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

QUARTERLY STATUS REPORT ON LAMPRE PROGRAM
FOR PERIOD ENDING NOVEMBER 20, 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.

LAMPRE I

Operation

During this report period the only reactor operation was a 100 watt reference critical measurement made on November 10. There had been a reactivity loss of about \$1.06 since the last previous reference critical measurement on August 9. Modifications being made to the shielding precluded other reactor operations.

The LAMPRE Operating Manual and Emergency Procedures have been extensively revised and amplified in preparation for extended operations at levels up to the design power of 1 Mw.

Shielding

Concrete block shielding has been placed around the fuel loading area to keep radiation levels in abutting areas at below tolerance levels during full-power operations. Various holes and thin places in the existing shielding were plugged and 2 in. of lead were added to part of the wall between the reactor and the control room.

Reflector Pin Activity

One of the stainless steel reflector pins was removed from the reactor and scanned for gamma activity. The predominant activity found was that due to Co^{58} formed by the $\text{Ni}^{58}(\text{n},\text{p})\text{Co}^{58}$ reaction; Co^{60} and Fe^{59} radiations were also observed.

Cover Gas Activity

A high concentration of A^{41} has been detected in cover gas samples taken shortly after shutdown of the reactor. Calculations indicate that the argon probably is due to the $K^{41}(n,p)A^{41}$ reaction; the sodium coolant contains about 200 ppm potassium as a contaminant.

FUEL AND CONTAINER DEVELOPMENT

LAMPRE I Fuel and Container Results

Experiments have continued, using various techniques, on the effects of adding 7000 ppm of silicon to plutonium-iron fuel on the corrosion of tantalum containers. One set of capsules was heated to 775°C and rocked for 16 hr, then exposed for 1400 hr at 635° followed by 700 hr at 675°C . Slight leakage was observed in the gas phase after the last exposure, but similar sets which were re-canned following the initial 16-hr alloying period did not show any failures as a result of the same exposure history. After several weeks of heating, radiographic examination of the capsules with silicon additions showed a low-density phase at the top of the liquid. The nature of this phase is being investigated.

After many months at gradually increasing temperatures, LAMPRE I stock capsules with low-purity fuel finally failed in the gas phase near the closure weld. A number of capsules are now being exposed in the inverted position to determine whether the residual impurities floating on the fuel are the cause of failure, or whether the capsule material is significantly different at this point.

Fourteen capsules made from experimental tantalum alloys produced at IASL have been under test for 2100 hr at temperatures up to 625°C . The experimental ingots contained various amounts of tungsten, thorium, yttrium, or hafnium with the tantalum. No corrosion failures have been experienced

so far; this performance is equal to that of pure tantalum. Continuing exposures at higher temperatures will determine if the materials being tested are superior to pure tantalum.

Four capsules made of commercial Ta-10% W sheet metal by hand rolling and welding have been tested for several months at 625°C. None has failed in the usual gas phase location, but two have developed trace indications of failure in the seam weld. This behavior is encouraging in view of the uncontrolled method of manufacture. Commercial tubing of this material is now available, and capsules with both ends welded in will be tested as soon as possible. Rolled and welded sheet metal of this alloy has also been chosen as the container for the PEX experiment (see below).

It is usually impractical to remove oxygen from fabricated tantalum by vacuum outgassing because of the extremely high temperature required (2200°C). Some interesting results have been obtained, however, in the use of chemical degassing agents at lower temperatures. Baths of molten cerium, calcium, calcium-magnesium alloy, and cerium-cobalt alloy have all shown substantial removal capability for oxygen from tantalum. At temperatures between 900 and 1000°C, removals of 95 to 98% of the initial oxygen content in less than a week were observed for cerium, cerium-cobalt, and calcium-magnesium. In all cases the removal is initially rapid, but becomes very slow at the end. So far it has not been determined whether there is a lower limit to the oxygen that can be removed.

Tensile tests of high-purity tantalum sheet were made at -196, -75, 25, 100, and 300°C, with specimens cut parallel to the rolling direction. At most temperatures yielding was discontinuous, as shown by a drop in load after the initial yield or by an upper and lower yield point. The yield strengths for specimens which displayed a smooth stress vs. strain curve were computed on the basis of 0.2% offset. The tensile strength was evaluated on the basis of maximum load achieved after yielding. In some cases this load was more than the lower yield point but less than the upper yield point. Thus, because all three stresses were computed using the initial cross sectional area, the tensile strength in some instances was less than the yield strength. Average values are given in Table I.

TABLE I

Results of Tensile Tests of High-Purity Tantalum After Annealing

Annealing Temp (°C)	Testing Temp (°C)	Yield Stress (psi)			Tensile Strength (psi)	% Elongation in 1 in.
		Upper	Lower	0.2% Offset		
900	-196			133,900	133,900	3
1150	-196			112,500	112,500	8
900	-75	77,200	72,100		73,800	14
1150	-75	61,500	53,000		54,700	32
900	25	52,250	52,150		57,450	15
1150	25	35,000	30,400	27,200	28,850	51
900	100			46,050	53,100	11
1150	100	15,950	15,150	15,850	34,450	42
900	300			44,800	53,300	7
1150	300			15,900	37,500	27

Dilute Plutonium Fuels and Containers

For high-performance reactor systems it is desirable to dilute the plutonium fuel below the 15 g/cm³ concentration used in LAMPRE I. Three main approaches to dilution are being studied, namely, plutonium alloys which do not contain uranium, molten uranium-plutonium alloys, and uranium-plutonium oxide paste.

Plutonium Alloys Without Uranium

Of the possible alloy systems, the Pu-Ce-Co system is best understood. The main portion of the work on the ternary phase diagram of this system has just been completed. Figure 1 represents a portion of this work as a projection of the liquidus surface upon the composition triangle. In this figure the liquidus valleys, as defined by the lines of univariant equilibria, separate the regions of the various primary phases. Each intersection of the liquidus valleys reveals the existence of a four-phase reaction, and the composition at each intersection is that of the liquid phase participating in that reaction. Temperature contours drawn at 500, 600, and 700°C define the

Fig. 1. Portion of Pu-Ce-Co ternary system phase diagram.

composition limits of the all-liquid region at each of these temperatures. The arrowheads on the boundaries indicate the direction of decreasing temperature. The interphase boundaries and compositions of all the solid phases of the system at various temperatures, which are not shown in the figure, have also been determined.

In preparation for full-scale corrosion testing and in-pile exposure of Pu-Ce-Co fuel, extensive studies have been made of preparation and purification methods for the alloy. Commercial cerium metal contains over 1000 ppm of oxygen, and two methods have been successfully developed for reducing this impurity. Vacuum filtration of cerium through tantalum frits gave oxygen levels of 45-90 ppm, and one trial of vacuum electron-beam melting gave 180 ppm. The beam melted material also showed desirable decreases in magnesium and hydrogen content. One of these techniques will be applied to the preparation of Pu-Ce-Co fuel.

Preliminary lots of this ternary fuel alloy have been tested in ordinary tantalum capsules for 1000 hr at 750°C and 600 hr at 850°C, with no failures so far. Although intended only as a preliminary test, these results are approaching the level required for a high-performance system. However, the effect of high flow velocities has not yet been investigated. Most of the ceramic refractories studied so far have also shown promising results for containment of the Pu-Ce-Co fuel; the materials tested were TaC, TiC, ZrC, HfC, TiB₂, and ZrB₂. In the case of ZrC, metallographic examination showed incipient intergranular penetration along certain grain boundaries. All the rest had unaltered structures and exhibited little or no surface reaction. Ceramic components are of interest for areas in a direct-contact reactor where erosion or other specific corrosion problems might be encountered.

Uranium-Plutonium Alloys

The use of a U-Pu alloy as a liquid fuel is expected to be a difficult technical problem, but the performance possibilities of this system justify an exploratory program. Table II gives composition and melting points for some of the ternary alloys which have been investigated by thermal analysis.

TABLE II

Compositions and Solidus Temperatures of Alloys in Some Ternary Uranium Systems

Metallographic Estimate of Alloy Composition in the Liquidus Valley (a/o)			Approximate Solidus Temperature (°C)
~ 28 Co	5 Cr	67 U	729
5 Cr	~ 24 Fe	71 U	728
5 Cr	~ 18 Mn	77 U	744
5 Cr	~ 32 Ni	63 U	735
15 Mn	5 Si	80 U	760
30 Co	10 Si	60 U	823
30 Fe	10 Si	60 U	809
18 Mn	10 Si	72 U	850
30 Ni	10 Si	60 U	786
5 Fe	23 Mn	72 U	746

So far none of the alternative alloys shows any superiority to the U-Mn eutectic system which has been used in the current tests. This alloy melts at about 730°C, or possibly lower with plutonium present. In tests at 800 and 850°C for 1000 hr, yttrium metal crucibles have stood up well, with tantalum and niobium containers performing next best in order. At 950°C, however, local attack of the yttrium was noted in 200 hr at isolated U-Mn globules. A compound of yttrium and manganese has been reported which may limit the resistance of yttrium. The solubility in molten uranium of tungsten has been found to be lower than for any other metal; scandium is also expected to perform well in this respect. Tests of these materials are under way, but are limited at present by the fabricability of tungsten and the availability of scandium. Fundamental studies of tungsten metallurgy are under way, since it is believed that this material is potentially fabricable.

A preliminary test of five ceramic crucibles with Pu-U-Mn fuel has been run at 900°C. The materials used were: Al₂O₃, Y₂O₃, MgO, BeO, CaO, and Y₂O₃. After 200 hr all showed so little effect of the fuel that no

choice could be made between them. Future tests will be run at temperatures over 1000°C and for longer times. Ceramics are potentially useful for one-region reactors as liners for metallic containers, allowing the container to operate at much lower temperature than the fuel.

PROCESS RESEARCH AND DEVELOPMENT

Reprocessing of UC-PuC

UC-PuC buttons, corresponding in weight to $U_{0.8}Pu_{0.2}Co_{0.95}$, dissolve in hot 13M HNO_3 -0.05M HF to form a reddish-brown solution. No residue was detected. The rate of dissolution was $150 \text{ mg cm}^{-2} \text{ min}^{-1}$ at 110°C. This solution was adjusted to 3M in HNO_3 , 0.1 in $Al(NO_3)_3$, 1.5 g U/liter and 0.38 g Pu/liter in order to extract both the Pu and U into 30 v/o TBP. A selective stripping agent, 0.03 M $Fe(NH_2SO_3)_2$ in 0.5 M HNO_3 , removed the Pu. In three cycles 90% of the total Pu was in the aqueous product. The U contamination was less than 500 ppm. U was stripped from the TBP with 0.01 M HNO_3 . The aqueous solution contained 90% of the initial U, with less than 50 ppm Pu. This method appears to be feasible for reprocessing UC-PuC.

Pyrometallurgical Processing

A volatilization-liquation run using irradiated Pu was completed in which 79.9 g of Pu were alloyed with 2.0 g Fe, held at 600°C for 48 hr and then filtered through a Ta frit. All equipment apparently functioned properly and the run was made without incident. Samples of the filtrate, filter residue, condensate and metal on the stirrer have been sent for analysis. The yield (charge weight/weight of filtered metal) was 0.63, appreciably lower than that of 0.87 obtained with the un-irradiated Pu in a prior run. The lower yield is due to Pu metal retained on top of the dashpot-type stirrer used. The weight of material remaining on the frit was about the same in both cases (12 g).

Analytical results indicate a Pu burnup of 0.24%, agreeing with the expected burnup based on reactor power and time of exposure. Dose rate measurements on samples of the feed and the filtered melt, made with a G.M. counter, show a proportionality in dose rate of 4:1 on an equal weight basis.

Prior to starting the run, the exhaust filters on the glove box were replaced with filters containing activated charcoal in addition to the high-efficiency filter media. The amount of activity in the effluent air was reduced to a trivial amount compared to that found during the previous run.

Paste Processing

A new processing method has been devised for recovery of UO_2 and Pu from paste blankets or paste cores. In recent experiments portions of this process have been tested in a preliminary way. In the proposed process the UO_2 , PuO_2 , and fission products would be separated from the paste sodium by heating in a vacuum retort at about 700°C . The dry residues would be dissolved in HNO_3 (U being oxidized to +6) and fractionally crystallized as the simple nitrates. Most of the U would be rapidly separated in this way, with about 10% of its former content of Pu and fission products. Heating of the $\text{UO}_2(\text{NO}_3)_2$ crystals would convert them to UC_3 in a form suitable for the LASL process for making densified UO_2 , which could then be returned to the reactor blanket.

The crude Pu fraction would be converted to the double ammonium nitrate and fractionally crystallized further, it being very inexpensive to continue the process until any desired purity is obtained. The experiments so far have tested U with 0.5% Pu and tracer quantities of fission products obtained from old water-boiler "soup." The salts are easy to form, and the separation efficiency obtained was approximately as expected. Although a larger scale of operations would be required for accurate design of a process, it is estimated that about eight stages of crystallization would be needed to obtain pure Pu.

CONCEPT DEVELOPMENT

Mobile Fuel System

In concept development, a design study and economic analysis of a full-scale, direct contact reactor station has been completed and is being

issued. The title is "Mobile Fuel Plutonium Breeders, A Study of Economic Potential," (LA-2644). A full description is given of the mode of operation and expected properties of direct contact cores. The reactor studied is based on a Pu-Ce-Co fuel and a UO_2 -Na paste blanket.

U-Pu alloy fuel could be utilized in a one-region, direct contact reactor. In a recently developed alternative concept, tubes of fuel are cooled by a molten blanket in which the core is immersed. The blanket is pumped through the core and then to a Na-cooled heat exchanger. Advantages of such a system include reduced activation of the Na, elimination of parasitic neutron capture in the Na, and lower container temperature than with a jet-cooled core of similar fuel. Also, leakage of neutrons from the reactor could be easily minimized. One of the more promising models studied is outlined below:

Fuel: U-Mn eutectic with 10.7% of U replaced by Pu giving 1.5 g Pu/cm³ fuel; assume 0.28% U^{235} in U.

Fuel Container: 0.020 in. thick Y sheets separating alternate layers of 0.150 in.-fuel and 0.096 in.-coolant.

Coolant and Blanket: U-Mn eutectic

Blanket Thickness: 60 cm, no reflector beyond.

Core Height: 95 cm

Core Diameter: 106 cm

Critical Mass: 660 kg Pu

Breeding (Pu Production) Ratios:

In fuel alone	0.78
Effective in core coolant	0.12
Effective in blanket	<u>0.94</u>
Overall	1.84

Utilization Ratio = (Pu made + U^{238} fission)/(Pu + U^{235}) destroyed = 2.10

Example of Operating Characteristics:

Specific power	500 watts/g Pu
Coolant velocity	10 ft/sec
Average temperature rise of coolant	230°C
Power generated by Pu	330 Mw
Overall power generation	475 Mw

CRITICAL EXPERIMENTS

PBX

A small-scale critical test of a circulating fuel, direct contact core is being designed for installation in the pit building, TSL-29. The assembly will be called the Pint Bottle Experiment (PBX), since the volume of the liquid core is almost exactly one pint. The experiment is to be ready for assembly in approximately 6 months, and will be designed for short-time operation only (not over 4 months). The assembly will operate at a maximum power of 20 kw, and will provide information in the following areas:

1. Zero-power transfer functions of the circulating fuel fast reactor.
2. Effects of ramp reactivity insertions (mechanical).
3. Reactivity changes caused by fuel circulation.
4. Effects produced by the extraction of delayed neutron precursors by Na.

The PBX will be a modest but essential step in the development of power reactor concepts of the direct contact type. It is expected to provide key information in confirming the theoretically predicted self-regulating properties of such cores, and in determining the degree of conservatism which must be used in separating the jet pumps from the core.

In order to keep the assembly as small as possible, the fuel used will be the same Pu-Fe eutectic as for LAMPRE I. In view of the short design life of the test, current tantalum materials and fabricating techniques are believed adequate.

In addition to design work, a Lucite test stand of the core circulating system has been constructed and used to optimize the design of the lift pump using mercury and water. With moderate pumping conditions, a core circulation rate of 2 liters/min is expected to be obtainable in the Pu-Na system.

2000-KW SODIUM TEST FACILITY

Operating Summary

Operation of the sodium test facility was terminated as of September 28, 1961. Total operating time logged on major items of equipment was as

follows:

<u>Item</u>	<u>Total Operating Time (hr)</u>
Steam Generator	9,509
Intermediate Sodium Heat Exchanger	11,920
Primary Loop Sodium Pump	11,920
Secondary Loop Sodium Pump	13,474
Gas-Fired Sodium Heater	10,650
Sodium Cold Trap System	13,662
Air-Operated Sodium Flow Control Valves:	
Primary Loop (two valves)	11,920
Secondary Loop (two valves)	13,474

Operation of the steam generator, intermediate heat exchanger, sodium pumps and other components tested was satisfactory, with the exception of a failure in the experimental, gas-lubricated shaft seal in the primary loop sodium pump.

A final report is currently being prepared in which test results and operating experience will be summarized.

Steam Generator

The steam generator has been removed from the secondary sodium loop and is presently being disassembled for visual and metallographic inspection.

Sodium Pumps

The experimental, gas-lubricated shaft seal in the primary loop sodium pump failed after an operating time of 1030 hr. Failure was detected by a sudden and excessive increase in helium leakage through the inner seal faces. The seal assembly has been removed from the pump and will be delivered to the manufacturer for final inspection.

CORE TEST FACILITY

Phase A

Architect-Engineer

Representatives of the Architect-Engineer, Norman Engineering Co., met twice with LASL and local AEC representatives for the purpose of discussing K Division's requirements. Representatives of the LASL Engineering Department visited the A-E to review informally the final Title I drawings. The Title I package is scheduled to be mailed November 20, 1961, and upon its receipt, appropriate sections will be forwarded to ALO for AEC approval and permission to begin work on Title II.

The areas in which there has been the most discussion are those involving the disassembly cell, core sodium-fuel (NaFu) cells, shield wall thickness, containment volume leak rates, and radioactive sodium (RaNa) room ventilation. General requirements in all of the above areas are more or less firm with the exception of the last. The problem is to locate ventilation ducts for the reactor pits, RaNa room, and RaNa dump tank pit so as not to conflict with the Phase B sodium system. A second problem involving these ducts is the economical design of a radiation trap for the duct penetrations of the wall between the RaNa room and the ventilating equipment room.

A meeting in Los Alamos with the A-E is scheduled for December 11 and 12, 1961. Review of the Title I drawings will take place at this meeting.

Containment

The leak rate requirements of the three volumes which, either routinely or accidentally, may be required to contain radioactivity have been reviewed and are discussed below.

The radioactivity to be contained by the RaNa room-reactor pit volume is that from Na^{24} and possibly solid fission products. In the event of a sodium system rupture, RaNa room temperature (and pressure)

would rise because of transfer of sensible heat from sodium to gas and from the oxidation of sodium. Calculations using pessimistic assumptions indicate that the temperature rise can be limited by the ventilating system water cooling coils so that the pressure rise will not exceed 2.5 psig. The original maximum leak rate of 2.6% per day at a pressure differential of 1-in. of water will still be permitted. The building structure must withstand 2.5 psig.

In the disassembly cell and connected NaPu cell the original specification of 2.6% per day leak rate at a pressure differential of 1-in. of water still stands. It is assumed that very little fission product gas would be released to this volume if there were a NaPu system rupture.

Containment by the gas cleanup cell of xenon and krypton escaping from a break in the cleanup system has a more stringent requirement. Assuming that 10^6 curies of gas might be released and that 10^3 curies per day is the maximum permissible release, a real or equivalent leak rate of 0.1% per day is established. It is believed that the 0.1% per day leakage specification for the Gas Cleanup Cell will greatly increase the construction and acceptance test cost and other methods of achieving the same containment are under consideration. One method would be to vent the cell to an evacuated tank so that the cell pressure could be maintained below that of surrounding areas for some specified length of time. For example, a 450 ft³ tank would be required to maintain reduced pressure in a cell with a 1.0% per day leak rate for a period of 10 days. Included in the above calculation is a large barometric pressure variation.

A second method would be to vent cell gas through a charcoal filter to the atmosphere. This scheme would permit a leak rate of 1.0% per day provided the filter retention of xenon and krypton exceeded 90%. Such filters can be designed to have retention efficiencies greater than 90%.

Phase B

Operating conditions for the four sodium loops have been stabilized to some extent and made a matter of record on a late schematic drawing of the systems.

One major cause for re-evaluation of operating conditions was a decision to use 270°F (150°C) for the temperature rise through the core and heat exchanger. This temperature rise has been used as a design criterion through all loops, causing a downward revision in flow requirements. The lower flow has permitted a decrease to be made from 8 in. to 6 in. in pipe size for the nominal 15-Mw loops. This will result in lower cost and lower pipe stresses. With the 6-in. piping system, the Seawolf pump is capable of producing flow considerably greater than required.

The blanket loop flow is at this time specified at 500 gpm for 270°F temperature rise in the loop. Using 4-in. pipe, this amounts to a pipe velocity of 13 ft/sec and loop pressure drops between 30 and 35 psi.

Revised operating requirements for the Seawolf pump have been forwarded to General Electric for their engineering evaluation of the use of these pumps in the two large loops of the Core Test Facility. These requirements are:

Flow	1500 gpm
Normal continuous operating temperature	750°F
Maximum continuous operating temperature	900°F
10-hr emergency temperature	1200°F.

Bids on a sodium pump for the 500 gpm blanket loop have been received from four vendors. Byron Jackson is apparently low with a bid of \$26,190.00.

The 6-in. RaNa system might, because of a core heat exchanger failure, contain those fission products that are soluble in sodium. Following such an accident, sodium would be drained to the RaNa dump tank. To permit access to the RaNa room the dump tank will require 4-1/2 ft of normal concrete shielding. This thickness was calculated on the assumption that 2% of the fission products from 1-year's operation are in the dump tank.

Recent changes to the sodium system include:

1. One cold trap serving both non-radioactive systems. This system will operate between the two surge tanks and the dump tank now functioning as the expansion tank.
2. Two Mark B Seawolf pumps installed in series in the radioactive 15-Mw loop.

3. Installation of a cold trap on the radioactive 15-Mw loop available for cleanup of circulating system but normally valved off.
4. Use of a 2000-gpm centrifugal pump in the non-radioactive 15-Mw loop.
5. Use of a 500-gpm centrifugal pump in the radioactive 5-Mw loop.
6. Use of NaK for the coolant of the Seawolf pumps as well as for the cold trap systems.

Answers to an inquiry to possible suppliers of sodium cold traps have indicated that Mine Safety Appliances will be the only bidder. As a result of a discussion on cold traps with Atomics International personnel, the following conclusions were drawn:

1. A cold trap would be an excellent way to remove fission particles from a sodium system.
2. It appears advisable that the whole cold trap be made removable rather than just a part.
3. The cold trap removal should be accomplished by means of mechanical joints rather than by cutting and rewelding, due to the plating out of strontium on pipe walls and the resulting contamination from welding and ensuing vaporization.
4. Compartmentalization of sodium components would be advisable for cleanup processes following a system contamination.
5. Consideration of a reactor bypass in the sodium system should be made for its usefulness during a possible cleanup operation.

The possibility was also discussed with A.I. for them to test the Seawolf pumps proposed for the CTF installation. The sodium loop under consideration had the following specifications:

1. Twelve-in. pipe size.
2. No capability of heat addition or removal.
3. A rather awkward pump installation.
4. 4000 gallons of hot sodium waiting to be circulated.

An alternative to having the Seawolf pumps (and possibly other sodium system components) tested by A.I. is to install a 6-in. test loop at TA-35 in the old Steam Generator Building. Whether to do the testing at IASL or to have it done under contract is still undecided.

Specifications for the 15 and 5 Mw, sodium-to-sodium heat exchangers are being prepared for submission to possible bidders. It is planned to let the contract for these heat exchangers before June 30, 1962.

Heat Dumps

The term "heat dumps" includes both the 15 Mw and 5 Mw sodium-to-air heat exchangers together with associated blowers, ducting, and stacks. A gas-fired heater will be incorporated into each heat dump for the purpose of maintaining sodium systems temperatures with no nuclear or electrical heating. Specifications for these two systems are being prepared. It is intended to write the purchase order prior to June 30, 1962.

Shielding

Shielding calculations for the RaNa room have been reviewed. The new calculations were based on the following assumptions:

1. Preliminary piping and equipment layouts, as of November 15, 1961, were used to determine the sodium distribution.
2. Major items of equipment and piping 4-in. in diameter and larger were subdivided into 23 equivalent point sources.
3. It was assumed that a centrifugal type sodium pump having a sump tank capacity of 20 ft³ will be used in the 5-Mw radioactive loop.
4. Sodium specific activity was assumed to be 30 mc/cm³. Only the 2.8-Mev gamma source from Na²⁴ activity was considered.
5. Self absorption in the sodium was neglected.
6. Shielding effects of steel pipe walls, tank walls, and the steel room liner were neglected.
7. No corrections to the calculations have been made for shield penetrations or other shielding defects.
8. A 6-ft thick, ordinary concrete floor was assumed between the non-radioactive and radioactive sodium rooms.
9. A 6.5-ft thick, ordinary concrete wall was assumed between the radioactive sodium room and the electrical equipment room.

Results of the calculations indicate maximum dose rates to be as follows:

- Nonradioactive Sodium Room - 2.1 mr/hr, located at floor level immediately above the sump tank for the centrifugal Na pump.
- Electrical Equipment Room - 0.75 mr/hr, located 5 ft above electrical equipment room floor and at a point closest to the tank for the centrifugal Na pump.

Although confidence in the calculations is probably not better than an order of magnitude, it is believed that the 6-ft floor and 6.5-ft wall dimensions are reasonable design figures. It should be recognized, however, that the piping and equipment layout is far from firm. Should the final design require further large concentrations of sodium, it is possible that some local shielding of equipment may be required.

Hazards

A report entitled "Siting Information for the Fast Reactor Core Test Facility, A Preliminary Study," (LAMS-2629) has been completed.

PUBLICATIONS

"Fabrication of Tantalum Fuel Containers for the LAMPRE-I Molten Plutonium Reactor," Trans. Am. Nuc. Soc., 4, 346 (1961).

"The Development of High Purity Tantalum and Alloys for Liquid Plutonium Containment in LAMPRE-I," *ibid.*, 4, 347 (1961).

"Corrosion Testing of LAMPRE-I Fuel," *ibid.*, 4, 347 (1961).

"Startup of the First Molten Plutonium Reactor," *ibid.*, 4, 364 (1961).

"Some Effects Peculiar to the Liquid Metal Fuel of LAMPRE-I," *ibid.*, 4, 365 (1961).