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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 NOVEMBER 20, 1963

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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

Reactor

The reactor has been in a standby condition during this quarter. The core loading has remained unchanged at 128 Pu-Fe loaded capsules plus three test capsules containing Pu-Co-Ce alloys. The Na coolant temperature has been held at 385°C and the flow rate at about 1/3 of the normal operating value.

Several fuel capsules in the reactor are known to be bonded to their handles. Preparations are being made to separate these capsules from their handles by applying higher torques than are possible with the installed equipment, and to remove a capsule and its handle through the top of the capsule charger, if necessary. Mechanical components and the necessary shielding (see below) are being assembled for these operations.

Calculations are being made to determine the nuclear characteristics of Core III proposals. It is planned to use this core to study the performance of Pu-Co-Ce fuels. For the alloys of interest, a major fraction of the core will have to contain Pu-Fe fuel in order to achieve criticality.

Calculations

A study is being made to evaluate the possibility of using overlapping neutron energy groups for fast reactor calculations. This proposal differs from previous usage of the technique (cf. G. P. Calame, et al., Nucl. Sci. and Eng., 10, 31 (1961)) in its intended application to an S_n formulation and to problems in which there is little resonance or thermal flux. The steady state and transient codes for studying the FRCTF dynamics have been combined to facilitate calculations, and the new code is being used to study FRCTF transients.

Fission Product Activities

A code has been developed for computing the activities of fission product chains in a circulating coolant loop. The code will handle up to 50 chains, will permit up to 10 members in each chain, and will compute activities in up to 10 loop components. Initial application of the code is being made to components of the FRCTF gas cleanup system in order to ascertain the distribution of thermal and radiation sources in the loop.

Shielding Studies

Several United Nuclear Corporation Monte Carlo codes have been acquired and modified for use on the IBM 7094 computer. They permit calculation of neutron or gamma-ray transmission through spherical or rectangular geometries. The codes are being used to design the shielding required for removal of the LAMPRE capsules which are bonded to their handles.

The code RAY has been written to trace rays between pairs of arbitrary points in a three-dimensional geometry composed of rectangular parallelepipeds of differing compositions. The code computes the sum of the products of path lengths and densities between a pair of points and is being used to check for possible thin spots in the shielding of the FRCTF gas cleanup cell.

Fuel Containers

Metallographic examination of LAMPRE capsule 1454 has been completed and shows severe wall thinning at the liquid level (see IAMS-2973) . Minimum wall thickness found was 13 mils, compared with the original wall thickness of ~ 24 mils. Intergranular penetration of the weld was observed, confirming the view that Ta weldments are more susceptible to intergranular penetration than is the base material. Approximately 80% of the LAMPRE Core II capsules examined have exhibited intergranular penetration of the weld region, with no indication of penetration at any

other point. Possible differences in composition and structure of the weld region and the parent material are under study.

Capsule 1524, which contained Pu-Co-Ce fuel (5 g Pu/cm³) in contact with approximately an equal volume of Na added to the capsule, has been removed from the reactor. Metallurgical examination revealed no change from the initial condition except for the presence of a reaction layer in the vapor phase portion of the capsule. Although this capsule had sustained 41 MWD radiation exposure, some doubt exists as to whether the fuel was ever completely molten. Resolution of this question pends examination of its mate which is still in the reactor.

Corrosion testing of LAMPRE fuel containment materials is continuing. Results of 2800-h tests of Pu-Fe containment in both electron-beam-melted and double-arc-melted Ta show definite superiority of the double arc-melted material for containment (Fig. 1). These tests have been carried out isothermally at 600°, 650°, and 700°C. Although both containment materials perform equally well at 600°C, the double-arc-melted product is better at the higher temperatures.

As mentioned previously (IAMS-2973), Ta welds produced by both electron beam and TIG processes fail by intergranular penetration before the base metal becomes penetrated. Surface films are being investigated as a means of protecting the internal Ta surface from the corrosive Pu-Fe fuel. Initial results indicate carbide coating of the Ta surface to be extremely beneficial in protecting the container from intergranular penetration. Two capsules fabricated from double-arc-melted tubing and pack carburized to form a thin TaC layer on the internal tubing surface have completed 3500 h of testing at 600° and 650°C (one at each temperature) without indication of Pu penetration. Additional capsules have been loaded and will be inserted into loops as space becomes available.

Fabrication of LAMPRE Core III Capsules

One hundred fourteen capsules for Core III of LAMPRE were made from arc-melted, high-purity Ta billet K658. Finishing steps remaining include

Fig. 1. Molten Pu-Fe Fuel Corrosion Tests in Tantalum Capsules

swaging the open end of each capsule, wall thickness measurement, and final annealing. The lot will then be made available for use.

Two hundred fifty capsules were fabricated from arc melted, high-purity tantalum billet K661. These capsules have been straightened and are ready for tip machining, trimming to length, and numbering.

The fabricated capsules are being held prior to annealing so that, if suitable facilities are located, they can be annealed at 1600°C instead of at 1450°C as was done for Cores I and II. In addition to annealing at 1600°C , consideration is being given to the inclusion of an internal TaC coating. The questions regarding further capsule processing for Core III will be resolved before January 1, 1964; completed, fueled capsules should then be ready for loading into LAMPRE approximately March 1, 1964.

Capsule Fabrication - General

A method is being sought for improving the closure welding of capsules. The closure joint is made by welding the cap and capsule at the junction of the cap shoulder and capsule wall. This is a butt joint for which electron beam and heliarc welding methods have been employed. A close fit of the two parts at the time of joining is advantageous. To improve the fit, both design and dimensions have periodically been altered in the rather extensive development program.

At present, a tight fit between cap and capsule has been secured by magnetic swaging of the capsule against the inserted cap. This has given a good fit without distortion of the parts. Both as-drawn and annealed capsules have been used with a forming pulse of 17 kV. Both types of material could be formed satisfactorily; however, the annealed metal was more tightly bonded of the two. Average reduction in diameter of the capsule was 10.5 mils with negligible deviation in out-of-roundness. Three magnetically swaged mockup assemblies were welded: one by electron beam, two by heliarc. All welds appeared visually to be sound. Metallographic examination of the welds will be made.

Projection ring specimens using a modified closure design were welded by Sciaky Bros., Chicago, and returned for evaluation. Heat treatment of the welded samples seems to have removed evidence of the bond interface. Grain size is much finer than is found in fusion welded samples.

A number of tube-tube, butt-welded Ta samples has been made for various studies including heat treatment, effect of a carbon addition, cold work effect, and cycle heat treatment. Capsules have also been made from Ta-5W and Ta-10W alloys for similar tests. This work is related to the need for longer capsules than can be made by the present impact extrusion and ironing process.

Pu-Co-Ce Fuels

Corrosion testing of Pu-Co-Ce fuels is continuing at temperatures from 600° to 700°C. Fuel compositions being employed are 5, 5.7, and 8 g Pu/cm³; the 3 g Pu/cm³ fuel is being eliminated from tests because of low interest regarding its potential breeding characteristics in large power systems. Fuels of 5 and 8 g Pu/cm³ contained in unalloyed Ta have logged up to 4200 h at 600°C and up to 2800 h at 650° and 700°C. No failures have been observed with 5 g Pu/cm³, but one failure (out of four tested) occurred at 650° and one at 700°C with the 8 g Pu/cm³ composition.

The reaction layer between Pu-Co-Ce fuel and Ta has been shown to consist of two distinct zones. One, near the Ta surface, is a diffusion layer consisting predominantly of Pu and Co dissolved in Ta with some Ce present. The outer layer, next to the fuel, is entirely Ta and Co with no Pu or Ce detected by electron microprobe analysis. This Ta-Co layer appears to precipitate from the fuel either during test or upon cooling from test temperatures. Experiments have shown that the outer layer will not form until the fuel is saturated with Ta; hence, the surface/volume ratio of the system and the composition of the fuel are major variables in determining the kinetics of film formation.

Pu-Mn-Ce Fuel Alloys

Differential thermal analysis work has been completed on five alloys of the Pu-Mn-Ce system having approximately a constant Pu concentration of 3 g Pu/cm³. The data indicate that there is a valley at $638 \pm 3^{\circ}\text{C}$ and a nominal composition of 25 a/o Pu-12 a/o Mn-63 a/o Ce. It appears that 5 and 8 g Pu/cm³ compositions in this system will probably have melting points below 638°C .

LIQUID METALS TECHNOLOGY

Introduction

A research and development program has been initiated to develop areas of liquid-metal coolant technology necessary to permit systematic design of high-temperature, ultra-pure reactor coolant systems. The problems are largely associated with the more exacting Na quality control methods required for reactor systems using refractory metal core materials or systems involving direct contact of fuel and coolant. The program includes the determination of the required degree of coolant purification, a study of various methods of achieving and maintaining minimum impurity levels, and the development of necessary instrumentation and analytical techniques to measure system coolant quality.

Hot Trap Evaluation Loop

The Hot Trap Evaluation Loop (HTEL) is an experimental loop designed specifically for studying the effectiveness of materials in "gettering" O from Na. The loop (Fig. 2) operates with a nominal flow of 1 gpm and contains three test sections in which materials may be exposed. The O content of the loop is maintained by a large by-pass cold trap and is measured by a plugging indicator and distillation sampler. The test sections shown in Fig. 3 were fabricated from 2-in. Schedule 40 pipe with a 5-in. tube with a 1/4-in. wall forming the outer annulus. A temperature gradient of 100°C is maintained along the 40-in. length with

Fig. 2. Hot Trap Evaluation Loop Flow Schematic

Fig. 3. Schematic of HTEL Test Section

the heaters at the bottom of the test section. Samples are introduced and removed from the test section through the 2-in. ball valve.

The test samples are 1/2-in. square tabs, 0.005 in. thick. They are degreased, vacuum annealed and recrystallized, weighed, and then attached to the sample tree with Ta wire. After exposure the samples are removed in a He atmosphere and immediately immersed in n-hexane. They are cleaned by immersing in acetone, water, and finally acetone; after drying they are stored in a vacuum desiccator. Materials presently under study are Zr, Zr-50% Ti, Ta, Ta-10% W, and Nb.

Preliminary results show that Zr and Zr-50% Ti alloy increase in weight in Na with 15 to 20 ppm O over the temperature range studied, i.e., about 550° to 700°C. On the other hand, Ta, Ta-10% W, and Nb exhibited a decrease in weight. It is possible that, in these latter cases, diffusion is occurring into the base metal. Further work is proceeding with the HTEL.

CORE TEST FACILITY

Phase A

Shop Drawing and Equipment Submittal

IASL personnel review most of the contractor's shop drawing and equipment submittals. Comments from IASL, AEC, and Norman Engineering, Los Angeles, are consolidated by Norman's field representative and transmitted to the contractor, Ets-Hokin and Glavan. In general, the quality and selection of equipment called out in the submittals has been quite satisfactory.

The submittals on welding procedures and welder qualification have, however, been unsatisfactory, particularly the submittals from Alpha

Engineering, the subcontractor for erection of the cell liner. The contractor has been notified that payments for welding will be withheld until approval of welding procedures and welder qualification.

Norman Engineering has prepared the schedule of administrative requirements which cover the first 35 sections of the specifications. For the most part, this schedule covers shop drawings and equipment.

Construction

Estimated completion of construction of the CTF is 6.7%, compared with the scheduled 8.5%.

The dump tank pit floor and walls have been poured and the first of three pours for the main foundation slab was made November 20. The two remaining pours are scheduled to be completed by November 27, 1963.

The Contractor's PERT network has been resubmitted and reflects some changes required by Norman Engineering. The resubmittal is awaiting approval by the latter before acceptance by the AEC.

Change Order No. 1

The change order request of August 8 was transmitted by IAAO to Norman Engineering for execution of the required work. In order to prevent purchase of equipment which might be inappropriate after the details of the changes are known, a broad hold order was given to Ets-Hokin and Galvan by IAAO.

Drawings and specifications for the change order were received by the AEC from Norman Engineering and delivered to the contractor for a proposal on November 4, 1963. So far, there has been no comment from the contractor concerning the change order.

Sodium Systems

Intermediate Heat Exchangers

The 12-in. and 20-in. stainless steel piping for the IHX shells was welded at SWEPCO Tube Corporation during October. Radiographs of the SWEPCO welds have been sent to LASL for evaluation. Tubing manufacture

by Allegheny-Ludlum for Baldwin-Lima-Hamilton has been completed and an inspection of the material and testing methods is being made by IASL personnel the week of November 18, 1963. A IASL conference with Baldwin-Lima-Hamilton representatives held at their Eddystone plant on September 25 revealed very little active work being performed in their shops.

Heat Dumps

A IASL shop inspection held at the Struthers Wells' plant in Warren, Pa. on September 23 and 24 revealed that fabrication of heat exchanger pressure parts is progressing satisfactorily.

A IASL shop inspection was also conducted at the Smith Moon Steel Company, Winfield, Kansas on September 26. This company is fabricating structural parts, ductwork, combustion chambers, stacks, etc., for the heat dump systems. The work had been reported as essentially complete and the fabricator was loading parts for shipment at the time of the plant visit. However, the inspection revealed numerous cases of unsatisfactory design and workmanship. Shipment has now been postponed until the necessary corrections are made.

One of the major problems is inadequate thermal insulation in the heat dump settings and stacks. A visit to Struthers Wells the week of November 18, 1963 will be made to discuss the problem of the insulated panels on surfaces that are contacted by 1300⁰F flue gas.

Shipment of the parts of the heat dump that are being supplied by Buffalo Blower are expected to arrive November 22, 1963. They are not involved, however, in the insulated panel situation. Work has begun on structural and electrical drawings for the installation of the heat dump.

Pumps

General Electric has submitted an outline drawing of the 500 gpm E.M. pump. This and the pumping section drawings were discussed during a visit to G.E. on November 19, 1963. General Electric has agreed to

consider LASL suggestions for strengthening the inlet-outlet region of the pump to enable it to withstand the higher stresses which may be imposed by the piping system.

The emergency pump proposals were discussed with Atomics International and with General Electric. Specifications will be prepared and an order placed for these pumps as soon as a selection is made between the proposals.

A visit was made to Superior Electric Company, Bristol, Conn., to discuss the possibility of using H-C Powerstats as power supplies for the two Seawolf pumps. It was decided that they are suitable and procurement specifications for these units are in preparation.

Electromagnetic Flowmeters

The two 6-in. and two 4-in. electromagnetic flowmeters have been received from Atomics International.

Surge Tanks

A preliminary calculation of pressure surge characteristics due to variable Na flow has been performed for the 15 MW RaNa loop in order to provide information for adequate design of the system surge tanks. The tanks have now been sized, based upon the following assumed conditions:

Range of Na flow rates	0-2000 gpm
Developed Na pump pressure at 2000 gpm	80 psi
Maximum reactor gas volume	250 ft ³
Total reactor pressure drop	29 psi
Hot trap pressure drop	4 psi
Maximum system pressure at 2000 gpm (pump out)	85 psi

System Dynamics

The steady state and transient codes have been combined. To start a transient calculation it is necessary only to specify power level, Na flow rates, and core inlet Na temperature. Steady state values of all

system parameters are then computed by the code and the specified transient is initiated with the initial conditions so established.

A simple interlock system capability has also been incorporated into the code:

- 1) Any scram causes the heat dump to be shut off.
- 2) High Na temperature at heat dump outlet causes a scram.
- 3) High Na temperature at reactor vessel outlet causes a scram.
- 4) Low Na temperature at reactor vessel inlet causes a scram.
- 5) Low Na flow in either loop causes a scram.
- 6) Low Na flow in either loop causes reduction of flow in the other loop.

This last feature was included because it prevents excessive Na temperatures at the pump throats.

Data System

The final bid extensions for the CTF Data System were cut off on September 9, 1963. Seven proposals were received out of 13 companies solicited. Four of the seven proposals are priced within the CTF budget for this system and all four of them, as amended, meet the specification with no apparent exceptions. However, there is still some question as to the detailed techniques to be used, particularly as regards input/output buffering and asynchronous operation of some of the peripherals.

Approval has been received from AL00 to proceed with the purchase of the Data System proposed by Honeywell. Although the specifications are not quite in a form acceptable to LA00, a letter of intent to purchase has been sent to Honeywell with the hope that they will initiate work on the job.

Hot Cells

Transfer Cell Mockup

Installation of a General Mills Model 300 bridge-type manipulator is complete except for control wiring hookup. This manipulator, a pair of

Model A master-slave manipulators, and a bridge-mounted hoist will permit check-out of the scheme for transferring alpha and gamma hot samples from the disassembly cell. Model A manipulator sleeves have been installed.

Pressure and Temperature Analysis

A hypothetical maximum credible accident (MCA), different from that in the Safety Analysis Report (SAR), has been postulated to establish the requirements of the vent filter train of ventilation system No. 2. The principal difference between the new MCA and one in the SAR is that dumping of fuel and radioactive Na from the vessel results from the absence of core cooling due to a site power outage. In the new MCA capability of space cooling by ventilation system No. 2 does not exist and temperatures are much higher.

The variation of pressure and temperature with time have been recalculated using the following assumptions and inputs:

- 1) A site power outage results in a release of fuel and radioactive Na to a catch pan on the reactor pit floor. In the absence of a specific core and vessel design, it is assumed that the release is instantaneous with the loss of power.
- 2) All the fuel drains out and the decay heat power is that of a 15 MW core operated continuously for one year.
- 3) Both the 15 MW and 5 MW radioactive loops are drained, and 10,000 lb of Na with an average temperature of 1000°F are released.
- 4) The catch pan has a mass such that the equilibrium temperature of pan and Na plus decay heat for the first hour is 600°F.
- 5) Only the oxygen in the pit is available for reacting with Na and the burning rate is 0.3 lb/ft²/h over the pit floor area of 484 ft². Burning proceeds at this rate until all oxygen is consumed.
- 6) Pit atmosphere cooling by the graphite thermal mass cancels the heat leak of the vessel to the pit atmosphere. (In other words, these two contributions to a heat balance are omitted.)

7) The contribution of the heat leak in the RaNa room to the average space temperature assumes a thermal mass of empty, insulated pipe.

8) The RaNa dump tank and pit are considered not to influence space temperature.

Twelve hours after the start of the accident some of the important temperatures are:

Reactor Pit Atmosphere	720°F
Reactor Pit Liner	690°F
RaNa Room Atmosphere	200°F
RaNa Room Liner	200°F
Average Atmosphere	250°F

Maximum pressure with vent open is 14 in. of water, occurring 3 min after the spill. At the end of 12 h with no venting space pressure is 2.2 psi. During the 12 h, 13% of the total atmosphere mass is vented through the filter train, a fraction comparable to that used in calculating the stack release dose in the SAR.

PHASE STUDIES OF PLUTONIUM ALLOYS

Plutonium-Neodymium Alloys

The addition of Nd to Pu causes a eutectic lowering of the melting point of Pu to 628°C. The solid solubility of Nd in ϵ -Pu at 600°C is between 2 and 3 a/o and in δ -Pu at 450°C is between 1 and 2 a/o.

The maximum solubility of Pu in β -Nd is about 33 a/o at 820°C where the α -Nd solid solution decomposes into liquid plus β -Nd solid solution, the latter containing about 27 a/o Pu. With decreasing temperature the solubility decreases to 26 a/o Pu at the eutectic isotherm (628°C) and to about 17 a/o at 450°C.

Plutonium-Praseodymium Alloys

The addition of Pr to Pu causes a eutectic lowering of the melting point of Pu to 625°C. The solid solubilities of Pr in ϵ - and δ -Pu are less than 2 a/o.

At 794°C on cooling, β -Pr containing slightly more than 30 a/o Pu in solution decomposes into the liquid phase plus α -Pr containing about 30 a/o Pu in solution. With decreasing temperature the solubility of Pu in α -Pr decreases to about 25 a/o at 600°C and to about 13 a/o at 400°C.

With 5 or more a/o Pu in solution the double hexagonal, close-packed, α -Pr solid solution transforms to a cubic close-packed structure at elevated temperatures. The transformation temperature is dependent upon composition. In the two-phase, ϵ -Pu plus α -Pr field it was found at about 534°C by thermal analysis, and in the single-phase region it increases linearly from 534°C and 21 a/o Pu, at the α -Pr solvus, to about 780°C and 5 a/o Pu.

Plutonium-Scandium Alloys

Delta-Pu dissolves a maximum of about 23 a/o Sc at about 663°C, the ϵ -Pu + α -Sc $\rightleftharpoons$ δ -Pu peritectoid temperature, and the solubility decreases to about 20 a/o at 400°C. Delta phase containing 5 or more a/o Sc in solution is readily preserved to room temperature, but the equilibrium (α + δ)/ δ boundary at 100°C lies between 10 and 15 a/o Sc.

Preliminary work indicates that the maximum solubility of Pu in α -Sc is of the order of 25 a/o.

Plutonium-Yttrium Alloys

The addition of Y to Pu results in the eutectic lowering of the melting point of Pu to 635°C.

Micrographic examinations of both cast and homogenized Y-rich alloys containing 5 to 20 a/o Pu indicate that a maximum of 15 to 20 a/o Pu is soluble in α -Y. The x-ray parametric method gave values of 16 a/o Pu at 600°C and 14 a/o Pu at 450°C. These results must be considered uncertain, however, until it is known if oxygen has a significant effect on the solubility of Pu in α -Y. Yttrium monoxide has been observed, both as

a Widmanstätten precipitate and also, in some instances, as primary dendrites, in the alloys and in the Y melting stock used in preparing the alloys.

Plutonium-Cerium-Cobalt Alloys

The crystal structure has been determined of the C phase $(\text{Pu,Ce})_5\text{Co}_3$ in the ternary Pu-Ce-Co system. This compound is tetragonal, space group $I4/mcm$, $a = 10.730 \pm 0.003 \text{ \AA}$, $c = 5.383 \pm 0.002 \text{ \AA}$, and has four formula units per unit cell. It is isostructural with Pu_5Si_3 (D_{8h}^m type). A structure which has vacancies, substitutional disorder, and positional disorder is proposed. The phase contains 37 a/o Co and its Ce content ranges from 4 to 23 a/o. The calculated and measured densities and the apparent thermal parameters support the proposed structure.