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SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT JANUARY-MARCH, 1956

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SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT JANUARY-MARCH, 1956

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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-1513	JULY-SEPTEMBER, 1955 (March 15, 1956)
NAA-SR-1582	OCTOBER-DECEMBER, 1955 (April 15, 1956)

SECTION A
TECHNOLOGY OF THE SODIUM GRAPHITE REACTOR

I. FULL-SCALE SGR

(E. F. Weisner)

An IBM code permitting calculations to be performed on a variety of lattice types was prepared and checked out. A parameter survey was begun comparing the hexagonal canned moderator (19 rod fuel elements) with the cylindrical canned moderator (10 and 13 rod fuel elements). These results may then be used as input to a two-group, two-region criticality problem, the IBM code for which already exists.

II. REACTOR PHYSICS

A. DETERMINATION OF POWER AND TEMPERATURE COEFFICIENT OF REACTIVITY (G. W. Rodeback)

A method of determining the metal temperature coefficient of reactivity of the SRE by measuring the shutdown power transient under various power conditions was studied. Negative reactivity is introduced as a step function to initiate the transient. Transient studies have been made with an electronic analog computer which had been set up earlier to study the nuclear-thermal time behavior of the SRE. These studies indicated that if the SRE is operated at half power, full sodium flow and then scrammed with $0.5 worth of negative reactivity, the power level (at say 15 sec after scram) will have an approximate linear dependence on the metal temperature coefficient of reactivity. Specifically, the above power level increases 14 per cent per unit of negative temperature coefficient of reactivity where a unit of the latter is defined as $10^{-5}/°C$. Since the coefficient is estimated to be of the order of $10^{-5}/°C$, the above effect should be readily measurable. The temperature transient as determined by the analog computer can be approximated by an exponential function. This transient in turn corresponds to an exponential change in reactivity, depending on the value of the metal temperature coefficient, which can be added to the original step function of negative reactivity. An input function of reactivity modified by the above temperature effect has been applied to a one delay group transient analysis. This will yield a qualitative picture of the effect of metal temperature coefficient of reactivity on the shutdown transient. The actual temperature coefficient determination will most probably utilize a six-group transient analysis solved by an analog computer.

B. SRE START-UP

The study of reactor physics start-up procedures for SRE showed the necessity of making an approximate two-region solution of the SRE core to determine the flux distribution in the unloaded condition with start-up source present. Measurement of reactivity of the sub-critical SRE during the latter stages of fuel loading is being considered. This would be done by a period measurement following the sudden removal of the start-up source from the core.

A satisfactory high temperature sealed-off fission counter for use during SRE start-up has been developed. It has dimensions of approximately 4 in. by 1/2 in. and a sensitivity of 0.017 c/sec-nv. The counter has operated with no change in sensitivity for over a week at 200° C. When returned to 200° C after temporary cooling at 20° C it regains its original sensitivity.

C. REACTIVITY STUDIES AND FLUX PERTURBATIONS (F. L. Fillmore)

A paper concerning the lethargy spread of resonance neutron in uranium was submitted for publication and a report[1] was written.

A study was made on the subject of changing the zirconium moderator cans in the SRE to stainless steel. It was found that if the core cans are made of zirconium and the reflector cans of 0.020 in. stainless steel, it will be necessary to load 39 process tubes with SRE fuel (2.75 per cent enriched) in order to achieve criticality. If all the cans are made of 0.020 in. stainless steel, it will be necessary to increase the enrichment to about 7 per cent if 37 process tubes are loaded.

The possibility of spiking the present SRE fuel with fully enriched uranium was also investigated. The fully enriched fuel elements were taken to be 0.030 in. uranium foils fitted over a solid beryllium cylinder of about 2.3 in. diameter. It was found that about 1/3 of the SRE fuel would have to be replaced by these elements if 37 process tubes were loaded in order to achieve criticality. This amounts to about 50 kg of fully enriched uranium. If the spikes were inserted into thimbles without replacing SRE fuel, about 30 kg of fully enriched uranium would be required.

III. DEVELOPMENT OF HOT CELL FACILITIES AND HANDLING TECHNIQUES

(J. M. Davis)

During this period, progress was made in the development of cell equipment and handling techniques for metallurgical studies of experimental fuel material.

Two full size SRE fuel rods, similar in all respects to actual rods except that the loading is normal U^{238}, were used in developing handling procedures. An all stainless steel cluster was obtained for developing disassembly equipment and techniques. Emergency disassembly tools have been designed and partially fabricated to insure removal of fuel rods from the cluster in case of unexpected disassembly difficulties.

Three basic methods of de-canning the fuel were considered. An ultrasonically vibrated cutting tool operating in a stream of abrasive was tested and found unsatisfactory due to excessive cutter wear. A commercial device utilizing a high frequency arc to remove metal and a movable abrasive saw type tool were also considered. The decision as to the method to be used awaits further comparison of performance data, costs and contamination problems.

The equipment for sectioning slugs has been completed except for the sludge trap and the coolant circulation. The controls of a lapping machine, to be used for grinding, have been adapted for remote operation. Modification and design work was carried out on a remote polishing device and on specimen mounting equipment.

A study of the photography problems and the binocular viewing of objects within the cell was completed and recommendations for procurement of suitable equipment made.

Tests of fabricated Type 2S aluminum cans for use in re-canning examined fuel material show that the use of all-crimp seals is unsatisfactory. Welded and extruded cans are presently under consideration.

Alterations in the design of the opening equipment previously built for the NAA 15-2 in-pile experiment irradiated at MTR were made. The alterations were necessitated by changes in the design of the capsule assembly. A new shipping cask for transporting the assembly has also been designed.

IV. METALLURGY OF SGR FUELS

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

A. EXPERIMENTAL FUEL SLUGS

The study of new uranium fuel material under high temperature operation and long life included cast uranium slugs and powder compacted slugs. Development work has been performed on fabrication of high density, homogeneous, uniform microstructure, dimensionally stable, materials with a few shape variations.

The centrifugal casting technique was used at Fernald in supplying 50 development slugs of each of three alloys, U-Mo, U-Zr, and U-Si. As a result of unfavorable irradiation results from Argonne, the U-Si alloy (0.5 w/o Si) has been eliminated from the program. The final batch of slugs in this development phase of the program were received and examined. The as-cast surface of both the alloyed and unalloyed slugs were unsatisfactory for use in the SRE. However, surface quality improvements have been made since receipt of the shipment. Heating of the mold to 500° F resulted in a better surface. It is planned at this time to cast oversize by about 0.020 to 0.030 in. and centerless grind to the finished dimensions. A few minor defects may remain in the slug surface. Further development of the casting method is expected to result in an as-cast surface which is satisfactory in a fuel element using a liquid metal bond. It is planned to reactor test a small number of cast and beta-treated unalloyed uranium slugs for performance comparison with the alpha-rolled beta-treated main fuel slugs.

The feasibility development work has been completed at SEP on the two types of powder compacted uranium slugs; (1) unalloyed large hollow slugs 2.42 in. OD by 1.35 in. ID by 4 in. long, and (2) cold pressed and sintered U-Mo (1.2 w/o Mo) alloy slugs 0.750 in. by 1 in.

A thorough metallurgical fabrication history will be maintained on all of these experimental fuel slugs. Representative samples of the input and final product will be obtained for evaluation of the irradiated slug performance.

B. CORED SLUGS FOR SRE FUEL TEMPERATURE MONITORING

The cored slug is expected to perform two principal functions in the SRE; (1) permit temperature mapping of the reactor core and (2) provide a design variation which may decrease distortion resulting from thermal stress and radiation damage. Three main fuel elements will each contain two stainless steel sheathed thermocouples, (0.125 in. diameter) located at different positions along the length of the fuel elements and centrally located within the fuel rod. The fabrication of the 3/16 in. hole consisted of preheating the slug to about 250° C, transfering the slug to a heated split sleeve mounted in a lathe, and drilling the slug half way from each end, using a high speed drill and Houghtons No. 35 cutting oil. The fabrication of the slugs for these three elements has been completed.

V. METALLURGY OF BREEDER FUELS

(B. R. Hayward, G. G. Bentle)

Thorium-uranium alloys are one of the principal types of experimental fuel materials to be studied in the SRE. Development fabrication work has been completed by Nuclear Metals for casting, extruding, swaging and finishing 0.750 in. by 6.00 in. thorium-uranium alloy slugs containing 5.4 per cent uranium. The principal problems were to fabricate a homogeneous and sound ingot. The ingots were sectioned, sampled, examined and compared with sections taken from the finished slugs.

Additional thorium-uranium alloys are being considered for the study of these alloys at higher temperature. Nuclear calculations indicate that for operating at a peak internal temperature of 1500° F and a slug surface temperature of 1100° F, approximately 11 per cent of fully enriched uranium is required. For a peak internal temperature of 1800° F and a slug surface temperature of 1145° F, approximately 16 per cent uranium is required. These alloys are of a higher uranium content that would be used in a large scale power reactor. The phase diagram indicates the same general type of structure should be present in each of the three alloys (5.4, 11, and 16 per cent uranium). In each case, a two phase alloy will exist. However, the constituents will vary in relative amounts. This factor will be reviewed in evaluation of the performance of these materials in the reactor.

VI. ENGINEERING EVALUATION OF GRAPHITE

(R. L. Carter, J. J. Gill, H. Taketani)

A 3-3/4 in. by 3-3/4 in. by 48 in. sample in two segments was prepared from one of the SRE engineering spare graphite hex logs. This sample was submitted for delta-in-hour comparison with a KC standard sample at the Hanford 305 Pile. Extreme care was taken to prevent contamination of the sample both during machining and during subsequent wrapping for shipment.

Detailed planning for the operation of the SRE vented moderator cans and vented dummy elements was continued with the compilation of the operational outline and with the finalization of equipment requirements preparatory to placement of orders.

Contacts were initiated to ensure increasing cooperative effort between this laboratory and the group carrying on the AEC sponsored research on gas absorption properties of graphite at Pennsylvania State University.

Degassing equipment was completed for the survey checks on SRE graphite gas content. The apparatus was employed in conjunction with an 8 kw induction heater to evaluate gas evolved during heating of samples of SRE moderator graphite to 1400° C. Data taken to date are in reasonable agreement with anticipated gas content based upon experimental work performed during development of grade TSP graphite.

Examination of techniques available for quantitative measurement of radiation induced hardness changes in graphite has commenced with a study of micro hardness profiles of large crystals of unirradiated graphite.

VII. ZIRCONIUM BEHAVIOR IN LIQUID SODIUM

(J. C. Bokros, R. L. Carter, R. L. Eichelberger, F. H. Eisen, R. B. Hinze, R. L. Mc Kissen)

Progress has been made toward the completion, operation and continued improvement of moderate scale liquid sodium facilities for the exposure of samples of structural material. Weight gains, hydrogen content, and fatigue life changes in these samples are being evaluated after exposure. Preliminary development of a zirconium or zirconium alloy getter hot trap is also being carried out.

A. FACILITIES FOR LARGE SCALE EXPERIMENTAL WORK (R. B. Hinze)

The two new test units added during the last quarter have been in steady operation. One is a static chamber with cold trap; the other, a dynamic loop with a hot trap, using zirconium-titanium alloy as a getter material. The addition of this equipment has tripled our facilities for exposing zirconium specimens to large (50 to 100 pounds) quantities of sodium.

Improved instrumentation and control on the test apparatus now gives us internal consistency in the data from the several exposure facilities. To improve the purity of the cover gas, a zirconium-titanium gettering device in series with the NaK bubbler has been added to the static test unit. Contamination due to sample changing has been eliminated by the addition of a gas lock.

The fourth large test loop is being redesigned to incorporate both a hot trap and a circulating cold trap to simulate the contaminant control system of the SRE.

B. HOT TRAP EXPERIMENTS (R. L. Eichelberger)

A sound approach to the problem of scavenging oxygen from a molten sodium system, to protect easily oxidized materials in the sodium, is to include in the system a quantity of material which will react preferentially with the oxygen. For a sodium-zirconium system, an abvious approach is to have this oxygen "getter" be zirconium or a zirconium alloy. Small scale capsule tests at 650° C with zirconium and an alloy containing 50 atom per cent titanium showed that the alloy oxidized at a significantly higher rate than the zirconium, when both materials were at the same temperature.

A dynamic test loop was constructed to investigate the protection afforded zirconium samples by the use of a "hot trap" filled with such getter material.

This system contained about 75 pounds of flowing sodium. The hot trap incorporated into this loop was packed with lathe turnings of zirconium-titanium 50 atom per cent alloy. The hot trap volume was packed to a getter density of about 3 per cent that of the solid alloy. Distributed in the hot trap were 12 weighed specimens.

After one month flushing with sodium at temperatures below 500° F, the loop was charged with fresh sodium. The hot trap was heated to temperatures of 1000° F and higher for several hundred hours to clean up the system before samples were added to the specimen pot. Plugging runs to determine oxygen concentration showed the plugging temperature dropped from 610° F to 430° F during this period.

Weighed specimens of chemically polished zirconium were inserted in the specimen pot and exposed at 1000° F with the hot trap at 1200° F. The surfaces of all the specimens were still bright after the 120 hours exposure. There was good agreement in the weight gains, and they were somewhat less than the values for the same length of exposure at 1000° F in cold trapped loops.

A second set of specimens subsequently exposed in the sample pot using the same hot trap contained four specimens previously exposed in the first test and seven new specimens. Duration of the test was about 220 hours with the specimens at 1000° F and the hot trap at 1200° F. The previously exposed samples -- which had gained 0.14 mg/cm^2 on the first exposure -- were still bright and showed no additional weight change after the second exposure, within weighing uncertainty of 0.0002 gm in 3.5 gm. The seven new samples were consistent with an average weight gain of 0.148 mg/cm^2. At the end of the test, an attempted plugging run showed no indication of plugging down to 300° F.

Several interpretations of these results have been considered, and only further experimentation will determine the most probable one. It is possible that some limiting surface concentration of oxygen in the zirconium is built up during short exposure, after which weight gain is slow. Another possibility is that the oxygen concentration in the sodium was slowly decreasing and had become so low by the end of the test that oxygen pick-up by the zirconium was slow. Further interpretation will have to wait for the results from other samples exposed during the second test and future experiments.

At the end of the second test the hot trap was opened and its contents examined. Of the 12 weighed specimens of getter coil in the hot trap, five were intact at the end of the test and were re-weighed. Weight gains, in weight per cent ranged from 1.29 per cent to 1.71 per cent averaging 1.48 per cent. Extrapolated to the amount of material in the hot trap, the weight gain for the entire trap is over 13 grams. This corresponds to 380 ppm of oxygen for the 75 pounds of sodium in the loop. Further tests of the hot trap getter are underway.

C. ANALYSIS OF NaZr LOOP DATA (F. H. Eisen, R. L. Mc Kisson)

The weight gain data for zirconium specimens exposed in the cold-trapped sodium loops are given in Fig. 1. These data have been fitted to a parabolic rate law given by the activation energy for the parabolic rate constant determined to be 24 k cal/mole or equivalent.

Log-log plots of the weight gain vs time were made at each temperature. The best straight lines were drawn through these points and the slopes determined. These are given in Table I and are seen to distribute themselves about 0.5, which would be the slope for a rate law of the form $W^2 = Kt$. It was assumed that this rate law should hold, and the best line with slope 0.5 was drawn for each set of data. The log-log plots, with these lines drawn are shown in Fig. 1. The intercepts of these lines with the line t = 100 were arbitrarily used to calculate K from the expression

$$K = \frac{W^2}{t}$$

K is a function of temperature and would be expected to obey an equation of the form $K = K_o \exp(-Q/RT)$. (T = temperature in ° K). Figure 2 shows log K plotted vs 1/T. The circled points (see Fig. 2) are for cold trap temperatures corresponding to an oxygen concentration of 10 ppm in the sodium, and are seen to fall fairly close to a straight line. This line corresponds to an activation energy of 24 k cal/mole.

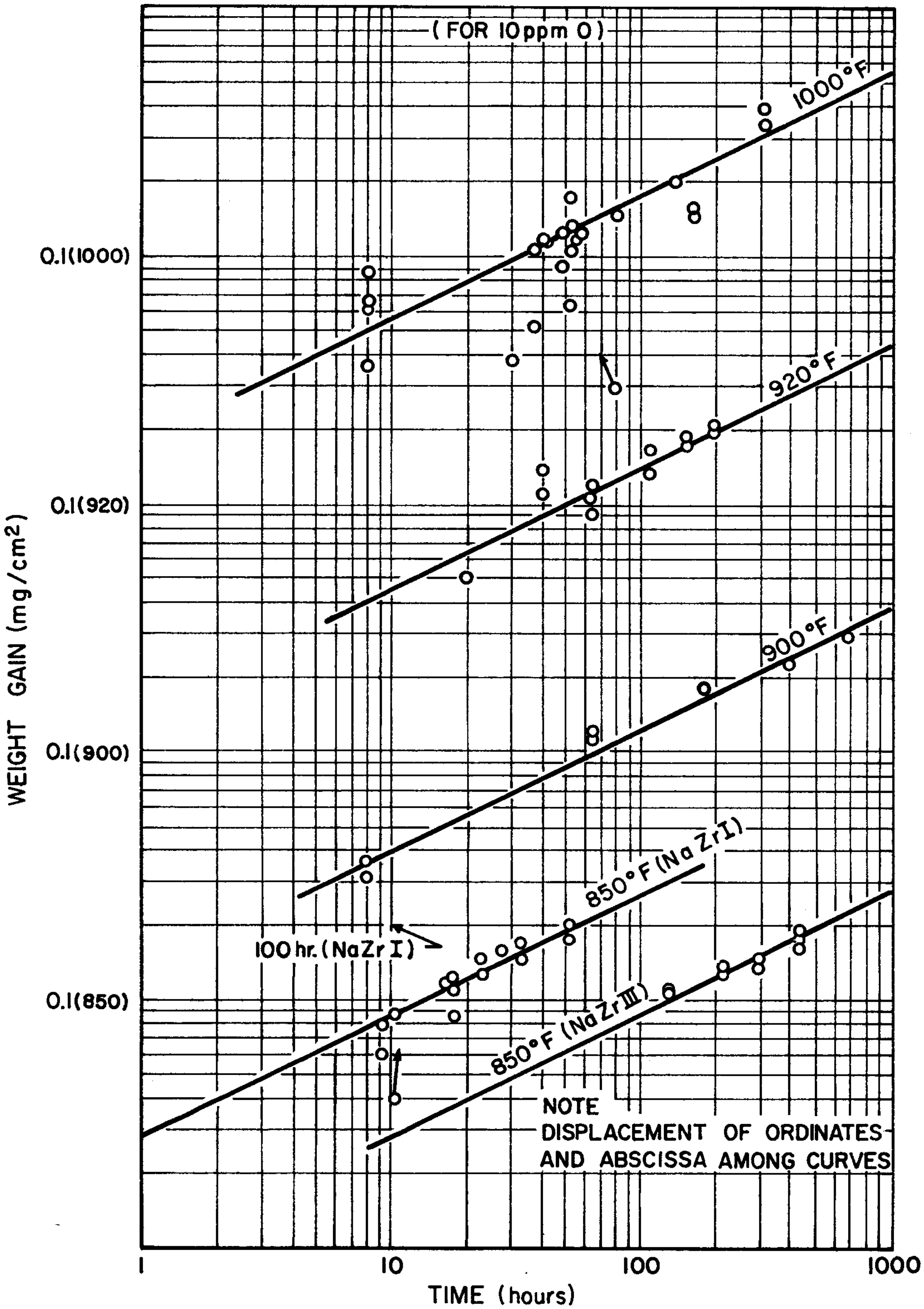


Fig. 1. Weight Gain vs Time for Exposure at indicated Temperatures to Sodium containing 10 ppm Oxygen

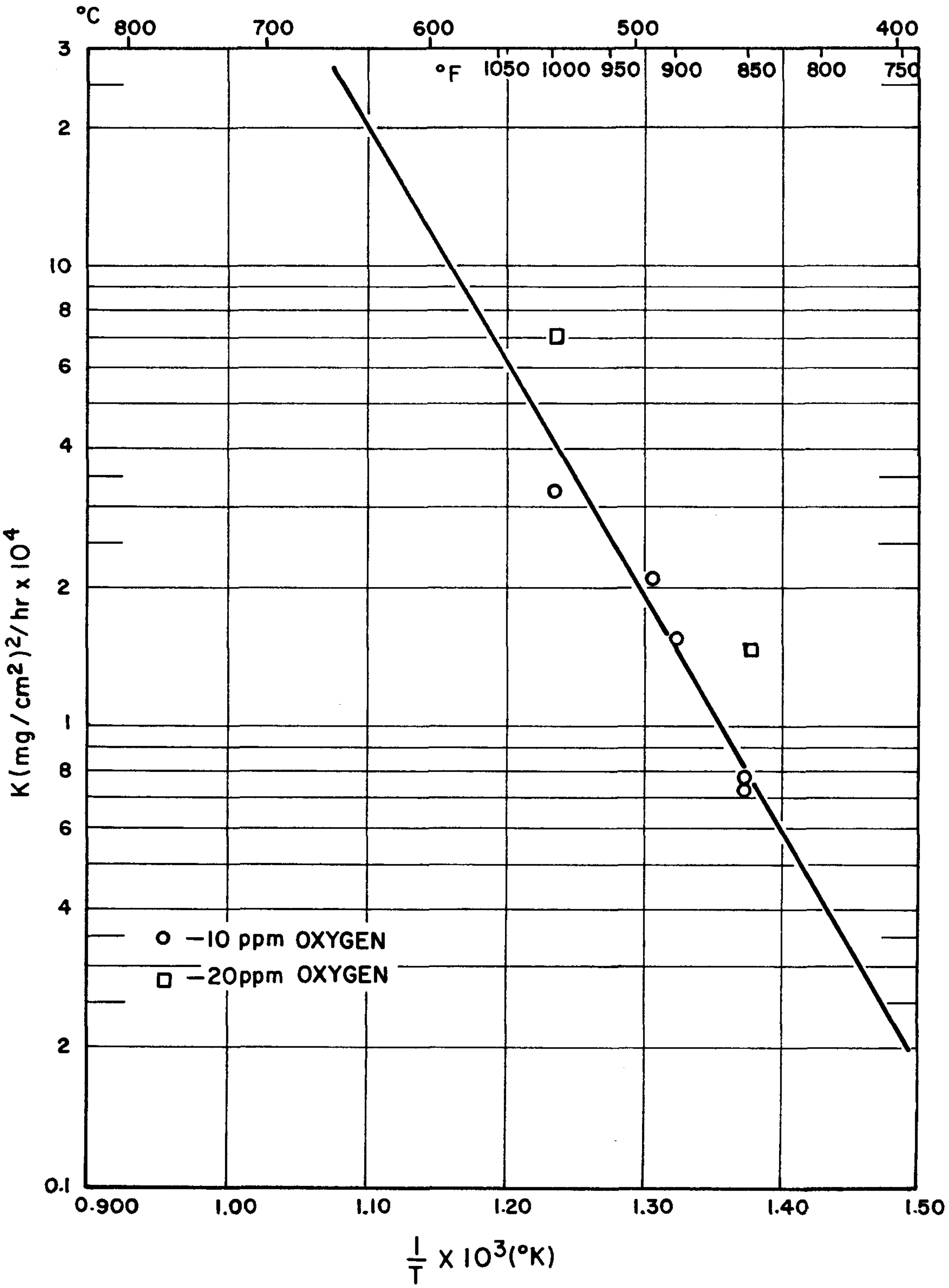


Fig. 2. Rate Constant K vs $\frac{1}{T}$

TABLE I

LOG-LOG PLOTS OF WEIGHT VS TIME AT GIVEN TEMP.
(Slopes of Best Lines Used)

T °F	10 ppm Oxygen Slope of Best Line	K $(mg/cm^2)^2/hr \times 10^4$
1000	0.56	3.24
920	0.53	2.10
900	0.50	1.56
850 (NaZr III)	0.42	0.77
850 (NaZr I)	0.48	0.74
	20 ppm Oxygen	
1000	--	7.18
850	--	1.48

Data are available at 1000° F and 850° F for cold trap temperatures corresponding to about 20 ppm oxygen. The log-log plots for these data are shown in Fig. 3 and the values of K are enclosed by a square on the graph of K vs 1/T (see Fig. 2). A line through these points would be approximately parallel to the line through the 10 ppm points. In view of the scatter in the data at 850° F and 20 ppm oxygen such a line should probably not be drawn until more data are available on the variation of K with oxygen concentration.

This analysis of the sodium loop data follows procedures which are common in corrosion work, and the assumption of a parabolic rate law is reasonable. It should be possible to use the dependence of K on temperature to predict the weight gain at temperatures beyond the range of the present experimental data so long as the mechanism does not change. There are indications of a change in mechanism in the oxidation and water corrosion of zirconium.[2] These data indicate a fit to a rate law of the form $W = Kt^n$, with n dependent on temperature. The values of n for corrosion of zirconium in sodium given in the Table (labeled slope of the best line) show a regular trend with temperature. While the temperature range for these measurements may be too small to conclude that there is a definite variation of n with temperature, this possibility should be considered in any extrapolation of the present analysis to temperatures outside the range of the measure-

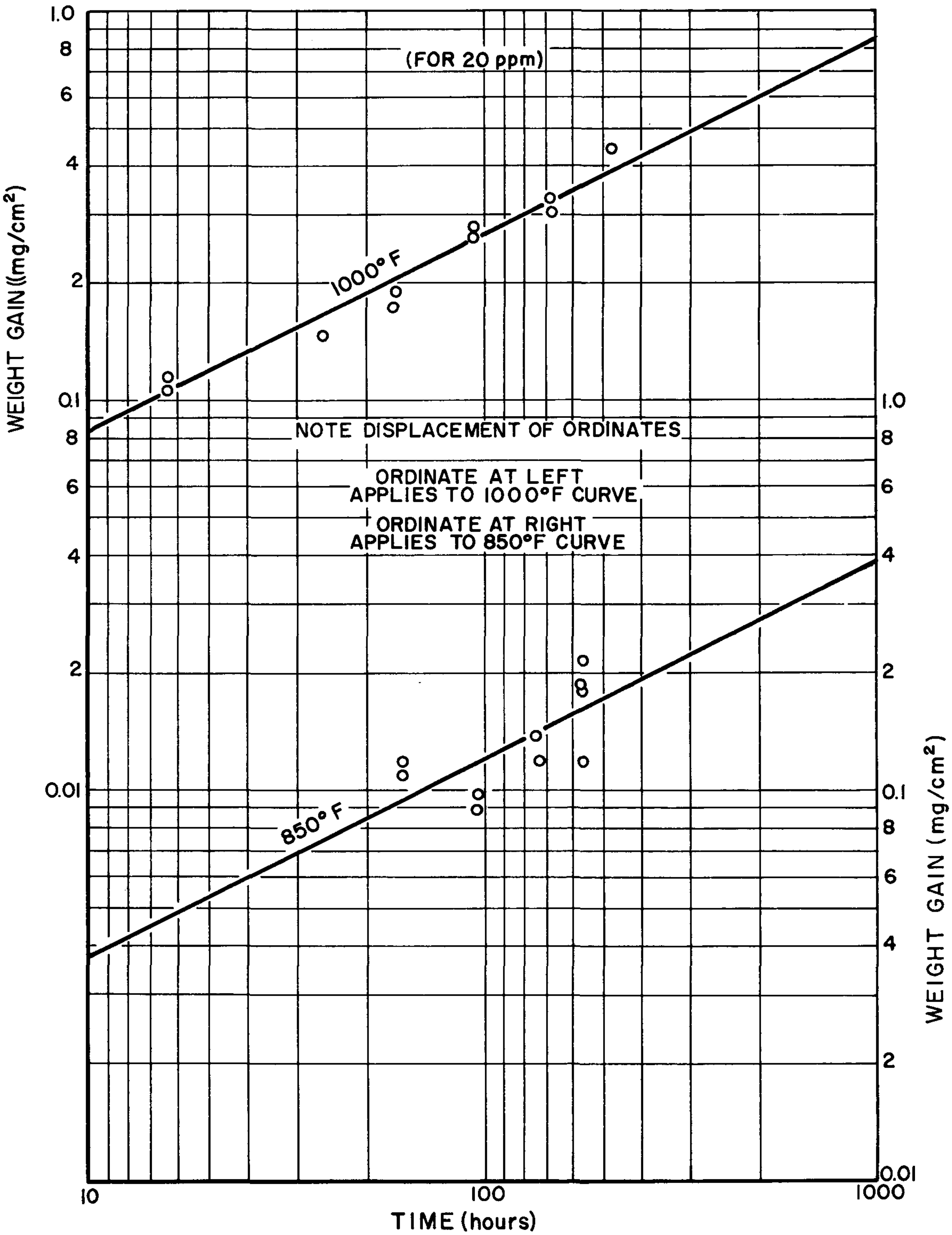


Fig. 3. Weight Gain vs Time for Exposure at indicated Temperatures to Sodium containing 20 ppm Oxygen

ments. It should be pointed out that the experimental data represent only the net change in weight of the zirconium samples, and the possibility that gases other than oxygen may be absorbed, or that some zirconium may be lost from the samples exists. Experimental information on these possibilities is desirable in order to confirm the interpretation of K as the parabolic rate constant for the oxidation of zirconium. Loop experiments at higher temperatures would also be desirable in order to check if there is a change in n at higher temperatures.

In view of the good work on the rate of absorption of hydrogen by zirconium which has been reported in the literature, additional investigations have not been considered necessary to date. A further consideration here is the fact that the rate of evolution of gases from the graphite can not be predicted. Two factors affect the rate of release of gas (principally hydrogen and carbon monoxide), the temperature and the radiation flux level. W. Greening[3] experimentally investigated the quantity of gas that was evolved from SRE graphite upon heating to various temperatures up to about 1700° C. He found at 1700° C that between 0.13 and 0.18 STP volumes of gas per volume of graphite were evolved. Further, the amount of gas evolved at temperatures below 1000° C was about 10 to 20 per cent of the total. These data can be used to estimate the amount of gas thermally desorbed from the graphite, but the amount and rate of desorption during reactor operation is unknown.

Since the rate at which zirconium absorbs hydrogen is a strong function of the partial pressure of hydrogen, an accurate prediction of hydrogen pick-up for this system cannot be made. A study of the schnorkel-closure question indicates that the zirconium moderator can walls can probably absorb the hydrogen released during pre-operational testing. Whether or not the absorption rate will exceed the release rate during pile operation is unknown. In any event, the zirconium will eventually absorb substantially all of the hydrogen released. For a total gas evolution of 0.2 STP volumes per volume of graphite the zirconium would contain about 190 ppm hydrogen if it becomes uniformly distributed.

Although no specific tests have yet been completed in the study of the problem of hydrogen in the SRE zirconium, the engineering tests on moderator can samples have produced a few pertinent observations. It was found that specimens of the zirconium cans used in the retort and can-head tests had hydrogen contents ranging from 30 to 200+ ppm. (Stock material averages about 35 ppm hydrogen).

Specimens with high hydrogen concentrations indicated a tendency for the hydrogen (hydride) to be concentrated in the grain boundaries. The observed breaks tended to follow these grain boundaries.

D. EFFECT OF SURFACE OXIDE ON FATIGUE AND RUPTURE PROPERTIES OF ZIRCONIUM (J. C. Bokros)

Data on the effect of surface oxide on the fatigue properties of zirconium at room temperature have been obtained and are plotted in Fig. 4. With a maximum fibre stress of 57,000 psi and a cycle rate of approximately 200 cycles per minute, the number of cycles to failure appeared to be independent of the amount of surface oxide in the range 0.5 to 1.6 mg/cm^2. Specimens in this range failed after 30,000 to 50,000 cycles. A concentration of 0.19 mg/cm^2 surface oxide resulted in failure at 83,000 cycles, and specimens free from surface oxide failed after 110,000 cycles or more. Metallographic examination of these specimens showed surface cracks in the specimens that had been oxidized while the specimens free of surface oxide showed no surface cracks near the point of failure. The number and depth of these surface cracks increased with increasing amounts of surface oxide.

At 515° F, specimens were tested under maximum stress about 50 per cent above the yield strength of annealed zirconium sheet (assuming strain hardening occurs and elastic behavior is realized). These showed a decrease of the fatigue life from 57,000 cycles for unoxidized zirconium to about 20,000 for oxidized zirconium. All of the specimens having surface oxide showed many cracks in the surface near the failure.

Stress-to-rupture tests of zirconium at 1000° F have been performed. The data are plotted in Fig. 5, together with a straight-line extrapolation out to 100,000 hours (11 years). The stress value corresponding to this time, 1700 psi, illustrates the need for the development of can and weld joint designs having minimum stresses. It should also be pointed out that these data are for zirconium which is not severely contaminated with oxygen or hydrogen, and that such contaminated material may have somewhat higher (100,000 hour) strength. Tensile tests on zirconium of the type to be used in SRE have also been made at room temperature, 400° F, 750° F, 1000° F and 1200° F. These results are given in Fig. 6.

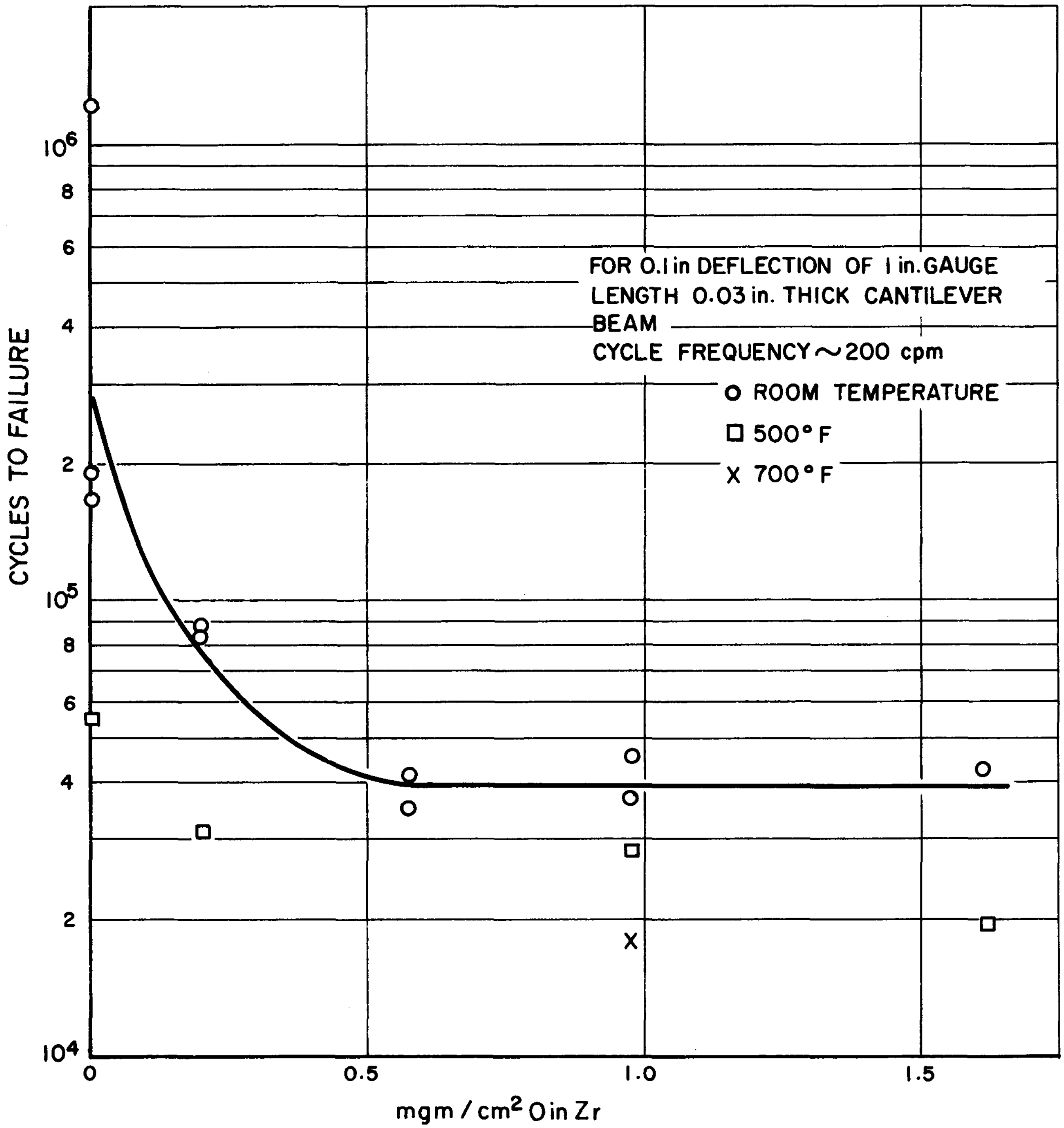


Fig. 4. Effect of Surface Oxide on the Fatigue Properties of Zirconium

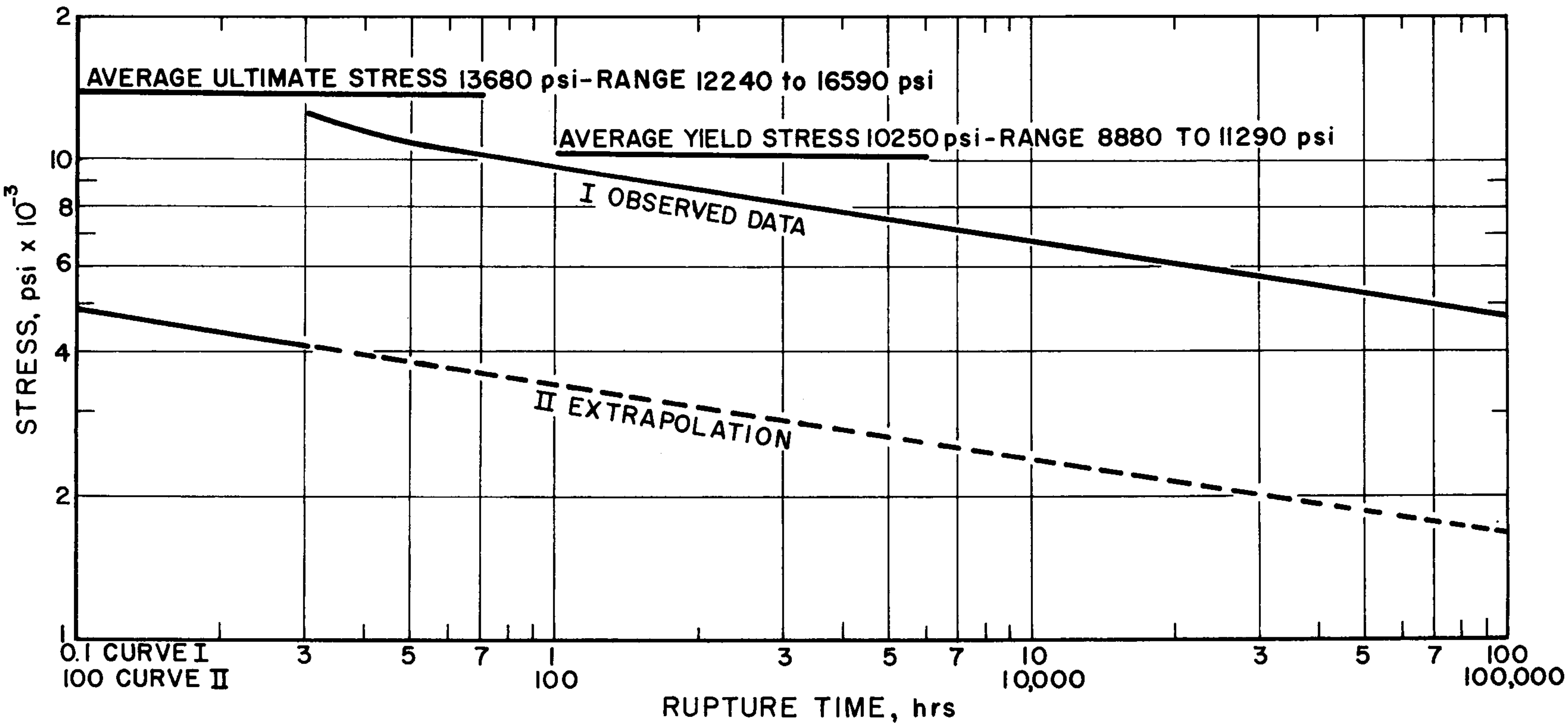


Fig. 5. Stress Rupture Data Hafnium Free SRE Zirconium 0.035 in. Sheet Longitudinal Grain, Helium Atmosphere Test Temperature 1000° F

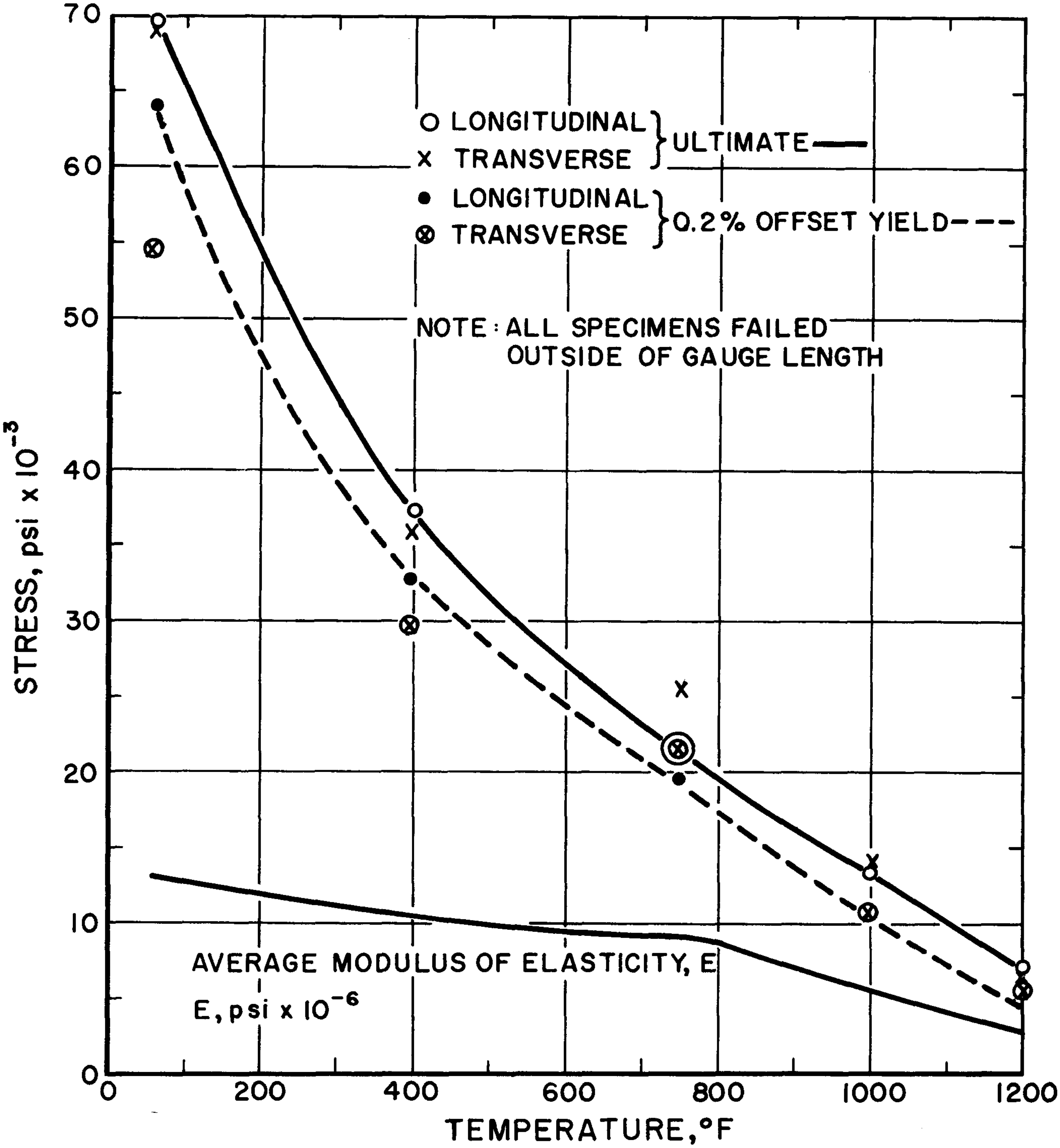


Fig. 6. Tensile Strength of Hafnium Free SRE Zirconium 0.035 in. Sheet Stock

SECTION B
SODIUM REACTOR EXPERIMENT

VIII. NUCLEAR ENGINEERING AND PHYSICS

A. SYSTEMS ANALYSIS (H. Dieckamp, D. J. Cockeram, L. Blue, E. Ash, D. Casey)

Analog computer studies of convective flow in the SRE after shutdown have been made. For purposes of this study it was assumed that the auxiliary loop was not operating and the main air blast heat exchanger (or the Southern California Edison steam generator) maintained a constant secondary sodium intermediate heat exchanger inlet temperature of 440° F. Heat capacities in the pipe and heat exchangers were neglected. The results of the analog computer studies yielded system temperatures and convection sodium flow rates as a function of time after shutdown. The analog computer simulated conditions existing for a 12 hour period following scram from full power operation. The results of the studies led to the following conclusions:

1. The primary convective flow immediately after shutdown from full power operation is 11 per cent of normal full power primary flow. Similarly, the secondary flow is 12 per cent of its full power value.
2. The convective flow alone should provide more than adequate cooling capacity for the reactor anytime after shutdown.
3. The cooling in the reactor immediately after shutdown yields a rate of temperature decay of the upper plenum sodium of about 5° F/min. Previous calculations have shown that a rate of 15 to 20° F/min was sufficient to cause yielding of the core tank. A review of the stress calculations will be made to make sure there is an adequate safety factor in tank construction at this rate of temperature change. Some additional convective flow studies will also be made to find how much effect simultaneous auxiliary system flow has on this rate of temperature change.

On the basis of these results, it is being considered that during normal operation the auxiliary loop will be operated at the lowest possible flow required to maintain the normal temperature gradient. After a scram, the addition of the auxiliary flow to the main convective will then not appreciably affect the rate of change of upper plenum sodium temperature. If the convective flow is stopped by valving off of the main loop, the auxiliary flow can then be increased as rapidly as necessary to remove the afterglow heat.

Several computer runs were made on the SRE analog to determine the feasibility of using regulating rod insertion for reactor shutdown following sodium pump failure. The study was made of a reactor in the steady state condition at full power (20 Mw) and full flow, with the sodium pump suddenly scrammed and one, two or four regulating rods inserted at the present design speed of 3 fpm, simultaneously with the pump scram. In addition, computer runs were made with insertion of four rods at twice and quadruple the present design rate, or 6 and 12 fpm respectively. In all cases, except for the 12 fpm run, design temperature limits in the fuel channel exit sodium temperature were exceeded (see Fig. 7). Reactor shutdown by regulating rod insertion following a sodium pump failure does not then appear to be feasible, unless very high rod rates were possible.

A study was made of the effect of different flow decays on the fuel channel outlet sodium temperature, Tc_{out}, after simultaneous sodium pump scram and rod scram. Previous studies have shown that Tc_{out} varies rapidly over a wide range after simultaneous rod and pump scram. In an attempt to minimize the Tc_{out} transient, it was decided to try faster flow decay rates after the simultaneous rod and pump scram. This could be achieved in the actual case on the SRE by dynamic braking of the sodium pump.

The SRE analog was run to simulate a reactor at full power (20 Mw) and full flow, with the rods and sodium pump scrammed simultaneously. Various forms of flow decay were used with a rod worth of $4\%\delta k$ and $10\%\delta k$.

The results of the above runs are presented in Fig. 8 and 9. The "E" run with rod worth of 4% looks promising. This run corresponds to a pump time constant of slightly less than one second and dropping of about half the safety rods. If all of the rods are dropped ($10\%\delta k$), then the dynamic braking must effect a time constant of about 1/2 second. However, in all cases, the final flow is excessive, resulting in too rapid cooling. For this method of scramming to be worth while, some decrease in the final flow would be necessary. This would be difficult to achieve, as this final flow is the convective flow.

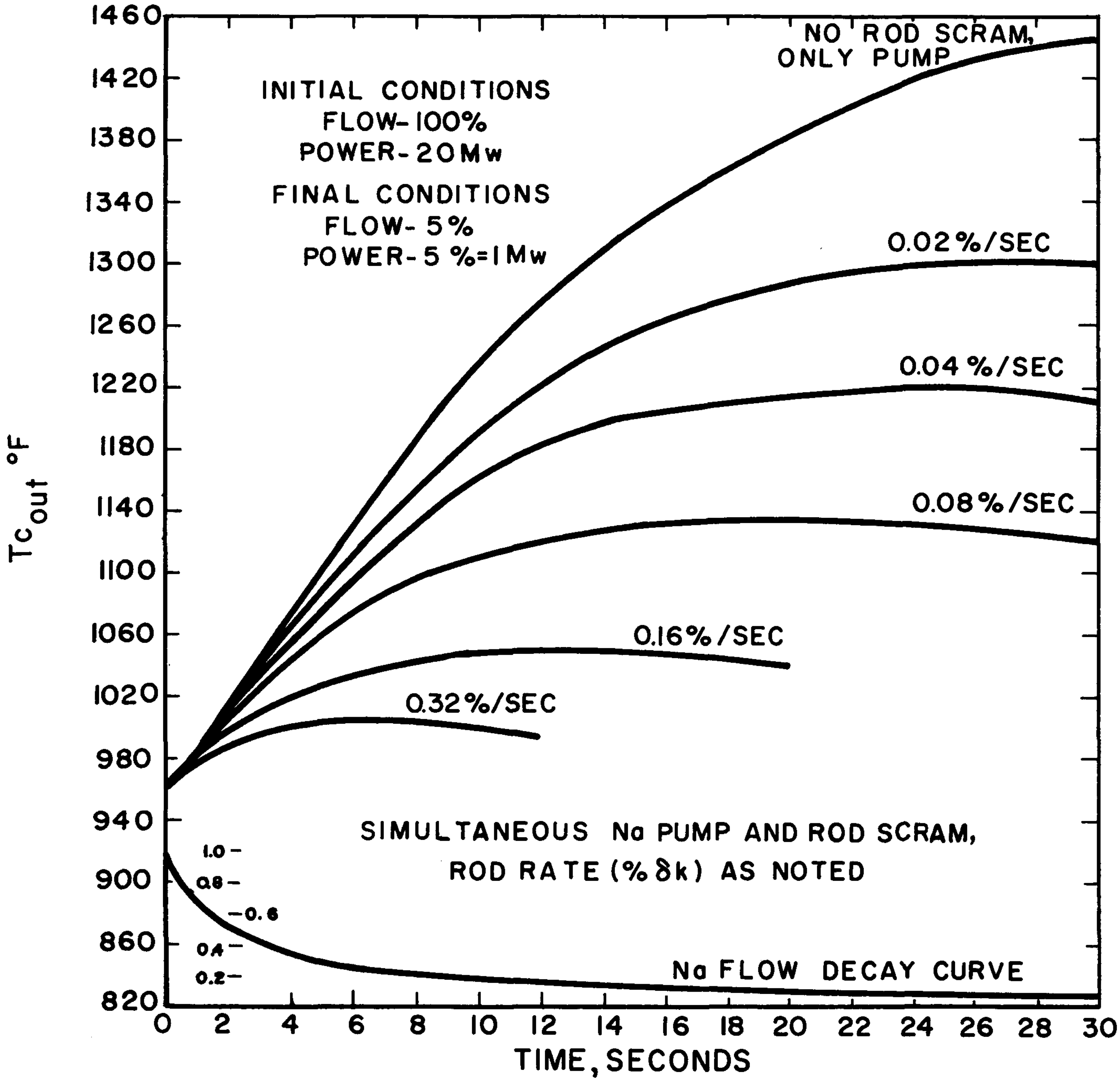


Fig. 7. Fuel Channel Exit Sodium Temperature following Reactor Scram

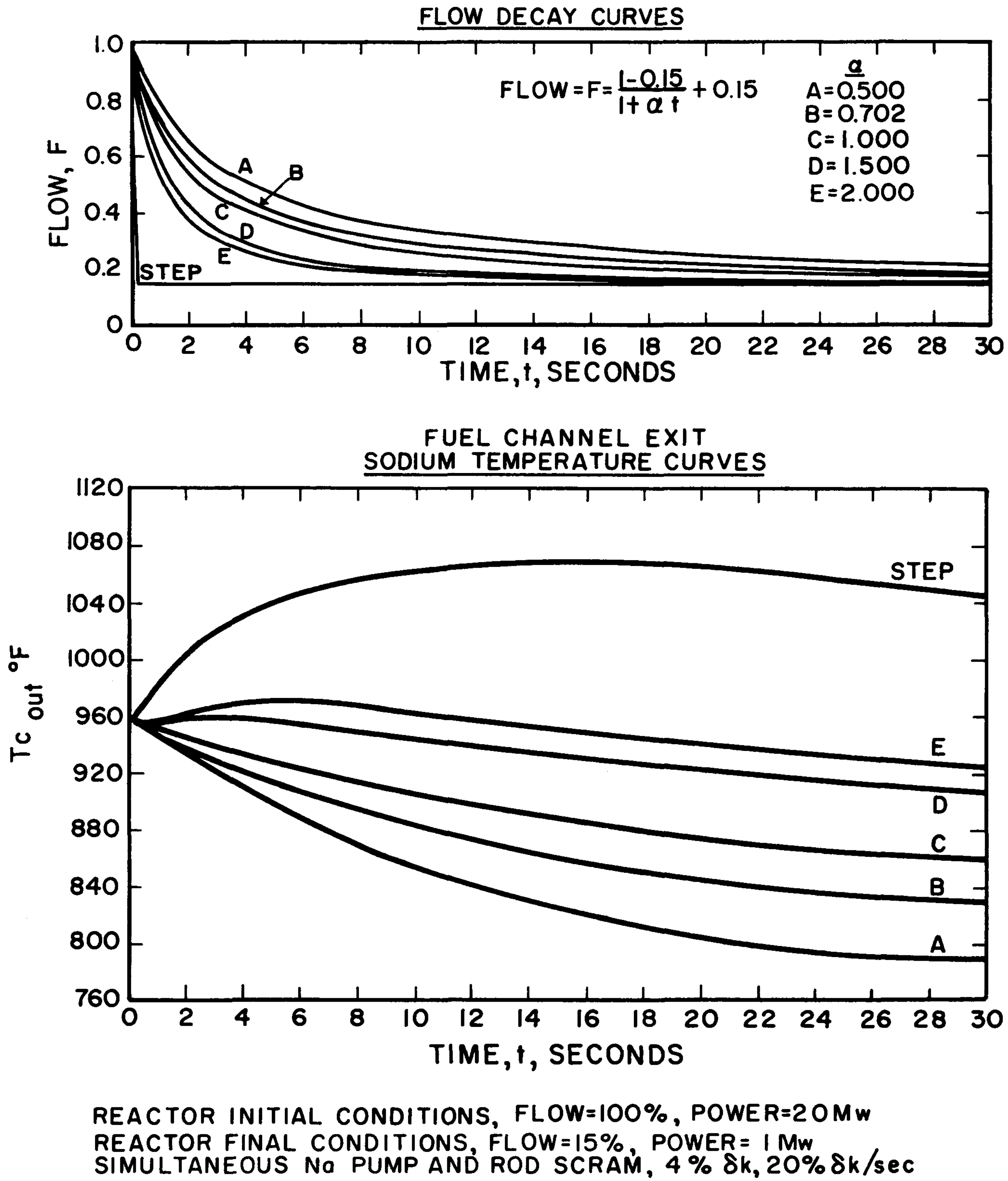
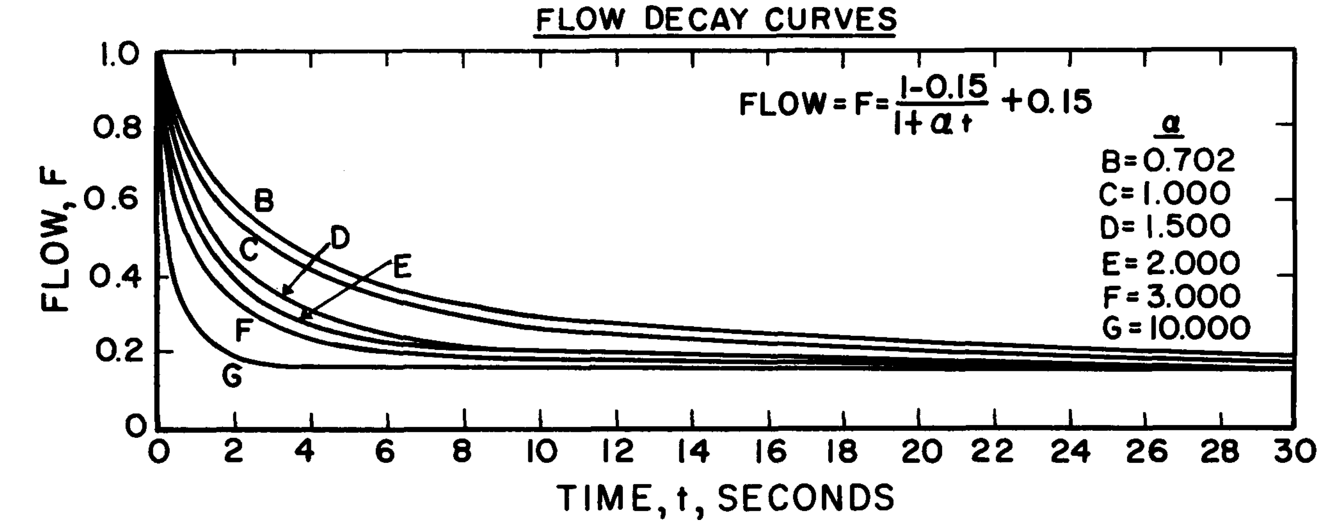
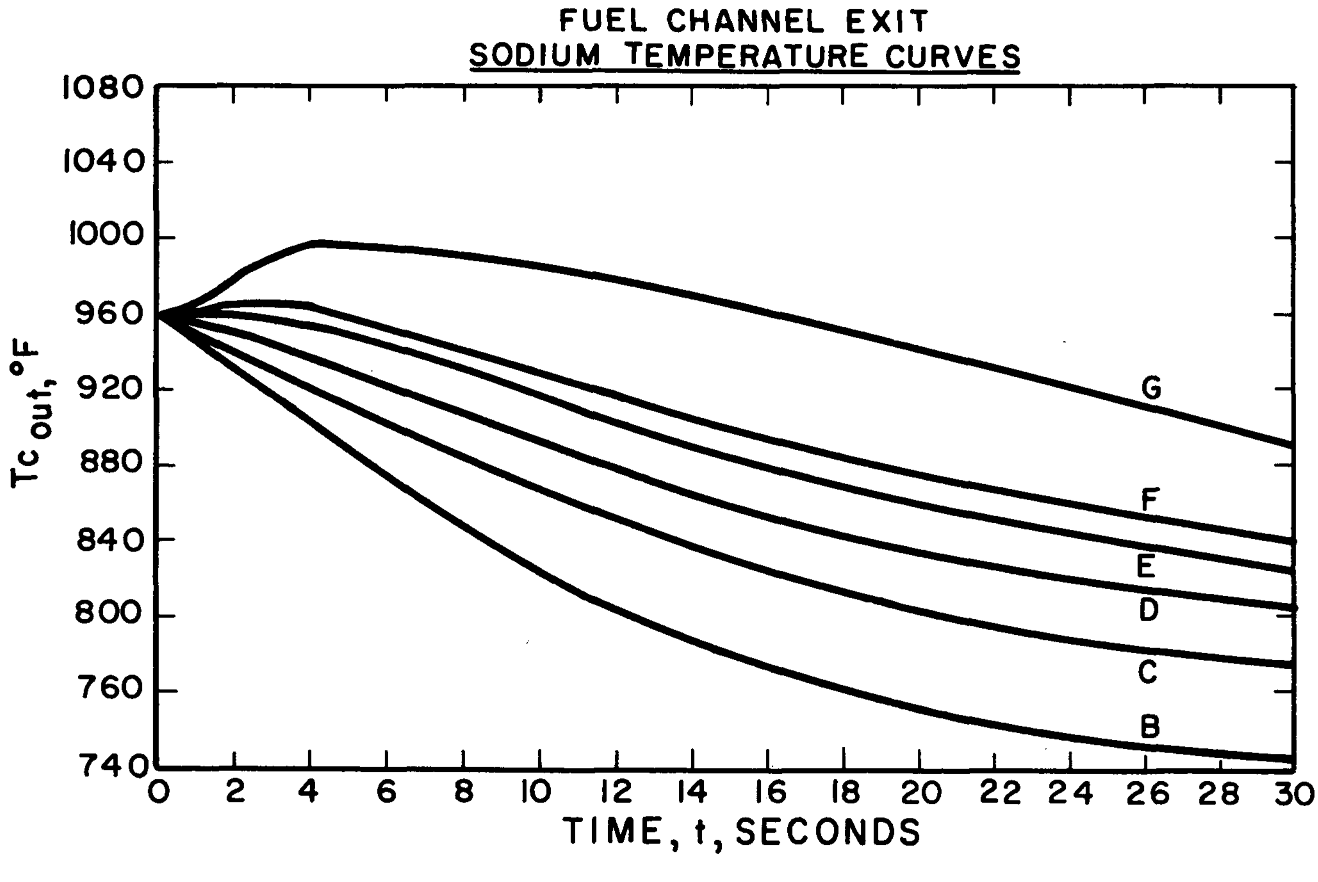


Fig. 8. Dynamic Braking Studies (Simultaneous Na Pump and Rod Scram, 4% δ k at 20% δ k/sec)





REACTOR INITIAL CONDITIONS, FLOW = 100%, POWER = 20 Mw
REACTOR FINAL CONDITIONS, FLOW = 15%, POWER = 1 Mw
SIMULTANEOUS Na PUMP AND ROD SCRAM, 10% δk at 20% δk/sec

Fig. 9. Dynamic Braking Studies (Simultaneous Na Pump and Rod Scram, 10% δ k at 20% δ k/sec)

An additional precaution in a dynamic braking of the sodium pump is that only on those scrams which were simultaneous rod and pump scrams, or those in which rods preceded pumps, would dynamic braking be desired. If the pump scram precedes the rod scram, then dynamic braking would only make the situation worse by raising Tc_{out} initially.

IX. LAND, UTILITIES AND BUILDINGS

(J. C. Cochran)

The general status of building and Site construction can be seen in Fig. 10 and 11. All of the side paneling has been erected and the roof has been installed.

The sodium service building is almost completed; the secondary fill tank has been located within the building (see Fig. 12).

The foundations in the secondary area have been poured. All of the flooring inside the building has been poured with the exception of the area in the vicinity of the cleaning cells. The main and auxiliary galleries have been lined with sheet metal and components are being put in place (Fig. 13).

The hot cell walls have been poured; the finishing of the cells, including installation of windows, is in process (see Fig. 14).

Construction in the Southern California Edison area is proceeding well; the turbo-generator, condenser and electrical equipment are in place. Construction of the cooling tower is well along.

Construction of an auxiliary water supply to the Site has begun.

All design, drawings and bid reviews for the entire reactor building have been completed and released. Detail drawings for fencing and fire protection are being completed. The preparation of as built drawings are now in progress.

As built drawings for building layout plot plan and the water distribution system have been prepared.

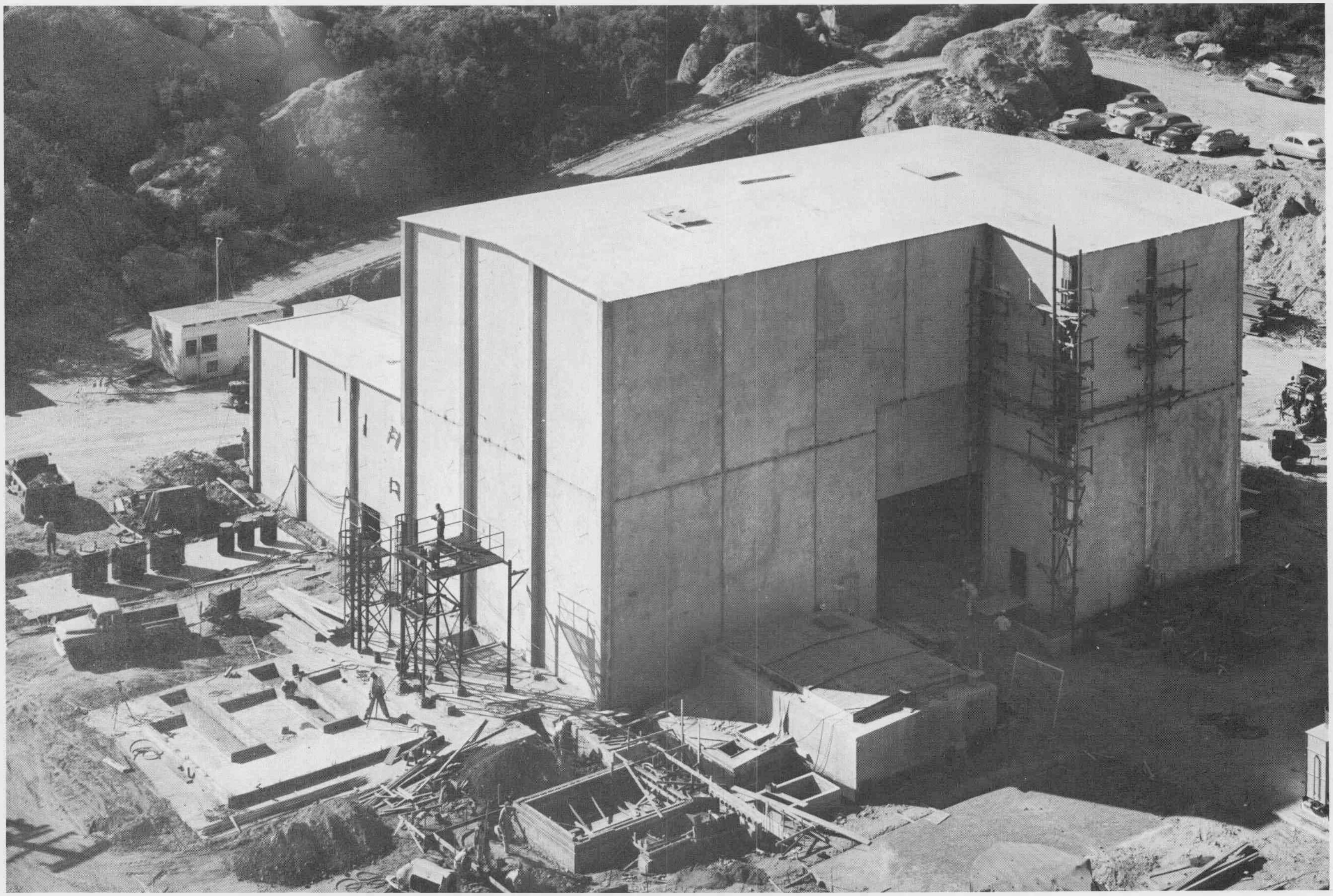


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Fig. 10. Building and Site Construction Progress (February-1956)

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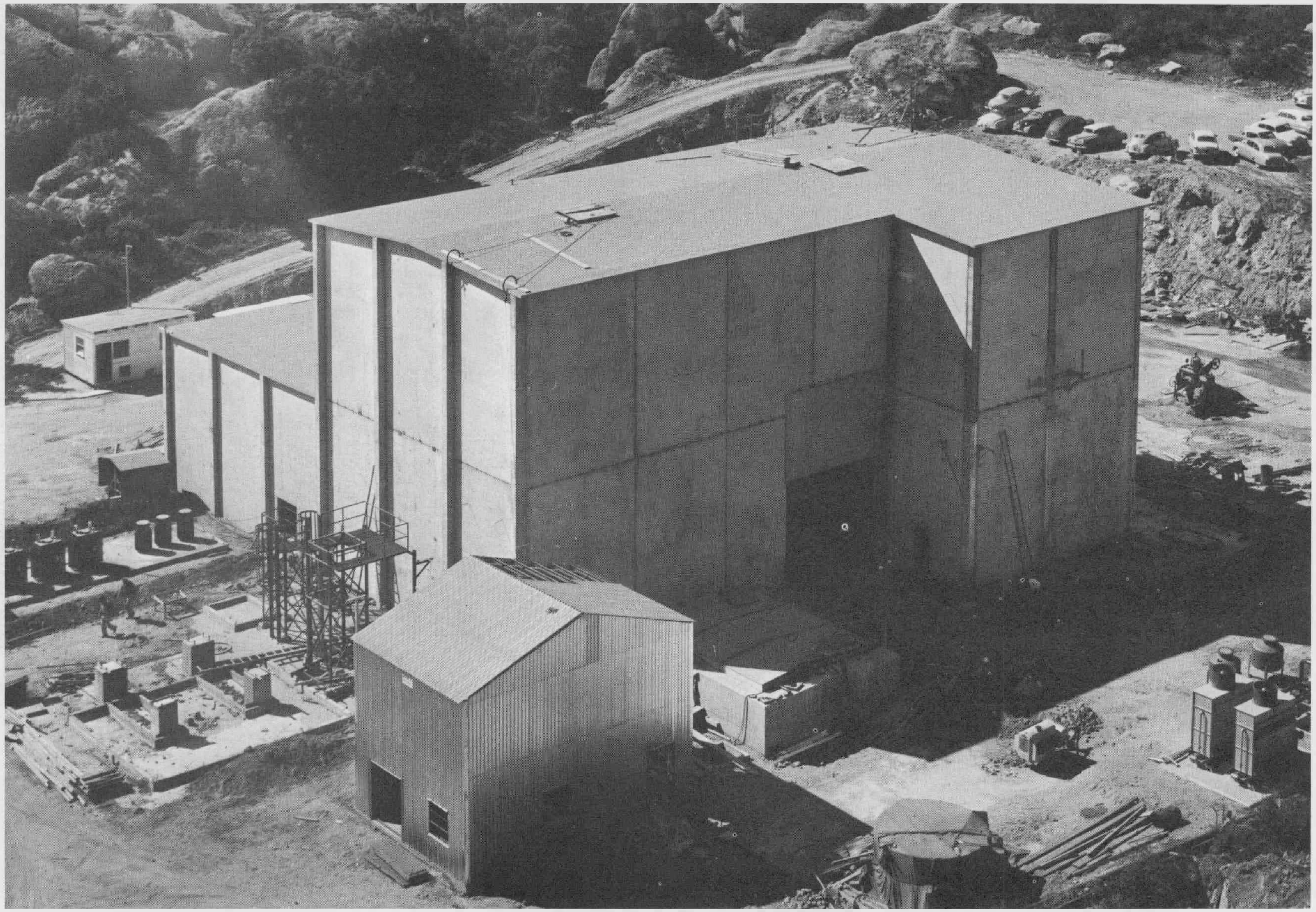

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Fig. 11. Building and Site Construction Progress (March-1956)

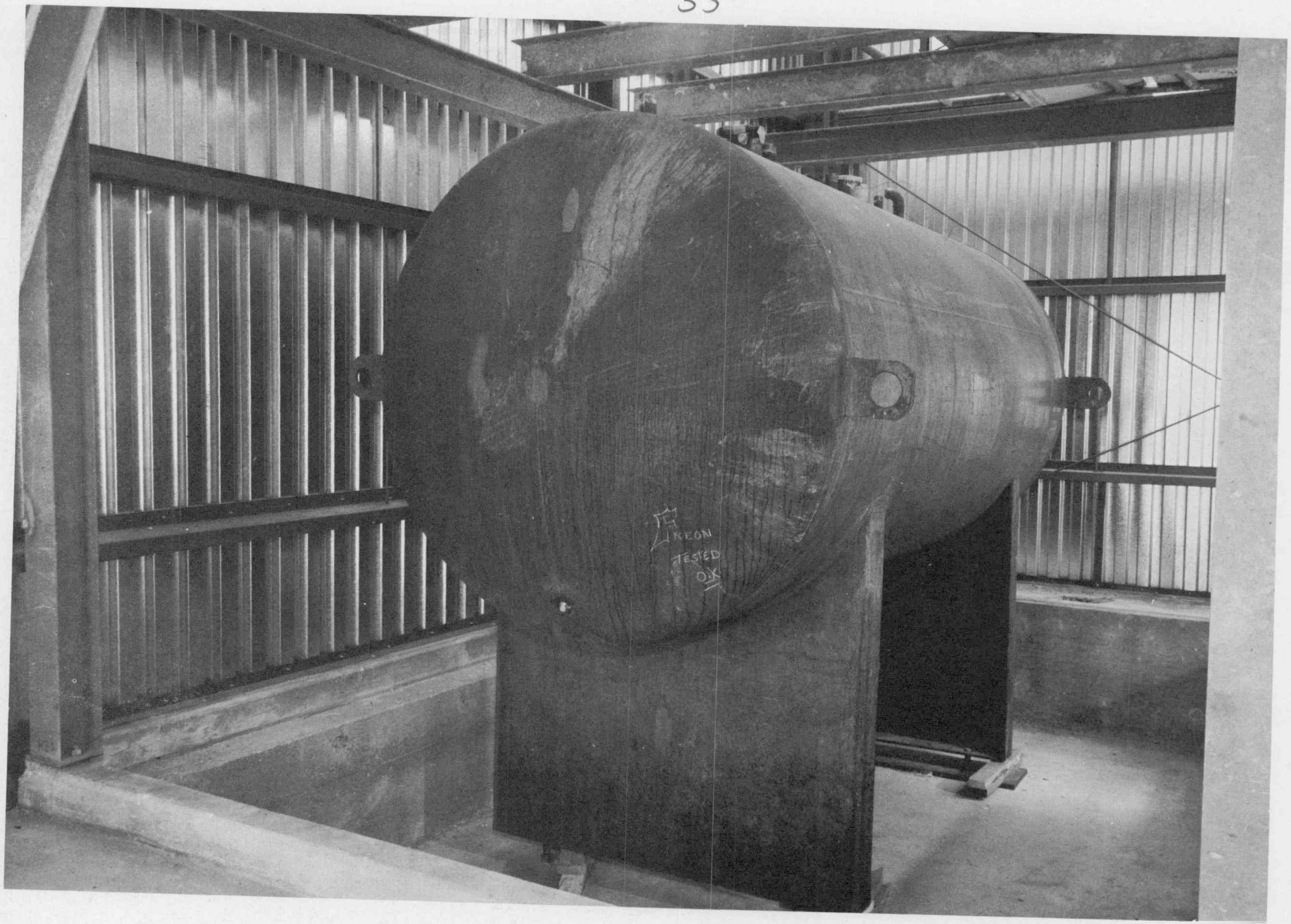

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Fig. 12. Secondary Sodium Fill Tank in Place

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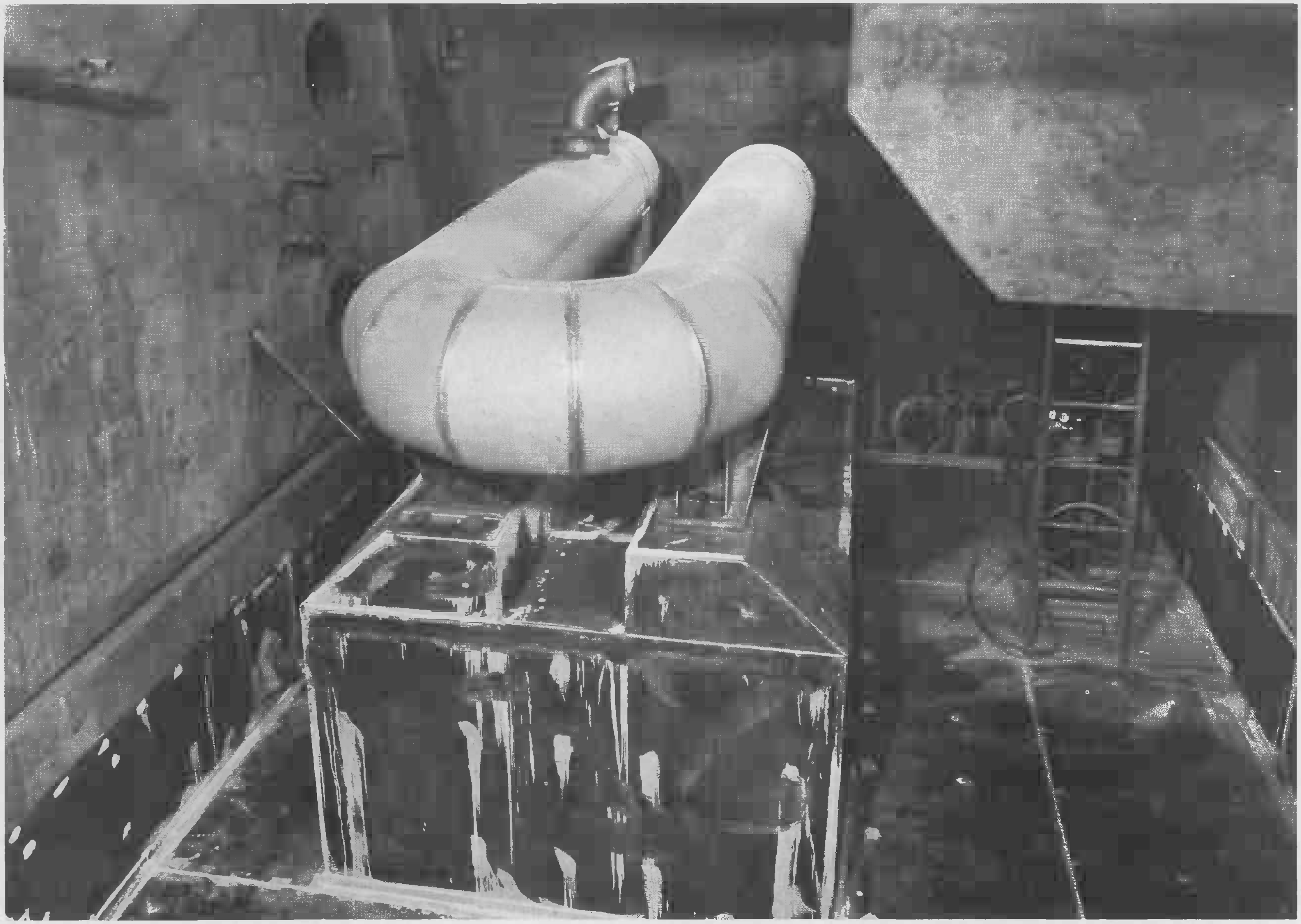

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Fig. 13. Main and Auxiliary Gallery Area

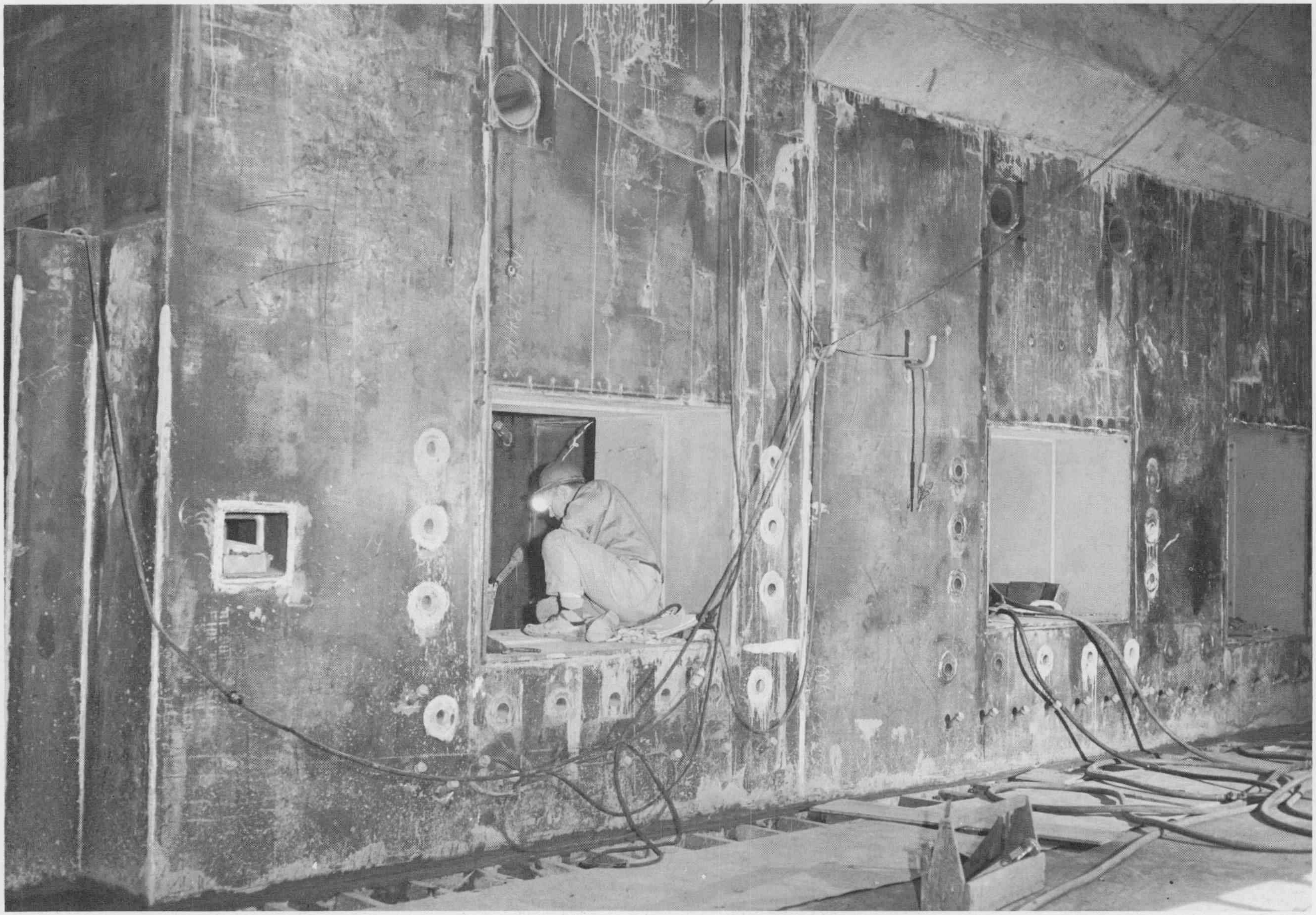

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Fig. 14. Hot Cell Construction

X. FUEL ELEMENTS

A. FUEL ROD FABRICATION (J. J. Droher and H. Strahl)

The fabrication of fuel rods has now attained a steady rate of output. To date 2268 slugs have been electropolished and outgassed. Of these slugs, 1764 have been processed into 147 fuel rods. This represents 50 per cent of the number of rods (including spares) requested for the initial loading. These rods have been stored in the vault in the Engineering Test Building (Fig. 15).

A fixture for leak testing the fuel rod in the vertical position was designed, fabricated and is now in use (see Fig. 16). This fixture utilizes a counterweight, which allows easy positioning of the fuel rod within the vacuum-tight test head. The bottom end cap is welded to the empty jacket tube. This weld is tested before the rod is loaded. Any indication of leakage with the helium mass spectrometer leak detector is cause for rejection. After loading with slugs, inserting the bond, and welding in the top end cap, the upper end of the loaded fuel rod is inserted in the test head and the weld of the top end cap to the thin-walled tubing is checked. Leak tests of fuel rods completed to date have detected nine welds which were not helium leak tight. This 5 per cent reject rate is considerably lower than that initially anticipated.

B. FUEL ROD DESIGN (M. Muller)

The concept of fuel handling and spent fuel processing in the SRE Hot Cell has been recently revised. It has changed to the extent that it is now desirable to disconnect easily a fuel cluster assembly in its entirety from the fuel hanger plug, rather than individual fuel rods from the cluster.

To allow for removal of an individual fuel rod the previous design called for the upper fuel cluster casting to be welded to the fuel plug hanger rod. If it became desirable to remove an entire cluster assembly this could only be effected by clipping the casting shank, at its smallest diameter, with a bolt cutter. This action would call for extensive rework of the fuel plug hanger rod to allow reattaching a new fuel cluster.

A design layout and stress analysis on this problem was completed in March 1956. Since the fuel element assemblies are still in the early stages of shop assembly, an advantageous change can be effected with a minimum of expense.

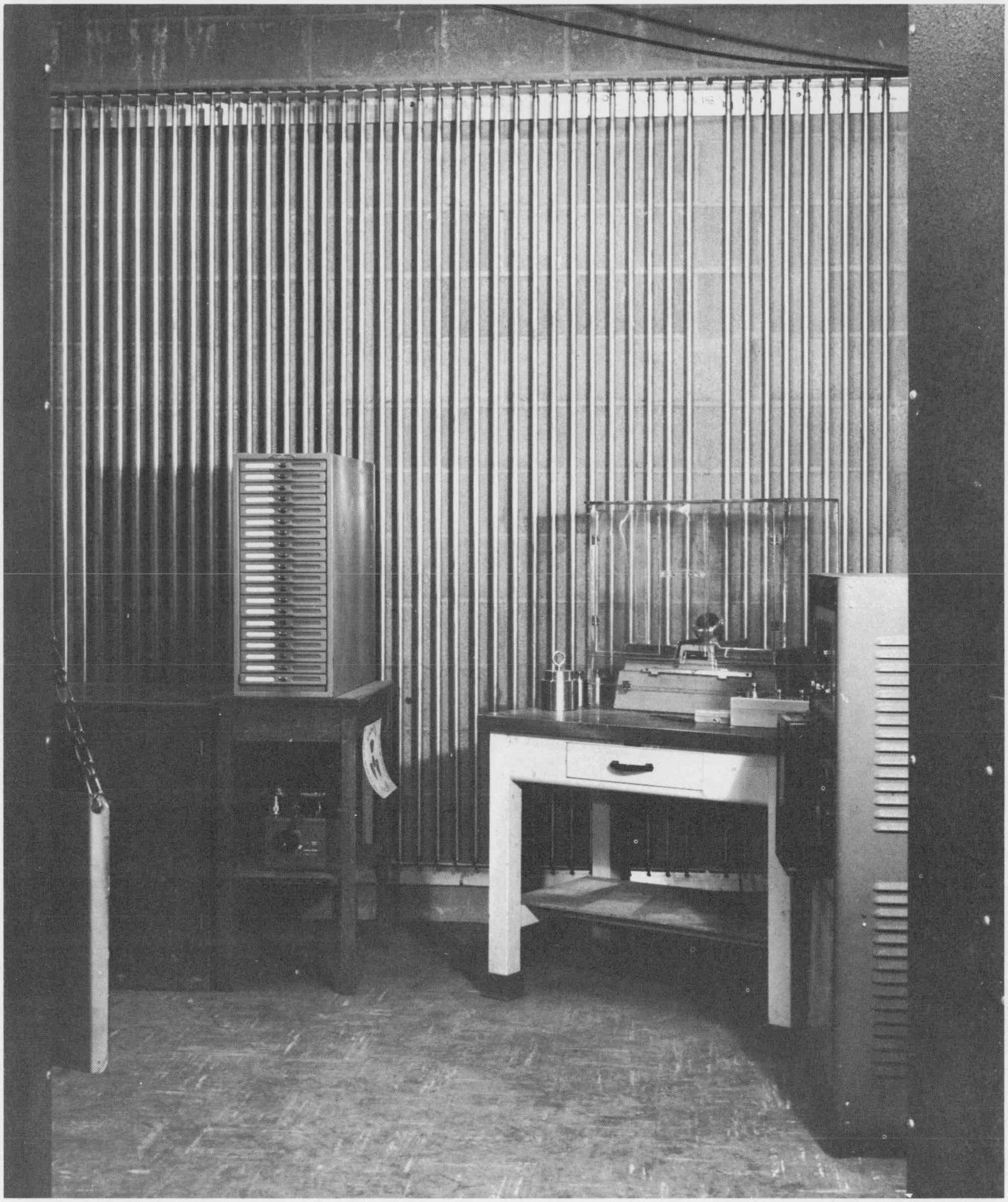

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Fig. 15. Processed Fuel Rods (stored in vault at ETB)

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Fig. 16. Fuel Rod Leak Testing Apparatus

In the new design (Fig. 17) a shear pin will be inserted through the hanger rod and cluster casting coupling point. A cotter pin will secure the shear pin. The shear pin is so designed that it is now the intended shear point in the event of hang up during fuel loading operations. If the shear point should break it will still be possible to seize and remove an entire cluster as a unit rather than attempting the recovery of individual fuel rods.

C. FUEL ROD TESTING (J. J. Droher)

A cluster of seven full scale NaK bonded rods was loaded with normal uranium and shipped to the Tower Facility for thermal cycling tests. The rods will be cycled in the process cylinder by withdrawing them from sodium at 1000° F into the 600° F helium atmosphere. The cluster will be inspected after 1, 10, and 100 cycles for warpage and elongation of the fuel rods, ease of disassembly, and corrosion.

To allow hot cell personnel to gain experience with the remote control handling equipment for the SRE hot cell, a second full scale cluster was loaded with normal uranium and shipped to the Raymer Facility. An additional cluster of shorter length rods was also fabricated for their use. This cluster of shorter rods was initially immersed in sodium at 1000° F for 100 hours. After removal, it was washed, soaked in water and dried. No difficulties in unscrewing threaded parts or removing close fitting sliding parts were encountered.

XI. MODERATOR CAN FABRICATION AND TESTING

A. MODERATOR CAN TESTING PROGRAM (J. A. Leppard and S. Facha)

The design of the moderator can head was modified as the result of a series of tests using the "Stress-Coat" technique. Four variations of the joint between the head and coolant tube were examined. Two similar joints indicated a noticeably lower stress value than the others and a compromise between the two was chosen for the final design.

Bend and hardness tests were performed on a number of trim ends of the production cans. It was noted that the longitudinal welds were slightly harder than originally anticipated, judged on the basis of development welding work.

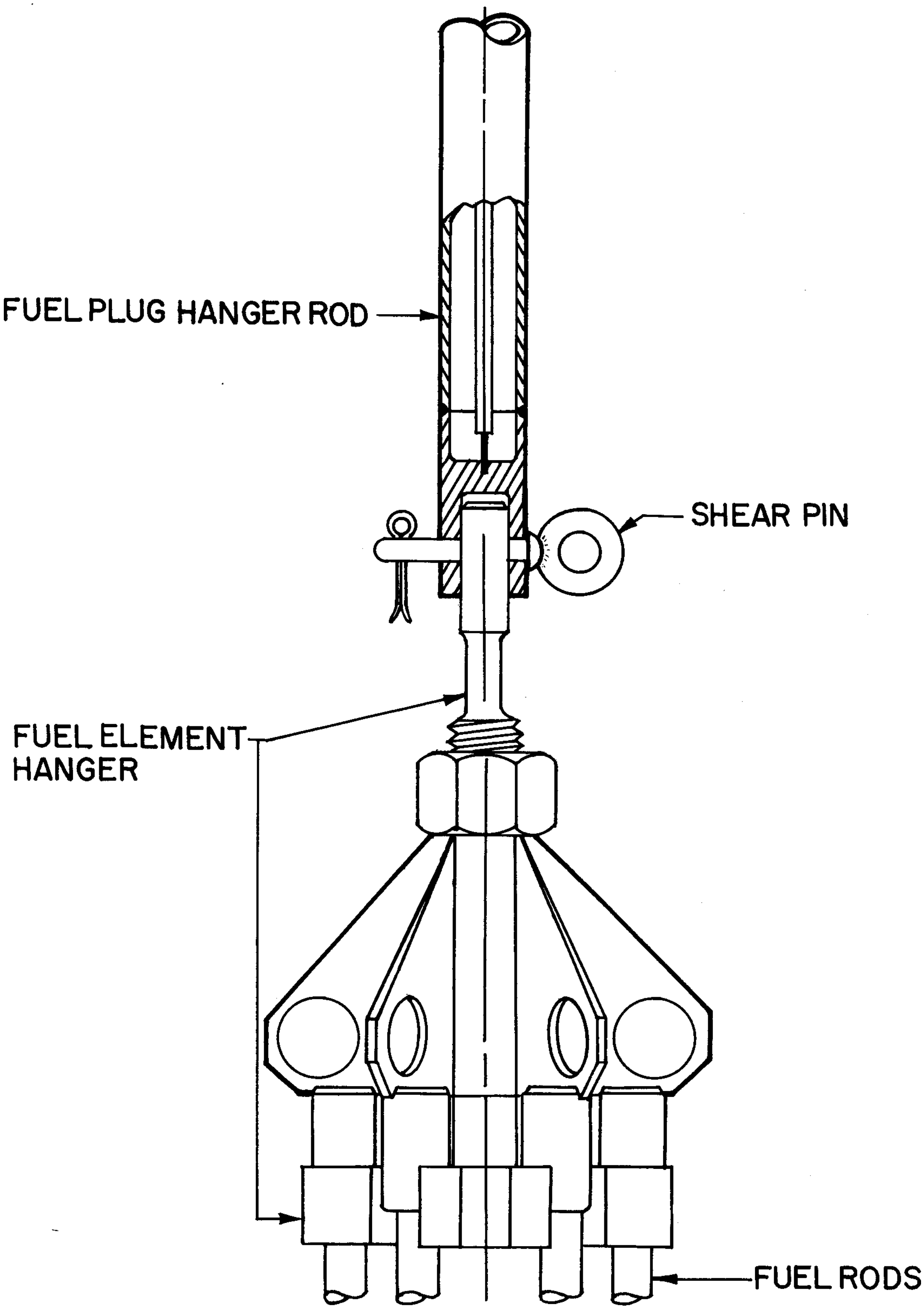

Fig. 17. Redesign of Fuel Cluster and Hanger Rod Coupling

Although very careful pre-weld cleaning procedures were followed, this effect was apparently due the inclusion of oxides in the welds.

A number of zirconium samples from various development test cans exposed to varying conditions of time, temperature, and atmosphere were bend and hardness tested. In all cases the welds and parent material were harder and more brittle after experimental tests. These same samples were vapor blasted and etched in a HF-HNO_3 pickle to remove 1 to 2 mils of stock. Subsequent welds performed on the samples were as ductile as welds made on original new stock. Details of an acceptable etching procedure have been worked out as a joint preparation for the closure welds made between the can head and hex shell.

Material intended for use in the moderator can heads was found to be defective, having overlaps, cold-shuts, and cracking, all due to a poor rolling job. Inspection by dye-penetrant test showed only 54 of a total of 187 heads to be crack free. The 54 heads were ultrasonically checked, with 50 proven sound and usable. Procedures for the stake welding of 100 mil stock have been developed as a part of a possible repair program for reusable heads. The remaining scrap plate on hand has also been ultrasonically tested to determine how much sound repair stock is available.

New zirconium ingots have been procured and are being prepared for rolling. The yield on this rolling will determine the extent of a repair program on existing heads.

B. MODERATOR CAN HEAD FLEXURE TEST (M. Tarpinian)

Based on the previous flexure tests and strain gauge measurements, modifications were incorporated into specimens No. 4 and 5. Since stress concentrations were very high at the head-to-tube weld joint, it was desirable to reduce the thickness of the can head in the vicinity of the weld, thus distributing stresses over a larger area. As a result, the head was modified as shown in Fig. 18 below.

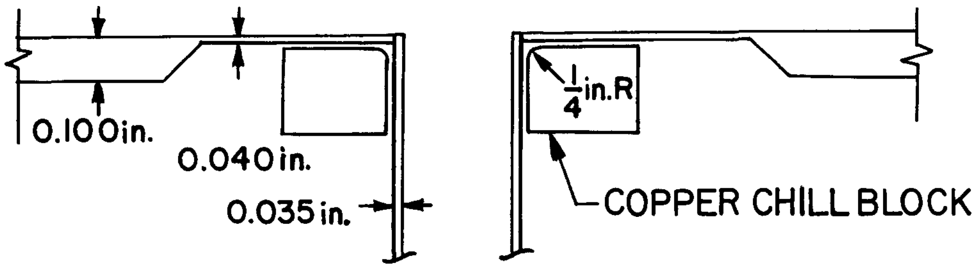

Fig. 18. Can Head Modification

During the welding process a copper chill block with a 1/4 in. radius was used to insure a more uniform fillet.

Can No. 4 with a graphite core, was tested under the following conditions:

Displacement of each head	0.09 in.
Temperature	975° F
Pressure differential on head	2-1/2 psi
Environment	Helium

To save time, the test was run without sodium. The test was terminated after 3080 cycles. There were no indications of failure in the can as determined by helium leak checking. Also, no deformations were evident. To check for superficial cracks and crazing, a dye-penetrant test was made with negative results. In general appearance, a slight blackening of the outside surface was apparent.

Based on revised calculations, a displacement of 0.07 in. per head was believed to be more realistic. Therefore, the test of Can No. 5, also with a graphite core, was set up to produce this movement. Failure occurred after 226 cycles. Subsequent examinations showed that the graphite core was cut oversize, thus not allowing sufficient clearance for the necessary displacement. The last 0.05 in. of movement was made with the graphite blocked up against the heads. On the basis of these observations, this should not be considered a representative test.

C. MODERATOR ELEMENT SHEATH STRESS ANALYSIS (W. F. Anderson)

Stress analysis was performed on the zirconium moderator element sheath to determine its suitability under a variety of loading conditions.

Flexibility constants were determined for the top head, the bottom head, the process tube and the outer sheath. These constants were combined in simultaneous equations for the rotation and translation of the points where these elements are connected. Solution of these simultaneous equations yielded values for the moments existing at these connection points and the axial force in the process tube. Stresses in the elements at these points were calculated from these moments and the axial force. These calculated stresses are shown in Table II.

Certain assumptions, considered important for this discussion, were made to provide a basis for the calculations.

TABLE II 45

CALCULATED STRESSES FOR ZIRCONIUM MODERATOR ELEMENT SHEATH

Loading Condition	Bending Stress Location: Top of Sheath psi	Top Head (at Sheath) psi	Top Head (at Process Tube) psi	Top of Tube psi	Bottom of Tube psi	Bottom Head (at Tube) psi	Axial Stress in Tube psi
Scram + Thermal Constriction of Tube	17,900	2,180	4,230	34,600	19,800	3,840	68 Compression
Scram	18,500	2,260	2,510	17,400	17,000	3,290	92 Compression
Static Sodium Pressure	4,850	590	-2,770	-15,000	4,850	920	276 Compression
Scram + Thermal Constr. + Fluid Flow Drag + Weight	17,100	2,100	3,930	32,600	24,000	4,660	55 Compression
Scram + Fluid Flow Weight	17,700	2,170	3,860	15,400	21,100	4,110	79 Compression
Sodium Pressure + Fluid Flow + Weight	4,100	500	-2,020	-16,500	685	138	264 Compression
Can Head Test Flexure Only. 0.090 in. Displacement	17,900	2,200	6,970	37,800	--	--	234 Tension
Can Head Test Pressure Only	26,000	3,240	5,500	29,600	--	--	--
Can Head Test Total (0.090 in. Displacement)	43,900	5,440	12,470	67,400	--	--	234
Can Head Test Flexure Only 0.070 in. Displacement	13,900	1,710	5,420	29,400	--	--	--
Can Head Test Total (0.070 in. Displacement)	39,900	4,950	10,920	59,000	--	--	--

It was assumed that the tube and outer sheath joined the heads at sharp right angles. The stresses calculated for the elements at this junction and noted in Table II would exist at the point where the center lines of the tube and head intersect. For the tube in particular, this stress is a localized phenomenon and decreases rapidly with distance down the tube, decreasing to zero in 0.25 in. In actuality a percentage decrease could be expected in this stress since an appreciable fraction of this 0.25 in. down the tube is reached before the tube wall is unsupported by head thickness or welding bead. Rounding the corner at the juncture of the tube and the head with a radius greater than 0.25 in. could appreciably relieve the stresses in the tube. The solution of the elastic equations for a rounded corner is being attempted at this date. These solutions should serve as a guide to further development and test work in addition to helping predict the expected life of these zirconium cans.

It was also assumed that this system behaves elastically. If stresses exceed the short time yield stress, or if stresses of longer duration cause creep, there is a possibility that some relief of these stresses will occur. The junctions of heads and tubes would become plastic hinges and a general redistribution of stress would occur. This would result in a decrease of stress at the plastic hinge to the yield point stress and an increase in stress at other points. This plastic behavior depends on the material involved remaining a ductile material. If considerable embrittlement or hardening of the material involved is to be expected, design should be based on stresses or change of stress calculated by elastic theory.

The Loading Conditions considered were as follows:

1. Static Sodium Pressure: The pressure due to the weight of molten sodium exerted against the top and bottom heads.
2. Static Sodium Pressure plus Fluid Flow Drag plus Weight: Same as 1 plus the drag force on the process tube exerted by sodium moving past plus the weight of the tube itself.
3. Scram: Same as 1 plus the force due to thermal shortening of the process tube when the temperature of the sodium coolant drops during a scram or shutdown.
4. Scram plus Fluid Flow Drag plus Weight: Same as 2 plus the effect of thermal shortening of the process tube.

5. Scram plus Thermal Constriction of Tube: Same as 3 plus the force due to the decrease in the diameter of the process tube by thermal contraction when the temperature of the sodium coolant drops during a scram or shutdown.
6. Scram plus Thermal Constriction plus Fluid Drag plus Weight: Same as 4 plus the effect of thermal constriction of the process tube.

Values of the strength properties of zirconium taken from AI date yield the following for 1000° F.

Short Time Ultimate Stress			12,500 psi
Short Time Yield Stress			7,000 psi
Creep Rupture Stress	0.3	hr	12,500 psi
	1.0	hr	9,500 psi
	10	hr	6,800 psi
	100	hr	5,000 psi
	300	hr	4,000 psi
Extrapolated	1000	hr	3,500 psi
Extrapolated	10,000	hr	2,500 psi

Comparison of the calculated stresses from Table II, with the strength data allows the following conclusions.

1. The process tubes of the present design are overstressed.
2. The stresses in the heads seem quite resonable under normal operating conditions (Loading Conditions 1 and 2). Under scram conditions the change of stress does not exceed the short time yield strength and suitable performance should be expected.
3. The can head tests to date have stressed the heads and the tube beyond those stresses which might be expected in service. Stress changes during the test exceed the stress change which might be expected during a scram (thermal constriction of the process tube neglected).

Some verification of the results of this analytical effort has been found in the results of the can head tests.

1. Failures occur in the process tube at the intersection of the tube and the head, the point of calculated maximum stress.
2. Photomicrographs show an equal number, if not a preponderance of cracks, of the type which lead to failure, starting from the graphite side of the tube where oxygen content of the sodium could have no effect.

XII. HEAT TRANSFER

A. SIX-INCH WEDGEPLUG VALVE TESTS (J. C. Flint)

The 6 in. oval port test valve was examined for pitting after its second run at 1200° F. The plug was quite clean and was not pitted. The body had a few new pits in the inlet and outlet throats but no new ones on the valve seat. This indicates that 95 to 98 per cent of the pits to be expected on a casting can be brought out in the first sodium soaking.

The 6 in. round port production valve soak showed extensive color change and a quantity of foreign material in several places. Figure 19 shows the seat surface after the sodium soak but before clean up. Figure 20 is the same area after two water baths and one steam bath.

Figure 21 shows the plug immediately after the sodium soak. Considerable black deposits are evident, but all of them came off in the steam or water baths.

Figure 22 is a view of the inlet pipe on the 6 in. round port production valve where a large area of carbonaceous material was found.

These tests resulted in the decision to soak all four SRE valves for 100 hours at 1200° F before final assembly into the reactor. All valves will be relapped after the soak test.

B. SODIUM LEVEL INDICATOR (J. C. Flint)

1. Crescent Level Gage - The second Crescent Level Gage was delivered for leak testing and was found to leak in the same place as the first unit. The unit was returned to the manufacturer and the contract was dropped due to the failure to meet required specifications and delivery date.

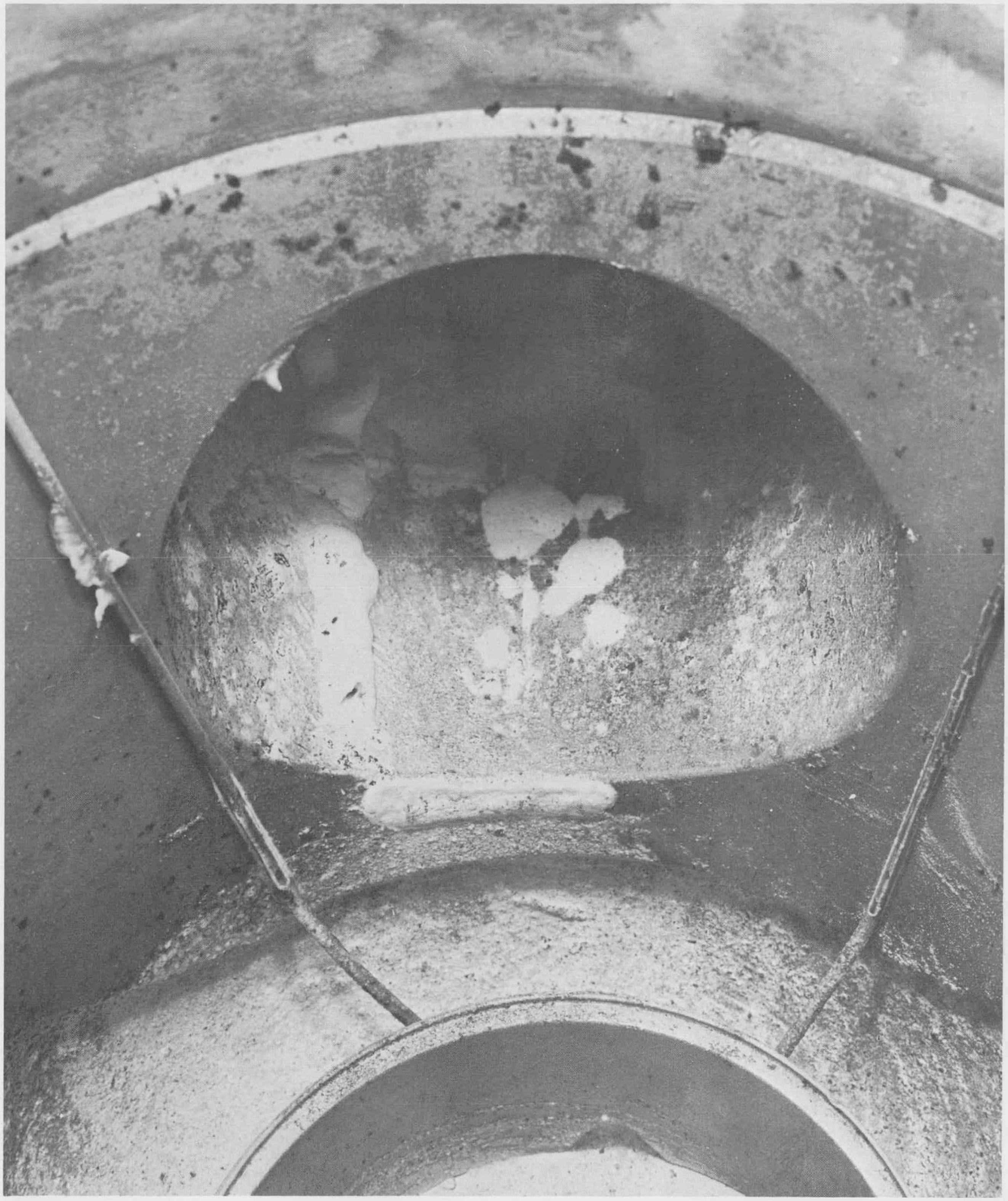

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Fig. 19. Seat Surface of the 6-in. Round Port Production Valve after Sodium Soak -- before Cleanup

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Fig. 20. Seat Surface of the 6-in. Round Port Production Valve after two Water Baths and one Steam Bath

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Fig. 21. Plug of the 6-in. Round Port Production Valve after the Sodium Soak

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Fig. 22. Inlet Pipe of 6-in. Round Port Production Valve -- showing large Deposits of Carbonaceous Material

2. Precision Probe Coil - The experimental precision probe coil, reported in the previous quarterly[3], has been in operation for three months on intermittent duty at temperatures up to 950° F. Additional coils of this type have been made and tested. All the test coils failed at coil temperatures of 550° F in an hour or less. It is therefore evident that the coil measures 1000° F sodium level only if the coil temperature is kept below 500° F.

3. Sodium Level Coil - The present sodium level coil development program has four objectives.

1. To test high temperature wire plus insulation combinations as well as insulating tapes and cloth for use in the SRE at 1000° F.
2. To obtain design data for modification of the ANL solenoid type coil for use in the SRE as long continuous reading level gages.
3. To construct a sodium tank, with thimbles, to test and calibrate the SRE units up to a length of 84 inches.
4. To obtain alarm coil design data for the SRE.

Three combinations of wire and insulation have been tested in air, in MgO powder and a protective layer of fiberglass tape in air at temperatures up to 1300° F. Two types of fiberglass insulated No. 22 copper wire have been tested. The first is Thermo-Electric No. DTP (0.048 to 0.50 in. diameter overall); the second is Varflex Co. No. 22 stranded copper unsaturated fiberglass covered (0.045 to 0.048 diameter overall). A small quantity of Refrasil insulated No. 22 solid silver wire was available and was tested. Refrasil covered No. 22 copper wire has been ordered and will be tested upon arrival.

All three insulated wire combinations tested held up well, insulation wise, in tests to 1200° F in air. The fiberglass insulations lost their original color and turned white after being at 1200° F for one day. At temperatures over 1200° F or test times over one day at 1200° F the insulation color changed to a light amber or light brown. After these tests the insulation did not crack with bending, provided the bends were not too sharp. The insulation melts somewhere between 1200° to 1750° F; it becomes rigid and brittle and will not yield to bending.

The copper wire in all cases did oxidize enough to give an indication in the resistance measurements and appeared to be in the dead soft condition after the first exposure to 1200° F. The silver wire showed far less oxidation than the copper.

Figure 23 shows: the Refrasil "A", Vargloss fiberglass "B", Thermo-Electric fiberglass "C", and Refrasil covered wire sample from ANL test pieces "D". The straight pieces on top are the unheated samples. In this case all samples were tested in MgO powder at 1000° F.

Long coils have been made in two general sizes according to their length to diameter ratio. The "long coils" (l/d=8 or more) all have lengths of 18 in. and were wound on 1-1/4 in. diameter forms except for the thin tube core which was 1-3/4 in. diameter. Core materials included alundum tubes, solid "magnetic" steel rods and a 1-3/4 in. OD by 0.062 in. wall "magnetic" steel tube. All the coils were wound with Thermo-Electric Type DTP, No. 22 solid copper wire insulated with unvarnished fiberglass with an OD overall of 0.050 in. With an OD of 0.050 in., 20 turns/inch, and two layers, all the coils had 720 turns. Figure 24 shows one of the solid core coils before heating.

Materials for core sizes of the type specified in the SRE design have been ordered. The following tests have been made:

1. Observing the coil resistance and inductance at room temperature.
2. Inductance vs Temperature to 1200° F (measured on a G.R. 650A Impedance Bridge) with no sleeve.
3. Inductance vs Sleeve position, or Level. An aluminum tube, 3 in. OD and 1/8 in. wall thickness, was used. A few tests were made with a copper tube 3 in. diameter and 1/16 in. wall. Additional material is on order for tests using various tube diameters and wall thicknesses.
4. Impedance vs Sleeve Position at 60 cps, 0.5 amp, 1.0 amp, 1.5 amp and 2.0 amp starting currents. Tests were started with the coil enclosed by the sleeve (position of maximum current and minimum impedance). Current was then recorded relative to sleeve position. In all 60 cps tests a Sorensen Regulator was used, to reduce line voltage fluctuations.

4. Test Tank - A 7 ft long sodium tank has been constructed of 8 in. Schedule 40 Type 304 stainless steel. This tank has three thimbles, a 1 in. Schedule 40 pipe, a 2 in. Schedule 40 pipe and 1.9 in. ID Schedule 10 pipe. The test tank and its dump tank are capable of temperatures up to 1200° F, so that level at constant temperature, can be raised or lowered as desired. Figure 25 shows the test apparatus before insulation and assembly.

5-5

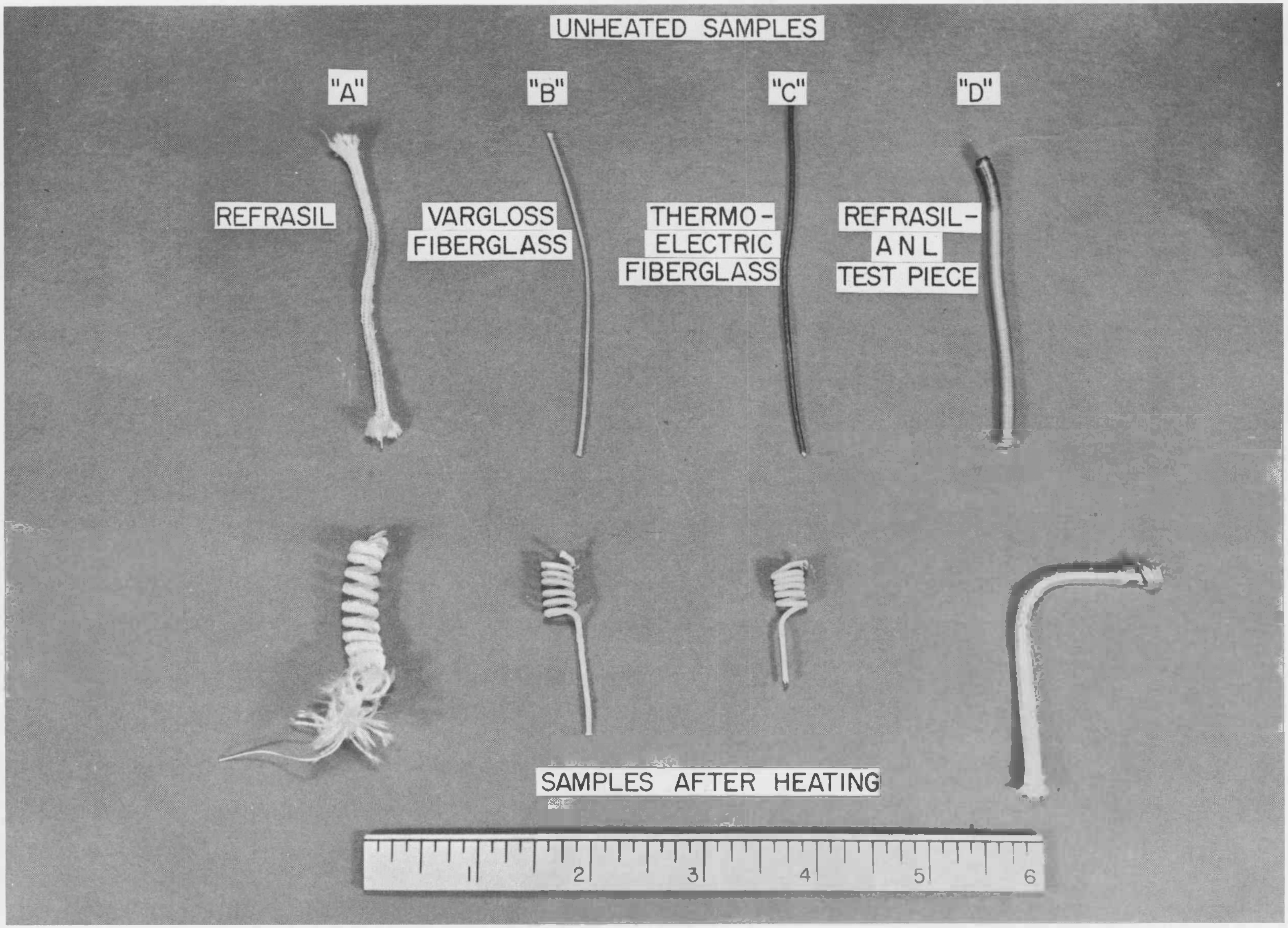

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Fig 23. Core Wires before and after Heating

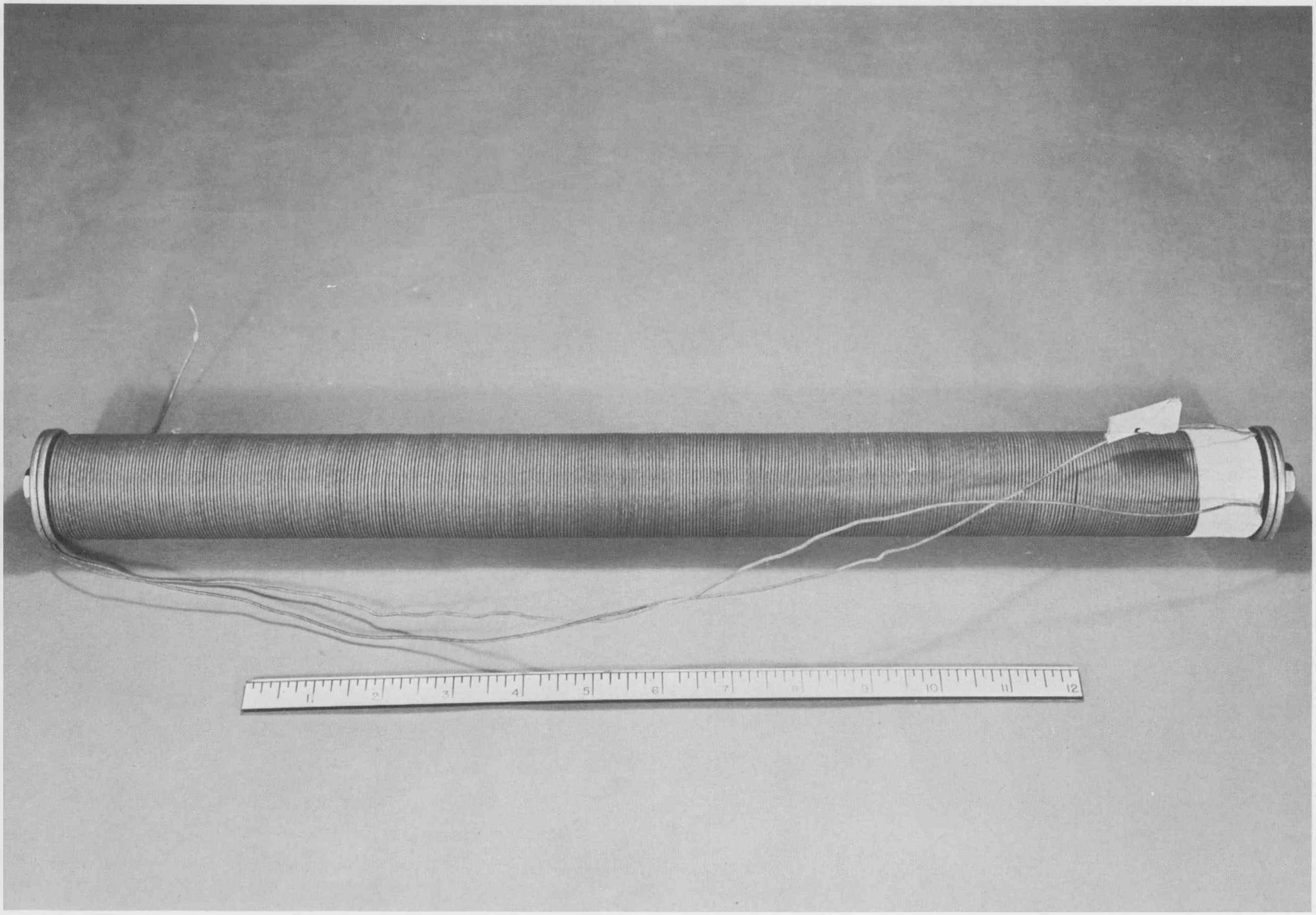

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Fig. 24. Long Solid Core Coil before Heating

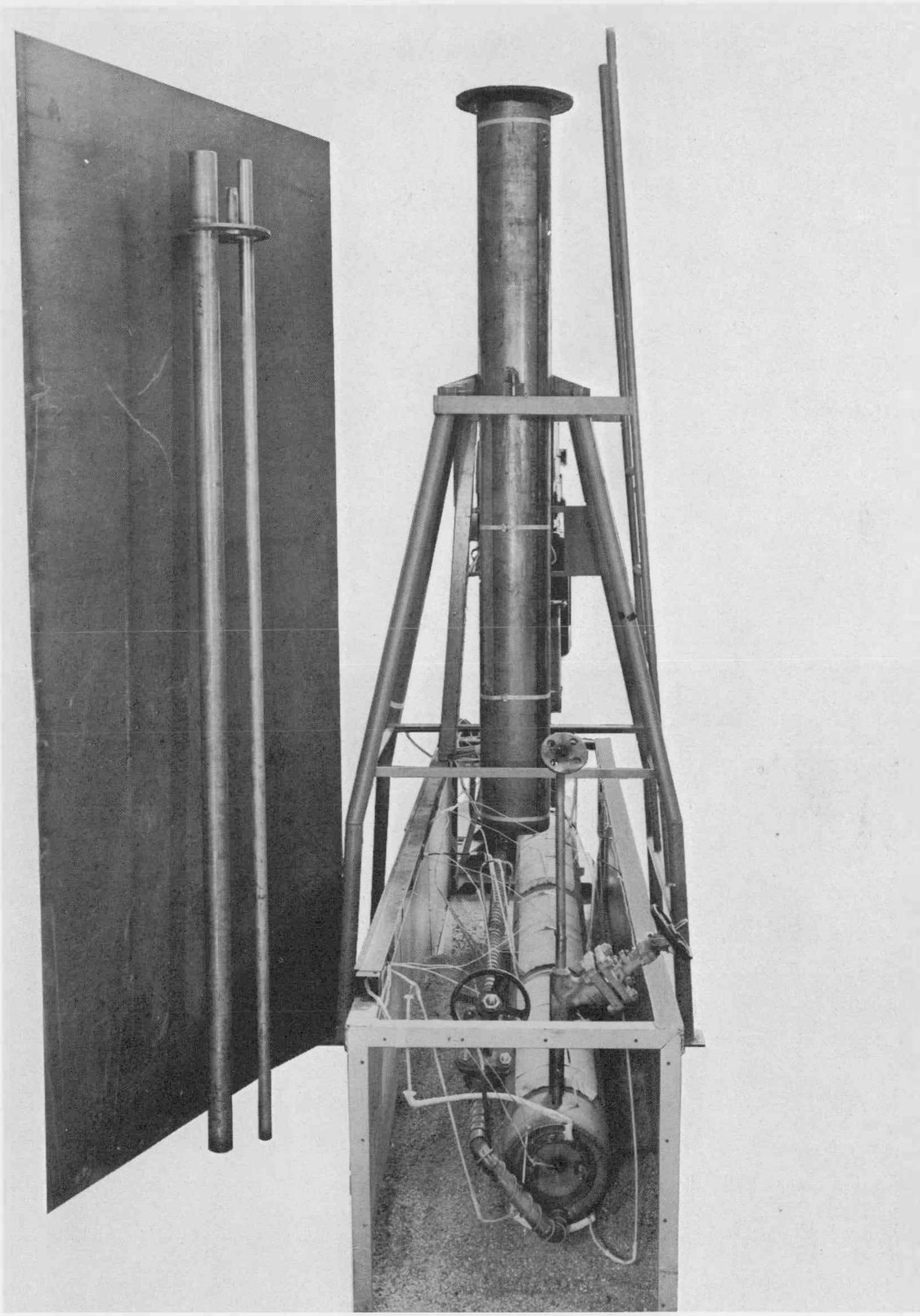

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Fig. 25. Coil Testing Apparatus before Insulation and Assembly

A few short coils have been constructed and tests made with coils operating in air at 1200° F. Figure 26 shows two of the coils after the tests. The center core is mild steel, similar to the core of coil "A".

C. DISTILLATION TYPE SODIUM SAMPLER (S. R. Rocklin)

A distillation type sodium sampler was constructed and operated experimentally. In this type of sampler the sodium is allowed to enter an evacuated chamber and fill a nickel cup. The cup is then heated until all the sodium is distilled, leaving the sodium oxide in the cup. The cup is then removed and its contents chemically analyzed for oxygen. Test results indicate that this type of sampler can be used to detect oxygen concentrations in sodium as low as 8 ppm.

An improved distillation sampler is being constructed and will be used to check plugging indicators.

D. PLUGGING INDICATOR TEST LOOP (M. Nathan)

Theoretical calculations to predict plugging meter sensitivity as a function of number of holes in the valve plug, hole size, and magnitude of bypass flow were initiated and completed in the early part of the quarter. The plugging meter test loop at the Santa Susana Tower Facility was modified to investigate the conclusions formulated by theoretical analysis. However, insufficient data were obtained due to failure of pressurized standpipes used to measure sodium pressure drops. One Callery and two General Electric pressure transmitters were obtained and installed to measure the system pressure drops. By the end of the quarter a new test series had been initiated.

E. PUMP SHAFT FREEZE SEALS (R. Cygan and R. W. Atz)

Three additional shaft freeze seal designs have been tested in the 6 in. pump loop. These included a two region 3/8 in. coil type, a two region long design, and the SRE prototype. All seals performed satisfactorily with a tetralin flow of 4 gpm (total for two region designs). Cooling loads for all designs were about 4 kw at 1200° F.

A tetralin leak from the shaft seal occurred due to an improperly installed thermocouple. This resulted in tetralin contamination of sodium. The results were the same as for oil leakage i.e., pressure build-up in loop due to vapori-

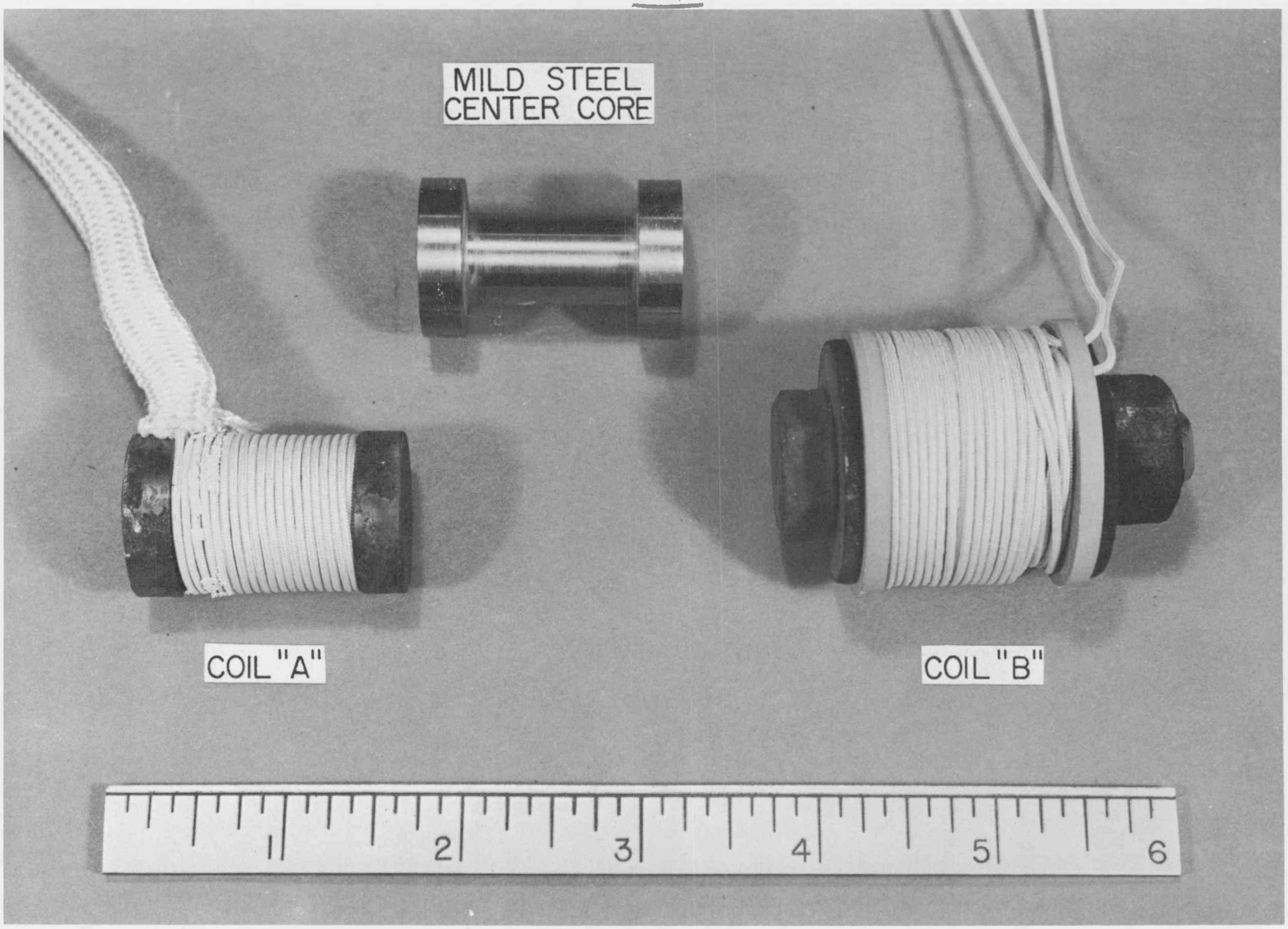

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Fig. 26. Short Coils after Test (coils operated in air at 1200° F)

zation, formation of carbon, and increased melting point of the sodium. This residue is not removed from the sodium by cold trapping or filtering.

F. SIX-INCH ASTM - A157 - C6 PUMP (R. Cygan and R. W. Atz)

After dismantling the six-inch pump loop for modification, severe corrosion and/or erosion of the pump case on the suction side was found. The largest and deepest pits were found on the bottom of the suction pump case on either side of the rear edge of the suction splitter. In the worst case up to 80 per cent of the original metal had been removed. Other damaged portions were concentrated in horizontal lines at or near the horizontal mid plane of the suction line. The beginning of these grooves was near the center of the flowmeter magnet and in line with the maximum magnetic flux produced by this magnet. Depth of corrosion in the ASTM-A157-C6 material was several times greater than in the adjacent 304L metal. Pits were found to be deeper in the heat affected zone where the 304L and the ASTM-A157-C6 were welded together.

A small mound of residue was found below the upstream edge of the suction nozzle splitter vane. After being chipped free it was found to contain mostly iron fragments in a clay-like matrix containing some free carbon.

Metallographic examination of the damaged areas as well as undamaged portions has been made. Both showed evidence of carburization with a depth of 0.003 in. on the damaged area and 0.001 in. depth in the parent metal. The cause for the peculiarly localized attack is not known but is under investigation.

G. STATIC POT SHAFT SEAL TESTS (A. N. Gallegos)

A static seal apparatus has been constructed. This consists of a sodium container with immersion heaters mounted under the mocked up pump stuffing box. A pump shaft mounted in the pump bearing bracket is motor driven. The sodium temperature below the seal has been cycled one hundred times between 500° and 1000° F. The prototype SRE seal thus tested showed shrinkage on the inside diameter. At the top of the seal this was 0.010 in. whereas the bottom (exposed to greater temperature changes) decreased 0.004 in. on the diameter. The external dimensions (fit into stuffing box) showed no change.

H. AUXILIARY PUMP TESTS (W. N. Bley and R. Cygan)

The two-inch pump loop has been completed and initial operation was begun. Remachining was necessary to increase the radial labyrinth clearances from 0.012 in. to 0.020 in. Cooling requirements of the shaft seal, using water as a coolant, were found to vary from 0.5 kw to 20 kw. Shaft speed did not seem to affect the cooling load. The cooling load of the shaft seal increased approximately 0.5 kw for each 200° F increase in temperature.

I. SRE FUEL ELEMENT ORIFICE CALIBRATION (C. R. Davidson)

Figure 27 is a schematic drawing of the piping installation of the orifice calibration apparatus.

This experimental data accumulated in the previous quarter[3], (the process of calibrating the orifice plates for the SRE fuel element), was translated into the form of pressure drop across the test section as a function of the water flow rate through the test section (see Fig. 28). From this data the discharge coefficient, corrected for frictional pressure drop in the test section, was computed as a function of the Reynolds' Number. The Reynolds' Number was based on the inner diameter of the test section, rather than the usual practice of using the orifice diameter, to avoid using an equivalent diameter necessitated by the particular configuration of the SRE orifice plate. Discharge coefficients ranging from 0.79 to 1.00 were obtained for Reynolds' Numbers varying from 2×10^4 to 3×10^5 (see Fig. 29).

Each SRE orifice plate actually has seven orifices, six holes in the plate and the fixed annular area between the pipe wall and the orifice plate. The annular area is equal for all the orifice sizes and only the flow area presented by the six holes is variable. Because of the thickness of the orifice plate (0.250 in.), it more nearly resembles a short tube than a conventional knife-edged orifice. This was confirmed by the higher discharge coefficients obtained. It was especially apparent in the case of the plate with the 0.250 in. holes where the hole diameter is equal to the orifice plate thickness. Coefficients based on sudden contraction - sudden expansion calculations for this range of Reynolds' Numbers vary from 0.88 for the 0.250 in. diameter holes to 1.13 for the 0.625 in. diameter holes. Values of around 0.8 for the discharge coefficient of a short tube would compare with previous experimental results in the field.

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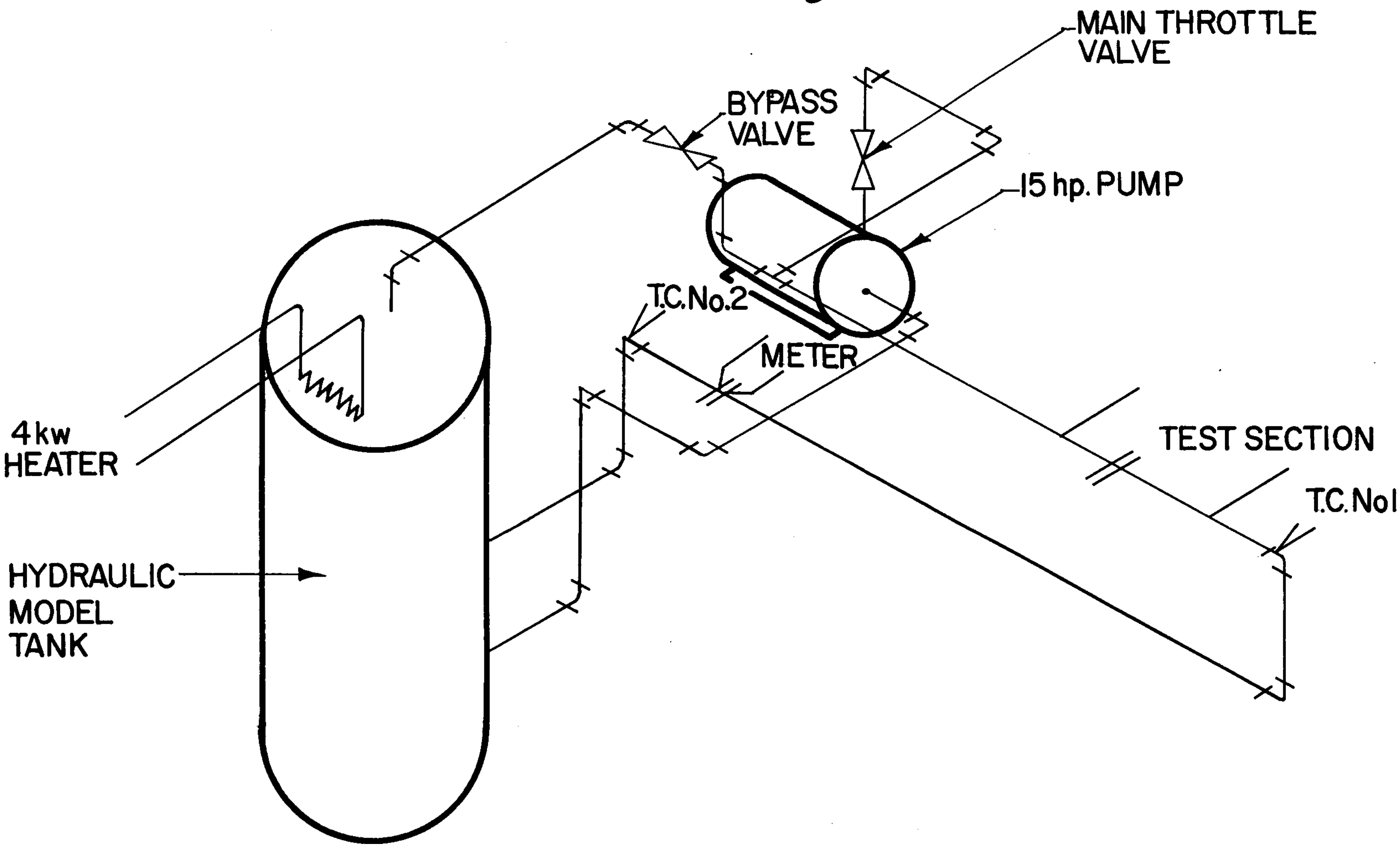


Fig. 27. Piping Installation Orifice Calibration Apparatus

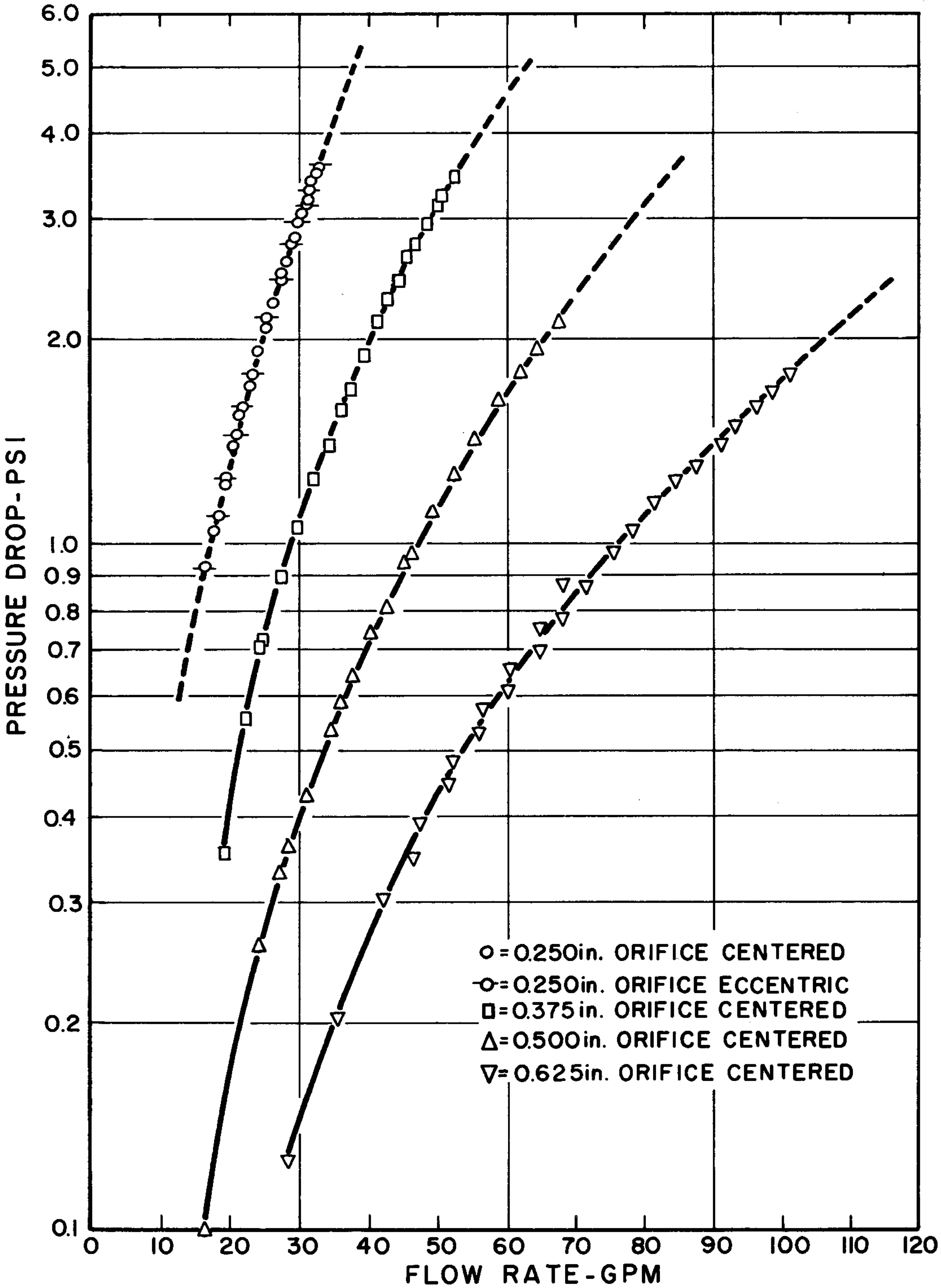


Fig. 28. Total Pressure Drop across Test Section vs Flow Rate for SRE Type Orifice Plate

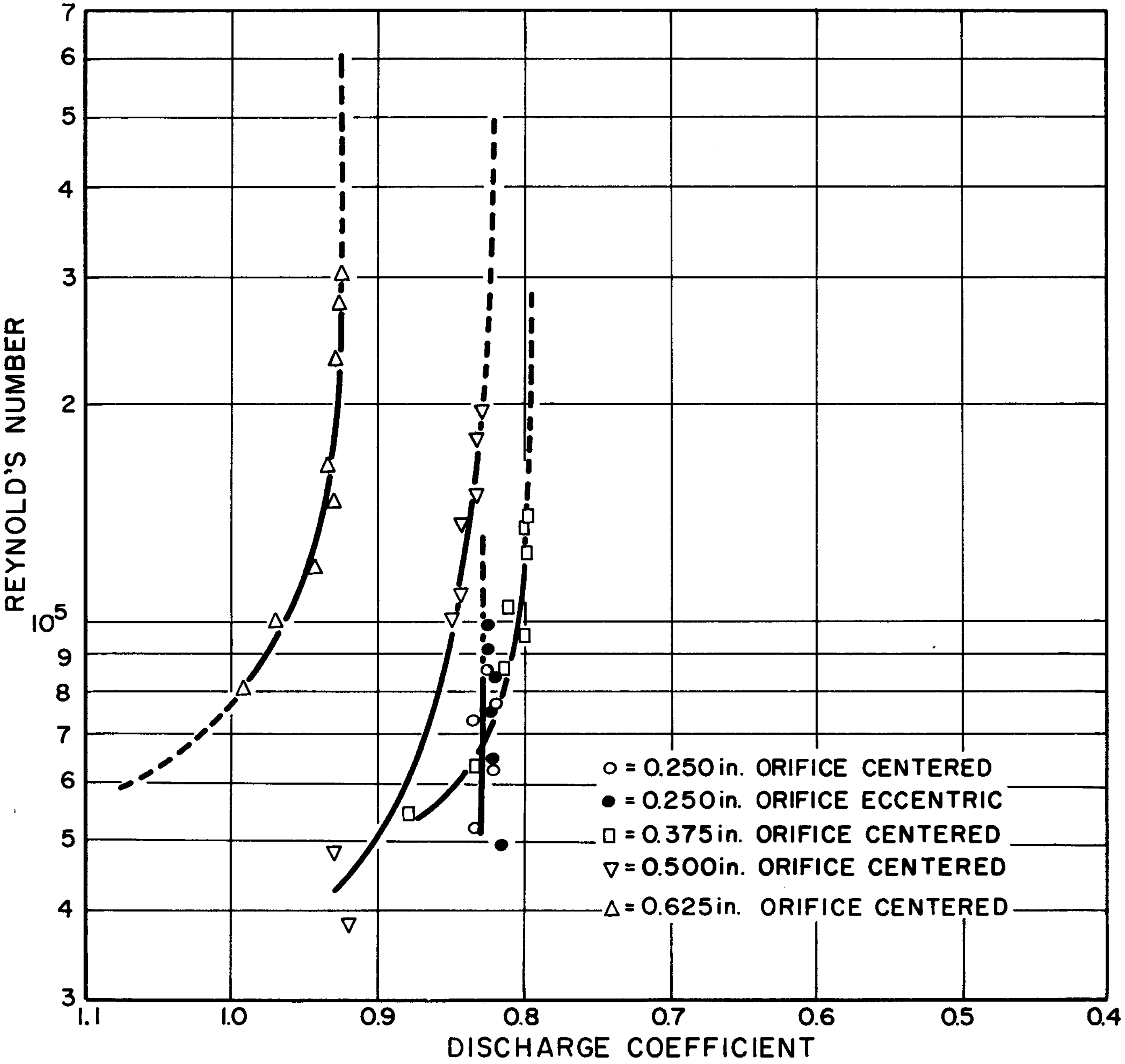


Fig. 29. Discharge Coefficient vs Reynolds' Number for SRE Type Orifice Plate

J. SRE VALVES (H. E. Richter)

1. Blocking Valves - The six inch Round Port Valves have been sodium soaked and are ready for final assembly. The six inch Rectangular Port Body Casting has been repaired and is awaiting acceptance by Inspection. The two inch Rectangular Port Valve is ready for sodium soaking. The two inch Round Port Valve is being repaired by Atomics International.

K. HEAT EXCHANGERS (H. E. Richter)

1. Main Air Blast Heat Exchanger - Revised tube bundles drawings have been received, checked, and released. The supplier of the finned tubes will start production on April 4, 1956.

2. Intermediate Heat Exchangers - These units have been installed.

L. SODIUM PUMPS (H. E. Richter)

The vendor's parts were reworked for greater wear ring clearances.

XIII. INSTRUMENTATION AND CONTROL

A. INSTRUMENTATION (W. Spalding)

1. Reactor Area - Release of the electrical wiring and installation drawings for the reactor area has been made. This included drawings of the north and south instrumentation pits, floor box No. 1 and No. 4 and all installations involving the top shield, with the exception of the cerrobend heaters.

2. Control Room - All design modifications brought about by the changes in the reactor control system and by changes to the plant instrumentation are nearing completion.

3. Sodium Piping Area - A check is being made of the Norman Engineering Co. drawings which show the piping heaters, thermocouples and heater power distribution. Changes to these drawings are being made to incorporate the latest piping additions and design requirements.

4. Instrument Pick-ups - All drawings for the fabrication of the sodium level and alarm pick-ups were released.

5. Electrical Power - The Phase "B" electrical contract was awarded to the E. E. Draucker Co. This contract calls for the work delineated in AI specification SRE-F-5-202. It includes building power and lighting, most electrical equipment installations, pipe and tank heating and all electrical instrument and panel installations.

The specifications for pump and fan drives and distribution panels have been revised to incorporate latest design requirements. The particular specifications for this equipment were written and released to the General Electric Co.

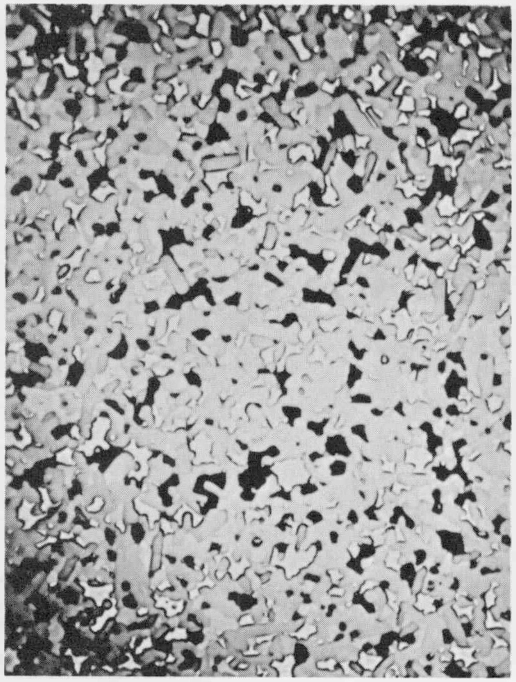
B. CONTROL ROD SYSTEMS

1. Control Rod Lead Screw Development (A. E. Miller, R. E. Douglas and E. L. Reed) - The Haynes 25 screw with nickel bonded chrome carbide balls, in test at Saginaw Steering Gear, failed after an additional 20 hours of operation. A total of 120 hours at 875° F was logged before failure. Failure was caused by excessive wear of the carbide balls and Haynes 25 races, this allowed the ball return tubes to rub on the screw race. Photomicrographs (Fig. 30) of the nickel-bonded chromium carbide balls reveal excessive porosity and a discontinuous binder stage. It is believed that poor bonding between carbide grains resulted in "spalling" of carbide grains into the race, and is responsible for the abrasive wear of the races. Better carbide material would have resulted in substantial improvement in screw life, however, it is felt that "spalling" of carbide particles would cause premature abrasive wear of the assembly.

To date all Haynes 25 screws tested were fabricated by grinding annealed material. The units were then "run-in" with hardened steel balls under a 3000 pound load in an effort to work harden the surfaces of the races. Photomicrographs (Fig. 31 and 32) of a race demonstrates insignificant differences in microstructure of the bearing and nonbearing surfaces of the race, indicating insufficient cold work. The inadequacy of the ball rolling technique was confirmed by hardness measurements of 29 Rc as compared to the desired hardness of 50 Rc. In an attempt to obtain sufficiently cold worked screws, an order has been placed for a thread rolled screw. This sample will be six inches long and should be adequate to demonstrate whether or not suitable hardness can be obtained. The threads in the mating nut for the above screw will be ground in a cold pressed slug.

The surface hardness of Haynes 25 balls similar to those used in previous Haynes 25 screw tests[4] varied from Rc 37 to 41 in the softer regions to Rc 55 in

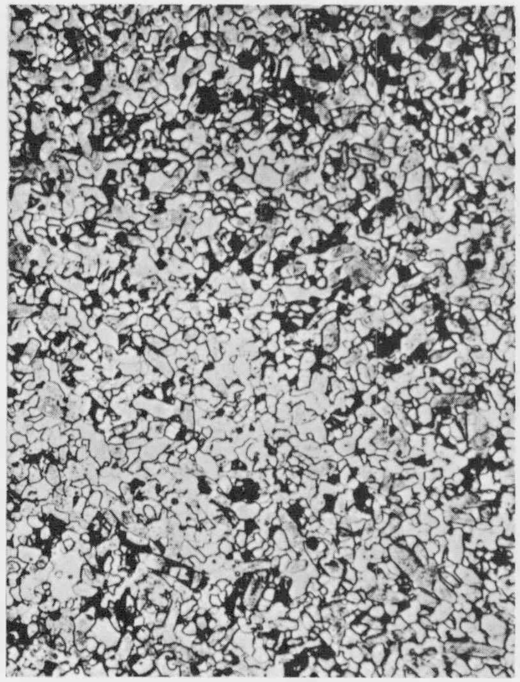
X500



(a)

Unetched. Shows size and distribution of voids which appear black. Nickel bond (white) also visible because of relief polishing.

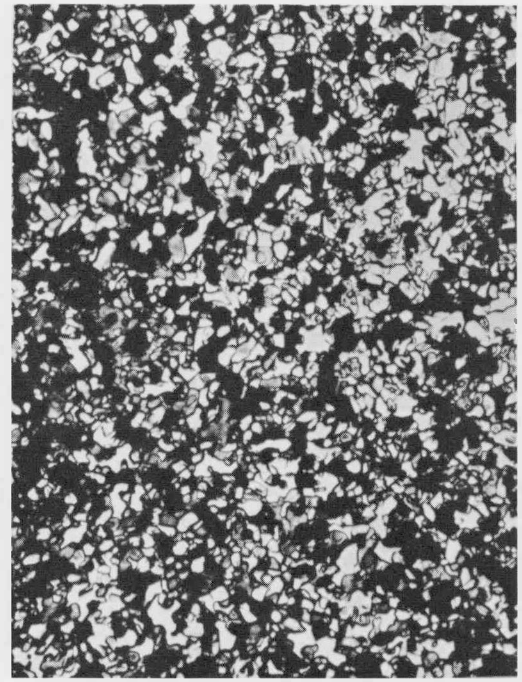
X500



(b)

Etched for approximately 3 minutes in a hot ferricynamide solution. The boundaries of the carbide particles and nickel bond material are visible and some of the carbides have been darkened.

X500

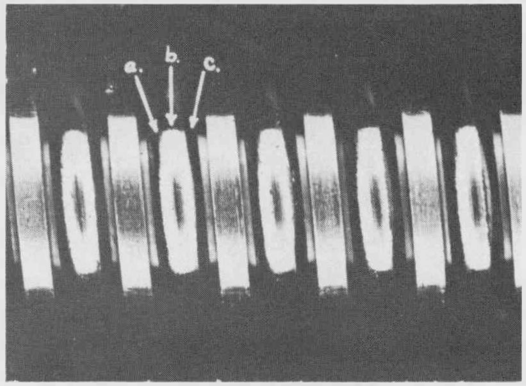


(c)

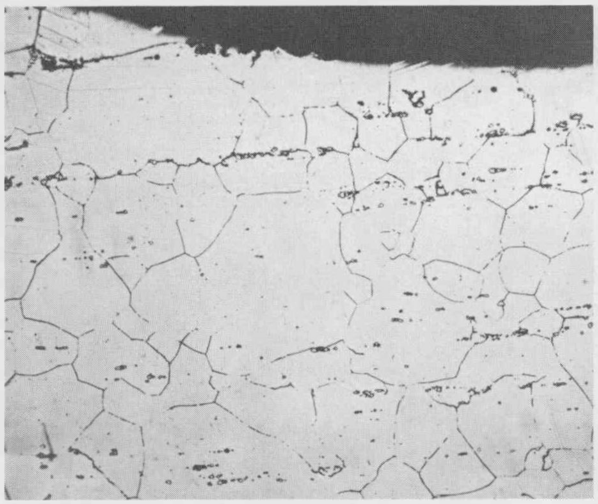
Etched for approximately 5 minutes in a hot ferricyanide solution. Carbide particles appear gray and the nickel bond material white. Note that the voids have been enlarged from etching.

Fig. 30. Microstructure of the Nickel-bonded Chromium Carbide Ball

KI-HCl-citric acid electrolytic etch. Note that the microstructures are essentially the same. The letters indicate the locations of the microstructures.

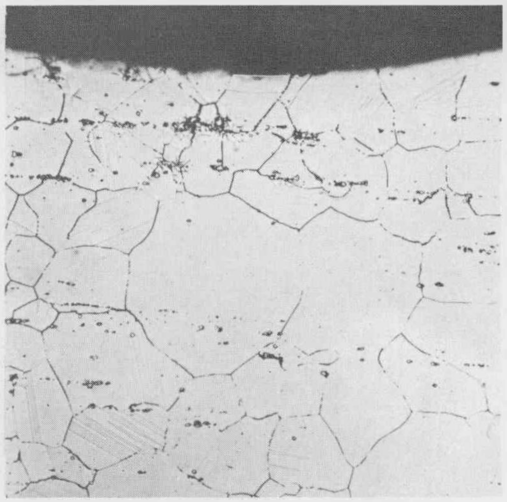


X100



(a) Lower portion of bearing surface

X100



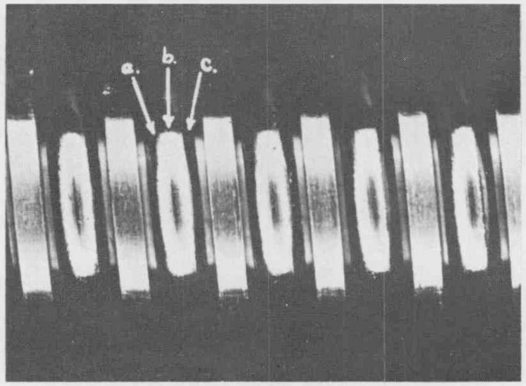
(b) Bottom of race

X100

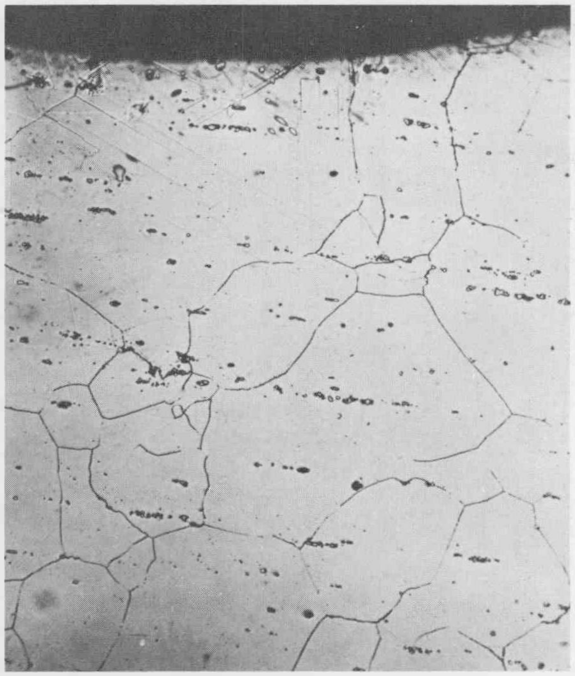
(c) Center portion of bearing surface

Fig. 31. Microstructure at various Locations of the Race of Haynes Stellite 25 Alloy Screw (full section)

X2

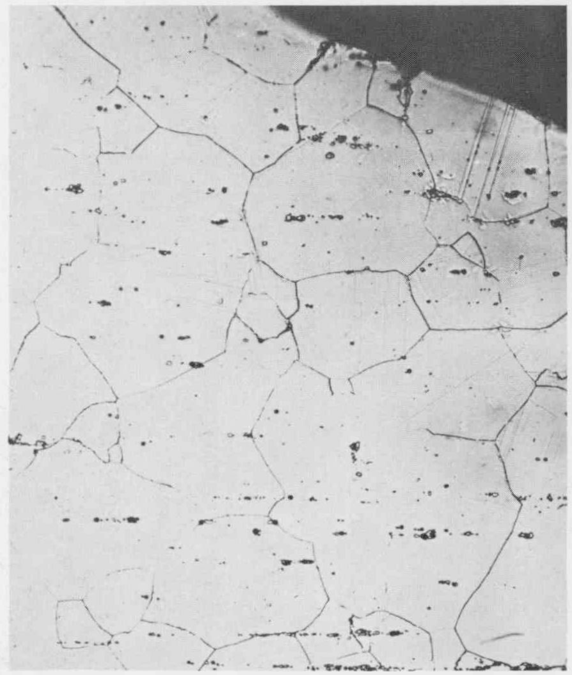


X100



(b) Bottom of race

X100



(c) Center of bearing surface

Fig. 32. Microstructures near the surface of the race of the Haynes Stellite 25 Alloy Screw (half section)

the harder regions. The balls showed "lopsided" etching characteristics similar to that of a non-homogeneous structure. A possible method of fabricating balls without the soft spots has been developed and a sample quantity ordered.

2. Prototype

a. Control Rod (A. E. Miller, R. E. Douglas, S. Elchyshyn and C. H. Kuhnhofer) - The repaired control rod prototype, with sulfided molybdenum keys, was tested for 19 hours, a total 413 cycles over a four-foot length at 300° to 800° F, before the torque requirements became excessive. Failure was caused by galling of the lowest shield rings. Material abraded from these rings (Fig. 33) fell into the screw causing damage to screw and nut. The keys were damaged slightly, possibly by this same material from the rings. It was found that the shield ring shaft, on which the shield rings are mounted, was bent. The bent shaft might have caused excessive bearing pressure on the lowest set of shield rings. The excessive pressure would explain the cause of the galling.

The prototype rod was assembled after the following operations were performed.

1. Damaged portions of screw machined
2. Damaged portions of nut machined
3. Damaged shield rings repaired
4. Bent shield ring shaft straightened
5. Additional MoS_2 rubbed on keys.

A test of 872 cycles over a two-foot length at temperatures between 400° and 800° F was made. This test was terminated after a large inert gas leak developed. The rotating "O" ring seal in the bearing housing had failed. Subsequent examination of the screw, nut, and keys showed little or no damage. In order to improve the rotating seal two new "O" ring grooves were machined in the shaft and the bore of the bearing housing was modified.

The rod was reassembled and put into test at 925° F. It was continuously cycled at this temperature for 54 hours at which time it failed. A total of 2224 cycles over a two-foot length was recorded. Examination of the disassembled rod revealed that the ball carrier nut was jammed on the screw. The nut could not be freed until all the balls were removed. The section of the screw adjacent to the jam was severely damaged (see Fig. 34). Galling between the inner (Fig. 35) and outer shield ring (Fig. 36) had occurred again. Alignment and straightness of the

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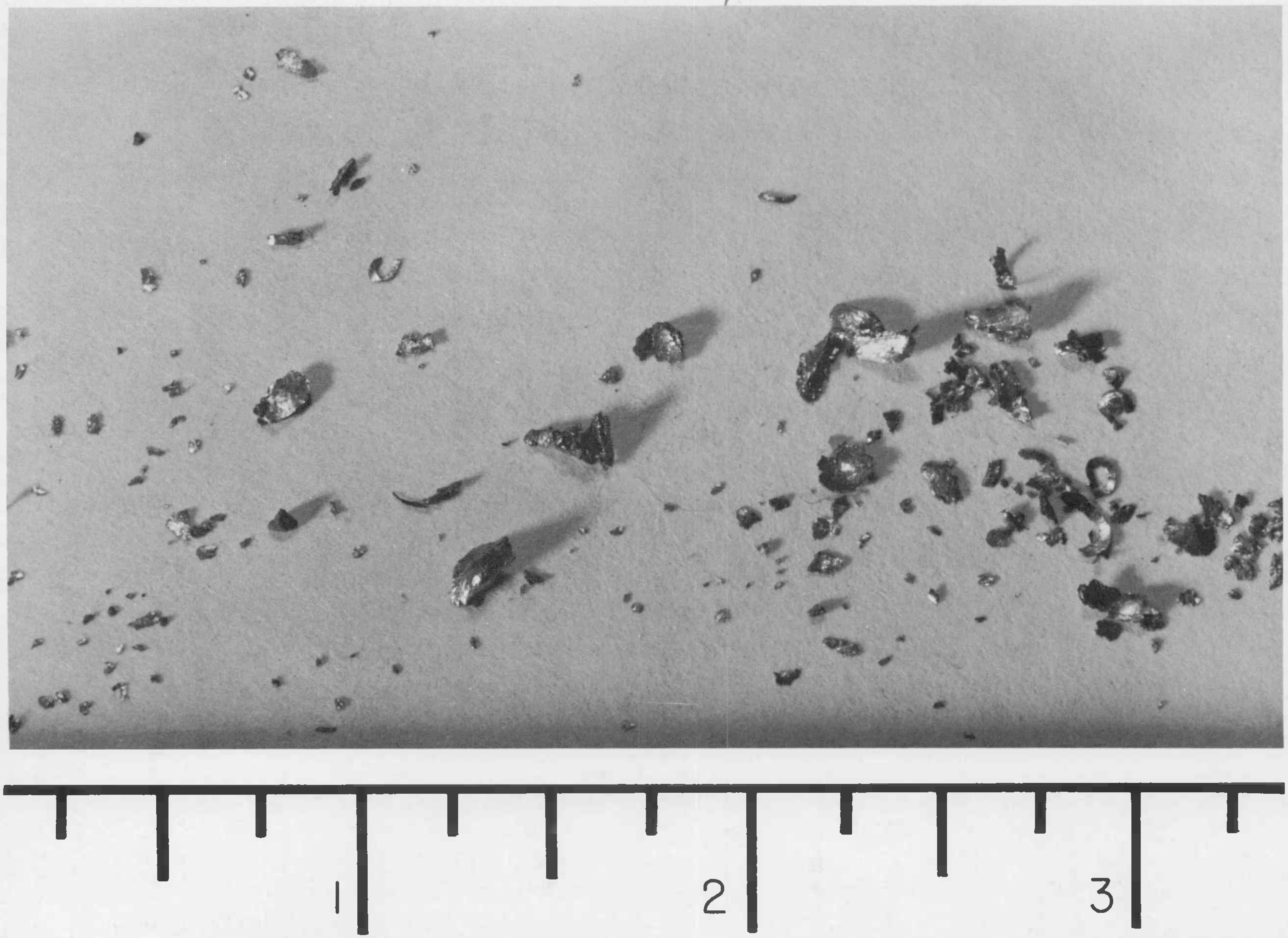

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Fig. 33. Material abraded from lowest Shield Ring of Control Rod Prototype

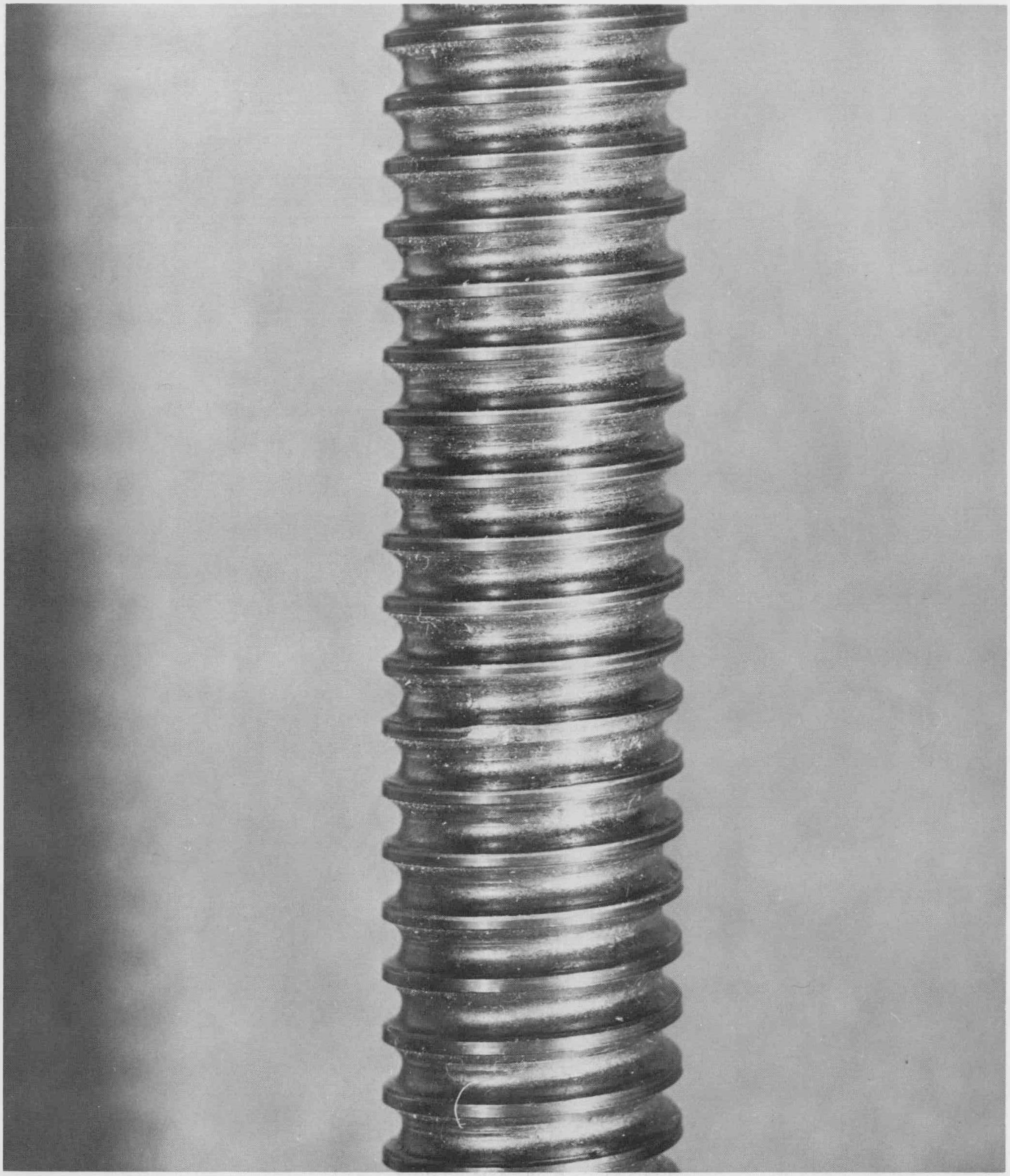

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Fig. 34. Damaged Portion of Screw adjacent to Jam

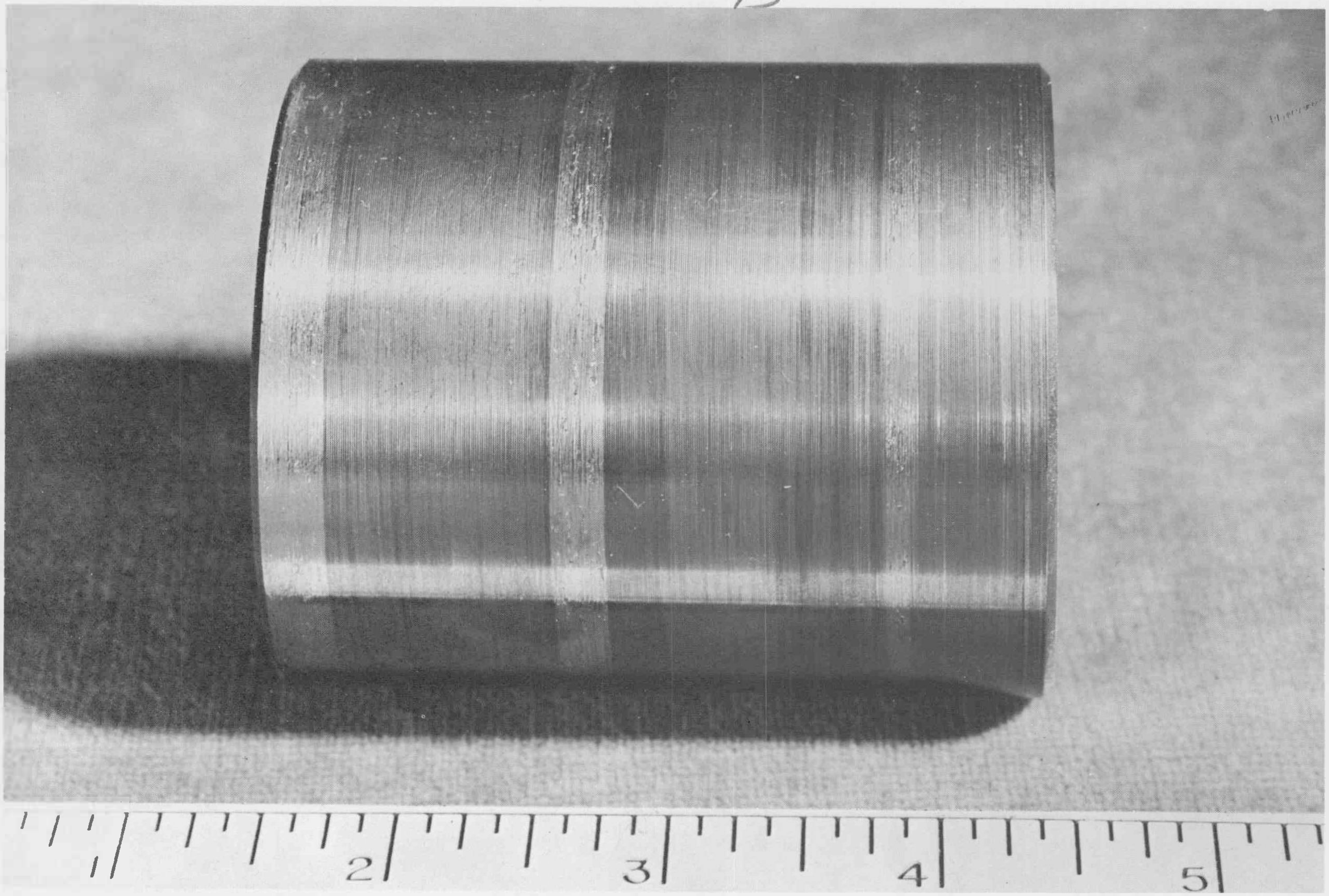

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Fig. 35. Inner Shield Ring (showing galling damage)

9693-55141B

Fig. 36. Center Shield Ring (showing galling damage)

shield ring shaft was checked and found to be satisfactory, eliminating it as a possible cause of the galling. Pickup of material was observed on the poison ring (see Fig. 37).

Examination of the thimble with a boroscope revealed slight damage to the thin wall section of the thimble. The molybdenum keys remained in good condition, showing some wear, after 90 hours of operations.

Clearance between inner and outer shield rings has been increased. The beveled edges of the poison rings have been replaced with a radius to eliminate pickup and the screw and nut were remachined. New stainless steel balls have been ordered to replace the damaged ones (see Fig. 38). Tests will resume on receipt of these balls.

b. Control Rod Drive (A. E. Miller) - The prototype double drive was satisfactorily used in conjunction with the three prototype rod tests made this quarter. The design of the control rod drives has been modified because of difficulties experienced in maintaining the rotating seals in the bearing housing of the prototype rod. A static seal between the drive and control rod thimble has been added. This backs up the rotating seal thus preventing any leak during reactor operation.

C. SAFETY ROD SYTEM

1. Latch (C. H. Kuhnhofer, E. C. Phillips and A. E. Miller) - After soaking for 500 hours at 1100° F, the capsule containing the silver plated latch was opened. Unfortunately, the latch had been released prior to capsule opening and no useful information was obtained. Provisions were made to prevent premature release and the test was re-run. A torque of 10 pound-inches was required for release after soaking, however, the latch did not open completely. The latch has been reworked to eliminate the cause of the hold-up and another identical test is in progress. The 10 pound-inch torque requirement is well below the 30 pound-inches that the torque tube can safely transmit.

2. Snubber (S. Elchyshyn, C. H. Kuhnhofer and H. L. Hornbaker) - The replacement Haynes 25 piston rings were received and put into test during this quarter. These rings are identical in size to those tested during the last quarter.

During this quarter the rings were subjected to 1000° F for 267 hours. A series of 155 "hot drops" and 84 "cold drops" were made with no damage to the rings.

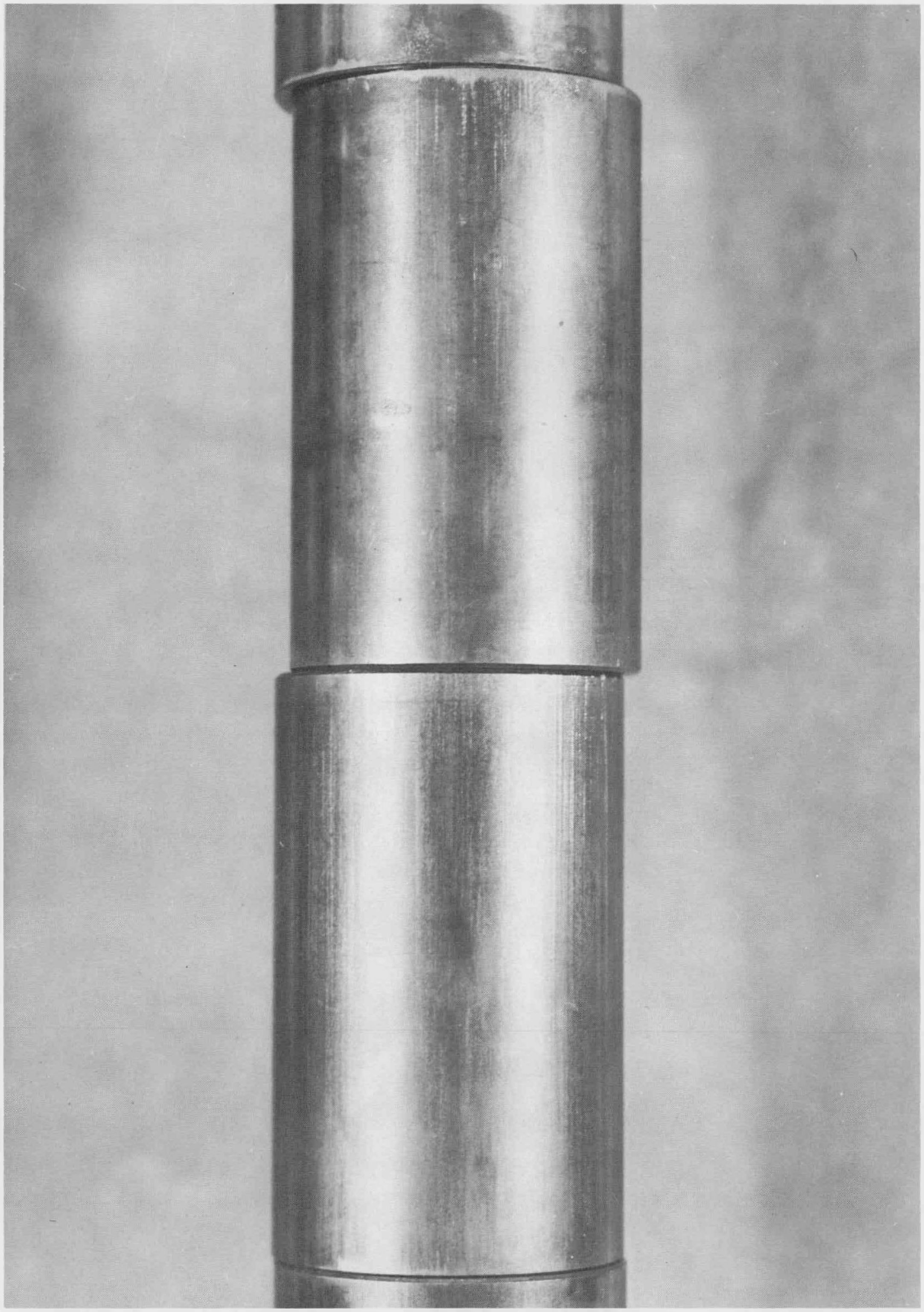

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Fig. 37. Poison Rings -- showing pick up of Material

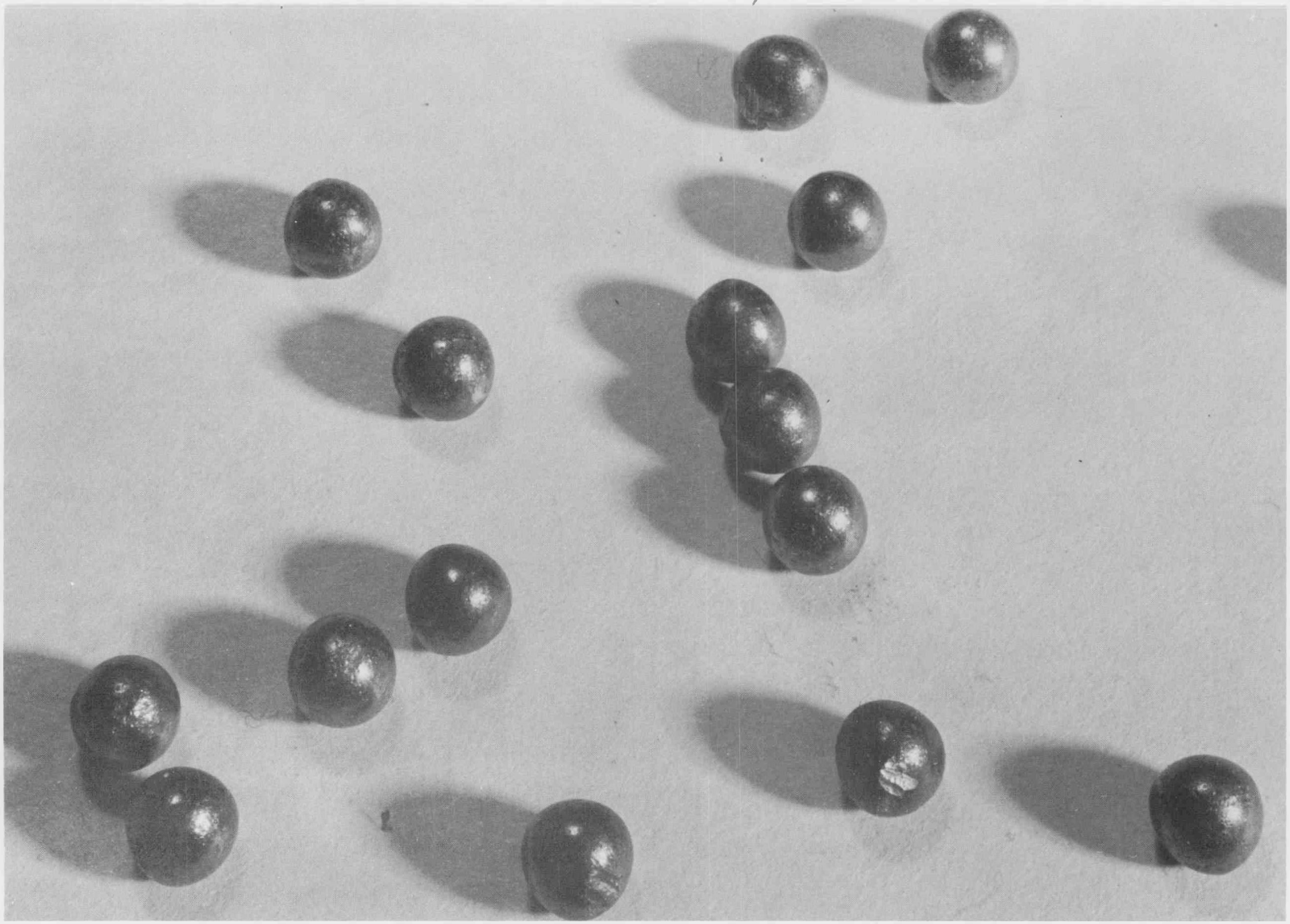

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Fig. 38. Damaged Stainless Steel Balls

Cold tests were made utilizing a piston made to the ring manufacturer's recommended dimensions. Inadequate snubbing was observed and this piston was replaced with one equivalent to that used in the previous tests. While snubbing definitely improved with the change in pistons, it was inferior to that previously obtained. Inspection of all component parts revealed that the upper portion of the pull-tube which passes through the lower cylinder bushing had become tapered. As a result the clearances between the bushing and pull-tube varied between 0.007 and 0.017 inches. The design clearance is 0.004 to 0.006 inches. It is postulated that the change in pull tube dimensions occurred when the previous rings failed. A new pull-tube is being fabricated in order to determine whether or not changes in pull-tube dimensions can be expected with adequate snubbing.

Snubber tests were continued with the deformed pull-tube. However, a packing gland using soft packing has been improvised to reduce the helium loss from the furnace. To date approximately 50 cold drops and 25 drops at 960° F have been made. The piston rings have been at 960° F for over 120 hours and are as yet undamaged.

3. Prototype (A. E. Miller, E. C. Phillips, and W. A. Davis) - The prototype safety rod was completely assembled. However, slight alterations have been performed on some of the component parts.

The latch plate in the prototype rod has been extensively reworked to eliminate severe galling which occurred between the torque tube and latch plate. An investigation is being made to determine what type materials should be used for the latch plate and torque tube at reactor operating temperatures.

Chipping of the edges of the poison rings which occurred during cold operation has been overcome. The chipping was eliminated by installing 1/8-inch thick stainless steel buffer rings between the boron nickel rings.

The limit switch system was found to be inadequate in stopping the latch at the proper point prior to latching. However, with the addition of another switch, operated by the switch rod, the system operates satisfactorily.

The prototype rod is now being instrumented for time response measurements at room temperature.

D. MoS_2 LUBRICATION AT HIGH TEMPERATURE (K. Horton)

The friction measuring apparatus was tested with MoS_2 coated specimens. A coefficient of friction varying between 0.18 to 0.29 was measured. These high values are believed due to the inadequacy of the apparatus. A meeting with Electrofilm Corp. has been arranged in order to discuss high temperature friction measuring apparatus. With our present knowledge on this subject it is felt that a suitable device can be designed to measure the coefficient of friction at elevated temperatures. Work to develop an apparatus will precede further friction measurements.

XIV. SHIELDING

A. FUEL HANDLING COFFIN (R. L. Ashley)

The density tests on several specimens of rolled lead plate being machined for the fuel handling coffin indicated densities of 11.23 gm/cc. This figure is within a fraction of 1 per cent of what had been assumed in the calculations determining the fuel handling coffin thickness. However, during the machining of this lead, it was found that most of the plates contained varying degrees of porosity in the central region.

The samples used in the above tests were selected so that they contained no porosity. A series of density measurements will be made on pieces containing the porosity in order to calculate the radiation levels which can be expected at the surface of the fuel handling coffin. Calculations now indicate that the maximum surface radiation level will be between 1.5 and 2.0 r/hr.

B. ACTIVATION OF MAIN SODIUM BLOCKING VALVES (R. L. Ashley)

Analysis of the extent of activation of the main sodium blocking valves in the galleries by neutrons leaking through the thermal shield has been completed. The calculations revealed that for operating times of one year and "cooling" times of more than two weeks, over 90 per cent of the induced gamma activity will be due to Co^{60}. The analysis indicated a maximum activity of 2 millicuries; the corresponding radiation level at a distance of three feet from the center of the valve was estimated to be 3 mr/hr. All other valves examined were found to be less active. The complex nature of the neutron distribution in the areas occupied by the valves has limited the analysis to an approximation. More accurate information on the induced activity will be available when neutron measurements have been performed.

C. GAMMA IRRADIATION FACILITY (R. L. Ashley)

The drawings of the facility were reviewed in regards to shielding integrity. Calculations indicated -- if the outer plug were removed immediately after shutdown, following full power operation, the radiation level over the open hole would be about 1×10^4 r/hr, while the general levels in the reactor room would be about 50 r/hr. With eight days decay, the level over the open hole would be reduced to 1 r/hr.

Several methods which could be utilized in removing the outer plug have been considered. (It would be necessary to remove the outer plug only if samples having a diameter greater than 3-1/2 inches were to be irradiated.)

D. INERT GAS SYSTEM (R. L. Ashley)

The problem of the potential uncontrolled release of radioactive gases to the SRE environs was reduced. This was accomplished by painting the gallery and vault walls and by eliminating the constant pressure tank.

E. MODERATOR REMOVAL SYSTEM (D. S. Duncan)

The proposed method for the safe removal of radioactive moderator and reflector cans, after several years of reactor operation, has been discussed. The method calls for the construction of a portable shielded compartment which would protect the operator during the removal operation. Since, during this operation the cans themselves will be unshielded, all other personnel in the building will be required to leave the area.

The shielding requirements for the operator's compartment have been determined. Analysis for the removal of zirconium moderator cans and stainless steel reflector cans were carried out. CS grade graphite and Type 304L stainless steel were used in the calculations. Actually, the SRE will use TSP graphite, the impurity composition of which is not completely known. However, the activation of the TSP graphite should be conservatively approximated by assuming a composition similar to that of CS grade graphite, the composition of which is known.

The calculations were based on the assumption that the reactor had been shut down for two weeks. This time is sufficient for the sodium in the reactor and other high level, short-lived activities to decay to the level of the longer lived components of the induced activity.

The results of the calculations are summarized in the Table III.

TABLE III

Shield Dimensions Required to Achieve Tolerance

	Heavy Concrete (inches)	Lead (inches)	Steel (inches)
Front shield	31	9-3/8	
Top shield	19-1/4	5	
Side shield	20	5-1/8	
Floor shield	< 20*		2-3/8
Rear shield			

* If the floor shield is supported 12 inches above the reactor floor, 20 inches of heavy concrete would be required. Since the floor shield will rest on the reactor floor, or almost so, the floor thickness should be only that required to support the shielded compartment.

Calculations were also performed to determine the dimensions of different viewing windows which could be inserted in either the front or top shields. The results indicated that either 14-1/4 inches of lead glass or 3 inches of lead glass and 28 inches of $ZnBr_2$ solution would be required in the front shield. The top shield would require only 8 inches of lead glass.

Since it was desired to estimate the distance from the unshielded source at which the radiation level would be reduced to tolerance, several additional calculations were performed. It was found that the dose rate would be reduced to tolerance at distances of 1530 and 1020 feet, when no shielding and when the effect of the building wall (5-1/2 inches of ordinary concrete) was considered.

F. LIQUID WASTE SUMP PUMP (H. E. Richter)

This assembly has been received. Modifications for additional piping connections are being made by Atomics International.

XV. REACTOR SERVICES

(A. M. Stelle, H. A. Ross-Clunis and G. R. Cogswell)

A. HEAT TRANSFER SYSTEM

The piping required for the Edison steam generator tie-in has been added to the main secondary circuit. The circuit now divides into two parallel loops, one to the steam generator and one to the air-blast exchanger. Each loop contains an angle valve in the supply leg and a plug valve in the return leg; both valves are of the frozen seal type. This permits dividing the flow in any desired ratio between the two loops. In addition, drain, vent, and purge lines are provided so that either loop can be drained for maintenance, and then refilled without disturbing the operation of the other loop.

The pressure drop through the steam generator loop is greater than the original drop through the air-blast exchanger. In this case the main secondary pump can be operated at a slightly higher speed to satisfy the requirements.

Provisions are now being made for the future addition of two hot trap units. Pipe stubs will be added to the six inch pipe on either side of the main primary pump. This will facilitate the addition of the hot trap circulating lines. Space is to be provided in the disposable cold trap vault for two future hot traps. Each trap will be in its own shielded compartment and will be of the disposable cartridge type. Disposal will be remotely handled, similar to that of the disposable cold trap.

The Pump Services P and I diagram has been changed to show packed bearings instead of a circulating lube oil system. Force feed lubricators are now provided for the gas seal at the top of the pump case.

B. SODIUM SERVICE SYSTEM

A transfer tank has been added between the melt tanks and the sodium filters. The use of this tank will allow the sodium to be pressured through the filters instead of depending upon gravity flow.

C. TOLUENE SYSTEM

The emergency water supply for the evaporative coolers has been changed from manual to automatic. It is actuated when the discharge pressure from the re-circulating water pumps is lowered.

D. INERT GAS SYSTEM

The vent compressors are now water cooled. It is a once-through system and the water is dumped into the toluene system evaporative cooler sumps as part of the make-up.

E. RADIOACTIVE LIQUID WASTE SYSTEM

The supply for the cleaning cells has been changed from treated water to distilled water and a distilled water storage tank has been added. The valves have been modified slightly at the sump tank so that the contents can be stirred by re-circulating the liquid through the sump pump.

F. PIPING

Changes and corrections to piping drawings are about 95 per cent complete.

The SRE-Edison tie-in piping has been laid out and the piping arrangement drawings are well under way.

Certified vendors' drawings of the gas compressors have been received and the piping to the compressors revised accordingly.

The insulation drawings and specifications have been sent out for bid.

The pipe hangers and supports are complete and all components have been delivered either to Atomics International warehouse or to the Site.

G. FUEL HANDLING (J. A. Leppard)

No handling experiments, as such, were performed during this quarter. The coffin and its associated process tube sodium circuit were used continually in carrying out certain experiments with the prototype control rod and the fuel cluster assembly.

A complete maintenance overhaul of the coffin was carried out. At this time the control panel was rewired, certain relays replaced, and the guide tubes and coffin body were thoroughly cleaned. Difficulties have developed in the use of the standard "O" ring gaskets and on the pickup device plugs. The gaskets have a tendency to roll in their grooves and ultimately shearing. Square type gaskets called "Quad Rings" have been substituted and are functioning properly. However, it is noted that they have a greater friction in the sliding seal.

H. HYDROGEN IN HELIUM (W. Bradshaw)

1. Effect of NaK bubbler on the Hydrogen Content of Helium - The temperature of the first tank was varied from room temperature to 600° F, (the highest temperature attainable with the external heater cable).

Partial hydrogen removal was obtained at low flow rate when pre-dried helium was passed first through the heated NaK tank and then through the cold NaK tank. At 100° to 300° F the hydrogen concentration was reduced from 3 ppm to 0.1 to 1.4 ppm. At 525° to 600° F the hydrogen concentration was reduced from 8 ppm to 2 to 3 ppm. As the temperature of the NaK in the second tank is increased, the hydrogen removal by NaK treatment becomes less effective.

Partial hydrogen removal by the NaK bubbler system is obtained only when the helium is pre-dried and when the NaK temperature is at equilibrium. No hydrogen removal is obtained when the gas flow rate is greater than one cubic foot per minute.

XVI. REACTOR STRUCTURES

A. SRE REACTOR CORE TANK (J. C. Cochran)

1. Welding Rods - In January 1956 it was discovered that a Type 410 (martensitic) welding rod may have been used in the fabrication of the core tank and certain other components of the SRE sodium system. The components in which this rod may have been used will be subjected to high temperatures and in some cases high stresses. Type 410 welding rod as deposited, forms a martensitic structure which is prone to fissuring even when deposited under ideal conditions. It was believed that when such a rod is deposited under uncontrolled conditions this tendency to fissure and crack, both at room and elevated temperatures, would be greatly increased. Therefore it was decided that all traces of Type 410 weld deposit be removed from the affected components.

The rod in question was a coated rod 1/8 in. diameter and 14 in. long. When received from the manufacturer, it was mixed with the Type 308 (austenitic) welding rod which was the type desired. Approximately 24 per cent of the mixed rod received by Atomics International was of Type 410. Examination of the welding records and of the design drawings determined that the mixed rod could have been used in the following components and locations:

1. Core tank
    (a) Repairs to girth welds (but not in the original welds)
    (b) Nozzle welds
    (c) Downcomer assemblies
2. Cold traps
3. Valve extension brackets
4. Six-inch piping spools
5. Three-inch piping spools.

2. Inspection - Type 410 welds and Type 308 welds differ in their magnetic permeabilities. The Type 410 welds have the higher permeability. No commercial type instruments were found which could detect differences in magnetic permeability much below the surface of the material. Accordingly, an instrument was developed by Atomics International for this purpose. By checking this instrument on samples containing known deposits of Type 410 material, it was found that such inclusions could be detected to a depth of approximately 3/4 of an inch. Complete inspections using this instrument were made. These inspections included the core tank seam welds, the core tank downcomer assembly welds and the core tank nozzle welds, the six in. and three in. piping spools, the valve extension brackets and the cold traps. In the inspection of the tank seam welds, six locations were judged to contain suspect areas. Drill samples were taken from these areas at increments of 1/8 in. depth and were chemically analyzed for nickel content. Type 308 weld rod is known to contain 9 to 10 per cent nickel, while Type 410 contains no nickel. In all the drill samples taken from weld seams no deficiency in nickel was found. However, two suspect areas were found on the surface of the parent metal in the vicinity of the nozzles. The chemical analysis of drill samples taken from these areas indicated that there was a deficiency of nickel at the outer surface. This was probably due to weld material left after removal of the temporary nozzle support assemblies used during fabrication of the

tank. The suspect material was removed and subsequent checks with the permeability measuring probe indicated that no Type 410 material remained.

Inspection of the downcomer assemblies indicated several suspect areas. Plug samples were removed from certain of these suspect areas and subjected to inspection by microscopic examination, X-ray diffraction, and chemical analysis. Microscopic examination of one of the plug samples revealed the presence of Type 410 weld material.

It was necessary to completely disassemble the downcomers "A" and "B" to inspect the internal welds. Certain of these internal welds also had areas of high permeability.

Inspection of the nozzle welds indicated areas of high permeability around all nozzles. Drill samples taken from certain of these nozzle areas and analyzed chemically indicated a deficiency in nickel. All nozzles have been removed and are being replaced. New downcomer assemblies are being made using as much of the original material as possible. However, none of the original material containing suspect weld areas is being used. No rework of the core tank welds is required with the exception of repairing the holes made in taking the drill samples.

Inspection of the piping spools indicated that there were approximately 13 questionable welds. These have been repaired by removal of the weld material, inspection with the permeability probe, and re-welding.

Inspection of the valve extension brackets and of the cold traps has not indicated that there are any questionable welds.

Another device was also developed to be used in the detection of Type 410 weld inclusions. It has been found that Type 410 material shows more ability to retain magnetism than does Type 308 weld material. The test procedure was as follows:

1. To de-magnetize thoroughly the area in question with an ac solenoid to establish a known magnetic history.
2. To magnetize the area by running a strong bar magnet slowly along the seam while the weld material was being tapped with a hammer.
3. To again de-magnetize the area and to plot the residual field using readings taken from an Oerstedmeter.

This apparatus was not available until near the end of the inspection period. It was used to inspect certain areas in the core tank welds already considered questionable following inspection using the permeability measuring probe. The findings with this device in this area were negative, corroborating the information received from the chemical analysis of the drill samples. However, certain other areas, considered satisfactory following inspection with the permeability probe, were adjudged suspect following examination with this latter device. Chemical analysis of drill samples taken from these areas did not show a deficiency in nickel.

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