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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 FEBRUARY 20, 1962

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

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FOR PERIOD ENDING FEBRUARY 20, 1962

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

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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

Operation

Reactor operation during the current report period was limited to a one-week run at 400 kw (November 29 - December 6, 1961) to serve as a final shakedown of procedures. Before mechanical deficiencies made a temporary shutdown necessary, 59 Mw-hr were accumulated making the total energy release to date 115 Mw-hr. Total reactivity loss has been approximately \$9.00, of which \$2.48 were lost during the December run; the average rate of loss was 1.7¢/hr.

Procedures as specified by the operation manuals appeared to be practical. Oscillator tests were repeated and no change in the transfer function, as compared to earlier measurements, was noted. As observed earlier, the reactor is extremely stable, and periods of several hours often elapsed between adjustments to the control rods, particularly at night when out-of-doors temperature and, hence, the heat dump temperature, was most constant. Shield modifications carried out in August 1961 were satisfactory and it appears that dose rates in operating areas will be acceptable at the design power of 1 Mw.

At the conclusion of the run a malfunction in the shim operation was noticed. In order to investigate this, the reactor loading was reduced by 1 kg of plutonium to make it completely safe against accidental criticality

and the shim operation observed under a variety of conditions. It was found that, under some conditions of high vessel temperature, there is a mechanical interference with the shim or its driving mechanisms, so that free motion of the shim is impeded, but not to such an extent as to immobilize it completely. The interference can be relieved by blocking the external cooling and reducing the temperature gradient between the vessel and the surrounding shield. Although it is not possible to determine the exact cause of the interference without an extremely difficult disassembly operation, the most probable source of trouble is the inward distortion of the guide tubes which causes them to contact the shim.

As corrective measures, the control-rod functions have been altered to provide the scram capability previously supplied by the shim drop, and air flow baffles will be installed to restrict the cooling in the annular flue surrounding the shim. With this revised operation, the reactor shutdown sequence will be a rapid drop of four control rods ($\sim \$9.20$) in about 1 sec, followed by a run-down of the shim ($\sim \$23$) in about 1 min.

Prior to further operation, the present core will be replaced. This will eliminate further delays in reaching design power, if the control modifications prove effective, and will replace the lost reactivity tied up in the present core.

Core Reloadings

After unloading a portion of the core, the remainder was frozen on January 22, 1962. The temperature descent from 466°C to 154°C was carried out with a multiplication of ~ 27 . Reactivity loss upon passing through the freezing point was only 42% , considerably less than expected. Overall reactivity gain in going from 466°C to 154°C was $\$3.5$. There was no indication that the reactivity losses tied up in the present core were not stable. The 160°C critical mass was remeasured, and the reactivity history of the present core, since the time it was first loaded, has been satisfactorily evaluated.

After freezing measurements were completed, the core was unloaded to 125 capsules, which will be the starting point for an approach to critical with the new core. Approximately 25 of these capsules have now been replaced by new ones; reloading operations are currently awaiting fabrication of more alpha cans required for the unloading operation. Unloading has proceeded smoothly so far, with all the handling gear working properly.

Preparations have begun for altering the air flow. The dose rate in the plenum under the reactor has been measured and is ~ 2 r/hr. Actual installation of the necessary air baffles will probably be delayed until a major part of the core replacement has been completed in order to take advantage of any reduction of the dose rate in the plenum.

When the modification has been completed, and appropriate AEC approval has been received, reactor operation will be initiated to evaluate the effectiveness of the modifications. It is hoped that with the new core composed of clean fuel, the annoying reactivity loss noted earlier will be reduced. The improved tantalum used in fabricating the new core will allow higher power operation.

Reflector Pins

The pause in loading Core II provides an opportunity to conduct such tests as the insertion of materials for environmental investigations and sodium sampling. In the first instance, peripheral reflector pins of 17-4 PH stainless steel are being replaced by pins made of other types of stainless steel (300 series). These steels contain 4% more nickel than do the original units; consequently a higher level of activation is to be anticipated.

Samples were obtained from the (stagnant) sodium in the upper portion of the reactor vessel (at the surface, from 2 in. below the surface, and from 13 in. below the surface). These will be analyzed for Ni, H, O, Cu, and C.

Tests of Fuel Handle Materials

During the 45 cycles of the fuel charger, one defective fuel handle has been found. The fuel element was successfully removed; however, the new fuel capsule could not be installed so the handle was replaced. A reddish discoloration was observed about 3 ft from the top. An analysis showed the deposit to be copper and nickel, with a preponderance of copper. The mechanism of this deposit formation is not understood.

Despite the elaborate thread treatment given the stainless-to-stainless threaded joint in LAMPRE, the above result indicates that some apprehension is justified when considering the problem of removing an irradiated fuel element with a frozen thread. The obvious solution is to eliminate stainless steel in this application. In LAMPRE I this is possible on one side of the joint, specifically the fuel element adaptor. This consideration has led to the order of 25 adaptors of 2140 steel (1.35% Ni and 0.62% Cr) and 25 adaptors of 4130 steel (0.95% Cr and 0.20% Mo.) These will be evaluated in a sodium environment to see if they are satisfactory in LAMPRE I.

Gamma Scanning of Fuel Capsules

The 12 in. variable slit and the new capsule scanner have been completed and used to scan capsule 1142. Figure 1 shows the La^{140} activity as a function of length compared with the x-ray photograph of the capsule. Since barium is the precursor of the lanthanum that is detected, references to the latter may really apply to barium. The self-shielding due to the plutonium does not explain the observed lack of correlation between the lanthanum activity and the plutonium position.

In particular, the thin diaphragm of plutonium revealed by the x-rays does not show a corresponding increase in La^{140} activity. This, coupled with the very sharp drop off in activity of the peak located at the fuel crud interface, suggests that the activity was located on the walls of the capsule and not in the body of the fuel. It is possible that the

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Fig. 1. La¹⁴⁰ activity as function of capsule length

barium tends to float to the top of the fuel and then migrates to the walls to form a ring at the liquid fuel surface. Upon cooling, the level of the fuel drops until the fuel freezes. When freezing, the fuel contracts further and may pull some of the barium on the wall with it, but leaves the barium ring above the solid fuel. The shape of the gamma scanning curve indicates that there is also a downward driving force on the barium (or barium-lanthanum) causing it to migrate toward the cool end of the capsule. This is a source of concern in view of the known deleterious effect of calcium and magnesium on corrosion just above the fuel level at high temperatures. Detailed microscopic examination of two capsules has shown slight evidence of a reaction layer in this region but no sign of intergranular penetration. Specific corrosive action of the fission products would not be expected to be detectable until a much greater burnup is attained with Core II.

FUEL AND CONTAINER DEVELOPMENT

LAMPRE I Fuel and Container Results

CORE I

The first core loading of LAMPRE I was melted in February, 1961, and was finally permitted to freeze in January, 1962, after about 8000 hr of operation at temperatures from 480° to 575°C. Most of the time the reactor was undergoing low power start-up testing or was shut down for shielding modification, so that the burnup experienced and the time at power were small. The capsules from the core that have been dissected have exhibited no corrosive effects. Bend tests of the capsule wall material show essentially the same properties as the original material, and metallographic examination gives no indication of penetration. These results corroborate the out-of-pile certification tests of similar fuel and containers, which predicted no visible attack at even higher temperatures.

The impurities present in the Core I fuel and the PuC added as a corrosion preventative tend to segregate at the liquid surface in the capsule as a second phase, presumably solid or semisolid. The disengagement of fission product gases from the liquid phase has been greatly impeded by this solid material. X-rays of the capsules removed from the reactor show that approximately 1/4 in. to 1 in. of the fuel and solid impurities has been pushed up toward the top of the capsule or entrained as a froth by the trapped gases. The continued loss of reactivity exhibited by the reactor during operation is consistent with the amount of fuel thus raised above the core. The few samples of improved fuel in the reactor showed a greatly reduced tendency to frothing and no indication of lifting of major quantities of fuel.

Core II

The fuel for the second core loading of LAMPRE I will be a high-purity Pu-Fe eutectic alloy with no additives and with calcium and magnesium content below 5 parts per million. The improved manufacturing process for the fuel was mentioned in a previous report; the last few months have shown that a consistent product is obtainable in this manner. Certification tests confirm that the new fuel gives improved results over the fuel made with cast-iron with any container, as long as the test temperature is below 700°C. The production process does not completely exclude dissolved carbon from the fuel, the average amount present being 60 ppm. This amount may be somewhat helpful, but the improved performance is mainly attributed to the lower calcium-magnesium content. The lower level of insoluble impurities is expected to improve gas disengagement and thus reduce reactivity loss in operation. A recently developed filtration process would provide the cleanest fuel in this respect, but reactor operation on Core II will determine whether the additional step is necessary.

Several types of starting material have proved equally successful for producing containers for Core II. These are high-purity, electron-beam melted unalloyed tantalum, arc-melted unalloyed tantalum, electron-beam-melted 0.1 w/o tungsten alloy, and an alloy containing 0.75 w/o tungsten and 0.2 w/o yttrium. The latter material has been mentioned in previous reports; it is produced by beam-melting tungsten and tantalum together, and arc-melting in the presence of yttrium. Very little of the yttrium remains in the final ingot, but the properties are reproducibly affected by this treatment. Capsules from all four starting stocks have been made by impacting and ironing, followed by annealing at 1450°C for 1 hr. The higher annealing temperature is believed to be a major contributor to the greatly improved performance of these capsules, but the reason for its effectiveness is not yet understood.

Samples of all four types of tantalum have been placed in standard certification tests aimed at simulating the reactor environment expected at full power operation. The tests were conducted for about 1400 hr at 625°C, followed by 500 hr at 675°C which is slightly higher than the expected capsule wall temperature peak at full power. Since there have been no failures so far, there is no means of determining if any one of these capsule materials is better than the others.

The capsules and fuel for Core II are all prepared, and about three-fourths of the number required have been closure-welded, weighed, and inspected.

Dilute Plutonium Fuels and Containers

In a high-performance reactor it will be desirable to dilute the concentrated plutonium fuel of LAMPRE I to obtain a high specific power without excessive power density in the core. Three types of dilute fuel are being studied -- Pu alloys without uranium, molten uranium-plutonium alloys, and uranium oxide-plutonium oxide paste.

Plutonium Alloys Without Uranium

The Pu-Ce-Co alloy system described in the preceding report is the only type of fuel without uranium receiving development attention at this time. The preparation of reproducible lots of this fuel by vacuum filtration has been found to be feasible, and a pilot production setup is being constructed.

A two-fold series of corrosion tests was set up to study in a preliminary way the effect of plutonium concentration on corrosion. In one series the plutonium concentrations tested were: 4, 5.7, 7.6, 9.5, 11, 13, and 15 g Pu/cm³. The 15 g/cm³ fuel is nearly as concentrated as the LAMPRE I fuel. After testing at 750°C for 200 hr, all the fuels more concentrated than 9.5 g/cm³ had penetrated their containers. After 400 hr, the 7.6 g/cm³ fuel showed some penetration, possibly due to a flaw in the container, while the more dilute fuels still showed no failures. The second series contained fuels with 1, 2, 3, 4, and 5 g Pu/cm³. These have been tested for 1000 hr at 750°C followed by 2000 hr at 850°C and by 200 hr at 900°C, with no failures. The containers for both series of tests were as-ironed (unannealed) unalloyed pure tantalum, a type used in many tests as a "standard corrodible container." It is much less corrosion-resistant than the annealed LAMPRE I capsules.

The two series indicate a marked difference in corrosive attack between dilute and concentrated Pu alloys. It is fortunate that the range of interest for high-performance fuels lies below 5 g Pu/cm³; the results with this material are already close to acceptable for experimental reactor use as static fuel. Loop tests using dilute fuel circulated by sodium jet pumps are under construction, using spare PTA assemblies. (The PTA setup is described below.) Container and fuel performance in the presence of high velocities, sodium, and simulated fission products are the principal remaining uncertainties in the evaluation of Pu-Ce-Co fuel.

In connection with the investigation of the Pu-Ce-Co system the following two-component phase and crystal structure results have been obtained:

The crystal structure of Pu_3Co was found to be orthorhombic, space group Cmcm with $a = 3.475 \pm 0.004 \text{ \AA}$, $b = 10.976 \pm 0.010 \text{ \AA}$ and $c = 9.220 \pm 0.008 \text{ \AA}$; $Z = 4$. The calculated density is 14.65 g/cm^3 .

The composition of the most cerium-rich compound in the cerium-cobalt system previously reported in the literature as Ce_3Co , has now been found to be $\text{Ce}_{24}\text{Co}_{11}$. Its crystal structure is hexagonal, space group $\text{P6}_3\text{mc}$ and $a = 9.587 \pm 0.003 \text{ \AA}$ and $c = 21.825 \pm 0.010 \text{ \AA}$; $Z = 2$. The calculated and observed densities are 7.67 and 7.59 g/cm^3 , respectively.

Uranium-Plutonium Alloys

In the table below are presented further results of the search for ternary eutectics of high uranium content having potential application as bases for one-region breeder fuels. The information was obtained by metallographic and thermal analysis procedures.

Compositions and Solidus Temperatures of
Alloys in Some Ternary Uranium Systems

Alloy Composition (a/o)	Approximate Solidus Temperature (°C)
15 Mn - 5 Si - 80 U*	851
20 Mn - 5 Si - 75 U	760
20 Fe - 5 Si - 75 U	806
15 Co - 5 Si - 80 U	820
5 Co - 22.5 Mn - 72.5 U	754
5 Ce - 22.5 Mn - 72.5 U	746
15 Mn - 5 Ni - 80 U	745
30 Co - 5 Ni - 65 U	743
5 Co - 30 Fe - 65 U	721
30 Fe - 5 Ni - 65 U	720
15 Mn - 5 Ru - 80 U	717
30 Fe - 5 Ru - 65 U	706

*In LAMS-2647 this composition was incorrectly reported as having a solidus temperature near 760°C .

All of the above mentioned alloys were selected as lying near a liquidus valley, but evidence for the existence of ternary eutectics was found only for the last four compositions listed in the table. The solidus temperatures for these four alloys, however, are only a few degrees lower than the lowest solidus temperatures in the related binary systems. Therefore it seems apparent that none of these ternary systems have solidus temperatures sufficiently lower than 725°C , the temperature of the high-uranium eutectic in the uranium-iron system, to offer much promise of producing a low-melting ternary eutectic of high uranium content.

The eutectic temperature of an alloy containing 78.5 a/o uranium and 21.5 a/o manganese was found to be $738 \pm 2^{\circ}\text{C}$, which is significantly higher than the 716°C reported in the literature. The temperatures of the peritectic reaction and of the β - γ uranium transformation in a sample of U_6Mn were also determined and found to be 744 and 753°C , respectively. The corresponding published temperatures for these reactions are 728 and 744°C .

Because the reported value for the eutectic temperature of the uranium-manganese system differed significantly from the recent LASL thermal analysis result, specimens in the binary uranium systems with iron, cobalt, and nickel were analyzed to confirm the published eutectic temperatures in those three systems. The results were found to be in good agreement with the literature values for the uranium-iron and uranium-nickel eutectic temperatures but for the uranium-cobalt eutectic $741 \pm 2^{\circ}\text{C}$ was obtained instead of 734°C . Moreover, the composition of the uranium-nickel eutectic, reported as 33 a/o nickel, was found to be close to 31.5 a/o nickel.

Certain ceramic materials are receiving attention as possible containers for the LAMPRE fuel. Ceramic compositions under consideration include: Al_2O_3 , ZrO_2 , CeO_2 , CaO , MgO , $\text{Al}_2\text{O}_3 + 2$ a/o Y_2O_3 , BeO , $\text{La}_2\text{O}_3 - \text{Al}_2\text{O}_3$, and $\text{Al}_2\text{O}_3 - \text{MgO}$ spinel. Crucibles of these materials will be made and tested with the fuels for compatibility and resistance to chemical attack.

Tests of a number of these materials with U-Pu-Mn are summarized in the accompanying table. Since a higher test temperature than 1000°C

Screening Tests of Containers for Alloys Based on UMn Eutectic

<u>Corrosion Specimen</u>	<u>Fuel</u>	<u>Temperature °C</u>	<u>Observations</u>
2nd stage Ta, ironed	76.7 a/o U + 18.6 a/o Mn + 4.70 a/o Pu + 500 ppm Si = TiPu 164	800, 850, 950	Leakage only after the last exposure, i. e., at 950°C
Al ₂ O ₃ container	TiPu 164	900	Unwetted. No visible attack. 170 ppm Al produced in fuel.
Container of 94 w/o ZrO ₂ + 6 w/o CaO	TiPu 164	900	No visible attack. Darkened container. Less than 5 ppm Zr in fuel.
CeO ₂ container	TiPu 164	900	No visible attack. Darkened container. Less than 25 ppm Ce in fuel.
Al ₂ O ₃ container	74.1 a/o U + 21.0 a/o Mn + 4.92 a/o Pu + ~70 ppm Si = TiPu 175	900	No visible attack.
Container of 94% ZrO ₂ + 6% CaO	TiPu 175	900	No visible attack. Darkened container.
CeO ₂ container	TiPu 175	900	No visible attack. Darkened container.
2nd stage Ta, ironed	TiPu 164	900	Wetted. Fairly strong IGP at the liquid-gas interface.
2nd stage Ta, ironed	TiPu 175	900	Wetted. Fairly strong IGP at the liquid-gas interface.
"Cerium Sulfide"	TiPu 175	900	Wetted. No visible attack.
Y container	78.3 a/o U + 21.7 a/o Mn	900	Wetted. 15 mil penetration near a liquid-gas interface.
Y container	78.3 a/o U + 21.7 a/o Mn	950	Wetted. Fluidization of Y at spots of fuel in gas phase.
BeO container (density = 2.90 $\frac{\text{gms.}}{\text{cm}^3}$)	78.0 a/o U + 22.0 a/o Mn	1000	Wetted near bottom. Metallography showed no attack. 84 ppm Be produced in fuel means 1/10 to 1/4 mil av. penetration
Container of 98% Al ₂ O ₃ + 2% Y ₂ O ₃	TiPu 184 = U + Mn + Pu like TiPu 164 and 175	900	Reaction layer between fuel and container. Possibly some volatilized Al.
Vitreous MgO container	TiPu 184	900	Very slight reaction layer.
BeO container	TiPu 184	900	Unwetted. Slight, if any, reaction layer.
Vitreous CaO container	TiPu 184	900	Slight dark reaction layer. Bulk CaO discolored but not penetrated.
Vitreous Y ₂ O ₃ container	TiPu 184	900	Slight, if any, reaction layer. Less than 5 ppm Y produced in fuel. Best appearance of S-1 to S-5.
Single crystal W in W-coated Al ₂ O ₃ container	TiPu 184	900	Flat meniscus. No obvious attack.
Sc container	TiPu 184	900	Wetted. No obvious attack.
Single crystal W in container of stock W	TiPu 184	900	Wetted. No obvious attack.
Y container	TiPu 184	900	Wetted. No obvious attack.

is needed to select the most promising container material for further testing, installation of furnace equipment capable of 1200°C test temperature has been started.

Uranium Oxide-Plutonium Oxide Paste Fuel

A paste flow apparatus has been operated consisting of two vertical reservoirs 2 in. in diameter connected at the bottom by a 3/8 in. i.d. U-tube 21 in. long. Uranium dioxide powder of about 10-μ particle size, with about 5% uranium metal powder, was slurried with excess sodium by bubbling helium through the mixture at 500°C. The slurry was easily transferred to the second cylinder by passing the helium into the first. The transfer was observed by means of a radioactive source and an x-ray image intensifier tube and television system. After settling for various times up to 16 hr, the settled slurry or paste could also be transferred with a small helium pressure. The observed flow rates were indicative of an apparent viscosity for the settled paste in the range of 10,000 to 30,000 centipoise.

A second paste flow apparatus using a positive displacement pump is in final assembly. A vacuum-type glove box is being renovated to serve as a loading facility for paste experiments.

CRITICAL EXPERIMENTS

PBX

PBX is a small-scale critical test of a circulating fuel, direct-contact core. The general objectives of this experiment were listed in the preceding report (IAMS-2647, p. 12). In the present period the objectives have been restated in the form of a tentative experimental plan; the design of the reactor and most auxiliary systems is nearing completion, and several components are in process of procurement or construction.

The fuel handling and circulation system of PBX is to be tested in a subcritical version called PTA, which contains all the hydraulic components of PBX but uses a core container only 2 cm deep instead of the 7-9 cm depth

of the reactor. PTA, which is in final assembly, will be operated in the laboratory using a source and x-ray image intensifier to examine the circulation and disengagement of fuel and sodium and to check the loading, melting and fuel handling procedures planned for the reactor.

The following specifications have been established for PBX:

Core diameter: 8.9 cm (3.5 in.)

Core height: Adjustable from 4.25 cm to 9.25 cm (8.25 cm calculated)

Fuel composition: 90 a/o Pu, 10 a/o Fe

Estimated critical mass: 7.93 kg

Estimated critical volume: 512 cm³

Total inventory: 10 kg

Separation of fuel free surface from top of core: 4 in. tantalum block

Circulation rate, fuel: 2 liters/min

primary sodium: 4 liters/min

secondary sodium: 75 liters/min (20 gpm)

Type of pump, fuel: One annular nozzle lift pump, 1/2 in. i.d., tantalum

primary sodium: Two dc electromagnetic pumps in series, 1/2 in. o.d.

secondary sodium: One ac induction electromagnetic pump, 1 in. o.d.

Rated maximum steady power: 20 kw thermal

Reactor vessel dimensions: 4 in. o.d. (lower half), 5-1/2 in. o.d. (upper half)

60 in. high (stainless steel)

Liquid fuel container: 3-1/2 in. i.d. x 30 in. high (tantalum-10% tungsten)

Control elements: 1. Adjustable core volume

2. Variable heat demand when at power

3. Large nickel reflector, hydraulically operated

4. Two smaller nickel reflector rods, hydraulically operated

5. Free-fall tube for nickel or uranium ramp slugs

Containment: Liquid fuel: Two layers of tantalum + catch pot + secondary sodium system.

Primary sodium system: Catch pot + secondary gas-tight enclosure for upper portion, secondary sodium system for lower portion.

Entire assembly: Sodium drip trays, controlled air access, stainless steel lined pit with controlled filtered ventilation, shielded with 5 ft of magnetite concrete + pit building enclosure.

Prototype assemblies for the PBX have been made from commercial tantalum. Deep drawing procedures were used to form the several tubes and cups required. Tubes were welded into the respective headers by the electron beam method. Where this joining method cannot be applied because of size limitations, the tungsten inert-arc process will be employed. It is intended that the final assembly be made from tantalum + 10 w/o tungsten alloy.

CORE TEST FACILITY

Phase A

Title I

The Title I drawings for the Core Test Facility were reviewed in considerable detail at a meeting between representatives of LASL, LAAO, and Norman Engineering on December 11 and 12, 1961. Minor changes to the drawings were made in the exterior fencing arrangements, the radioactive sodium rooms, and the space for the nonradioactive sodium.

A review of the ventilation systems indicated that duplicate equipment should be provided for certain portions in order to insure the required reliability. Under design heat load conditions, the ventilation system serving the disassembly cell can be out of service no longer than 15 min without exceeding the maximum allowable temperature for the space. The necessary changes were made in the ventilation system requirements.

Title II

Following the review of the Title I package, Norman Engineering has been authorized by the AEC to proceed with the Title II work. The date for completion has been set as March 14, 1962. The controls and alarm requirements for the ventilation system No. 1, disassembly and sodium-plutonium cells, were established and sent to Norman Engineering. A graphic control panel and annunciator will be placed in the electrical equipment room in order to provide a central control location easily accessible from the control room and outside of all areas subject to contamination.

Phase B

Sodium Systems

Pumps

The two Mark B Seawolf pumps were inspected. One has the latest model throat spacers, appears to be in excellent shape, and, if the expansion bellows is adequate, should be completely satisfactory for use. The second has no throat spacers and for this reason will not be used.

It has been learned that two more Seawolf pumps are on hand at Atomics International in Canoga Park. At this time it is planned to obtain these two pumps and to install them in both the radioactive and nonradioactive 15-Mw loops, retaining the third as a spare. The use of an EM pump in the radiator loop will assure natural convective flow in both the radioactive and nonradioactive 15-Mw loops.

A request for a proposal to cover the cost of performing tests on the Seawolf pump was sent to AI. The estimated cost was near that of the LASL budget estimate, but the decision as to whether or not to test the pumps has not yet been made. As a result of discussions held with General Electric Co., after completion of their study of Seawolf pump operation at elevated temperature, it has been concluded that this type pump is satisfactory up to 900°F.

Piping

Design is continuing on the radioactive and nonradioactive sodium piping systems. Piping flexibility and stress analysis have been completed and the piping layout finished on the radioactive system. The radioactive sodium piping drain system design has been established. The nonradioactive sodium system design is in progress.

Heating

It is proposed to control the heat to the 6-in. diameter and 4-in. diameter sodium pipe by the use of selector switches which will give eight heat levels ranging from 16 w/ft to 133 w/ft. Calculations indicate that

approximately 55 w/ft will be required to raise the temperature of the pipe (6-in. and 4-in. diameter) and solid sodium to 450°F in 52 hr.

Intermediate Heat Exchangers

Specifications for the intermediate heat exchangers have been prepared and proposals have been solicited from seven manufacturers.

Heat Dumps

Plans and specifications have been released to Supply and Property Department for soliciting bids on heat dump equipment. Proposals on this equipment should be received about March 15, 1962.

Heat Dump System

It has been estimated that the 150-hp and the 500-hp blowers that are to be located on the 95 ft x 36 ft equipment pad in Phase B may produce 122 db to 127 db and 128 db to 133 db, respectively, of noise at the fan inlets within the most critical frequency band of 150 cps to 300 cps (based on charts, tables, and equations given in the ASHRAE Guide, 1961, for determining the sound generated by centrifugal fans). The blower noise may create an intolerable condition in some rooms of the Core Test Facility and an undesirable nuisance in adjoining facilities. The A-E was asked to study this problem, and has retained the firm of Bolt, Beranek, and Newman, Inc., as consultants.

Although the report from the consultants has not been received, a preliminary discussion at Norman Engineering about the blower noise problem indicated that a fan enclosure building might be necessary. This would require sound attenuators on the fan intake and exhaust, increasing the static pressure requirements for the blowers. The additional cost of combating this noise problem has not been estimated, but will be made when the consultant's report is received.

Disassembly Cell Mockup

A mockup of a section of the disassembly cell, the alpha transfer mechanisms, and the sample transfer cell is in the planning state. It is to

be built in the recently vacated steam generator building. The purpose of the mockup is to try transfer procedures with a General Mills manipulator and alpha transfer mechanisms, help design alpha can conveyance and positioning mechanisms, and to test the integrity of Model A sealed manipulators in the transfer procedure and minor maintenance of the General Mills manipulator.

FUEL REPROCESSING

A third volatilization and liquation experiment was conducted on a Pu specimen which had been irradiated to a burnup of 0.24%. After 10 days cooling, the irradiated 80-g Pu foil (1.25 in. x 14 in. x 0.02 in.) was loaded into a Ta condenser tube attached to a crucible containing a porous Ta frit. Enough Fe powder to form the 90 a/o Pu-Fe eutectic was added to the container, which was held at 700°C for 2 hr to permit the Pu to alloy with the Fe. The temperature was then lowered to 600°C and held for 48 hr. Helium was then bled into the furnace to a pressure of 15 psig while the filtrate was drawn into the Ta receiver by means of a vacuum. Samples of the filtered melt were taken by drilling an axial hole in the ingot. The filter residue and the condensate were dissolved in 4M HCl and samples taken from these solutions.

The results of these runs are shown in the accompanying table. Run 1 was essentially a proof test of the equipment with a non-irradiated Pu foil. Runs 1 and 2 were done without stirring and Run 3 was made with stirring, using the dasher. Yields in all the runs were modest (63 to 88%) but this was, in part, related to the small scale of the experiment.

Decontamination of Zr and Ce was greater than that reported in the literature (Footnote b on accompanying table) where the runs were of comparable temperature but at much higher concentration of simulated fission products. This is contrary to expectation. Oxidation of the foil by air during the loading and unloading of the capsule would contribute to

SNP. AERO " PLUTONIUM FROM FISSION PRODUCT ELEMENTS BY VOLATILIZATION AND LIQUATION

Run	Conc. in product per g Pu/Conc. in charge per g Pu									Pu Recovery	Hr at 600°C ^(a)	% Pu Burnup
	Ba	Sr	Ce	Pu	Zr	Cs	Fe	β	γ			
1	--	--	--	--	--	--	1.00	--	--	88	48	none
2	--	0.0042	0.13	0.93	0.0007	0.0003	0.92	0.34	0.37	73	48	0.12
3	0.0026	0.0017	0.15	0.52	0.016	--	1.30	0.10	0.14	63	48	0.24
8 M ^(b)	--	--	0.77	0.94	0.20	--	0.91	--	--	~ 90	22	10.0 ^(c)

(a) Furnace pressure = 6×10^{-5} mm Hg.

(b) Typical results in the literature.

(c) Fission product elements added to simulate 10% burnup.

oxidative slagging, and the Ce removals are consistent with this supposition. The Ru decontamination in Run 3 is anomalous, having no basis in liquation or slagging theory and is probably due to poor sampling and/or sample dissolution techniques in which RuO_2 volatilizes. The poor material balance on this element tends to support this contention.

From these considerations it would appear that slagging plays a significant part in separation of these fission products from the Pu and that it would be most difficult to prevent where the burnups are so low and oxidation of the charge so likely.