NAA-SR-1941

COPY

SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT

JANUARY – MARCH, 1957

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

Printed in USA. Price 40 cents. Available from the

Office of Technical Services
U. S. Department of Commerce
Washington 25, D. C.

NAA-SR-1941
REACTORS-POWER
54 PAGES

SODIUM GRAPHITE REACTOR
QUARTERLY PROGRESS REPORT
JANUARY - MARCH, 1957

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ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

CONTRACT: AT(04-3)-49
ISSUED: JULY 1, 1957

DISTRIBUTION

This report is distributed according to the category "Reactors-Power" as given in the "Distribution Lists for Nonclassified Reports," TID-4500 (13th Edition), January 15, 1957. A total of 595 copies of this report was printed.

This report is based upon studies conducted for the Atomic Energy Commission under Contract AT(04-3)-49.

THE TWO PREVIOUS QUARTERLY PROGRESS REPORTS ARE:

NAA-SR-1760	JULY-SEPTEMBER, 1956 (January 15, 1957)
NAA-SR-1875	OCTOBER-DECEMBER, 1956 (May 15, 1957)

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SUMMARY

Significant items of progress in the Sodium-Graphite Reactors program for this period are outlined as follows:

1. Criticality and nuclear parameter calculations have been initiated and expanded for edge-loaded hexagonal moderator can core configurations. It is found that a 0.3-inch radius fuel rod yields the smallest core radius for a given fuel to moderator ratio. (Section I).
2. Design studies on a 200-Mw (electrical) SGR plant were commenced. (Section I).
3. The SRE was loaded, without sodium in the core, to a point where inverse multiplication measurements indicated that one additional fuel element would achieve criticality. The extrapolated critical mass under these conditions was 22.2 fuel elements. (Section II).
4. Flux maps were made in the dry, subcritical, core. (Section II).
5. Sodium was loaded into the main secondary heat transfer system and circulated. (Section XIII).
6. The SRE hot cell was contaminated by a fire which occurred while examining an MTR-irradiated fuel specimen. Clean-up is proceeding. (Section III).
7. Eleven fuel specimens have been irradiated in MTR under thermal stress conditions. Six specimens have been examined to date. Tests tend to substantiate British data on metal swelling under hot irradiation. (Section IV).
8. An experimental hollow fuel element has been assembled for use in SRE experimental physics work during SRE criticality. (Section IV).
9. Thorium-5.4% Uranium experimental fuel rod fabrication was commenced. (Section V).
10. Uranium slug casting in thin-walled molds using pressure and static casting techniques has been satisfactorily demonstrated on an experimental basis. (Section VI).
11. Sufficient fuel for dry, subcritical SRE tests has been fabricated and delivered to the reactor. (Section IX).
12. A moderator can has been tested under reactor temperature gradient conditions. No serious deficiencies were observed. (Section X).

13. Preheating of the SRE heat transfer system prior to sodium loading indicated that recircuiting of existing heaters, and addition of heaters, was required. This was accomplished. (Section XI).
14. Mark II production control rods were obtained and tested satisfactorily. Prototype Mark II control rod tests have indicated areas in which packing and cleaning procedures should be improved. (Section XII). Mark II safety rod prototype tests indicate that this type is an improvement over the Mark I rod. (Section XII).
15. Preoperational tests of the SRE reactor service systems were carried out. Minor deficiencies were corrected in preparation for critical operation. (Section XIII).

SECTION A

TECHNOLOGY OF THE SODIUM GRAPHITE REACTOR

The material used in this Section was contributed by the following persons:

L. R. Blue
J. P. Carlino
J. M. Davis
C. Gerber
C. Guderjahn
W. T. Hayes
B. R. Hayward
R. B. Hinze
H. E. Johnson
G. Landler
J. E. Mahlmeister
J. J. McClure
W. T. Morgan
G. W. Rodeback
H. N. Royden
J. A. Stanley
J. Walter
L. Wilkinson

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I. FULL-SCALE SGR

A. ALTERNATE CORE CONFIGURATION STUDIES (J. E. Mahlmeister, J. J. McClure, G. Landler, C. Gerber)

1. Edge-Loaded Hexagonal Moderator Cans

The results of criticality calculations that were given previously[1] have been revised and are shown in Fig. 1 through 3. The curves have been expanded to include a larger range of lattice spacing. They also contain revised values of the thermal neutron flux in the large moderator cans.

Core surveys are included in Fig. 4. It may be noted that the material buckling is highest (and thus the critical radius is lowest) for 0.3-inch radius rods for the same fuel to moderator ratio.

An enrichment survey for a 37-rod fuel cluster is presented as Fig. 5.

2. 200 Mw (electrical) Study

A plan of action for the preliminary design of a 200-Mw (electrical) SGR plant has been devised and is currently being followed. This plan is to make seven studies for the first phase. These include:

a) Reactor characteristics
b) Reactor heat transfer and hydraulic requirements
c) Combined nuclear, heat transfer and hydraulic study
d) Steam cycle study
e) Analysis of 75 Mw SGR costs
f) Materials evaluation
g) Selection of the plant parameters.

3. Sodium System Cost Comparison

A study was made for a sodium graphite type reactor to compare the cost of the heat transfer system using all Type 304 stainless steel vs 2-1/4 Cr 1 Mo alloy steel throughout. The criteria are outlined in Table I.

The study indicated no appreciable difference in cost for the main coolant piping when using either 304 stainless steel or chrome-molybdenum material. This was because the saving due to the differences in the unit prices of the two

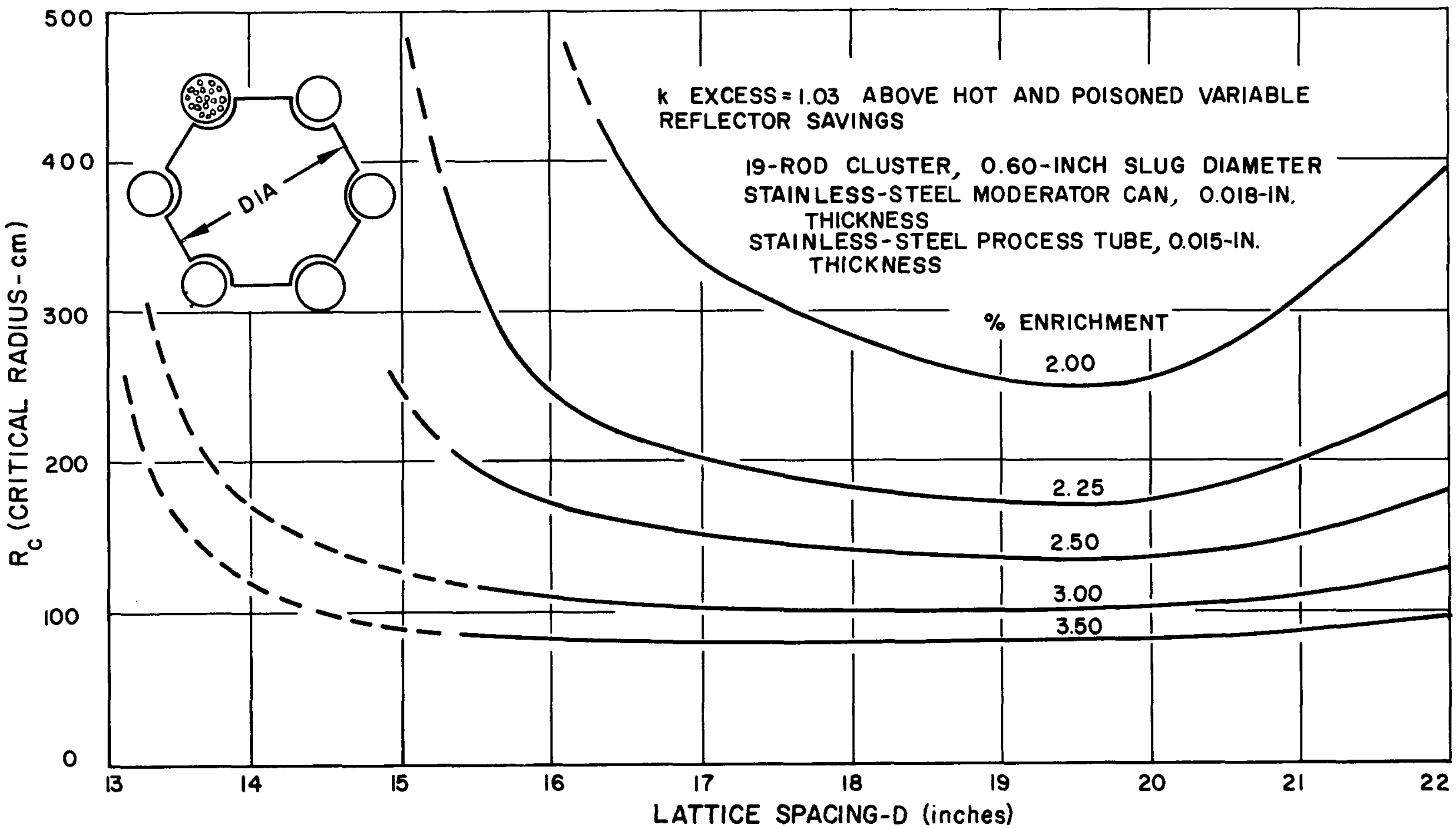

Fig. 1. Core Radius vs Lattice Spacing

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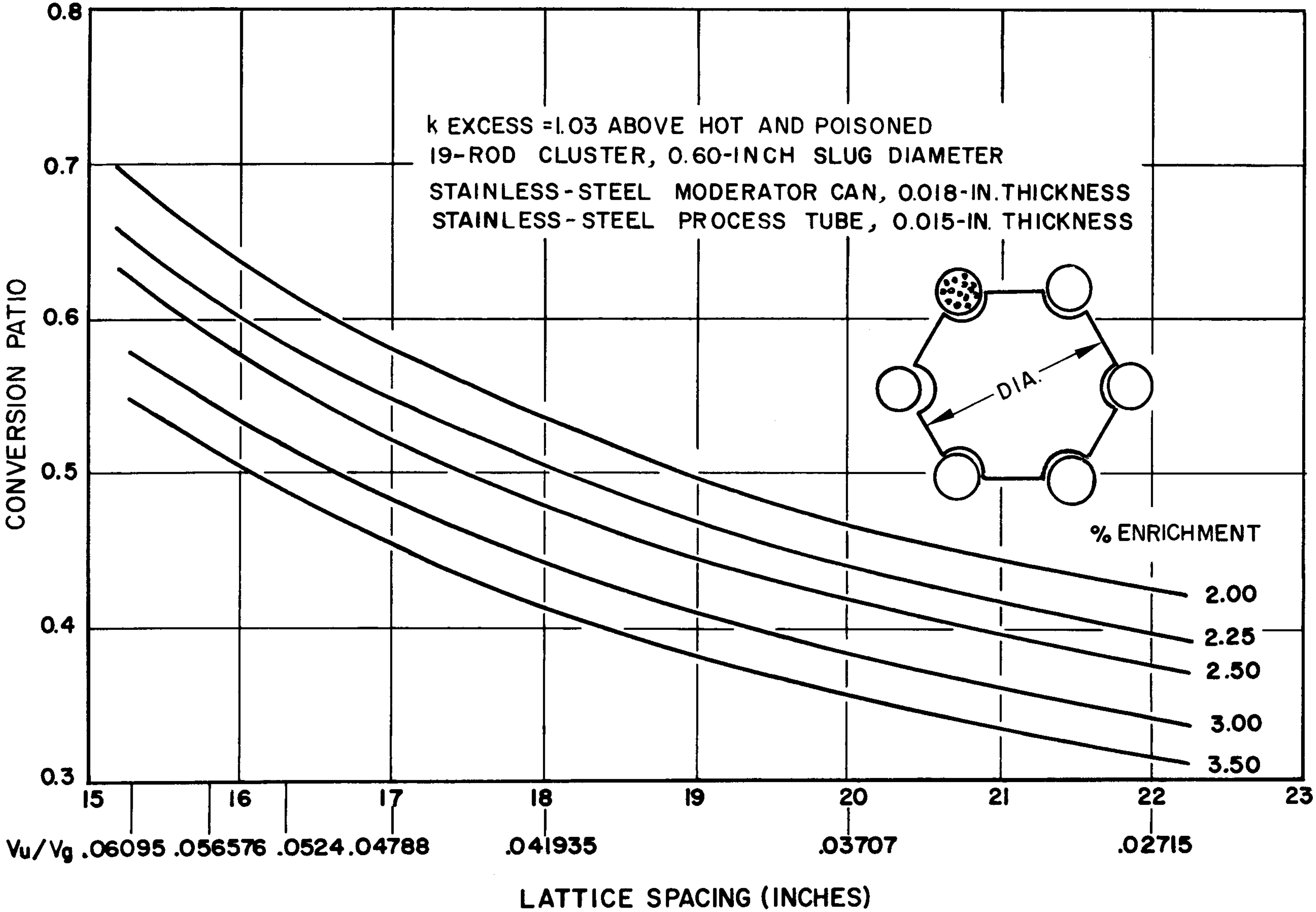

Fig. 2. Conversion Ratio vs Lattice Spacing

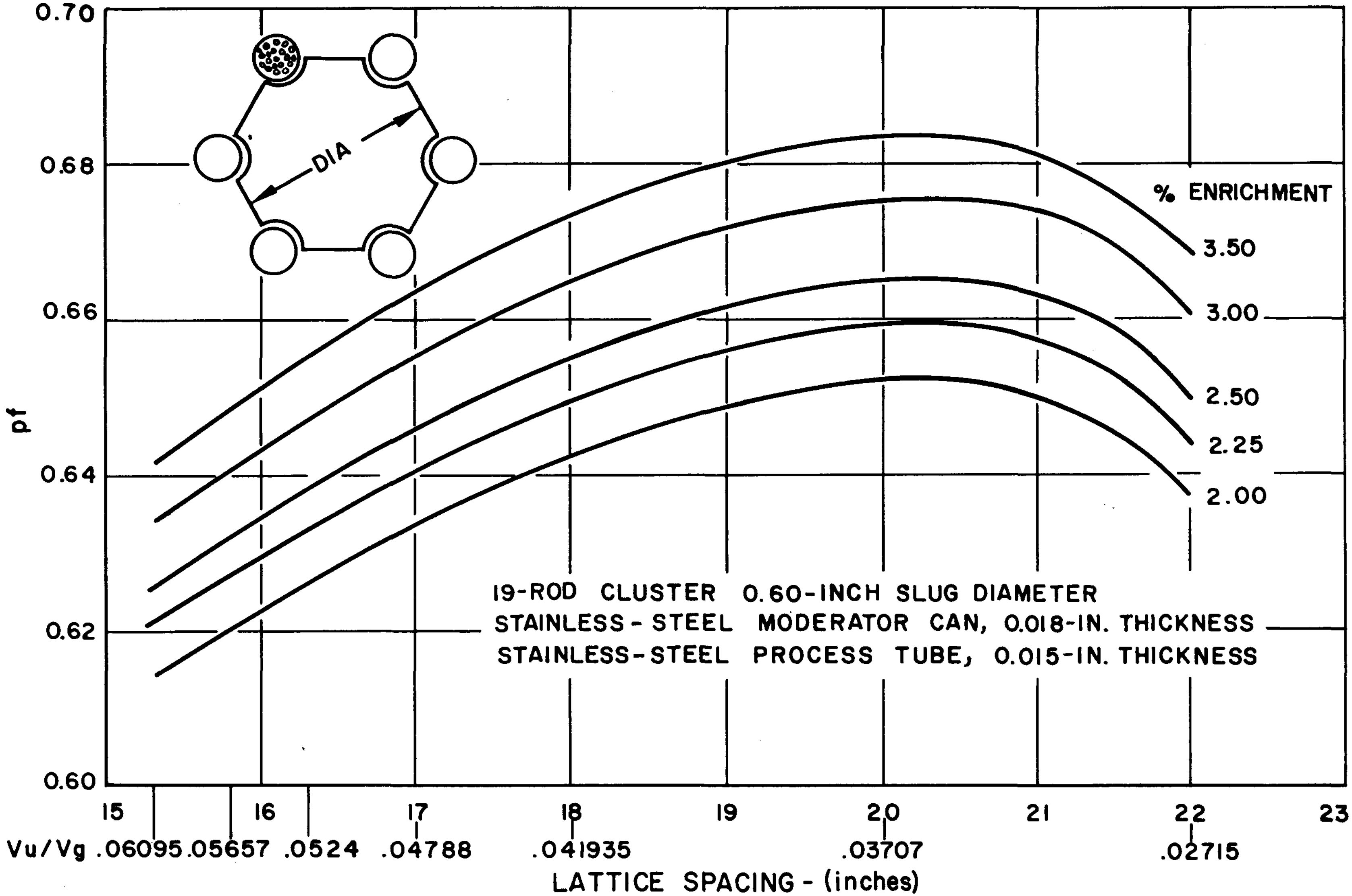

Fig. 3. p^f vs Lattice Spacing

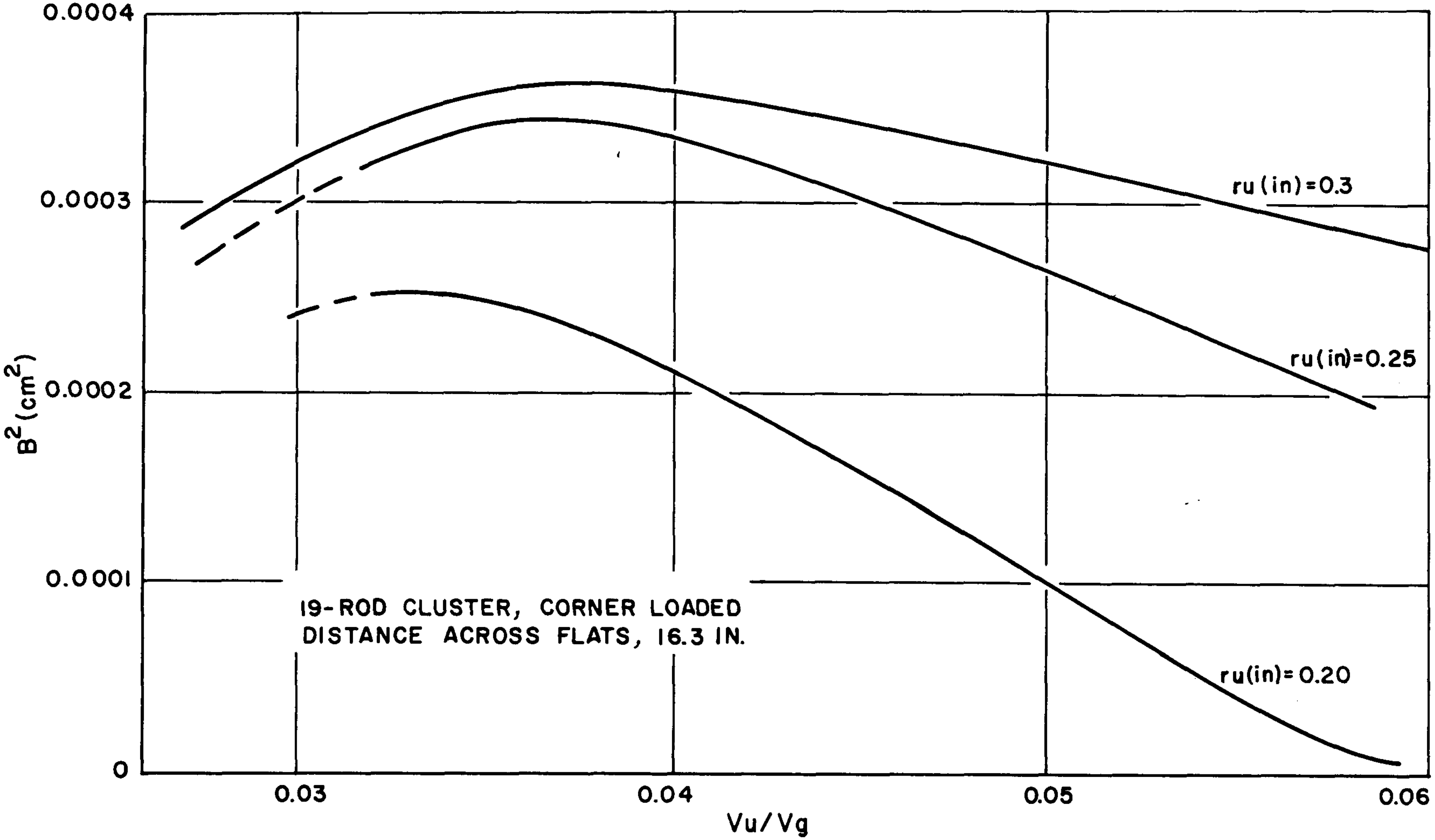

Fig. 4. Buckling vs Uranium Moderator Volume (Fraction for Variable Rod Diameter)

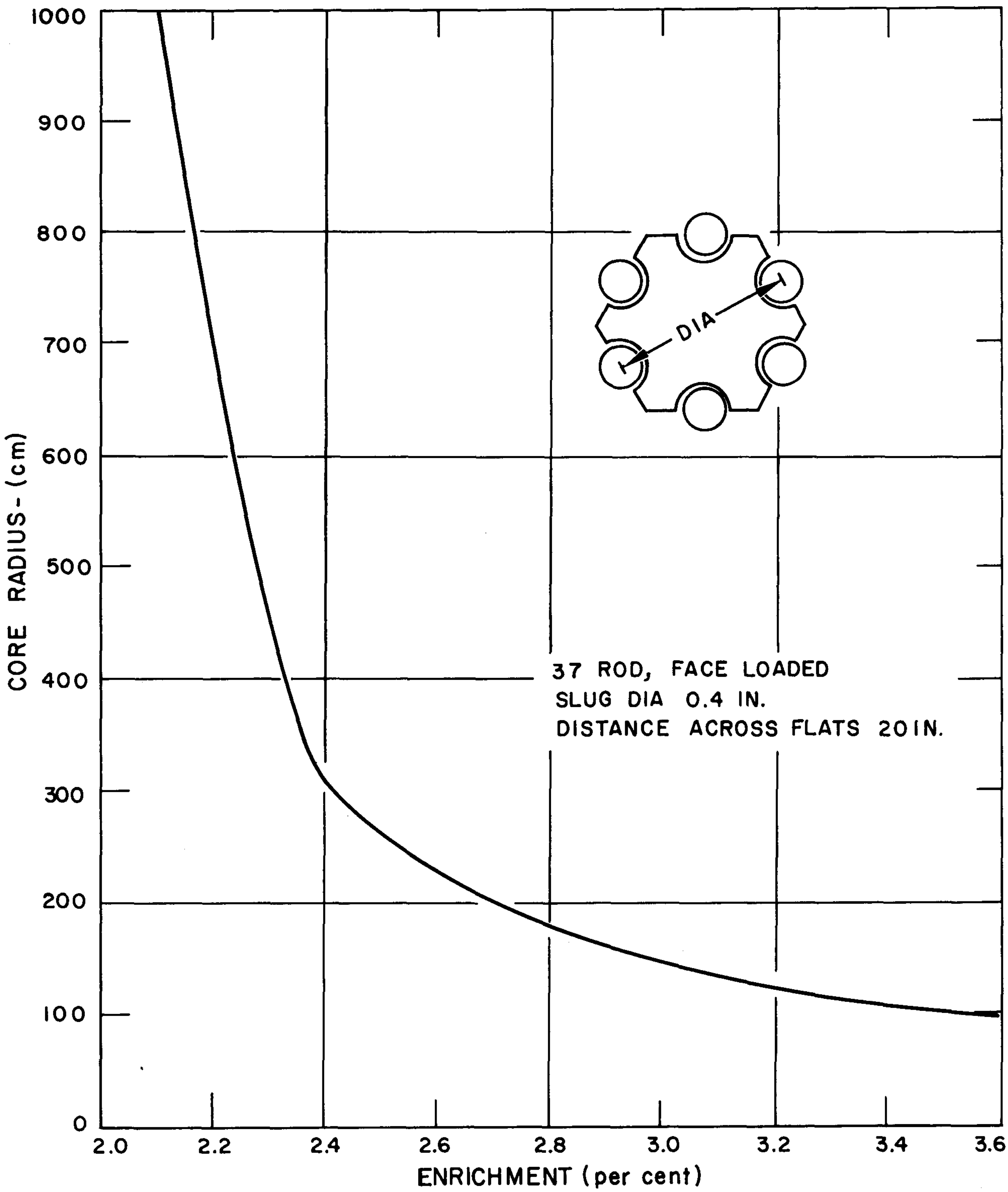

Fig. 5. Core Radius vs Enrichment

materials was offset by the increased pipe size and the cost of preheating and stress relieving required for the chrome-molybdenum alloy welding.

TABLE I

HEAT TRANSFER SYSTEMS
STAINLESS STEEL VS CHROME-MOLYBDENUM STEEL

	Stainless Steel	Chrome-Molybdenum Steel
Reactor Rating:	248 Mw	248 Mw
Reactor Temperature, Inlet:	500° F	525° F
Reactor Temperature, Outlet:	925° F	850° F
Piping Design Temperature:	950° F	900° F
Steam Conditions (throttle):	825° F/800 psig	770° F/760 psig
Main Piping Size:	12 inch	14 inch
Piping Arrangement:	Calculations indicated that there was about a 16 per cent stress reduction using chrome-molybdenum steel in place of stainless steel; therefore, the same piping arrangement was used for both systems.	

The chrome-molybdenum pump assemblies proved to be more expensive, since the motor cost increased due to the increased power requirement and the saving for material cost was offset by the increased pipe size.

No saving was indicated with chrome-molybdenum in the valve cost due to the increase in size.

Use of chrome-molybdenum alloy for the Intermediate Heat Exchanger indicated a saving over 304 stainless steel, as the smaller surface required due to a higher heat transfer rate and a lower material cost more than offset the additional cost of preheating and stress relieving chrome-molybdenum welding.

The steam generator study indicated a saving using chrome-molybdenum due both to lower material cost and the elimination of complications for providing galvanic corrosion protection where double wall tubes of dissimilar materials were attached to the tube sheets on the superheater in the 304 stainless-steel system.

The cost of the core tank was assumed to be the same for either system. Although no chrome-molybdenum material data was available for the core tank, the saving using chrome-molybdenum would about offset the additional fabrication and erection costs due to preheating, stress relieving and possible field difficulties.

A lower cost for preheating was indicated for the chrome-molybdenum system due mainly to the elimination of steel cladding for the application of the induction heating on the coolant piping.

A summary of the cost comparison is given in Table II.

TABLE II

COST COMPARISON

	12-inch Stainless Steel	14-inch Chrome-molybdenum Steel
Main Coolant Piping	$ 341,255	$ 349,665
Pumps (6)	341,200	380,000
Valves	208,625	208,625
Intermediate Heat Exchangers (3)	217,000	150,000
Steam Generators (3)	875,000	552,000
Core Tank	201,489	201,489
Preheating	71,814	30,557
Total	$ 2,256,383	$ 1,872,336

Total Saving - $ 384,047

B. PLANT CONTROL (W. T. Morgan, L. R. Blue)

The steady-state, variable-load, plant control study was coded for the IBM-704 computer. Early runs failed to show a correct energy balance, even at the design load. It was determined that this was due to selecting a poor value for the average specific heat of superheated steam. Since this specific heat varies greatly with temperature and pressure, a subroutine was inserted to analyze the superheater on a point-by-point basis. Correct values for the properties of steam are taken from the steam tables for each point thus eliminating the troublesome average specific heat. This change produced a satisfactory computer solution of the design load case.

Attempts to run cases at other loads or sodium flow ratios were unsuccessful, the machine being stopped by one of the iteration limits before a solution was reached. Correcting a minor coding error resulted in a satisfactory solution for the 80 per cent load, balanced sodium flow case. The logic system on the iteration loops is being examined to determine why the remaining cases do not run.

II. REACTOR PHYSICS

A. SRE START-UP (C. Guderjahn)

A total of 21 fuel elements were loaded in the SRE in order to determine the dry room temperature critical mass. During the loading, the flux was measured with fission counters placed in several channels in the reflector. The critical mass was obtained from an extrapolation of the plot of inverse counting rate vs number of fuel elements loaded. The average of the data from three symmetrically placed counters is plotted in Fig. 6. The extrapolation of this plot gives a critical mass of 22.2 fuel elements.

The temperature coefficient of the dry SRE was estimated from subcritical multiplication measurements. Gold foils were placed in the reflector both before and after heating the reactor from room temperature to 305° F. The foils were counted on a 2π flow counter and correction was made for source decay. The temperature coefficient was positive in this temperature range and equal to $4.4 \times 10^{-5}/°C$.

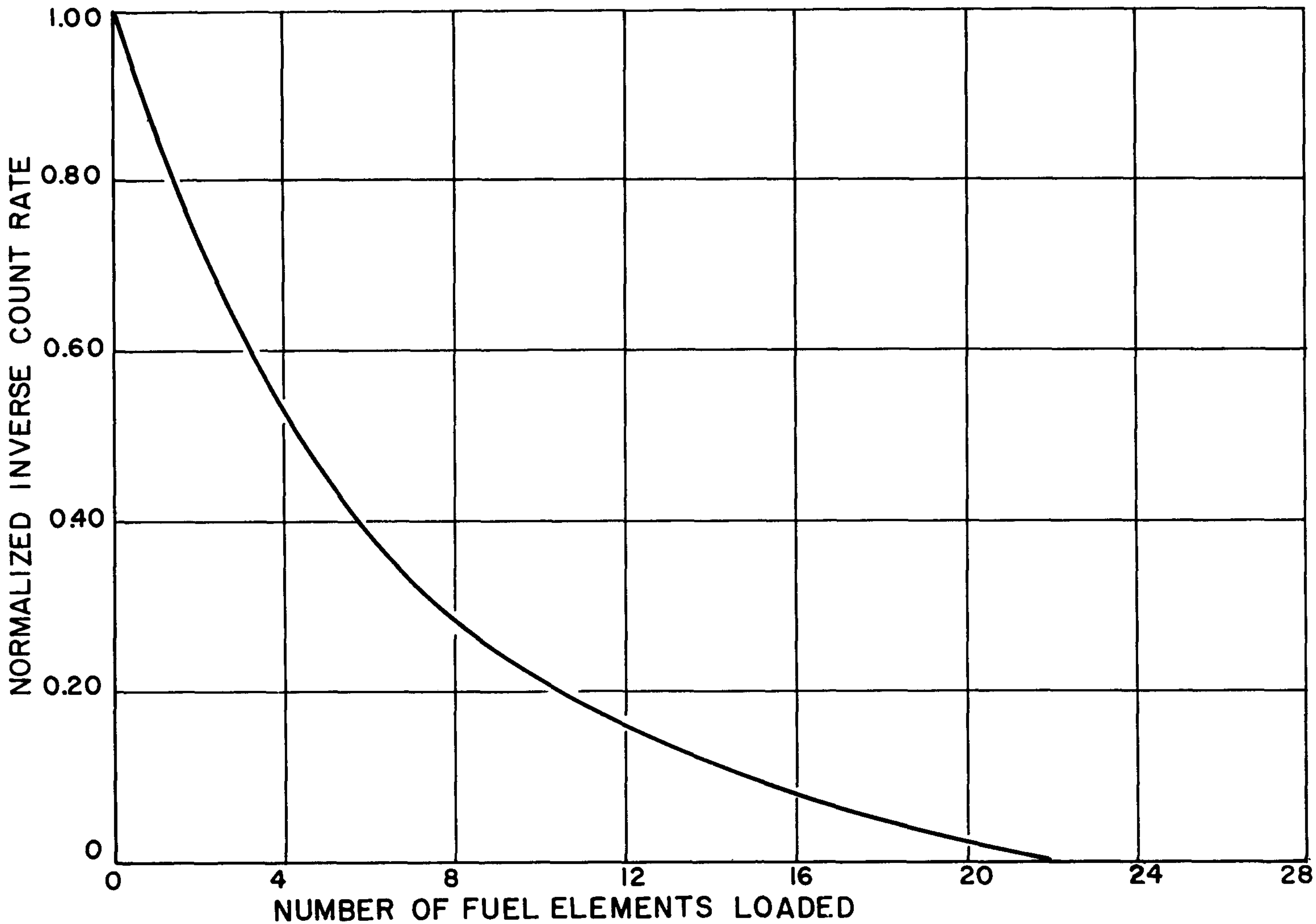

Fig. 6. Normalized Inverse Count Rate of 3 Symmetrically Placed Counters vs Number of Fuel Elements Loaded

The worth of several types of elements was also estimated from subcritical multiplication measurements. The following results were obtained:

20th fuel element loaded	+ 1.2%*
21st fuel element loaded	+ 1.2%
Dummy fuel element near center of core	+ 0.4%
Control rod	- 4.0%
Th-5% U^{235} fuel element in place of an uranium element	- 0.25%

A flux map of the dry subcritical SRE was made by exposing bare and cadmium covered gold foils in various channels of the core and reflector. The average radial and axial thermal flux is shown in Fig. 7 where it is compared with the theoretically predicted flux distribution.[2]

B. PARTIALLY INSERTED CONTROL ROD (F. L. Fillmore)

The IBM-704 code is now giving satisfactory results for the value of the buckling and flux distribution for the partially inserted control rod problem. A study of the flux distribution around the rod is under way.

A perturbation theory estimate of the revised SRE dry critical mass was made which took into account the difference between the actual SRE and the reactor assumed for the calculations given in another report.[2] The result was a critical mass of 17.1 fuel clusters, which still neglected the effect of neutron streaming in the voids. Recent calculations indicate that the effect of streaming requires an additional 1.7 fuel clusters for criticality. The resulting value of 18.8 clusters is 18 per cent below the experimental value of 22.2 clusters. In view of the rather large differences between the actual SRE and the reported reactor,[2] a two-group recalculation of the critical mass for the dry, wet and hot poisoned cases is in progress.

In order to investigate epithermal effects, a ten-group criticality calculation was done on a simplified SRE type reactor and the results compared with the two-group solution. The ten-group solution gave $k_{eff} = 0.99$ compared to the two-group solution of $k_{eff} = 1.00$.

*Change in reactivity caused by insertion of the element.

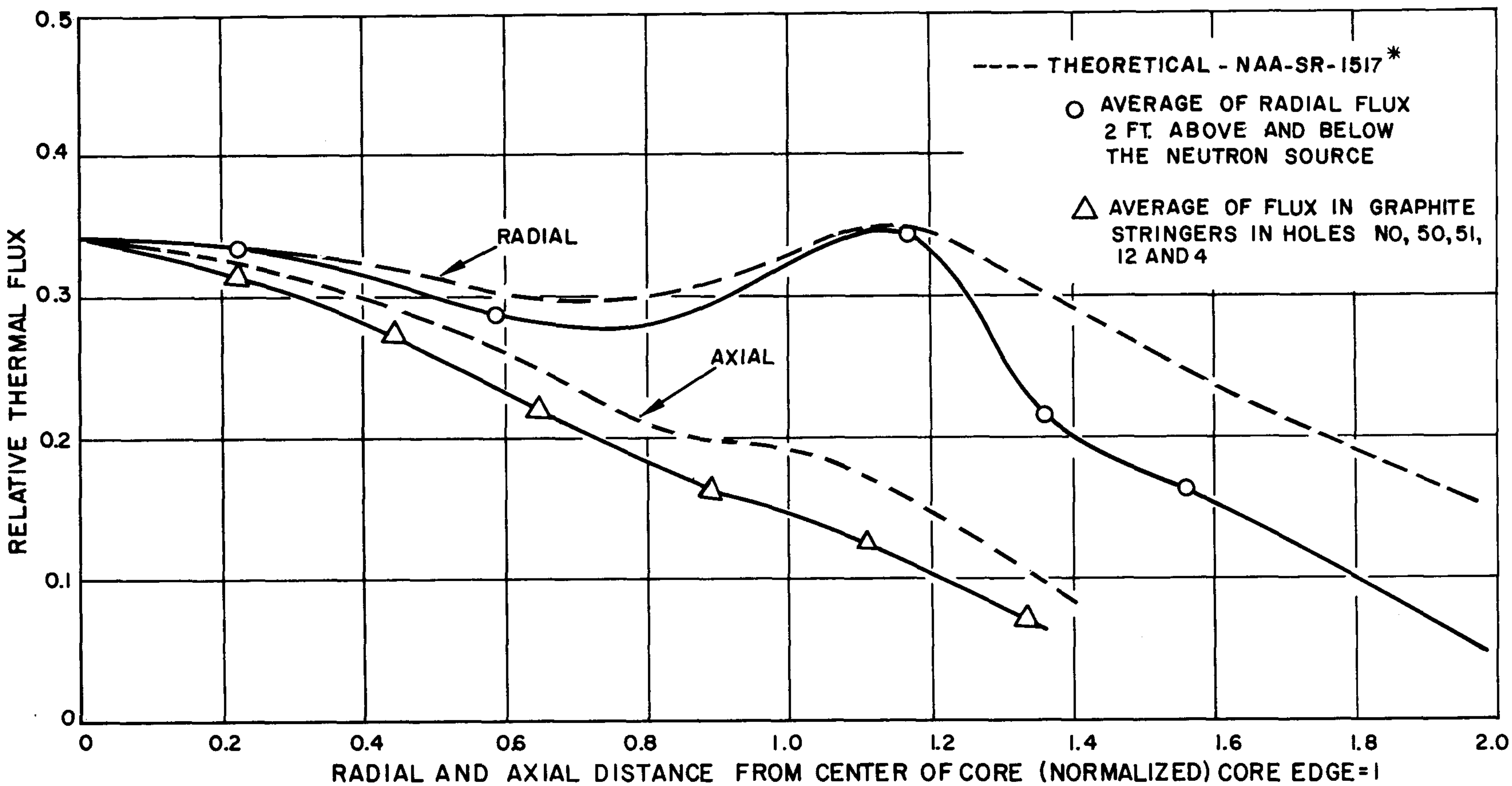

Fig. 7. Radial and Oxide Thermal Flux Plot of Dry SRE

*F. L. Fillmore, "Two-Group Calculation of the Critical Core Size of the SRE Reactor", NAA-SR-1517, July 1, 1956.

A two-group flux calculation which takes into account the axial temperature distribution in the reactor has been started. The result will be useful in evaluating reactivity changes and burnup.

C. DETERMINATION OF SRE POWER AND TEMPERATURE COEFFICIENTS OF REACTIVITY (H. N. Royden, W. T. Hayes, G. W. Rodeback)

Approximate solutions to the one-delayed-neutron-group reactor kinetic equations have been obtained in connection with analysis of flux-transient experiments to be performed on the SRE. The one-group kinetic equations can be expressed as a single second order differential equation with variable coefficients, for reactivity which varies with time. For reactivity varying sufficiently slowly, the "Liouville approximation" may be used to obtain fairly accurate solutions. The solution for a ramp input (reactivity linear with time) is given below:

Ramp reactivity function: $\rho = Mt$

$$n(t) = \frac{1}{2}\left(\frac{T_o}{T}\right)^{1/4} \left[(1 - A)\, e^{r_1} + (1 + A)\, e^{r_2}\right]$$

where $n(t)$ is the thermal-neutron density at time t, and $n(o) = 1$, $n(o) = 0$. The other quantities are:

$$T(t) = (\gamma + \lambda)^2 + 2m[1 - (\gamma - \lambda)t] + m^2t^2; \quad T_o \equiv T(0)$$

$$A = 2\,\frac{\gamma + \lambda - \dfrac{m(\gamma - \lambda)}{T_o}}{\dfrac{(\gamma - \lambda)^2}{\sqrt{T_o}} + \sqrt{T_o} - \dfrac{2(2\gamma\lambda + m)(\gamma - \lambda - \sqrt{T_o})}{T_o - (\gamma - \lambda)\sqrt{T_o}}}$$

$$r_{1,2} = \frac{1}{4}\, mt^2 - \frac{t}{4}\left[2(\gamma + \lambda) \pm \sqrt{T}\right] \mp \left(\frac{\gamma - \lambda}{4m}\right)\left(\sqrt{T_o} - \sqrt{T}\right)$$

$$\pm \frac{\gamma\lambda}{m} + \frac{1}{2}\, ln\left[\frac{\sqrt{T_o} - \gamma + \lambda}{\sqrt{T} + mt - \gamma + \lambda}\right]$$

where: $\gamma = \beta/l$, $m = M/l$, λ = average decay constant for delayed neutrons, l = prompt-neutron generation time, β = fraction of fission neutrons which is delayed.

Even for a reactivity rate of one dollar per second, which is hardly "slowly varying", the above solution agrees with an analogue-computer solution to within 5 per cent at prompt critical.

Solutions have also been obtained for reactivity varying as $(1 - e^{-bt})$. Efforts are being made to find solutions which may be less accurate but are also less cumbersome numerically.

III. DEVELOPMENT OF HOT CELL FACILITIES AND HANDLING TECHNIQUES

(J. M. Davis)

A. NAA-15-3 (High Temperature MTR Irradiation Experiment) EXAMINATION

This MTR-irradiated assembly of capsules containing specimens of various experimental uranium fuel materials was disassembled and the specimens examined in the SRE Primary Hot Cell. The operation was marred by two incidents which heavily contaminated the cell: (a) a fire (caused by a NaK plus butyl alcohol reaction) got out of control and not only spread contamination but damaged some equipment, and (b) one of the fuel specimens, which was badly distorted in irradiation broke into two pieces and generated a quantity of dust which spread over a wide area within the cell.

B. NEUTRON SOURCE

Following the completion of the NAA-15-3 examination, the neutron source was attached to its hanger-rod for insertion into the reactor, for use in the dry, sub-critical experiments. An alternate handling method was used due to the high cell contamination, but the operation was completed quickly and without incident.

C. CELL CLEANUP

Cell cleanup has proceeded slowly but without significant contamination of the operating area and without personnel dosages exceeding weekly tolerance. Because such massive contamination was not anticipated, the planned logistics of cleanup were not adequate for the situation. The problem has delayed cleanup, but the process is accelerating as procedures become better established and personnel learn to use them.

D. METALLOGRAPH AND MICRO-HARDNESS TESTER

The shielding cans for this installation have been received and the mockup has been constructed.

IV. METALLURGY OF SGR FUELS

(B. R. Hayward, J. Walter, L. Wilkinson)

A. FUEL IRRADIATION TESTS

Eleven fuel specimens have been irradiated in the MTR under thermal stress conditions and temperatures approximating SRE conditions. The specimens include both alloy fuels and standard unalloyed SRE fuel. Six of these specimens have been examined in the SRE hot cell. Of these six, only three could be completely removed from the capsules for measurement. The three removed were measured and photographed. (Fig. 8B-8C-8D). The photographs show a growth at the top end of the specimen and a fairly smooth surface. The end distortion is believed to be caused by the specimen extending above the NaK level of the capsules and becoming molten due to poor heat transfer. The diameter changes measured are listed in Table III together with the estimated total atom burnup.

Figure 8A shows the bright machined finish of the specimens before they were loaded into the NaK filled capsules for irradiation.

The examination of five additional specimens will be made in May. These include three U-1.2 w/o Mo specimens, one U-2 w/o Zr specimen and one unalloyed beta treated specimen. Normal irradiation temperatures for these has been between 1000° and 1200° F at the axis of the specimens.

A test assembly containing five Th-10 w/o U specimens has been shipped to the MTR for irradiation. The peak temperature expected in these specimens is 1400° F.

The results of this test indicate a serious problem exists in the irradiation of solid uranium metal dilute alloy fuels at elevated temperature to high burnup.

Fig. 8-A. Typical Specimen before Irradiation

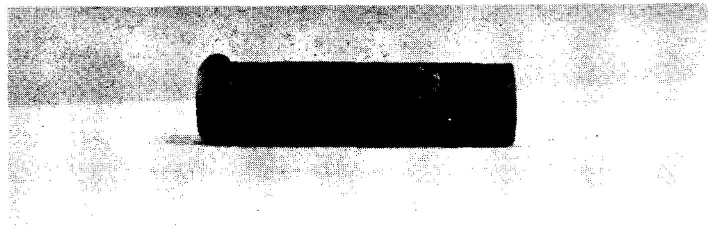

Fig. 8-B. Effect of Irradiation on a Cast 2 w/o Zr-U Alloy Specimen. Estimated total Atom Burnup, 0.25 per cent. Maximum Measured Irradiation Temperature, 640° F

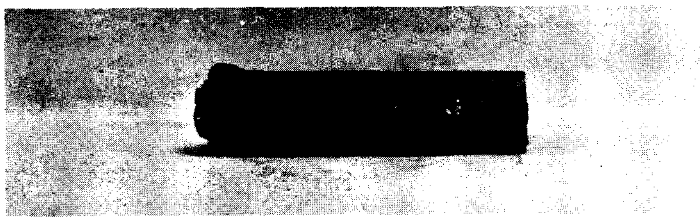

Fig. 8-C. Effect of Irradiation on a Powder Compacted 1.2 w/o Mo-U Alloy Specimen. Estimated total Atom Burnup, 0.3 per cent. Maximum Measured Irradiation Temperature, 790° F

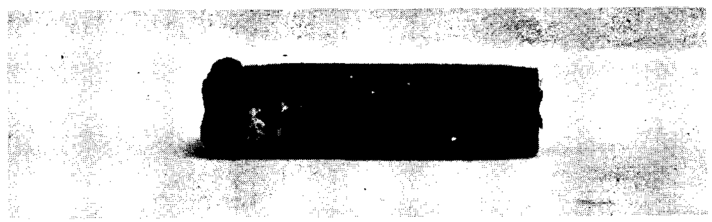

Fig. 8-D. Effect of Irradiation on a Powder Compacted 1.2 w/o Mo-U Alloy Specimen. Estimated total Atom Burnup, 0.4 per cent. Maximum Measured Irradiation Temperature, 1050° F

TABLE III

DIAMETER CHANGES AND ESTIMATED TOTAL ATOM BURNUP OF FUEL SPECIMENS

Figure	Specimen* Composition	Estimated Per Cent Burnup	Irradiation Temperature** Maximum	Irradiation Temperature** Normal	Diameter Increase Average	Diameter Increase Maximum
8B	U-2 w/o Zr	0.25	640° F	600° F	4%	4%
8C	U-1.2 w/o Mo	0.3	790° F	790° F	9%	10%
8D	U-1.2 w/o Mo	0.4	1050° F	970° F	12%	13%

*The original specimens were nominally 0.375 in. diameter by 1.500 in. long.

**The temperatures were measured along the axis of the specimens 0.500 in. above the bottom end of the specimen. Specimen temperatures were measured only during the early phases of the test.

The increase in diameter is greater than could be permitted in SRE fuel rods. Although these results are few, the data tends to substantiate British data on swelling.

B. LARGE HOLLOW SLUG FUEL ELEMENT

The large hollow fuel element is an experimental fuel element which represents a major design change from the SRE seven rod cluster. The element consists of a cylinder made up of hot pressed uranium cylinders clad internally and externally with thin wall 304 stainless steel tubing and NaK bonded. The element was designed with a 2.430 in. ID, 0.010 in. wall outer tube and a 1.300 in. OD by 0.015 in. wall inner tube.

A review was made of the stresses expected in the large hollow fuel element at a bond temperature of 1000° F. The calculated element internal pressure, due to NaK expansion and to the increase in temperature of the gas space above the fuel, is 44 psi gauge. Critical collapsing pressures for unsupported thin wall tubing with the design dimension of the element inner tube are estimated to be only slightly above 44 psi. It was also calculated that the 18 inch unsupported section of the outer tube would collapse at external pressures of between 5 and 10 psi. External pressures of this magnitude could be encountered in the fuel element cleaning cell.

Based on these calculations, the design wall thickness of the inner tube has been increased to 0.020 in. A short internal stainless steel sleeve will be used to prevent the unsupported section of the outer tube from collapsing.

Since a large hollow fuel element is needed for physics measurements during SRE criticality testing, an element of the old design plus the internal support sleeve has been assembled. After criticality testing, this element will be reassembled using the improved design.

The end cap welding problem in this fuel design which was previously reported[1] to be satisfactory, has developed additional complications. Microcracks in the fuel element tubing adjacent to the end cap welds have not been completely eliminated by varying the welding procedure. A more thorough study of this problem will be conducted before the fuel elements for reactor operation are assembled.

V. METALLURGY OF BREEDER FUELS

(J. A. Stanley, B. R. Hayward)

The extruded Th-5.4 w/o U fuel slugs for irradiation in the SRE have been analyzed for uranium content and density. The analytical results for uranium content confirm the results from the manufacturer in that the material is within the specirication of 5.4 ± 0.1 w/o uranium. However, there is no agreement with the manufacturer as to which ingots resulted in slugs on the high and low sides of the specifications. To try to resolve the extent of any variation in uranium content from slug to slug and ingot to ingot, danger coefficient tests will be made on representative slugs in the water boiler neutron source equipment. Samples from the front, center and rear of each extrusion rod had uniform densities of 11.90 ± 0.02 gr/cc with the exception of samples from a rod of high carbon content (1500 ppm carbon). Samples from this rod had a density of 11.84 gr/cc.

Slugs for 21 SRE fuel rods are in the process of being jacketed. These rods will include two temperature monitored fuel rods. The three fuel elements assembled from the Th-U alloy rods will be used for nuclear measurements and long time irradiations.

Several feet of the enriched Th-5.4 w/o U alloy rod are being shipped to Hanford. A cooperative program on the effects of irradiation on the properties of these reactor fuels will be made at Hanford.

VI. ADVANCED FUEL MATERIALS

(J. A. Stanley, B. R. Hayward)

According to recent AEC progress reports, the only method so far which appears to be successful for producing fuel materials in thin-walled molds has been the centrifugal technique. However, our technical personnel have been exploring other multiple casting techniques, namely, pressure and static, which not only produce acceptable fuel materials but offer a simplicity of design and operation (see Fig. 9 and 10). Preliminary work on several slugs from a single casting operation into a thin-walled graphite (1/16 in. thick) or Vycor mold (1/16 in. thick) using pressure and static techniques has shown good metallurgical results. These data have resulted in preliminary tests for multiple casting of fuel materials.

The procedure for the pressure casting of 0.250 in. to 0.750 in. diameter slugs into thin-walled unsupported Vycor tubes is as follows:

1) Use a massive charge of clean metal for melting
2) A 1375° C pouring temperature
3) Properly coated Mg_2ZrO_4 Vycor molds
4) A preheated mold temperature so controlled that a 300° F temperature is maintained on top portion of molds
5) A pressure of at least 25 psi of helium released within 30 seconds into casting chamber
6) A residence of at least 3 minutes for Vycor molds in molten bath after pressurization prior to mold ejection for solidification.

The above factors are very important to obtain repeat results of fuel slug sizes ranging from 1/4 in. to 5/8 in. diameter.

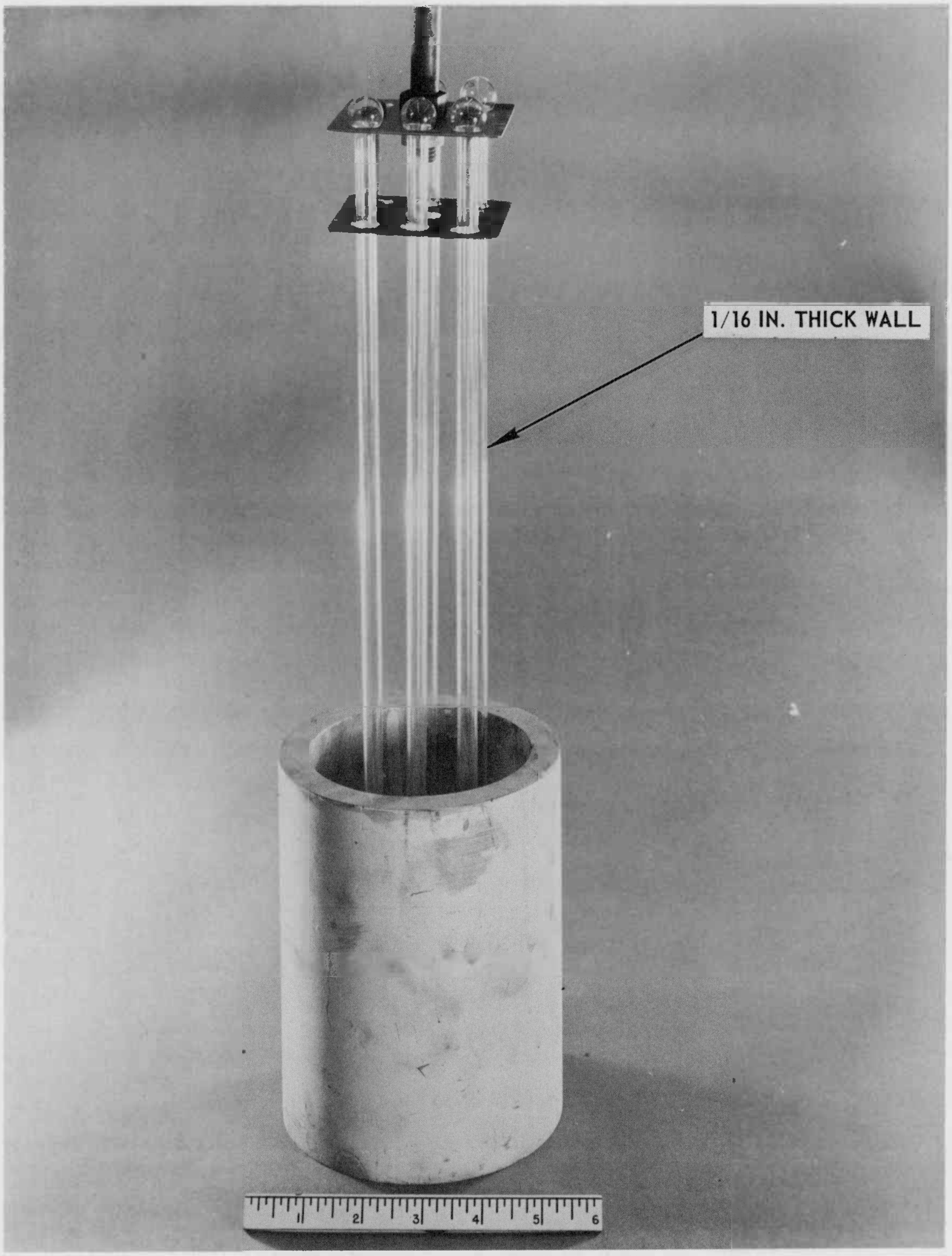

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Fig. 9. Vycor Cast Setup

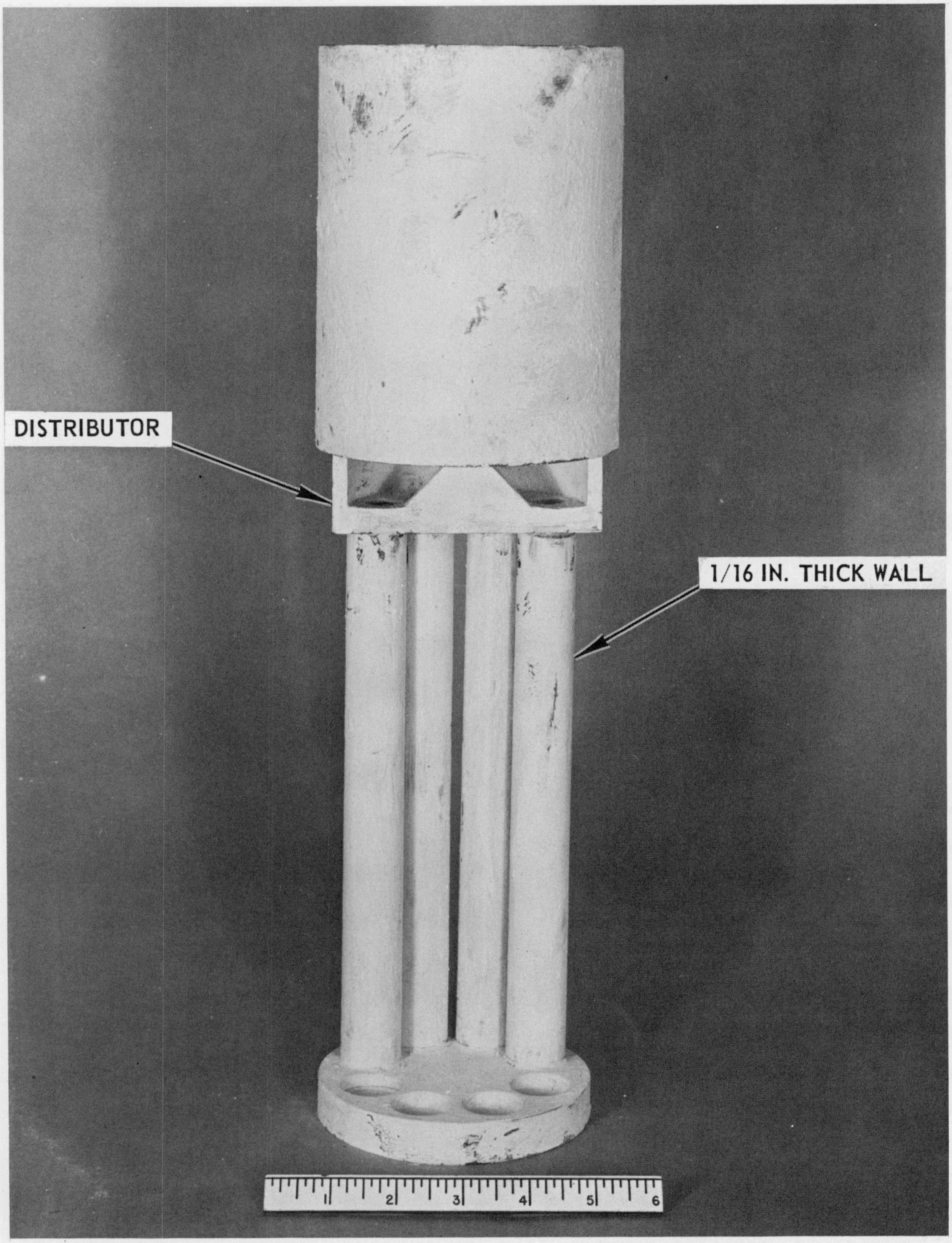

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Fig. 10. Static Casting and Melting Assembly

Initial metallurgical data on previously reported casting tests have shown unalloyed uranium slugs with:

1) Uniform, coarse grained, crystal structures
2) Heavy oxide inclusions distributed preferentially at top end of slugs
3) An overall casting density of 18.96 g/cm^3 per 9-inch long casting
4) No appreciable warp
5) Very little surface porosity
6) A shrinkage of 2 to 3 per cent
7) An as-cast surface roughness of better than 125 rms
8) A remarkable metal soundness (supported by gamma-graph results)
9) A metal finished casting yield of approximately 85 per cent.

The static "gang" casting method for normal uranium metal has resulted in several castings of eight, 3/4-inch diameter by 9-inch long, and sixteen, 3/8-inch diameter by 9-inch long slugs. The procedure for melting prior to casting is similar to the above technique. However, a special type distributor has been designed (Fig. 10) to permit proper molten metal flow into graphite molds.

Metallurgical data have shown the above static cast slugs to have:

1) Coarse-grained heavily twinned crystal structures
2) Light oxide inclusions along the entire slug length
3) 18.76 g/cm^3 density per 8 in. specimen
4) A badly warped casting
5) A heavy peripheral distribution of porosity
6) A metal shrinkage of 2 to 3 per cent from the mold diameter.

It appears that the pressure approach offers more promise than static method with respect to warp, surface roughness, and metal soundness. However, both systems offer a rapid means for variable studies, i.e., mold materials. These preliminary results indicate that the pressure casting method produces good quality slugs at 5/8 in. diameter and smaller (Fig. 11). For fuel slugs greater than 5/8 in. diameter, the static and centrifugal methods appear more adaptable (Fig. 12).

The third method now considered is the centrifugal casting technique for both uranium and thorium fuel materials. A unit consisting of a 40 in.-diameter rotor table, a 30-in. diameter mold and a graphite distributor system has been designed.

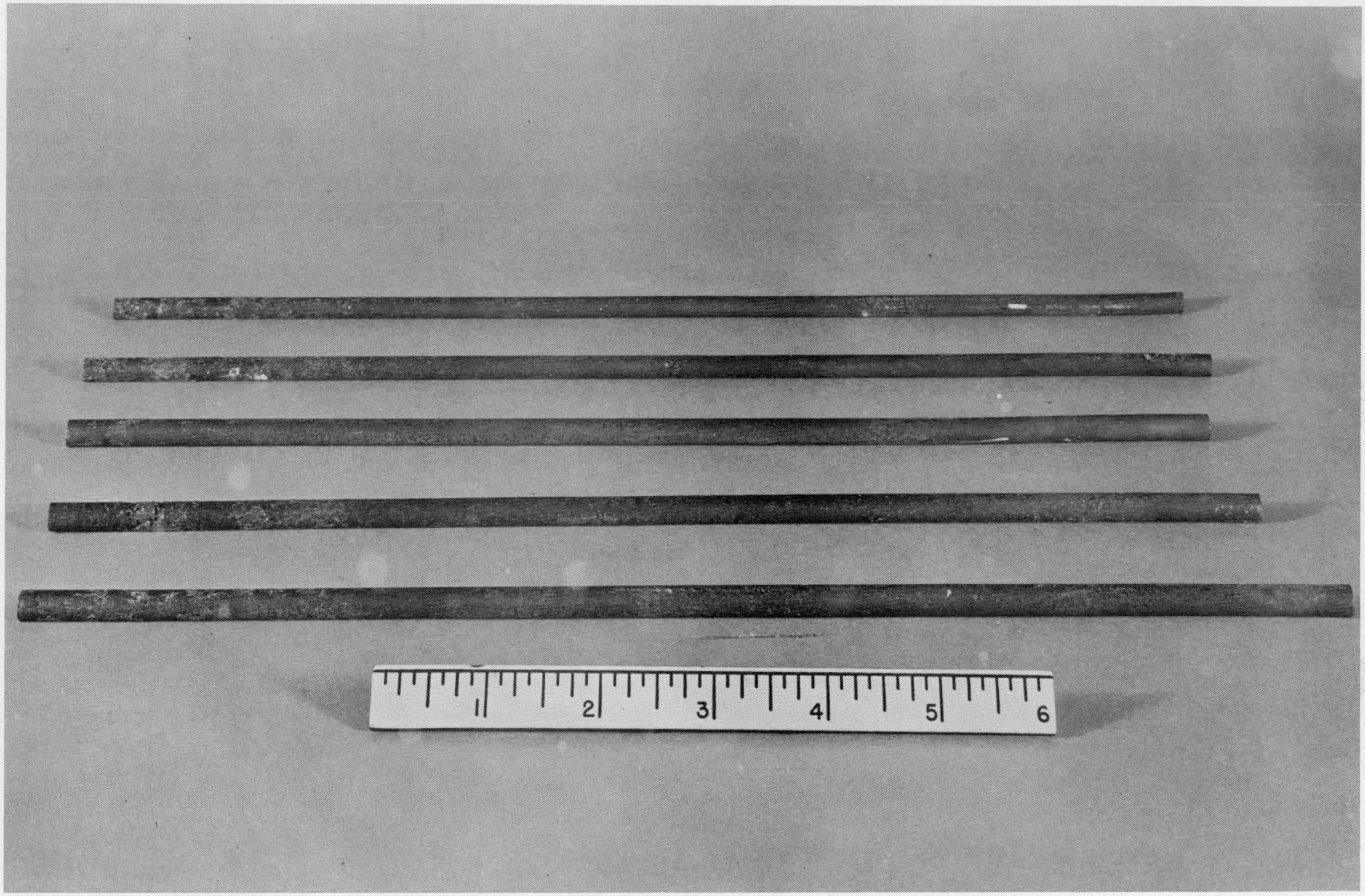

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Fig. 11. Pressure Cast 1/4-inch Diameter by 12-inch Long Uranium Rods

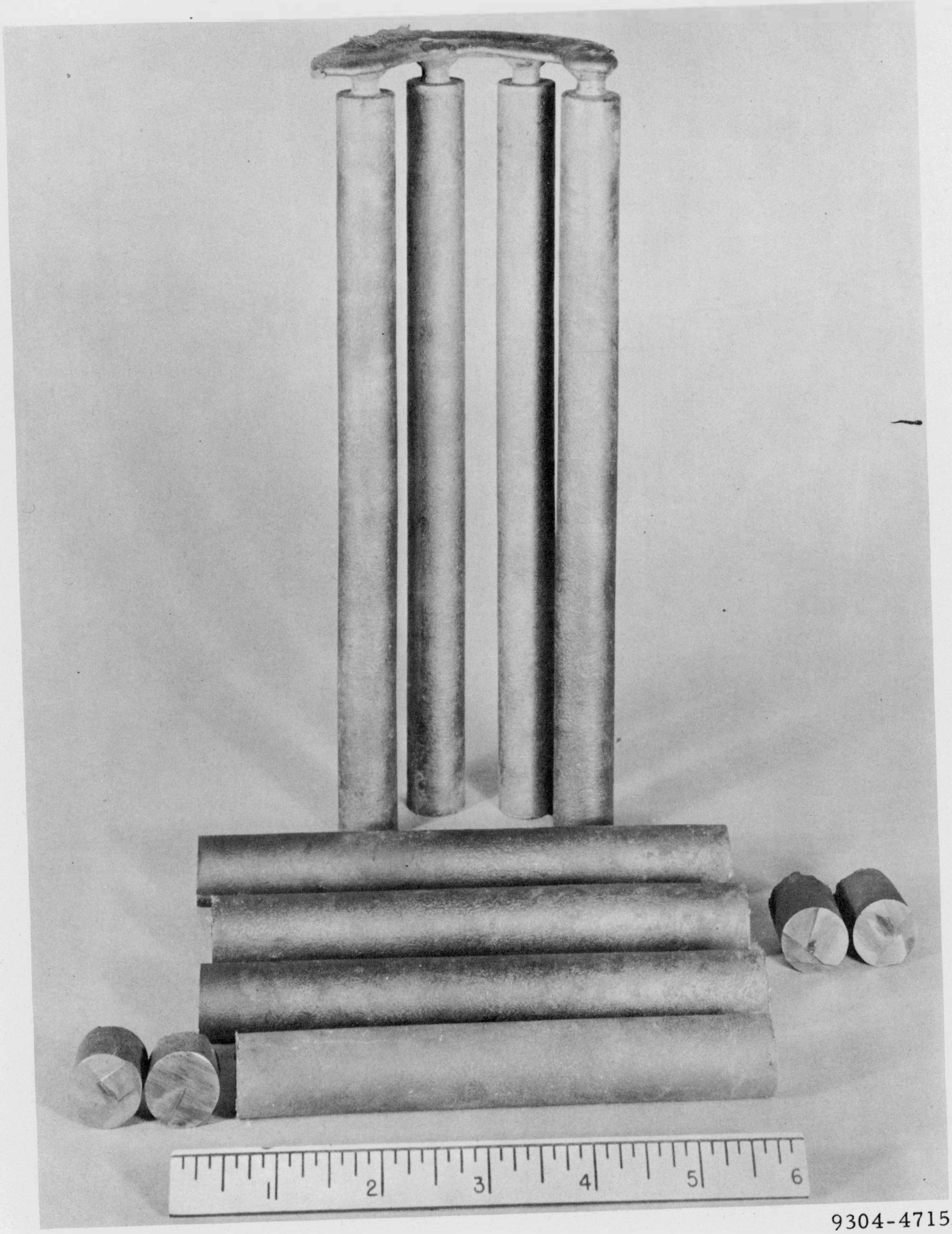

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Fig. 12. Static Cast 3/4-inch Diameter by 9-inch Long Uranium Bars

The construction of the machine and subsequent components is now nearing completion.

Several thorium-uranium static slug castings have been made using a three chamber mold arrangement with moderate success. However, further examination of the product slugs have shown some possibly discouraging aspects of cast slugs. Some of the more discouraging factors are metal brittleness and carbon pick-up, reactivity of thorium with graphite (during melting operations) and thoriums tremendous wetting of graphite which restricts the molten flow during casting. These results are very limited due to insufficient power capacity for adequate melting, superheating, and distributor and mold pre-heating. A larger power supply (100 kw) has been ordered.

VII. CORROSION IN SODIUM

(R. B. Hinze, H. E. Johnson, J. P. Carlino)

Beryllium is used in the SRE core as part of the source element and in two temperature monitoring corner channel elements. In both these applications, the beryllium metal is in contact with primary sodium. While behavior of beryllium in sodium has been reported in the literature, the oxygen concentration of the sodium in the experiments has been above SRE oxygen levels. To survey the effect of sodium of low oxygen concentration, four specimens of beryllium (0.011-in. sheet approximately 2 in. by 0.5 in.) were exposed. The following weight changes are shown in Table IV.

TABLE IV

WEIGHT CHANGES OF SAMPLES EXPOSED IN COLD TRAP AND HOT TRAP

Sample	Exposure Time	Weight Change
Exposed at 950° F in static chamber with cold trap at 290° F		
Be 1	260 hours	gain 0.11 mg/cm^2
Be 2	500 hours	gain 0.09 mg/cm^2
Be 3	1024 hours	loss 0.12 mg/cm^2
Exposed at 1000° F in dynamic loop with hot trap at 1200° F		
Be 4	507 hours	gain 1.7 mg/cm^2

Samples 1 and 2 exposed in cold-trapped sodium (nominal oxygen concentration 10 ppm) exhibited weight gains after short exposures, and were slightly tarnished. Sample 3, after more than 1000 hours in cold trapped sodium, had a loosely adhering white film, identified by X-ray diffraction as BeO. This sample showed a weight loss.

Sample 4, exposed to hot-trapped sodium (very low oxygen concentration) gained appreciably. The sample surface had an oxide film, but in this case the film was adherent. From a comparison of the appearance of Sample 4 with Sample 3 one would expect that longer exposure in hot-trapped sodium would result in a loose film, with consequent weight loss.

SECTION B

SODIUM REACTOR EXPERIMENT

The material used in this Section was contributed by the following persons:

R. W. Atz
J. J. Droher
S. Elchyshyn
R. B. Hall
R. Hoff
D. H. Johnson
J. A. Leppard
A. E. Miller
E. C. Phillips
H. E. Richter
H. Schroeder
J. F. Stolz

VIII. LAND, UTILITIES AND BUILDINGS

(J. F. Stolz)

A. HOT CELL VENTILATION

In order that the existing hot cell ventilation system can be used during the initial experiments, a prefilter has been installed in the hot cell. This prefilter will minimize contamination of the ducts and fan.

Portable air locks were built for use around the hot cell doors when they are open. Fans will then move 1000 cfm of air at all times. This removes the former requirement of two speed fan operation. All manipulator parts are closed as much as possible. In-cell filters are used; they are 60 per cent efficient.

The existing ventilators pull the hot cell service area atmosphere into the operating area. This condition has been corrected by installing an exhaust fan and filter out of the service area.

B. GALLERY STEEL SHOT TROUGHS (Below Shield Block Joints)

The troughs in the auxiliary gallery were changed to plates to allow more room for the finned cooling tubes. Additional troughs were placed in the main gallery for the additional capacity required at joints between blocks. At these points steel shot can flow from behind blocks towards the joint.

C. EM PUMP VAULT SURFACE DRAINAGE

The sleeve around the manhole in the EM pump vault was raised. The area between the primary fill tank vault and the reactor building was paved. The paving carries surface water away from the EM pump vault, rather than allow the water to soak into the earth and seep into the vault.

D. GALLERY SHIELD BLOCK ANCHORAGES

In order to provide a minimum factor of safety of 6 in the design of the gallery shield block hoist anchorages, the following steps were taken: (1) Four holes were drilled in each of 9 blocks to receive redesigned anchorages. (2) Redesigned anchorages were furnished and installed. (3) A hoisting rig with an equalizing bar design was furnished.

IX. FUEL ELEMENTS

(J. J. Droher, S. Elchyshyn)

A. FUEL ELEMENT PRODUCTION

Further refinements during tests in the wrapping of fuel rods with spacer wire allowed a reduction in the point of closest approach of wires on adjacent rods to 3/8 inch. Using this spacing, all of the outer rods for SRE fuel elements have been wrapped with wire. One fuel element containing two thermocoupled fuel rods has been held aside and will be wrapped with equally spaced distances between wires on adjacent rods.

The wire wrapping jig (Fig. 13) consists of a brass tube in which a helical groove has been milled completely through the wall. An indexing device at the top locates and clamps the wire relative to the slot in the top end of a fuel rod. After welding the wire to the top end cap, uniform tension is applied to the wire by means of dead weights. The other end of the wire is cut to the required length and is welded to the bottom end cap. The wire wrap corresponds to the thread of a screw, and the milled groove to the internal thread of a nut. The jig rotates while it is being raised by a hoist, thus permitting the jig to be removed.

Two jigs, one with a left hand helix and the other with a right hand helix, are required. This permits alternate left and right hand threads of the wire wrap on adjacent rods.

Cyclograph tests of all spare rods were performed. This completes such tests on all rods loaded to date. Satisfactory bonding was observed on all standard rods obviating the necessity for any heat treatment.

Twenty-eight standard fuel elements were assembled and delivered to the SRE for dry sub-critical tests. The last of the six thermocoupled rods requested for use as spares was fabricated and tested.

The current series of test welds made on end caps for Hollow Fuel Elements was completed. The amount of NaK bond required was established. A 15 inch long tube was placed above the slugs to support the outer tube in case of collapse. The thermocouples were omitted and this element will be used for preliminary nuclear tests pending further revisions. Loading of this element was completed.

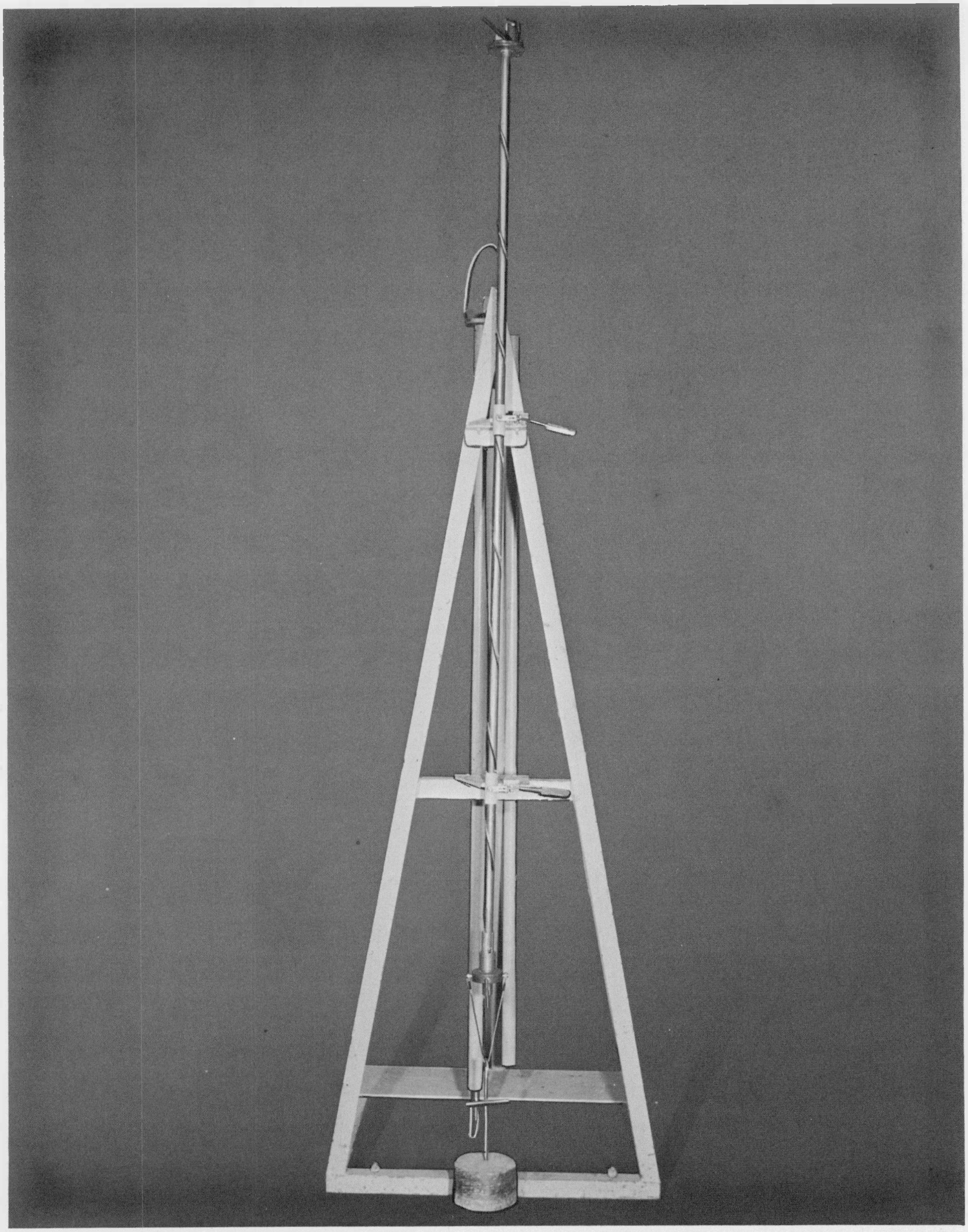

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Fig. 13. Wire Wrapping Jig

X. MODERATOR CAN FABRICATION AND TESTING

(J. A. Leppard)

An additional moderator can has been soaked in the vacuum furnace for 96 hours at reactor gradient, i.e., 960° F at top to 500° F at bottom, and with 2.5 psi differential pressure across the can. This can was one that was damaged during manufacture. The shell was slightly bulged at several points from the damage but during the test these panels became flat and have remained so after cooling. The heads of the can were also flattened during the test although dimensional changes were less than 1/16 inch.

Preparations are being made to flood this can with sodium in order to experimentally determine the effect of sodium filling a cracked can.

XI. HEAT TRANSFER

A. SIX-INCH PUMP LOOP (R. W. Atz)

Tanks and supporting structures were assembled and installed. The control system has been rebuilt to accommodate future installation of automatic control equipment. Assembly of the loop is continuing. Target date for completion is expected early next quarter.

B. TWO-INCH PUMP LOOP (H. Schroeder)

The control system for continuous unattended operation of the 2-inch pump loop has been built and installed. This equipment contains high-low sodium temperature controls, freeze seal temperature alarms, coolant flow alarms and sodium level alarms. An automatic dump valve is included in the system. In the event of coolant flow failure to the freeze seals, loss of compressed air for the pressure transmitters or excessive sodium temperature, the loop will be automatically drained and the heaters shut off.

A plugging meter is being built for installation in the system. This instrument will permit continuous monitoring of the sodium oxide concentration in the loop. A diffusion cold trap has been installed to collect the oxide. A special test section of schedule 40, 304 stainless steel pipe is being prepared for installation in the loop as a part of the alternate materials research program.

C. TETRALIN SYSTEM EMERGENCY GASOLINE ENGINE (H. E. Richter)

The present emergency gasoline engine is designed as a "last resort" method of supplying power to the tetralin pump. It will be operated in the event of a power failure and failure of the emergency power. At a time when the above failures occur, the reactor will be in a scram condition. All operators will be occupied with reactor and sodium flow controls and will not be readily available for manual starting of the engine. Since the existing air cooled engine does not lend itself to automatic starting and control, an automatic starting four cylinder, water cooled engine has been selected for replacement. This engine will increase the reliability of the tetralin coolant system.

D. AIRBLAST HEAT EXCHANGER (H. E. Richter)

During preheating of the heat exchanger, difficulty was encountered in reaching the design temperature of 250° F. This was caused by the large air leakage area inherent to the louver design. A temporary solution to the problem consisted of an insulating blanket on the top row of tubes, and sheet metal closures for the fan opening. Design of a permanent method of reducing air leakage is in progress.

E. HEATERS (R. B. Hall)

1. Tank Heaters

Following a study of existing conditions, heaters were added to several tanks in the system. Because of a change in type of insulation and method of installation, heat losses were considerably greater than originally calculated. Existing heaters were recircuited, and heaters added to the transfer, condensate, flush and drain, primary fill and secondary fill tanks. Heating capacity is now satisfactory.

2. Pipe Heaters

A major recircuiting of heaters was accomplished, including the addition of a number of 120 volt panels to provide individual circuiting and operation of many of the heaters. The present design will allow for control of the pipe heating in any of 15 different sequences of operation.

3. Valve Heaters

All previously unheated 2-inch and smaller valves have been individually heated by wrapping them with tubular heaters.

XII. INSTRUMENTATION AND CONTROL

A. MARK II PROTOTYPE CONTROL ROD (A. E. Miller)

The Mark II prototype has been operated for an additional 17 days under a temperature gradient corresponding to full power SRE operation. This rod has been tested for a total of 27 days "hot" and 8 days "cold" (room temperature). Subsequent examination of the rod and support column has shown both to be in good condition.

The wear rate of the four support rods which connect the screw to the poison column has decreased. Spectrographic analysis of the wear products revealed them to be the components of austenitic stainless steel. An examination of the first of the four production Mark II rods after its functional test revealed no particulate matter anywhere within the thimble. This indicates that the wear as well as the observed boron in the prototype may be attributed to boron carbide spilled in the guide tubes during packing of the prototype shield assembly (see Fig. 14). The boron carbide serves as part of the radiation shield in this shield assembly. Exceptional care was taken to insure that the corresponding tubes in the production rods were properly cleaned prior to final assembly.

B. MARK II PRODUCTION CONTROL ROD (A. E. Miller)

Four production Mark II control rods and support columns were fabricated, cleaned, lubricated and assembled. These rods were functionally tested prior to delivery to the SRE. The functional test consisted of cycling the rod over a 2-foot stroke at regulating rod speeds for 4 hours at room temperature and from 4 to 6 hours at temperatures corresponding to full power operation of the SRE. Tests were satisfactory.

C. PRODUCTION DRIVES (A. E. Miller)

Three single-speed and 3 two-speed drives are required for the SRE. One of each type is a reactor spare.

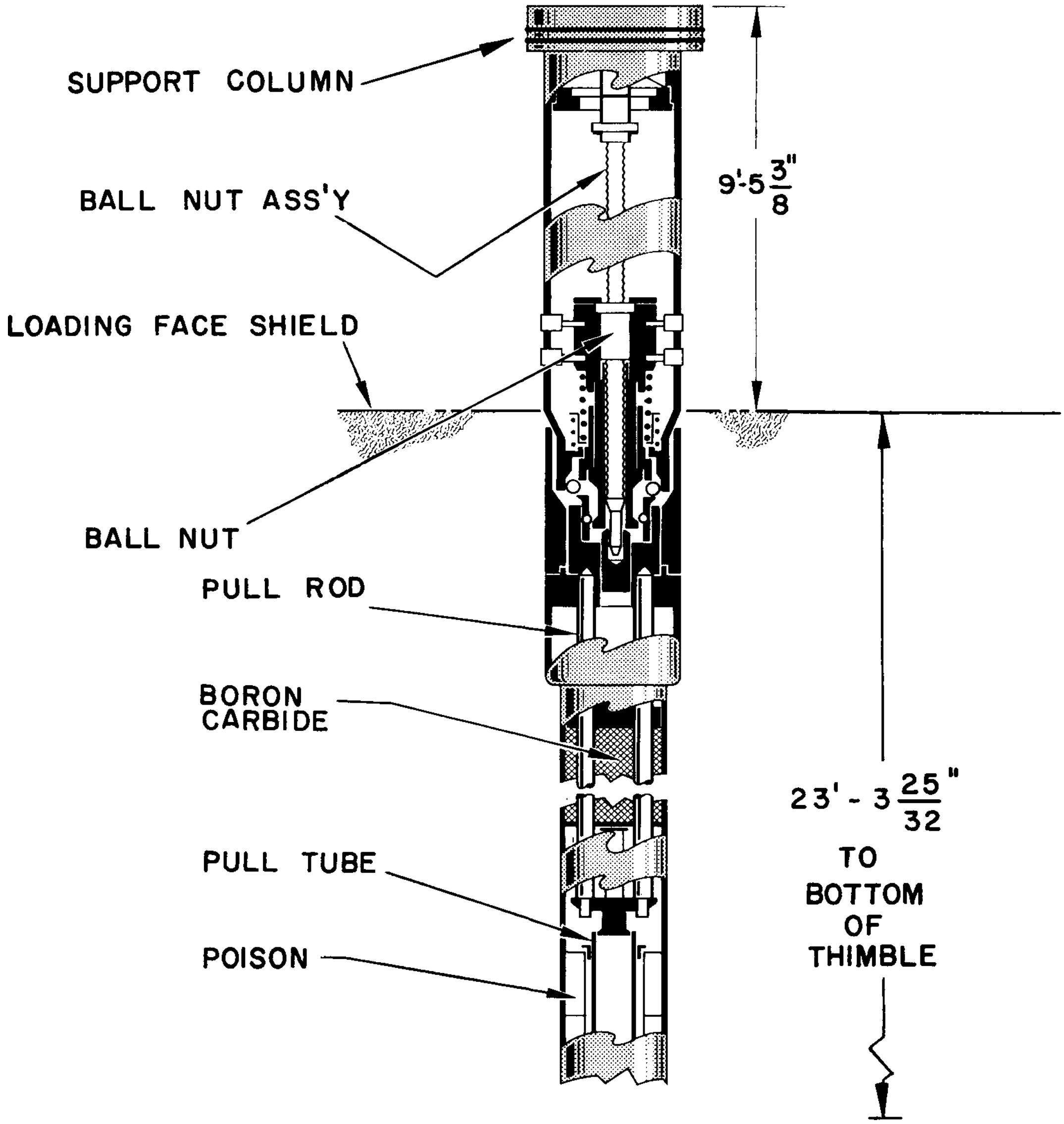

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Fig. 14. Mark II SRE Control Rod

Three single-speed drives were fabricated. One drive was inoperable because of misalignment between the worm and worm gear. The gear box for this unit is being re-made. Minor modifications were made in the other two drives to prevent oil leakage prior to a successful functional test. The test consisted of operating the drives continuously for 24 hours.

None of the two-speed drives were acceptable because a large proportion of the component parts were not fabricated within design tolerance. One unit was assembled from the best components (of the three assemblies) and delivered to the SRE. Two new units are being fabricated.

D. MARK I SAFETY ROD PROTOTYPE (E. C. Phillips)

The stainless steel components of the safety rod prototype which operate at elevated temperatures were carburized and chromed to reduce galling and friction.

An extended test of the safety rod prototype was attempted at 800° F. This test was terminated by the failure of the drive shaft after 364 cycles of dropping and retrieving the poison element. During the test, it was necessary to adjust the prototype to compensate for wear of the silver bushing in the latch plate.

Upon removal of the prototype from the test furnace, the thin walled section of the thimble was found to have fallen off. This was due to extensive damage to the transition piece between the heavy walled and the thin walled section of the thimble (Fig. 15). A shoulder in the transition piece receives the impact of the snubber cylinder when the poison element is dropped.

Prior to failure, this thimble has been at 1100° F for 200 hours, and at 600° F or more for 102 hours. A total of 589 drops of the poison element had been made in this thimble including 161 at 1100° F, 364 at 800° F, and 21 at 600° F.

The stainless steel components of the safety rod prototype which had been carburized and chromed showed no wear.

Because of the failure of the thimble in the above test and because of the addition of a weld joint in the Mark I production safety rod thimbles, which was not included in the prototype thimble, a test of 500 drops of the poison element at 600° F was made with the internal mechanism of the Mark I safety rod prototype in a Mark I production thimble. After the test, the production thimble

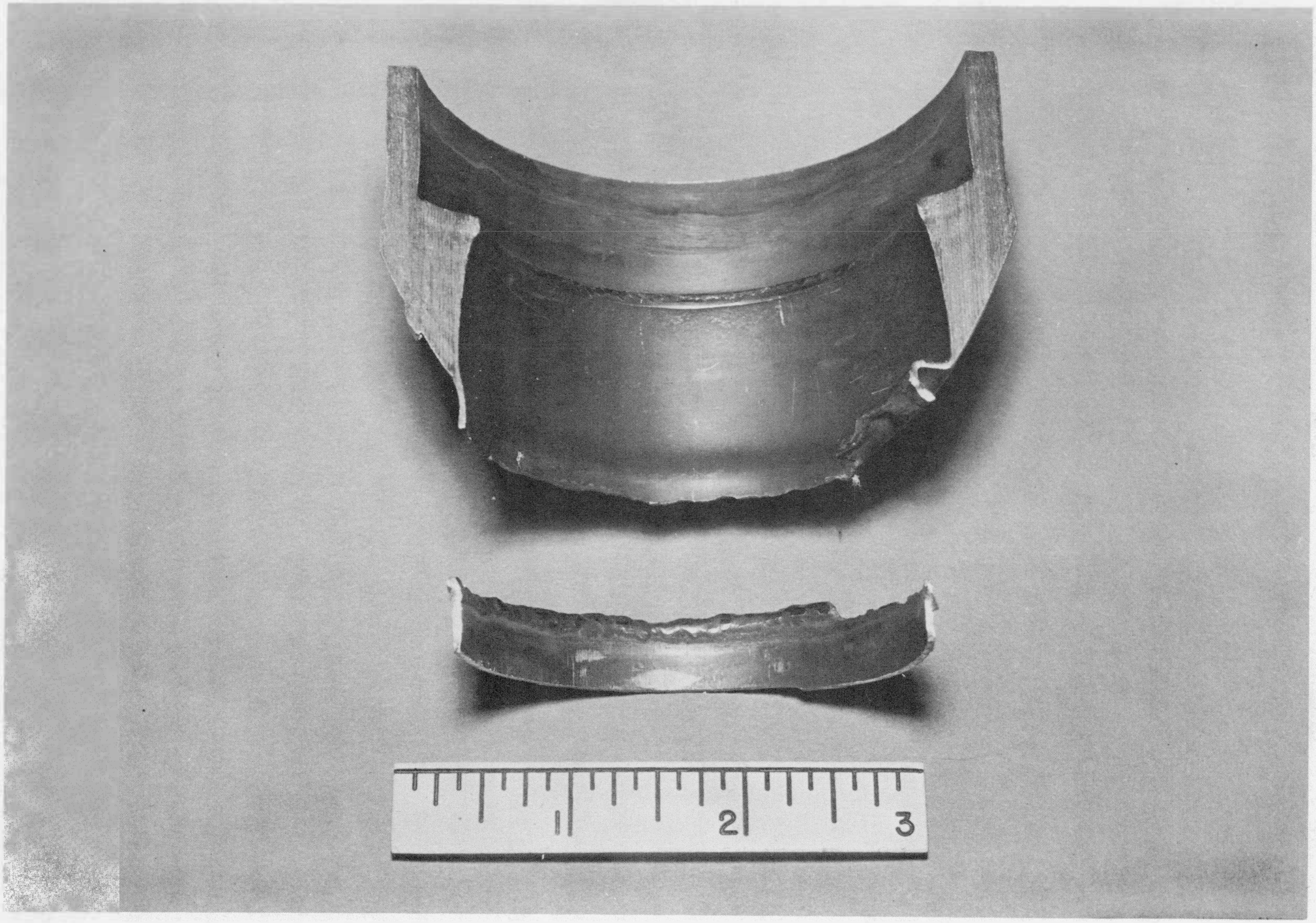

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Fig. 15. Damaged Thimble (Mark I Safety Rod Prototype)

showed evidence of damage, but the damage was not as extensive as that which occurred to the prototype thimble.

A latch plate, flame plated with aluminum oxide, was used in this test. This plate showed no wear after 500 drops. The linkage between the solenoid and the latch became disconnected during the test and the rod was operated manually for the greater part of the test.

E. MARK I PRODUCTION SAFETY ROD (E. C. Phillips)

Four Mark I production safety rods were assembled. During proof testing one of the rods jammed. Upon removing this rod from the furnace, it was found that the thimble had cracked in the center of the broached section. This crack occurred at a step welded joint; this joint is not present in the prototype thimble. X-ray inspection of the remaining production safety rod thimbles showed that the step welds in all of the thimbles were faulty.

After the thimbles were repaired and checked by X-ray, the four Mark I production rods were re-assembled, proof tested for 10 drops at 600° F, and delivered to SRE. These rods were installed in the reactor and the control circuits to the safety rods were checked. During this check, the holding coils in all four solenoids were damaged due to excessive current. The solenoids were replaced and the control circuits were corrected.

F. SRE MARK II SAFETY ROD (R. Hoff)

The Mark II Safety Rod is provided as an alternate to the Mark I which has encountered serious high temperature material problems due to the location of the drive mechanism in the rod thimble. The Mark II rod avoids these temperature problems by locating the drive mechanism above the reactor face where the mechanism can be at room temperature. Figure 16 is a schematic diagram of the Mark II safety rod.

The rod has been dropped 300 times from full height and 100 times from intermediate heights. These drops were made with 15 psig helium pressures and rod temperatures up to 800° F.

Time response measurements indicate that the rod falls approximately at the acceleration of gravity. In Figure 17 the solid curve was obtained from a recording

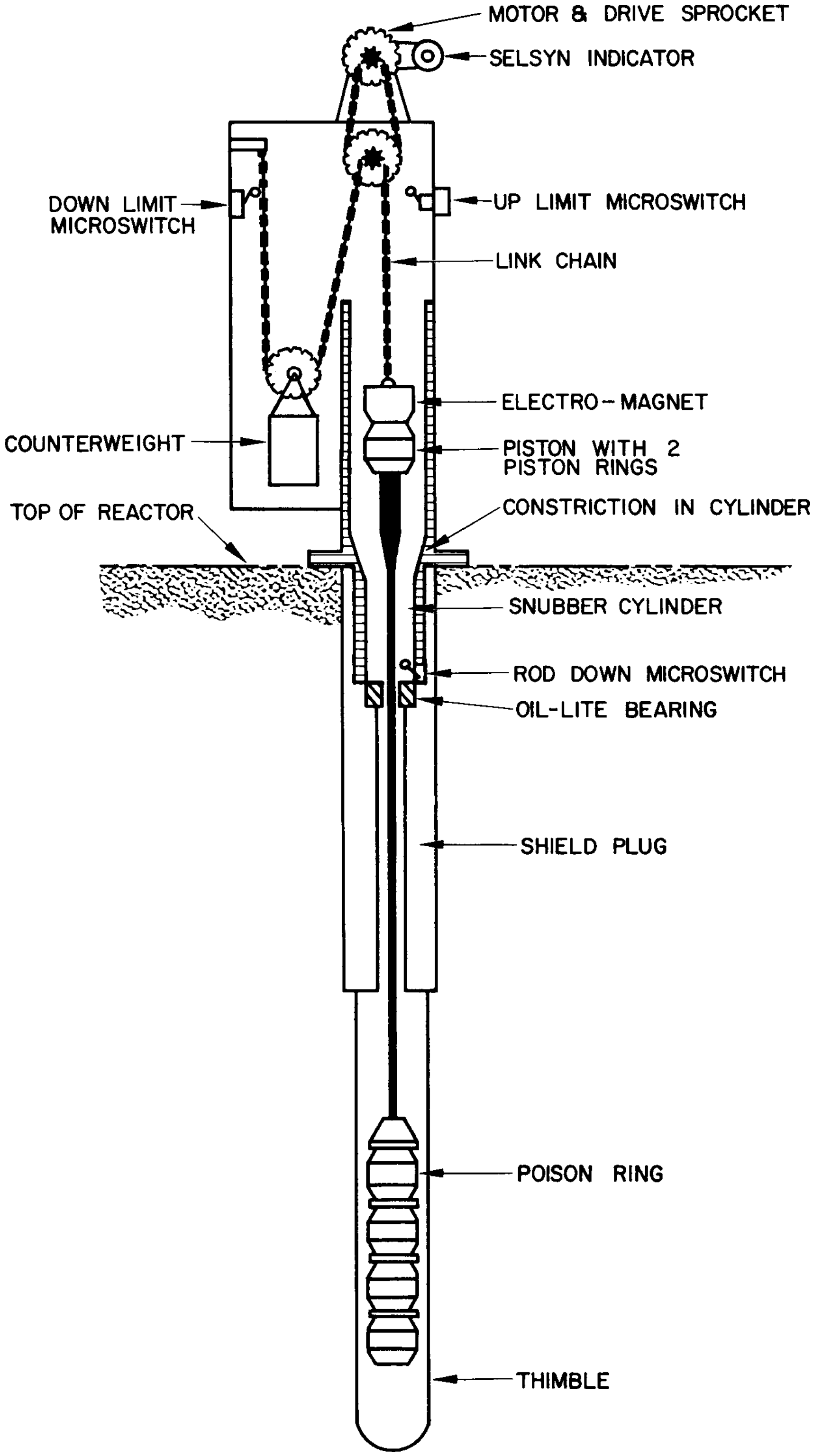

Fig. 16. Schematic Diagram of Mark II Safety Rod

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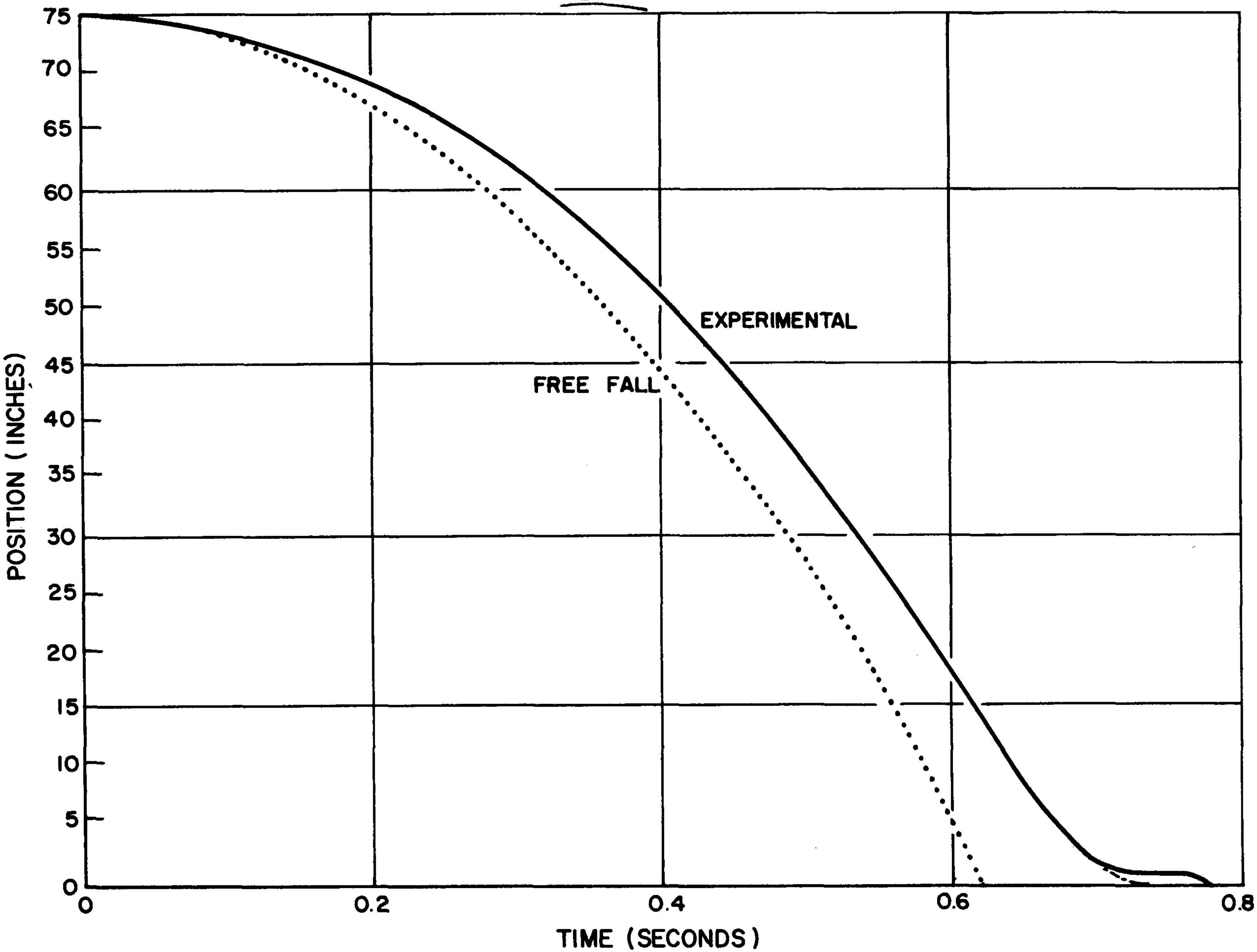

Fig. 17. Mark II Safety Rod Drop, Position vs Time

of rod position vs time when the rod was dropped from full height in 15 psig helium after 100 drops. This curve shows that the rod is almost completely inserted within 0.70 of a second. The dotted curve is a calculated curve for free fall due to the acceleration of gravity and shows that 0.62 second is required for the rod to drop fully in. The 0.70 second actually required to insert the rod indicates that very little time is consumed in releasing the rod, in friction, or in snubbing.

There is a noticeable decrease in the amount of snubbing available as additional drops are made. The solid curve shows snubbing achieved after 100 drops while the dot-dash extension is that obtained after 400 drops. Changing the piston rings made no effect on snubbing. The point of wear is the Oil-Lite bearing at the bottom of the snubber cylinder.

XIII. REACTOR SERVICES

A. PREOPERATIONAL TESTING (D. H. Johnson)

1. Electrical System

a. Heater

All electrical components in the pre-heating operation are being tested to verify the installations as specified in the design drawings.

Preliminary heating of various tanks, lines and valves indicated non-uniformity of heat distribution on sections of piping. In order to correct this non-uniformity in heat distribution, considerable rework has been performed. The extent of this rework has included: additional insulation in certain areas, addition of more heaters, replacement of faulty heaters, recircuiting heaters, and replacing heaters with those of a higher output.

b. Emergency Power System

Rework has been done to provide better efficiency for controlling the generator frequency output at increased emergency power loads. This rework consisted of placing additional resistors in the field coils of the diverter pole generators. The diverter pole generators have been tested and are working satisfactorily.

The diesel generator was connected and performance was satisfactory during 150 hours of continuous duty. Periodic tests are being conducted on the diesel alternator to determine drift under load and temperature sensitivity.

c. Sodium and Airblast Fan Drive Systems

The fabrication of the control circuits for the main primary and main secondary sodium pumps is 90 per cent completed. The main secondary sodium pump is now being operated by manual rheostat which will be used as a standby control in the normal control circuits.

d. Control Room

All thermocouple signal leads that terminate at the reactor top shield have been checked for grounds, thermocouple polarity and signal continuity to the readout instruments in the control room. There are a number of discrepancies in the wiring at the top shield. These errors in wiring are being corrected.

2. Sodium Systems

a. Auxiliary Secondary Sodium System

At the end of this quarter operational tests were still in progress on this system. Trouble was encountered with freeze seals. During one test the freeze seal failed at 1000 rpm with 8 psig on the case and 3.1 on the expansion tank.

b. Main Secondary Sodium System

Difficulty was encountered in heating the airblast heat exchanger. In order to correct this discrepancy, the control louvers were covered with insulating material to reduce heat losses.

Operational tests on the main secondary sodium pump are being conducted under manual control; it was run as high as 1500 rpm. The pump has a tendency to bind when the temperature is increased. Correction of this malfunction was accomplished by shimming up the motor. A slight leak was noted at the pump shaft syntron seal. The shaft seal is being reworked.

Sodium filling of this system was started March 22, 1957.

c. Primary Sodium System

Difficulties experienced with the syntron seals in the secondary pumps has prompted an investigation for modifying these seals on the primary pumps. Work on the main primary sodium system included: rework of temperature recorders, valve repairs, installation of new valves, and repairs to leaks in the primary fill tank.

3. Toluene System

Tetralin leaks were noted and repaired in the mass transfer system. The gallery cooling lines were helium leak checked and several leaks were repaired.

Other tests on this system included: the primary cold trap, auxiliary primary cold trap, main secondary cold trap, and sodium service disposable cold trap. With the exception of tests on the sodium service disposable cold trap, tests were completed and operation was satisfactory.

A sample of tetralin was taken from the circulating stream; it will be irradiated to determine the expected level of radiation from the tetralin supply tank during normal operation.

4. Core Elements

a. Source

The source was inserted in the reactor on March 11, 1957. Radiation was measured at 500 r, one foot from the source. Radiation above the top shield was measured at 0.5 mr/hr; no neutrons were observed.

b. Special Moderator Cans

The special moderator can plugs were reworked and installed. Leaks were detected in the tube fittings on the special moderator cans; they are in the process of being repaired.

c. Dummy Elements

Twenty-three dummy elements were loaded into the reactor core. Center channel dummy element 71605-006 hung up in hole 23 due to spacer protrusions on the lower tube. All elements of this design were reworked by installing guide pieces and grinding off the square corners.

d. Control Rods

Four Mark II and two Mark I control rods were cleaned and checked. Two Mark II and two Mark I control rods were loaded in the reactor and the drives mounted.

e. Safety Rods

Safety rods Nos. 5, 4, 3, and 1 were installed in core holes Nos. 63, 48, 40, and 27 respectively and operationally checked.

f. Fuel Elements

Twenty fuel elements were received and stored for the first dry sub-critical runs. Eighteen were loaded into the reactor core. Eight additional fuel elements were received, assembled to plugs and stored in the fuel storage area in preparation for the second dry sub-critical run.

5. Dry Sub-critical Test

Phase 1 of the SRE ambient, dry sub-critical test was conducted from March 16 through March 18, 1957. Predictions[2] indicated dry critical loading to be 13.8 fuel elements. The experiment was expected to be terminated after compiling data from a 12 element loading.

Extrapolation of data after the 12th element was loaded indicated a critical loading of 20 elements. Experimentation was continued until 18 elements had been loaded. Activities were temporarily suspended to allow for further review of the data.

Tests were resumed and the dry sub-critical test was completed on March 23. The test was terminated with 21 fuel elements. The indication was that 22 or 22+ fuel elements would be required for criticality.

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2. F. L. Fillmore, "Two-Group Calculation of the Critical Core Size of the SRE Reactor", NAA-SR-1517, July 1, 1956.