325
7963
LAMS-2915

MASTER

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

QUARTERLY STATUS REPORT ON LAMPRE PROGRAM
FOR PERIOD ENDING MAY 20, 1963

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

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

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

LAMS-2915
UC-80, REACTOR TECHNOLOGY
TID-4500 (20th Ed.)

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

REPORT COMPILED: June 1963

REPORT DISTRIBUTED: June 25, 1963

QUARTERLY STATUS REPORT ON LAMPRE PROGRAM
FOR PERIOD ENDING MAY 20, 1963

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 25.9 MWD of operation during the quarter to give a total integrated power with this core of 96.1 MWD. Except for brief periods, the power level has continued to be limited to 500 kW. The reactivity loss rate characteristically amounts to several cents per hour at the start of a run, drops below a cent per hour within the first few days, and continues to get smaller during the remainder of the run; at the end of run 66, the longest continuous run with Core II, the loss rate was about 0.16 ¢/h . The operations during the quarter are summarized in Table 1.

TABLE 1

Summary of LAMPRE I Operations

<u>Run No.</u>	<u>Start Date</u>	<u>End Date</u>	<u>Power, MW</u>	<u>Accumulated MWD</u>
66	3/8/63	4/3/63	0.5	82.9
67	4/5/63	4/9/63	0.5	84.8
68	4/11/63	4/30/63	0.5	94.0
69-72	(Operator Training)		≤ 0.8	94.4
73	5/17/63	--	0.5	96.1

Prior to Run 66 the core loading was increased by one capsule to give a total of 138 Pu-Fe capsules; in addition there are three capsules containing small amounts of Pu in a Pu-Co-Ce alloy. The total fuel in the core is 23.53 kg Pu less burnup.

Irradiated Capsule Scanning

A capsule which was placed in the core just prior to Run 66 was removed immediately following the run to freeze the fuel and fission products in their locations at the time of shutdown. The capsule was stored for about a month to reduce the total source strength to a manageable level; the capsule was then scanned to determine the locations and strengths of various γ -ray emitters. About 91% of the Ba-La¹⁴⁰ activity was concentrated in a band 0.1 in. wide at about the top of the fuel. Previous capsule scans have shown a double band of this activity in the same region. It is thought that the two bands correspond to the fuel levels during operation and during shutdown, respectively, and that the latter band was eliminated by the rapid freezing of the fuel. Over most of the length of the fuel the Ba-La¹⁴⁰ activity was relatively low, but at the bottom half inch of the capsule it was about that expected from burnup in that region; apparently the Ba-La transport mechanism was not effective near the bottom of the capsule. The Zr-Nb⁹⁵ activity along the fuel seems to follow the fission distribution, indicating that there was little transport of these nuclides during reactor operation. An interesting result of the scan was the identification of I¹³¹ in a narrow band between the fuel and the top closure plug of the capsule; this isotope has not been identified in scans of other capsules.

Heat Transfer Calculations

Because of the similarity of the basic equations involved in viscous flow and heat conduction, it was possible to use the Astra HEATING code to calculate the relative velocity distribution that would exist in the LAMPRE core Na channels under laminar-flow conditions. A Na heat sink distribution was specified which was proportional to these relative velocities and which was normalized so as to balance total heat sink in Na with total heat source in fuel. This laminar-flow heat sink was then used in the corresponding LAMPRE heat conduction calculation performed by the same HEATING code.

As a comparison, an otherwise identical problem was run using a constant heat sink per unit volume of Na, or so-called "slug" flow, i.e., velocity uniform over the Na channels. The actual temperature distribution and the actual heat transfer coefficient should lie somewhere between these limiting cases. Some of the more important results of the calculations are given in Table 2.

TABLE 2

Conditions at Center of LAMPRE Core at 1 MW

<u>Item</u>	<u>Laminar Flow</u>	<u>Slug Flow</u>
1. $\Delta T^{\circ}\text{C}$ (Av o.d. Ta to Av Na)	19.3	14.2
2. $\Delta T^{\circ}\text{C}$ (Av i.d. Ta to Av Na)	46.9	42.1
3. $\Delta T^{\circ}\text{C}$ (+ from mean, o.d. Ta)	9.1	3.9
4. $\Delta T^{\circ}\text{C}$ (+ from mean, i.d. Ta)	5.2	2.2
5. Ratio Item 4 to Item 2	0.11	0.05
6. h ($\text{w}/\text{cm}^2\text{-}^{\circ}\text{C}$)	10.72	14.49
7. h ($\text{Btu}/\text{h}\text{-ft}^2\text{-}^{\circ}\text{F}$)	18,880	25,510

The heat transfer coefficient is based on the Ta average surface temperature to average Na temperature.

The results show that the circumferential temperature variation at the inner surface of the Ta is only about half that at the outer surface and that at a given cross section the hottest Ta to average Na

temperature difference is increased by 5 to 11% by the circumferential temperature variation.

RESEARCH AND DEVELOPMENT

Core Calculations

Excellent results have been obtained so far in applying two-dimensional transport theory codes to a detailed model of LAMPRE I. Critical mass data at 160° and 480°C are given in Table 3. The

TABLE 3
Comparison of Calculations with Measurements

<u>Temp. (°C)</u>	<u>Critical Mass (kg Pu)</u>		
	<u>TDC</u>	<u>DDK</u>	<u>Experimental</u>
160	23.07	23.11	22.98
480	23.12	23.06	23.37

calculated total worth of the control rods and shim was found to be 5.20 kg Pu, in very good agreement with the experimental worth of 5.04 kg Pu. During this quarter the new IASL code DDK has become operative and is being used instead of the IASL code TDC for most two-dimensional fast reactor calculations. In addition, the UKAEA two-dimensional multigroup diffusion code CRAM has been employed for shielding evaluations and thermal reactor calculations.

The Fortran code DPC, that converts reactor engineering data into punched card input for the TDC, DDK, and CRAM codes, is now in use. This code facilitates the accurate preparation of input data for complex reactor systems and provides permanent records of the reactor models assumed in the calculations.

CORE TEST FACILITY

Survey of Core Calculations

A survey of various cores for potential use in the Core Test Facility (CTF) is being made, using a simplified core representation. The major current effort is directed toward the inclusion of thermal limitations, in conjunction with the nuclear core properties. The survey includes all combinations of:

Fuel content: 16 g/cm³ in Fe; 8, 5.7, 3 g/cm³ Pu in Co-Oe

Capsule Diameter: 1.0 and 1.5 cm

Capsule Wall: 0.025 in. Ta

Capsule dia/pitch: 0.75, 0.85, 0.9

At present, three generalizations are available from the analysis which has been completed: (1) Nuclear performance of the core is insensitive to core height-to-diameter in the range from 0.75-1.25. (2) The Pu utilization (power produced/Pu destroyed) is essentially constant within the range of parameters considered. This indicates that the neutron spectrum is similar in all of the cases considered. (3) Neutron economy is significantly affected by the Ta/Pu content, and slightly by the Na/Pu content. In both cases, an increase in the relative Pu content improves neutron economy.

Phase A

Title II plans and specifications for the CTF have been reproduced and were mailed by Norman Engineering to prospective bidders on April 15, 1963. The bid opening date has been set by the AEC for May 29, 1963.

The contractor is required to plan and schedule his operations using a critical path network made up of a summary network and such subnetworks as required to define each feature of his proposed plan of operation. All networks will be activity oriented and drawn so that the length of an arrow represents graphically the expected duration of each activity.

Calculations indicate that the sizing by Norman Engineering of the Barnebey-Cheney Co. radioactive iodine traps for Ventilation System No. 2 exhaust is unacceptable. The criteria specified that the filters, charcoal trap, exhaust line, etc., should be sized to provide 1000 cfm flow at +5.0 in. of water pressure within the space. Norman Engineering estimates that a space pressure of 24.7 in. of water is required to provide an exhaust rate of 1000 cfm through the exhaust system. The pressure drop chargeable to the iodine trap is 15.6 in. of water. Norman Engineering has been requested to increase the capacity of the traps and to consider the suitability of the new Barnebey-Cheney filter tested at Savannah River. The pressure drop through this filter is appreciably less than that through the presently specified charcoal filter.

Sodium Systems

Intermediate Sodium Heat Exchangers

The Baldwin-Lima-Hamilton Co.'s assembly drawings, welding specifications, and proposed inspection procedures have been reviewed. The following modifications to the manufacturer's design are being negotiated:

It appears that the proposed shell flow baffle arrangement could result in operating problems. In particular, a Na inlet baffle is being provided for the purpose of preventing direct Na impingement on the tube bundle. The baffle location is such that Na will be deflected up into the gas thermal barrier below the tube sheet. Such an arrangement may result in gas entrainment which could remove the gas from the thermal barrier area.

The proposed radiographic specifications for the tube-to-tube sheet welds permits porosity defects of up to 0.035 in. or roughly 42% of the tube wall thickness. Negotiations are proceeding to reduce this acceptance standard for defect size.

The test procedure calls for a 1560-psig pneumatic pressure test for both heat exchangers. For safety reasons, it is proposed that hydrostatic pressure tests using demineralized water be substituted for the pneumatic tests.

Heat Dumps

Approximately 80% of Struthers Wells design drawings have now been checked and approved. The heat exchanger tubes are now being finned by the G.M. Jackson Co.

A review of Struthers Wells' revised heat transfer calculations indicates that the equipment is marginally sized for the specified load and temperature conditions. The manufacturer has been advised that the units may prove to be undersized. Struthers Wells is investigating what can be done to increase cooling air fan capacity and to increase heat transfer surface.

Preliminary design of heat dump equipment foundations has begun. These foundations are planned to be included in Phase B, but the building contractor will be asked to give a price for making them.

Pumps

Specifications have been written for the two emergency dc pumps which will provide flow in the two radioactive Na systems during a site power outage or primary pump failure. Some of the pump requirements are tabulated below. Requirements, except for flow, are the same for both pumps. Flow for the 15-MW loop pump is 50 gpm and for the 5-MW loop pump 20 gpm.

Design life, years	10
Maximum Na ²⁴ activity	2 x 10 ⁹ with 2.8 MeV
γ's/sec/cm ³	2 x 10 ⁹ with 1.4 MeV
Developed head, ft of Na	0.55
Normal operating temp., °F	1000
Design Temperature, °F	1200
Max. system pressure, psig	150
Power Supply	Storage batteries

Control Room and Instrumentation Data System

Preliminary specifications for the data system have been sent to possible suppliers for engineering comments and estimates; replies are due May 27, 1963.

Nuclear Safety System

Preliminary contacts were made with three possible suppliers of this system. Based on some very tentative design concepts, it seems as though the cost will be several times the amount originally budgeted.

System Analysis

The following transient problems have been investigated:

1. Reactor scram followed by shutdown of heat dump. The purpose is to observe the effect on core inlet temperature and obtain information regarding maximum acceptable time for achieving shutdown of heat dump.
2. Accidental shutdown of heat dump followed by reactor scram, in order to observe effect on E.M. pump throat temperature.
3. Loss of flow in secondary Na loop followed by reactor scram and shutdown of heat dump, to determine the effect on E.M. pump throat temperature.

Safety Analysis Report

In a number of instances, Title II plans and specifications are different from the Facility description appearing in the Safety Analysis Report. One difference which affects maximum credible accident calculations is the method of reducing the radioactive Na space pressure after the MCA. In LA-2735 it was assumed that when space pressure exceeded -6.34 in. of water the static pressure control blowers of Ventilation System No. 2 started and removed gas from the space at the rate of 1000 cfm. Because of routine operation requirements, the present exhaust system design shows a 25/100 cfm rating through the blowers, but a valve bypassing the blowers automatically opens when space pressure exceeds +1.0 in. of water. Calculations indicate that 24.7 in. of water space pressure are required for 1000 cfm exhaust flow. The most severe radiation hazard is that of whole body gamma dose from the cloud. At 1150 m (nearest uncontrolled area) the dose given in LA-2735 is 26 r. The calculated MCA dose for the system shown by Title II plans is 23 r. In both cases thyroid and bone doses are negligible.

Preliminary copies of a revision to CTF Safety Analysis Report (IA-2735) reflecting the above change were mailed to copy holders in Albuquerque and Washington who are reviewing the report.

FUEL AND CAPSULE DEVELOPMENT

Examination of LAMPRE Capsules

Metallographic evaluation of LAMPRE fuel capsules has been continued on a limited scale. Examination of one capsule indicated heavy precipitation, both intergranular and transgranular, the precipitates being of the same morphology as tantalum nitride. The capsule was deformed during the cutoff operation and evidence of intergranular fracture occurred which was indicative of brittleness. No intergranular penetration was observed; however, a reaction zone was found in the vapor phase section of the capsule.

Gas analysis of LAMPRE cover gas has shown for some time that N_2 builds up at almost a linear rate in the cover gas system. Because of this observation and the fact that precipitates, similar to tantalum nitrides, are observed in all LAMPRE capsules, a calculation was made to determine if the N_2 entering the reactor system could be due to diffusion through the stainless steel portion of the reactor system. The results show that diffusion cannot be ruled out as the source of N_2 in the system. This fact is important for reactor systems in which the coolant portion is of stainless steel and the reactor core is built from a strong nitride former such as Ta. Investigation of the effects over long periods of time of small amounts of N_2 in Na on Ta and other refractory metals will be continued.

Additional Na samples, taken from LAMPRE, have shown no consistent trend for Fe, Cr, or Ni content; however, Pu levels are below the 1 ppm limit of detectability. Duplicate analyses taken from the same Na sample have shown large variations in Fe content, i.e., 60 ppm vs 20 ppm, indicating that there is a problem in obtaining a uniform sample representative of the Na coolant. This matter is being studied further.

Out-of-pile testing of Ta containers indicates that double arc-melted Ta is better than electron-beam melted material for the containment of Pu-Fe fuel. In addition, a technique of autoradiomicroscopy is being developed to determine when Pu is present in Ta grain boundaries. In general, when a capsule shows gross Pu failure, i.e., Pu on the outside surface of the container, a general penetration either in the gas phase or below the liquid level is not observed. Therefore, it is quite difficult to locate metallographically the areas in which the Pu has penetrated. Moreover, it is impossible to determine by normal metallographic techniques whether or not a grain boundary has Pu in it. Therefore a technique like that of the autoradiographic microscopy, or something similar, is required for absolute proof of Pu penetration. Considerable effort is being given to this problem.

Fabrication of LAMPRE I Capsules

A pilot lot of 50 capsules is in production to study the effect of process variables on the formation of defects. The material is double arc-melted, high-purity Ta from billet K658. The billet was formed into rod from which slugs were cut. The slugs were impact extruded into cups which were machined and polished on the outer surface. The cups will be processed through the several ironing stages. At each stage of ironing, the capsule will be examined and repolished if necessary to remove surface defects. It is hoped to produce final capsules which will have smooth surfaces free from defects. If this pilot lot is approved, production will begin on the next core loading for LAMPRE.

The bottom thickness of impacted cups has excess material which tends to create a double curvature in the closed end of finish drawn capsules. Removing excess metal by machining the impacted cup seems to prevent the defect. After the second stage draw the capsule is repointed to the 60° included angle. The remaining stages are drawn by the usual procedure.

In the past, impact extrusion through the five stages from slug to cup was performed using one punch 2-3/4 in. long. Deflection in the punch during impacting at 30-45 tons caused variation of wall thickness which exceeded 10 mils for 50% of the cups. Once the wall variation was established, it was maintained through the ironing stages. There was no means other than machining available to correct the situation.

To reduce punch deflection, three punches were used having different lengths: 1-3/8 in. for the first three stages; 1-7/8 in. for the fourth stage; and the standard 2-3/4 in. punch for the final stage.

Considerable improvement in wall thickness resulted: 60% of the impacted cups were within a 5-mil variation, 25% within 8 mils, and 15% within 12 mils.

Experimental capsules have been fabricated by drawing and spinning a closed end on a length of 0.433 in. o.d. by 25-mil wall Ta tubing. The spun end was then heliarc welded shut. Tests of these capsules by a dye penetrant indicated defects in the welded conical tip. Metallographic examination of the tip showed metal folds apparently caused by the spinning operation. Moreover, the tip did not have full weld penetration, giving rise to hairline cracks on the inside of the tip.

Additional capsules were processed by the same technique. Each capsule was bored out at the tip, and fitted with a 0.062-in. diam Ta plug which was welded in place. Although this process for making capsules seems attractive, there is some question of reproducibility especially in respect to tip formation. Further test capsules will be prepared to acquire a better feeling for this aspect of the procedure.

Experiments are under way to examine the electron-beam welding of capsules using Ta-10 w/o W seamless tubing. To a short tube length, a machined conical tip was butted and the joint welded by the electron beam method. The welds will be examined metallographically to determine how well the closure was made. Satisfactory welds may indicate a new fabricating process which will yield a final capsule of quite uniform wall thickness.

Pu-Co-Ce Fuels

Pu-Co-Ce fuels of 3, 5, and 8 g Pu/cm³ have been tested for 1000 h at 700°C in pure Ta, Ta-10 w/o W, and Nb. No failures were obtained in the Ta-base alloys; however, the 5 and 8 g/cm³ fuels failed in Nb containers. Additional tests are being made at 700°C and life studies at 600° and 650°C. These tests are designed to determine the effects of long-term corrosion and isothermal temperatures, i.e., 10,000 h or greater.

The formation of the Ta₆Co₇ layer in Ta capsules containing Co alloy fuels appears to involve a complex mechanism. An inner layer, next to the Ta surface, contains islands of pure Ce, whereas the outer layer is pure Ta₆Co₇. In order to understand the mechanism of formation from a theoretical standpoint, the solubility of Ta is being measured in liquid fuel. The information obtained so far indicates that the total Ta solubility in Co-Ce binaries is less than 30 ppm, which is the limit of chemical detectability. This problem is also being studied by using radio-Ta tracers and preliminary indication of this work is that the Ta solubility at 700°C is less than 1 ppm.

The observation that TaC coating on the internal surface of the Ta capsule results in a marked decrease or a total absence of the Ta₆Co₇ layer has led to testing capsules coated with TaC at 600 and 650°C with Pu-Co-Ce and Pu-Fe fuels. In addition, attempts are being made to develop a W coating in the vicinity of a 1 mil thickness to determine the effect on film inhibition.

The melting point of filtered Ce was found to be 765°C, which is 35° lower than that for pure Ce, indicating the presence of impurities. Since all the Ce now being used for the preparation of Pu-Co-Ce alloys is of this type, an unknown effect of the impure Ce is present. This problem is being studied.

Alternative Ce-based Alloys

A minor effort has been initiated to determine the possibility of using Ce-based alloys other than Pu-Co-Ce as reactor fuel. These are the

Pu-Cu-Ce, Pu-Ni-Ce, and Pu-Mg-Ce alloys. Preliminary indications are that there are severe segregation problems involved with cooling of the Pu-Cu-Ce alloy; it appears, therefore, not to be suitable for use in a reactor system in which a fuel will be cooled intermittently. Preliminary determination of the liquidus surface of the Pu-Ni-Ce system indicates it to be of the same shape as the Pu-Co-Ce system. Melting points appear to be somewhat lower than 450°C for fuel compositions of interest in reactor work. Preliminary corrosion screening tests with this fuel are being initiated.

U-Mn-Pu Alloys

Screening tests have continued in an attempt to find the most suitable container for U-Mn-Pu alloys. Tests at 900°C indicate that a static fuel system dissolves Y at a slower rate than does a mobile system, the latter being simulated by tipping the fuel from one end of the container to the other periodically under isothermal conditions. With the U-Mn binary alloy, the mobile system dissolved the Y at approximately twice the rate as did the static. Corrosion tests on this system will be continued.

PLUTONIUM ALLOY SYSTEMS

Pu-In Alloys

Preliminary lattice parameter measurements of the ζ phase (Pu_3In , face-centered cubic, AuCu_3 type of structure) indicate that its composition could depart considerably from the stoichiometric ratio. The $\zeta/(\zeta + \eta)$ boundary at 900°C was successfully placed at 27.4 a/o In, however, by volume fraction analysis of the two-phase microstructures of alloy specimens containing 28 and 30 a/o In. It is of interest that above about 850°C the ζ field does not include the stoichiometric composition representative of the phase. The $(\text{L} + \zeta)/\zeta$ boundary extrapolates to about 27 a/o In at 960°C , the $\text{L} + \eta \rightleftharpoons \zeta$ peritectic point.

Pu-La Alloys

The γ -La peritectic point at 20 a/o Pu corresponds to the maximum solubility of Pu in γ -La, and falls essentially at the end of the peritectic isotherm at 835°C. Therefore, 20 a/o also corresponds to the maximum solubility of Pu in β -La. This solubility decreases to about 8 a/o Pu at 500°C.

Pu-N Alloys

Lattice parameter measurements that were made on PuN which had been equilibrated with excess N₂ showed that no large change in the lattice parameter had occurred. The value was close to 4.906 Å in all cases. The largest lattice parameter observed has been 4.913 Å for PuN melted in impure N₂. This sample contained α -Pu and Pu₂O₃ in addition to PuN.

Pu-Pr Alloys

The solubility limit of Pu in α -Pr is 30 a/o at 770°C. The peritectic point for the $L + \beta\text{-Pr} \rightleftharpoons \alpha\text{-Pr}$ reaction is 794°C with 31 a/o Pu; this also corresponds to the maximum solubility of Pu in α -Pr. The Pr-rich end of the peritectic horizontal is also at about 31 a/o Pu, corresponding to the maximum solubility of Pu in β -Pr. The earlier conclusion that the solubilities of Pu in α and β Pr are the same within the accuracy of the experimental determinations is supported by the finding that the solution of Pu in these allotropes influences the transformation temperature very little. The peritectic horizontal lies at $794 \pm 3^\circ\text{C}$, whereas 792°C is reported as the $\alpha \rightleftharpoons \beta$ transformation temperature of pure Pr.

Pu-Nd Alloys

Experimental data on the Pu-Nd system indicate that the general form of the phase diagram is similar to that for Pu-Pr alloys. The most important difference is that the β -Nd solid solution on cooling decomposes into liquid plus α -Nd, whereas the β -Pr solid solution reacts with the liquid phase to form α -Pr.

Pu-Si Alloys

Melting point data for the intermediate phases in the Pu-Si alloy system are shown in the accompanying table.

<u>Intermediate Phase</u>	<u>Melting Point, °C</u>
Pu ₅ Si ₃ (ξ phase)	1377 ± 10
Pu ₃ Si ₂ (η phase)	1441 ± 10
PuSi (θ phase)	1576 ± 10
Pu ₂ Si ₃ (ι phase)*	1646 ± 10
PuSi ₂ (κ phase)*	1638 ± 10

*Both ι and κ exist over temperature dependent composition ranges.

The two eutectic reactions, $L \rightleftharpoons \theta + \iota$ and $L \rightleftharpoons \kappa + \text{Si}$, occur at $1521 \pm 10^\circ\text{C}$ and $1234 \pm 5^\circ\text{C}$, respectively.

NON-PLUTONIUM ALLOYS OF REACTOR INTEREST

Co-Ta Alloys

X-ray examination of arc melted Co-Ta alloys in the composition range 0.5 to 50 w/o Co (1.68 to 75.43 a/o) has confirmed the tentative data published on this system. In addition, the following new information was obtained: (1) The most Ta-rich intermediate phase was found to be Ta₂Co. It has the body-centered tetragonal CuAl₂ type of structure. The lattice parameters of Ta₂Co in an alloy specimen composed of Ta₂Co and Ta are $a = 6.121 \pm 0.002$ and $c = 4.961 \pm 0.002$ Å. The calculated density is 15.04 g/cm³. (2) The phase which has the ideal crystallographic composition of Co₇Ta₆ exists over a homogeneity range of approximately 10 a/o.

Co-Nb Alloys

The crystalline deposit formed on a Nb surface in contact with a molten Ce-12 at/o Co alloy specimen at 700°C for 240 h was identified as the Co_7Nb_8 phase, isostructural with Co_7Ta_8 .