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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 FEBRUARY 20, 1961

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

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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 AND PRODUCTION

Development

Ten reactor fuel capsules which had been held in flowing sodium at 525°C in certification tests have been examined again after another exposure period. None showed leaks after a total exposure of 2520 hr. Two of these will be dissected and the remaining eight exposed further. Another lot of reactor capsules has been tested at 625°C to determine the effect of various annealing temperatures. Under the conditions of this test, capsules annealed at 1050 and 1200°C leaked, but not those annealed at 1350°C . After 12 melt-freeze cycles, it has been concluded that silicon added to the fuel is effective in preventing capsule freeze-up damage.

Production

A total of 159 fuel elements were accepted for the initial core loading; an additional 31 processed capsules are on hand but not loaded. The fuel composition varies as follows:

15 elements	LCX III fuel (cast iron), stabilizer and carbon disc added.
27 elements	Plutonium plus iron plus stabilizer, PuC added at assembly.
117 elements	LCX III fuel (cast iron), stabilizer and PuC added at assembly.

The loading history of this run of fuel elements is good. One capsule was lost in welding, one was rejected for excessive melting in the capsule wall, and two failed the helium leak check.

An additional 115 capsules are being fabricated from the same billet as spares for LAMPRE I. Fifteen of these are being pickled between each step of the fabrication process to remove any metallic pickup which may remain on the capsule from the draw tools. However, there is some evidence of hydrogen embrittlement from the pickling treatments and procedures for removing hydrogen may have to be initiated.

REACTOR SYSTEM

Sodium System

During this period the sodium system was operated in the temperature range 150 to 450°C with only minor difficulties in instrumentation. With the system at low temperature, hot trap No. 1 was frozen and removed. An examination showed that only the leading edges of one section of zirconium fuel had any appreciable pick up of oxygen. It is believed that the two remaining hot traps are in good shape and will be adequate for several years for normal operations.

Cover Gas System

Hydrogen in the cover gas has been monitored regularly, and daily flushes of the cover gas have been resumed. The hydrogen concentration has stabilized at 45 ± 20 ppm. Insertion of zirconium foil and massive zirconium slugs into the core have failed to change cover gas hydrogen content. A 3-day run with zirconium-filled hot traps at 450°C also did not affect hydrogen content. Before going to hot (475°C) critical, six zirconium-filled capsules will be put into reflector pin locations at the edge of the core. One fuel-filled capsule was put in the core and held at 450°C for one week then removed and the fuel analyzed for hydrogen. The analysis showed that hydrogen impurity was not appreciably above that in the original fuel. Fuel from capsules removed during hot critical experiments will be analyzed for hydrogen.

Capsule Charger

Electrical breakdown at several points within the capsule charger has resulted in wiring modification of the circuit in order to reduce maximum voltages to ground and in the installation of sodium splash shields at critical locations. A total of five dummy capsules could not be unscrewed by the charger mechanism and the elements had to be removed from the charger. It was found that the threads had galled and that the galling occurred mostly on the dummy capsule thread. Consequently, capsules or reflector pins now being loaded are being reworked to remove all burrs or sharp edges.

Shielding

Installation of the lead walls in the cell and sodium equipment room as well as the dump tank shield has been completed. The cell doorway has been filled with concrete blocks and polyethylene in order that neutron shielding at this point is similar to the adjacent wall. Access to the cell is now through the ceiling lift hole.

Requirements for the remainder of the shielding will be determined by a low-power run for sufficient time to obtain saturated sodium activity. Areas involved in this low-power survey are block storage, fuel transfer area, boiler room, and cell corridor.

FUEL STORAGE FACILITY

The storage facility has been turned over for use although some work remains to be done. This includes installing the electric heater, paving the excavation around the outside of the facility, final connection to the telephone cabinet in the basement, check-out of the intrusion alarm system, check-out of phonet alarm system, and installing a lock cylinder in the facility entrance door.

Examination of the storage sleeves showed that a number contained varying amounts of water which apparently leaked in during construction in snowy weather. The sleeves are being dried and the rust removed.

STARTUP

The loading of the LAMPRE I core commenced on January 20, 1961 with the sodium temperature set at 160°C . The operations at this temperature are primarily aimed toward determining the shim-down cold critical mass. This mass will set the upper limit on loadings which may be achieved during power operations since the configuration must always be such that the reactor will remain shut down in the event of an inadvertent freeze-up of the core. Thus, the amount of excess reactivity loading for losses resulting from power operations, and which are recoverable upon shutdown, will be established by the measurement of the shim-down cold critical mass.

At present, 136 capsules (23.166 kg Pu) are installed in the core. The reactor was first brought to critical on February 17, 1961 with 135 capsules (22.996 kg Pu) installed, with the shim and two rods fully up, and with the No. 3 rod set at 13 in. of insertion. Full shim and rod insertion corresponds to 16 in. of travel, but the shim stop is set at an indicated insertion of 12.67 in. to increase the initial rate of reactivity depression during a scram. The fourth control rod position is empty since it will be about 2 weeks before the oscillator rod will be ready for installation. Figure 1 shows the curve of the extrapolated critical mass vs the position of the No. 3 control rod taken during the approach to critical.

Preliminary results on the worth of the sodium have been obtained by comparing current measurements with LCX III, as follows:

	<u>Pu, kg</u>
LCX III shim-up, rods up critical mass	25.35
Correction for temperature difference ($80-160^{\circ}\text{C}$).	+0.15
Correction for oscillator rod	<u>+0.20</u>
Corrected LCX III result	25.70

Comparing this to the value 22.98 kg Pu obtained currently for the shim-up, rods-up critical mass, a net worth of 2.7 kg is found for the sodium. This agrees well with the calculated value of 3.1 kg.

During the loading of the first 91 capsules, shim-down neutron count rates only were observed. These data are illustrated in Fig. 2 in which reciprocal multiplication is plotted against the square root of the mass in kg.

Fig. 1. Extrapolated critical mass vs position of No. 3 control rod

Fig. 2. Reciprocal multiplication vs square root of plutonium mass

CHAPTER II

SODIUM TEST FACILITY

OPERATING EXPERIENCE

Summary

Operation of the Sodium Test Facility has been continuous this quarter except for a total of six maintenance and inspection shutdowns resulting in 680 idle hours. Other than the replacement of a flange gasket in the steam superheater outlet header, all of the down time has been required for maintenance of steam plant auxiliary equipment and the gas-fired sodium heater. Essentially no down time has been required for repair of sodium equipment being tested. As of February 20, the following total running hours have been logged on major items of test equipment:

Item	Total Operating Time (hr)
Steam Generator	5024
Intermediate Sodium Heat Exchanger	7039
Primary Loop Sodium Pump	7039
Secondary Loop Sodium Pump	8188
Gas-Fired Sodium Heater	6131
Sodium Cold Trap System	8276
Air-Operated Sodium Flow Control Valves:	
Primary Loop (two valves)	7039
Secondary Loop (two valves)	8188

Intermediate Sodium Heat Exchanger

Operation of the intermediate sodium heat exchanger has continued to be satisfactory. Tests have been conducted to determine overall heat transfer performance at loads varying from 1000 kw (50% of design load) to 2000 kw (100% of design load). A typical set of heat transfer test results are shown on Fig. 3. The curves show (1) average heat flux, (2) average overall heat transfer coefficient, (3) log mean temperature difference, and (4) sodium mass flow rate, all plotted against percent of rated design load. The results indicate an average overall heat transfer coefficient of roughly 1500 Btu/hr,ft², °F at 100% of design load. Sodium flow condition for the 100% load represents a tube sodium velocity of 9.1 ft/sec in the 3/4 in. o.d. tubes and roughly 1.5 ft/sec in the 8-in. i.d. shell.

Steam Generating Unit

Operation of the steam generating unit has continued to be satisfactory. Heat transfer tests similar to those described for the intermediate sodium heater have been conducted on all four components of the steam generator, namely, the tempering sodium heat exchange section, superheating section, evaporating section, and feedwater heating section. All tests were conducted under design operating conditions as follows:

Superheated Steam Outlet Pressure	890 psig
Superheated Steam Outlet Temperature	925°F
Feedwater Inlet Temperature	400°F

Curves showing a typical set of test results are presented in Figs. 4 through 7. Average overall heat transfer coefficients at 100% of design load were found to be roughly as follows:

	<u>U</u> <u>Btu/hr,ft², °F</u>
Tempering Sodium Heat Exchanger (Fig. 4)	1200
Steam Superheating Section (Fig. 5)	280
Steam Evaporating Section (Fig. 6)	1600
Feedwater Heating Section (Fig. 7)	450

Fig. 3. Heat transfer test results for intermediate sodium heat exchanger.

Fig. 4. Heat transfer test results for tempering sodium heat exchanger.

Fig. 5. Heat transfer test results for steam generator-cylindrical section

Fig. 6. Heat transfer test results for steam generator-evaporator section.

Fig. 7. Heat transfer test results for steam generator-feedwater heating section.

It should be pointed out that results for the tempering sodium heat exchanger are plotted against percent of the steam generator design load rather than percent of design rating for the tempering sodium heater. Actually, the total heat transfer and tube-side mass sodium flow in the tempering heater remained essentially constant for all test conditions while the shell-side mass sodium flow was varied linearly with the steam generator output.

Attention should also be called to the test points plotted for 100% load conditions in the evaporating section and feedwater heating section of the steam generator (see Figs. 6 and 7). It will be noted that test results indicate a sudden drop in evaporating section overall heat transfer coefficient from 1620 to 1010 Btu/hr,ft²,°F between 90% and 100% loads. Further, this should be compared with the feedwater heating section which shows a rapid increase in overall heat transfer coefficient from 425 to 610 Btu/hr,ft²,°F between the 90% and 100% load conditions. It is believed that the 100% load test points are questionable, because of errors in determining the actual distribution of heat transfer surface area between evaporating and feedwater heating sections. In the steam generator being tested, the feedwater heating section and evaporating section are not separate and distinct pieces of equipment, but rather are a single heat exchanger with a continuous tube bundle for both feedwater heating and evaporation. At best, it is very difficult to determine a precise point in the steam generator tube bundle at which the feedwater temperature has been raised to the saturation point and evaporation begins.

Instrumentation for determining the relative "feedwater heating" and "evaporating" areas consists of thermocouples welded at intervals along the entire length of the heat exchanger shell. It is then assumed that the temperature gradient along the length of the shell approximates the mean longitudinal temperature gradient of the sodium passing from inlet to outlet. From additional instrumentation, shell sodium flow, steam flow, sodium inlet and outlet temperatures, and feedwater inlet temperature can be determined. Using these data, a heat balance computation can be made which

results in a theoretical sodium temperature drop through the feedwater heating section. This temperature difference is then matched against the shell metal temperature gradient to determine the point in the heat exchanger where feedwater heating should be complete and evaporation begin. Although admittedly a rather crude test procedure, the results produced have generally appeared to be reasonable. It is believed, however, that in the case of the 100% points shown on Figs. 6 and 7, the computations have given an inaccurate estimate of the relative heat transfer areas, thus resulting in a value for evaporating overall heat transfer coefficient that is too low with a correspondingly high feedwater heating overall heat transfer coefficient. For this reason, the two heat transfer coefficient curves have been extrapolated from 90% to 100% loads.

Observed test results taken to date have failed to indicate conclusively any appreciable reduction in overall heat transfer coefficients due to tube fouling.

Centrifugal Sodium Pumps

The two centrifugal sodium pumps have continued to perform satisfactorily without requiring maintenance. The secondary loop pump has not been shut down since initial start up and has now operated 8188 continuous hours. Lubricating oil leakage past the shaft seals in both pumps has remained constant at about 1 to 5 cm³/day.

The Byron Jackson Pump Co. has recently developed a gas-lubricated shaft seal assembly which is adaptable to the sodium pumps now being tested in the Sodium Test Facility. The new seal assembly eliminates the requirement for lubricating oil and hence is attractive for use in nuclear reactor sodium pumps. The Byron Jackson Co. is presently fabricating one of the gas-lubricated seals to fit the primary loop sodium pump. The new seal design will be tested and evaluated under the existing test program. It is anticipated that the pump modifications will be made during the first part of April, 1961.

Sodium Flow Control Valves

Operation of the 3-in. diaphragm-operated sodium flow control valves has been satisfactory. The valves have all provided excellent flow control and have required no maintenance.

Gas-Fired Sodium Heater

Maintenance on the refractory setting in the gas-fired sodium heater has continued to present a problem. The heater is now scheduled for a shut down during the first part of April at which time a new setting will be installed.

CHAPTER III

RESEARCH AND DEVELOPMENT FOR THE LAMPRE PROGRAM

FUEL ALLOY PROGRAM

LAMPRE I Fuel

Further work has shown that 5 at/o silicon added to high purity plutonium-iron fuel is more effective than carbon in reducing corrosion. The silicon also protects the capsule from melt-freeze damage.

Direct Contact Fuels

A survey has been made of the possible ternary alloy systems which may be useful for dilute fuels in direct contact cores. At present there are only four such systems about which enough is known to undertake corrosion studies. There are the alloys of plutonium and cerium with manganese, cobalt, nickel and copper, respectively. The cobalt system is best known, but the copper system is potentially less corrosive to containers of tantalum and niobium. Preliminary tests of the nickel and cobalt systems have indicated a reduced rate of attack on standard tantalum specimens compared to that by the plutonium-iron eutectic fuel. Further tests on all four systems are now under way.

CONTAINER ALLOY DEVELOPMENT

LAMPRE I Capsules

Tests of capsules with tungsten and yttrium additions to the tantalum ingot prepared under laboratory conditions have continued to

exhibit improved structure and better corrosion resistance. In order to produce the same material on a larger scale, a new alloy was cast into an ingot at National Research Corporation and rolled to intermediate size rod. This ingot (K-600) contains 0.75 w/o tungsten and 0.2 w/o yttrium and will be the starting material for the second LAMPFE I loading. The alloy was prepared by first electron-beam melting the tantalum + 0.75 w/o tungsten alloy into small ingots, joining these to form the consumable electrode for the vacuum arc-melting operation, inserting the yttrium metal into grooves machined along the length of the consumable electrode, and then arc-melting the electrode to produce the final ingot. This technique was used to reproduce as closely as possible the experimental ingots prepared at LASL.

A number of tantalum-tungsten alloys, ranging up to 11 w/o tungsten were cast, rolled into sheet and subjected to the LASL cup test. The results indicated that all alloys tested possessed satisfactory deep-drawing characteristics. However, the hardness of the alloys containing 6 w/o or more of tungsten was high enough so that difficulties would be anticipated with the aluminum bronze dies used in deep drawing operations.

The same alloys were also tested for strength and elongation characteristics. Although it is difficult to draw exact conclusions from the data because of the number of variables, there is a trend indicated toward an increase in strength and a decrease in ductility as the tungsten content increases. However, even the 11 w/o tungsten alloy still showed appreciable ductility in the tensile tests.

The plutonium-cobalt-cerium alloy at about 3 to 4 g plutonium/cm³ has shown relatively less attack on tantalum than the concentrated plutonium-iron fuel. The cobalt-cerium binary and cerium alone did not attack tantalum in fairly short tests. A loop in which the ternary alloy was circulated by a sodium jet for about 24 hr disclosed no tendency for the fuel to segregate and no corrosion in the short exposure. A test of niobium in plutonium-nickel alloy showed a mode of corrosion similar to tantalum and similar protection by carbon.

A capsule made of commercial 10% tungsten-tantalum alloy sheet was fabricated by rolling and seam welding. The longitudinal weld had an excellent structure and the capsule has survived 700 hr exposure so far to LAMPRE I fuel at 625°C. This is an important result since the large containers required for direct contact cores may have to use welded seams exposed to the fuel.

DIRECT CONTACT CORE STUDIES

The study of mercury-water jet pump systems has proceeded actively. When nozzles which are wetted by the fuel are used, there is a tendency for bubbles of coolant to be produced in fuel both with mercury-water systems and plutonium alloy-sodium systems. The use of loosely woven mesh screens of a material wetted by the mercury is effective in coalescing such emulsions in the mercury-water system. If such screens are effective with plutonium emulsions as well, they would be a practical component in a direct contact system.

A small scale jet pump loop operated successfully in circulating plutonium-cobalt-cerium alloy by means of sodium. Strong wetting of tantalum surfaces by the fuel was noted, as well as the formation of sodium bubbles mentioned above. Another loop suited to the study of phase separating devices is being planned.

Parametric studies and nuclear calculations have been extended to both very large and very small cores of the direct contact type. One study has been started of the properties of a 300-Mwe plant using such a reactor. The details will be reported separately, but important benefits have already been obtained in clarifying the range of parameters to be considered in the development work. It is apparent that plutonium concentrations in the fuel lower than 1 g/cm³ are not likely to be economically attractive, since decrease in breeding gain is more rapid than increase in specific power.

The distribution of fission products between fuel and sodium phases is important to the direct contact concept. Experimental work on this has not been started, but theoretical estimates indicate that the dilute

fuel will tolerate a very high burnup, perhaps as much as 200%, without any reprocessing except cleanup of the sodium phase and replenishment of the plutonium burned. The few elements which are not extracted by the sodium may cause changes in the viscosity or wetting properties of the fuel, and so the tolerable burnup level must be determined by experiment.

The rapid extraction of many fission products leads to another important problem, namely, the loss of delayed neutron precursors. An estimate for a 225-Mw core at high specific power showed that perhaps only 30% of the delayed neutrons would be emitted in the core. It is not yet clear whether the nuclear stability would be adversely affected.

DEVELOPMENT OF LIQUID FUELS

Plutonium-Cerium-Cobalt Alloys

The work at present is directed toward an investigation of the plutonium-cerium-cobalt ternary system. Additional information that makes it possible to locate the liquidus valley connecting the $\text{Ce}_7/\text{Co}_3/\text{Ce}$ and the $\text{Pu}/\text{Pu}_3\text{Co}$ binary eutectics was obtained during the quarter. A tentative liquidus projection of the ternary system has been prepared; the results obtained for the cerium-rich side of the valley are believed to be reliable but those for the cobalt-rich side are based on insufficient data and will undoubtedly be revised in the future. The data indicate the composition range of alloys which are completely molten at 500°C.

Fuel Alloy Examination

Metallographic examinations and evaluations were made during the quarter of 16 fuel samples, including those of binary alloys of plutonium with carbon, manganese, and osmium, and of plutonium-iron eutectic with additions of aluminum, fluorine, hydrogen, silicon, and sulfur. All samples had uniform microstructures and were considered to be homogeneous.

CONTAINER MATERIALS FOR REACTOR FUELS

Metallographic Examination

The work related to studies of container materials for reactor fuels included examination of 173 samples from 42 tests. Some of the tests were designed to investigate the effects of molten plutonium alloy fuels, and of small additions to the plutonium-iron eutectic fuel, on the corrosion of pure tantalum. Others were made to evaluate the effects of molten plutonium-iron eutectic on various tantalum alloys, and of molten plutonium-cerium-nickel alloy on niobium.

Plutonium-Iron Eutectic Fuel

Additions of carbon and of silicon to the plutonium-iron eutectic continue to appear beneficial in retarding the rate at which intergranular penetration of tantalum proceeds although the attack is not entirely eliminated. Additions of beryllium and boron were not found to be beneficial in this respect.

On the basis of one test in which pure magnesium was held in a tantalum capsule for 200 hr at 750°C, magnesium, when present as an impurity in the plutonium-iron eutectic, would not seem to contribute significantly to intergranular penetration. Only a trace of attack of the tantalum was observed in this test. A layer of sodium on top of the plutonium-iron eutectic did not appear to be beneficial in retarding attack, and a layer of calcium-magnesium alloy, intended to act as an internal getter, was found to be detrimental.

The results of tests in which molten plutonium-iron eutectic was held in contact with various tantalum alloys may be summarized briefly. Thorium additions ranging from 0.5 to 3.0 w/o were not beneficial. Intergranular penetration was observed to become more severe with increasing thorium content. Additions of < 0.01 and 0.015 w/o iron offered no significant improvement over the resistance of pure tantalum to attack, and the addition of 0.5 w/o hafnium, while apparently not detrimental in this particular test, also resulted in no significant improvement.

A tantalum alloy sample containing 3 w/o tungsten was somewhat more resistant to attack than one containing 0.1 w/o tungsten. A recrystallized capsule sample made from 0.1 w/o tungsten-tantalum alloy was found to be attacked to a lesser extent than was a similar sample in the as-ironed condition.

Ternary Alloy Fuels

A ternary plutonium-cerium-cobalt alloy appeared to be somewhat less corrosive than the plutonium-iron eutectic. This result may be due to the relatively low plutonium content (41.8 w/o) that existed in the ternary alloy. Thus, there is some hope that the alloys containing low plutonium concentrations, now being considered for use as fuels for future reactor cores, may prove to be less corrosive than the plutonium-iron eutectic. Binary alloys of plutonium containing 3 w/o cerium seemed to be about as corrosive as, and those containing 2/5 w/o manganese and 8.5 w/o osmium were slightly less corrosive than, the plutonium-iron eutectic.

A niobium capsule and coupon that had been in contact with molten plutonium-cerium-nickel alloy for 200 hr at 750°C were severely attacked intergranularly. In a companion test, the addition of 160 ppm carbon to the ternary alloy appeared to result in less severe attack on a similar niobium capsule although the coupons in both tests were completely penetrated.

Certification Test Results

In the certification test samples that were examined, only slight attack was observed in an NRC 0.1 w/o tungsten-tantalum (with residual yttrium) as-ironed capsule after it had been in test in sodium for 2800 hr at 625°C, and in a recrystallized capsule of similar composition after 2100 hr at 625°C. On the other hand, Temescal 0.1 w/o tungsten-tantalum (without yttrium) capsules were completely penetrated by the plutonium-iron eutectic after 500 to 1000 hr at 675°C. Pure tantalum capsules were found to have suffered moderate to complete penetration after 700 hr at 625°C.

It is noteworthy that in the certification capsules that have been in test for long periods of time (1000 to 4000 hr) significant buildup of gases and inclusions occur in the fuel. Some fuel samples from these tests, particularly those samples taken from the top of the fuel, are so reactive in air that it has been impractical to attempt to examine them metallographically in the open laboratory because of the plutonium contamination hazard. It would seem that in these tests gaseous impurities might have passed from the sodium, outside the capsule, to the tantalum and through it where they could have been picked up by the plutonium fuel.

The sequence of attack that has now been observed in many tantalum samples also seems worthy of noting here. This sequence appears to be: (1) the formation of "worm holes" at isolated locations, (2) general invasion along the cold work flow lines, (3) recrystallization of the tantalum, and (4) severe intergranular penetration.

FUEL REPROCESSING

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). Further experiments were directed toward the complete removal of the balance of the plutonium (~ 0.2 g/l) from the stripped solvent using 0.3M HNO_3 in the existing extraction equipment. Batch tests show that of the reagents tested, viz., FeSO_4 , $\text{H}_2\text{C}_2\text{O}_4$, KF , and K_2CO_3 , only KF was acceptable for the complete removal of plutonium. The resulting precipitate may be recycled with the next dissolver batch.

A procedure has been developed for the disposal of aqueous wastes, especially the raffinate which is 2M HNO_3 and 1.5M $\text{Al}(\text{NO}_3)_3$, and contains β, γ activity in addition to plutonium at ~ 0.01 g/l. The raw solution is mixed with appropriate amounts of cement and vermiculite in a disposable transfer can. After solidification of its contents, the can is encased in a separate concrete jacket.

The equipment required for a pyrometallurgical fuel reprocessing experiment utilizing volatilization, liquation, and slagging is being tested on non-irradiated plutonium.

CHAPTER IV

CORE TEST FACILITY

All efforts during this past period have been devoted to preparation of the design criteria book. The necessary drawings have now been completed and the book is in the final stages of preparation. Selection of an architect-engineer for Phase A should be accomplished in March.

Atmospheric specifications for all of the inert rooms, i.e., the two reactor cells, the radioactive sodium room and the disassembly cell, have been established as nitrogen plus 2% oxygen maximum. Dew point has been set at 6°F in all areas except the disassembly cell which will be held at -40°F.

Four electromagnetic pumps each capable of pumping 3000 gpm of liquid sodium have been obtained as surplus equipment through the AEC Chicago Operations Office. These pumps were originally made for the SIR reactor for the SEAWOLF.