LAMS-2720

MASTER

395
7-2-64

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

QUARTERLY STATUS REPORT ON LAMPRE PROGRAM
FOR PERIOD ENDING MAY 20, 1962

DISCLAIMER

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

LEGAL NOTICE

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

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

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

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

Printed in USA. Price \$.50. Available from the
Office of Technical Services
U. S. Department of Commerce
Washington 25, D. C.

LAMS-2720
REACTOR TECHNOLOGY
(TID-4500, 17th Ed.)

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

REPORT COMPILED: June 1962

REPORT DISTRIBUTED: June 19, 1962

QUARTERLY STATUS REPORT ON LAMPRE PROGRAM
FOR PERIOD ENDING MAY 20, 1962

Prepared from material supplied by
CMB, CMF, and K Divisions

Contract W-7405-ENG. 36 with the U. S. Atomic Energy Commission

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

LAMPRE I

REACTOR OPERATION

LAMPRE I reactor operations were resumed in April 1962 with a replacement core. The new loading is characterized by improved purity, obtained by controlled preparation and absence of anti-corrosion additives. The capsules have been checked in laboratory certification tests to temperatures significantly higher than will be experienced in normal operation. Hot critical (472°C) was reached on April 20, 1962 with an extrapolated minimum critical mass (shim plus 4 rods inserted) of 21.42 kg. Reactivity measurements taken during the subcritical melting of the fuel are shown in Fig. 1. (Compare LAMS-2564, Fig. 1, for corresponding data of Core I.) Period measurements verified the rod reactivity calibration as unchanged from the previous loading.

Some of the characteristics of the core during the temperature ascent are listed in Table 1 along with the same information for Core I. The differences in the coefficients for the two cores are in part attributable to the intermediate position of the shim during the second operation. With the shim in such a position, the thermal expansion of components with the rise in temperature counteracts a part of the negative coefficient due to the fuel.

During the report period, 110 Mw-hr of operation were completed at various power levels and observations were made of control element performance and reactivity losses. Reactivity loss during reactor operation is shown in Fig. 2 for both the present core and the first loading. The curve is based on

Fig. 1. Reactivity Characteristics During Melting of Mark II
Fuel Loading: 132 capsules = 22.483 kg Pu

Table 1

Temperature Ascent and Fuel Melting in LAMPRE I*

	<u>Core I</u>	<u>Core II</u>
Date	3/27/61	4/18/62
Average temperature coefficient in solid state	-1.3¢/°C	-0.71¢/°C
Total reactivity change during melting	+\$5.60	+\$17.3
Temperature coefficient during melting	+1.5¢/°C	+21.5¢/°C
Rate of reactivity addition during melting	+0.41¢/min	+7.0¢/min
Average temperature coefficient in molten state**	-4.7¢/°C	-1.8¢/°C
Temperature when melting commenced	~ 416°C	~ 416°C
Core loading:		
No. of capsules	121	132
kg Pu	20.61	22.48
Shim position	0 in. (down)	6.7 in. (~ 1/2 way up)
Rods:		
Number	4	4
Position	0 in. (down)	12.7 in. (up)

* All characteristics are based on data taken at relatively low multiplications and plotted in Fig. 2 for Core II and Fig. 1, LAMS-2564, for Core I.

** Temperature coefficient of both cores under normal operating conditions is about 3 ¢/°C.

Fig. 2. Reactivity Loss During Reactor Operation

reactivity measurements made at 1 kw reference power prior to the start of higher power operations. Steady state operation at 600 kw is now continuing on a routine basis.

Air flow baffles have been installed to permit control of the flow of cooling air in the annular flue surrounding the shim. During the recent operations the air flow through the flue was stopped and the flow through the graphite shielding around the core was varied to determine the optimum operating conditions.

CORE LOADING

At the end of 21.5 Mw-hr of Core II operation, one capsule was removed for inspection. X-ray examination and visual inspection disclosed no unusual appearance. A thin layer of slag was present at the top, but it had not disengaged from the fuel. A large number of small bubbles were observed in the upper fourth of the fuel. They are believed to be helium trapped in the fuel during melting and their localization indicates that they are migrating upward. At the time the capsule was removed its source strength was estimated to be about 800 curies. During the removal, the biological shielding of the fuel handling equipment was checked and found to be adequate.

During reactor operations early in May, activity build-up was observed in the cover gas. Analysis of the spectrum and decay characteristics of a sample of the gas indicated that the activity was probably due to xenon isotopes from a leaky capsule. The two most recently installed capsules were removed from the core for examination.

FUEL ELEMENTS

Metallographic examination of a Core I capsule having 120 Mw-hr of irradiation exposure disclosed no major attack. Four locations within the capsule were examined, viz., immediately below the cap weld in the gas phase, in the frothy region, just above the fuel froth interface, and well within the fuel zone. Only slight pitting of the capsule was observed in the frothy

fuel region. The froth contained many large voids, some of which were lined with a light-colored phase having the appearance of a carbide.

Core II utilizes four tantalum alloy melts and 13 fuel melts, all of which have been tested in laboratory furnaces. Closure plugs were machined from commercial grade rod and all adaptors are Type 304 stainless steel. All current capsules are impacted and ironed to final configuration with a final anneal at 1450°C for 1 hr.

The following table summarizes the starting materials for Core II capsules:

<u>Ingot No.</u>	<u>Alloy</u>	<u>No. in Core</u>
K-574	Electron beam melt Ta, 0.1% W	66
K-596	Electron beam melt high-purity Ta	33
K-600	Arc-melted Ta, 0.75% W, 0.2% Y	19
K-606	Arc-melted, high- purity Ta	10

The K-600 Ta and fuel from two melts which have given anomalous results will be programmed out of the reactor during the next quarter. The cause of the unexpected corrosive behavior of this fuel, which meets chemical specifications, is not understood.

An artificial fission alloy containing 27 elements in sufficient quantity to simulate 1% burnup was prepared and loaded into four LAMPRE capsules. These capsules have withstood an initial out-of-reactor test of 700 hr at 625°C without leaking plutonium. Radiographic evidence of poor homogenization of these elements with the Pu-Fe alloy, even after 48 hr of rocking at 700°C, leads to some reservations as to the validity of the results.

FABRICATION OF LAMPRE I CAPSULES

Sixty-five capsules, fabricated from Ta-0.1 w/o W alloy by impact extrusion, are receiving final processing. These capsules are to be used as replacements for in-core fuel elements in the LAMPRE reactor.

Seven tantalum-base alloys (1.6, 0.5, 0.1 w/o W; 0.8 w/o W-0.2 w/o Th; 0.1 w/o W - 0.2 w/o Y; 0.2 w/o Y; and 0.2 w/o Hf) were rolled into sheet from arc-cast billets. Two blanks were cut from each alloy sheet and deep drawn into fuel capsules. These capsules are presently undergoing environmental testing in the sodium cell for purposes of material evaluation.

Fusion welding experiments of Ta-10 w/o W employing heliarc inert gas in electron-beam welding procedures were completed. Tensile tests of the welds produced gave results which showed considerably more consistency for the electron-beam process than for the heliarc method. The appearance of the electron-beam welds was also superior.

RESEARCH AND DEVELOPMENT

GAMMA TELEVISION SYSTEM (GTS)

A gamma-ray conversion television system was utilized to observe for the first time the flow of molten plutonium in the PTA experiment. The system allowed continuous observation of the pumping outlet tubes, reservoir, and core section. Motion pictures were taken of the television screen, and provided a permanent record of this milestone in the program. Some still radiographs were also taken of the pump section in non-operating (Fig. 3) and pumping (Fig. 4) conditions.

A new system is being assembled which will utilize a Rowland Gamma Intensifier and a Bendix BX-7 high-resolution, high-sensitivity, image orthicon television chain. A remotely operated, simplified vertical scanner is being designed and will be constructed soon.

Fig. 3. PTA Experiment in the Non-Operating State as Viewed Through the Mark I Gamma Television System

Fig. 4. PTA Experiment Pumping Molten Plutonium as Viewed Through the Mark I Gamma Television System

FUEL AND CONTAINER DEVELOPMENT

DILUTE FUELS AND CONTAINERS

Currently three types of dilute fuels are being studied for application in a high-performance reactor. These fuels are Pu-Co-Ce alloys, ternary U-Pu alloy systems, and UO_2 - PuO_2 paste.

Plutonium-Cobalt-Cerium Alloys

Molten alloys of Pu-Co-Ce have heretofore been studied on a survey basis. Efforts are being intensified to develop these alloys into a high-performance, fast reactor fuel. The optimism associated with this material is primarily a result of preliminary corrosion tests which indicate that Pu-Co-Ce alloys, at least up to 6 g Pu/cm³, are less corrosive to Ta containers than Pu-Fe alloys under corresponding environmental conditions.

Three corrosion test series form the basis for the above statement regarding corrosion. In the first series, second stage, as-ironed Ta capsules have been employed for containment of alloys with 1, 2, 3, 4, and 5 g Pu/cm³. A series of ten capsules (two of each concentration) has withstood a thermal history of 1000 hr at 750°C, 2000 hr at 850°C, and 700 hr at 800°C. Pu-Fe eutectic alloy would have penetrated this type of capsule in less than 200 hr at 750°C.

The second corrosion series used capsules similar to those of the preceding series and contained fuels of the following concentrations: 4, 5.7, 7.6, 9.5, 11, 13, and 15 g Pu/cm³. After 200 hr at 750°C, all capsules having in excess of 7.6 g Pu/cm³ had shown leakage. Capsules containing 4 and 5.7 g Pu/cm³ fuels have undergone additional test periods of 400 hr at 750°C and 1000 hr at 850°C without evidence of fuel leakage.

The third series has fuels of identical compositions to the second series, but they are contained in high-purity, electron-beam melted tantalum capsules which have been annealed at 1450°C for 1 hr. To date this series has had a thermal history of 700 hr at 625°C followed by 700 hr at 700°C.

No leakage has been experienced, although the containers with the 15 g Pu/cm³ fuels were fractured during the cooling period after the initial period of 700 hr at 625°C. This series is being subjected to further testing.

An increased emphasis will be placed in the future on this fuel system. Experiments are being planned for irradiation testing of 3 g Pu/cm³ fuel, determination of selected physical property data, and development of capsules made by welding bottom tips and top caps to seamless tantalum tubing. Studies of the welding process with tantalum and of the susceptibility of the welds to plutonium penetration are required for the development of complex reactor core configurations.

Tentative results from the U.K. (AERE-R-3122) indicate that the alloy 14 a/o Pu-25 a/o Co-61 a/o Ce has a negative coefficient of linear thermal expansion ($\alpha = -25 \times 10^{-6}/^{\circ}\text{C}$). The results of quenching experiments, with an alloy having a composition close to the eutectic valley in this system, seem to confirm the British report. Tantalum tubes, which contained Pu-rich alloy specimens during heating and quenching to room temperature from above their melting ranges, were found to have ruptured during the experiments, as if the specimens had expanded during cooling. Because of this apparent evidence for an undesirable positive temperature coefficient of reactivity (negative expansion coefficient), four alloy specimens having compositions in the range from 60 to 30 a/o Ce and from 15 to 45 a/o Pu were examined dilatometrically and were found to exhibit normal, positive expansion behavior on heating from room temperature to 400°C. The results are summarized in the following table:

Alloy Composition (a/o)	Coefficient of Linear Thermal Expansion * ($\times 10^{-6}/^{\circ}\text{C}$)
15 Pu - 25 Co - 60 Ce	+ 27
25 Pu - 25 Co - 50 Ce	+ 23
35 Pu - 25 Co - 40 Ce	+ 27
45 Pu - 25 Co - 30 Ce	+ 10.5

* The coefficients reported are averages of the values observed for duplicate heating and cooling runs for each specimen.

It is noteworthy that the absolute magnitude of the coefficient for the U.K. alloy ($- 25 \times 10^{-6}/^{\circ}\text{C}$) is approximately the same as that for the LASL specimen of nearly the same composition, but the difference in sign cannot be easily explained. The rupturing of the tantalum capsules was probably caused by something other than expansion of the homogeneous alloys during cooling, but still lacks adequate explanation.

The British investigators found positive expansion on melting and subsequent heating of their alloy in the liquid state. Moreover, the dilatometric work described above indicates positive expansion of alloys in the solid state. It seems likely, therefore, that the Pu-Co-Ce alloys, at least those containing 45 a/o or less of Pu and lying near the bottom of the eutectic valley, will have a safely negative temperature coefficient of reactivity at all times.

Uranium-Plutonium Ternary Alloy Systems

The search for ternary eutectics of high uranium content has been continued and another system, namely, Co-Ru-U, in addition to those mentioned in earlier Quarterly Reports, was found to contain a ternary eutectic. The eutectic has a composition near 30 a/o Co, 5 a/o Ru, and 65 a/o U and melts at about 714°C . The addition, however, of the third component (Ru) to the U-Co binary eutectic does not result in lowering the melting temperature appreciably.

Only one more ternary system, viz., Ni-Si-U, seems to be worth studying by thermal analysis. Because microstructures of these alloys have been unusually difficult to interpret, a probable eutectic valley has not yet been located. The pertinent U-Ni binary eutectic was found to contain 39 ± 0.5 a/o Ni, rather than 31.5 a/o reported in the literature.

Because of the unpromising results that are being obtained in the search for low-melting ternary eutectics of high uranium content, the U-Pu-Mn system is being investigated. Some 30 alloys in the area bounded by 10 and 35 a/o Mn and by 5 and 25 a/o Pu, have been thermally examined.

Although the Pu-U alloy is expected to be more difficult to handle than alloys containing no uranium (by virtue of the higher melting points involved), its potential as a high-performance reactor fuel makes it worthy of attention.

In previous reports corrosion results have been given for various containers with U-Mn binary alloys. During this report period, initial survey experiments were completed using, in most cases, a fuel of composition 75.1 a/o U-4.5 a/o Pu-20.4 a/o Mn. These tests were run at 900°C for 200 hr and are preliminary in nature and scope. The results are given in Table 2.

Uranium Oxide - Plutonium Oxide Paste

A paste flow apparatus using a positive displacement pump as described in the preceding report (IAMS-2681) has been completed. An attempt to extrude a column of settled paste 0.5 in. in diam and 2 in. in length through a tube 0.170 in. in diam and 5.5 in. in length at a temperature of 300°C was unsuccessful. A piston pressure of 8500 psi proved insufficient to extrude this settled 37.5 v/o, < 10 μ UO₂, plus 1.5 v/o U metal paste.

The pump was modified by employing a 1/2 in. diam piston. The paste specimen previously described was allowed to settle for several days at which time an attempt to move it was made at a temperature of 490°C. A pressure of 27,000 psi proved insufficient to extrude the paste although the piston rod bent elastically. The rod could be removed from the piston without difficulty, indicating that internal binding of the assembly did not cause the high resistance force.

A further experiment using 105-150 μ UO₂ powder will be performed at a future date. The amount of effort on this oxide system will be de-emphasized to allow intensified development work on the Pu-Co-Ce system.

Table 2

Exposure of Test Containers to 75.1 a/o U-45 a/o Pu-20.4 a/o Mn
Fuel at 900°C for 2 hours.

<u>Test Container</u>	<u>Observations</u>
Yttrium (1100 ppm O, 70 C, 35 H ₂ , 30 N ₂)	Not more than 1 mil of solution attack anywhere. No intergranular attack.
Scandium	Complete intergranular attack of 40-mil wall. Some solution attack.
Calcium oxide (10 ³ -10 ⁴ ppm Mg, 100-1000 Si, Fe, Al)	Slight reaction layer. 10 ppm Ca present in fuel. Metallography showed no attack.
Yttria	Slight, if any, reaction layer. Metallography showed very little penetration by fuel. Less than 5 ppm Y in fuel.
Cerium sulfide, CeS _{0.99} * (1000 ppm Si, 100 Mg, Ca)	Wetted. No attack observed by metallography or visually. 80 ppm Ce, 10 S in fuel.
Beryllia, 2.90 g/cm ³ 99+% (3000 ppm Al, 500 Ca)	Slight, if any, reaction layer. Metallography showed microcracks or intergranular penetration up to 1 mil deep. 10 ppm Be in fuel.
Magnesia (10 ³ -10 ⁴ ppm Si, 2000 Ca, 500 Fe, 200 Ni, 100 Mn)	Very slight reaction layer. Up to 3 mils of intergranular attack. 10 ppm Mg in fuel.
98 w/o Al ₂ O ₃ + 2 w/o Y ₂ O ₃ (10 ³ -10 ⁴ ppm W, 5000 Si, 1000 Ca, 100-1000 Zr, 200 Fe)	Definite reaction layer between fuel and container. Up to 2 mils of pitting attack. 250 ppm Al and less than 5 ppm Y produced in fuel.

* Fuel was 74.1 a/o U, 4.9 a/o Pu, 21.0 a/o Mn in this case.

It appears that yttrium withstands attack by the Pu ternary alloy better than by U-Mn binary alloys. All the container materials, except magnesia and the Al₂O₃-Y₂O₃ ceramic, will be subjected to further test. In addition, new ceramics and alloys of the system W-Re will be tested in the effort to find a corrosion-resistant container material.

FUEL REPROCESSING

Electrorefining equipment, similar to that used for the electrorefining of unirradiated Pu metal (LA-2666), but modified to permit remote handling, has been installed in a hot cell. Two experiments have been completed and a third is in progress. In order to check the operation of the equipment, a 170-g specimen of unirradiated Pu metal was used as feed material in the first experiment. The experiment was successful and 117 g of Pu were collected in the product ring with a collection efficiency of 96% and an anode efficiency of 87%.

In the second experiment, Pu-10 a/o Fe alloy was the feed material. Irradiated Pu was used in the preparation of the alloy to give a feed equivalent to a burnup of 0.06% of the Pu, with a 9-month cooling period. Current efficiency and product collection were also good in this run. The decontamination factors obtained are shown in Table 3.

Table 3
Decontamination of Plutonium Metal by Electrorefining

<u>Nuclide</u>	<u>C_o/C*</u>
Zr-95	13
Ru-106	350
Ce-141, -144	> 380
Cs-137	> 93

* Concentration of nuclide in the feed over that in the product.

Three solvent extraction runs were made on dissolved irradiated Pu using a secondary (LA-1) amine. The results of these runs are shown in Table 4.

Table 4

Decontamination of Plutonium by Solvent Extraction

<u>Run No.</u>	<u>Activity</u>	<u>Pu Recovery</u> ⁽¹⁾	<u>β Decontamination</u> ⁽²⁾	<u>γ Decontamination</u> ⁽²⁾
12-1	~ 1 r/hr	95%	96.3%	94.1%
13-1	~ 16 r/hr	90%	97.7%	97.1%
14-1	~ 150 r/hr	96%	97.9% ⁽³⁾	96.8% ⁽³⁾

(1) $\text{Pu recovery} = \frac{[\text{Pu}] \text{ product}}{[\text{Pu}] \text{ feed}} \times 100.$

(2) β and γ decontamination percent = $\left(1 - \frac{[\beta, \gamma] \text{ product}}{[\beta, \gamma] \text{ feed}}\right) \times 100$ and does not take into account the γ activity due to Pu.

(3) Even with this decontamination, the product solutions read 10r(γ)/hr at contact. The major fission product present is Zr-Nb⁹⁵ with a trace of Ru¹⁰⁶. Removal of these will be attempted by an oxalate precipitation procedure.

CORE TEST FACILITY

PHASE A

Title II

During this period IASL representatives met four times with Norman Engineering; three meetings were in Los Angeles and one was in Los Alamos. A summary of the discussions appears below.

Building and Grounds

Because of the climate, ordinary roof draining schemes cannot be employed in Los Alamos. It is required that water from the roof be collected, taken into an inside downspout, and carried underground to a spill point at such a location that an accumulation of ice is not a hazard.

A new noise study has been completed by Bolt, Beranek, and Newman with the recommendation that building modifications and extensive noise attenuators will not be required to attain satisfactory working conditions in and near the FRCTF.

Utilities

Cathodic protection of underground pipes and conduits has been added by the AEC. Criteria for this system are being prepared and will be transmitted to Norman Engineering.

Heating and Ventilating

Consideration of the consequences of failure of the blowers associated with ventilation systems No. 1 and No. 2 have led to the requirement that these blowers have a paralleled automatic alternate. In both systems the recirculated nitrogen is used for heat removal, No. 1 from the disassembly cell and No. 2 from the radioactive Na area. Loss of cooling in either area would result in prohibitively high temperatures for concrete and installed equipment. Water cooling coils are also to be installed in parallel.

Requirements for control and instrumentation of systems No. 1 and No. 2 have been given to Norman Engineering. Each will have a semigraphic panel in the electrical equipment room where an operator can diagnose and take corrective action if a malfunction alarm is sounded.

Hot Cells

Norman Engineering is not being asked to design the hot cells completely and is being supplied with fairly detailed drawings. Cell equipment will mostly be supplied by LASL for contractor or user installation.

An unresolved problem in the Na-Pu cells is installation and subsequent removal of alpha barrier between the cells. This barrier, installed before disassembly operations contaminate the space, will permit the west Na-Pu cell equipment to be installed in a clean work space. By removing the barrier (remotely because one side is dirty) after equipment installation, the west Na-Pu cell will have available to it the east Na-Pu cell manipulators, hoist, and equipment transfer cell.

Gas Cleanup Cell

Drawings, in some detail, of the gas cleanup cell have been delivered to the A-E. The cell width has been increased in order to make possible the use of a vehicular manipulator.

Control Room

The line defined by the front of the racks has been specified so that the A-E may locate the rack cooling exhaust duct. The duct will be installed without openings which will be cut to fit the needs of the individual racks when their layout is known.

The extent to which automatic data processing is to be employed is under study. The minimum requirement is a several hundred channel data logger.

Penetrations

General location, size, and number of penetrations into controlled atmosphere areas have been furnished Norman Engineering. These penetrations are in three general classes: (1) leak-tight piping for gas or liquid metal, (2) conduit, and (3) sleeves for remote valve operation. A tabulation of these penetrations and their requirements is in preparation.

PHASE B

Sodium Systems

The engineering study report of General Electric concerning the use of Seawolf pumps for the LASL application has been received. It appears that the Seawolf pumps should be satisfactory, and so it has been decided to use two, one in each of the 6-in. loops. The two Type AF dockside spares at Atomics International have a superior type of mounting bracket, and as they have not been through the fairly violent temperature tests that have been imposed on the two pumps now on hand, a decision has been made to install the two Type AF pumps in the loops and keep the Types AJ and AK for spare parts. This will give 100% spare status.

Studies have been made to determine convection flow potentials of the four main sodium loops using the best estimates of core loop and blanket configurations and the latest piping layouts. These calculations show that the 15-Mw radioactive Na loop will not sustain convective flow because of the small difference in elevation between the core heat exchanger and the

intermediate heat exchanger. The 15-Mw nonradioactive Na loop, and both 5-Mw loops, however, will sustain convection and it appears the latter two should be satisfactory for removal of 500 kw shutdown heat from the core region. As yet undetermined is the unbalance of convection between the 5 Mw radioactive Na and nonradioactive Na through the intermediate heat exchanger.

Single line piping drawings of the two nonradioactive Na systems have been completed. Major components have been located with respect to building column lines. These drawings are detailed to the extent that material take-offs can be made. The layouts were used to complete pipe stress analyses. Single line piping drawings of auxiliary systems, such as drain lines, hot traps, and cover gas supply, can now be started.

Gas Systems

The gas distribution system has been laid out sufficiently to allow positioning of floor penetrations in the building and piping for these penetrations is being ordered.

A design study is being made of a condenser to be mounted above the core cover gas space in the exit line to the gas cleanup system. Its purpose is to condense Na vapor from the outgoing gas and return it to the core. The design being studied first will involve cooling the outgoing gas with the returning gas from the cleanup system.

The gas cleanup system is being designed to circulate $1 \text{ ft}^3/\text{min}$ or 10% of the cover gas volume per minute, whichever is less. It is assumed that krypton and xenon will be the only gaseous fission products present in the cover gas, since other fission products will remain in the Na system or the fuel alloy. In order to dispose of some of the very short-lived gaseous fission products, the cover gas region will be designed in such a manner that fission products emerging from the Na will remain in the cover gas region at least 10 min before being removed to the gas cleanup system.

The gas cleanup system will be designed to keep hydrogen and nitrogen down to acceptable concentrations and to delay krypton for at least 48 hr.

A system adequate for krypton removal will also be adequate for xenon removal. Oxygen, carbon monoxide, carbon dioxide, and hydrocarbons are not expected to be present in the cover gas, but the system will be capable of removing them.

Control Room

LASL personnel visited the ETR, MTR, and EBR II installations at the National Reactor Testing Station for the purpose of obtaining engineering information on data logging, nuclear instrumentation, control systems, and cable penetrations.

Critical Path Network

Phase B effort is to be scheduled and followed using the "critical path" method. Two members of NASA visited LASL and assisted in setting up critical path programs for the Fast Reactor Core Test Facility and UHTREX. A network for the Na systems has been completed and a successful IBM run made. Using the Na network as a starting point, the remainder of Phase B work will be incorporated to complete the diagram.

Hazards Study

The time needed to prepare a hazards study plus the time for the Division of Licensing and Regulation consideration and approval will result in the earliest start of construction date of September 1, 1962. Since the Title II review can be completed by July 15, the letting of a construction contract can be held up by hazards approval. Preparation of the hazards study is being expedited.

It is obvious that a complete hazards summary in the usual sense cannot be made until a detailed design of the reactor core exists. At this time there are two promising core concepts, but the design of a specific

core must await the results of current research programs. Consequently, most of the effort for the hazards study in preparation will be devoted to a description of Phases A and B, with only a general evaluation of hazards from an operating reactor with a definite, engineered core. In particular, the possible troubles and dangers concerned with the Na coolant systems and the radioactive core cover gas systems are being examined.