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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, 1960

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

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FOR PERIOD ENDING AUGUST 20, 1960

Compiled and edited

by

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From reports written by members
of CMB, CMF, and K Divisions

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All LAMS reports are informal documents, usually prepared for a special purpose. This LAMS report has been prepared, as the title indicates, to present the status of the LASL LAMPRE program. It has not been reviewed or verified for accuracy in the interest of prompt distribution. All LAMS reports express the views of the authors as of the time they were written and do not necessarily reflect the opinions of the Los Alamos Scientific Laboratory or the final opinion of the authors on the subject.

CHAPTER I

LAMPRE I

CAPSULE DEVELOPMENT

The certification tests (LAMS-2438, p. 15) have been continued. Most of the capsules tested were impacted and ironed material made from an arc-melted tantalum ingot containing 0.1 w/o tungsten (plus 0.2 w/o yttrium which is vaporized). Two of the four capsules which were tested in the as-worked condition failed, as indicated by the detection of α -activity outside, after between 1400 and 2100 hr in static sodium at 625°C. The remaining two capsules from this test which had been annealed, will be examined after 2800 hr.

A second ingot has been received which was prepared by melting in an electron beam furnace a mixture of tantalum and 0.1 w/o tungsten, both in the form of powders of a high degree of purity. No yttrium was used in this electron-beam melted ingot. The 3.5 in. ingot was rolled to a 1.25-in. rod and then swaged to 5/8 in. thickness. Although chemical analysis indicates no significant differences in composition, the electron-beam melted material is softer, work hardens more slowly, and anneals at an appreciably lower temperature than does the arc-melted product. In order to compare the two alloys, capsules made from both materials were put into one of the certification tests and the temperature of the sodium increased to 675°C. The eight arc melted capsules showed no alpha count after 500 hr, but three of the four electron-beam welded capsules had failed.

SODIUM SYSTEM

The temperature of coolant leaving the core is to be measured by thermocouples with junctions in the region of the splined section of the capsule handle. There is a connector at the top of the handle which, when removed, will permit the fuel element to be withdrawn for capsule replacement. Tests of the original version of the connector showed it to be unsatisfactory and the time necessary for the development of an acceptable alternative has made it necessary to postpone filling the coolant system with sodium.

The most promising solution to the problem is to use two modified coax (Microdot Corporation) receptacles. A female receptacle is set into the handle and a mating receptacle into the removable connector. The bodies of both receptacles are insulated from the element handle so as to have both thermocouple leads insulated from ground. The chief modification is the substitution of rhodium for the normal gold plating in order to reduce the possibility of damage should the connector become exposed to sodium. A number of connectors have been ordered and upon their receipt the core thermocouple assemblies will be completed.

LCX III

During LCX III (cold critical without sodium) operations, it became apparent that the use of a single safety chassis containing trip circuits for all level and rate scrams was undesirable. Level trips will be obtained from three independent and individually contained circuits (trips from any two of three necessary for a scram). Four low-level current amplifier circuits employing transistors have been ordered. Evaluation of a commercial chamber and counter has led to an order for five of each. These detectors were found to perform quite well at 150°C, the ambient temperature in some of the test thimbles.

GAS SYSTEM

A supply of 20 bottles of analyzed helium has been accumulated for the LAMPRE I cover gas system. The helium often contains over 6 ppm of oxygen, hydrogen, and moisture, even after selecting the best bottles. A portable cleanup system has been developed which can reliably reduce these impurities to less than 1 ppm, with the hydrogen being undetectable. In view of the reported observations on the gettering of hydrogen by tantalum in sodium systems, it is believed that precautions should be exercised to keep hydrogen out of the reactor sodium. The most certain means for accomplishing this is by means of a recirculating purifier or by frequent purging.

FUEL TRANSFER AND STORAGE

Title II engineering (working drawings) for the storage facility is completed.

The problem of transfer of fuel between the reactor charger, working areas, and the storage facility is being re-examined. It is desirable to be able to take capsules directly from the reactor to storage as well as to the cutoff box. Capsules to be prepared for metallography are moved first to a laboratory cave where the specimen is cut from the capsule and the specimen is then transferred to a second shielded facility. All parts of a sectioned capsule eventually are moved to storage. In addition, a portion of a stored capsule might have to be retrieved from storage for further examination. All these operations are complicated by the existence of two different types of alpha containers and the necessity for loading transfer shields from the top but discharging from the bottom into the storage facility. Methods for satisfying these requirements, employing a single complex shield as opposed to two simpler shields, are being evaluated.

PUBLICATIONS

"Fabrication of Tantalum Capsules for LAMPRE-I," LAMS-2434.

"Preparation and Fabrication of Plutonium Fuel Alloys for LAMPRE-I,"
LA-2439.

CHAPTER II

SODIUM TEST FACILITY

OPERATING SUMMARY

The 2000-kw Sodium Test Facility was initially placed in operation on March 15, 1960. Since startup, operation has been continuous except for 12 forced shutdowns required for maintenance of auxiliary steam system equipment and repair of refractory setting in the gas-fired sodium heater. Essentially no maintenance or adjustment has been required on sodium components or the steam generator. As of August 20, the following total running hours have been logged on major items of equipment:

<u>Item</u>	<u>Total Operating Time (hr)</u>
Steam Generator	1554
Intermediate Sodium Heat Exchanger	1554
Primary Loop Sodium Pump	2960
Secondary Loop Sodium Pump	3776
Gas-Fired Sodium Heater	2570
Sodium Cold Trap System	3960
Air-Operated Sodium Flow Control Valves:	
Primary Loop (two valves)	2960
Secondary Loop (two valves)	3776

Typical observed flow, temperature and pressure conditions for 100% load operation are shown in parentheses on Fig. 1. For comparison purposes, predicted design conditions are also indicated.

Steam Generating Unit

Fig. 2 is a photograph of the completed steam generating unit prior to the application of thermal insulation.

Design Heat Transfer Criteria

Design relationships used originally for predicting steam generator heat transfer coefficients are tabulated below. Due to the many acknowledged unknowns involved in predicting shell side sodium film coefficients and forced-circulation boiling water film coefficients, no exhaustive attempt was made to attain a fine degree of refinement in the original design computations. In all cases, fluid properties were evaluated at bulk temperatures. Unbaffled, counterflow conditions were assumed for all heat exchange surfaces. The effect of subcooled boiling in the feedwater heating section of the evaporator was neglected, thus leading to conservative predictions. No tube scaling factors were used for sodium side heat transfer; however, a scale resistance factor of 0.0003 was introduced for tube side computations in the evaporator and superheater.

Design heat transfer relationships were then as follows:

Shell Side Heat Transfer Film Coefficients

Evaporator, superheater and tempering sodium heat exchanger:

$$Nu = 5.8 + 0.023 (Pe)^{0.8} \quad (\text{Modified Seban equation})$$

Tube Side Heat Transfer Film Coefficients

Evaporator:

$$\text{Feedwater heating section: } Nu = 0.023 (Re)^{0.8} (Pr)^{0.33} \quad (\text{Colburn Equation})$$

0-70% (wt) Steam section: Assumed to be 6000 Btu/hr-°F-ft²

70-100% (wt) Steam section: Assumed to be 2000 Btu/hr-°F-ft².

Fig. 1. Comparison of Design and Operating Test Data of Sodium Test Facility. (Observed test data are in parentheses; other numbers are design data).

Fig. 2. Erected Steam Generator Prior to
Application of Thermal Insulation

Superheater:

$$Nu = 0.023 (Re)^{0.8} (Pr)^{0.33} \text{ (Colburn equation)}$$

Tempering Na IX:

$$Nu = 7 + 0.025 (Pe)^{0.8} \text{ (Martinelli \& Lyon equation)}$$

The symbols Nu, Re, Pr, and Pe refer to the familiar Nusselt, Reynolds, Prandtl, and Peclet moduli, respectively.

Observed Heat Transfer Data

The steam generator has been operated under steady state loads from 1500 kw (75% rating) to 2000 kw (100% rating). Overall heat transfer coefficients based on the observed 100% load operating data accumulated so far are tabulated below and compared with predicted design values:

	Overall Heat Transfer Coefficient Btu/hr-°F-ft ²	
	<u>Observed</u>	<u>Predicted</u>
Evaporator:		
Feedwater heating zone	~ 1000	430
0-70% steam zone	~ 1800	1100
70-100% steam zone	~ 1000	800
Superheater	300	240
Tempering Na heat exchanger	1450	1600

The values in the table were computed by heat balance using a temperature gradient along the longitudinal axis of the evaporator shell to determine the zones of feedwater heating, 0-70% steam and 70-100% steam. It is recognized that the individual heat transfer coefficients in the three evaporator zones are, at best, rather questionable estimates, since it is difficult to isolate one zone from another. However, since the observed coefficients in each zone are all in excess of predicted design values, it may reasonably be deduced that the combined overall heat transfer in the evaporator is well in excess of the predicted design value.

It will also be seen that the superheater overall heat transfer is calculated to be ~ 12% better than predicted. On the other hand, computations show the overall sodium-to-sodium heat transfer in the tempering heat exchanger to be roughly 10% less than predicted.

Tests have also been conducted on both evaporator and superheater to determine the heat transfer contribution resulting from the vertical sections of the tube bundles. The vertical legs are located in essentially stagnant sodium and it has been assumed that the surface is relatively ineffective as a heat transfer area. Preliminary results of tests made with varying sodium levels in the vertical portions of both evaporator and superheater shells tend to confirm the correctness of this assumption.

Observed operating data taken during the first 1500 hr of steam generator operation do not indicate any appreciable loss in heat transfer due to tube fouling.

Typical Feedwater Treatment

Feedwater treatment for the steam generating unit consists of the following:

- 1) Pre-treatment of makeup water in a two-bed, anion-cation demineralizer.
- 2) Continuous "polishing" treatment in a mixed bed demineralizer designed to pass roughly 10% of the total system circulation rate.
- 3) Continuous oxygen removal through a conventional deaerating feedwater heater.
- 4) Continuous oxygen scavenging by introduction of an aqueous hydrozine (N_2H_4) solution at the boiler feed pump suction.
- 5) pH control through addition of ammonia to the system.

Fluid Pressure Losses

Observations have been made to determine the overall sodium and steam pressure losses through the steam generating unit. Steam pressure drops include the moisture separator as well as interconnecting piping

between evaporator and superheater. Observed and predicted values are tabulated below:

	Flow Rate lb/hr	Fluid Pressure Loss	
		Observed	Predicted
Shell Side (Na flow)	90,500	3.9 ft of Na	4.0 ft of Na
Tube Side (Steam flow)	6,250	~ 17 psi	20 psi

INTERMEDIATE SODIUM HEAT EXCHANGER

Operation of the intermediate sodium heat exchanger has been satisfactory. Computations for 100% load conditions show the overall heat transfer coefficient to be roughly 1460 Btu/hr-°F-ft², as compared to a predicted design value of 1560 Btu/hr-°F-ft². Hence, it appears that the observed value is roughly 7% less favorable than predicted. It should be noted that predicted values were based on use of the Martinelli-Lyon equation for computation of the tube side film coefficient and the modified Seban equation for the shell side film coefficient. The intermediate heat exchanger results, together with the lower than predicted heat transfer in the tempering heat exchanger, supports the view that the Martinelli-Lyon equation for computing heat transfer coefficients may perhaps be somewhat optimistic.

CENTRIFUGAL SODIUM PUMPS

Startup of the two centrifugal sodium pumps was somewhat troublesome due to lubricating oil leakage past the pump shaft seals. The seals installed initially were of the mechanical type with "Stoodite No. 1" rotating faces and graphite stationary faces. A seal oil leakage rate of as much as 220 cm³/day was observed in the primary pump seal after about 350 hr of operation. This was caused by scored seal faces as well as by porosity in the graphite half of the seal. A redesign of the seal, replacing the graphite with a bronze face, has proved to be effective. Oil leakage rates on both secondary and primary pumps are now

~ 1 to 2 cm³/day. Otherwise, operation of both pumps has been quite satisfactory. The "Dynamatic" variable speed control makes possible excellent control of pump capacity and has required no maintenance during the first 3000 hr of operation.

GAS-FIRED SODIUM HEATER

The refractory setting in the sodium heater firebox has required almost continuous maintenance since startup. Serious cracks have appeared in the setting side walls and floor, thus permitting overheating of the thermal insulation. Surface temperatures of up to 600°F have been observed on the outer steel casing. It is anticipated that a new refractory design (or extensive repairs) will be required in the near future.

The capacity of the heater has been adequate and operation of gas burners, controls, and heat transfer equipment has been satisfactory.

SODIUM FLOW CONTROL VALVES

Operation of bellows-sealed, air-operated sodium flow control valves has been very satisfactory. Three 3-in. valves and one 2-in. valve are being tested. All valves have globe type bodies and air-diaphragm operators with remote controllers. No instability has been observed through the full throttling range from closed to 100% open.

SODIUM PURIFICATION COLD TRAP

Operation of the forced-flow, air-cooled, cold trap system has been satisfactory. Initial oxygen removal in the secondary loop required approximately 60 hr to reduce sodium plugging temperatures from 320°F to 215°F in a system containing about 2700 lb of sodium. Cold trap circulation rate was roughly 700 lb/hr. Plugging temperatures of less than 215°F are presently being maintained in both primary and secondary sodium systems.

CHAPTER III

RESEARCH AND DEVELOPMENT FOR THE LAMPRE PROGRAM

CORE DEVELOPMENT PROGRAM

DIRECT CONTACT CORE

General

Parametric studies of large and small direct contact core systems (LAMS-2438, p. 41) indicate that the plutonium concentrations of interest cover a small range. If the fuel contains less than about 2 g Pu/cm³, the calculations indicate a rapid decrease in the breeding ratio. On the other hand, if the concentration of Pu exceeds 5 g/cm³, the hold up of material external to the core results in a very low specific power on the total inventory. Even with optimum hydraulic performance, it is unlikely that a 15 Mw core of this type can achieve more than about 250 w/g based on total inventory. Since the out-of-core inventory may not scale directly with overall size, the in-core inventory performance is probably a more significant number for optimization of a small core. Large cores appear capable of specific powers in the range of 1000 w/g based on total inventory with even less than optimum hydraulic performance.

Direct Contact Studies

The first experiment in pumping and movement of molten fuel by liquid sodium was partially successful. Continuous operation was prevented by obstruction of a syphon tube, but transfer operation, lift pump operation,

and level detection by radioactive floats were successfully demonstrated. The following observations were made:

- 1) The sodium in contact with the fuel does not seem to be alpha-contaminated.
- 2) The fuel surfaces **frozen** under sodium appear mirror-bright. In the catch pot the plutonium meniscus is convex upward, indicating non-wetting, whereas in the upper reservoir it is concave upward.
- 3) The tantalum flare fittings operated perfectly, and opened easily after the test.

Equipment is being set up for jet pump design development using mercury and water. Theoretical analysis of jet pump operation is also under way with the objective of achieving control of particle sizes produced in the transfer of momentum by the jet.

CORE TEST FACILITY

Criteria are being developed for the design of the Core Test Facility. Preliminary calculations have been made on a dual-loop sodium heat dump system which will permit stable operation over a wide range of nuclear powers, using electrical heating as necessary to maintain temperatures.

A layout of the sodium piping and all of the associated equipment is required to permit the specifications of the building and shielding to be established. The rescheduled construction time scale indicates that there will be some delay in completion, largely as a result of uncertainty over the status of the proposal and of a corresponding delay in LAMPRE I. It should be pointed out, however, that the Core Test Facility design and its usefulness does not depend on the LAMPRE I operation. LAMPRE I will serve as an irradiation facility to augment the existing laboratory evaluation techniques in the materials development program.

MOBILE BLANKET STUDIES

Two proposed blanket systems for LAMPRE type reactors are currently being investigated. One uses a settled paste of 10-micron UO_2 particles in sodium, the other uses a molten alloy of uranium-manganese.

There are two possible ways of utilizing the UO_2 paste system for a blanket. In one, the paste moves very slowly through pipes or tubes about 1/2 in. to 1 in. in diameter which are immersed in the main reactor coolant (sodium). Heat generated in the blanket paste is conducted through the tube walls, and paste circulation is needed only at the rate required for gas purging. The other approach is similar to the direct contact core concept; the paste moves downward continuously through a structureless blanket region at a rate sufficient to remove the heat generated. The paste is removed and cooled by sodium jets upon discharge from the bottom of the region, and carried as a dilute slurry to a centrifugal separator above the blanket. It then settles again to a paste and begins another cycle.

A project has just been completed in which the flow properties of settled pastes of UO_2 in water and acetone were measured and correlated with previous work. Although the theory of such non-Newtonian hydraulics is not well developed, it is believed that the type of behavior to be expected with sodium as the liquid phase can be predicted. The measurements made apply to the first paste concept, in which flow through small tubes is used.

The second concept of bulk flow is being studied initially by means of sand and water pastes. The crucial problem in this method is the design of a nozzle which will permit unimpeded paste discharge into a rapidly moving jet of liquid without permitting turbulence to extend back into the settled bed.

Scanning equipment has been set up for measurement of the settling rate of UO_2 particles suspended in sodium. Other equipment has been

operated in which such pastes were moved by gas pressure from one vessel to another, but further work is awaiting the development of better diagnostic instruments. A sintering test at 750°C for 31 days showed no change of UO₂-Na paste contained in stainless steel.

Although still in a preliminary stage, the uranium-manganese eutectic alloy (mp 716°C) has shown some promise. Yttrium is being investigated as a possible container material. A capsule and coupon made from electron-beam melted metal have survived a 1000-hr run with the uranium-manganese alloy at 850°C. Upon being sectioned, the yttrium container and coupon were found to be free from intergranular attack, and were apparently unaffected except for one small pit which may have been a sectioning injury.

FUEL ALLOY DEVELOPMENT PROGRAM

EFFECTS OF ADDITIVES

The first data have been obtained in the systematic program which has been initiated to determine the effects of various additives on the interaction between plutonium (and its alloys) and tantalum. The results outlined below are preliminary and some of the experiments will be repeated when purer materials are available.

Oxygen Addition

The addition of plutonium oxide to pure plutonium, and to plutonium-iron alloy, resulted in no increase of intergranular attack on standard container material (high-purity tantalum). If oxygen participates in the corrosion mechanism, it must somehow be available to the tantalum. Both this test and thermodynamic data indicate that oxygen which has reached the fuel is not available to the tantalum.

Carbon Addition

Carbon additions to fuel have been made in several ways, namely, as PuC, as graphite, and by using cast iron instead of pure iron. The carbon addition reduces destructive mass transfer attack to negligible levels, and it also reduces intergranular corrosion of the tantalum.

Diluents

For the direct contact core concept, the desirable fuel concentration, as seen above, is in the range from 2 to 5 g plutonium per cm^3 . At such dilutions, the use of ternary systems becomes possible, and one such system, of which the phase equilibria are being studied, is plutonium-cerium-cobalt. Some tests had been made earlier in the rocking (TiPu) furnace at a plutonium concentration of 2.7 g/cm^3 in a tantalum container and there was no indication of intergranular attack after 1000 hr at 700°C followed by 330 hr at 800°C . The tantalum in use at that time was a commercial stock with both history and carbon content being unknown. A recent test of the Pu-Ce-Co alloy in high-purity tantalum using 14 g Pu/cm^3 showed some penetration of the container after only 100 hr at 750°C . In a test of cobalt-cerium eutectic alloy, however, there was no attack after 1200 hr at 700°C . Further tests are planned with the dilute alloy, using carbon addition to the fuel and tantalum of known quality.

PHASE STUDIES OF TERNARY SYSTEMS

Previous work in the plutonium-cerium-cobalt system has been concerned with alloys containing high concentrations of plutonium, because it seemed that these would be of special interest for the LAMPRE program. Fairly complete studies have been made of systems containing 70 and 80 a/o, respectively, of plutonium. The bottom of the liquidus valley, representing the alloy of lowest melting point, was found to be 414°C for 80 a/o and 410°C for 70 a/o plutonium. It appears that there is probably a maximum between 70 and 80 a/o, and from this maximum the temperature of the bottom of the valley decreases with increasing plutonium content. For an alloy containing 50 a/o plutonium, the temperature has fallen to 400°C .

In view of the interest in fuels of relatively low plutonium content, e.g., 10 to 30 a/o, for direct contact cores, as stated earlier, attention is now being turned to the study of such ternary systems. One consequence of this change is that there are now other alloys, in addition to the

plutonium-cerium-cobalt system, which may have practical interest. For example, although ternary alloys of plutonium with cerium and copper are known to have high melting points at high plutonium concentrations there is a possibility that alloys having sufficiently low melting points may be found at low plutonium compositions. Other suitable alloys involving more noble metals than those at present under consideration may also be found.

SEGREGATION IN ALLOYS

A test made with a plutonium-cast iron eutectic alloy indicated that liquation occurs upon heating. The bottom portion of the fuel was found to be clean and free of inclusions while the top portions were relatively dirty. The portions in between contained intermediate amounts of inclusions. However, since no clear demarcation line was seen separating the clean and dirty fuel, it was concluded that the liquation had proceeded for either an insufficiently long time, or at an insufficiently high temperature, for maximum effectiveness.

In the course of the preparation of plutonium fuels, it was observed that segregation of impurities generally occurred at the tops of plutonium-iron alloy rods. With rods of plutonium-cerium-cobalt alloys, however, the segregation was less serious; the rods were uniformly cleaner metallographically than were those of the iron alloys.

CONTAINER MATERIALS DEVELOPMENT PROGRAM

FABRICATION

Tantalum-Tungsten Alloys

Alloys of tantalum containing a nominal 0.1, 1.0, 1.5, 3.0, 5.0, and 10.0 w/o tungsten have been made by either electron-beam or inert arc melting methods. These alloys are being studied to determine the effect of tungsten addition on forming procedures and on heat treatment for recrystallization or stress relief. The evaluation procedures include

rolling of sheet to examine response to deformation, cup tests to determine drawability of the alloys, and welding. Results thus far for the lower tungsten additions indicate that the alloys can be formed and joined satisfactorily. Some adjustment in the heat treating temperatures seems required to secure full recrystallization or maximum softness from stress relieving. Cup tests of the tantalum-10 w/o tungsten sheet indicate that this alloy will be difficult to form if it can be formed at all. Perhaps some intermediate tungsten composition of those being examined will prove suitable from the fabrication view point.

Production of Alloys

As a result of the development of a procedure for the recovery of tantalum scrap, in the form of chips and turnings, a possible method for producing alloys of tantalum with other metals has been proposed. The alloying element is introduced into a batch of tantalum chips in as uniform a manner as possible. The mixture is compressed in a die into 6-in. bars, several of which are welded together lengthwise. The resulting ingot is then used as a consumable electrode for arc melting.

TANTALUM METALLURGY RESEARCH

A recently completed study gives experimental values for the diffusion rates of oxygen and nitrogen in tantalum and tantalum-tungsten alloys containing up to 10% tungsten. The results differ only slightly from the literature values for pure tantalum and serve to confirm the calculations used to predict gas pickup, with the actual types of metal in use in the program. Tungsten addition to tantalum did not decrease the rate of oxygen diffusion in these tests. The rate of carbon diffusion is being measured.

Tantalum which is immersed in sodium is readily hydrided if the sodium has access to hydrogen. Several runs have been made to study the effect of oxide and nitride films on tantalum in inhibiting hydrogen pickup. Contrary to the results obtained in gas phase studies, nitride or oxide films give no useful protection against hydriding in sodium.

GETTER ELEMENTS

The addition of thorium, zirconium, and hafnium as getter elements to the tantalum showed no improvement in corrosion resistance if more than 1% was added, although there may be some advantage in the use of lower concentrations. The work is being repeated because there is some doubt concerning the quality of the alloys.

TECHNIQUE DEVELOPMENT

Extensive comparisons have indicated a correlation between bend failure tests of exposed tantalum and metallographic indications of intergranular attack. The bend test has the advantage of averaging over a larger area than it is possible to examine microscopically in a routine manner, and is also much quicker.

The radioautographic technique for detection of plutonium penetration is being improved. The alpha contamination has been observed to spread on the outside of the container in some cases to areas considerably beyond the actual point of penetration. It has been found that this so-called stained area can be mechanically removed, and then a new radioautograph will reveal the points of failure. A technique has been worked out for applying liquid emulsion directly to the capsule, so that the blackened areas can guide the metallographer to make sections at the point of failure.

CORROSION TESTING

Extensive testing has continued of the corrosion of tantalum by plutonium-iron eutectic alloy and by pure plutonium fuels. Capsules of tantalum are used and frequently, in addition, coupons of the metal are placed in the molten alloy. Plutonium-iron fuel alloys made with cast iron are less corrosive than pure plutonium, whereas high-purity iron alloy is most corrosive of all. The greatest attack on the tantalum generally occurs near the top of the molten fuel.

In a comparison between tantalum coupons which has been subject to either (a) normal fabrication annealing followed by acid cleaning or (b)

degassing treatment at temperatures between 2600 and 2700°C, the latter were found to be in significantly better condition after 200 hr heating at 750°C. As a general rule, although not always, severest attack of the annealed coupons occurred near or above the top of the fuel, but the capsules themselves suffered somewhat more advanced attack below the fuel level.

FUEL PROCESSING

Work is in progress for the development of a solvent extraction method for the recovery of plutonium residues from various pyrometallurgical processes. The solvent under study is a mixture of a secondary amine (IA-1) and tetrachloroethylene (TCE). Difficulties have been experienced in connection with the formation of solids and evolution of gas when the solvent, which has been scrubbed with sodium hydroxide, is exposed to nitric acid. Although the reason for this behavior is still unknown, it has been established that it is not caused by either reaction between the amine and nitrous acid nor by hydrolysis of the TCE. Preliminary work with Gulf Solvent BT as a diluent in place of TCE has shown that the former is somewhat better with regard to gas evolution; there is little difference in the formation of solids. Attempts to remove the solid-forming components from the organic solvent by repeated contacting with nitric acid were unsuccessful.