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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, 1963

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OF THE UNIVERSITY OF CALIFORNIA LOS ALAMOS NEW MEXICO

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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

LAMPRE I (Core II) accumulated an additional 29 MWD of operation to give a total integrated power of this core of 70.2 MWD. The temporary limitation of 500 kW on the power level has been continued. The reactivity loss rate at the end of the quarter was about 0.24¢/h. Table I gives a résumé of recent operations with Core II.

TABLE I

Summary of LAMPRE I Operations, Core II

<u>Run No.</u>	<u>Start Date</u>	<u>Stop Date</u>	<u>Power, MW</u>	<u>Accumulated MWD</u>
62	11/15/62	11/28/62	0.5	46.3
63	11/29/62	12/17/62	0.5	55.0
64	1/10/63	1/28/63	0.5	62.0
65	2/1/63	2/18/63	0.5	70.2

Reactor Dynamics

A FORTRAN code (LTC) to predict LAMPRE I power and temperature transients resulting from changes in power demand or flow, or from changes in reactivity was written to check the validity of the calculational model. Figure 1 shows the experimental and calculated results for a 100 kW power increase initiated with the reactor operating at a steady state level of 50 kW. The agreement is considered satisfactory.

Fig. 1. Response of LAMPRE I to ~ 100 kW Power Demand for Initial Power of ~ 50 kW.

FUEL AND CONTAINER DEVELOPMENT

LAMPRE I Fuel and Containers

Metallographic examination of LAMPRE fuel capsules has continued since the leak of September 15, 1962 (see LAMS-2815). Two additional capsules (numbers 1273 and 1290), removed from the core subsequent to the leak, have been examined. A conspicuous absence of intergranular penetration was observed for all areas of these irradiated fuel capsules except in the weld zone. All three capsules exhibited intergranular penetration of Pu-fuel in the heat-affected zone of the closure weld. In addition, the capsules showed evidence of Na attack of the weld metal surface. Similar attack has been observed in Ta containing appreciable amounts of oxygen. Since these observations suggest oxygen contamination of the Ta during closure welding, extensive investigations have been initiated to improve welding procedures and to re-examine the effects of oxygen on Pu-fuel corrosion of Ta.

Since the failure of capsule 1257 allowed approximately 75 g of Pu-Fe fuel to enter the coolant, samples of Na have been taken to determine if Pu is being carried throughout the system. Table II summarizes the data obtained.

TABLE II

<u>Date of Sample</u>	<u>Location of Sample</u>	<u>Pu(ppm)</u>	<u>Fe(ppm)</u>
Prior to failure of 1257	*	1	35 $\pm$ 5
October 9, 1962	*	10 $\pm$ 5	19 $\pm$ 2
November 29, 1962	In core region	3 $\pm$ 1	30 $\pm$ 3
January 10, 1963	Top of reactor	1	20 $\pm$ 1
January 10, 1963	In core region	1	400 $\pm$ 8

*Sodium sampler filled at top of sodium level, then lowered into core for a variable period of time and extracted.

An interpretation of these data is that the Na dissolved or entrained some Pu after the capsule failure and that it has since been cleaned up. An unexpected observation is the high Fe content of the main Na stream circulating through the core. Additional Na samples will be taken at each programmed shutdown of the reactor (every 2 weeks) to follow Pu and Fe content of the coolant as a function of location in the reactor.

IAMPRE I Capsules: Fabrication

In a program to improve the surface condition of drawn Ta capsules, impacted Ta slugs were machined on all surfaces. The polished cups formed were drawn into capsules. Surface condition of these capsules was poorer than on capsules drawn without prior surface treatment. These tests indicate the tendency of Ta to exaggerate tool marks. Polishing may only smear metal over a tool mark which later becomes apparent when the metal is cold worked.

A second problem in capsule forming is to maintain uniform wall thickness through all processing stages. New tooling has been made and mounted in a precision die set in a 100-ton press. These tools were designed to obtain an impacted cup possessing a uniform wall thickness, a feature not possible with previous tools. By starting with a uniform-wall cup and exercising good control in subsequent operations, it may be possible to maintain the regularity through the final forming stages.

To simplify capsule forming, it has been proposed that capsules be made from tubing by spinning one end closed and welding to secure the present shape. This scheme was tried using a 0.433 in. o.d. by 25 mil wall standard Ta tubing. One end of a tube length was drawn down in successive conical lead-in dies to 0.125 in. o.d. The narrowed part was removed and the cone-shaped section closed by spinning. The capsule was cleaned and welded shut. Six capsules fabricated by this method are being tested.

Two Ta-10 w/o W capsules have been made for evaluation from rolled and welded tubing. In butt welding the tubing, there was noticeably less bulging at the weld than has been present in pure Ta welds made under the same conditions.

A new design of the cap for LAMPRE fuel elements has been successfully welded upon every attempt. The cap will be subjected to tests.

Pu-Co-Ce Alloys

The first irradiation experiment of the Pu-Co-Ce fuel alloy has been removed from LAMPRE after 40 MWD of irradiation. The alloy contained 3 g Pu/cm³ with 5 cm³ of Na in contact with the fuel. Metallographic investigation will follow the completion of gamma scanning to determine the distribution of fission products between fuel and Na.

At the present power level limitation of 0.5 MW for LAMPRE I, long times are required to obtain high fuel burnups of interest in establishing the worth of using molten Pu fuels for power reactor applications. In view of this limitation, a series of calculations has been made to explore the feasibility of using the 5 MW (thermal) OWR reactor at IASL for irradiation experiments involving Pu-Co-Ce fuels. Initial calculations show that advantages, from the standpoint of increased specific power, power density, and burnup rate, can be derived by using the OWR. The OWR will give a 1% burnup in 48 days, as compared with 400 days in LAMPRE at 0.5 MW and 40 days in the projected CTF at 15 MW.

A possible disadvantage in the use of the OWR is the unavoidable surface heating in the capsule caused by thermal neutrons. Effort is therefore being directed at heat transfer calculations and preliminary experimental design to determine the engineering feasibility of performing such in-pile tests.

Corrosion testing of Pu-Co-Ce alloys in Ta has continued at a reduced rate pending completion of additional Na environmental loop and Pu handling facilities. Work on these facilities is expected to be completed by June 15, 1963. Test results continue to indicate excellent performance of alloys containing up to 5.7 g Pu/cm³ at temperatures to 700°C.

Preliminary experiments to study the possibility of inhibiting the growth of the Ta_6Co_7 layer have been completed using Pu-Co-Ce alloys of 5 g Pu/cm³ with C additions from 0-1000 ppm. Tests run at 800°C for 1000 h indicate a pronounced reduction in reaction layer thickness with increasing C content. These studies will be continued over the range of fuel alloys of interest, i.e., 3 to 8 g Pu/cm³.

Co-Ce Alloys

Detailed studies of the Ta-Co reaction layer are continuing. A strong influence of Co concentration on the kinetics of film formation has been established using binary Co-Ce alloys contained in high-purity Ta tubing. Co concentrations in excess of 25 a/o in the binary alloy cause a marked increase in the rate of film growth.

Density determinations of Co-Ce alloys are being made using the NaK volumeter. Pure Ce and Ce-25 a/o Co alloy both undergo an approximate 0.75% volume decrease on melting. This decrease is similar to, although less than, that observed with Pu, and indicates that melt-freeze cycling of fuel capsules may present a difficulty with respect to capsule deformation and/or rupturing.

Uranium-Base Alloys

Attempts to find a suitable container for U-Mn-Pu alloys have continued. Yttrium (Y) capsules containing both binary U-Mn alloy and ternary Pu-U-Mn were run at 900°C to determine if the beneficial effect of Pu on the corrosion resistance of Y, as first reported in LAMS-2720, is correct. The fuels used were 78 a/o U-22 a/o Mn in one case, and 73 a/o U-5 a/o Pu-22 a/o Mn in the other. Observed penetrations of the Y walls, after 500 h at 900°C, were 2.5 mils with the binary U-Mn fuel and less than 1 mil with the ternary fuel. Hence it appears that the initial test observations were correct and that Pu additions increase the corrosion resistance of Y to U-Mn fuels. Tests are in progress to investigate this effect at 950°C up to times of 1500 h.

Plutonium Alloy Preparation and Fabrication

Molten Pu casting feed has been successfully agitated in a vacuum furnace using a modified dash-pot stirrer to alloy materials, to increase ingot yield, by breaking the oxide film on lathe turnings, and to outgas the melt. On the upstroke of the stirrer, a trough lifts metal from the melt and the holes allow drainage back into the melt. On the downstroke, metal is pushed from the top of the melt into the bottom by being trapped in the volume inside and above the trough. The stirrer is used with a 3.25-in. diameter melt crucible employing the frozen-plug technique for pouring. The cycle is about 10 strokes/min for ~ 5 min.

A 5.5-kg ingot of Pu-2.5 w/o Fe alloy was prepared with a 97% yield from a charge composed of 45% lathe turnings. This yield, normal for massive feed, is up substantially from an anticipated 90% yield without stirring. The usual outgassing observed during the pouring of a melt did not occur; rather, the melt outgassed during stirring.

The preparation of a Pu-Ce-Co alloy containing 8 g Pu/cm³ was also accomplished by the same method. The three components were placed in the melting crucible, heated to 800°C, stirred, and poured into a rod mold. Chemical analysis of top and bottom samples showed the material to be homogeneous within the precision of the determinations.

CORE TEST FACILITY

Phase A

A revised cost estimate received from Norman Engineering reflects corrections and modifications growing out of the October 1962 Title II review. This estimate shows an increase of \$7,662 over the September 1962 estimate. The present estimate of Phase A costs are as follows:

A-E	\$ 315,000
Building estimate	2,609,000
10% building contingency	261,000
Zia services	15,000
	<u>\$3,200,000</u>

Seven sets of Title-II drawings and specifications were received on February 18, 1963 for compliance review. A final conference with Norman Engineering, AEC, and LASL personnel is scheduled for February 26 and 27, 1963.

On the basis of a favorable review of the Safety Analysis Report by the Albuquerque Operations Office, the Los Alamos Operations Office has sent out advance notice to possible bidders. The present schedule calls for transmission of Title-II drawings and specifications to bidders about March 15, opening of bids May 3, and start of construction June 3, 1963.

Sodium System

Piping. One-line isometric drawings of 6-in. and 4-in. piping are now being prepared for use in purchasing shop-fabricated piping, and work is continuing on the radioactive and nonradioactive Na two-line drawings. Relocation of two penetrations in the Nonradioactive Sodium Room floor was required to reduce the forces imposed upon the intermediate heat exchanger by nonradioactive Na lines. The required changes in line locations have been made.

Intermediate Heat Exchangers. Revised general arrangement drawings have been completed by the Baldwin-Lima-Hamilton Corporation and have been approved by K Division. Detailed assembly drawings scheduled for completion February 1, 1963, have not been received.

The B-L-H Corporation also reports that an examination of LASL's Na piping layout reveals that certain modifications will readily solve the calculated overstressed conditions at the intermediate heat exchangers.

Heat Dumps. A review of progress with Struthers Wells revealed that problems have been encountered in the design of inlet and outlet Na headers for the heat exchanger sections. Design work has been delayed pending final stress analysis calculations which have been subcontracted to the Arthur D. Little Co.

Pumps. General Electric has stated that the E.M. pump for the 5 MW radioactive Na loop can be designed for liquid-cooling. This will result in a considerable saving in space and equipment over the air-cooled design. Actual design of the pump is still being delayed pending accumulation of more data by GE who have pumps of similar design in operation. Delivery of the pump is scheduled for December, 1963.

Byron Jackson reports that the 5 MW nonradioactive Na pump tank is being reworked because some of the welds failed to pass radiographic inspection. The tank is expected to be ready for leak testing in March, and delivery is scheduled for June, 1963.

A preliminary study has been made of the requirements for an emergency, battery-operated E.M. pump to be used, in the event of power or main pump failure, to remove decay heat from the core. Natural thermal convection in the nonradioactive loops makes emergency pumps in those loops unnecessary. Calculations indicate that for the 15 MW radioactive Na loop a flow of 50 gpm with a developed head of 0.5 ft of Na is required.

Flowmeters. Delivery has been made of the two 4-in. and two 6-in. Foster flow tubes. These four flowmeters will be radiographed in an attempt to determine cleanliness of the inner passages. Delivery of the four electromagnetic flowmeters is expected in March.

Hot and Cold Traps. It has been decided to await results of hot-trap developments before starting detail design of cold traps. It may be that all necessary initial cleanup can be performed in the dump tank and subsequent system cleanup done on both radioactive and nonradioactive systems by hot traps.

Pressure Transducers. A 6-psi Consolidated Controls Corp. high-temperature differential pressure transmitter has been received and tested at room temperature. The data were quite linear. Troubles with cable and connectors at temperatures above 150°C have precluded any calibration at elevated temperatures because of excessive zero offset. A new high-temperature cable has been received for evaluation.

Sodium Level Detectors. A decision has been made to mount the Na level indicators required on the surge tanks and intermediate heat exchangers in the nonradioactive Na room. This will allow replacement of coils as well as maintenance of the drive system without entrance into the radioactive Na room. As a result, four additional penetrations will be required in this room through the 6-ft thick ceiling; each penetration consists of a piece of 1-in. standard weight pipe with a 12-in. square plate at the top.

Sodium Leak Detector. An investigation of methods for Na leak detection has not been encouraging. Several types of detector cables have been tested, but all have proved to be unsatisfactory. Investigation will be made of other methods for Na leak detection.

Sodium Smoke Filters. As a part of Phase B, construction filters for the removal of Na smoke from a system leak will be installed in Ventilation System No. 2. Two suppliers of Dynel fiber have indicated the availability of materials with 1.7 lb/ft² density and 15 to 30 μ fiber diameter, suitable for the smoke filter. A drawing of a filter housing has been completed. Installation drawings for the smoke filter will be initiated upon receipt of Norman Engineering Title-II construction drawings.

Core Cover Gas Cleanup System

The tentative instrumentation system for the Core Cover Gas Cleanup System has been reviewed with instrumentation personnel to insure compatibility with the over-all instrumentation. Design of components for the Na vapor condenser test loop is nearly complete and components are being obtained for construction. Contact has been made with the Linde Company for molecular-sieve (M-S) data in the temperature and concentration ranges of interest. It appears that it may be possible to remove N₂ in a refrigerated M-S bed, if a more desirable method is not found.

The brine selection for the -20⁰F system has been complicated by the possible radiochemistry of glycols and alcohols. These compounds are said

to form explosive polymers upon exposure to gamma and neutron radiation. The rate of polymerization and the extent of the hazard are being investigated. The refrigeration cycle may be simplified by the use of DuPont Freon R-502, a new refrigerant. It has been reported that a simple vapor cycle using DuPont R-502 may displace cascaded and compounded cycles.

The possibility is being investigated of purifying He by passing it through a silica glass diffuser. Data provided by the Bell Laboratories indicate that 1000 scf/day of He may be purified using a 2-ft³ bundle of silica glass capillary tubing at 400°C with a 30-psi pressure difference. If suitable, such a diffuser would allow removal of both chemical and fission product impurities in a single piece of equipment.

Control Room and Instrumentation

Data System. The preliminary specification for the data system is still in preparation. The problems with the input characteristics for low-level analog signals have either been by-passed or postponed. The pertinent relationships for a fourth-order Bessel Filter were then calculated for a 1.0 cps 3 db point. Unfortunately, the values for the inductances, when desiring an input impedance of the order of 1 megohm, were extremely high. Since signal conditioning for a large number of channels driving low-input impedance filters is too expensive, the use of passive filters alone (at these low frequencies) does not seem feasible. It was therefore decided to wait and see how many input signals will contain high enough frequencies of significant amplitude to cause intolerable folding errors.

Instrumentation System Study. A rough draft has been completed for the introduction, scope, and the design status of the core and Na system. IBM channel title and channel requirement listings have been made up for all presently known channels. It is estimated that the System Study is now about 35% complete.

Hot Cells

Windows. Specifications for the 10 hot-cell windows have been completed and mailed to possible bidders. Bids are to be submitted by April 1, 1963.

In the disassembly and NaPu cell window specifications submitted to vendors, the following data have been included.

1. Source Geometry - point source. Distance to hot side of window - 5 ft

2. Gamma Ray Source Strengths

<u>Gamma Energy (MeV)</u>	<u>Curies</u>	<u>Gamma Energy (MeV)</u>	<u>Curies</u>
0.4	4.3×10^4	1.8	8.2×10^3
0.9	3.9×10^5	2.2	4.1×10^3
1.2	5.8×10^5	2.6	1.1×10^3
1.35	5.8×10^4	3.2	2.8×10^2

3. Dose rate at cold face of the window to be no greater than 0.2 mrem/h. However, the density-inches provided by the window must not be less than that corresponding to a 4-1/2 ft thick magnetite concrete shield having a density of 230 lb/ft³.

4. The integrated dose at the hot face of the window will be 10⁹ roentgens over a 2-yr period.

Specifications for the shielding requirements for the viewing windows in the gas cleanup cell were as follows:

Density inches provided by the window must not be less than that corresponding to a 3-1/2 ft thick magnetite concrete shield having a density of 230 lb/ft³.

Manipulators. Specifications for bridge-mounted manipulators, bridge-mounted hoists, and Model A manipulators are being written.

PLUTONIUM ALLOY SYSTEMS

During the current reporting period, further details of several Pu alloy systems were elucidated. Although much of this information does not have immediate practical application in the field of reactor development, it adds to the general fund of knowledge about phase relationships in Pu alloys.

Plutonium-Carbon-Oxygen System

Heat treatments at 1000°C of several alloys containing 50 a/o Pu, 25 to 43 a/o C, and 25 to 7 a/o O suggest that the alloys do not decompose at that temperature, since no CO evolution was observed. Present indications are that the 1000°C isothermal section of the phase diagram includes a single-phase region comprising a solid solution of O in PuC, which extends at least as far as 50 a/o Pu-25 a/o C-25 a/o O. This solid solution coexists with liquid Pu along one of its boundaries. Lattice parameters of four alloys consisting of the solid solution phase were found to lie between 4.964 and 4.970 Å. Thus, no great change in the lattice parameter of PuC seems to be caused by dissolved O.

Plutonium-Europium System

Thermal analysis and metallographic and x-ray examinations of a 5 a/o Eu alloy specimen have confirmed the earlier report that Pu and Eu form immiscible liquids. The melting point of the Pu-rich layer was 642°C, essentially the same as that of pure Pu. A thermal effect at 813°C was apparently the melting point of the Eu (reported as 824 ± 10°C). The lattice parameter of a sample from the Eu-rich layer, is in agreement with that of pure Eu. Thus there appears to be little or no solubility of Pu in Eu. The absence of significant effects on the transformation temperatures of the Pu allotropes indicates that Eu is not soluble in them.

Plutonium-Lanthanum System

Differential thermal analysis of a 5 a/o La alloy specimen has given results which confirm and extend the general construction of the tentative phase diagram proposed by U.K.A.E.R.E. investigators. The temperatures of 450 and 475°C, respectively, found for the $\delta \rightarrow \delta'$ and $\delta' \rightarrow \epsilon$ transformations in the alloy are essentially the same as those found for pure Pu and suggest that very little La is soluble in the δ and δ' phases. The solubility limit of Pu in β -La at 600°C was determined by the lattice spacing method to be 11 ± 1 a/o.

Plutonium-Nitrogen System

The decomposition pressure of PuN increased from 0.01 atm at 2587°C to 12 atm at 3032°C. It appears that the presence of several atomic percent of O in PuN does not significantly affect the decomposition pressure.

Plutonium-Praseodymium System

Liquidus temperatures of Pu-Pr alloys have been obtained for alloys containing 5 to 90 a/o Pr. Alpha-Pr solid solution forms peritectically, i.e., $L + \beta\text{-Pr} \rightleftharpoons \alpha\text{-Pr}$, the peritectic horizontal extending probably not beyond the composition interval of 25 to 90 a/o Pr. The average observed temperature of the eutectic reaction $L \rightleftharpoons \epsilon\text{-Pu} + \alpha\text{-Pr}$ is $626 \pm 2^\circ\text{C}$ and that of the peritectoid reaction $\epsilon\text{-Pu} + \alpha\text{-Pr} \rightleftharpoons \delta\text{-Pu}$ is $484 \pm 1^\circ\text{C}$.

The solubility of Pu in Pr increases with temperature and may reach a maximum of 30 a/o at about 800°C. At 450°C the maximum solubility was found to be 16 ± 1 a/o Pu.

Plutonium-Scandium System

Microstructural and x-ray data indicate that the maximum solubility of Pu in Sc is about 20 a/o. Differential thermal analysis results obtained for 5 and 25 a/o Sc alloys suggest that ϵ -Pu and β -Sc form a

continuous series of solid solutions in the temperature range of 564° to 693°C for the alloy containing 5 a/o Sc, and 724 to 874°C for the 25 a/o Sc alloy.

The peritectoid reaction $\epsilon\text{-Pu} (\beta\text{-Sc}) + \alpha\text{-Sc} \rightleftharpoons \delta\text{-Pu}$ was found to occur at 663°C, and the peritectoid point, which corresponds to the maximum solubility of Sc in $\delta\text{-Pu}$, was estimated to lie at about 25 a/o Sc.

Only δ phase was observed in the microstructure of a 15 a/o Sc alloy that had been homogenized at 450°C for 7 weeks. The lattice parameter of the cubic close-packed unit cell was $a = 4.6532 \pm 0.0001 \text{ \AA}$, which leads to a calculated density of 13.84 g/cm³, as compared with the measured density of 13.85 g/cm³ of the arc-melted ingot.

Plutonium-Yttrium System

Delta Pu was not retained to room temperature in filings from an alloy specimen containing 15 a/o Y. Thus the maximum solubility of Y in $\delta\text{-Pu}$ appears to be low. The maximum solubility of Pu in Y is about 20 a/o.