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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, 1963

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

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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

Operation

LAMPRE I (Core II) accumulated an additional 8.4 MWD of operation during the quarter to give a total integrated power of 104.5 MWD. During Run No. 73, which was started on May 15, 1963, two instances of abnormal build-up of gamma activity in the cover gas system were observed, accompanied by a total loss of reactivity of 10 to 20%. The reactor was shut down, and nine fuel elements were removed from the central part of the core for examination. Several of the capsules appeared to have Pu contamination on their surfaces, and radiography indicated that about 2 cm³ of fuel was missing from a capsule which had been about halfway out in the core. Further operations at power with this core did not seem warranted, so the core loading was reduced to 21.8 kg Pu (less burnup) and the remaining fuel was frozen by reducing the ambient temperature to 385°C. Extrapolation measurements indicate that the system is subcritical by about \$65 with the shim and four control rods withdrawn.

Fission Gas Bubbles and Response to Large Transients

From the observed accumulated reactivity loss of \$12 in the Mark II loading of LAMPRE I, it is estimated that a volume fraction of 0.05 fission gas bubbles was contained in the fuel of this core at the end of its irradiation. Since the gas bubbles greatly increase the compressibility of the molten fuel, it would be expected that the bubbles would affect the fuel expansion during a power transient in which inertial effects are significant. Estimates were made of the effects of the bubbles on energy releases, peak pressures and peak temperatures for various reactivity insertion rates.

It was found that, for a given ramp insertion rate, the bubbles produce greater energy releases and higher temperatures, but smaller pressure excursions. Moreover, the deviations from linear energy release due to inertial effects are observed at lower reactivity insertion rates when bubbles are present. For reactivity insertion rates above \$10/sec, the pressure rise during the course of the transient will cause a significant increase in the acoustic speed; this should result in an associated decrease in the effects due to the bubbles.

General

It was reported in LAMS-2915, p. 10 that Na samples taken from LAMPRE had not given consistent results with respect to Ni, Cr, and Fe analyses. It has now been established that the discrepancies can probably be explained by a combination of sampling at different times (at intervals of days) and positions in the reactor. Samples taken over a short period of time (within a few hours) in a single location gave consistent analyses.

The buildup of N₂ has been observed in the LAMPRE cover gas (LAMS-2915, p. 10). Since the reactor was frozen, as mentioned earlier, the N₂ buildup rate has decreased.

LAMPRE FUEL CAPSULES

Metallographic and X-Ray Examination

Metallographic examination of five fuel capsules removed from Core II of LAMPRE was performed and the results are recorded in Table I. All capsules with the exception of No. 1268 (low reactor exposure) showed evidence of Pu intergranular penetration (IGP) immediately below the closure welds in the heat-affected zone. This location of intergranular penetration is typical of all such leakage observed in irradiated fuel capsules and suggests a major problem exists in obtaining a satisfactory welding technology for sealing Pu-fueled Ta containers. Investigations have been initiated to understand the effect of welding on the degree of impurity pickup and grain boundary segregation in the heat-affected zone of closure welds.

TABLE I
Examination of Pu-Fe Fuel Capsules

Capsule No.	<u>1268</u>	<u>1393</u>	<u>1260</u>	<u>1251</u>	<u>1454</u>
MWD (Reactor)	2.3	64.7	104.5	104.5	104.5
Container	Ta-0.1W	TA-0.7W- 0.2Y	Ta-0.1W	Ta-0.1W	EB Ta
Platelets (weld)	No	No	Yes	Yes	(incomplete)
Precipitates (body of cap)	Yes	No	No	Yes	(incomplete)
IGP (fuel) (weld)	No	Severe	Slight (~ 0.0002")	Severe	(incomplete)
Soln. attack (anywhere)	No	No	No	No	Yes
Comments	Short reactor exposure	One of few capsules without platelets	No apparent damage	Large sections of Ta lost due to platelets intersect- ing in weld	Fuel (~ 30 g Pu) leaked out of capsule

Another phenomenon regularly observed in irradiated capsules is the formation of platelets near the external surface of the weld metal which is in contact with Na coolant. These platelets for the most part appear optically active when observed under polarized light and have the morphology of oxide precipitates (see Fig. 1). The platelets form on a set of orthogonal planes in the Ta lattice and in some instances a large block of Ta may be dislodged from the surface where platelets intersect. To date, it has not been possible to obtain platelets in out-of-reactor tests with any regularity.

X-ray examination of capsule 1454 showed evidence of gross fuel leakage of approximately 30 g of Pu. Upon puncturing of the capsule a gas pressure of approximately 200 psi (corrected to reactor operating temperature

Fig. 1. Capsule 1251 (x300): weld metal, etched with $\text{HNO}_3\text{-HF-N}_2\text{SO}_4$, shows platelets in weld.

of 500°C) was present; this is approximately one third that expected for a gas tight capsule of equivalent reactor exposure (~ 10⁴ MWD). Metallographic examination of the fuel-vapor interface level disclosed approximately 40% wall thinning from the inside of the capsule. Since this effect at the fuel-vapor interface has been observed on only three capsules of ten examined, an attempt to determine the common features of these three capsules is in progress.

Tantalum Evaluation

Out-of-pile Na loop testing of Ta containers continues to show the superiority of double arc-melted Ta to electron-beam melted material for containment of Pu-Fe fuel. Isothermal tests at 600°C, 650°C, and 700°C have progressed to 2800 h and have just been interrupted for examination. Results of the examination are not yet available.

Electron microscopy of unalloyed Ta grain boundaries has indicated a pronounced effect of annealing temperature on the amount of grain boundary precipitates present in electron-beam melted metal. Annealing at 1600°C for 1 h produces a maximum amount of precipitation in a series of specimens annealed for 1 h in 200°C increments from 1200°C to 2200°C. Specimens annealed at 1200°C and above 2000°C exhibit precipitate free boundaries, whereas at 1400°C and 1800°C some precipitates are observed although considerably fewer than at 1600°C. This observed maximum precipitation corresponds to a minimum intergranular penetration observed with Pu-Fe fuel in small laboratory investigations.

Evaluation of butt-welds, made with double arc-melted Ta tubing, and Pu-Fe fuel has shown intergranular penetration in the vicinity of nearly all welds whether made by electron-beam welding or the TIG process. Failures were observed after 700 h at 700°C, 1400 h at 650°C, and 2100 h at 600°C. These results again suggest that the changes to the metallurgical or chemical properties of the weld area are the key to the intergranular penetration by Pu-Fe fuels. It is of interest to note that in none of the weld tests has the closure weld at the top of the capsule failed. This apparent anomaly is not understood.

Fabrication

A pilot lot of 40 capsules was made from K658 high-purity, double arc-melted Ta billet to check several fabrication changes to which reference was made in LAMS-2915, p. 11. Three punches were used in forming the starting slug into the cup ready for ironing. It had been standard procedure to use one long punch for the five impact stages. Deflection in this punch during impacting tended to form cups of uneven wall thickness. By using shorter punches in the first several stages, a more uniform wall was obtained which carried through to the final capsule. Surface machining and/or polishing at intermittent stages of processing improved the surface of the finished capsule. Wall thickness was increased to secure an average thickness of 28 mils at the capsule open end.

Based on the improved quality of the cups as shown by the pilot lot, it was decided to fabricate approximately 400 capsules to furnish a full core loading. Material from billet K658 and a second high-purity billet K661 was cut into plugs. About 150 capsules from K658 are in the final ironing stage of fabrication and some 250 slugs from K661 are in the impact extrusion stages.

There has been a general tendency for the o.d. of the capsule to enlarge slightly upon welding the end cap. Since the final o.d. is a critical dimension, an excessive increase could cause rejection of the loaded fuel element. Consequently, experiments were made on Ta tubes and capsules to reduce the diameter of the open end by squeezing in a split die. The procedure proved feasible and so it was adopted for the 40-capsule pilot lot mentioned above. Reduction in o.d. from the as-drawn 0.429/0.430 in. to 0.426/0.427 in. provided an additional few thousandths leeway on the diameter to offset the increase from welding.

A group of capsules made from Ta tubing by spinning one end of the tube closed and welding is being evaluated. If the process appears suitable for further testing, additional capsules will be made. Fabrication from tubing would yield capsules with a more even wall thickness than is generally obtained from the present process. The major problem when processing tubing

is to make the closure at the tip end of the capsule. A process in which the spun tip is drilled out, fitted with a plug, and welded seems feasible but requires evaluation.

Four projection-ring, Ta samples welded in air were received from Sciaky Bros., Chicago. There was some evidence of misalignment during welding. X-radiography indicated a sound weld but metallography picked up fine porosity and some evidence of contamination in the weld area. Sciaky Bros. will weld additional samples, but in an inert atmosphere.

Attempts have been made to size the capsule open end using magnetic swaging. One three-piece and one solid mandrel were used with better results from the latter. One test sample was as-ironed, the second annealed. The results showed a satisfactory reduction in inside diameter of about 3 mils. Spring back was a uniform 1 mil from the 0.369 in. diam solid mandrel, permitting easy removal of the mandrel.

A new drawing lubricant is in process of evaluation. It consists of 100 g of acrylic powder dissolved in 1000 cm³ of trichloroethylene. The metal is dipped into the solution and drained. Room temperature drying leaves a layer of lubricant ~ 0.2 mil thick. The material seems to act by melting under the draw pressure thus wetting the tools and draw-piece.

FUEL AND RELATED ALLOYS

Pu-Co-Ce Fuels

Plutonium-cobalt-cerium fuels with 3, 5, and 8 g Pu/cm³ have been tested at 600°, 650°, and 700°C up to 2800 h without indication of failures. One test at 600°C, employing electron-beam melted Ta, has proceeded to 3500 h without failure indication. The tests will be continued to 10,000 h to determine the out-of-reactor corrosion performance of these fuels.

The Ta₆Co₇ reaction layer has been shown to form on the fuel-Ta interface only with the 8 g Pu/cm³ fuel tested at 700°C for 1000 h. However,

a reaction layer formed on the vapor-Ta interface with 3, 5, and 8 g Pu/cm³ under the same conditions.

Ce-Co Alloys

When carburized and uncarburized Ta are tested simultaneously in a given melt of Co-Ce, no Ta₆Co₇ reaction layer is formed. The carburized layer partially dissolves in the melt and coats the bare Ta with a uniform carbide layer. This result indicates that a carburized layer on Ta may be self-healing and provide corrosion resistance to the container. Ta capsules with carburized layers are being tested with Pu-Fe fuel to determine the protective capacity of carbide layers at 650°C; so far 2800 h of testing have been completed.

A sample of Ce-18 w/o Co was held at 900°C for 53 h and then passed through a Ta frit filter to remove any particulate matter present. Analysis showed less than 25 ppm Ta, which is the limit of detectability of the method used. A similar sample which was not filtered showed a Ta analysis of 1100 ppm. It is evident that the Ta₆Co₇ reaction layer spalls from the Ta surface and is present in the melt as particulate reaction layer product.

Evidence of a low temperature coefficient of Ta solubility in Co-Ce alloys has been provided by a radio-tracer technique. The differential solubility of Ta in a Ce-12 w/o Co alloy between 600° and 700°C has been determined as 3 ppm. Although this differential solubility is small, mass transfer effects might be experienced with a mobile fuel operating in a temperature gradient.

Other Ce-Based Alloys

Density determinations of Ce-Ni alloys up to 21 w/o Ni has established the Ce-Ni eutectic melting temperature to be 485°C rather than 453°C as reported in the literature. This result has been substantiated by differential thermal analysis work.

Preliminary differential thermal analysis of the Ce-Mn binary system is complete. The invariant temperatures obtained near the lowest melting eutectic and up to 33 a/o Mn are as follows:

<u>Temp. (°C)</u>	<u>Composition</u>	<u>Tentative Interpretation</u>
638 $\pm$ 3	Low-Mn side of eutectic	Delta-gamma Ce
622 $\pm$ 3	6-33 a/o Mn, minimum	Eutectic temperature
625 $\pm$ 5	High-Mn side of eutectic	Beta-alpha Mn

Differential thermal analysis equipment is now ready for use in studying the Pu-Ce-Mn liquidus surface to determine if compositions in the range of 5-8 g Pu/cm³ have low enough melting temperatures to be of interest in molten Pu reactors.

The low melting point of Ce determined by differential thermal analysis and reported in IAMS-2915 (p. 15) has been established as being the result of a 1% Co impurity inadvertently added to the sample during fabrication.

PLUTONIUM ALLOYS

Plutonium-Scandium Alloys

The existence of a continuous series of solid solutions between ϵ -Pu and β -Sc, indicated by thermal data, has been confirmed by microstructural evidence.

Plutonium-Lanthanum Alloys

The solubility of Pu in β -La was determined by the x-ray lattice spacing method. The results, believed to be accurate to only about ± 1 a/o, are given on the following page.

<u>Temperature (°C)</u>	<u>Solubility (a/o Pu)</u>	<u>Temperature (°C)</u>	<u>Solubility (a/o Pu)</u>
250	2.4	600	10.7
350	3.8	700	13.8
400	5	800	17.7
500	6.9		

Plutonium-Praseodymium Alloys

A Pu alloy specimen containing 1 a/o Pr was examined dilatometrically. The temperatures obtained for the $\alpha \rightarrow \beta$, $\beta \rightarrow \gamma$, and $\gamma \rightarrow \delta$ Pu transitions were found to be essentially the same as those determined for Pu of ordinary purity examined under similar conditions; the peritectoid temperature of $488 \pm 3^\circ\text{C}$ agreed with that given by thermal analysis of other alloys.

The coefficient of linear thermal expansion obtained on heating the specimen in the δ -phase temperature range varied between -17 and $-38 \times 10^{-6}/^\circ\text{C}$ for different runs but averaged $7.8 \pm 1.4 \times 10^{-6}/^\circ\text{C}$ on cooling in the same temperature range.

REACTOR CALCULATIONS

Cores for Testing in CTF

Selection of tentative core parameters for testing in CTF from survey calculations depends on finding a balance between two objectives. Based solely on neutron economy, the following are clearly indicated: high fuel density, high diam/pitch ratio, and large capsule size to wall thickness ratio. Alternatively, a desire for high specific power with no limit on power or total flow indicates the following: low fuel density, open lattices, and small capsules. It appears likely that, for the objective of obtaining a short doubling time, a midrange compromise is indicated.

Further calculations are underway to describe cores in the range selected with somewhat more detail, and to determine the relationship between fueled region neutron economy and breeding ratio for this type of core. These calculations are being concentrated on midrange fuel densities, relatively high diam/pitch ratio, and relatively small fuel capsules.

Large Core Study

A preliminary survey of large, molten Pu-fueled reactors has been completed. This survey considered several modular designs which would have power capabilities in the range of 1000 MW (th) or greater. The driver regions represent a selection from the molten Pu cores which have been considered with respect to the CTF. The blankets are stationary lattices of UC_2 and UC, stainless steel clad, and cooled with Na. The particular lattice parameters have been arbitrarily selected. The significant results of this survey are as follows:

1. The use of molten Pu in a modular arrangement appears to be quite feasible from a nuclear point of view.
2. For either blanket considered, a driver fuel containing ~ 5 to 6 g Pu/cm³ appears to be optimum.
3. The decreased moderation of UC rather than UC_2 as a blanket fuel allows a distinctly ($\sim 5\%$) higher breeding ratio. The spectrum in a UC blanket is still quite soft relative to the driver spectra, indicating that further improvement would be possible, if a blanket having less moderation were available.
4. The breeding ratio of a given driver and blanket appears to increase monotonically as the thickness of blanket surrounding each driver is increased.
5. Breeding ratios of ~ 1.4 with doubling times of ~ 20 yr appear to be possible for such systems of ~ 1000 MW (th) size.

CORE TEST FACILITY

Phase A

Construction Contract

Phase A construction bids were opened May 29, 1963 and the low bidder, Ets-Hokin and Galvan, was selected as the contractor. The seven bids ranged from \$2,851,705 to \$3,557,700 with the AEC estimate of \$2,608,600. Notice to proceed was given to Ets-Hokin and Galvan July 29, 1963. Construction began about August 12, and on August 20 the temporary security fencing had been installed and earth moving work had begun.

Design and Maximum Credible Accident

The maximum credible accident postulated in the Safety Analysis Report (LA-2735) assumed that the dumping of fuel and primary Na onto the reactor room floor occurred without impairing the operation of ventilation system No. 2. If it is assumed that ventilation system No. 2 is not operating at this time, the pressure and temperature transient is considerably more severe.

An analysis was made of the Title II plans and specifications with regard to pressure and temperature characteristics of structure and equipment that would be involved in the MCA. Present design of the radioactive Na room hatch shield cover and equipment access opening seal limits internal pressure from a structural standpoint to about 5 psig. Piping and equipment in ventilation system No. 2 are designed for a working pressure of 2.5 psig, but will withstand 5 psig without damage. The equipment is specified for a working temperature of 175°F and should be good for temperatures up to 250°F.

Change Order

A change order request was sent to LAAO on August 8, 1963, and covers modifications which grew out of the Advisory Committee on Reactor Safeguards safety analysis review (see page 17) as well as changes of a more usual nature. The important items in the order are the following:

1. Modification of the cooling tower water to conform with local requirements and maintenance practice.
2. Deletion of contractor installation of all government furnished bridge-mounted, hot-cell manipulators, certain bridge-mounted hoists, certain Model A manipulators, and the gas cleanup cell periscope sleeves.
3. Modifications required for the installation of the government furnished transfer drawer, located in the gas cleanup cell.
4. Removal from the contract of the ventilation system No. 2 vent filter train and running the vent lines underground for future Phase B installation of the filter train.
5. Separating the power supply to each alternate motor pair so that failure of a single motor control center will not result in the loss of power to both motors.
6. Addition of hold-down bolts around the edge of the radioactive Na room hatch and an increase in the size of the bolts around the reactor pit.
7. Change of ventilation system No. 2 upper temperature requirement from 175°F continuously to 400°F for 6 h.
8. Substitution of a 20 kW uninterrupted emergency power supply for data system and instrumentation for the specified 25 kW conventional unit.

Sodium Systems

Intermediate Heat Exchangers

The intermediate heat exchanger designs are complete and all drawings have been approved. The Baldwin-Lima-Hamilton Corp. has been unable to obtain seamless, stainless steel pipe for use in fabricating the heat exchanger shells. It therefore will be necessary to weld the shells from rolled plate. B-L-H further advises that they cannot perform such welding with their production facilities. Hence, they are subcontracting the shell fabrication to the Swepeco Tube Corporation, Clifton, N.J. Shell fabrication will start about the first or second week of September.

Design has been modified to include shell Na impingement baffles at the inlets. These baffles are required to prevent gas entrainment from the He thermal barriers which are located between Na and tube sheets.

Heat Dumps

Fabrication of the heat exchanger pressure parts was started on August 7 at the Struthers Wells, Warren, Pa. plant. These parts are scheduled for completion the last week of October. Fabrication of the heat exchanger enclosure, ducts, stacks, and structural parts has been started at the Smith Moon Steel plant in Winfield, Kansas. Completion date for this work is scheduled for the middle of October. It now appears that the heat dump delivery should be completed some time in November. Shipments of some minor components were received during August.

Pumps

Byron Jackson has completed the 500 gpm mechanical pump; it has passed the performance test and is being prepared for shipment. General Electric has provided a preliminary sketch of the 500 gpm EM pump showing external configuration and dimensions. Actual design drawings are being prepared and should be submitted for IASL approval within the next few weeks.

Specifications were written and a purchase inquiry initiated for procurement of dc, electromagnetic pumps for both 5-MW and 15-MW radioactive Na systems. These pumps are low-flow, low-head units that are designed to supplement natural Na convection in the event of a failure in the normally operated pumps. The dc pumps are designed only to remove reactor decay heat following a reactor shutdown.

Piping and Auxiliary Equipment

Work has been resumed on piping layout, pipe flexibility studies, and design of tank components. Much of this work is still seriously hampered by lack of firm information on reactor, hot traps, cold traps, and emergency dc Na pumps. Sodium piping design is progressing on a limited basis by the examination of auxiliary system details.

Control Room and Instrumentation

Data System

Engineering comments and estimates on the data system preliminary specifications were received from ten companies. An addendum to the preliminary specifications has been prepared which, together with the preliminary specifications, constitutes the final specification. The addendum contains clarification of sections of the preliminary specifications about which vendors had questions. In addition to answering these questions, the addendum includes some modifications, based on vendor's engineering comments and estimates, that will permit one or more vendors to make proposals utilizing their own particular product; this could result in a lower cost for the same performance.

The final specifications were sent out for firm proposals and bids on July 3 and July 8 from Los Alamos and Los Angeles, respectively. The proposals were due in five weeks, but extensions have been granted.

Preliminary flow diagrams have been prepared for the following typical CTF problems: (1) Analog Scan Routine; (2) Digital Scan Routine; (3) Alarm Routine; (4) Nuclear Measurement Subprogram; and (5) Sodium System Startup Subprogram.

Core Cover Gas Cleanup

Possible manufacturers of diffusion equipment were visited, in order to gain an idea of the state of development of such equipment and to determine whether it was considered feasible to manufacture diffusers suitable for the CTF application.

Samples of fused quartz-microtubing were provided for a gamma irradiation test of 10^9 r; there was no visible effect. It is concluded that radiation damage to this type of glass will not constitute a problem. Specifications are being prepared for development models of a He diffusion purification unit and for a He circulator.

Safety Analysis Review

Representatives from LASL met on June 11, 1963 with a subcommittee of the Advisory Committee for Reactor Safeguards to discuss CTF safety problems. Also in attendance were DRD and DIR representatives. Preliminary to consideration of CTF hazards, two LASL presentations were made; one concerning LAMPRE I operating experience, the other concerning the health hazard of plutonium.

Discussion of hazards was focused on the following points:

1. Filter efficiencies of both the absolute and iodine filters.
2. Filter performance at elevated temperatures.
3. Release of radioactive iodine during the MCA.
4. Release of plutonium during the MCA.
5. The assumption that ventilation system No. 2 remained operative during the MCA.

The ACRS met in Los Alamos July 11, 1963 and consideration of the CTF was on the agenda. The Committee decided that since no core was specified and since DRD had already authorized construction, they would make no formal review of safety. It is understood by all concerned that a complete review will be necessary prior to operation of Core I.