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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 AUGUST 20, 1961

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Printed in USA. Price \$.50. Available from the

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

LAMS-2620
REACTOR TECHNOLOGY
(TID-4500, 16th Ed.)

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

REPORT COMPILED: September 1961
REPORT DISTRIBUTED: September 29, 1961

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FOR PERIOD ENDING AUGUST 20, 1961

Prepared from material submitted by
members of CMB, CMF, and K Divisions

Contract W-7405-ENG. 36 with the U. S. Atomic Energy Commission

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.

LAMPRE-I

OPERATIONS

Introduction and Summary

All basic experiments planned for the LAMPRE I startup program were completed during this report period. The tests included operation at 100 kw and 200 kw utilizing both normal coolant flow (133 gpm) and half flow, and full flow operation at 400 kw. At each power level, transfer function measurements were made, and a continuous run of approximately 60 hr duration was carried out to determine the characteristics of the reactivity loss first observed at 50 kw. Various power demand tests were made in order to measure the power coefficient and to determine suitable standard procedures for achieving a specified power.

When this experimental program had been finished, the reactor was shut down but kept at a temperature of $\sim 480^{\circ}\text{C}$ by means of the electrical heaters. Since that time, work on the LAMPRE I system has consisted of necessary improvements to the reactor shielding and instrumentation. The scheduled date for resumption of operations is October 1, 1961. At that time, it is planned to begin continuous operation at an initial level of one-third design power.

Startup Experiments

In Experimental Plan Number 5 (EP No. 5), "Power Ascension in LAMPRE I," a series of power demand experiments were performed using the air heat dump to supply the demands. Transfer function measurements were performed at the power levels 50 kw, 100 kw, and 200 kw, for both normal

sodium flow (133 gpm) and half flow (67 gpm), and at 400 kw for normal flow. A series of runs of approximately 60 hr duration was performed at each of the above power levels to examine the longer term reactivity variations. In general, the reactor exhibited excellent short term stability and performance characteristics.

Power Coefficient and Oscillator Data

In making power demand experiments, the reactor was stabilized at an equilibrium power level of about 50 kw by turning off the electrical heaters and opening the exhaust vanes on the air heat dump. Then from this level the demands were initiated by opening the intake air vanes to various settings.

Some typical results are shown in Fig. 1; the initial power was 45 kw and the demand level was 100 kw. It is seen that, in contrast to the power demands from 1 kw (LAMS-2564, Fig. 3), there is no power overshoot.

Improved experiments were conducted in which the system temperature changes were made slowly with time to reach equilibrium conditions. A new value for the isothermal temperature coefficient has been measured as $-2.88 \text{ } \mu/\text{ }^{\circ}\text{C}$. This total temperature coefficient may be broken down into contributions from the fuel, the Na, and the structure, as follows:

	<u>Derived Values</u>	<u>Calculated Values</u>
$\left(\frac{\partial k}{\partial T}\right)_{\text{Pu}}$	$-2.0 \mu/\text{ }^{\circ}\text{C}$	-1.8
$\left(\frac{\partial k}{\partial T}\right)_{\text{Na}}$	-0.5	-0.5
$\left(\frac{\partial k}{\partial T}\right)_{\text{Structure}}$	-0.4	-0.9

Fig. 1. Core inlet and outlet temperatures for 100-kw power demand starting from 45 kw

Power coefficient data for the full flow conditions using the revised value of isothermal temperature coefficients are tabulated below:

<u>Demand (kw)</u>	<u>Core inlet temperature change ($^{\circ}\text{C}$)</u>	<u>Power coefficient (¢/kw)</u>
24	-3.9	-0.47
48	-7.7	-0.47
74	-12.8	-0.51
100	-17.3	-0.50
57	-9.9	-0.51

In view of the rod worth of ~ 2.25 for each of three rods (LAMS-2564, page 6), it appears that sufficient control is available to permit reaching a power in excess of 1 Mw if the reactor is loaded so that zero-power critical is achieved upon insertion of a portion of the first control rod.

Reactor transfer function data obtained at the various power levels are plotted in Fig. 2 for a Na flow of 133 gpm. The gain curves illustrate the increase in stability as the power level increases.

A good fit to all the oscillator data obtained on LAMPRE I to date is obtained by a feedback function of the type

$$H(\omega) = c/(1 + j\omega\tau),$$

where c is a power coefficient, τ is the reactor thermal time constant and ω is the frequency in radians/sec. Effects due to the sodium are seen in a comparison of the results for full (133 gpm) and half (67 gpm) sodium flows:

	<u>Full Flow</u>	<u>Half Flow</u>
c	0.47 ¢/kw	0.63 ¢/kw
τ	1.3 sec	1.8 sec

The short time constant indicates that the principal contributions to the data must come from core parameters.

Fig. 2. Transfer functions of LAMPRE I for various power levels

Reactivity Effects

Total energy release during LAMPRE I operation to date is about 56 megawatt hours. The reactivity loss observed at 50 kw (LAMS-2564, page 8) has in general persisted throughout the operating history. Most of the total integrated power was accumulated during weekend runs at powers of 50 to 400 kw.

In general, significant reactivity loss accompanied the high-power runs and smaller changes occurred during shutdown periods. This effect is presumed to be due to gaseous fission products retained in the fuel. Radiographs and gamma scans show that there is a frothy, low-density region in the upper portion of the fuel which seems to be growing larger.

In some cases, notably after a 400-kw run, the reactivity loss continued after shutdown. A loss of 35¢ occurred during the first 14 days following the last previous operation at power on June 5, and an additional 51¢ were lost during the next 40 days. A possible explanation for this loss may be that in this interval fuel was redistributed in the capsules by the formation of froth not associated with gaseous fission products.

Detailed measurements of reactivity were made at power during the last two weekend runs. The first part of a 200-kw run followed the pattern of earlier runs - an initial period of relatively rapid reactivity loss, followed by one of flatter and apparently constant slope. During the last hours of May 27 the sodium coolant flow was reduced from normal to half value. The first reactivity measured at the lower flow was more negative (requiring rod insertion) by about the amount to be expected because of the higher core and outlet sodium temperatures. However, succeeding measurements indicated that the negative reactivity decreased during the next 3 or 4 hr, then remained about constant for approximately 10 hr, and finally increased again at about the same slope observed at full flow. This transient effect is not understood, but presumably implies that some gas bubbles were expelled by the temperature increase (about 20°C on average core temperature) and the larger vertical temperature gradient in the core. On June 1 and 2 an attempt was made to examine this transient

reactivity recovery during a short (17 hr) run at 200 kw. Both temperature and flow were varied separately, but no large recovery of reactivity was induced.

On June 2, the 17 hr run was terminated at 1200 hr, reactivity was followed at 1 kw for 5 hr (recovering about 4%), then a weekend run at 400 kw was begun. This run, unlike previous ones, commenced by regaining some 12% of reactivity during the first 8 hr. It might be supposed that during the preceding 17 hr run at 200 kw the gas bubbles had been so disposed as to be readily disengaged by the 400 kw operating conditions. Later in the run, the usual reactivity loss was observed. This run was followed by a 7 hr period at 1 kw during which about 13% of reactivity was recovered.

The observed loss of reactivity, though not the pattern noted, might be attributed to leakage of Pu out of some capsules in the core. However, when a sample of the coolant Na was examined for the presence of fission products, none were found, although the sensitivity of the apparatus was estimated to be such that 0.01 micro-curie could have been detected. (Ta and possibly Mn were found in the Na, however.) Also, a 180 cm³ sample of reactor system cover gas was similarly examined. The only radioactive species observed was identified (on the basis of gamma-ray energy and half life) as A⁴¹; A⁴⁰ has been independently found by analysis to be present in the He cover gas. Therefore, it is concluded that the reactivity loss is not attributable to fuel leakage.

Examination of LAMPRE I Fuel Capsules

During the course of the reactor startup, selected fuel capsules were withdrawn from the reactor for examination by means of radiography, gamma ray scanning, and metallographic techniques.

While the first LAMPRE core loading was being fabricated, there was evidence that the presence of a few hundred ppm of C in the Pu fuel alloy was effective, at 750°C, in reducing intergranular attack in Ta. In most of the LAMPRE I fuel capsules, an appropriate addition of C was therefore

made by inserting a small quantity of PuC into the Ta capsule together with the Pu-Fe fuel alloy. It was subsequently found that whatever the benefits associated with the presence of PuC, there were also other effects, notably a tendency for a low-density "crud" to form near the top of the fuel in a capsule. In some cases formation of this crud caused the normal density fuel to separate axially into two sections. Such crud occurred to a much smaller degree in so-called "clean" fuel containing no PuC. When reactivity losses began to occur as LAMPRE I power was raised, it was suspected that the capsules containing PuC might be in part responsible. Therefore, prior to power runs at 100, 200, and 400 kw, a pair of capsules, one containing standard Pu-cast Fe fuel with PuC, the other containing fuel made from Pu and pure Fe, without PuC, were inserted into the reactor. At the completion of the run, the capsules were removed from the reactor for examination and a new pair was inserted for the next run. Some of the results of these tests are indicated in the x-ray photographs which follow.*

The capsules shown in Fig. A and A' were inserted into the reactor prior to a 100 kw run of approximately 60 hr duration. Capsule 1094, Fig. A, contained fuel with PuC, while capsule 1194, Fig. A', contained only the so-called "clean" fuel. Clearly, the fuel in the capsule of Fig. A' is confined to a much more closely-defined region than is that in Fig. A, which has suffered an accumulation of unidentified crud of low density near the top of the fueled portion of the capsule. Either crud, or a void region, also occurs at the bottom tip of the capsule containing PuC. This bottom void probably results from the fact that the PuC was placed in the bottom of the capsule during fuel loading, and complete filling of the capsule tip did not occur when the fuel melted. Jolting the capsules against the core locator plate is effective in filling these tip voids (LAMS-2564) but this was not done with the pair of test capsules shown in Figs. A and A'.

*The "wasp waisted" appearance is a defect in the photographic reproduction.

A

A'

Fig. A. Capsule 1094, containing PuC, after 6 Mw-hr operation

Fig. A'. Capsule 1149, "clean" fuel, after 6 Mw-hr operation

Figures B and F' illustrate respectively PuC-containing and clean fuel run for a similar period at 200 kw. Figures C and C' show fuel present in the reactor during a 60 hr, 400 kw run. In all cases (Figs. A, B, C) the PuC-bearing fuel contains a low-density region at the top which contributes to core distortion and hence loss of reactivity. A comparison of Figs. A, B, and C with Fig. C'', which shows a PuC-containing capsule used in the reactor during zero power operation, indicates that this crud formation and fuel separation is more drastic when the reactor is operated at substantial power.

The PuC-containing capsule shown in Fig. D was in the reactor (at the outer edge of the core) throughout the startup operation (approximately 2-1/2 months) and for the full 56 Mw-hr of integrated power. Capsule 1172 (Fig. E) has the same history as the one of Fig. D, but was located at the center of the core.

A comparison of data obtained from La^{140} gamma-ray scans (collimated scintillation and photomultiplier, multichannel analyzer) showed that in a clean-fuel capsule (No. 1098) a major portion of the La^{140} activity is concentrated near the top of the fuel at the interface between the fuel and the small amount of crud present in this capsule. In a PuC-containing fuel a much smaller fraction of the total La^{140} activity has been concentrated at the interface between fuel and crud. It is not known whether or not similar considerations would apply in the case of gaseous fission products; an experiment to investigate this is planned, however.

FUEL AND CONTAINER DEVELOPMENT

FUEL ALLOY PROGRAM

LAMPRE I Fuel

The presence of traces of Ca and Mg in the Pu-Fe fuel alloy for LAMPRE I was reported earlier to have a major effect on the occurrence of intergranular penetration of Ta containers (LAMS-2564, page 18). This effect has been confirmed with further tests in which standard containers

B

B'

Fig. B. Capsule 1164, containing PuC, after 12 Mw-hr operation

Fig. B'. Capsule 1099, "clean" fuel, after 12 Mw-hr operation

C

C'

C''

Fig. C. Capsule 1043, containing PuC, after 24 Mw-hr operation

Fig. C'. Capsule 1098, "clean" fuel, after 24 Mw-hr operation

Fig. C''. Capsule 1076, containing PuC, used only for zero-power operation

D

E

Fig. D. Capsule 1031, containing PuC, located at outer edge of core; exposed during whole startup operation (56 Mw-h4)

Fig. E. Capsule 1172, containing PuC, located at center of core; exposure same as Fig. D

were exposed to lots of fuel containing very low and very high levels of Ca and Mg impurity. The samples were examined for failures after each 700 hr. The high levels of Ca and Mg caused failure in the first period, while the low-level samples have shown no failures so far, and are still operating after 2800 hr.

The improved behavior exhibited by the fuel with these impurities controlled, coupled with the increase in container lifetime, gives assurance for LAMPRE I operation at temperatures up to about 650°C. At higher temperatures, however, solution attack of Ta is known to occur. In the preliminary stage of fuel development for this project it was shown that the presence of C or Si in the fuel can control solution attack (mass transfer) up to about 900°C. An increasing amount of effort is now being devoted to these additives, especially Si. The most time-consuming part of these tests is verification that the additive does not increase intergranular attack at moderate temperatures. With high-purity fuel and improved containers a comparative test of this type, now under way, is expected to require many months for a definitive answer. Meanwhile high-temperature tests, assessment of viscosity effects, and mass transfer tests are under way. The higher temperatures are used to accelerate corrosion, and thus to guide the long-term tests at operating temperatures.

Dilute Fuels

The Pu-Fe fuel of LAMPRE I contains about 14 g/cm³ of Pu. Although a large-scale reactor using such fuel is potentially possible, it is at present considered more promising for high-performance reactors to explore the use of Pu in a diluted but still mobile form. A literature search has been made to find alloy systems which might have potential value as bases for low-Pu fuels. In more than 350 binary systems, liquidus temperatures at 600°C or lower extend over sufficiently large composition ranges to offer the possibility that enough Pu might be dissolved to make ternary fuel alloys containing about 5 at/o Pu. However, when the criterion of compatibility of the alloy components with Na is considered, the number of

attractive binary systems is greatly reduced. Perhaps a dozen may be worthy of further investigation, but only two seem to offer outstanding possibilities: the Al-Mg system, which has a eutectic at 437°C and $\sim 70\%$ Mg, and the Ba-Mg system which has a eutectic at 358°C and 65 a/o Ba.

Plutonium-Aluminum-Magnesium Alloys

Metallographic examination of an alloy containing 5 a/o Pu, 28 a/o Al, and 67 a/o Mg was performed as a preliminary step toward evaluating the alloy for use as a dilute Pu fuel. Evaporation of some of the Mg during alloy preparation made the composition questionable and the cast ingot was microstructurally inhomogeneous. However, the predominant microconstituent was found to have a eutectic structure and the system thus offers some possibility of providing useful fuel alloys.

Plutonium-Cerium-Cobalt Alloys

Pu-Ce-Co alloys, mentioned in previous reports, are perhaps the best known ternary systems containing Pu. Corrosion tests in Ta have been made with a series of such alloys containing 1, 2, 3, 4, and 5 g/cm³ of Pu. At 750°C none of the tests failed after 1000 hr, although the concentrated Pu-Fe fuel survived less than 200 hr under these conditions.

Cerium-Cobalt Alloys

The composition of the compound given as Ce_3Co in Hansen's "Constitution of Binary Alloys" has now been determined, by means of single-crystal studies, to have the more probable composition of $\text{Ce}_{48}\text{Co}_{22}$, which corresponds to 31.4 a/o Co. Thermal analysis and microstructural data indicate that the compound decomposes peritectically rather than congruently (as shown in Hansen's diagram) but the data are not entirely consistent.

Plutonium-Uranium Base Alloys

Another diluent being explored is U. Uranium-base alloys have much higher melting points than those in the Ce system, but some success has been obtained in their containment in preliminary tests. Calculations have shown that such fuels in a direct-contact core would yield a very high breeding gain when used in a one-region reactor. In such a concept, processing of fuel and blanket could be combined.

Detailed investigations of multi-component systems are obviously exceedingly difficult and much too time-consuming to offer a practicable approach to disclosing possibly useful alloys. Instead, spot checks will be made of various systems to gain some idea of the direction that the effort should take. It is likely that U will have to be fluxed with one or more of the transition metals, e.g., Mn, Fe, Co, Ni. Further additions should include metals which do not form strongly bonded, high-melting compounds with the other components. In this respect such metals as Zr and Ti may be useful.

Some preliminary results obtained for multi-component alloys containing Pu and U are described below.

The addition of less than 10 a/o Pu to U-Mn alloys appears to be sufficient to obliterate the eutectic microstructure that is characteristic of the binary alloys. Interpretation of the microstructures of the ternary alloys, however, must await x-ray diffraction or other identification of the three phases which are consistently present after ingots have been heat treated at temperatures up to 700°C. Liquidus and solidus temperatures have not been established but the percentage of liquid present in the alloy at 700°C seems to increase with increasing Pu content. Therefore, alloys having compositions lying in the liquidus valley might be quite fluid at 700°C, if the Pu content exceeds about 15 a/o.

A quaternary alloy containing 5 a/o Pu, 74 a/o U, and 17 a/o Mn, and 4 a/o Ce was so inhomogeneous that it may be miscible to a limited extent in the liquid state.

Oxide Pastes

The use of a paste of UO_2 in Na has been proposed for the blanket in LAMPRE type reactors. Recently improved methods of calculation indicate that for large reactors an attractive breeding gain is possible using UO_2 - PuO_2 paste for the core as well. Methods of obtaining motion of the paste within the core or blanket are the principal uncertainties with this material.

CONTAINER DEVELOPMENT

LAMPRE I Capsules

The use of a higher annealing temperature for finished Ta capsules has continued to provide major improvements in corrosion resistance. As mentioned in LAMS-2564 (page 18), distinct levels of improvement are exhibited at 1050°C , 1200°C , 1350°C , and 1450°C . Tests are under way of capsules annealed at still higher temperatures, but so far no capsules have failed, regardless of the kind of fuel, if annealed at 1450°C or above. Several more months may be required to show any differences. Both high-purity Ta and Ta-W alloy exhibit this greatly improved behavior upon annealing.

The capsules used for the first loading of LAMPRE I were all annealed at 1300°C . As these were being placed in the reactor, a set of capsules from the same stock but annealed at only 1050°C was loaded with identical fuel to that in the reactor. These have had 7 months exposure to low temperatures, but exceeding those experienced by the reactor loading. It is expected that the failures encountered with this set, having lower annealing temperature, should be at least as severe as could be expected from the reactor loading. These capsules have survived 2520 hr at 525° , followed by 2125 hr at 575° , and are now being exposed at 625°C . So far there have been no failures. The reactor loading itself has been molten for about 3600 hr, mostly at 480°C , with excursions up to about 525°C for the central capsules during operating periods. Two capsules have been removed so far and sectioned for metallographic examination; neither showed any corrosive attack of the Ta.

Alloy Development

Tests have been made on Ta and Ta-W alloys to determine the degree of reduction by rolling which establishes a condition of "critical strain." Test results show 15 to 18% reduction is sufficient to create the condition in pure Ta. For alloys containing 0.1, 1.0, and 3.0 w/o W, the amount of reduction apparently diminishes with increasing W; further work is necessary to verify these results.

Capsules prepared from various experimental Ta alloys are under test. These include additions to the ingot of Y, Hf, and varying amounts of W from 0.1% to 10%, as well as Th + W and Y + W. None of these experimental capsules showed leakage after the first month's exposure, and the test is continuing.

The improvement in fuel and containers recently indicated will be confirmed by placing fuel elements in the reactor and by further laboratory testing. The projected core lifetimes are now sufficiently high, however, to assure that LAMPRE I can be operated as designed. Study of radiation induced corrosion, if found, remains one of the objectives of LAMPRE I operation.

Containers for Dilute Fuel

The work on containers for dilute fuel is still exploratory, but preparations are being made to expand this effort as testing facilities are released from the LAMPRE I materials development. As mentioned above, the dilute fuels based on Ce appear to be less corrosive toward Ta than are Pu-Fe, while for the U-diluted systems Y and Ta are both promising. Ceramic container materials are receiving some attention. Crucibles of CeO_2 , Al_2O_3 , Y_2O_3 , CaO , and ZrO_2 have been made for experimental evaluation in this connection. The greatest effort at present is directed toward preparation of pure starting materials, developing analytical methods and handling techniques, and revising some of the furnaces for higher testing temperatures.

DIRECT CONTACT CORE STUDIES

The information obtained from a detailed investigation of the behavior of jet and lift pumps using mercury and water, already reported, makes possible the design of such pumps for any application. The loop (Experiment No. 5) using Wood's metal and hot water, mentioned in LAMS-2564 (page 20), has been used to study methods of separation and collection of two immiscible fluids after dispersion in a jet nozzle. More than 15 different configurations have been set up and operated, including some successful separators. This work is continuing in order to develop a separator which is compact, effective, and easily manufactured.

A larger mercury-water loop is under construction, having a water flow rate up to 300 gpm. It will be used to mock up tentative core geometries for small cores, or segments of larger cores. A preliminary test with Na has been made with components of a 25-gpm loop for jet pumping of cobalt-cerium alloy. This loop will be known as Experiment No. 6.

An x-ray image amplifier device for studying the behavior of fluid dispersions inside an opaque system at high temperature has been constructed and tested in a breadboard model. This model was used to observe the operation of Experiment No. 5. The results were a considerable improvement over static x-ray diagnosis, and a finished model of the device is being recommended.

SODIUM TEST FACILITY

OPERATING EXPERIENCE

Summary

The sodium test facility was operated continuously except for seven shutdowns (total of 593 hr) required by maintenance, inspection, and equipment modifications. As of August 20, the following total running times have been logged on major items of test equipment:

<u>Item</u>	<u>Total Operating Time (hr)</u>
Steam Generating Unit	8,619
Intermediate Sodium Heat Exchanger	11,002
Primary Loop Sodium Pump	11,002
Secondary Loop Sodium Pump	12,532
Gas-Fired Sodium Heater	9,760
Sodium Cold Trap System	12,620
Diaphragm-Operated Sodium Flow Control Valves:	
Primary Loop (two valves)	11,002
Secondary Loop (two valves)	12,532

Intermediate Sodium Heat Exchanger

Operation of the intermediate sodium heat exchanger has continued to be satisfactory. Further tests have been conducted to confirm overall heat transfer performance at loads varying from 1000 kwt to 2000 kwt (100% of design load). An average overall heat transfer coefficient of roughly 1500 Btu/hr-ft²-°F has been attained while operating at 100% of design load.

STEAM GENERATING UNIT

Heat Transfer Data

Additional tests have been conducted on the steam generator to confirm average overall heat transfer characteristics in the four components of the unit, namely, the sodium tempering heat exchange section, steam superheating section, evaporating section, and feedwater heating section. Testing was performed at loads varying from 1000 kwt to 2000 kwt (100% of design load), with sodium flows slightly increased over that reported earlier (LAMS-2564). Operating conditions for all loads were as follows:

Superheated Steam Outlet Pressure	890 psig
Superheated Steam Outlet Temperature	925°F
Feedwater Inlet Temperature	400°F

It was noted that the average overall heat transfer coefficient in the evaporating section appeared to vary inversely with load. This is contrary to the normally expected increase in heat transfer coefficient resulting from increased steam and sodium flow velocities. It is believed that the reduction in evaporating heat transfer coefficient at the higher loads is probably not a function of flow characteristics, but rather is due to the elevated sodium temperature required for high load operation. At 100% load, the temperature difference between sodium inlet and saturated steam outlet is roughly 169°F. At 50% load, this temperature difference is reduced to roughly 124°F. It is thus concluded that the elevated sodium temperature accompanying higher loads results in a tube-wall temperature adequate to cause appreciably more film-type boiling than the predominantly nucleate boiling which occurs at lower sodium temperatures associated with reduced loads.

In contrast with the evaporating section, the heat transfer coefficient in the feedwater heating section increases quite rapidly with increase in load. This again is believed to be a result of increased sodium temperature associated with increased load. At higher loads, the relatively large temperature difference between sodium and water apparently results in a surface boiling condition at the tube wall, thus producing heat transfer coefficients more typical of boiling water than for liquid-to-liquid heat transfer.

Observed test results have failed to indicate any appreciable reduction in overall heat transfer coefficients due to tube fouling.

Evaporator Modifications

The evaporating section of the steam generator has been modified by the installation of flow control orifices at the water inlet ends of each of the evaporator tubes. The objective of the orifice installation was an attempt to more nearly equalize steam flow through each individual tube. The orifices were designed with tapered plug bodies about 1-3/8 in. long. The plugs were secured on the inside of the tube ends by means of a press fit. Single orifice holes in each plug were drilled and reamed to a 0.089 in. diameter such as to produce a predicted pressure drop of 20 psi across each tube when the evaporator is operated at rated load of 6250 lb/hr of steam. Since installing the orifices, the steam generator has been operated for 110 hr. Tests are presently being conducted to study the influence of orificing upon heat transfer, pressure drop, and temperature stability. To date, the evaporator has been operated at loads between 50% and 75% of design load. Study work has not proceeded to a point where definite conclusions may be drawn, but the following general observations have been made.

Steam flow distribution in the evaporator tubes appears to have been improved. Prior to installing the orifices, the exit steam temperature from the top evaporator tube when operating at 75% of design steaming load was observed to be 25°F to 30°F above the exit steam temperature from the bottom tube. This temperature difference is assumed to result primarily from unequal steam flow through the tubes. After installation of the orifices, the difference in steam temperature between top and bottom tubes has been reduced to from 1 to 6°F. It has further been observed that the evaporator outlet steam temperature and steam flow reach a steady state condition more rapidly following a steam load transient than prior to orificing the tubes.

CENTRIFUGAL SODIUM PUMPS

Operation of the secondary loop sodium pump has continued to be satisfactory with no maintenance required during 12,500 hr of continuous

running time. The oil leakage rate past the lower shaft seal has remained essentially constant at from 2 to 5 cm³/day.

The original oil-lubricated shaft seal has been removed from the primary loop sodium pump after 10,700 hr of operation. Inspection of the seal indicated a slight scoring on the lower seal faces. However, oil leakage through the lower seal at the time of shutdown was less than 1 cm³/day. The upper seal faces exhibited essentially no wear, with polishing marks from the original lapping still visible.

A new experimental, helium-lubricated shaft seal assembly was installed in the primary loop pump on August 1. After operating the new seal assembly for approximately 150 hr, the helium consumption rate increased to an unsatisfactorily high level. The seal was then removed from the pump and returned to the factory. Inspection of the seal revealed that one of the graphite seal faces had been damaged beyond repair and the other seal face was badly worn. Modifications were made in the seal design and it has now been reinstalled in the pump and returned to service. Since modification, the helium leakage rate for the new seal assembly has been approximately 10 standard ft³ per day.

CORE TEST FACILITY

PHASE A

Norman Engineering Company, Beverly Hills, Cal., has been selected as the Architect-Engineer for Title I, II, and III of Phase A. This company was officially authorized to proceed with Title I on August 1, 1961 with completion requested by October 16, 1961. The TA-35 location has definitely been selected for the facility, the operating convenience for K Division being the dominating factor. The construction cost will be less than at an alternative site.

PHASE B

Sodium Systems

The sodium systems have been redesigned to incorporate Type 304 stainless steel throughout. The advantages of the material are its reduced cost and its comparative freedom from the effects of long term aging at 1200°F. The only disadvantage is that the piping loops must be somewhat larger to reduce piping stresses. A layout has been completed which appears to be conservatively designed to ASME code specifications, and still fits comfortably within the available space.

Sodium Pumps

A study contract has been negotiated with the manufacturer (General Electric Co.) to determine what changes are necessary in the surplus SEAWOLF electromagnetic pumps to permit operation at CTF conditions. The normal rating for these units is 3200 gpm of Na at 700°F. In the CTF they must continuously pump 2800 gpm at 850°F with intermittent operation to 1200°F.

Of the various types of power supplies available for these pumps, the Type HC 3-phase autotransformer as manufactured by Superior Electric Co. seems to be most promising. These units will permit continuously variable operation of the pump to 2800 gpm. Cost is about \$10,000 per pump.

FUEL REPROCESSING

Pyrometallurgical Processing

Analytical results on the first volatilization and liquation experiment (IAMS-2564, page 22) with Pu irradiated to 0.12% burn-up showed only a few percent of the expected quantities of Sr and Ba volatilizing. A study of these elements was chosen for ease of analysis and because each has volatility properties such that a measurable separation

could be expected. It is felt that the difference between the theoretical and observed volatilization was mainly due to the formation of a tenacious slag at the surface of the melt. Evidence of such a slag was found in the melting crucible after cooling. A dash-pot stirrer to break up this film is being installed in the system before a second run is made.

Because of the previously reported failure in filtration equipment, no liquation results were obtained in this run. Suitable repairs in this system are being effected.

Solvent Extraction Recovery

Work on the development of a solvent extraction method for the recovery of Pu from various pyrometallurgical processes (IAMS-2564, page 23) has been directed toward finding the relationship between acidity and Pu extraction efficiency. The extraction coefficient of Pu into the solvent from $\sim 7M$ NO_3^- feed solutions is greatly affected by the acidity in the range of 0.5 to 1.5M; the lower the acidity, the higher the coefficient.

Fe^{++} has proved to be effective in the stripping of Pu from the solvent at low (0.3M) acidity and without NO_3^- . A fission-product spiked batch extraction experiment showed this stripping method to be reasonably efficient in β - γ decontamination.