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LOS ALAMOS SCIENTIFIC LABORATORY
OF THE UNIVERSITY OF CALIFORNIA • LOS ALAMOS NEW MEXICO

QUARTERLY STATUS REPORT ON LAMPRE PROGRAM
FOR PERIOD ENDING MAY 20, 1961

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LOS ALAMOS SCIENTIFIC LABORATORY
OF THE UNIVERSITY OF CALIFORNIA LOS ALAMOS NEW MEXICO

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FOR PERIOD ENDING MAY 20, 1961

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CMB, CMF, and K Divisions

All LAMS reports are informal documents, usually prepared for a special purpose. This LAMS report has been prepared, as the title indicates, to present the status of the LASL LAMPRE program. It has not been reviewed or verified for accuracy in the interest of prompt distribution. All LAMS reports express the views of the authors as of the time they were written and do not necessarily reflect the opinions of the Los Alamos Scientific Laboratory or the final opinion of the authors on the subject.

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LAMPRE-I

STARTUP

Introduction

The startup procedure for LAMPRE-I has been divided into three distinct stages. The first stage is concerned with the attainment of criticality and making various measurements at 160°C , at which temperature the sodium is liquid but the fuel is solid. The second stage is the approach to hot critical and the determination of the critical loading and other characteristics of the system at operating temperatures. The final stage will involve the increase from low power to the full operating power of the reactor.

Measurements at 160°C

The first stage of operation of LAMPRE-I at 160°C was completed during the current reporting period, electrical heating of the sodium being used to maintain (or vary) the temperature. Period measurements made by displacing the No. 3 rod above and below its critical position (7.1 in. insertion at 160°C) showed the rod to be worth 25.51¢/in. in the vicinity of the critical position. The critical mass measurements indicate the rod is worth 37.65 g/in. in the same region. Thus, the mass reactivity calibration is 0.68¢/g. From the reactivity calibration of the No. 3 rod a calibration of the neutron detectors in terms of

multiplication was obtained and, from data on the spontaneous fission source, a low-level power calibration has been determined.

A measurement of the temperature coefficient in the range 160°C to 180°C was made by observing the displacement of the No. 3 rod required to maintain the neutron level constant at a high multiplication as the sodium temperature was varied. The rod insertion was plotted against the core outlet temperature and from the slope of the curve and the known rod worth, the temperature coefficient at 160°C was found to be -1.32% .

The shim-down extrapolated mass was measured in order to determine the safe limit on future reactor loadings in the event an emergency arises which requires a shutdown to room temperature. In performing this measurement, extrapolated critical mass measurements at intervals on the shim position were taken as the core loading was increased from 136 capsules to 154 capsules (26.232 kg Pu).

With regard to the cold shutdown limit on core loading, a further correction of +0.22 kg is applied to the 160°C shim-out case to allow for the error introduced by extrapolating from an actual loading that is 1.6 kg below critical. The net critical mass of 27.72 kg is obtained for the shim-down case at 80°C with the oscillator rod inserted. Sodium worths at 80°C of 2.94 kg and 3.88 kg were calculated for the shim-up and shim-down cases, respectively. These compare favorably with the 3.1 kg and 3.9 kg reported in LA-2327 from calculations of the corresponding cases at 20°C .

Hot Critical Results

In the approach to hot critical the electrical heaters were first used to raise the system temperature from 160°C to 500°C . While the temperature was increasing, the core loading was 20.61 kg Pu (121 capsules), and the control elements (shim and rods) were maintained in their down positions. Multiplication measurements, inferred from the count rates of two counters, were obtained as the core temperature increased and are plotted in Fig. 1. It is seen that a marked increase in multiplication occurred as the core melted. This was due to (1) the increase in density of Pu that occurs at the melting point (see

Fig. 1. Multiplication versus temperature and time during temperature increase of LAMPRE-1 from 160° to about 500°C

Fig. 2.15, LA-2327), and (2) the effective increase in core fuel density that is caused by the slumping of the fuel into the clearances required during the initial insertion of the fuel slugs into the capsules. The time variation of the multiplication in the vicinity of the fuel melting point is also shown in Fig. 1.

Data obtained from the multiplication curves at 500°C are tabulated below:

Average temperature coefficient in solid state	-1.3¢/°C
Total reactivity change during melting	+45.60
Temperature coefficient during melting	+41.50/°C
Rate of reactivity addition during melting	+41¢/min
Average temperature coefficient in molten state	-4.7¢/°C
Initial temperature at start of melting	416°C
Temperature at end of melting	424°C

After the fuel had been melted the core loading was increased to 23 kg (135 capsules) yielding a shim-up, rods-up multiplication of about 100. At this multiplication there was adequate sensitivity for the observation of any reactivity changes which could be produced by jarring the capsules to move (or remove) the bubbles and voids usually observed when the fuel is first melted. Jarring was produced by raising individual fuel capsules 0.5 to 1.5 in. with the capsule charger and releasing them so that they fell to their original position on the capsule locator plate.

The results obtained are summarized below :

<u>Core region</u>	<u>No. of capsules dropped</u>	<u>Estimated reactivity gain</u>	
		<u>1st drop</u>	<u>2nd drop</u>
Central	20 out of 37	7.0¢	4.0 ¢
Intermediate	46 out of 54	50.0¢	19.3¢
Outer	39 out of 44	36.7¢	9.8¢
Total reactivity gain			\$1.27

Of the 135 capsules in the core, 27 could not be dropped owing to frictional contact with adjacent elements. It is seen in the table that in

the second pass around the core the reactivity gained by jarring was significantly less than in the first pass. Consequently, it was concluded that remaining voids or bubbles in the capsules, if any, should be in a relatively stable configuration. In addition, radiographs of capsules removed from the core indicated very little void content in the bulk of the fuel. However, a foamy or frothy region, approximately 1 in. in height, was seen at the top of the fuel in the radiographs.

The reactor was first brought to hot critical on March 27, 1961. Extrapolated mass data taken during the approach to critical yielded a critical mass of 23.37 kg Pu (137 capsules) at a core temperature of 480°C for the case of the shim plus three rods, all inserted to their standard up positions. This may be compared to the mass of 22.98 kg obtained at 160°C for the corresponding case. S_4 calculations predicted that the critical mass should have been reduced by 0.17 kg in raising the temperature from 160°C to 480°C, but the effect of the frothy region in the capsules was not allowed for in these calculations.

Zero Power Measurements at 480°C

In the approach to the final loading of 139 capsules (23.68 kg) extrapolated M_c vs rod position curves were taken on rods No. 2 and 3, and an extrapolated M_c curve was also measured for the upper 4 in. of shim travel. When the loading was critical, the period measurements were repeated to obtain mass and rod reactivity calibrations. These results are compared below with those obtained earlier for LCX III (cold, dry LAMPRE-I) and at 160°C:

<u>Case</u>	<u>Rod No.</u>	<u>ℓ/g</u>	<u>Total mass worth of rod (g)</u>	<u>Total reactivity worth of rod (β)</u>
LCX III	3	0.62	370	2.29
160°C	3	0.68	328	2.23
480°C	3	0.75	330	2.47
480°C	2	0.57	393	2.24

It is seen the average of the two β/g values at 480°C agrees closely with the measurements at 160°C , and the total reactivity worths for all four cases are in reasonable agreement. Accordingly, the results for the two rods at 480°C were averaged to obtain a reactivity vs rod position curve. The maximum reactivity point was found to occur at 15 in. of rod insertion at 480°C . This point, which was at 16 in. of insertion at 160°C , was evidently shifted downward due to the relative expansion of the core and rod supporting materials during the temperature increase. Stops have been placed on the rod drives to limit the insertion to 14.5 in.

The extrapolated mass curve on the upper 4 in. of shim travel (8.25 to 12.25 in. insertion) was taken to verify the initial reactivity depression rate available during scrams. This rate was found to be $\$1.29/\text{in.}$ of shim withdrawal at the normal fully inserted up position (12.25 in. console indicator reading) of the shim. At 160°C the indicated shim-up position was 12.67 in., and at 80°C it was 12.85 in. These changes in shim-up indicated position result from the downward thermal expansion of the reactor vessel, whose bottom end determines the upper limit of shim travel. Any actual shift in the relative positions of shim and core is much smaller than the indicated variations in the up position of the shim.

A temperature coefficient measurement was made by varying progressively the sodium temperature --as read by the vessel inlet thermocouple--from 400°C to 433°C , then to 500°C and back to 480°C . Reactivity was held constant (at 10 watts power) by manipulating the No. 2 rod. A plot of the reactivity vs temperature was made and from this the temperature coefficient of $-3.21\beta/^{\circ}\text{C}$ at 430°C was obtained. The results were taken from the rising branch of the curve, i.e., from 433°C to 500°C , where the temperature was brought to equilibrium at each point.

An apparent hysteresis was observed in the reactivity vs temperature curve. This could be accounted for by variations in temperature distribution within the system while the core temperature was changed. Furthermore, when the temperature was being raised after cooling to 433°C , there was a reversal of sign in the trend of the reactivity with temperature at 437°C . It is

possible that some of the fuel had started to freeze, but temperature instrumentation in the core is not sufficiently accurate to confirm or disprove this suggestion.

Reactor Operation up to 50 kw

Power coefficient measurements were performed over the power range of 0 to 45 kw for sodium flows, 33, 67, 133, and 170 gpm. The measurements were made by turning off the electrical heaters sequentially and attempting to hold the vessel inlet sodium temperature constant by suitable adjustment of the nuclear power. In practice difficulties were experienced in obtaining good data because of the time required to reach equilibrium, particularly at low flows and because of inadequacies in the instrumentation. The observations indicate that there are no significant positive power coefficients contributed by fuel element or rod handle bowing effects as discussed in LA-2327.

A series of power demand experiments, initiated by turning off electrical heaters on the sodium loop and/or opening the exhaust vanes on the air heat dump, demonstrated the effectiveness of the negative temperature coefficient in controlling power transients in the 0 to 50 kw range of operation. Three extended runs of the reactor at 50 kw for 38, 61, and 19 hours have been performed. During these runs an average reactivity loss rate of 0.9¢/hr has been observed. The total 7.3 Mw-hr of operation accumulated to date has produced a total reactivity loss of about \$2.00. It is speculated that the reactivity loss may be due to holdup of fission product gases in the fuel and/or changes in the configuration of the frothy region above the bulk fuel region. It has been estimated that at 50 kw gas holdup could be expected to cause up to 0.8 ¢/hr reactivity loss. Future operations at increasing power should provide additional data for evaluating this phenomenon.

The 50-kw transfer function data are plotted in Fig. 2. The calculated zero power and 50 kw gain and phase shift curves were computed by means of a FORTRAN code, RIF. Power demand experiments were made for

Fig. 2. 50-kilowatt transfer function for LAMPRE-I

several different powers and the results for 50 kw are shown in Fig. 3. In addition to the power response curves, plots of the core inlet and outlet temperatures are shown. Estimates of the power coefficient were obtained by comparing the net change in power with the net change in reactivity inferred from the inlet temperature. The results are as follows:

<u>Demand (kw)</u>	<u>Average Power Coefficient (β/kw)</u>
10	0.58
18	0.59
30	0.61
50	0.56

In addition to the continuous monitoring of the reactivity during the runs, the 100 watt critical point was checked during startup and shutdown of each run. These check points provide an inventory of the overall reactivity change and confirm the power coefficient data given above.

SHIELDING

Inadequacies in the control room shield wall were discovered as soon as the reactor power was increased to a few kw. Fortunately, most of these shield weaknesses were the result of improperly plugged conduit penetrations, etc., and the deficiencies have been, for the most part, rectified.

Shielding beneath the reactor, which was thought to be the weakest link in the whole biological shield, appears to be much more effective than expected. The fuel transfer area above the reactor, however, is under-shielded. Exact source of the neutron leakage is not known, but it may be the result of too high temperature (and consequent water loss) from the main concrete shield plug. Methods for improving the top shield are being investigated. Suggestions to date all involve much tedious fitting of various supplemental shield materials around existing piping, etc. A test of sodium system shielding was made after a 40-hr run at 50 kw reactor power. The shielding was found to be adequate.

Fig. 3. 50-kilowatt power demand from 1 kv and core inlet and outlet temperatures

FUEL PIN SCAN

Fuel pin No. 687 was withdrawn from the reactor after a run of a few hours at 100 watts of power. It was scanned with a 3-in. NaI crystal scanner, and pulse height distributions were taken every inch along the length of the capsule. The scanner was set up to count gamma rays whose energy was within desired limits and the following conclusions were reached:

- 1) The fuel had not circulated.
- 2) The effective length of the fuel was 6.20 in.
- 3) There was no detectable Ta¹⁸² activity.
- 4) La¹⁴⁰ was concentrated 0.60 in. from the top surface of the fuel.
- 5) Above the lanthanum were segregated some unidentified fission products.
- 6) Xe¹³⁵ was found in the gas phase.
- 7) The ends of the fuel exhibited higher than expected activity.

REACTOR SYSTEM

Sodium System

During this reporting period the sodium system was operated between 155 and 180°C for the warm critical experiments, and then the temperature was raised to the 450-500°C region for hot critical measurements and approach to power experiments described above. The only serious malfunction of the system during this time was a short-circuit in the primary winding of the heating transformer which supplies some 20 kw of heat to the sodium. The short was caused by transients occurring when the transformer was turned on at full power. The damage has been repaired and adjustable autotransformers installed in such a way as to prevent full voltage from being applied to the primary suddenly.

An attempt to measure the sodium transit time of the system at two different flow rates was performed by introduction of a thermal transient at the heat exchanger. Although the observations were consistent, the transit time was about twice that calculated from the 3-in. flowmeter and

system volume. The calculated sodium flow necessary to cause the pressure drop observed between the reactor vessel and the surge tank is in agreement with the flow as read from the electromagnetic flowmeter. A third check on flow may be obtained from the reactor power data. The power, as calculated from temperature rise across the core and flow rate, has been shown to be in good agreement with the value determined from the fission chambers. From the data available, it is concluded that the 3-in. electromagnetic flowmeter is correct and that some error is involved in the transit time measurement.

A test was made to determine the time before loop freezing of the cold trap system if the cold trap loses power. It was found that the system did not freeze after 10 min of pump-off operation, but that a plug had formed after 15 min.

Data gathered from the operation of the system during the past months were examined to determine the pump operation characteristics. It appears that over-all operating efficiency is about 3% at rated flow of 133 gpm and that at this flow there are about 10 kw of heat put into the sodium by the pumps.

Capsule Charger

The lead shielding for the capsule charger is essentially complete. Two removable lead rings will be placed around the housing before removing capsules of high activity. Additional lead may be required at several points and will be added, based on future radiation surveys during operation. The tunnel viewing window now has a reading of ~ 70 mr/hr with a 20 curie capsule in the back of the tunnel.

The periscopes for the fuel transfer system have been modified to use a 6X monocular in place of a 4X galvanometer telescope to facilitate location of a rod handle. Five hot capsules were removed during May, using the completely remote and shielded procedures. Design has started on facilities to remove a fuel element and attached handle through the top of the capsule charger remotely without loss of cover gas.

Capsule Examination

Caves for the puncturing, gas collection, sectioning, and metallurgical examination of LAMPRE I capsules after removal from the reactor have been completed. A new type of etching device was developed which permits direct acid etching of specimens without introducing acids inside the box. The examination of irradiated capsules will begin as soon as shakedown trials have been completed.

MATERIALS TESTING

It has been found that Type 17-4 PH stainless steel will undergo additional grain boundary precipitation during long-term exposure to temperatures considerably below the hardening temperature. Thus, material in the H-1100 condition exposed at a temperature of 800°F will continue to precipitate until it has properties of material heat treated at 875°F or below. This is the situation which exists in the core region of LAMPRE I, affecting the reflector pins, the lower portion of the rod handles, and the fuel element adaptors. When these parts were put in the reactor they were believed to be in condition H-950. The property changes involve increase in strength, increase in hardness and decrease in ductility and impact strength.

Measurements of the properties of 17-4 PH stainless steel were obtained as follows:

<u>Item</u>	<u>Impact*</u> <u>Strength</u> <u>(ft-lb)</u>	<u>Ultimate</u> <u>Tensile</u> <u>Strength</u> <u>(psi)</u>	<u>Elongation</u> <u>(% in 2-in.)</u>	<u>Hardness</u> <u>(R_c)</u>
Reflector Pins	12.0			
Condition H-950	12.0			45

	3.5			
Reflector Pins	2.0			
Condition H-950	6.5	190,000	7.6	
Exposed to Na at	2.5		6.2	45
Temperatures up	2.0		3.6	
to 500°C	2.5		5.6	

	19.0			
Rod Handle	8.5			
Condition H-950	8.0			45
	3.5			

Handbook Value	30.0	180,000	14.0	40
"ARMCO"				

* Izod impact test using modified Type Z specimen 0.400 in. in diameter.

It should be noted that all tests were performed at room temperature. Under service conditions (approximately 450°C) the material is undoubtedly less brittle than these data indicate.

FUEL AND CONTAINER DEVELOPMENT

FUEL ALLOY PROGRAM

LAMPRE-1 Fuel

A study of the history of fuel materials used in many comparison tests has revealed that different production methods were used in what were thought to be identical lots. A recent effort has produced much greater understanding of these variables, and many former conclusions must be re-evaluated.

It has been demonstrated that any detectable quantities of calcium and magnesium in the fuel alloy can be a major contributor to corrosive failure of the containers. Manufacturing methods for fuel have been developed which reduce these impurities to a low level (< 10 ppm) with greatly improved test results. The desire to retain as large as possible a carbon content in the fuel precluded the use of these methods earlier. Although carbon is still regarded as a desirable additive for operation at temperatures above 700°C , it has been removed from the list of specified constituents for the present. Tests have shown that up to at least 625°C carbon addition confers no benefit, and may even be a detriment. In the present stage of reactor operation the fuel temperatures are mostly below 625°C , carbon solubility is low, and excess carbon or PuC tends to collect on the surface of the fuel, hindering the disengagement of gas bubbles. Although testing is not completed yet, it appears that silicon addition may provide equally good high-temperature protection at levels which will remain soluble in the fuel at the operating temperature. Since silicon also confers some protection against melt-freeze damage, it would be preferred over carbon if it proves to be effective.

A testing temperature of 625°C seems to give the most corrosive condition for the impure fuel presently in the reactor. At higher temperatures the carbon additive becomes effective, and at 525 and 575°C the fuel does not attack the container.

Ternary Alloy Fuels

Work has been continued on determining details of the ternary plutonium-cerium-cobalt phase diagram. Success was achieved in identifying some of the four-phase reactions in the system. The existence of the ternary eutectic $L \rightarrow \eta + \delta\text{-Pu} + \beta\text{-Ce}$ at the Pu-rich side of the diagram at 408°C is now well established, and another ternary eutectic $L \rightarrow B + \lambda + \beta\text{-Ce}$ is believed to exist at the Ce-rich side at about 420°C . The $L + A \rightarrow \eta + \beta\text{-Ce}$ reaction at 425°C , the $L + \theta \rightarrow \eta + A$ reaction at 443°C , and the peritectic reaction $L + K + \theta \rightarrow C$ at 480°C seem quite certain. Reactions that are not yet fully proved are: $L + C \rightarrow A + B$ at $\sim 450^\circ\text{C}$, $L + K \rightarrow B + \lambda$ at $\sim 440^\circ\text{C}$, $L + A \rightarrow B + \beta\text{-Ce}$ at $\sim 445^\circ\text{C}$, $L + \epsilon + C \rightarrow A$ at $\sim 450^\circ\text{C}$, and $L + K + C \rightarrow B$ at $\sim 480^\circ\text{C}$.

National Research Corporation has produced a quantity of Co-Ce using a special alloying procedure specified by IASL. This material will be used for direct contact loops studies, and, by remelting with plutonium, for corrosion studies also. The development of a commercial source of supply will relieve a large burden upon laboratory furnace facilities.

Analytical methods for Co-Ce alloys have been developed, including the first accurate method for oxygen determination in cerium. Cobalt has been determined by an x-ray absorption edge technique in which secondary x-radiation from a Cu-Ni target was transmitted through a solution of Co-Ce alloy and the K-absorption of the Co measured. Linear results were obtained with Co concentrations between 0 and 10 g/l. Relative coefficients of variation of 1.8 and 0.6% were obtained for Co concentrations between 1 and 5 g/l. Ce and U do not interfere.

CONTAINER ALLOY DEVELOPMENT

LAMPRE I Capsules

Failure of LAMPRE capsules in certification tests over the past several months fall into three general classes:

- 1) Spotty failure below the liquid level, often in lines or rows of spots corresponding to the direction of ironing of the capsules.

2) Attack at the fuel level or in the gas phase just above, usually in bands extending most of the way around the capsule.

3) Diffuse failure at a narrow band just below the weld at the top of the capsule.

Type 1 failure, if it occurs at all, seems to appear in a few hundred hours. Bend tests and metallography have shown that the metal nearby is in good condition. The failures have been attributed to residual fabricating stresses, local concentrations of impurities in the original metal, or to pick-up of traces of contaminants from the fabricating dies, etc. A major discovery of the current period has been the demonstration that higher annealing temperatures for the capsules give progressively higher resistance to type 1 failure. Distinct improvement has been shown for each temperature as annealing was done at 1050, 1200, 1350, and 1450°C.

Type 2 failure seems to be strongly correlated with the presence of impurities in the fuel, especially calcium and magnesium. It has not been observed so far in testing high-purity fuel. This type of failure has also been shown to be diminished by higher annealing temperature for the capsules.

Type 3 failure takes longer to appear, and is not usually observed unless the capsule is resistant to the other forms of attack. It is related to the ability of impurities to make the fuel climb up the wall of the capsule. It has not been observed in high-purity fuel, which does not climb the wall in a static test. Since the heat-affected zone is a region of abrupt change in metallurgical structure, higher annealing temperatures should reduce the sensitivity of this region. Capsules annealed at 1450°C, which are characterized by complete recrystallization and extensive grain growth, have been tested up to 2000 hr so far without failure of any kind, even though calcium and magnesium were present. Further tests are under way using annealing temperatures from 1200 to 2200°C.

Tests of a group of capsules subjected to a variety of surface treatments, such as polishing, bright-dipping, grinding, etc. show no marked effect of surface condition upon corrosion resistance.

Tantalum Alloy Tests

Small scale testing of hundreds of specimens containing various amounts of binary alloying agents with tantalum has been extended over a wide range of elements. So far this work has yielded only two additives which show a promising improvement in corrosion resistance, namely, yttrium and tungsten. Both additives seem to function by producing a finer grained tantalum with a higher recrystallization temperature, accompanied in the case of yttrium by an increased strain hardening rate. The beneficial effect of yttrium has been confirmed several times in both large and small ingots, and persists after remelting of the ingot, although the element does not remain in the metal. Capsule production is now proceeding from a large ingot containing yttrium and tungsten additions of 0.2 and 0.7% respectively. Tungsten additions up to 10% are under study in other ingots.

Apart from the effect of the additives the corrosion resistance of tantalum has improved as higher purity was attained. Many metallic elements are believed to be relatively innocuous in small quantities, but the interstitial impurities H, N, C, and O have strong effects on fabricating properties. Oxygen is definitely deleterious to corrosion resistance, and is a key agent in one of the most promising hypotheses for the mechanism of plutonium attack on tantalum. Large amounts of carbon are also detrimental to corrosion resistance, but amounts of this element in the range of 25 to 60 ppm are tolerable. Hydrogen and nitrogen in the tantalum do not appear to affect corrosion resistance.

Spectrophotometric Determination of Tungsten in Tantalum

In analytical technique development, tungsten has been determined when present in tantalum in concentrations as low as 200 ppm by complexing with thiocyanate and measuring the absorbance at 400 mμ. Tantalum and fluoride interferences are eliminated by complexing with tartaric acid and boric acid. Of the 27 elements studied, interference was caused only by Cu, Pt, Rh, and V. It is believed the method can be used for lower W concentrations.

DIRECT CONTACT CORE STUDIES

A new loop for liquid fuel hydraulic studies has been completed and operated. This loop, known as Core Development Experiment No. 5, uses a Wood's metal alloy as the fuel and hot water as the propelling agent. The density ratio and hydraulic properties of this system are similar to the cobalt-cerium-plutonium and sodium system. Several runs with the loop have led to the development of improved phase separators, which can be easily replaced. Studies of mercury-water jets have also continued in other loops.

SODIUM TEST FACILITY

OPERATING EXPERIENCE

Summary

The sodium test facility was operated continuously this quarter except for six maintenance and inspection shutdowns resulting in 64 idle hours. As of May 20, 1961, the following total running hours have been logged on major items of test equipment:

<u>Item</u>	<u>Total Operating Time (hr)</u>
Steam Generator	7,004
Intermediate Sodium Heat Exchanger	9,165
Primary Loop Sodium Pump	9,165
Secondary Loop Sodium Pump	10,324
Gas-Fired Sodium Heater	8,124
Sodium Cold Trap System	10,412
Air-Operated Sodium Flow Control Valves:	
Primary Loop (two valves)	9,165
Secondary Loop (two valves)	10,324

Operation of the steam generator, intermediate sodium heat exchanger, sodium pumps and other sodium components being tested has been entirely satisfactory with no maintenance required on these items. Heat transfer studies are continuing in order to determine the characteristics of all heat exchanger components under varying conditions of sodium flow, steam flow, sodium temperature, and steam temperature. The final results of these studies will be presented in a future report.

Sodium Pumps

The gas-lubricated shaft seal for the primary pump has been received and is ready for installation. It is planned to install this seal assembly during the month of July, at which time the loop will be shut down for maintenance work on the gas-fired sodium heater.

Variations in the monthly oil consumption by the primary pump indicate that a foreign particle may become lodged between the seal faces, causing increased oil leakage. After the particle is dislodged, it appears that the seal faces again wear in and bring the leakage rate back to normal.

Sodium Cover Gas Purity

Helium used as a sodium cover gas in both primary and secondary loops has been analyzed to check for build-up of oxygen, hydrogen, and methane. The cover gas in the secondary loop has not been vented since startup of the facility (about 10,000 hr operation). Helium in the primary loop was not vented for roughly 3600 hr prior to conducting the gas analysis. Results of the gas analysis were as follows:

	<u>Primary Loop Helium</u>	<u>Secondary Loop Helium</u>
Oxygen	Undetectable	< 3 ppm
Hydrogen	Undetectable	< 50 ppm
Nitrogen	40 ppm	300 ppm
Methane	Undetectable	< 15 ppm

The contaminants found in the analyses are believed to be well below tolerable limits for the test facility. It appears that operation of the continuous cold trap system has achieved satisfactory removal of oxygen and hydrogen from both the sodium systems.

CORE TEST FACILITY

The basic design data book for Phase A has been completed and was submitted to LAAO on March 23, 1961.

Proposals have been received from 12 engineering firms. These proposals have been evaluated and submitted to a selection board composed entirely of AEC representatives. Choice of the A-E now rests with this selection board.

The general configuration of the 1000 gpm and 3000 gpm sodium systems has been established. Preliminary piping stress calculations have been completed and layout drawings are now being prepared.

The surplus SEAWOLF electromagnetic pumps have been received. Their general condition and their suitability for use in the Core Test Facility are being investigated. Inquiries to determine the cost of suitable power supplies have been initiated. Mechanical changes necessary to permit these pumps to operate safely at 1100°F are being considered.

FUEL PROCESSING

Pyrometallurgical Reprocessing

The first experimental run with neutron irradiated Pu for the investigation of pyrometallurgical reprocessing by volatilization and liquation has been made, although no analytical results are yet available. The procedure was to melt in a vacuum 82 g of Pu irradiated to an estimated burn-up of 0.1 to 0.2% and to alloy it with sufficient Fe to make the eutectic composition of Pu-10 at/o Fe. This was held in the molten state at 600°C for 46 hr, and then an attempt was made to filter it through a Ta frit. A water-cooled condenser in the off-gas stream was designed

to collect those materials (having volatilities similar to Cs) leaving the melt. The only apparent failure in the course of the run was an inability to filter the melt at the conclusion of the experiment. The reasons for this are not clear and are being investigated.

Solvent Extraction Recovery

Work is in progress for the development of a solvent extraction method for the recovery of Pu from various pyrometallurgical processes. The solvent under study is a mixture of a secondary amine (IA-1) and tetrachloroethylene (TCE). Recent experimental work has been directed toward the development of a more efficient stripping solution (0.3M HNO₃) than has been previously used. A Fe⁺⁺ stripping cycle is presently being evaluated. The first attempt was made with a solution having the same H⁺ and NO₃⁻ concentration as the feed, to prevent Zr and Ru stripping, but the Pu stripping efficiency was not particularly high and considerable β-γ activity also stripped out. There are indications that a Fe⁺⁺, low H⁺, low NO₃⁻ strip has a high Pu stripping efficiency. If unacceptable amounts of β-γ activity are also stripped, as is expected, it will be necessary to remove the troublesome components from the solvent prior to stripping.

Disposal of Organic Residues

A suitable method for the disposal of organic residues has been developed. Briefly, it involves the absorption of the waste solvent in vermiculite and the forced evaporation of the volatile components by heating and blowing air through the mixture. The final disposal of the waste container will be by a method identical with that previously reported for aqueous waste disposal.