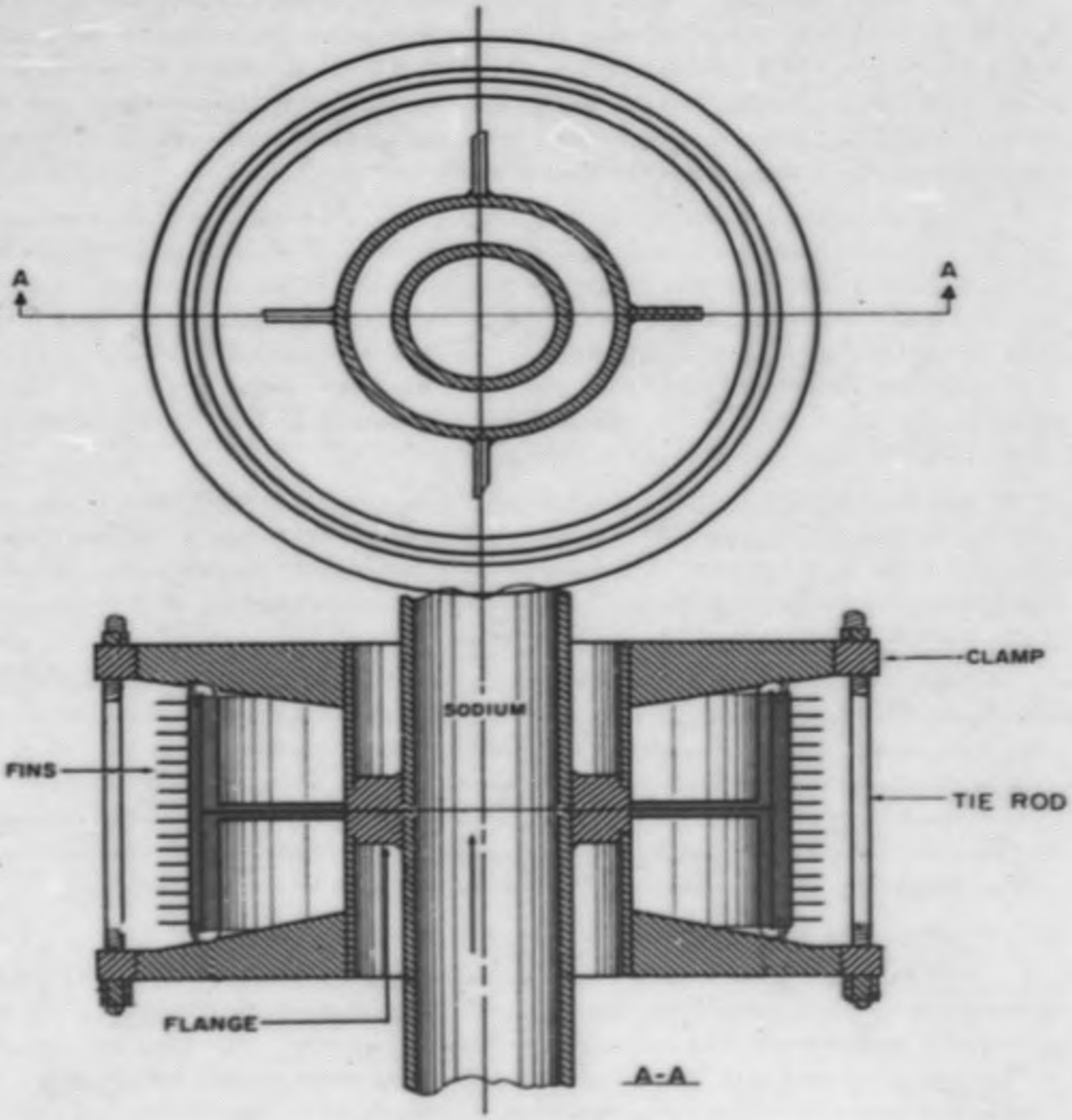


Fig. 11. Freeze-Seal Pipe Connection

The free convection coefficient between the fins and the ambient air is taken as 1 Btu/hr-ft^2-°F. Heat transfer from the pipe to the underside of the finned surface is neglected as it can be minimized by the use of radiation shields or insulation. The calculations are based upon conservative assumptions with the exception of the assumption that there is no heat transferred through the pipe insulation to the underside of the finned surface.

This analysis is for a standard 2-inch pipe. For the same extended surface (finned cylinder diameter minus pipe outside diameter) on a larger pipe the heat transfer area of the fins will be larger per unit area of pipe surface than for the smaller pipe. This tends to indicate that this type of connection is more favorable for small pipe and that more finned surface is required for larger pipes with the same sodium temperature. On the other hand, the ratio of over-all diameter to pipe diameter decreases as the pipe size increases so that the finned surface relative to the pipe size is less for larger pipe.

The relatively large diameter of the connection in comparison to the size of the pipe (approximately 11 inches for a 2-inch standard pipe) may be undesirable for some applications. If a diameter as large as 11 inches is not practical and smaller diameters must be used, a satisfactory design could be developed using forced circulation of air over the fins.

A disadvantage associated with forced convection cooling is the reliance on a pumping system to insure a leak tight connection. Failure of the pump would require that the whole sodium system be drained. Dependence on forced air convection appears less important in applications where the connection is used in a space which ordinarily would require forced ventilation. Where it is necessary to remove heat to lower the ambient temperature, it may be possible to direct cool inlet air or other gases over the finned sections of the connections.

E. Coolant Tube Replacement (M. Mueller, E. B. Hecker)

The development of equipment, techniques, and welding conditions required to remotely weld together two sections of a 2.80 OD by 0.025-inch wall 347 stainless steel coolant tube was discussed in the last report.[1] The tube sections being welded were located in a glass pipe and the method was developed by visual observation.

In order to adapt the method to permit completely blind welding, the glass pipe was replaced by a 5-foot long section of 6-inch diameter steel pipe (Fig. 12-B).

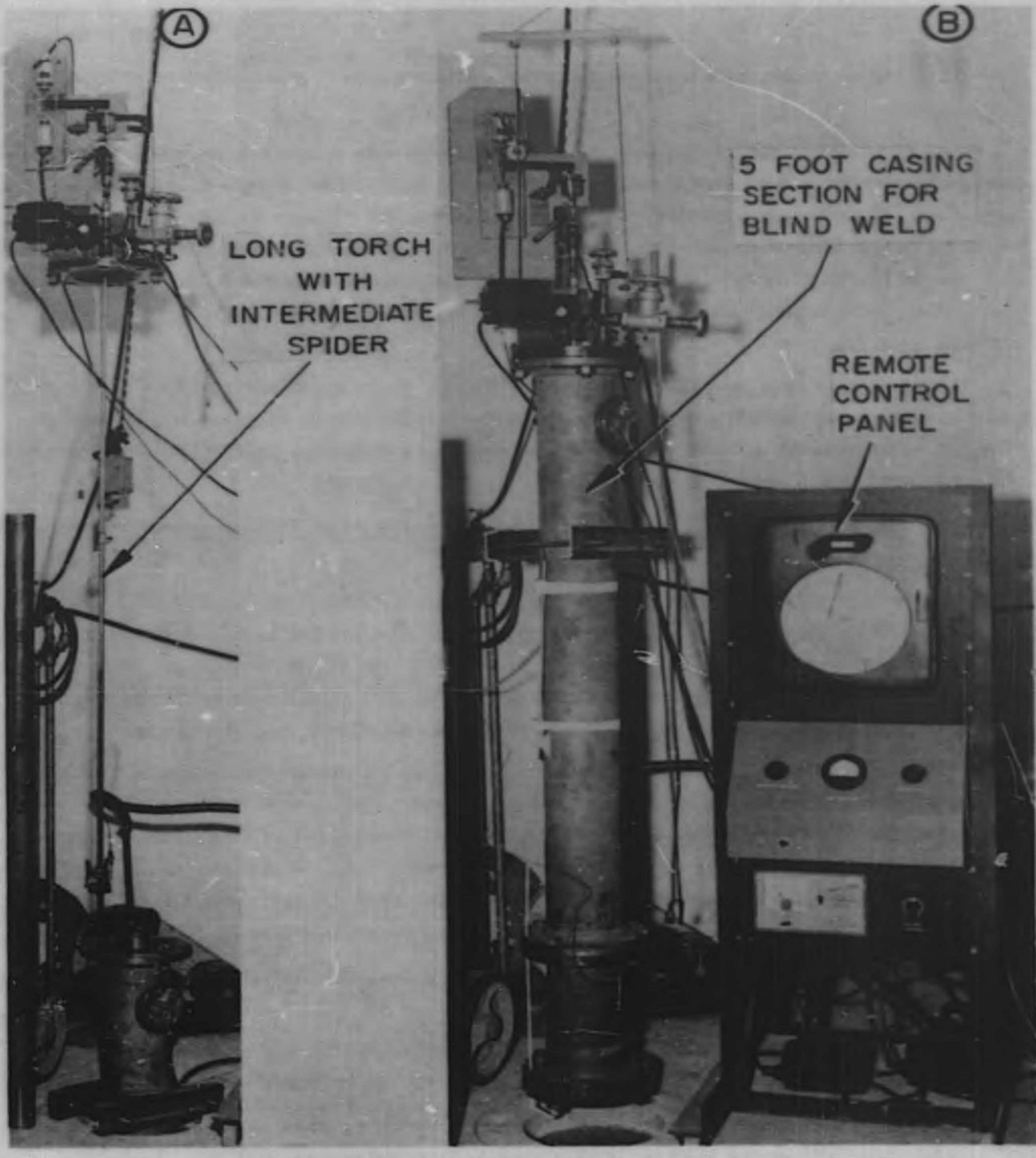


Fig. 12. Remote Welding Apparatus

A longer torch shaft was used; an intermediate spider was added to minimize deflection.

A short section of the tubing to be welded was placed at the bottom of the equipment casing with the flange at the upper end. The long section of tubing was then placed on top of the short section with the flange section down. The welding torch was then lowered into the casing, gas tight connections were made, and the weld was made. A satisfactory weldment, better than could be produced manually, was made (Fig. 13) which was found to be leak proof when tested with a mass spectrometer.

The equipment and techniques developed can be adapted to any angle. This experiment successfully concludes the development of a method for internal, remote controlled, tube welding as a means of repairing and replacing damaged reactor coolant tubes.

F. Heating Effects in Control Rods - A Comparison of Hafnium and Boron Steel (J. Howell)

A hollow thimble type control rod has been proposed for the SRE (Ref. 1, page 49). The amount of heat generated in the rod while in the reactor is of great importance. Because of the heat transfer characteristics of this configuration, a comparison of the heat generation rates in hollow control rods of two possible materials, hafnium and 3 per cent boron steel, has been made.

The procedure used for calculating the heat generation rates in the control rods assumes that the sources are proportional to the local thermal flux near the rods; this is derived from the volumetric power generation in the core. Such a method leads to a slight overestimate of the heating, since the thermal flux is actually depressed near the rods. However, the error is small and will not affect the comparison since the heating is related to the flux.

Heat is generated in a neutron absorbing control rod in an operating reactor due to the following reactions:

(1) Slowing down (moderation) of fast neutrons in the rod material.

(2) Absorption of energy of gammas from distributed sources (fission gammas, as well as those emitted following parasitic neutron capture in the moderator and other core materials).

(3) Absorption of kinetic energy of particles which may be produced as a result of neutron capture in the rod.

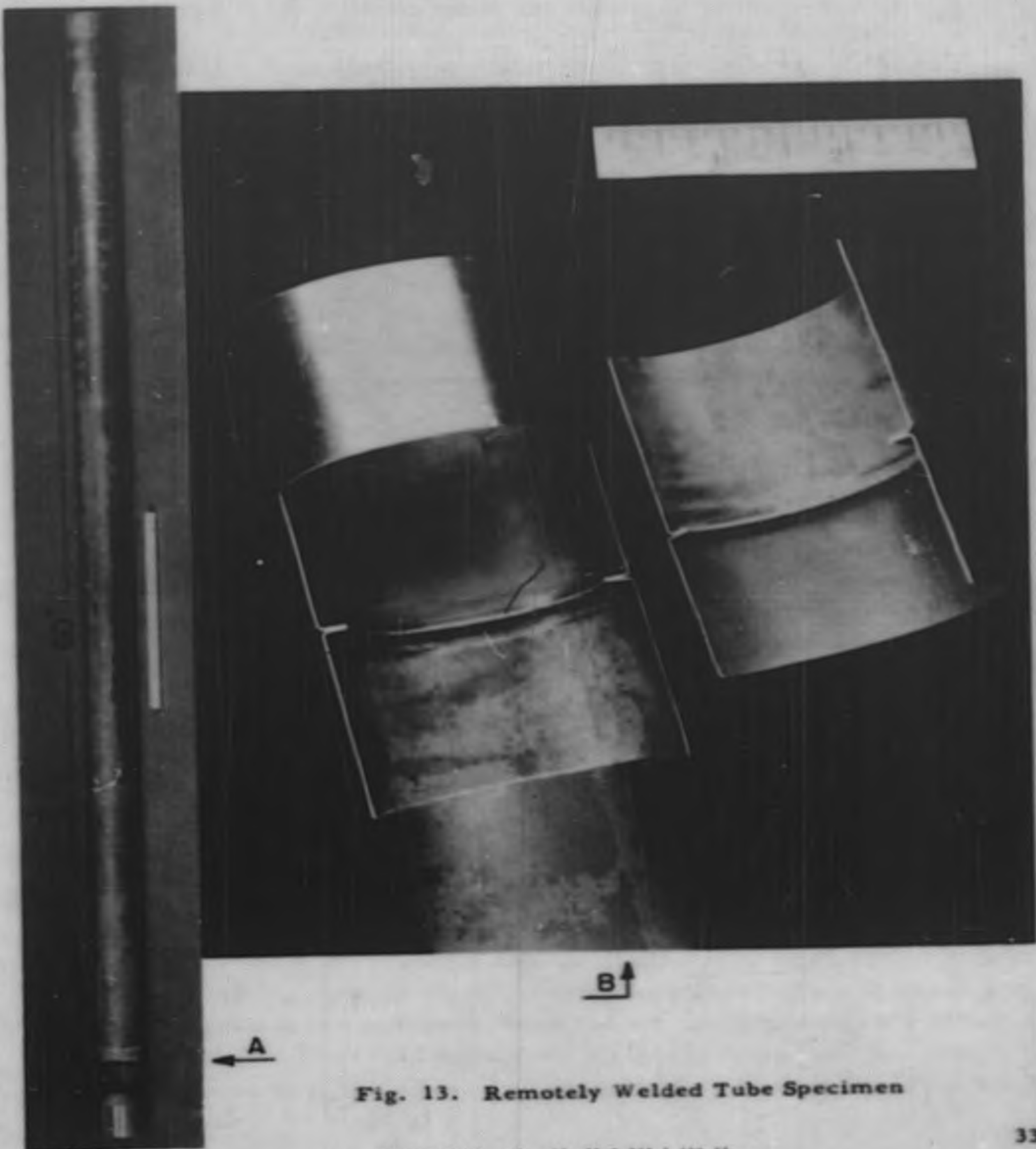


Fig. 13. Remotely Welded Tube Specimen

(4) Self-absorption of prompt and decay gammas which may be produced as a result of neutron capture in the rod.

For the rods being considered, which are of relatively high mass materials and small physical cross section, the contribution of the fast neutron moderation effect is small and was neglected in these calculations.

A comparison of the calculated maximum amount of heat generated in control rods of the proposed SRE design is presented in Table IV.

TABLE IV

HEAT GENERATION IN HAFNIUM AND BORON STEEL CONTROL RODS

Heat Source	Maximum Heat generated in rod $\left(\frac{kw}{ft}\right)$	
	3 per cent boron steel	hafnium
Fission and core gammas	0.87	2.17
Kinetic energy of particles	5.93	-
Gammas generated in rod	0.38	6.07
Total	7.18	8.24

From this comparison it is seen that the heating in the hafnium rod may be expected to equal that in the boron steel rod, and possibly to exceed it by as much as 15 per cent. It therefore appears that hafnium possesses no advantage over boron steel for this type of control rod.

G. Intermediate Heat Transfer Systems for Use with Sodium Cooled Reactors (J. R. Wetch)

A preliminary study has been made to examine alternate intermediate heat transfer media. A mercury fog convection intermediate heat transfer system may have advantages over a liquid sodium system. The advantages and disadvantages of organic vapor, and organic liquid intermediate heat transfer systems were also studied. An intermediate system which looks attractive for reactors with high outlet temperatures and low inlet temperatures (Fig. 14) uses mercury fog for the high temperature section and organic vapor for the two

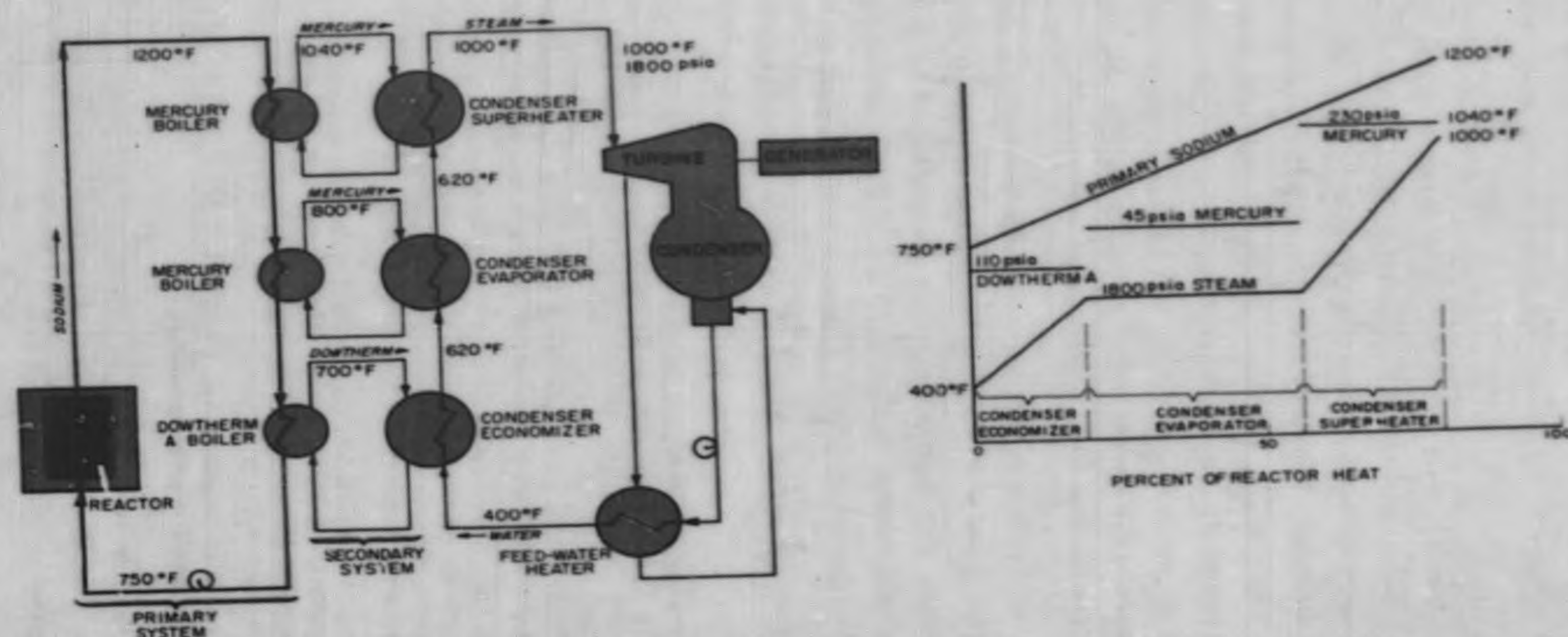


Fig. 14. SGR Cooling System with Mercury Fog-Dowtherm A Intermediate

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low temperature sections. The advantages of these systems are a substantial increase in safety and significant reductions in equipment investment.

1. Mercury Fog Convection Circuit - The mercury fog convection principle is well known. A liquid droplet-vapor mixture is the fluid which occupies the mercury boiler tube volume. A mercury boiler tube that has 80 per cent fog surface is by definition a tube which is only 20 per cent filled with liquid mercury at start-up. Properly treated mercury in deoxidized ferritic tubes will wet the heat transfer surface with a highly tenacious film through which heat is transferred and from which vaporization occurs. The expansion of the vapor as it passes upward through the tube tears liquid droplets from the relatively thick film layer at the bottom (inlet) portion of the tube and carries the liquid (feed) to supply the film in the upper (outlet end) of the tube.

Experience indicates it is perfectly safe to operate hydrocarbon fired mercury boiler tubes at 90 per cent fog surface or greater without overheating the boiler tubes. For the application we are interested in, it is impossible to overheat the boiler tubes, and fog surfaces of 95 per cent should be perfectly safe as long as circulation ratios of 1.3:1 are maintained.

The advantages of a mercury fog convection intermediate heat transfer system are:

a. The danger of a violent sodium-water explosion is eliminated. This allows relatively close coupling of the reactor and the steam generators. It also allows "single tube" heat exchanger construction. Savings may be realized from going to single-walled steam generators of carbon steel close coupled with the reactor and using steel alloy shells in the Na-Hg exchanger.

b. The fog convection type heat transfer loop can eliminate the necessity for intermediate fluid pumps. The system will operate by thermal convection and will make the system independent of auxiliary power scources.

c. Materials which are compatible with both sodium and mercury are much cheaper than stainless steel and will provide considerable economy in the intermediate heat transfer circuit at the operating pressures.

d. Mercury is liquid at room temperature and pipe and equipment heaters will not be necessary.

Before further design study is undertaken it is necessary that experimental pressure drop data be collected for boiling mercury at low circulation ratios in vertical forced convection tubes.

2. Organic Vapor Convection Circuits - Dowtherm "A", which is a eutectic mixture of biphenyl and diphenyl oxide, and Dowtherm "E", which is o-dichlorobenzene, are the two most commonly used organic heat transfer fluids. Both of these fluids are liquids at room temperature, are compatible with most structural materials, and are noncorrosive and nontoxic. Neither will react with water, and only a mild reaction ensues when in contact with hot sodium.

The recommended temperature range of operation for Dowtherm "E" is 0 to 550° F and for Dowtherm "A" 60 to 775° F. Both fluids are subject to damage by gamma radiation.

If mercury vapor were used in the superheat region as is shown on Fig. 14, the temperature to which Dowtherm "A" is subjected in the boiler section is not much above that which is now common practice. Only a very small portion of the fluid is ever heated to temperatures above 650° F which is the temperature where detectable thermal decomposition begins to occur. Either Dowtherm "A" or Dowtherm "E" may be used in the economizer section because the temperatures there are no greater than 550° F.

The advantages of a Dowtherm vapor convection heat transfer system are similar to those of the mercury fog convection system. The disadvantages are the thermal decomposition and fouling, the relatively minor decomposition due to gamma radiation, and the low heat transfer rates. The Dowtherm condensation rates are half those of mercury condensation rates and thus require the use of a higher reactor inlet coolant temperature.

In summary, it appears possible to use mercury fog circuits to cool the primary coolant sodium to 750° F. The sodium then passes through heat exchangers cooled by an organic such as Dowtherm "A". The sodium exits from this exchanger at a temperature of approximately 500° F.

H. SRE - Evaluation of Dowtherm as a Secondary Coolant (J. R. Wetch)

The SRE currently being designed uses sodium as the coolant for both the primary and secondary circuits. It is felt desirable for the initial operation to develop experience and experimental data using only one coolant. At some future date the reactor may be operated using an organic fluid such as Dowtherm for the secondary coolant. Data regarding the radiation stability and fouling factors of organic coolants must be obtained before a detailed design study for the high temperature SRE system may be prepared. However, a feasibility study was made to evaluate Dowtherm (both "A" and "E") for future application in the SRE.

It is possible to use Dowtherm in two thermal siphon loops in series with the main primary sodium circuit. The boiler in the high temperature loop will cool the primary sodium from 960 to about 700° F. The boiler in the low temperature loop will further cool the primary sodium from 700° F to the reactor inlet temperature. The fluids will be used at temperatures of 650 and 450° F for Dowtherm "A" and "E", respectively.

The proposed organic secondary system has several attractive features. Both loops will be charged with fluids that are liquid at room temperature. The completely independent lower temperature Dowtherm "E" loop is liquid at 0° F and thus will be operable under all temperature conditions. Both the high and the low temperature loops may be thermal siphon loops that are completely independent of circulation pumps and auxiliary sources of power.

The condensing temperatures are high enough that 10 per cent of the reactor power may be removed without the aid of the fans on an attached fin-fan unit. The maximum pressure of either loop is about 70 psia. The two fluids and their thermal decomposition products are noncorrosive to carbon steel. Both Dowtherms are nontoxic and have high flash points. They are considered safer than sodium from the standpoint of fire. Leak-tightness is necessary, but not to the high degree which is required in sodium systems.

1. <u>Thermal Stability</u> - The effluent sodium of the 20-megawatt SRE is at 960° F and the inlet sodium is at 500° F. If the sodium were cooled by two boilers, the maximum temperature to which any of the Dowtherm "A" would be heated is 750° F, whereas the bulk fluid temperature will always be near 650° F.

Dowtherm "A" is ordinarily used in commercial practice until the decomposition products reach 15 per cent by weight. With this as a criterion the period of service to be expected from Dowtherm "A" at 750° F before purification is desirable is 3-4 months.

Many Dowtherm "A" boilers are in commercial use at 750° F; available information indicates it is used in boilers for indefinite lengths of time at 700° F mean fluid temperature. Dowtherm "E" is used in boilers indefinitely at a vapor temperature of 500° F with no apparent fouling or corrosion of heat transfer surfaces.

2. Radiation Stability - The energy release of the primary sodium during continuous 20-megawatt operation of the SRE has been calculated to be about $8 \times 10^9 \frac{\text{Mev}}{\text{cc-Na-sec}}$.

In a double walled tube boiler about 20 per cent of the gamma energy is absorbed in the organic fluid. Since less than 10 per cent of the total fluid in the system will be in the heat exchanger, the total energy absorbed in the Dowtherm is calculated to be $0.174 \times 10^{16} \frac{\text{Mev absorbed}}{\text{cc-system-yr}}$. Based upon experimental data obtained from the irridation of biphenyl, it is calculated that the expected life based upon radiation decomposition is about 400 years. This figure is corroborated by other calculations based upon the "G" value of biphenyl. Evidence indicates Dowtherm "A" is about half as stable as biphenyl and Dowtherm "E" is about one-tenth as stable in a low temperature ionization radiation field. Even so, it is believed that radiation deterioration is not serious compared to thermal decomposition.

I. Control Rod Studies (J. Howell)

As previously reported (Ref. 1, page 59) a thimble type control rod design is being developed for use with sodium graphite reactors. A Thimble Rod Experiment is being assembled to determine the heat transfer characteristics of such a design.

An electrically heated helical graphite heating unit is mounted vertically inside of a stack of boron coated steel control rod rings as shown in Fig. 15. Surrounding the rings is an annulus filled with static NaK which will be cooled by Dowtherm A. The Dowtherm in turn will be water cooled as shown in Fig. 16. The required information will be calculated from the flow, pressure, and temperature data.

All of the components for the experimental system have been fabricated and are now being assembled.

J. SRE - Ball Type Safety Device

A Ball Type Safety Device (Fig. 17) is being designed and studied for use in the SRE. Balls of boron steel or some other material containing a nuclear poison are stored in an annular hopper just above the top reflector of the reactor. To scram or shut down the reactor, a gate is opened by deenergizing a solenoid. The balls fall from the hopper to the bottom of an 8-foot long thimble.

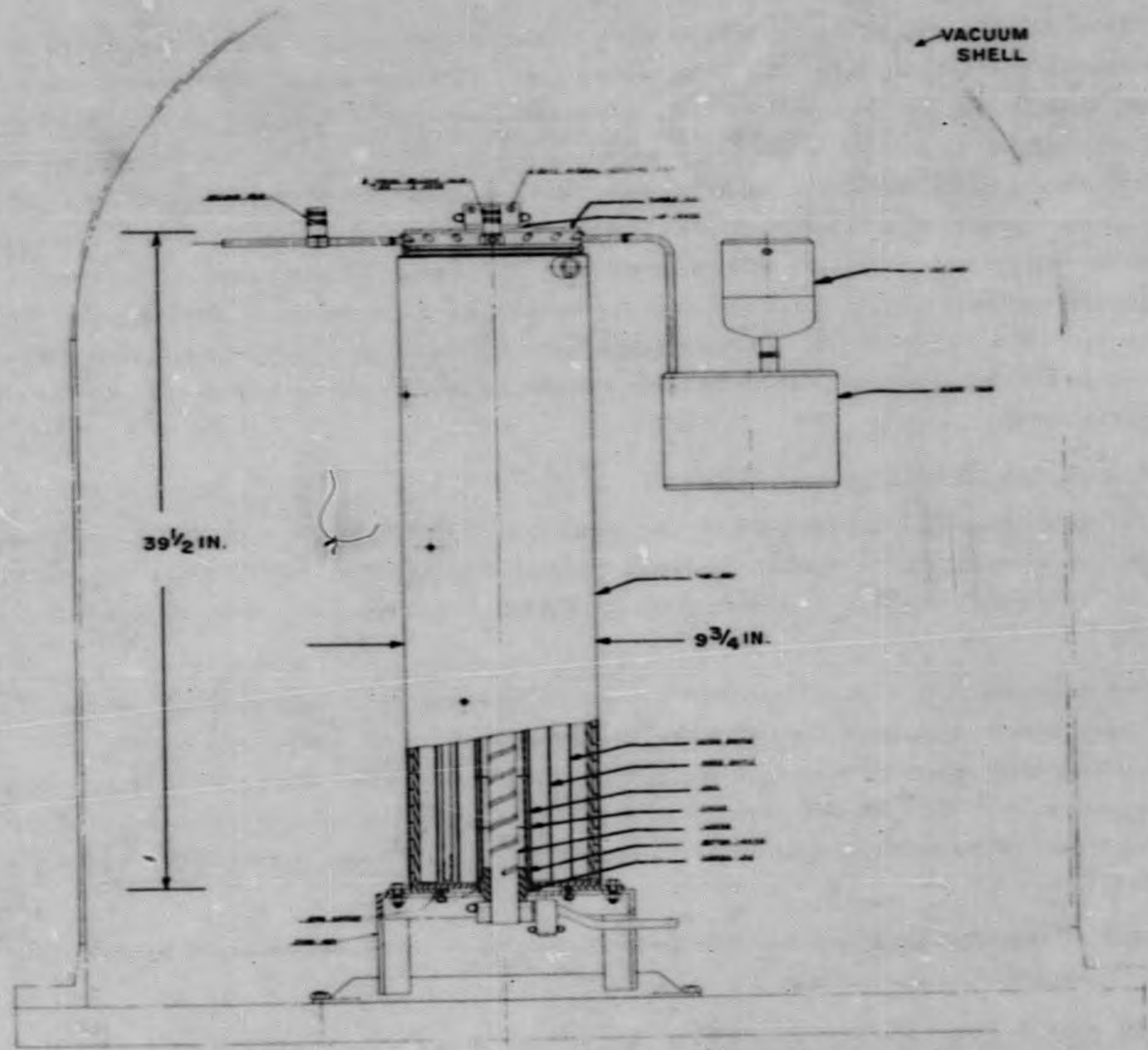

Fig. 15. Thimble Rod Experiment Test Vessel

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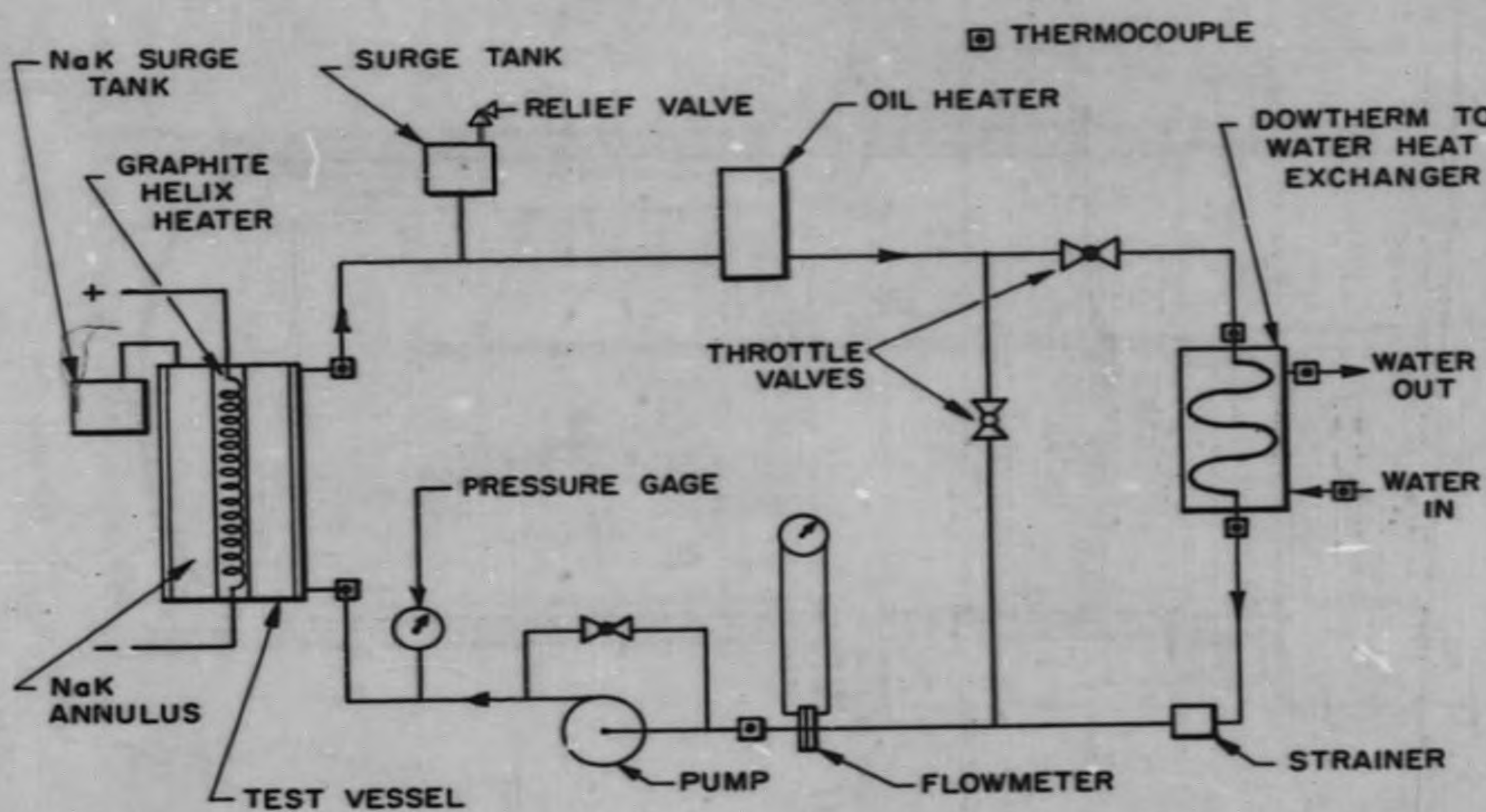


Fig. 16. Thimble Rod Experiment Heat Transfer System

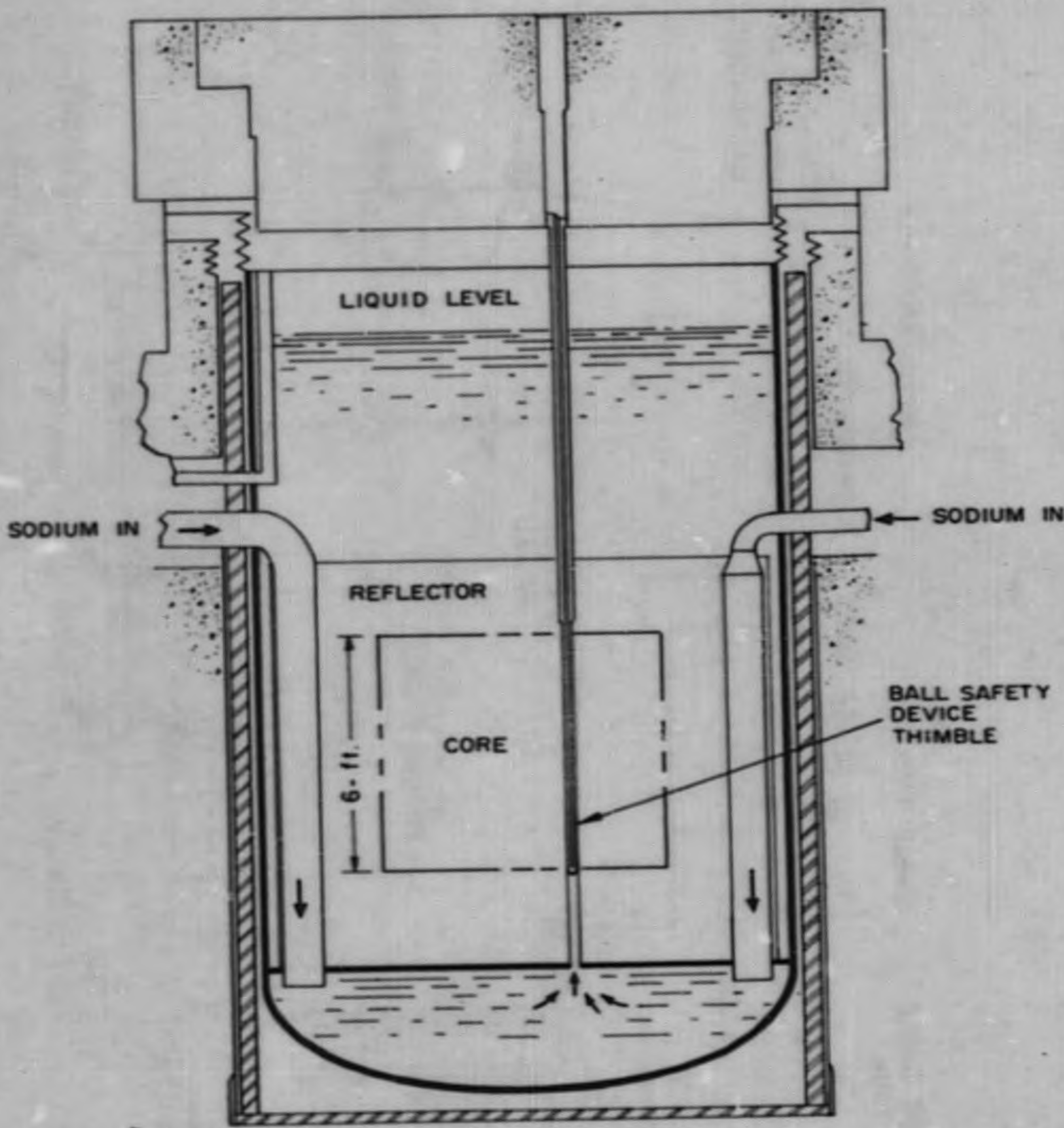


Fig. 17. Ball Safety Device Arrangement

1. Heat Transfer Characteristics (H. Sletten) - Assuming that gravity is the only force acting on the balls after they leave the hopper, the time required for a ball to fall 2 feet to the upper reflector-core interface is 0.350 second and the time required for a ball to traverse the 6-foot long core is 0.354 second. During the time that the first ball falls through the core region only a slight negative reactivity has been introduced and, consequently, this first ball is probably subjected to the maximum, full power, neutron and gamma fluxes. Due to the release of 2.8 Mev of energy (in form of α-particles) from the neutron capture in boron-10, heat is generated in the ball as it traverses the core. Other sources of heat such as gamma radiation and moderation are relatively small and are neglected. The ball temperature will change due to a difference between the rate of heat generation and the rate at which heat is liberated from the ball by radiation to the walls of the annular space and by convection to a helium atmosphere.

From the standpoint of the effects of temperature in the boron-steel balls, there appeared to be three conditions which may be critical: (1) heat generation and temperature reached by the first falling ball, (2) temperature of the balls in their static position in the hopper above the reflector, and (3) the temperature of balls that may remain at the bottom of the thimble after recovery. These three situations were studied for the Sodium Reactor Experiment, in which 1/8-inch diameter steel balls containing 2 per cent by weight boron are considered. It was found that the temperature of a first ball will increase not more than 8° F as it passes through the core at power level of 20 megawatts. The temperature of the balls in the hopper will be about 120° F above the temperature of a 1000° F hopper wall. A ball at the bottom of the thimble which is bathed by 500° F sodium at the core-reflector interface will have a temperature less than 1010° F.

2. Experimental Studied (E. Phillips) - A series of experiments are being performed in order to determine whether or not the steel balls may be recovered from the bottom of the ball thimble. The balls will be recovered by being blown out by gas pressure. It is necessary to determine the flow rate and gas pressure required.

The ball type safety device (Figs. 17 and 18) consists of three concentric tubes positioned vertically in the reactor core. The balls will be contained in a

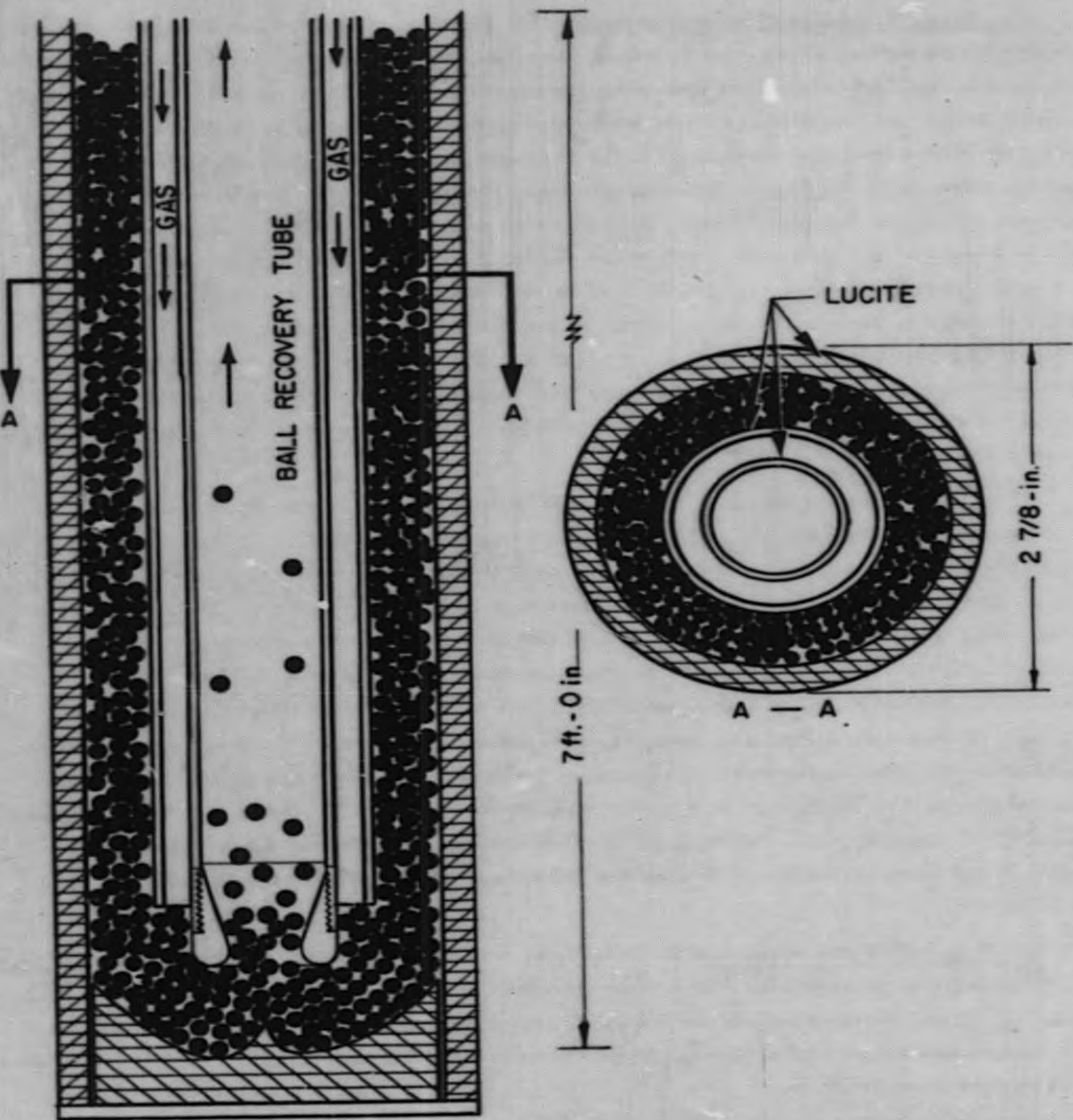


Fig. 18. Ball Safety Device (Schematic)

hopper which is attached at the top of the three tubes and located in the reflector above the core. A solenoid operated gate mechanism permits the balls to fall into the outer annulus of the ball thimble when the power is shut off. At the bottom of the ball thimble is a deflector cup which directs the balls from the outer annulus into the center tube. Gas flowing down the inner annulus forces the balls up the center tube.

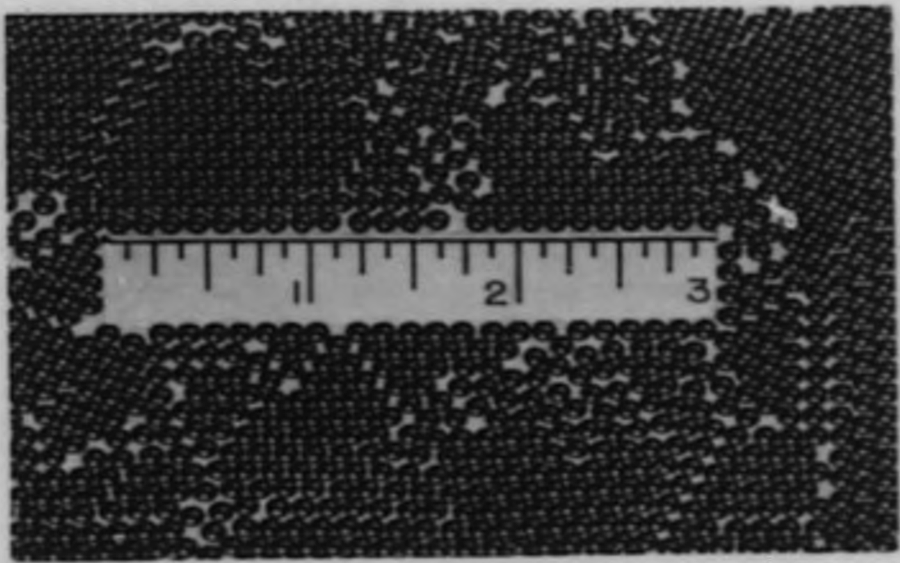
A prototype test apparatus has been fabricated of transparent plastic. The ball removal tests consist of varying air flow rates and the relative positions of the concentric tubes. It has been found that at lower flow rates the flow of balls tends to pulse and some are left in the deflector cup. At higher flow rates the balls flow smoothly and are completely removed. The tests have been performed using both steel balls and irregular iron shot (Fig. 19). An air pressure of 2.5 psig and a flow rate of 18 cfm removes 35 pounds (the required amount) of the balls in 6 minutes. A pressure of 9.5 psig and a flow rate of 24 cfm removes the balls in 23 seconds.

The design of the device is now being modified so that the gas flows down the center tube and the balls flow up the inner annulus. Preliminary tests indicate a much greater flow rate of gas is required. Modifications will be made to the deflector and the tests will be continued.

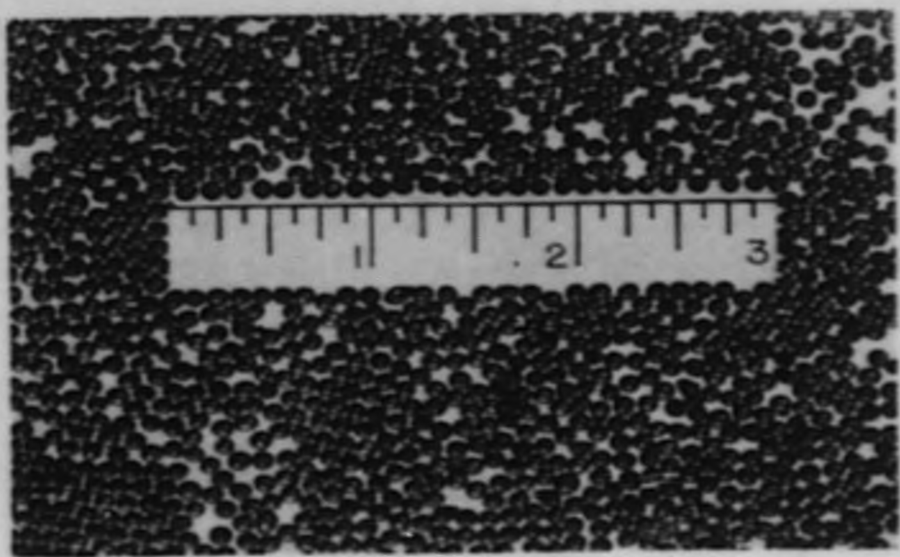
K. <u>SRE - Hydraulic Tests on Fuel Elements</u> (R. Hummel, T. Shimazaki)

The fuel element for the SRE consists of seven individual fuel rods which are bound together to form a bundle or cluster as shown in Fig. 20. The rods are approximately uniformly spaced by means of a spacer wire which is wound around the center rod and the alternate perimeter rods. The seven rods are held together by a tie wire at the upstream end and by a flexure section at the downstream end. The element is suspended in the reactor coolant tube by means of a hanger rod.

Because of the spacer wire wound on some of the rods, the pressure drop in the section of the coolant tube containing the fuel element cannot be accurately calculated. As previously reported (Ref. 1, page 77) a hydraulic test loop was erected to experimentally determine the pressure drops. The measurements were obtained under isothermal conditions using water as the fluid and with the pressure taps located as shown in Fig. 21. The pressure drop measurements

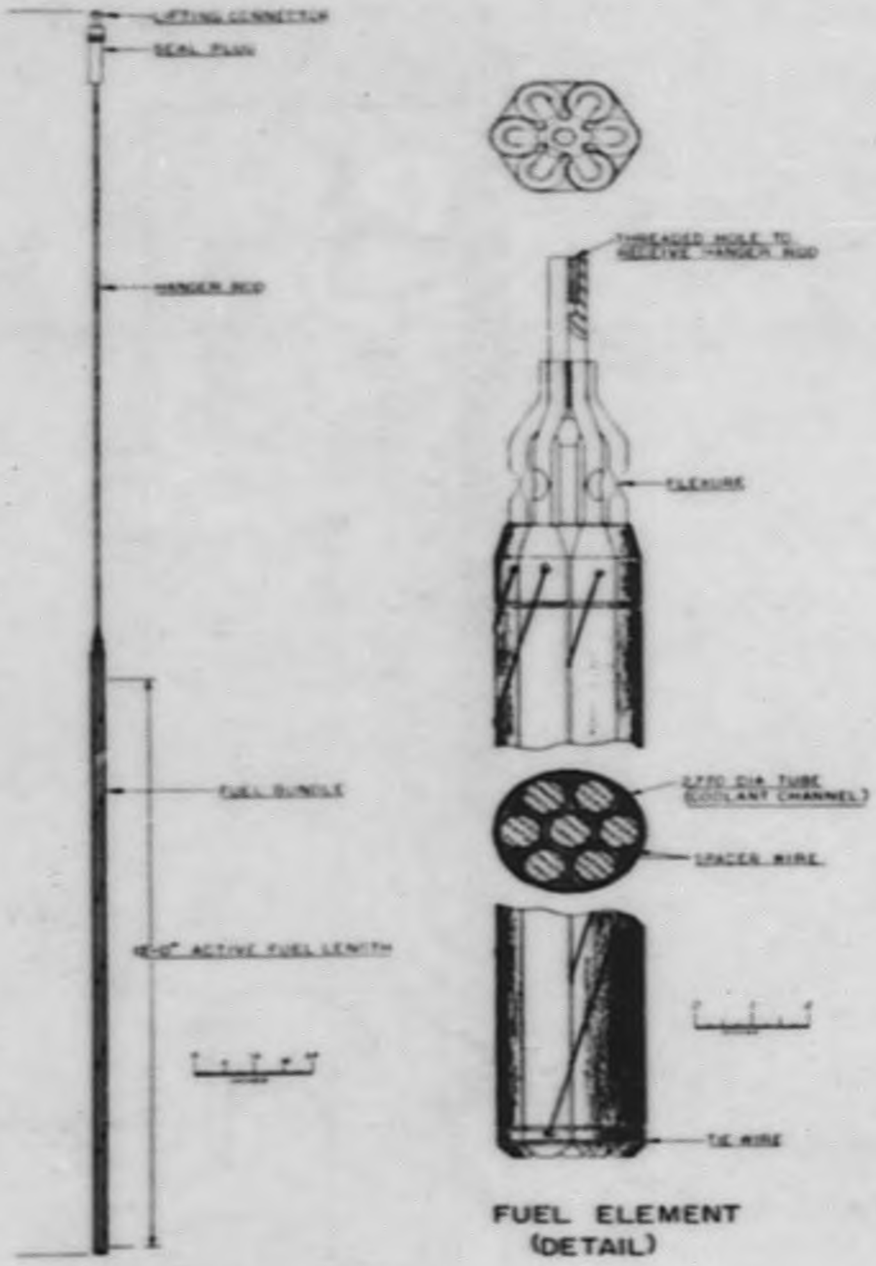


0. 125-inch diameter Grade 1 Chrome Steel Balls



0. 094-inch diameter Chilled Iron Shot

Fig. 19. Balls Used in Ball Safety Device



FUEL ELEMENT (DETAIL)

FUEL ELEMENT

Fig. 20. SRE Fuel Element Used for Hydraulic Tests

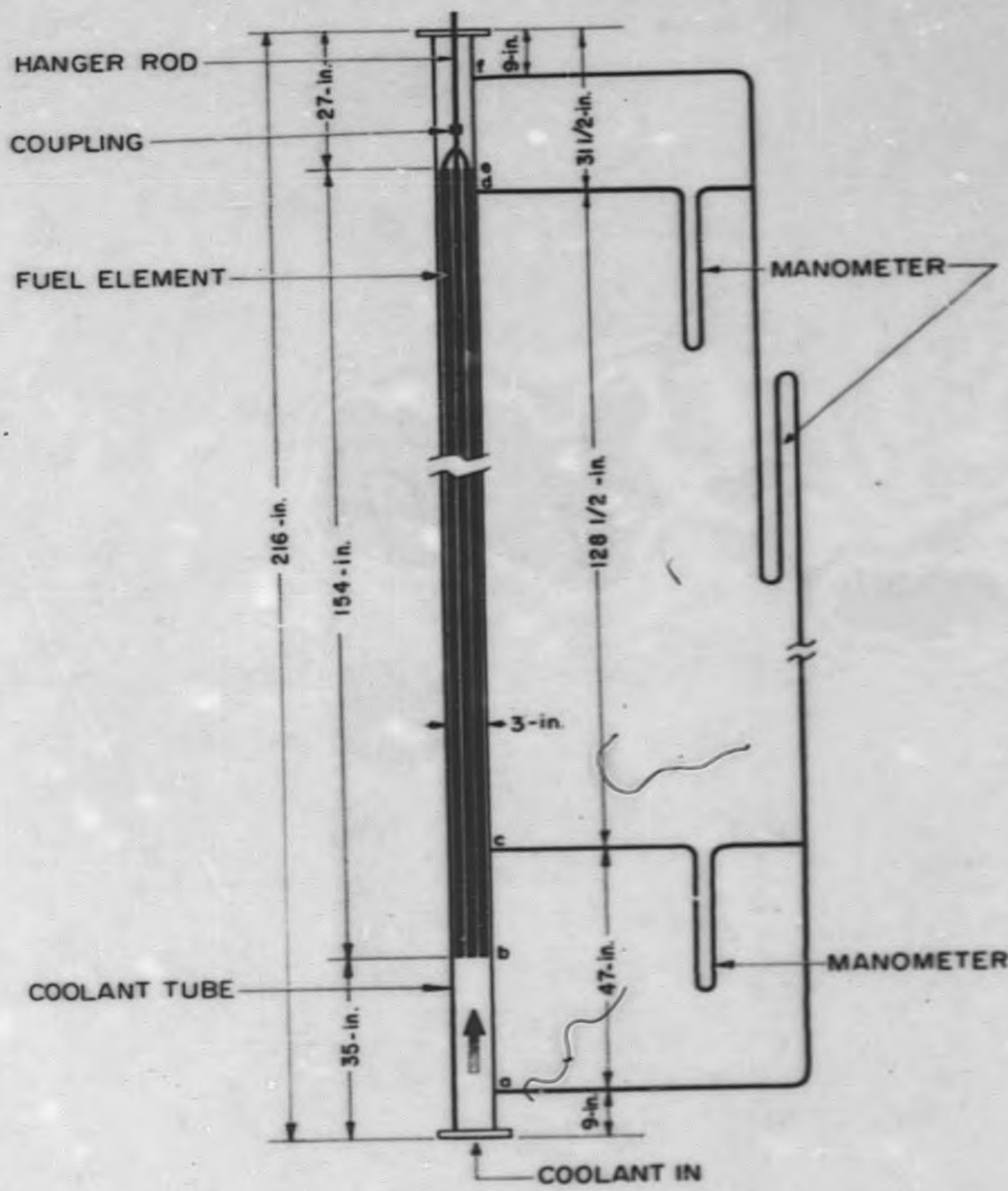


Fig. 21. Location of Pressure Taps for SRE Fuel Element Hydraulic Tests

taken and the range of Reynolds number were:

Pressure Drop	Pressure Taps	$Re = \frac{D_e v_a \rho}{\mu}$
Over-all	a to f	975 to 190,000
Entrance	a to c	8,700 to 190,000
Enlargement (exit)	d to f	18,200 to 190,000
Fuel element section	c to d	8,700 to 43,600

The fuel element section pressure drop measurement checked to within ± 5 per cent of the difference between the over-all pressure drop measurement and the sum of the entrance and enlargement pressure drop measurements. The Reynolds number is based on the equivalent diameter of the flow area in the fuel element section.

1. Friction Factor for Flow Through Fuel Element Section - The friction factor obtained for isothermal flow through the fuel element section is shown in Fig. 22. The data fall quite closely on the recommended friction factor curve (Ref. 2) for a pipe with a relative roughness, e/D_e, of 0.0010.

Experimental data indicate that the friction factor for isothermal turbulent flow through noncircular ducts annuli, and longitudinal banks of tubes or rods is approximately the same as for the circular pipe if the equivalent diameter is used as the diameter of the noncircular flow passage. Friction factors for isothermal flow parallel to banks of tubes of various diameters and spacings without spacers[3] are found to be slightly lower than the friction factor curve for smooth tubes shown in Fig. 22. The difference between the results of Ref. 3 and the present results can be attributed to the presence of the four wire spacers in the latter case.

2. Enlargement Head Loss. - The enlargement head loss is plotted in Fig. 23. In calculating the results shown, the pressure drop between points e and f (see Fig. 21) was assumed to be that for the annulus formed by the hanger rod in the coolant tube. The measured enlargement head loss can be expressed as:

$$h_e = \frac{1.23\,(v_d - v_f)^{1.85}}{2g} \qquad \ldots (1)$$

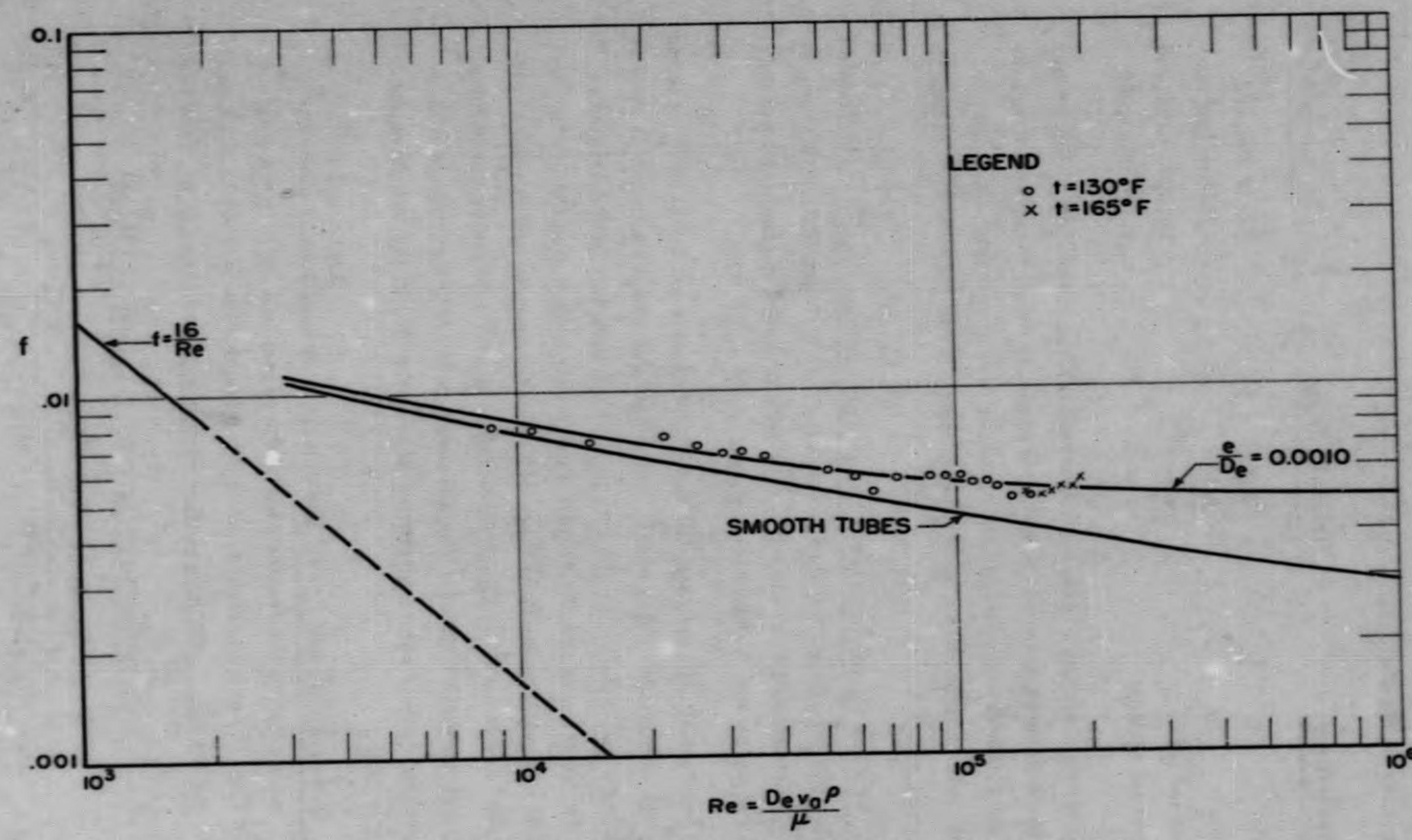


Fig. 22. SRE Fuel Element Friction Factor for Isothermal Flow

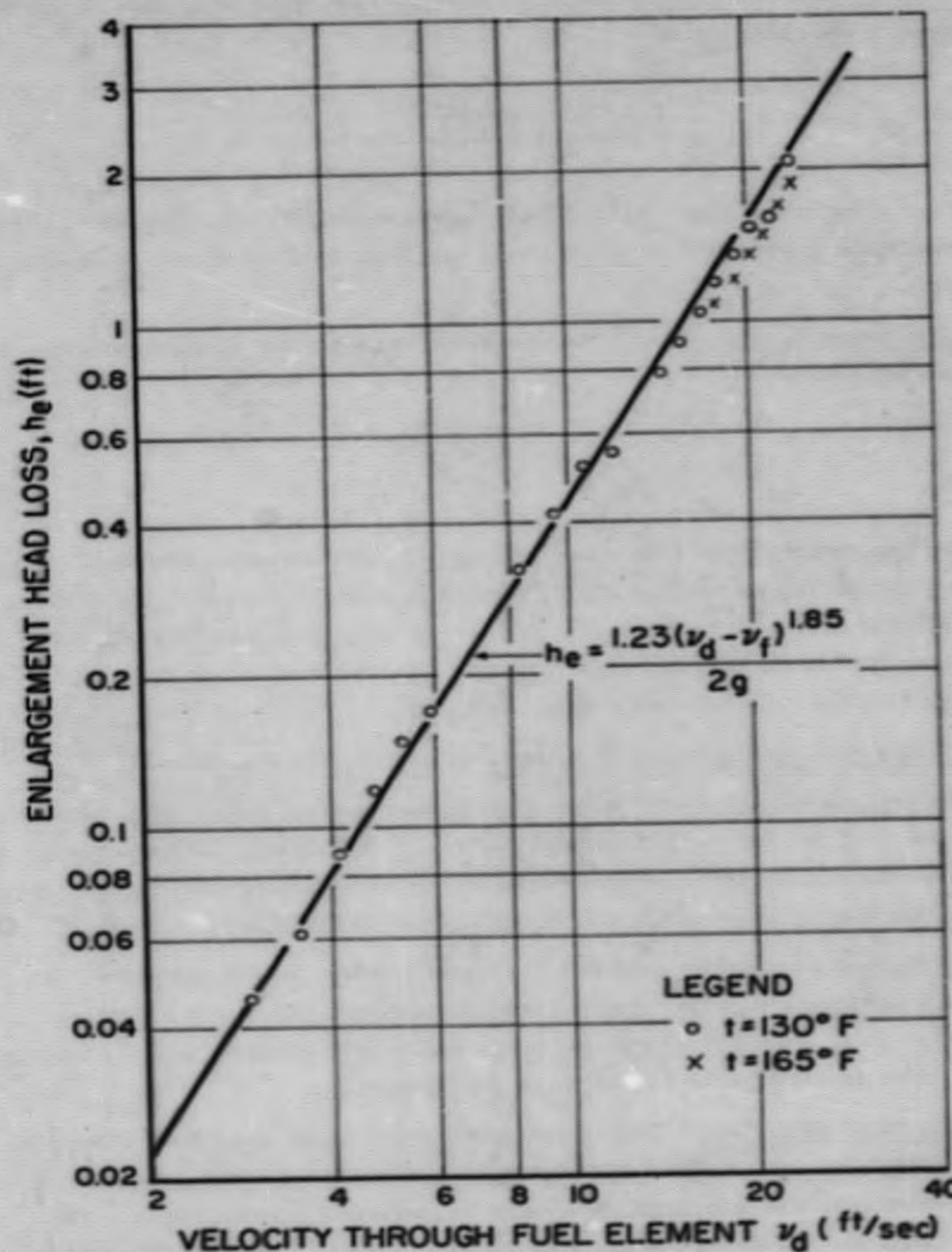


Fig. 23. SRE Fuel Element Enlargement Head Loss

On the basis of experimental measurements,[4] the head loss due to sudden enlargement in a circular pipe is

$$h_e = \frac{1.098\,(v_1 - v_2)^{1.919}}{2g} \quad . \qquad \ldots (2)$$

The difference between Eqs. (1) and (2) is undoubtedly due largely to the somewhat complicated geometry at the end of the SRE fuel element where the enlargement occurs.

3. Entrance Loss - The entrance head loss at the upstream end of the fuel element can be expressed as:

$$h = K \frac{V_d^{\,2}}{2g} \qquad \ldots (3)$$

where K is the entrance head loss coefficient. The results obtained show an apparently significant variation of K with Reynolds number. This is in variance with the well established experimental fact of only a relatively small effect of Reynolds number on K. The scatter of data in the region of Re $<10^5$ suggests that the data in this region may not be reliable.

L. SRE - Thermal Analysis of the Moderator (C. H. Robbins)

An investigation has been made of temperatures, stresses, and other thermal problems of the graphite moderator of the SRE. There appear to be no particular thermal problems involved in using zirconium canned graphite as the moderator, so long as the design is kept within certain broad limits.

Graphite temperatures are not likely to create any problems. In particular, high graphite temperatures are not likely to weaken the zirconium can unless the sodium is also hot, since the temperature of the zirconium will be essentially the same as the sodium with which it is in contact.

1. Thermal Stresses - The maximum estimated thermal stress in the graphite, 820 psi, occurs when all the heat generated in the graphite flows to the coolant tube and the graphite has been irradiated a long time. This is somewhat lower than the minimum expected tensile strength of AGOT graphite of 1000 psi. Stresses for varying amounts of heat flow to the outside of the graphite

block may be estimated by using the maximum figure of 820 psi and the relative values shown on Fig. 24.

2. Graphite Growth Due to Irradiation Damage - TID 5021 Fig. 25.1 shows that the maximum expansion of a single graphite crystal (increase in C_o lattice spacing) amounts to about 0.3 per cent if the graphite is irradiated at temperatures in excess of 275° C. The expansion of a large piece of graphite composed of many crystals randomly oriented is likely to be about one-sixth of the increase in the C_o lattice spacing of a single crystal. All the graphite in the SRE will probably be hotter than 275° C, since the minimum sodium temperature will be 500° F (260° C) and a temperature difference must exist to transfer heat from the graphite to the sodium. On the foregoing basis, the expansion across the flats (10.84 inches) due to radiation damage would amount to 5.42 mils. This increase is small compared to tolerances required in manufacturing the cans.

3. Conclusions - If thermal stress exceeded the strength of the graphite, cracks would probably appear at the surface. These cracks would be vertical, even though the calculations show the axial and circumferential stress to be equal at the surface, because graphite is stronger in the direction of extrusion. Probably the development of cracks would relieve the stress before the structural properties of the graphite were seriously impaired.

There is considerable error possible in the calculation of thermal stresses, because of some of the assumptions and because the properties of graphite after irradiation at high temperature are not well known. When better values for the properties of irradiated graphite are available it may be desirable to calculate graphite stresses using more exact methods. For example, the effect of variation of properties with direction can be considered.

It is desirable to design the moderator cans to provide a safety factor for thermal stresses. Thermal stress in the graphite will not be excessive according to the calculations (Fig. 25) and even if it does exceed the graphite strength, the moderator blocks will probably not be badly damaged. However, since the zirconium cans are to have sodium flowing around the outside, it is easy to provide a safety factor for thermal stresses. This can be done by forcing most of the heat generated in the graphite to flow to the outside of the block. If no other factors influence the design, it would be desirable to aim for the minimum possible tensile stress by having 30 per cent of the graphite heat carried off by the coolant tube.

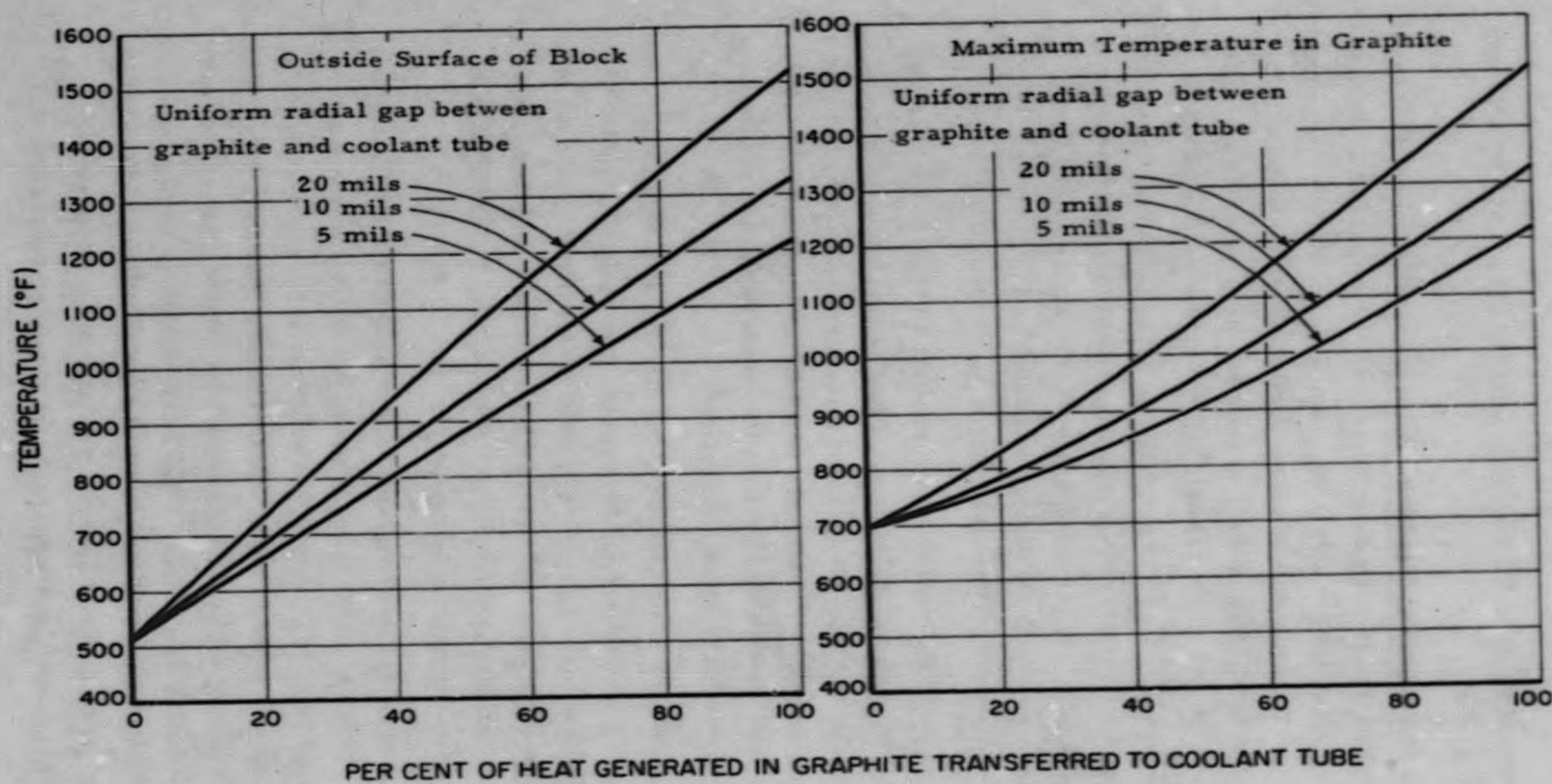


Fig. 24. Temperatures of Radiation Damaged Graphite in the Horizontal Mid-plane of a Central SRE Moderator Cell

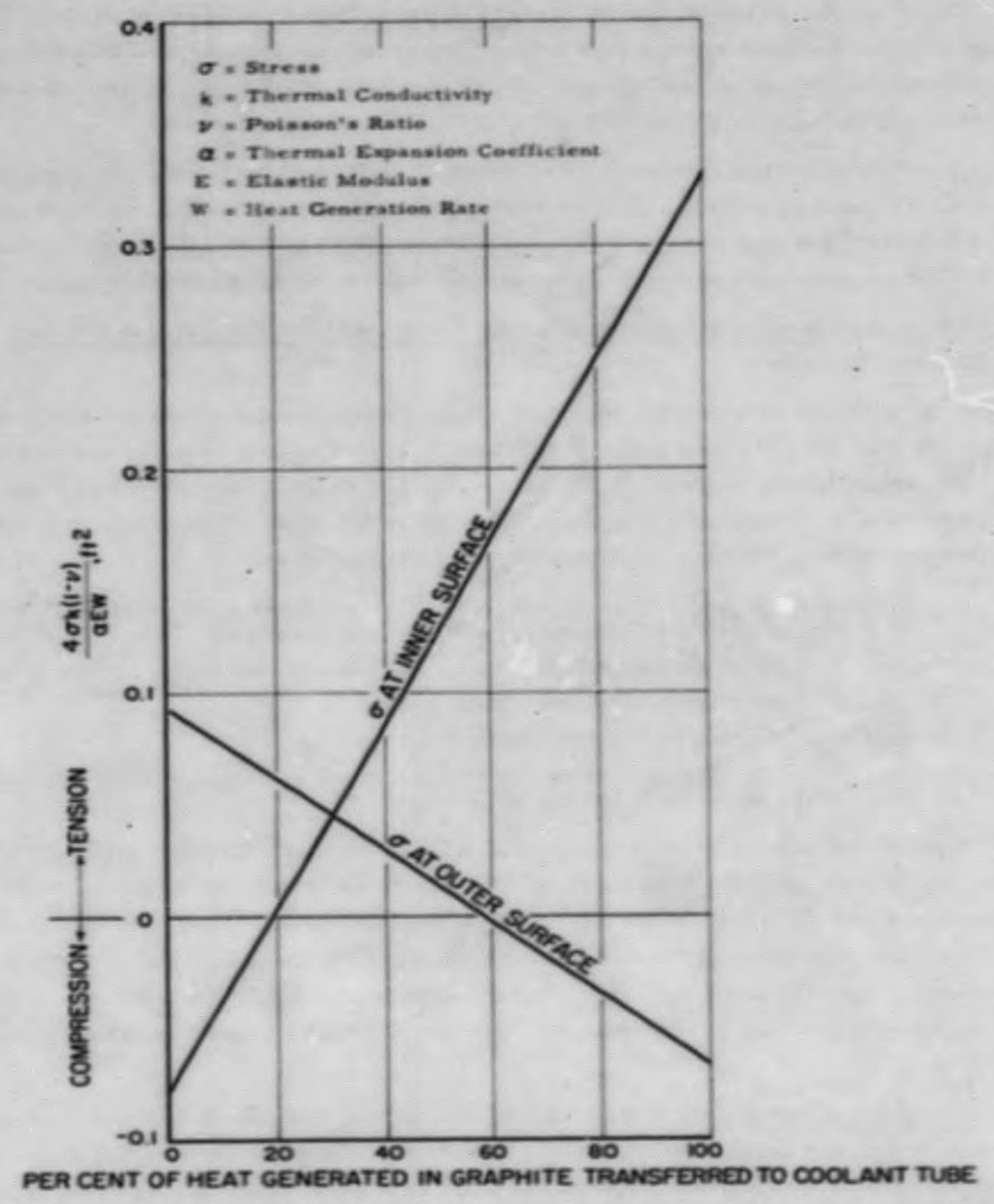


Fig. 25. Thermal Stress Due to Uniform Generation of Heat in a Long Hollow Cylinder Equivalent to the SRE Graphite Block

None of the dimensions of the graphite or the can appear critical from the standpoint of thermal stress and temperature of the graphite. Therefore, the dimensions and clearances should be determined primarily by the requirements for manufacturing and assembling the can and the graphite.

A cold diametral clearance of about 0.040 inch between the graphite and coolant tube is suggested. To provide a safety factor for thermal stress, it is desirable that the gap between the outside of the block and the can be not more than twice as much as the gap between the tube and the graphite hole.

M. SRE - Cooling Requirements of the Thermal and Biological Shields

(M. P. Heisler)

A preliminary analysis of the cooling requirements for the SRE thermal and biological shields has been completed. The cooling system utilizes toluene which flows in pipes imbedded in the biological shield concrete and attached to the 0.25-inch steel plate Core Cavity Liner (Fig. 26). The arrangement and thicknesses of the various components are as follows:

1. Thermal Shield - The surfaces of the reactor will have 7-3/4 inches of ferritic thermal shielding divided as follows:
   a. 1.5 inches of stainless steel in Core Tank wall
   b. 5.5 inches of iron Thermal Shield
   c. 0.25 inch of steel in Outer Tank wall
2. Thermal Insulation - The outer tank wall is lined with 9 inches of insulating firebrick or its equivalent.

Below the bottom of the core tank 4.5 inches of insulating firebrick or its equivalent is sufficient to reduce the heat loss to an acceptable value. The full thermal shield heating load flowing across the core tank wall produces less than 7° F temperature difference. The thermal stresses arising from this are negligible. Hence, it is not necessary to protect the core tank from heat generation caused by gamma ray absorption and no thermal shield is needed within the tank.

3. Biological Shield - The thickness of the Portland concrete biological shield has not yet been determined. However, the cooling requirement for the shield should be adequately represented by the data given under Item 6 as long as the thickness is greater than 2 feet. At this distance from the Core Cavity

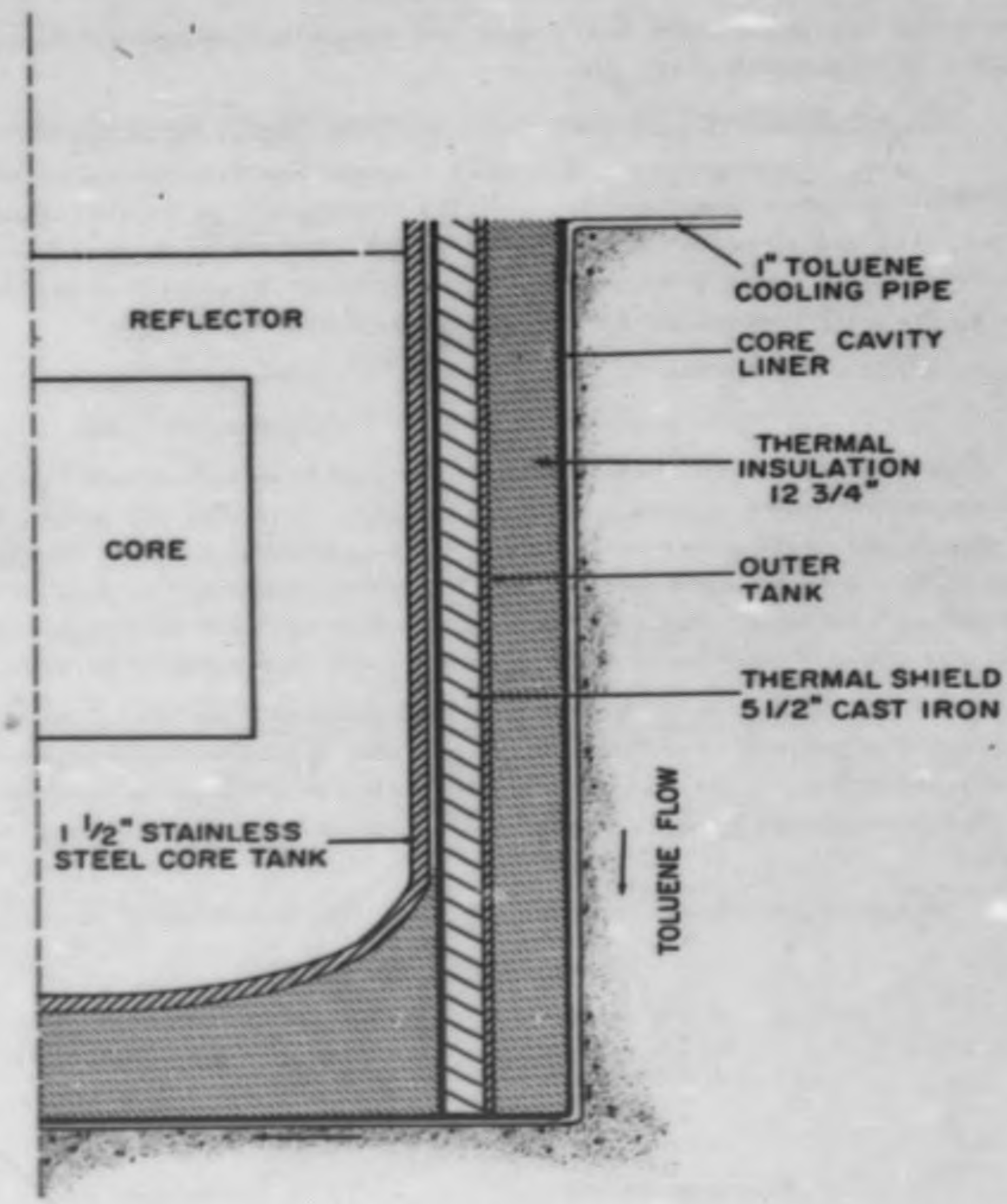


Fig. 26. SRE Core Arrangement

Liner the heat generation in the concrete virtually disappears for the calculated radiation intensities (Fig. 27).

4. Cooling Requirements of the Various Shield Components

a. Assumptions - The peak thermal and fast neutron leakages from the reflector have been calculated on the assumption of an average core diameter of 6.2 feet and a radial reflector of 2.4 feet. On the basis of a power level of 20 megawatts, which corresponds to a maximum thermal flux of 6×10^{13} in the core, the peak leakages out of the reflector were found to be

fast group	6.87×10^{10} neutrons/cm^2-sec
thermal group	5.45×10^{11} neutrons/cm^2-sec

As a step toward conservatism in design it was assumed that the neutron fluxes quoted above represented the average rather than the peak values as far as the shield cooling was concerned. This assumption should offset to some extent the inherent inaccuracy of shielding calculations. In view of the fact that the shield cooling system is a very minor item in the total reactor cost, conservatism costs very little in this instance and is therefore advisable.

b. Thermal Shield Heating Calculations - On the basis of the above assumptions the total heat load in the thermal shield was computed to be approximately 1200 Btu/ft^2-hr. In addition about 300 Btu/ft^2-hr is returned to the reflector in the form of capture gammas which escape absorption in the thermal shield.

The heat generation is proportioned within the thermal shield as follows:

Thermal Capture Gammas	
Core Tank	620 Btu/ft^2-hr.
Thermal Shield and Outer Tank	300 Btu/ft^2-hr.
Fast Capture Gammas	
Thermal Shield	120 Btu/ft^2-hr.
Core and Reflector Gammas Thermal Shield	160 Btu/ft^2-hr.
Total	1200 Btu/ft^2-hr.

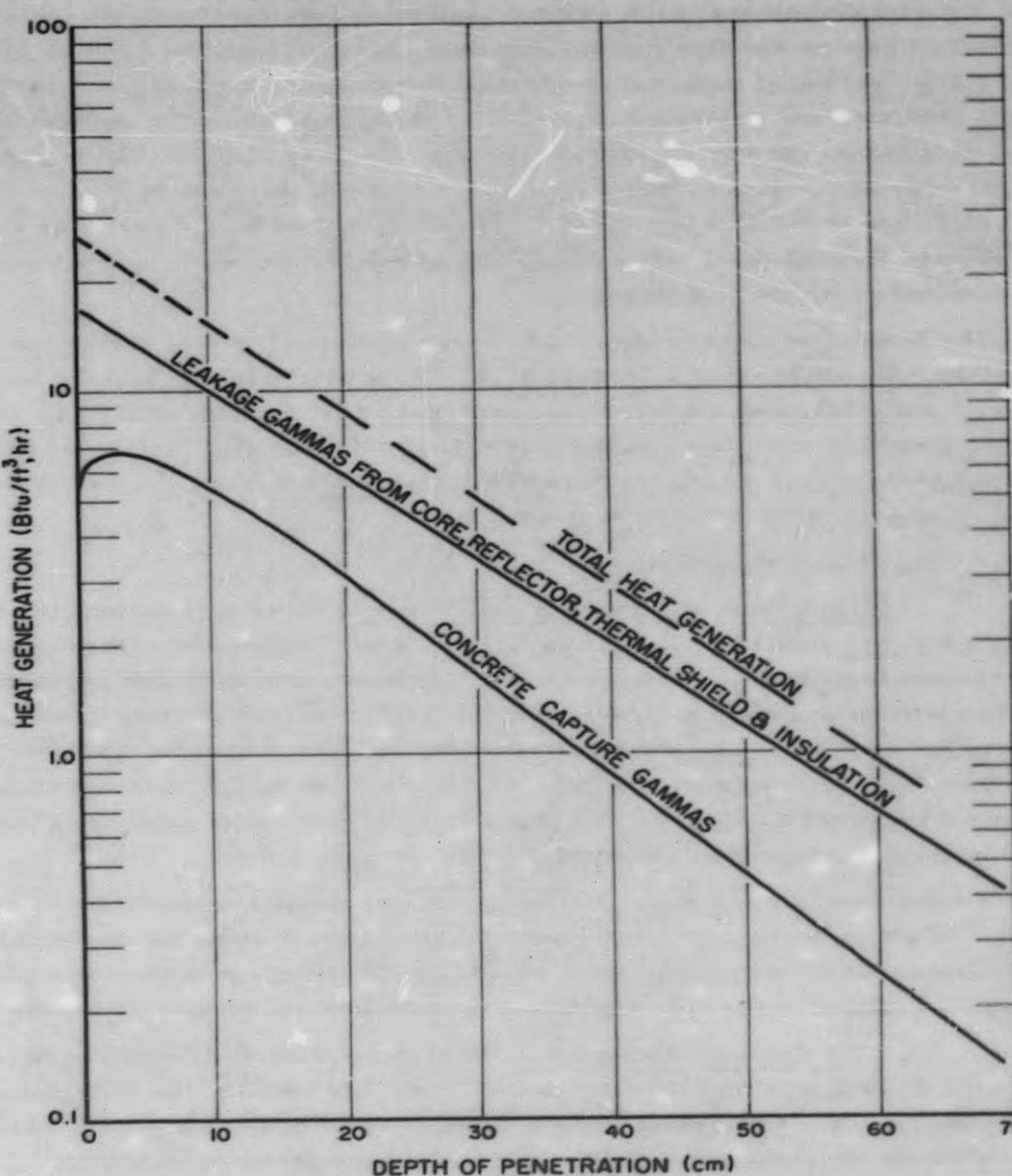


Fig. 27. Heat Generation in SRE Biological Shield Concrete

These figures assume that each neutron capture in iron results in the emission of an 8 Mev gamma and that gamma absorption occurs on the basis of single flight collision-both of these are conservative representations of the actual gamma emission and absorption process. The thermal shield unloads its heat in two directions. The larger part of the heat, roughly 1140 Btu/ft^2-hr, is carried away by the sodium which flows upward along the inner face of the stainless steel core tank. The remainder, approximately 60 Btu/ft^2-hr. flows radially outward through the thermal insulation and is carried away by the toluene in the biological shield cooling pipes.

The amount of sodium required to carry away the thermal shield heat is negligible. On the basis of 1200 Btu/ft^2-hr °F, a heat transfer surface 11 feet high by 1 foot wide, and a temperature increase in the sodium of 300° F, each circumferential foot of tank would require 4.5 cubic feet of sodium per hour. The data assumed are rough but nevertheless are adequate for illustrating the small magnitude of the coolant flow involved.

c. Thermal Insulation

(1.) Choice of insulating material - In these calculations the thermal insulation is assumed to be insulating fireclay brick. They are essentially mullite crystals embedded in silica glass, and there is some basis for presuming that they will be satisfactory under radiation since experimental data indicate glass is stable. Also, the insulation is placed outside the thermal shield and is essentially subjected to the same intensity of radiation as the concrete biological shield. If concrete does not deteriorate under the radiations existing in the SRE, then insulating firebrick will probably not deteriorate either.

Substitution of steel wool or other fibrous or powder insulation will not materially affect the cooling requirements stated herein since the thicknesses and densities involved are relatively small. Some change in dimensions may be necessary because of the different thermal conductivities of the various materials.

(2.) Cooling calculations - Heat generation due to gamma absorption within the firebrick is less than 4.5 Btu/ft^2-hr. In contrast, the heat lost by conduction from the hot thermal shield amounts to 60 Btu/ft^2-hr on the basis of a mean shield temperature of 750° F and coolant temperature of 100 °F. It is evident that the heat generated internally within the insulation will not materially affect the insulating effectiveness of the firebrick.

The gap thickness between the thermal shield and the Core Tank will also have only a small effect on the heat loss. Assuming only radiant heat exchange (emissivitiy of stainless steel 0.4 and iron 0.6)--i.e., no conduction across a gas layer between the shield and the tank--we find that a temperature difference of 90° F between shield and tank will transfer all the heat across the gap by radiation. This will increase the loss through the insulation by 15 per cent.

d. Biological Shield - Roughly 15 Btu/ft^2-hr are generated in the biological shield. The breakdown of this between capture gammas in the concrete resulting from a 7-Mev gamma emitted upon neutron capture and gammas coming from the core, reflector, thermal shield, and insulation is shown in Fig. 27. Fast neutrons were assumed to be attenuated exponentially with a neutron relaxation length of 11.1 centimeters and a gamma attenuation coefficient of 0.058 cm^{-1} was used.

From Fig. 27 it is seen that very little heat generation occurs beyond a depth of about 2 feet. Also, the heat generation is exponential except for a small region near the surface. It was calculated that the maximum temperature difference in the biological shield would be less than 10° F. With this small temperature difference no stress problem exists in the biological shield.

5. Conclusions - The shield cooling requirements for the SRE can be summarized as follows:

a. The requirements for the shield cooling system are quite nominal. A liquid cooling system, such as toluene, can be installed at small cost.

b. Radial temperature gradients across the tank wall are quite small and internal protective devices, such as placing portions of the thermal shield inside the tank are not necessary from the standpoint of internal heat generation in the tank wall.

c. The load on the cooling system is only slightly affected by the presence of gaps between the thermal shield and the tank. Radiation alone can transmit the heat across the gaps with a relatively small increase in thermal shield temperature.

d. Temperature gradients and thermal stresses are quite small in the biological shield. There is a possibility that since this is the case the thermal shield can be reduced in thickness from the 7-3/4 inches assumed here.

IV. ALTERNATE REACTOR DESIGNS

A. A 145-Megawatt Sodium-Graphite Plutonium Plus Power Reactor

1. Introduction - Conceptual design studies had been carried out earlier on sodium-graphite reactors intended for the plutonium plus power and for the power only objectives.[5] As a result of renewed interest on the part of the Reactor Development Division of the Atomic Energy Commission in the large-scale production of plutonium, consideration has been given to the design of a reactor intended for plutonium production only. This reactor is a sodium-graphite unit of intermediate size, (145 thermal megawatts) and capable of the production of approximately 141 grams of plutonium per day. Basic data for the reactor are given in Table V.

This reactor uses low enrichment uranium fuel assembled into 19-rod fuel elements. The core tank which contains the moderator and reflector is located below the floor level. All control rods and fuel elements enter the reactor from above. Four parallel sodium circuits remove heat from the core. Each circuit contains a primary radioactive loop coupled by a sodium-to-sodium heat exchanger to a secondary nonradioactive loop. The present design calls for the discharge of the heat to the atmosphere through a sodium-to-air heat exchanger.

2. Reactor Arrangement - The reactor is of the tank type, which is installed in the ground so that the top of the reactor shield is flush with the ground level (Fig. 28). Using this arrangement the fuel handling coffin which is used for charging and discharging the reactor can be carried by an overhead bridge crane to the fuel storage and cleaning area.

Surrounding the core tank is a steel thermal shield, a steel cavity liner, thermal insulation, and the biological shield of concrete. The upper shield is essentially a concrete plug with the necessary openings.

Individual graphite moderator units are canned in zirconium cans and are positioned vertically in the core tank. The fuel elements are hung vertically in coaxial channels in the moderator cans. Coolant sodium flows upward through the coolant channels.

3. Afterglow Heat Removal - The safety of this reactor would be increased if the coolant would continue to flow and thus remove the afterglow heat following

TABLE V

NUCLEAR AND PHYSICAL CHARACTERISTICS FOR A 145-MEGAWATT SODIUM-GRAPHITE PLUTONIUM PLUS POWER REACTOR

Moderator	Graphite
Coolant	Sodium
Cladding	0.015 inch Zr
Coolant Tube	0.035 inch Zr
Fuel rod diameter	0.455 inch
19 rods per element	
Lattice Constants	
η	1.661
ϵ	1.047
p (hot)	0.7023
f (hot)	0.9424
k hot poisoned	1.151
L_f^2 hot	369 cm^2
L^2 hot	111 cm^2
B^2 hot poisoned	290×10^{-6} cm^{-2}
Initial conv. ratio	0.80
Enrichment (initial loading 2% k_{excess})	0.022
Power Distribution	
Reflected length/unreflected length	0.703
Central/average rod power	0.670
Peak/average thermal flux	1.85

Reactor Assembly	
Active fuel length	7.0 ft
Total number of fuel elements*	81
Total core volume	1470 ft^3
Active core diameter	7.5 ft
Diameter of graphite stack	12.5 ft
Weight of uranium	6450 kg
Weight of graphite	73.5 tons
Center-to-center rod spacing	9.00 inch
Triangular lattice	
Data for Central Rod	
Power	2640 kw
Specific power	1500 kw/kg 25
Max. uranium temperature	1200° F
Coolant inlet temperature	300° F
Coolant outlet temperature	640° F
Coolant velocity	20 ft/sec
Coolant pressure drop	24 psi
Operating Conditions	
Total reactor power	145 megawatts
Average specific power	1020 kw/kg 25
Pu production rate	38.5 kg/yr
Average thermal flux	3.82×10^{13}
Peak thermal flux	7×10^{13}
Mixed mean sodium temp.	780° F
Total coolant flow rate	3.32×10^6 lbs/hr

*91 channels, 10 for control and safety rods

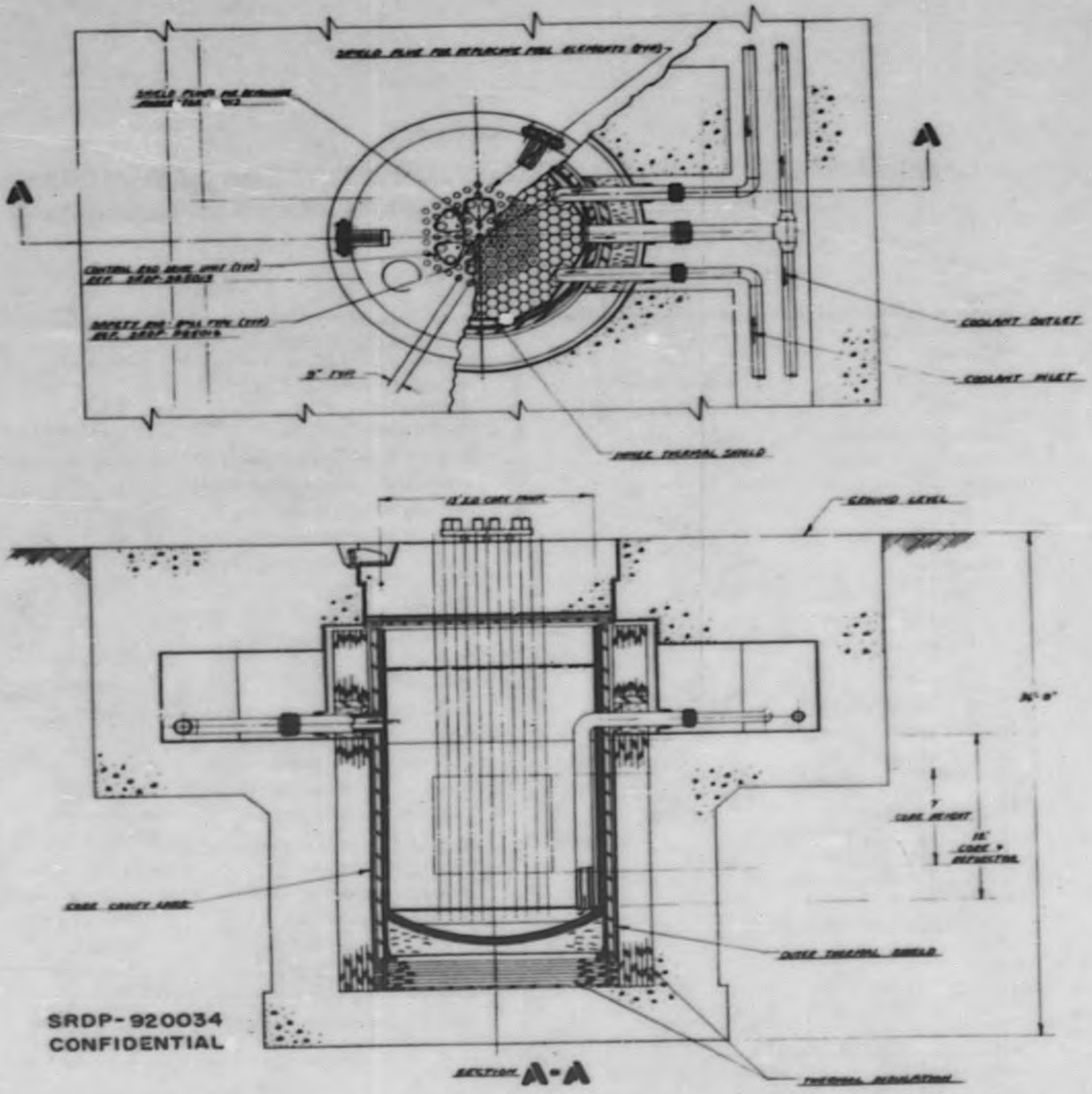


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Fig. 28. Basic Dimensions of Sodium Graphite Plutonium Plus Power Reactor

a reactor scram and pump power failure. Assuming a maximum sodium temperature of 1200° F, and further assuming that thermal convection need remove only 1 per cent of the maximum reactor thermal power, a preliminary analysis of the cooling system arrangement has been made. The driving head pressure required was calculated to be approximately one-half that which is available from the vertical distance between the heat exchangers. It appears that thermal convection in only one of the four sodium coolant loops will handle the required 1 per cent of the reactor power.

B. A Natural Convection Sodium Cooled Reactor (T. Shimazaki)

Power cost studies for nuclear power plants show the importance of low first cost in the achievement of competitive electric power from nuclear power plants. Novel approaches seem to be needed if lower first cost is to be attained in any of the current power reactor designs in the "immediately feasible" class. A natural convection sodium cooled reactor in which the heat from the sodium is transferred to the working fluid through the walls of the tank containing the reactor and sodium has been suggested as an approach to lowering the first cost of sodium cooled reactors. The feasibility of a sodium cooled reactor contained in a hermetically sealed tank, in which the coolant flows by natural convection to transfer heat from the fuel elements to the working fluid (water) has been investigated for a 10 thermal megawatt reactor. Beryllium was assumed for the moderator and reflector in order to make the reactor as compact as possible. The effect of using graphite for the moderator and reflector would be simply to increase the size of the reactor and, therefore, the reactor tank. Two cases were considered. In Case A, the heat is transferred from the sodium to the working fluid through the reactor tank wall. In Case B, the heat is transferred from the sodium to the working fluid through a heat exchanger enclosed in the reactor tank.

The data assumed for the reactors studied are given in Table VI.

1. Case A - For the reactor studied as Case A the heat is transferred from the sodium to the working fluid through the reactor tank wall, and the tank must be approximately 6 feet in diameter by 55 feet long if saturated steam is generated at 150 psi. The reactor tank (Type 347 stainless steel) is completely jacketed and is used as a heat transfer surface.

The reactor is located at the lower end of the tank. The net power plant efficiency for this case, assuming a 2-inch Hg condenser pressure, is about

TABLE VI

BASIC DATA FOR A NATURAL CONVECTION SODIUM COOLED REACTOR

1. Core and Reflector	
a. Material	beryllium
b. Over-all dimensions	55 in. diam. by 54 in.
c. Effective core diameter	31 in.
d. Number and arrangement of fuel channels	21 (square array)
e. C-C fuel channel spacing	6 in.
f. Fuel channel diameter	3.25 in.
2. Fuel Element	
a. Fuel rod diameter	0.455 in.
b. Cladding	0.015 in. zirconium
c. Bond	mechanical
d. Number of rods per fuel element	19
e. Active fuel rod length	30 in.
f. Number of fuel elements	21
3. Power Distribution	
a. Reflected length/unreflected length	0.655
b. Central/average fuel element power	1.39
c. Peak/average thermal flux	1.67
4. Data for Central Fuel Element	
a. Power	660 kw
b. Maximum fuel temperature	1200° F
c. Coolant inlet temperature	450° F
d. Coolant outlet temperature	824° F
e. Coolant velocity	3 ft/sec
f. Pressure drop	0.07 psi
5. Operating Conditions	
a. Total reactor power	10 megawatts
b. Fuel channels orificed for maximum permissible fuel temperature	
c. Coolant inlet temperature	450° F
d. Coolant mixed mean outlet temperature	900° F

20 per cent. The non-superheat, low pressure steam cycle is employed in order to keep the size of the reactor tank to a minimum. Superheating will increase the required heat transfer area substantially due to the low heat transfer coefficient of steam. Increasing the steam pressure will also increase the required heat transfer area since:

1. The tank wall thickness, which is the major thermal resistance between the sodium and water, must be increased, and
2. The temperature drive between the sodium and the water is reduced.

The height of the reactor tank is dictated by the required heat transfer area. Increasing the tank diameter will not result in any significant reduction in height due to the fact that when the tank diameter is increased the tank wall thickness must be increased.

By using a reactor tank with a corrugated wall, as shown in Fig. 29, the over-all size of the reactor tank can be substantially reduced. For example, if the corrugation is 3 inches high with 1-inch radius bends, the side wall area is approximately doubled and the wall thickness is halved so that the over-all height of the reactor tank can be reduced to about 25 per cent of the original height of 55 feet or to about 16 feet.

2. Case B - The arrangement for the reactor considered as Case B is shown in Fig. 30. A "once-through" type boiler is located in this tank near the upper end. The boiler consists of 72 helium filled double wall tubes. Boiler feed water enters the tubes at 220° F and leaves as steam at 700 psig and 775° F. For a 2-inch Hg condenser pressure, the net power plant efficiency is approximately 30.5 per cent.

If graphite is used for the moderator and reflector, the over-all size of the core and reflector becomes approximately 7.5 feet in diameter by 7.5 feet high. This requires that the reactor tank diameter increases from 6 feet to 9 feet.

3. Conclusions - The 10 thermal megawatt, natural convection sodium cooled reactors considered in the present study offer the following advantages:

a. A substantial reduction in first cost due to elimination of coolant pumps and intermediate coolant loop and heat exchanger.
b. Relatively compact.
c. Possiblity of factory assembly.

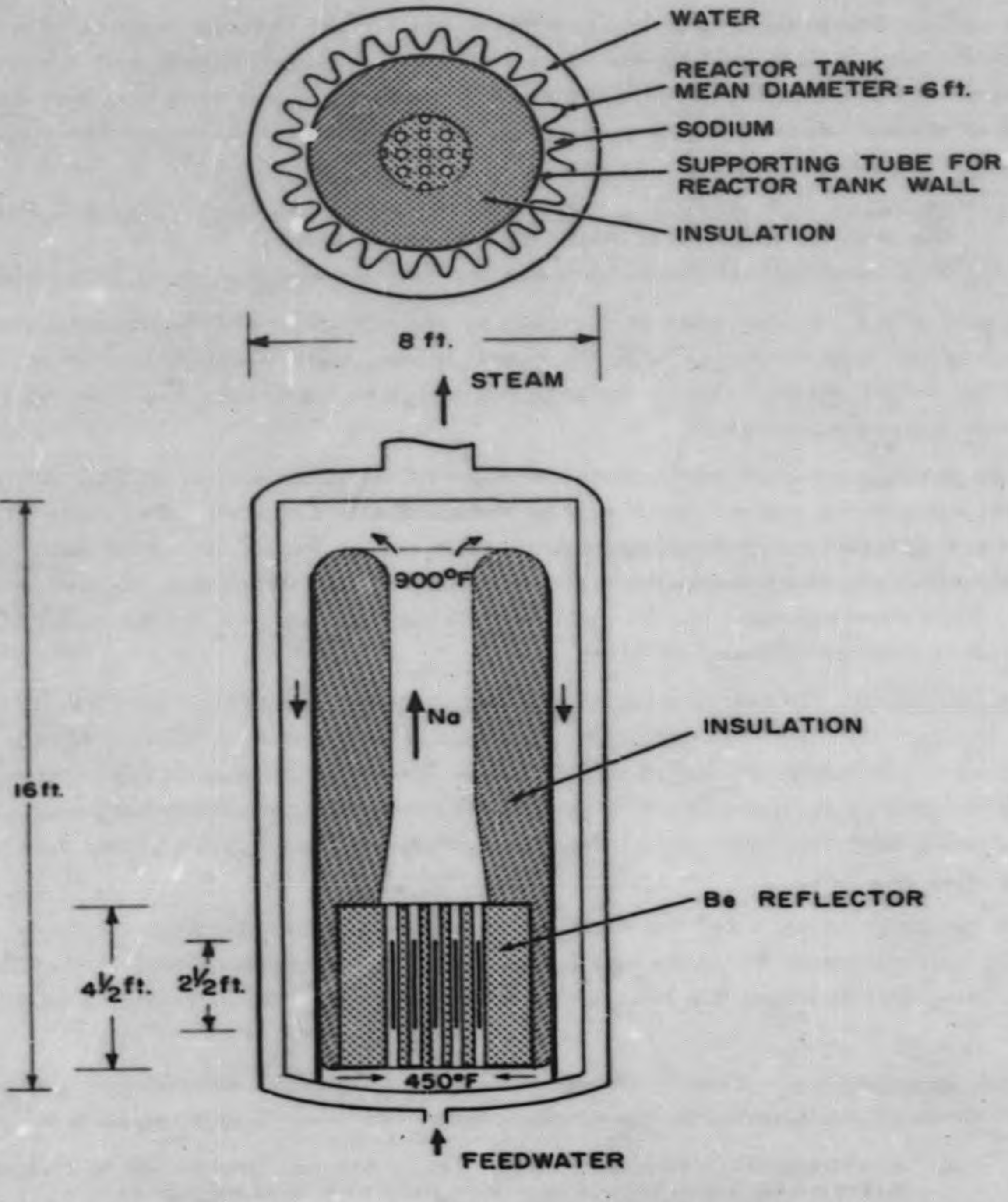

Fig. 29. Natural Convection Sodium Cooled Reactor - Case A

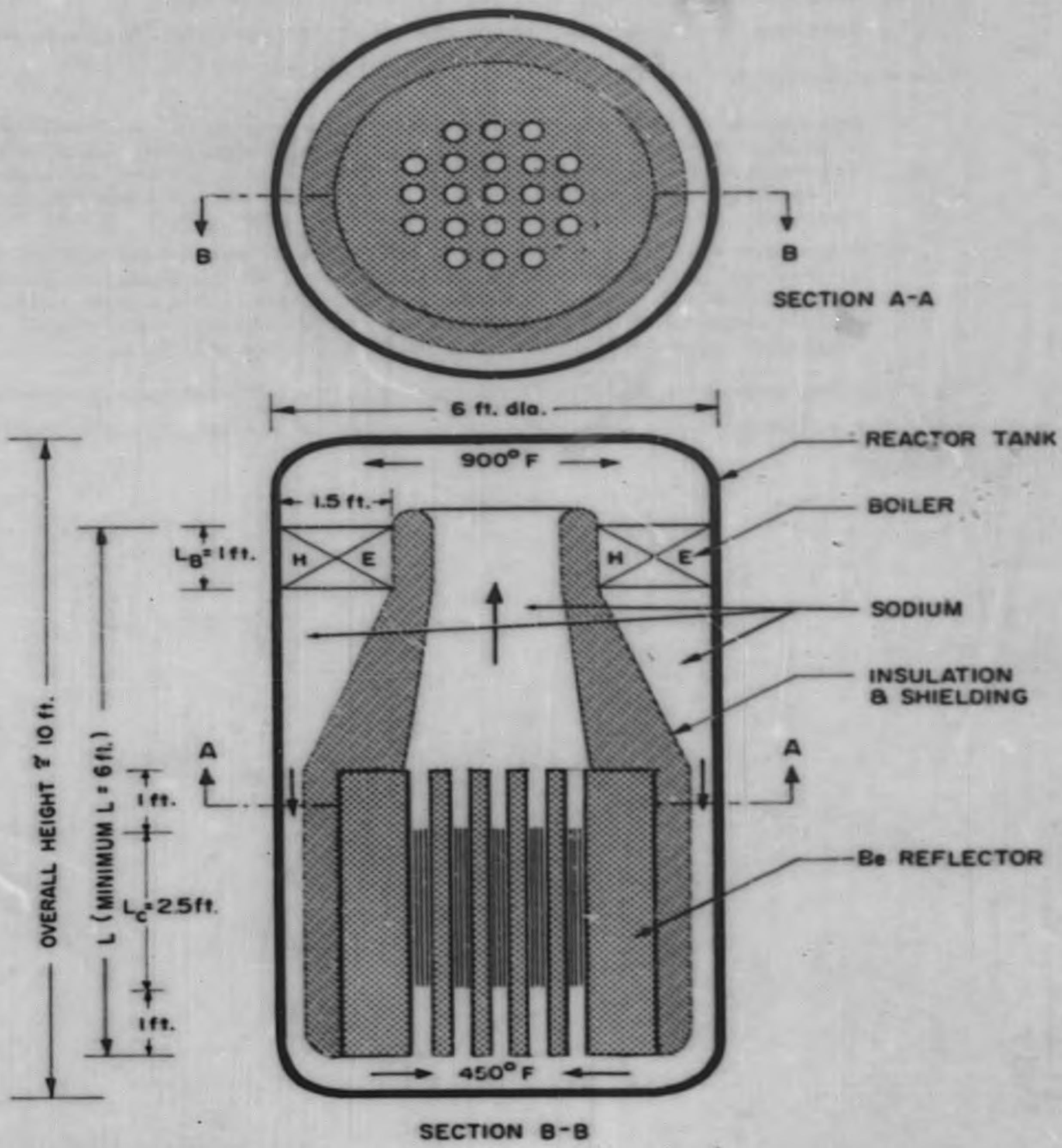

Fig. 30. Natural Convection Sodium Cooled Reactor - Case B

d. Sodium is contained in a hermetically sealed tank.

e. Increase in reliability since there are no coolant pumps to fail.

The disadvantages are:

a. Presence of water in the immediate proximity of radioactive sodium. If this is of real concern, an intermediate loop can be interposed between the water and the sodium, or the sodium can be replaced with some other liquid metal, such as lead bismuth eutectic, which does not react violently with water.

b. Since the water is not shielded from the reactor and will become activated, at least in Case A, the power conversion equipment will require shielding. If an intermediate loop is employed, shielding will be required only for the loop and the intermediate heat exchanger in addition to the shielding for the reactor tank.

The results of this preliminary thermal analysis of natural convection sodium cooled reactors indicated they are feasible with regard to the engineering considerations.

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