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EXPERIMENTAL BREEDER PROJECT
Report for the period March 1, 1949
through January 31, 1950

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EXPERIMENTAL BREEDER PROJECT

Report for the Period March 1, 1949 through January 31, 1950

Contractor's Work

Design work of The Austin Company has gone steadily forward with the following percentages of completion estimated at the end of the month.

<u>Date</u>	<u>Percentage Completion</u>
March, 1949	5 1/2
April, 1949	23 1/2
May, 1949	38
June, 1949	45
July, 1949	60
August, 1949	70
September, 1949	75
October, 1949	80
November, 1949	83
December, 1949	85
January, 1950	87

It is expected that this design work will be completed by late spring.

Some work was done at the site under subcontract, pending selection of the prime contractor. The well has been finished, the pump installed, and the water supply proven. The 8 inch water line from the pump to the building is well under way and should be completed within the next month. The excavation for the building is now essentially complete. A power line has been brought in, and the main substation is nearing completion. The Bechtel Corporation has been selected as a prime contractor and is now in the process of setting up field offices and selecting staff for various key positions. It is expected that concrete work will get under way in the near future.

Construction Design

A large proportion of the time of our mechanical design group has been devoted to the design and preparation of construction drawings of the component parts to be constructed by the Laboratory. The design schedule has been subdivided into thirteen major components representing the reactor proper and allied control equipment. These are now largely completed with the exception of minor changes dictated by tests conducted on preliminary units.

Much of the special tooling required for the fabrication of the various components, such as the reactor plates, has been completed. The special dies required for drawing the outer blanket brick jackets are now being made, and a series of weld studies is in progress to determine the most advantageous weld design for each of the many welded joints required in the various components. The lower shield is being fabricated and is scheduled for completion in March. This unit will be assembled in the laboratory, together with the external safeties and controls.

The reactor structure members and the jackets for the critical material are to be of stainless steel, and all of the parts subjected to high flux will be fabricated from a single heat of type 347 stainless steel made to our specifications. The first of this order is due for delivery in mid-January.

Measurements of Gamma-Ray Attenuation in Concrete (E. N. Pettitt)

Using the experimental "hot" laboratory described in ANL-4274, measurements were made of the attenuation of gamma rays by concrete. The gamma-ray sources were placed approximately in the center of the "hot" laboratory, and the intensity of gamma rays penetrating one of the walls was measured. A Victoreen thimble ionization chamber was placed inside the "hot" laboratory to measure the source intensity. As the gamma-ray emitters were small compared to the distances involved, they approximated point sources. Because of this, it was possible to compare measurements made with different source-to-instrument distances by means of the inverse-square law. Concrete thickness was increased by stacking additional concrete blocks against the wall.

Attenuation of gamma rays from three sources has been investigated: Na^{24} , which emits 1.4- and 2.8-Mev gamma rays in cascade; Co^{60} , which emits 1.1- and 1.3-Mev gamma rays in cascade; and fission products of an "X" slug, which emit a variety of gamma rays.

Intensity measurements were made of Na^{24} gamma rays after penetrating 24 to 43 inches of concrete (specific gravity = 2.2). The results of these measurements, made with a Lauritsen electroscope calibrated against a standard cobalt source, are shown in curve A, Figure 1. It can be seen that within the range of these measurements the attenuation of Na^{24} gamma rays in concrete is exponential with thickness. Measurements for the same thickness of concrete made with the electroscope close to the wall (approximately 4 cm) and as far as one meter from the wall agreed (within experimental error) with the law of the inverse square of the distance from the source. In the range of 2 to 4 feet of concrete, 3.75 inches will reduce Na^{24} gamma-ray intensity by a factor of 2.

FIG. 1

For comparison, the results of three different calculations for attenuation are also plotted in Figure 1. Curve D is a plot of the attenuation factor for Na^{24} gamma rays, calculated by using the simple point-source formula of chapter V of CL-697. The linear absorption coefficients for 1.4- and 2.8-Mev gamma rays were deduced from the half-value-layer curves given in the Radiation Hazard Control Manual. The aluminum curve was used and adjusted to specific gravity = 2.2. This method neglects multiple scattering and any gamma rays which are scattered into the detector from regions outside the solid angle subtended between the source and detector. Curve D thus gives optimistic attenuation factors.

Curve C is calculated by using the same formula as for curve D, the difference being that in an empirical attempt to allow for radiation scattered into the detector, the half-value layers have been increased by 20% in accordance with the method suggested in CRC-341.

Hirschfelder, *et al.*, in two papers in the Physical Review, 73, 852 (1948), have made extensive calculations for gamma rays of 1, 3, and 5 Mev traversing lead, iron, concrete, and water. From their tabulated results, curve B has been deduced for Na^{24} gamma rays. Their calculations assumed concrete with a specific gravity of 2.3, which would tend to make the attenuation greater than for the concrete of specific gravity of 2.2 used in the measurements which are plotted in curve A.

The results of similar attenuation measurements made for the gamma rays of Co^{60} are shown in Figure 2. It will be noticed that only measurements made with the electroscope close to the concrete (higher-level measurements) follow a pure exponential attenuation with increasing thickness of shield. The deviation from an exponential attenuation for measurements made at points removed from the concrete is probably not a real effect but is due to the room background, which was not subtracted from these low-level measurements. This effect did not appear in the Na^{24} measurements, all of which were made at higher levels possible because of the stronger sources available. Also, it will be noticed that for thicknesses of concrete less than 30 inches (measurements made at level much higher than room background) the attenuation in each case is less for points close to the concrete. This is probably due to soft radiation scattered from large angles which does not reach points far from the shield. Further indication of this effect was obtained by covering the wall with a sheet of 1/4 inch lead. Calculations from tabulated half-value thicknesses of lead for Co^{60} gamma rays indicate the gamma ray intensity should be reduced by a factor of 1.6, whereas measurements gave a factor of 2.2 with the electroscope close to the wall. From the measurements made with the electroscope close to the concrete, it is found that, at least in the range of 2 to 3 1/2 feet of concrete, 2.65 inches will reduce the intensity of Co^{60} gamma rays by a factor of 2.

FIG. 2

Both the glass window and the zinc bromide solution window have a specific gravity = 2.5. Some rough measurements indicate that these windows have an attenuation greater than the concrete by about the amount to be expected by the density increase.

Rod Distillation Experiment No. 2 (R. A. Cameron and H. E. Stanton)

Rod No. 19 from the pile mock-up was pulled and installed in a vertical tubular furnace on February 1, 1949. The furnace was wound with Nichrome wire with several electrically independent coils spaced along the length of the furnace. The coils were operated at different currents to simulate the temperature distribution along the central axis of the metal in the central rod of the pile. The rod had been filled with radioactive NaK, and autoradiographs showed it to be filled to the proper level (about 5/8 inch above top of the uranium) and free of voids in the annulus. One small gas pocket in the annulus disappeared after preliminary heat-treatment to induce wetting.

X-rays showed a well-defined meniscus (Figure 4) at the position noted. During the course of the experiment, from time to time, the furnace was cooled to room temperature and lowered from the rod, and X-ray exposures were made to ascertain the position of the meniscus. The interval between these exposures was progressively increased.

On April 5, 1949, the temperatures along the rod were changed to simulate those at twice the power generation in the pile. During the lower temperature period no change in NaK level was noted (provided the rod was at room temperature along its length). This constancy in level also persisted (to within about 1/64 inch) through May 11, 1949. However, the next picture, taken on July 12, showed a complete disappearance of the meniscus.

It was thought that this apparent lowering of the alloy level down below the tops of the U slugs was caused by evaporation of the NaK. At about 450°C the vapor pressure of the components of the alloy would be a few mm. Hg, but at a temperature 50°C lower the vapor pressure would be about 5 times smaller. In the experiment, except for a few brief intervals, this temperature did not exist. In any event, if the meniscus were

FIG. 4

MENISCUS -

maintained at a higher temperature than the upper end of the gas chamber, one might expect some transport of alloy and consequent lowering of the meniscus.

In order to check this possible mechanism, the rod was disassembled (July 19, 1949), and the interior examined. The break was made in helium at the joint where the fuel-bearing portion is spun over and welded to the lower end of the by-pass section. No appreciable amount of alloy could be seen in the chamber adjacent to the slugs.

The two parts were cleaned separately and carefully observed during the process. The by-pass section was cleaned by inserting a long "needle" on a hypodermic syringe through the 1/8 inch hole and slowly filling it with alcohol. There was no trace of any reaction to indicate the presence of NaK. It is certainly true that a very few cubic millimeters of NaK would produce an observable reaction, whereas the lowering of the alloy level indicated a displacement of about 3/4 c.c. The by-pass section showed no weight change on cleaning within an error of about 0.1 gm.

In order to at least qualitatively account for the disappearance of the alloy, the slugs were measured with a micrometer, and these measurements checked with the original values. It was found that the slug diameters had decreased for 9 out of the 10 slugs by amounts varying from about 0.5 to 2.5 mils. The average was 1.5 mils. In addition, measurements made on the outside diameter of the jacket indicated the possibility that it had expanded. Assuming then that the annulus had increased 1.3 mils in width, or $\frac{1.3}{22} \sim 6\%$, the volume change for this reason alone would amount to about 0.5 c.c., since there are about 8 c.c. in the annulus. This calculation is, of course, rough. The slugs also increased in length; this would have a further tendency to lower the relative position of the meniscus. This was shown in the X-ray films, since the lengthwise expansion sheared off the top tantalum spacer.

The rod was again assembled using the same uranium slugs, filled with NaK to the proper level, and inserted again into the vertical tubular heater. The same temperature distribution as before has been applied along the surface of the fuel rod, and periodic checks have been made with X-rays, one exposure a week to check the position of the meniscus at room temperature. With the exception of the periodic cooling of the fuel rod for X-ray exposures, the fuel rod has undergone continuous heat-treatment for 140 days. Measurements have been taken on each film to within 1/64 inch, and no noticeable change has been measured.

It seems reasonable to conclude that there is not sufficient evaporation of NaK alloy and subsequent condensation anywhere in the by-pass section to in any way affect pile design. The following experiments are being carried out to verify this point.

Rod Distillation Experiment No. 3---Ball Check Rod (R. A. Cameron and H. E. Stanton)

In the present design of the CP-4 fuel rod, the .020 inch annulus between the fuel section and the jacket is filled with NaK alloy and blanketed with argon gas at atmospheric pressure. If a leak developed in one of the jacket welds, or a jacket ruptured, this NaK bond plus fission gases could escape to the pile coolant. To eliminate this possible loss, it had been suggested that fuel rods be filled at a pressure of half an atmosphere. Then if a leak developed, pile coolant would be taken into the rod. At these lower pressures, however, trouble might be experienced with distillation of the bond and condensation at a cooler point.

The ball check rod experiment was set up with distillation purposely induced in two rods and one fuel rod designed to check or lower the distillation rate when filled at pressures lower than atmospheric.

The apparatus for this experiment consisted of three dummy fuel rods similar to the CP-4 fuel-rod design except for the section where the fuel-bearing jacket is spun over and welded to the lower end of the by-pass section. Three plugs were machined and pressed into the top of a jacket section (see Figure 5). Two plugs each had a .040 inch diameter hole drilled through with the top end of one of the plugs chamfered to seat a 3/32 inch diameter ball. The third plug had some constriction in it where it was welded. These three rods were placed in a vertical furnace with their by-pass sections above the top of the furnace. The furnace was wound with Nichrome wire into two electrically independent coils spaced along the length of the furnace. The rods had no uranium, were 2/3 filled with NaK alloy, evacuated, and sealed. Two thermocouples were attached to the jacket of one of the dummy rods, and the coils were operated so as to obtain a temperature of 375°C along the surface of the fuel rods.

Fiducial points were marked on each rod, and an X-ray exposure was made to show the original positions of the NaK levels before heating. Once a week the furnace was lowered, the rods were allowed to cool to room temperature, and an X-ray was taken to ascertain the NaK levels. Figure 6 shows a curve of time vs. height of the NaK column in ball check rod.

As can be seen from Figure 6, the curve of time vs. column height for the ball check rod has a slight negative slope. This NaK level has slowly dropped at a steady rate for over a 2 1/2 month period. It appears that NaK has been evaporating up past the ball check, but the condensate has not been allowed to drop back past the ball to the fuel section. The NaK levels in the two other rods behaved erratically. Condensate in the by-pass section dropped back at different periods altering the height of NaK level with an irregular time pattern. X-rays showed a real transport

FIG.5

BALL CHECK DISTILLATION EXP. NO.3

BALL CHECK DISTILLATION EXPERIMENT No. 3
TIME VS. NaK COLUMN HEIGHT CURVE

HORIZONTAL AXIS - TIME (days)
 VERTICAL AXIS - NaK COLUMN HEIGHT (inches)
 in BALL CHECK ROD

FIG. 6

of alloy in the by-pass section. According to the X-rays, the largest abrupt changes in alloy level have been around 1/4 inch or less, whereas the ball check rod shows a very steady level height over the same period. The experiment is being continued.

A second experiment, which has not been running long enough to record sufficient data, consists mainly of observing the distillation rate in three dummy rods filled at three different pressures; namely, Rod No. 1 at vacuum pressure (29.5 inches of Hg.), Rod No. 2 at half atmosphere (17 inches of Hg.), and Rod No. 3 at atmospheric pressure, but without any ball checks.

If CP-4 fuel rods are to be pulled out from the pile every two months, it would seem that the small amount of alloy evaporated over a two-month period need not change the fuel-rod design.

Mock-up Reactor (L. J. Koch)

The mock-up of a reactor section was in operation for a total of 3900 hours from March 1, 1949, through December 31, 1949.

On June 21, 1949, four fuel rods---No. 4, No. 15, No. 17, and No. 18---were removed and inspected. No difficulties were experienced, and X-ray photographs did not reveal any indication of growth or distortion. These rods had been in the mock-up for a total of 2626 hours. Rods No. 15 and No. 17 had been previously removed and inspected; rods No. 4 and No. 18 had not been removed since the original installation. The rods were reinstalled, and the test was continued.

On December 6, 1949, all the fuel rods and blanket rods were removed from the mock-up, and the internal plate assembly was removed and completely disassembled. Considerable difficulty was experienced with the hold-down latches on the fuel rods, and since this feature had been previously removed from the fuel-rod design, the locking mechanisms were removed from all the fuel rods. The rods were inspected, the mock-up was reassembled, and the test continued. There was no indication of uranium growth or distortion.

Cup and Shield Assembly Mock-up (B. C. Cerutti)

This mock-up, which was described in report No. ANL-4274, is being used to test and improve the safety and control mechanisms which actuate rods in the outer blanket or cup.

The construction design of the safety rod mechanism, which was also described in report No. ANL-4274, has been completed, with modification in the design to improve the deceleration characteristics and provide more rapid assembly and installation. One of these units has been built and is now operating in the mock-up.

The construction design of the safety and control rods has been completed, with minor modifications, and two of these units have been fabricated and are now operating in the mock-up.

A velocity and acceleration curve and travel vs. time curve of the safety rod with the preliminary design are shown on Figure 7 and Figure 8. The velocity and acceleration curves were arrived at by successive differentiation of the measured travel vs. time curve. Similar data will be taken in the near future for the construction design of this assembly. The final design should have a greater acceleration owing to a change in the design of the drive spring.

An experimental control-rod drive mechanism has been built and tested in the mock-up. The construction design of this unit has been completed and will be released for fabrication in the near future.

This unit consists of a screw drive, directly coupled to the control rod and driven through a worm-drive gear box. The worm is driven by a two-speed, differential-type gear box through which a 100 to 1 speed change is effected. This speed change is accomplished with a magnetic clutch-brake combination. The unit is driven by two (2) series-type motors, coupled to drive in opposite directions, which provide approximately a 3 to 1 speed variation. This combination thus provides an over-all speed variation of approximately 300 to 1 in both directions. A special dial indicator has been designed and built which indicates rod position to .001 inches.

A mock-up of a section of the control panel has been built to operate the external safety and control-rod drives and the internal safety-rod drive mock-up. This has also provided a check on the final wiring circuit and operational sequence to be used in the final design.

In the near future all of these mechanisms will be connected so they can be cycled continuously to obtain endurance data.

Internal Safety Rod Mock-up (W. B. Gintner)

A mock-up has been designed and built for testing the internal safety rod and its drive mechanism. There will be two of these rods in CP-4. These rods, containing enriched boron, replace two internal blanket rods and will be dropped into the reactor section adjacent to the core. Each rod

FIG. 7

FIG. 8

travels in a thin-walled tube and is held in place by a magnet. The safety will be accomplished by breaking the magnet circuit, permitting the rod to fall into position. The mechanism then follows up to guarantee that the rod does not stop because of any friction which should develop.

The magnet is designed to operate on the twelve-volt circuit that is used in operating a number of other controls. A thin disc of nonmagnetic stainless steel is held in place between the magnet and the top of the rod. The use of this disc tends to reduce the force due to residual magnetism, decreasing the break-away time between the magnet and rod. The magnet also contains an internal safety switch. This switch is so designed that, should the magnet fail to pick up the rod on the upward travel, it will turn on a warning light and at the same time lock out the rest of the control system until it is brought into contact with the rod once more.

The thin-walled tube, or thimble, which carries the rod is designed to provide a hydraulic cushion for the rod to fall into. The coolant used in the pile will serve as the hydraulic fluid for this cushion. A positive stop at the top of the reactor will prevent damage, in the event that the cushion does not function properly. At the time of this report a separate mock-up is being designed to test different types of dash-pot constructions. These tests, using NaK as the hydraulic cushion, will be conducted to determine the most suitable design.

The drive mechanism provides an extremely slow speed out and a high speed in. The high speed in acts as a follow up to insure that the rod does not stick or stop short of its full travel. This two-direction, two-speed drive is accomplished with a differential-type gear box similar to the one used for the control-rod drive. Two magnetic clutch-brake combinations are used in conjunction with the gear box and a constant-speed, three-phase motor. The actual drive travel is transmitted through a rack and pinion. A velocity, an acceleration, and a time vs. distance curve of the rod fall are included in this report (Figures 9, 10, and 11). The time vs. distance curve was measured by means of a scaler and pulse generator. The velocity and acceleration curves were arrived at by successive differentiation of this curve.

Heat-Transfer Characteristics of Vertical Composite Tube Steam Generator and Water Heater (F. D. McGinnis, M. Novick)

Further experiments were made with the falling water film type of steam generator which was described in the previous report, ANL-4274.

In previous tests, heat exchange was carried out in tubes where water was below its boiling point in the upper portion and at boiling temperature in the lower portion. This mixed condition made it necessary to

FIG. 9

FIG.10

FIG. 11

estimate relative areas for the two types of heat exchange involved in order to determine temperature distribution on the water side.

Experiments described here were done under pure boiling and no boiling conditions separately. The first was accomplished by preheating the feed water to saturation temperature in direct contact heat exchange in the steam drum. For the water heating test with no boiling, it was necessary to cool the recirculating water by means of an external heat exchanger.

Piping and fittings in the original equipment were changed so that the generator could operate at up to 400 p.s.i. ga. Tests were carried out with boiling at pressures up to 350 p.s.i. ga. and with a recirculating water rate of four times the steam rate.

Calculations of over-all heat-transfer characteristics were made, as in the previous report, using log mean temperature differences and were based on the inside area of boiler tube. The quantity of heat flow was calculated from NaK circulation rates and published values* of its specific heat. These values are somewhat lower than reported in ANL-4274 and are used so that calculated values of heat transfer will be on the safe side in view of the uncertainty of the correct specific heat values.

Table I is a summary of the measured data and calculated results for heat transfer under pure boiling. Table II lists measured data and calculated results for heat transfer to water below its boiling point.

Figure 12 is a plot of over-all heat-transfer coefficient for pure boiling vs. NaK flow rate and illustrates the effect of varying the latter. Figure 13 is a plot of over-all heat-transfer coefficient vs. water flow rate where the NaK flow rate was held constant and no boiling occurred.

As can be seen from the tables, over-all heat transfer coefficients as high as 1000 at a modest NaK velocity of 2.5 ft./sec were obtained in pure boiling. Values were obtained as high as 700 for water heating with no surface boiling at a water flow rate of 60#/hr/inch of wetted periphery and a NaK flow rate of 2 ft/sec.

The water heating runs were accomplished by allowing the pressure within the equipment to rise while cooling the recirculating water until equilibrium was established and the pressure remained constant. The atmosphere within the heat exchanger then consisted of a mixture of steam and air. It is possible that some of the incoming water was being preheated by the steam which caused some boiling down below to make up for the condensed steam. This effect would indicate a higher average coefficient

*(Table III, Preliminary Report 4, Supplement 2, N6 ori-146, Mine Safety Appliance Company, June 24, 1948.)

Table I
VERTICAL COMPOSITE TUBE STEAM GENERATOR

NaK Alloy						Boiling Water				
Run No.	Flow Rate (g.p.m.)	Re 1000	Velocity (ft/sec)	Temperatures (°C)		Power Input (BTU/hr. 1000)	Steam Temperature (°C)	Steam Pressure (p.s.i. ga.)	Δtm (°C)	U (BTU hr °F ft²)
				In	Out					
1	14.2	8.	1.45	273	225	105.	222	339	17.9	834
2	5.5	3.3	.562	336	222	97.5	220	322	28.2	490
3	9.9	5.65	1.01	291	222	107.	220	322	20.2	750
4	12.4	6.99	1.265	276	222	106.	219	315	18.5	811
5	15.9	8.75	1.62	264	223	103.5	220	322	16.9	866
6	24.9	13.5	2.54	246	220	103.	215	291	15.3	955
7	20.6	11.3	2.1	261	226	116.	222	339	17.	965
8	11.9	6.78	1.215	283	223	113.5	220	222	20.2	795
9	7.6	4.38	.776	305	217	108.5	215	291	24.4	627
10	5.9	3.54	.603	335	221	104.	220	322	28.1	523
11	9.7	5.54	.99	289	220	107.	216	298	21.4	705
12	14.3	7.85	1.46	268	221	106.	218	307	18.	834
13	18.8	10.25	1.92	254	220	103.5	216	298	16.2	900
14	22.5	12.2	2.3	250	220	106.	215	291	15.6	960
15	26.6	14.4	2.72	242	217	108.	212	274	15.1	1010
16	26.	14.3	2.66	254	227	108.5	222	339	15.3	1005
17	20.8	11.4	2.13	260	227	107.5	222	339	15.7	973
18	16.3	9.1	1.67	266	226	104.	222	339	16.6	883
19	12.2	6.95	1.25	289	224	124.5	220	322	20.8	850
20	25.	13.25	2.56	212	239	108.	208	250	14.9	1025
21	24.7	14.1	2.52	285	226	229.	220	322	25.	1290

Table II

VERTICAL COMPOSITE TUBE WATER HEATER

Full Power										
Run No.	NaK				Water				Δt_m (°C)	U $\left(\frac{\text{BTU}}{\text{hr ft}^2 \text{ } ^\circ\text{F}}\right)$
	Flow (g.p.m.)	Temperatures (°C)		Power $\left(\frac{\text{BTU}}{\text{hr}} \times 10^{-3}\right)$	Temperatures (°C)		Water Flow			
		In	Out		In	Out	G.P.M.	Γ #/hr/in. periphery		
1	20.6	145	110	122.	58	135	1.76	55.7	24.8	695
2	20.6	158	123	123.5	68	147	1.76	55.7	27.2	642
3	20.8	163	128	124.	73	152	1.76	55.7	27.7	608
4	20.7	152	117	122.5	60	141	1.69	53.6	27.0	641
5	21.	164	130	121.	52	152	1.35	42.8	34.8	492
6	21.	185	151	119.	48	170	1.09	34.5	45.5	370
7	20.9	231	197	114.	41	203	.788	24.8	75.2	214
8	21.0	202	167	121.	50	181	1.04	32.8	55.7	297
9	21.0	182	147	124.	52	164	1.24	39.2	45.	389
10	21.0	162	126	127.	60	149	1.59	50.3	32.5	555
11	20.7	128	93	123.	50	116	2.06	65.2	24.3	716
12	20.6	127	95	113.	51	114	2.00	63.4	25.1	637
13	20.7	151	117	120.	49	131	1.63	51.7	39.0	435
14	20.9	132	101	110.	50	120	1.77	55.5	26.0	600

Half Power										
Run No.	NaK				Water				Δt_m (°C)	U $\left(\frac{\text{BTU}}{\text{hr ft}^2 \text{ } ^\circ\text{F}}\right)$
	Flow (g.p.m.)	Temperatures (°C)		Power $\left(\frac{\text{BTU}}{\text{hr}} \times 10^{-3}\right)$	Temperatures (°C)		Water Flow			
		In	Out		In	Out	G.P.M.	Γ #/hr/in. periphery		
15	20.6	162	147	53.5	32	152	.496	15.7	42.7	177
16	20.4	129	113	57.5	38	123	.755	24.0	28.7	283
17	20.3	109	90	66.8	42	103	1.22	38.9	19.9	474
18	20.8	95	76	67.6	50	89	1.87	59.6	13.5	705
19	18.4	95	74	65.5	50	89	1.84	58.7	13.9	666
20	23.5	89	72	68.5	43	83	1.91	60.9	13.7	708

FIG. 12

FIG.13

of heat transfer than would be expected by pure water heating. However, in the runs below the boiling point of water at atmospheric pressure (No. 18, 19, and 20) where this effect would not be present, the heat-transfer coefficients fell in line with the others. This would indicate that the above effect, if present, is small.

In preparation for a full-scale test on a single tube of a proposed design heat exchanger, the power input was raised to 67.2 K.W. Run number 21 on Table I was made at this level. A value of 1290 was obtained for the heat-transfer coefficient at this power. This seems to indicate that a more favorable temperature difference for boiling exists between the water and wall at the higher power level.

Test of Rotating Liquid Seal (F. D. McGinnis, M. Novick)

One proposed method of designing a liquid-metal pump with a totally enclosed motor and bearing assembly provides a rotating-plate type of liquid seal which is mounted on the shaft and enables the suction to be at a negative head without pulling blanket gas into the pump. The purpose of this experiment is to determine the maximum pressure difference which can be maintained across the seal without leakage and the factors influencing it.

Figure 14 is an exaggerated sketch of the test setup. Liquid is put into the test tank to a level slightly above the upper stationary plate, and the shaft rotated by means of an external motor. The liquid in the clearances between plates is caused to rotate by means of the revolving plate and produces a pressure as measured by P_3 . If the top and bottom clearances are the same, the pressures produced on each side of the rotating plate are the same and are functions of the square of the speed of rotation of the liquid and of the differences of the squares of the O.D. and I.D. of the plate clearances. Thus the head produced on each side of the rotating plate varies as

$$\frac{n^2}{2} (O.D.^2 - I.D.^2)$$

where n is shaft speed in revolutions per minute and O.D. and I.D. are the effective liquid radial dimensions. The average liquid speed is taken as the average of the stationary and rotating plates and is $\frac{n}{2}$.

In operation, a negative suction head is applied to the pump chamber. The liquid in the gas chamber flows down into the pump chamber because of this applied pressure difference. When the liquid surface falls into the upper clearance between plates, its effective I.D. dimension

FIG. 14

ROTATING LIQUID SEAL

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changes. This causes a lower developed pressure in the top clearance and a net pumping action upward because of the higher pressure in the bottom clearance. The upper effective I.D. dimension stabilizes when the developed pressure difference equals the pressure difference between the gas and pump chambers. When the I.D. dimension equals the O.D., the liquid is lost from the upper clearance, and the maximum pressure difference for the seal has been reached.

Factors which affect the operation of a given set of plates and the power loss in the seal with a given fluid are plate clearances, surface finish of plates, and speed of rotation.

The test was carried out with kerosene having a density of about .8, which is comparable to NaK alloy. The O.D. plate dimension was 10 inches, and I.D. was 4 $\frac{1}{2}$ inches. One set of tests was carried out using a smooth machined rotating plate $\frac{1}{4}$ inch thick. Another set of tests was with a similar plate except that it had six flat-bottomed grooves $\frac{1}{4}$ inch wide and $\frac{1}{8}$ inch deep cut radially into each side.

Table III lists the results of runs at various clearances at different speeds with the two types of rotating plates. Power required was determined by measuring power input to the motor and represents the difference between running dry and with liquid.

Corrosion Experiments (L. J. Koch)

An investigation of the effects of irradiation on uranium in sodium-potassium alloy has been undertaken. On November 14, 1949, a stainless steel capsule containing a .364 inch diameter x 1 inch long cylindrical uranium slug immersed in NaK alloy was placed in the central thimble of CP-3. On December 27, 1949, after 209,662 KWH pile operation, the sample was removed and inspected, with the following results:

0.7 mg weight loss	-- (.002% weight loss)
.006 inch reduction in diameter	-- (-.16%)
.0014 inch increase in length	-- (+.14%)

The length and diameter measurements are subject to considerable error because of the difficulty of measuring to these accuracies by remote means. The surface of the slug was lightly oxidized, with no indication of any unusual surface phenomena. This result is more encouraging than those reported by G. E. of their Beta experiments in KAPL-266. An approximate comparison of G. E. foil E-4 with our slug EB-1, correcting for total irradiation and surface area, indicates that our slug corrosion was approximately 2% of theirs. Analysis of the alcohol with which the NaK was reacted and of the stainless steel can by the Chemical Engineering Division showed .095 mg. of uranium.

Table 101
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Smooth Rotating Plate

Speed (RPM)	Side Clearance		Maximum Head Across Seal (inches of fluid) $P_2 - P_1$	Head Developed at Edge of Plate $\rightarrow P_1$ No suction applied (inches of fluid)	Power Required Maximum H.P.	Remarks
	Top (inches)	Bottom (inches)				
1344	.002	.002	144	133	.36	
1344	.010	.010	143	175	.36	
1470	.002	.002	219	204	.78	
1770	.002	.002	247	237		
1344	.002	.002	144	134		
1344	.010	.010	134	133		
1344	.013	.013	133	129		
1344	.025	.025	140	134		
1344	.033	.033	91	133		
1344	.045	.045	93	132		
1344	.002	.002	130	130		

Radially Blasted Rotating Plate

Speed (RPM)	Side Clearance		Maximum Head Across Seal (inches of fluid) $P_2 - P_1$	Head Developed at Edge of Plate $\rightarrow P_1$ No suction applied (inches of fluid)	Power Required Maximum H.P.	Remarks
	Top (inches)	Bottom (inches)				
1344	.010	.010	101 -- 130	230	.36	Tended to be unstable, and break seal
1344	.025	.025	93 -- 101	202	.38	
1344	.045	.045	102	211	.43	

This is probably more accurate than the weighing, which was done on two different balances and by remote control. The weight is also subject to error due to oxide pickup. This would show the corrosion rate to be .1% of the value found at KAPL. Further experiments are being planned to verify this point.