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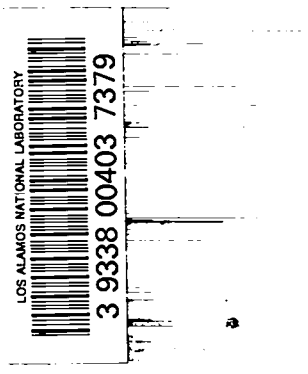
LOS ALAMOS SCIENTIFIC LABORATORY

OF THE UNIVERSITY OF CALIFORNIA o LOS ALAMOS NEW MEXICO

QUARTERLY STATUS REPORT ON PLUTONIUM REACTOR FUEL DEVELOPMENT FOR PERIOD ENDING NOVEMBER 20, 1964



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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

NOTE

In the future, progress in activities relating to the development and testing of plutonium reactor fuels will be included as part of the Advanced Reactor Technology Quarterly Progress Report. The first issue of this new report should appear in February 1965.

LAMPRE

Postmortem Examination

Examination of capsules from LAMPRE Core II is complete. A report is being prepared summarizing observations of effects on fuel pins at burnups up to 0.6 a/o Pu.

Radiochemical analysis of capsule 1536, containing 20 g of irradiated 3 g/cm^3 Pu-Co-Ce alloy in contact with approximately 2 g of Na, is complete. Subsequent to irradiation the capsule had been heated for 8 h at 550°C with vibration to obtain equilibration of fission products between fuel and Na phases. Following the equilibration, a 60-h soak at 550°C allowed the phases to separate; they were then analyzed and the results are shown in Table I.

Table I

Relative Amounts of Fission Products in Fuel and Sodium

Fission Product	Fuel (HCl Soluble) (c/min/g Pu)	Sodium (Butanol Soluble) (c/min/g Na)
Cs	2.1×10^8	5.9×10^9
Ru	3.97×10^9	7.9×10^7
Zr	3.5×10^7	1.85×10^6
Eu	9.6×10^7	7.3×10^8
Sb	1.85×10^8	3.6×10^7
Sr	3.7×10^8	3.2×10^9

FUEL CONTAINMENT STUDIES

Plutonium-Iron Alloys

Three Pu-Fe/Ta-0.1 W+Y corrosion tests at 700°C have attained 4900 h without failure. However, at 650°C, two of the capsules failed at

5600 h with the other satisfactorily completing 8400 h of operation. When compared to the 100% failure of Pu-Fe/Ta-0.1 W capsules at 650° and 700°C in 700 h, it is seen that addition of Y has had beneficial effects on Pu-Fe corrosion of Ta.

To investigate further the effects of Y additions, capsules of Ta-33 ppm Y and Ta-80 ppm Y have been fabricated (cf. LA-3163-MS). These capsules will be tested with both 6.2 g/cm^3 Pu-Co-Ce and Pu-Fe fuels.

The first accelerated corrosion test (ACT) of Pu-Fe containment by pure double-arc melted Ta has been completed. Ten capsules were run at 1100°C and Pu leakage was determined with an α-particle monitoring system. Typical values of activity on the external capsule surface ranged from 700 to 15,000 cpm. Since containment failure is defined as measurable α activity on the capsule surface, it is seen that the failure criterion is somewhat variable. However, all failures occurred in a similar manner. A typical time of 2.5 h elapsed from first indications of Pu activity, as detected by a 400-channel analyzer, to ACT equipment-trip which shut down the test. The data and results are given in Table II

Table II

Tantalum Capsules Tested at 1100°C

Run No.	Time at 1100°C (h)	Location of Major Leak	Approx. Activity on Capsule (cpm)	No. of Melt/ Freezes Prior to End of Test
1	34.5	Gas phase, below weld	15,000	0
2	20.1	Bottom, tip	2,500	4
3	16.4	Gas phase, below weld	750	0
4	16.8	Bottom, tip	10,000	1
5	31.6	Gas phase, below weld	5,000	0
6	27.2	Bottom, tip	2,500	0
7	47.9	Bottom, tip	1,200	0
8	21.7	Gas phase, below weld	1,500	0
9	23.6	Below fuel level,1 in. from tip	2,500	1
10	23.0	Gas phase, below weld	1,500	0

Average time to failure: 23.9 h (apart from Run No. 7, rejected by Chauvenet's criterion).

Standard deviation: $\pm$ 5.9 h.

As seen in Table II, two areas of the capsule were significant in the Pu leakage, namely, the gas phase below the heat affected zone of the weld and the bottom tip area. Although gas phase penetration is observed in capsule tests at temperatures of 750°C and below, bottom tip failures are rare. Metallographic and autoradiographic examination of sectioned capsules indicated spotty Pu penetration in the leak areas. This type of failure indicates Pu penetration along triple points (line where three grains intersect) and not along grain boundaries themselves (surface between two grains). In a few instances, partial penetration was noted in non-leakage areas of the capsules.

Interpolation between the 1100°C data obtained by ACT and that of 700°C Na loop tests predicts a mean failure time of 60 h at 1000°C. To date, one test at 1000°C has been completed and failure obtained at 46 h which is well within the predicted statistical limits.

Plutonium-Cobalt-Cerium Alloys

Thermal cycle test equipment has been fabricated for studying Pu-Co-Ce solidification behavior. NaK volumeter measurements of the density changes of the alloys between solid and liquid states for 6.2 and 8.0 g/cm^3 Pu concentrations were reported previously (LA-3163-MS). Subsequent work shows that for 5.0 g Pu/cm^3 Pu-Co-Ce the densities are 10.05 g/cm^3 (solid) and 10.23 g/cm^3 (liquid), i.e., 1.8% volume expansion on freezing.

It is seen that Pu-Co-Ce alloys all expand on freezing from 1.8% for 5.0 g Pu/cm^3 to 3.0% for 8.0 g Pu/cm^3. This expansion behavior is in contrast to that of Pu-Fe alloys which expand 1.2% for unstabilized Pu-Fe eutectic to a low of 0.43% for Pu-Fe eutectic Ga stabilized. The fact that stabilized Pu-Fe fuels do not distort capsules in a gross manner is undoubtedly due to the low volume expansion on freezing.

In an effort to obtain lower volume expansions with Pu-Co-Ce alloys, a series of experiments has been initiated with Mn additions replacing

equivalent amounts of Ce. Pure Ce expands approximately 1% on freezing, but the addition of 6 w/o Mn causes a 1% contraction on freezing. Differential thermal analysis data indicate as much as one half of the Ce of any particular ternary Pu-Co-Ce alloy may be replaced by Mn and yet maintain the liquidus temperature below 450°C. Based on this fact, measurements will be made of the volume expansions of these alloys on freezing.

Although Na loop testing of Pu-Co-Ce fuels has been hampered by gross distortion of the test capsules due to thermal cycling during the inspection periods, other tests in short capsules are continuing. Table III shows the results of a small set of capsules tested with 6.5 g Pu/cm^3 and 5.0 g Pu/cm^3, with and without carbide surface layers. TaC surface layers were put on the basic container by pack carburizing in vacuum at 1800°C. The Ta-10 W capsules have not been penetrated by 5 g Pu/cm^3 fuel at temperatures of 750° and 850°C up to 3000 h. These tests are being continued.

Table III

Corrosion Tests on Carburized and Uncarburized Capsules

Capsule Material	No. of Capsules	Surface	Pu Content (g/cm^3)	Temp. (°C)	Time (h)	Remarks
Ta	1	Uncarburized	6.5	750	2000	No leaks
Ta	1	Carburized	6.5	750	2000	No leaks
Ta-10 W	2	Carburized	5.0	750	3000	No leaks
Ta	1	Uncarburized	6.5	850	1000	Leaked
Ta	1	Carburized	6.5	850	2000	No leaks
Ta-10 W	2	Carburized	5.0	850	3000	No leaks

In addition to the foregoing tests, one 8 g Pu/cm^3 Pu-Co-Ce/pure Ta capsule was tested in the ACT apparatus for 660 h at 1000°C without failure. The run was terminated after the fourth inadvertent shutdown because of a spurious electronic signal. A total of 5% radial expansion was obtained in the capsule below the fuel level due to these four melt-freeze cycles. It appears from this single observation that it will be

necessary to test Pu-Co-Ce fuels at least at 1100°C to observe failures in reasonable periods of time.

In order to obtain fundamental data necessary for mass-transfer analysis in flowing fuel systems, a Ta diffusion experiment has been operated for 2807 h with 5 g Pu/cm^3 Pu-Co-Ce fuel. A temperature gradient is imposed on the capsule so that the bottom is 700°C while the top is 608°C. Such a gradient transports Ta from a radioactive tab in the bottom by both diffusion and convection processes. Gamma scans indicate active Ta^{182} is transporting from the tab up the 4-in. fuel column to the gas fuel interface where it is depositing on the wall as Ta-Co crystals. The maximum rate of transport observed has been 9×10^{-3} mil Ta/yr. In the last 500 h, the rate has slowed to approximately 2×10^{-3} mil Ta/yr.

The stability of Pu-Co-Ce fuel in contact with Na coolant is being studied by means of an open-ended capsule in a circulating Na loop. The temperatures of the system are maintained at 650°C at the capsule and 500°C in the cold leg with a Na flow of 0.48 gpm. To date, a 5 g Pu/cm^3 Pu-Co-Ce fuel has operated for 250 h in contact with Na. Radiographic inspection of the capsule shows a well defined fuel-Na interface that has not changed shape or position with time. No precipitated material has been detected in the system by the plugging indicator, and no α or γ activity has been found on the surfaces of Na samples removed for chemical analysis.

Blanket Systems

Further work on molten salts for blanket systems has been deferred because equipment has been diverted to the Pu-Co-Ce-Mn alloy investigations mentioned earlier.

FUEL CONTAINER FABRICATION

Material Procurement

Special Ta-W-Y alloys for container fabrication, melted at Wah Chang Corp., Albany, Oregon, were scheduled for delivery to LASL the latter part

of November, but when the cast billets were machined they proved to be quite porous and will have to be remelted. Delivery of these metals is now expected sometime in February 1965.

One of the Ta-W-Y alloys prepared at National Research Corp. showed 1000 ppm Mo on analysis. Since the specification calls for 20 ppm Mo, a re-analysis of the alloy will be made. If the specification is not met, the material will be rejected and another ingot produced.

Joining Development

Eighty tubes were processed from three Ta-0.1 W+Y billets, namely, K-687 - 18, K-697 - 32, K-698 - 30. The billets were rolled to 25-mil sheet from which blanks 1.15 in. by 5 in. were cut. The blanks were press-formed to tubular shape and electron-beam welded along the longitudinal butt joint. Welded tubes were trimmed to 4-in. lengths. Visual inspection indicated the welds to be quite good; no x-ray inspection was performed.

Commercial strip from NRC, 20 mils thick, was formed into tube in the same manner. Two billets, K-695 - 10 (Ta + 84 ppm Y) and K-692 - 22 (Ta + 33 ppm Y), provided the strip. Each tube was fitted with two end caps made from the same billet material as the tube. Tubes and caps are being evaluated.

Fabrication Development

For future reactors it is intended that Pu fuel containers be 18 to 24 in. long by 0.4 in. o.d. with a 25-mil wall. It is desirable to use seamless tube, one end of which is closed as in the LAMPRE capsules. In initial experimental work, bench drawing over a mandrel and tube sinking have been examined as possible forming methods. Third-stage LAMPRE capsules, 1/2 in. o.d. with a 35-mil wall, were used as starting tubes. Slip fit mandrels of solid Cu, mild steel tubing, and Al tubing were placed in capsules and the composite bench drawn. The diameter reduction achieved during drawing varied, but all capsules eventually failed by tension in the wall. Sinking was more successful with the capsule being reduced from

1/2 in. o.d. to 3/8 in. without failure. The sunk capsule showed 5 to 8 mil wall thickness increase. The Al mandrel was dissolved from the drawn capsule and the wall thickness measured; little change from the original 35 mils was noted. Although sinking was successful (no failure in tension) and the method permits control of o.d., control of wall thickness and wall thickness uniformity might be difficult to achieve.

A second procedure will be tried in which coextrusion is employed. A first-stage LAMPRE capsule will be filled with mild steel, cased in steel, and extruded at 950°C. A second test will employ Al as the filler with extrusion at room temperature.

Diffusion Bonding

To secure intimate contact between the fuel tube and the closure cap, prior to a thermal diffusion treatment, magnetic swaging was tried. The process appears unsatisfactory as the swaged tube tended to spring back from the cap. As a result, poor contact existed at the interface and bonding was intermittent.

Various materials used at the tube-cap interface to promote bonding were partially successful. Materials tried included Ta powder-varcum-lampblack; Ta powder, V foil; Y foil; and 50 Pt-50 Pd foil. Powders or powder mixes seem to yield intermittent bonding with no assurance of a leak-tight joint. Foils proved better and under suitable conditions may give a leak-tight, well-bonded joint. Of the foils used, V gave the best joint; Y and Pt-Pd do not appear to be satisfactory. Any material selected to promote diffusion bonding of Ta should be metallurgically compatible to prevent formation of brittle components in the joint interface. Vanadium seems to be satisfactory, but other metals may be equally suitable. Further work in this direction seems warranted.

OMEGA WEST REACTOR EXPERIMENT (OWREX)

Design and Construction of Facility

Design is complete and construction is well advanced on the facility for testing reactor fuels in the core of the Omega West Reactor. Normally, the fuel will be a molten alloy of Pu encapsulated in Ta, but other fuel and cladding combinations could be tested, if desired.

A cylindrical fuel test specimen, not exceeding 0.430 in. in diameter by 8 in. in length, may generate up to 5 kW of fission heat which can be removed by Na circulating in natural convection. The temperature at a selected reference point can be maintained, using thermostatically-controlled electrical heaters, at any desired value in the range of about 300° to 800°C. Temperature control may also be maintained, to prevent freezing the fuel when the OWR is shut down.

Neutronic calculations indicate that the average power density in the fuel may be as high as 1000 W/cm^3. It also appears that the central power density, governed by neutron flux penetration, will be reasonable, especially for the Pu-Co-Ce fuels of lower density.

The design of the facility incorporates minor modifications suggested by experience with the successful operation of an early mockup. Major features of the mockup, which operated for over 900 h, are retained. The convecting Na is doubly contained by stainless steel. The molten fuel is doubly contained by Ta, with the space between monitored by a fission-gas activity sensor to indicate failure of the primary fuel containment. On completion of a test, the fuel specimen is to be separately withdrawn into a gas-tight section of tubing within a γ-ray shield of depleted U. This shielded specimen is taken to hot cells for examination, and a new specimen is inserted into the facility.

Numerous auxiliary systems, shields, etc., are required to accomplish the Na charging, out-of-pile checkout of the facility, in-pile control of the experiment, and removal of irradiated specimens. Such systems have been designed and are either completed or nearly so.

CODES

Shielding

A general Monte Carlo code (MCG) for γ rays, developed by T Division at LASL, is more flexible than the United Nuclear Monte Carlo codes, ADONIS and SAGE, in that it can handle an arbitrary three-dimensional configuration of first and second degree surfaces; ADONIS can treat only systems composed of rectangular parallelepipeds and SAGE treats problems in spherical geometry. Options provided by the MCG code allow processing of the source histories to obtain information on photon crossings at surfaces, energy deposition in regions, and other special tallies of interest in a specific problem.

In order to test the operation of the code, a 7-region spherical model of LAMPRE I was chosen. A 2-MeV γ-ray source, spatially distributed according to fission density, was specified in the core region. MCG and SAGE calculations were compared. Results obtained for absorptions in selected regions and the percent escape from the system are shown below:

	MCG	SAGE
Absorptions in core (%)	74.56	73.67
Absorptions in shim (%)	12.34	12.43
Absorptions in borated graphite (%)	0.269	0.276
Escape from system (%)	0.0796	0.0621

In view of the good agreement obtained in the test problems, it is concluded that the MCG code is operating satisfactorily.

Statistical Analysis of Experiments

A FORTRAN Capsule Statistics code (CPS) has been written for evaluating the statistics of capsule tests. In the code, the failure rates in two sets of capsules are compared, and the probability (or confidence) that the change in failure rates observed is a real and not chance effect is computed. As many such comparisons as is desired may be

computed in sequence; hence, the code will be useful for evaluating previous capsule data and may be used in planning future experiments for constructing tables covering all possible results.

Data Preparation Code

The DPD code is being used to convert basic input specifications for a reactor into a form suitable for input to the 2-D neutronic codes, DDK and CRAM. DPD incorporates the Hassitt-Mitchel input routine (DECIN), which uses a very flexible format. Considerable further flexibility in input has been included to permit simplified and adaptable input specifications. Provision has been included for sequence problems, arbitrary input order, and substantial internal data checks and diagnostics.

Thermal Calculations for a Pin Lattice

Codes have been completed for calculating the temperature distributions in triangular lattices of clad fuel pins. The coolant may be in laminar or slug flow, and the fuel may be static or in laminar convection. In the solid-fuel, slug-flow coolant case, the difference between the peak interface temperature in a given plane and the average coolant temperature in the same plane is obtained directly. It is hoped that these codes can be combined with present thermal analyses to obtain a greatly improved estimate of thermal performance in the Fast Reactor Core Test Facility.

Gamma-Ray Spectrum Analysis

A code has been developed for fitting a complex γ-ray spectrum by a sum of lines, each contributing a Gaussian full-energy peak and an exponential low-energy tail. A least-squares technique is used to obtain the best values of the parameters. The code was applied to estimate the activity of Co^{60} relative to that of Ta^{182} in a Ta capsule, irradiated in LAMPRE, which contained Pu-Co-Ce fuel. This was done by analysis of the many lines, unresolved by a 2.3-in. diam by 6-in. long NaI crystal, of Ta^{182} and Co^{60} in the neighborhood of 1.2 MeV.

FAST REACTOR CORE TEST FACILITY (FRCTF)

Nuclear Design Calculations

Several nuclear design calculations for a proposed FRCTF first core have been run to provide a rational basis for certain general design features. The results of these generalizations indicate that:

1. A substantial decrease from previously estimated FRCTF specific powers is associated with the increase in pitch/diameter rates from 1.1 to the more conservative 1.2 or 1.25.

2. Specific power cannot be significantly increased without increasing one or more factors beyond presently assumed limits. Among such factors are the following: (a) Increased volume fraction of fuel, presently limited to about 0.4 to 0.5; further increase would raise serious questions as to the feasibility of heat removal. (b) Increased fuel density; increases in this raise questions as to material capabilities. (c) Increased core power level. (d) Increased Na flow capacity.

Further calculations have been run to compare 19-pin and 7-pin module designs and to estimate control capability. It was found from 1-D survey calculations that, if a reflector control element has a certain worth (difference in reactivity between Ni and Na in the control region), then a Ta follower (instead of Na) doubles the worth of the control rod, and B_4C triples it (approximately). One 2-D design calculation indicates that an external 4-in. thick annular reflector of Ni followed by Ta is worth approximately 3.3% Δk. It is estimated that additional reactivity reduction should be available for shutdown.

Doubling Time of Breeders Using FRCTF-Type Cores

Although early FRCTF cores will not necessarily be blanketed, it may be advantageous to orient their design toward systems which appear promising for fast breeder applications. The minimum doubling time may be used to indicate some of the design factors that are important and to

establish ranges of interest for certain parameters--in particular, the Pu density in the fuel.

Calculations for a series of FRCTF-type cores yield an empirical relation between breeding ratio and the atom ratio of Ta/Pu, which is represented by R. In addition, specific power is expressible in terms of core-average heat flux HF, Ta wall thickness b, and the Ta to Pu atom ratio R. In terms of these, the usual doubling time (DT) relation is transformed into

$$DT(yr) \approx \frac{66{,}000\ b\ (cm)}{HF\ (W/cm^2)} \cdot \frac{1}{R(0.9-0.3\ R)} \cdot$$

Or, in terms of core-average thermal stress S_{Ta} in the wall,

$$DT(yr) \approx \frac{1.6 \times 10^7 b^2}{S_{Ta}(psi)} \cdot \frac{1}{R(0.9-0.3\ R)} \cdot$$

The relations emphasize the well-known facts that the DT is reduced by: (a) thin Ta wall, (b) high heat flux, (c) high wall stress. They demonstrate the less obvious fact (d) that there exists, for a given HF or S_{Ta}, an optimum value of R (in this instance, about 1.5). This latter fact may be used to designate a fuel of optimum Pu density corresponding to any given Ta capsule size. For example, for Ta capsules of 0.40-in. o.d. by 0.025-in. wall (about like LAMPRE) a fuel of about 4.5 g Pu/cm^3 is indicated. If a limit of 5000 psi on S_{Ta} (core average) is allowed with an 0.025-in. wall, the relations give a DT of 21 years, SP of 440 W/g Pu, and HF of 320 W/cm^2 (1 x 10^6 Btu/h-ft^2).

The foregoing analysis may be extended to capsules made of other materials. For a material whose effective neutron absorption cross section is x times that of Ta, the following comparisons may be made with the case for Ta walls, assuming $x < 1$, which is the usual case and the attractive one: (a) the optimum fuel contains less Pu (by factor x), (b) doubling time is shorter (by factor x). For example, x for Nb is

about 0.27. If Nb is used instead of Ta in the case given above, the fuel density decreases from 4.5 to 1.2 g Pu/cm^3, and the doubling time from 21 to 6 years.

FRCTF CONSTRUCTION AND EQUIPMENT

Phase A

Shop Drawing and Equipment Submittals

It is estimated that the Contractor's shop drawing and equipment submittals are 90-95% complete. Submittals have been received on all major components.

Nondestructive Testing

Approximately 1800 radiographs of piping welds have been reviewed since the beginning of the Phase A contract. It is estimated that radiography is about two-thirds complete. The problems of obtaining proper radiographic technique, darkroom procedure, and film evaluation still exist.

Construction Modifications

Modifications Nos. 11 through 16 have been signed. The total cost of these modifications is $54,000.

The four 16-in. and 18-in. contractor-furnished and installed sleeves for nonradioactive Na piping between the radioactive and nonradioactive Na rooms and the four companion sleeves through the south wall of the nonradioactive Na room were changed to 20-in. government-furnished, contractor-installed sleeves, at no cost to the government.

The north 17 in the east row of cable sleeves in the control room subfloor were increased from 3 in. to 4 in., at no cost to the government, to facilitate interconnection of the data system without removal of cable connectors. A request was made for the addition of permanent sheet metal guards over mineral-insulated cable penetrations to protect the cable from possible breakage at the point of penetration through liner plate.

Construction Progress

All walls have been poured to the 97 to 100 ft level. Roof slabs over the electrical equipment and radioactive Na rooms have also been poured. Piping is being installed in the mechanical equipment rooms for ventilation systems No. 1 and 2, and part of the blowers for these systems have been set on location. Construction was scheduled to be 68% complete on November 14, 1964, and was estimated to be only 47% complete. The contractor's current PERT printout indicates that completion of construction will be 7 weeks later than scheduled.

Phase B

Intermediate Heat Exchangers

The 5-MW IHX tube bundle installation has been completed and the shell is being installed; shipment is scheduled for mid-December. The 15-MW IHX tube bundle installation is approximately 75% complete and shipment is scheduled for early in 1965. The vendor is experiencing considerable difficulty in producing porosity-free tube-to-tube sheet welds. About 35% of the welds have required repair to remove porosity which exceeds LASL's specification of 0.035 in. maximum dimension as measured on radiographic film. Consequently, the estimated production time has been greatly extended.

Recommendations have been solicited from the vendor with regard to LASL's proposal for increasing shell side Na flow rates by 25% above the original design specifications. It appears that the originally specified flow rates result in baffle velocities that are already about 40% greater than would normally be recommended. Some concern is therefore being expressed as to whether the present design is on the conservative side. The main problem is the possibility of tube vibration which could accompany excessive baffle velocities. Hence, the vendor has recommended that shell flow rates not be increased above the original design values unless pre-operational tests are conducted to demonstrate that vibration will not occur.

Sodium Piping Systems

Because present fuel-container requirements limit core inlet temperature and maximum Ta-fuel interface temperature, methods of increasing flow through the core are being examined. It appears that by using the same major components, i.e., pumps, intermediate heat exchangers (IHXs), and heat dumps, but paralleling the IHX primary piping, a core flow of 2500 gpm can be achieved. If the 6-in. pipe is replaced by 10-in. pipe, the available vessel pressure drop at 2500 gpm is about that for 1500 gpm in 6-in. pipe. The 15-MW IHX would take about 1865 gpm and the 5-MW IHX about 635 gpm. With such an arrangement core power will be increased one third and core ΔT was $270^{\circ}F$.

Paralleling the primary sides of the IHXs is, in one respect, a relaxation of an old requirement--that of having core and blanket loops separate. If a breeding blanket is installed in the future, core coolant flow would have to be reduced in order to have more flow available for blanket heat removal.

Revised piping layouts based on paralleled IHX primaries and 10-in. pipe to the vessel are being studied. Pipe stress calculations for the first layout indicated that stresses would be unacceptably high. Work is being continued to find a layout having acceptable stresses and pipe lengths.

Data System

The major tests on the computer main frame, magnetic drum memory, magnetic tape system, analog to digital converters, etc., appear to be satisfactory. The vendor is making an effort to ship the Data System to LASL by the end of November 1964. A partial set of racks for the CTF operator's console has been set up in the system's temporary location. Several minor problems still need attention, but it should be possible to install the Data System as soon as it arrives at LASL.

LIQUID METALS TECHNOLOGY

Hot Trap Evaluation Loop

Analysis of the kinetics of hot-trapping Na with Zr (or other comparable systems) can be approached on theoretical grounds, but the present state of O analysis technology does not permit the performance of meaningful physical measurements on operating systems. The following approach to understanding hot-trap kinetics is therefore being tried.

Since a concentration profile along the length of the hot trap should exist, and if the weight change of materials along this length is a function of concentration, then the weight change, which in the final analysis is the parameter being sought, is a function of concentration. This line of reasoning led to an experiment in which samples of Zr, 50 Zr-50 Ti, Ta, Ta-10 W, Nb, and U were exposed for 2000 h at 600°C in positions along the flow path in a Zr hot trap. Sodium flow rate was nominally 1 gpm in a cross-sectional area of 2.4 $in.^2$. The area of Zr surface was 1645 $in.^2$. The total weight gain of the hot trap was 1.29 g. The results showed that the weight change of all materials except U decreases along the Na flow path. Furthermore, the hot-trapping efficiency of a particular material is directly related to its weight change; of the materials studied, 50 Zr-50 Ti was found to be the most efficient for a hot trap.

PLUTONIUM AND ALLOY STUDIES

Self-Diffusion in Plutonium

A preliminary experiment was made to determine the order of magnitude of self-diffusion in γ-Pu. Two diffusion couples containing 0.008% Pu^{238} on one side and 0.5% Pu^{238} on the other were annealed for 1030 h, one couple at 218.8° and the other at 299.1°C. The distribution of Pu^{238} in the annealed couples was then determined by radiochemical analysis of 0.001-in. thick sections obtained by turning consecutive layers from the samples. Although the observed penetration was quite

small at the lower temperature, a diffusion coefficient, D, for each temperature was computed:

$$D_{218.8} = 1.13 \times 10^{-12} \text{ cm}^2/\text{sec}$$

$$D_{299.1} = 9.29 \times 10^{-12} \text{ cm}^2/\text{sec}.$$

A tentative calculation of the temperature dependence of the diffusion coefficient is

$$D = 3.5 \times 10^{-6} \exp\left(\frac{-14,600}{RT}\right) \text{ cm}^2/\text{sec}.$$

The self-diffusion coefficient for ϵ (bcc) was approximated crudely by extrapolating the self-diffusion data for δ-Pu (fcc), assuming that both phases have the same activation energy for diffusion and that D is increased about two orders of magnitude as the result of the change from fcc to bcc; this is the magnitude of the change in D for Fe and Th for such a change in structure. The value thus obtained is 1.5×10^{-7} cm^2/sec for ϵ-Pu at 530°C. The complete homogenization of a 0.060-in. thick by 3/8 in. diam diffusion couple during 8 h at 530°C confirmed that self-diffusion in ϵ-Pu is at least as fast as estimated. Further experiments involving shorter annealing times of 5 to 30 min are being planned to determine the self-diffusion coefficient for this phase more accurately.

Plutonium-Americium Alloys

The existence of a continuous series of solid solutions between δ-Pu and β-Am, as inferred from the constitution of Pu-rich alloys, was verified experimentally with alloys containing from 59 to 95 a/o Am. Results of equilibration heat treatments of appropriate alloys indicate that δ-Pu (β-Am) solid solution between the composition limits 5 to 85 a/o Am is stable at room temperature.

Plutonium-Scandium Alloys

The homogeneity range of the intermediate phase in the Pu-Sc system (IA(MS)-3096) was tentatively estimated from microstructural and x-ray data to be between 37 and 40 a/o Sc. Differential thermal analysis results indicate that the phase is formed peritectically between ϵ-Pu solid solution (~ 25 a/o Sc) and α-Sc solid solution (~ 53 a/o Sc) at 753°C.

Compositional Control of LAMPRE Fuel

Continuation of the study of compositional control of Pu-Ce-Co alloy fuel (LA-3163-MS) has resulted in development of a simple mathematical expression, based on an idealized model, which relates sample variance, sample size, concentration of a second phase, and particle size of a second phase. Rearrangement and extension to include concentration and density factors has led to the following tentative expression pertaining to a two-phase or two-microconstituent system:

$$V_s = \frac{Vf_\alpha(1 - Vf_\alpha)\, P_{v\alpha}}{\sigma^2_{vf_\alpha}} \, (C_{A\alpha} - C_{A\beta}) \left(\frac{\rho_\beta}{\rho_\alpha}\right) ,$$

where V_s = sample volume; Vf_α = volume fraction of a phase alpha; $P_{v\alpha}$ = volume of one particle of phase alpha; σ = desired standard deviation in Vf_α (σ^2 = variance); $C_{A\alpha}$ and $C_{A\beta}$ = fractional concentration of component A in phase alpha and phase beta, respectively; and ρ_α and ρ_β = density of phase alpha and phase beta, respectively.