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FOREWORD

The Reactor Development Program Progress Report, issued monthly, is intended to be a means of reporting those items of significant technical progress which have occurred in both the specific reactor projects and the general engineering research and development programs. The report is organized in a way which, it is hoped, gives the clearest, most logical over-all view of progress. The budget classification is followed only in broad outline, and no attempt is made to report separately on each sub-activity number. Further, since the intent is to report only items of significant progress, not all activities are reported each month. In order to issue this report as soon as possible after the end of the month editorial work must necessarily be limited. Also, since this is an informal progress report, the results and data presented should be understood to be preliminary and subject to change unless otherwise stated.

The issuance of these reports is not intended to constitute publication in any sense of the word. Final results either will be submitted for publication in regular professional journals or will be published in the form of ANL topical reports.

The last six reports issued
in this series are:

September 1961	ANL-6433
October 1961	ANL-6454
November 1961	ANL-6473
December 1961	ANL-6485
January 1962	ANL-6509
February 1962	ANL-6525

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I. WATER COOLED REACTORS

A. EBWR

1. Reactor Modifications

The original rack, pinion shaft, and seal shaft components in the EBWR control rod drives were fabricated of 17-4 PH stainless steel aged at 900°F. These components were replaced this month with components made of 17-4 PH aged at 1100°F (see Progress Report, July 1961, ANL-6399).

The design of the new pinion shafts and seal shafts were modified to permit a control rod to scram that might stick by reason of debris caught between the teeth of the rack and pinion gear.

2. Reactor Operation

a. Soluble Poison Control System - The soluble poison control system was modified for 100-Mw operation. The modified system has been tested and the performance results are now being evaluated.

b. Transfer Functions - Transfer function measurements have been made using the rod oscillator at 32 selected frequencies from 0.0012 to 9 cycles per second with the reactor operating at the following conditions:

- (1) Zero power, room temperature.
- (2) 10 and 20 Mw, 600 psig, makeup water through lower feedwater ring, and boric acid concentration about 4 gm/gal.
- (3) 20 Mw, 300 psig, makeup water through lower feedwater ring, and boric acid concentration about 4 gm/gal.

Three different methods were used to measure the transfer function. The first method uses the original null analyser. The second method is based on analog sine-cosine cross correlation and uses a small analog computer to perform the multiplications and integrations required in the computation. The third method uses a sine-cosine cross correlation, which involves the use of an analog-to-digital conversion and data storage service. Raw analog data is digitally converted and stored on a magnetic tape. The magnetic tape is later read into the IBM-704 digital computer and processed according to a prearranged program. The output from the IBM-704 gives the amplitude and phase at each frequency point.

The digital data system is also being used to analyze the statistical variations in neutron flux (noise) that are normally observed in

the reactor with stationary control rods. Auto-correlation and power density spectrum calculations are being made of the noise data at each reactor operating condition.

The root mean square value of the reactor noise is being measured and an analog computer is used to perform the required squaring and integrating functions.

The transient response of the reactor is being measured by running the oscillator rod from its full down position to its full up position at a velocity corresponding to 4 cycles per second.

B. BORAX-V

1. Operation

The loading of boiling fuel assemblies continued with the 60th (and final) assembly loaded on March 9, 1962. Figure 1 is a photograph of the fully loaded core. Boron stainless steel poison rods were added to reduce the flux at the center of the reactor and to correct local flux peaking. Thirty-two poison rods were distributed throughout the central assemblies.

Figure 1. BORAX-V Loaded with Boiling Core

Reciprocal count rate curves were plotted for loading position, central rod fully withdrawn plus intermediate rod half withdrawn, and central rod fully withdrawn. As the final loading was approached the reactor was made critical with the central rod fully withdrawn and one of the intermediate rods partially withdrawn. Period measurements were

made to determine rod worth. These measurements were used to predict reactor critical with only one rod withdrawn. However, in the course of tests, it was found that the reactor could go critical with one intermediate control rod withdrawn. This condition was immediately changed by adding four additional poison rods and relocating eight others to reduce the flux around the four intermediate rods (see Figure 2). The reactor now will not go critical on the withdrawal of any single control rod.

Figure 2. Fully Loaded Boiling Core BORAX V

When the full 60-assembly loading was reached, including 36 boron stainless steel poison rods, the temperature effect was measured and found to be $-0.004\% \Delta k/k$ per $^{\circ}\text{F}$ for a temperature change from 67°F to 105°F .

To determine the reactivity effect of substituting water-filled rods for fuel rods, four water rods were installed in the center of the fuel assemblies in core positions 33, 34, 43 and 44 (see Figure 2). The water rods were found to be worth $+0.006\% \Delta k/k$ per rod as determined by the change in bank critical control rod position. Replacing two fuel rods with water rods in the center of the fuel elements in core positions 23 and 24 produced a reactivity effect of $+0.018\% \Delta k/k$ per rod.

Flux wire irradiations using 3.4 w/o U-Zr wire were made to determine the radial flux distribution across the reactor in three radial traverses. These preliminary measurements indicated a maximum-to-average ratio of 1.28 radially.

In addition, a single assembly was modified for detailed flux mapping by making one of the aluminum sides removable. Data were obtained at the surfaces of selected rods, by using short flux wires supported in a square polyethylene block (drilled so that all 49 rods go through it). Two diagonal plots through a central assembly are shown in Figure 3. Both show the effect of a boron stainless steel rod. The effect of a water rod at the center is compared with a fuel rod in that position. The maximum-to-average value within this single assembly was 1.59.

Figure 3. Diagonal Flux Traverses through Boiling Assembly

Differential worth of control rods was measured for various configurations of control rod vertical position. As measured in the core without boric acid by the deviation of the rod from the bank critical position, the maximum worth for the center control rod was less than $0.5\% \Delta k/k$ per inch. Each intermediate rod is worth 57% of the center rod, and each outside rod is worth 17% . These values effectively decrease when the core is poisoned with boric acid.

Attempts were made to measure the shutdown worth of control rods by the rod drop method. These measurements were unsuccessful because of the effect of mechanical vibration on the chambers located within the reactor. The preliminary estimate of shutdown reactivity worth, based on subcritical count rate changes and differential rod worth measurements, is about 7% . Some error is known to be present due to interaction of control rods. Partial information from boric acid poison experiments also indicates the shutdown margin is about $7\% \Delta k/k$.

Boric acid solution has been added to the reactor to calibrate control rods and to measure the reactivity effect of boric acid. The worth of boric acid is measured to be about $-0.6\% \Delta k/k$ per gram/gal.

2. Installation and Modification of Reactor and Components

Based on the control rod worths measured above, and making allowances for increase of rod worth at operating temperature and the change in worth of a control rod when it is far from the bank critical position, new control rod withdrawal speeds were established to give a maximum reactivity addition rate of less than $0.05\%/sec$. New gears were installed to produce rod speeds of 4 in./min. for the center rod, 5.04 in./min for individual intermediate rods, and 6.12 in./min for outside rods.

A number of plant modifications have been completed: The batch feed discharge piping has been revised. Nine control rod seal thermocouples were installed. A new, larger pressure gauge for television viewing has been installed, and insulation on the reactor-water-level gauge glass was modified. A seal-water differential-pressure control system for supplying rod drive seals, auxiliary pump seals, and forced-convection pump seal water was installed and calibrated. A spring-check valve for the feedwater system has been installed close to the reactor vessel to prevent water hammer in the feedwater system.

The suspended solids filter in the water sample panel was modified to facilitate removal of the sample.

3. Procurement and Fabrication

a. Superheater Fuel - Inspection continues on superheater fuel plates for the peripheral superheater. Of the 230 HPE (half-enriched peripheral) plates received, 106 have been inspected, 101 accepted, one rejected, and four are being corrosion tested.

Final brazing development work on the four plate superheater fuel elements continued with a total of 14 elements brazed this month. Two batches of three and five elements each appeared to be brazed satisfactorily. However, another batch of five elements was not satisfactory. Four of the elements were sectioned transversely at 4-in. intervals, and the brazed joints examined. Porosity and incompletely filled side plate channels were found in some sections of each element. The spacer wire-to-fuel plate joints were sound in two elements; however, completely nonbonded sections were found in two other elements. The fifth element was prepared for a hydrostatic pressure test, but excessive leakage through brazed joints prevented the test.

During the pre-braze assembly of the elements, braze alloy has been applied along both sides of the spacer wire-to-bottom plate joints. By inverting the completed assembly, the alloy was supposed to flow not only under the wire where applied, but downward and under the opposite side of the wire. Either this flow was not reliable or the mixture of powdered brazing alloy and carrier was not uniform, resulting in spots with insufficient alloy.

Another single element was prepared with the brazing alloy and carrier well mixed, and the alloy was applied along the top of each spacer wire as well as along both sides. After brazing, this element was sectioned transversely many times and examined. All brazed joints were satisfactory. A second single element was brazed while resting on edge (instead of flat); this element also appears to be satisfactory.

At month's end another batch of five developmental elements had been prepared for brazing. A total of 32 HPD (half depleted peripheral) and 32 FPD depleted uranium plates have now been brazed. Assembly of the superheater fuel assembly nozzles and risers is proceeding.

b. Reactor Components - The stainless steel type 304L cladding angles for the new control rods have been formed and hydrogen annealed. Spot welding and fusion welding development work was completed, and spot welding assembly of the new control rods was started. The slug-to-cladding spot welds are being radiographed before the edge of the cladding is seal welded.

Fabrication of ten 17-4 PH stainless steel control rod extension shafts has been completed. Fabrication of one more spare shaft is complete except for chrome plating and brazing. Ten shafts have been tested, at operating pressure and temperature, for 100 motor-driven cycles and 50 scram cycles each, with no significant wear or scratching.

A dummy control rod with the new design of follower-to-cruciform extension mechanical joint has been scrammed 100 times at 600 psig, 489°F, with no damage to the joint or rod. These scram tests continue.

A total of 22 boiling fuel rods which did not meet specifications were found during the loading of the boiling core. The three defective boiling fuel assembly boxes found last month have been repaired.

c. Experimental Components - Assembly of the instrumented boiling fuel assemblies continues. A successful brazing technique has been developed for the pressure seal joint between the stainless steel sheathed instrument leads and the pressurized terminal box seal plate for the instrumented boiling fuel assembly. Preparations are being made for the actual brazing of these plates.

Fabrication of the movable steam probe is continuing.

Fabrication of the drive unit for the oscillator has been started. Zircaloy-2 tubes for cladding the rotating oscillating rod have been rolled, welded, and machined. A welded sample is being corrosion tested. All components have been ordered or are on hand for the oscillator and drive unit.

Fabrication of the H_2 - O_2 recombiner test rig is complete with the exception of the Knallgas (H_2 , O_2) generator and piping system. The differential pressure cells for measurement of pressure drop through the demister and for void fraction and liquid level determination have been calibrated. The system has been hydrostatically tested at 900 psi. A run utilizing the heaters to produce 600 psig saturated steam has been made to check operation of the system before installing the platinized demister and the oxygen and hydrogen system.

Construction of the four pigs for the fuel rod gamma-scanning system has begun. The root-mean-square noise analysis equipment is now operational.

4. Development and Testing

a. Plant Tests - The boron-addition heating system was completed including tubular heaters in the bottom of the boron tank, tape heating on valves in the piping, and strip heaters on the top water level gauge glass. These heaters and their controls were tested at room temperature and operating temperature (489°F). The necessary boric acid can be contained in solution at temperature between 200° and 489°F. Considerable adjustment was required in the placement of the heating elements to maintain uniform temperatures throughout the system.

Dissolving and sampling equipment has been designed, constructed, and put into use for the experimental boric acid addition to the reactor. Samples of reactor water are being analyzed for boric acid concentrations.

The feasibility study and investigation of Polybor (di-sodium octaborate) continues in an effort to provide a boron solution of sufficient concentration at room temperature. It has been determined that the pH of the Polybor solution can be controlled with boric acid. Attempts to remove Polybor from the solution may not be feasible using mixed bed resins because of the formation of an unidentified gas containing boron. Fabrication of static corrosion test equipment was completed and is ready for use. Strips of Type 304 stainless steel and Type X-8001 aluminum combined in couples will be tested in various combinations of Polybor and boric acid at 600 psig, 489°F conditions.

b. In-Core Instrumentation - Improvements in high temperature thermocouples are still desired to extend their operating limits to 5000°F. Specifications have been prepared for the purpose of obtaining alloyed tungsten/rhenium-sheathed thermocouples utilizing high purity thoria insulation and tungsten-tungsten/26% rhenium thermoelements. Vendors now are doing tungsten alloy sheath fabrication work but have not yet successfully compacted high purity thoria. The use of thoria/10% magnesia is not considered satisfactory since in earlier tests magnesia was shown to be undesirable due to its high vapor pressure and low chemical inertness.

c. Corrosion Tests - Testing of stainless steels in oxygenated superheated steam at 650°C and 600 psi continues (see Progress Report, January 1962, ANL-6509). Samples were not inspected during this period. However, oxygen content of the effluent condensate remained constant at 23.8 ppm. Input oxygen concentration during this period was 31.7 ppm.

d. Exponential Experiment - Several additional measurements were made on the subcritical experiment using 4.95% enriched BORAX V fuel in the TREAT shield. Using a special slotted fuel rod, a determination of the typical flux depression within a rod was made. This rod was slit transversely in such a manner that a $\frac{3}{8}$ -in. -long U-Zr wire could be mounted to the slit. After irradiation, the wire was cut into small segments, counted, and weighed. By this method, the ratio of flux at the surface of the rod to the radial average was found to be 1.12.

The effect of placing eight 10% enriched fuel rods dispersed symmetrically about a central 5% rod was assessed. A peak-to-average thermal flux within the assembly of about 1.5 was thus obtained.

The exponential experiments have been concluded since the fuel was required for the boiling core loading in BORAX V. A report covering this work is in preparation.

II. LIQUID METAL COOLED REACTORS

A. General Research and Development

1. Evaluation of Carbide and Oxide Fuels for Large Fast Breeder Reactors

A preliminary study has been made to assess the ultimate potential of the uranium oxide and carbide fuels in terms of achieving minimal fuel cycle costs in large fast breeder reactors. A parametric heat transfer analysis was made for fast reactor breeder cores using these fuels. This heat transfer data was then used in conjunction with the CYCLE and COST code to calculate fuel cycle costs for typical breeder cores using each type of fuel.

The five reference problems (see Progress Report, February 1962, ANL-6525) have been analyzed using the CYCLE and COST codes and the results are shown in Table I. The fuel cycle cost calculations are given in the last column. Two of the uranium carbide-fueled systems show a saving on net fuel cycle cost of 0.60 to 0.80 mills/kw-hr over the two oxide systems. The third carbide system is one in which the thermal efficiency was limited by a low outlet temperature. Consequently, it has a high fuel cycle cost of 2.46 mills/kw-hr.

Table I. Results of Cycle Calculations for the Five Reference Problems*

Problem No.	Volume (liters)	Material	Net Power (Mwe)	Blanket Power Fract.	Velocity (ft/sec)	Rod Diameter (in.)	Total Fuel Loading (kg metal)	Avg Pu Loading (kg) All isotopes
179.40001	6080	Oxide	589	0.068	42	0.25	29,459	2,575
179.40002	601	Carbide	704	0.225	109	0.25	3,821	418
179.40003	1404	Carbide	567	0.299	30	0.25	3,837	655
179.40004	2676	Carbide	640	0.154	30	0.50	14,426	1,226
179.40005	2676	Oxide	677	0.188	30	0.125	8,194	1,193

Problem No.	Enrichment, %		Total Breed Ratio	Thermal Eff. (%)	Time Between Fuel Loadings (days)	Sodium Volume Fract.	Fraction of Pu in Core Feed Matl.	Core and Blanket Cost (mills/kw-hr)
	Initial	Final						
179.40001	7.70	10.84	1.6698	41.1	258	0.1910	-0.35784	2.30
179.40002	10.59	11.91	1.9665	40.8	33.5	0.3438	-0.07398	1.47
179.40003	17.54	17.46	1.8350	29.7	33.5	0.7173	+0.18568	2.46
179.40004	7.51	10.22	1.8913	40.5	126	0.4422	-0.30046	1.51
179.40005	14.51	15.44	1.5941	41.1	72	0.4911	+0.019606	2.08

*The thermal power output of all cores is normalized to 1338 Mw; the fuel cycle analyzed is a U^{238} - Pu^{239} cycle.

a. Conclusions of the Study - The results presented here complete the study. The following conclusions are drawn: the heat transfer results show that carbide fuels can give a significant increase in over-all core

power density over oxide fuels for equivalent rod sizes. It is possible to take advantage of the higher permissible thermal rating of carbide fuels by going to larger diameter rods. This results in a lower fuel cycle cost for carbide over oxide fueled cores. The comparison gives 2.08 mills for the best oxide price as against 1.51 mills for one of the lower carbide costs.

An interesting result is that there appears to be no compelling incentive, from a purely economic standpoint, to go to higher power densities and higher flow rates for the carbide systems, since increased core operating costs are about balanced by a larger fraction of cheap power from the blanket. For example, a carbide core with rod diameter = 0.50 in. gives a fuel cycle cost of 1.51 mills/kw-hr, whereas a core with rod diameter = 0.250 gives a cost of 1.47 mills/kw-hr. On the other hand, with the oxide systems considered here, there is an incentive to go to higher power density and smaller rod diameters, e.g., the cost for an oxide core with rod diameter = 0.250 in. is 2.30 mills/kw-hr whereas a rod diameter of 0.125 gives 2.08 mills/kw-hr.

2. Isotopic Power Sources

An exploratory study to see if it might be profitable to recover certain higher isotopes produced in a fast power reactor has been concluded. The particular isotopes considered were the alpha emitters Pu^{238} and Cm^{242} which are useful as power sources. It was found that optimum production rates would be obtained when Np^{237} and Am^{241} were chemically separated from recycled fast reactor fuel and then used as target material in a thermal reactor to obtain Pu^{238} and Cm^{242} in a second irradiation. Production rates for the target isotopes were calculated (see Progress Report, February 1962, ANL-6525, p. 10) for the Fermi reactor using a U^{235} and a Pu^{239} fuel cycle. Without considering the additional costs of the second irradiation, the values of Np^{237} and Am^{241} as by-products from the fast reactor were estimated. To make a 1 mill/kw-hr reduction in reactor power costs, the profit (sale price minus separation cost) for Np^{237} and Am^{241} must be \$500/gm and \$1000/gm, respectively. Addition of separation costs would then make the required sale price roughly \$1000/gm for Np^{237} and \$2500/gm for Am^{241} . Since these figures are higher than current market prices, the fast reactor cannot compete with existing production sources.

3. ZPR-III

In this period Assembly 38 was studied, this being a spherical system of the same composition as the cylindrical Assembly 24 ($L/D = 0.927$), i.e., 7.6 v/o U^{235} , 72.9 v/o U^{238} , and 9.4 v/o stainless steel. The principal object was to obtain the critical mass of the sphere and thus to derive an experimental value of the Shape Factor appropriate to Assembly 24. This could then be compared with the calculated value given by Loewenstein and Main in ANL-6403.

A limited number of fission chamber and central reactivity measurements were also made.

a. Critical Mass - Assembly 24 was critical at a loading of 460.7 kg of U^{235} , and the Shape Factor calculated in ANL-6403 was 0.985 so that a spherical critical mass of 454 kg U^{235} was expected.

In order to minimize differences between Assemblies 24 and 38, which could introduce ambiguities in the experimental Shape Factor, the exact fuel and diluent arrangements used in Assembly 24 were used as far as possible in Assembly 38. The spherical core was designed by computing the mass of U^{235} that should be present in each 1 in.-thick layer of the core at the assumed critical mass of 454 kg and then terminating the fuel columns so that in any 1 in.-thick layer the appropriate mass was present and the outer boundaries of the fuel lay on a circle. For layers at and near the horizontal "equator" of the sphere the adjacent 1 in.-thick layers differed very little in U^{235} content, and as the minimum variation in U^{235} column length is $\frac{1}{2}$ in. the design requirements could be met using only 2 in.-high fuel pieces. As the "polar caps" were approached, however, it was necessary to use 1 in.-high fuel pieces.

Using this technique each 1 in.-thick layer was designed so that it contained within 0.1% of the planned mass of U^{235} .

However, it was found that the assembly became critical at a loading significantly less than the planned loading of 454 kg and consequently this critical core resembled a slightly flattened sphere, the main fuel omissions being at the polar caps. It was therefore necessary to determine the edge worth of fuel at a number of positions in order to evaluate the critical mass of a true sphere.

b. Edge Worth Measurements and Spherical Critical Mass - The edge worth of U^{235} was measured at four positions, the results being listed below:

Position	Worth (Inh/kg)
End of drawers, equator	14.5
Side drawers, equator	13.1
Polar cap	14.1
"Latitude 45°"	13.6

It can be seen that the edge worth of fuel does not vary greatly, which is to be expected as the assembly is almost spherical. There is, however, a variation of 0.7 about the mean value of 13.8 lh/kg and this must be reflected in an uncertainty in the critical mass of a true sphere.

The assembly was first critical at a fuel loading of 428.85 kg U^{235} with control rod #10 withdrawn 5.85 in. This is 54 lh supercritical with the control rod completely inserted.

The corrected critical mass of the sphere is estimated to be 424 ± 1 kg U^{235} and thus the apparent experiment Shape Factor for Assembly 24 is $424 \pm 1.0/460.7$, i.e., 0.920 ± 0.002 . It should be noted that this figure does not take into account any systematic differences between Assemblies 24 and 38 such as in heterogeneity or edge-roughness effects.

c. Fission Chamber Measurements - A number of absolute fission chamber measurements were made at the core center, the count rates being corrected for the presence of isotopes other than the principal using some of the data of Assembly 24. The resulting fission ratios are presented below. In all cases the errors derived from counting statistics are about 1%.

Table II. Assembly 38: Central Fission Ratios

Exp	Fuel Arrangement	Chamber	Ratio
1	$2 \times \frac{1}{16}$ in. plates symmetrical in center drawers	Kirn	$\frac{Pu^{239}}{U^{235}}$ 1.17
2	$2 \times \frac{1}{16}$ in. plates symmetrical in center drawers	Kirn	$\frac{U^{238}}{U^{235}}$.0306
3	$2 \times \frac{1}{16}$ in. plates symmetrical in center drawers	Thick Wall Gas Flow	$\frac{U^{238}}{U^{235}}$.0312
4	$2 \times \frac{1}{16}$ in. plates symmetrical in center drawers	Thin Wall Gas Flow	$\frac{U^{238}}{U^{235}}$.0335
5	$2 \times \frac{1}{16}$ in. plates at side of center drawers	Thin Wall	$\frac{U^{238}}{U^{235}}$.0352
6	$2 \times \frac{1}{16}$ in. plates at side of center drawers	Kirn	$\frac{U^{238}}{U^{235}}$.0318
7	$4 \times \frac{1}{16}$ in. plates in center drawers: 9 central drawers de-bunched	Kirn	$\frac{U^{238}}{U^{235}}$.0309
8	$2 \times \frac{1}{8}$ in. plates in center drawers: 9 central drawers de-bunched	Kirn	$\frac{U^{238}}{U^{235}}$.0307

For comparison, the Pu^{239}/U^{235} and U^{238}/U^{235} ratios given for Assembly 24 are respectively $1.14 \pm .01$ and $.0284 \pm .0003$. These are probably comparable with experiments 1 and 2 of the above table.

The inferences are (a) that the repeatability of measurements (assuming that the central ratios are the same for 24 and 38) is appreciably

worse than the statistical errors, (b) the U^{238}/U^{235} ratio is dependent on chamber wall thickness, and (c) the U^{238}/U^{235} ratio is generally insensitive to fuel arrangement, but experiments 5 and 6 indicate about a 5% change with a slightly asymmetric loading.

d. Central Reactivity Measurements - The central reactivities of U^{233} , U^{235} , Pu^{239} , S, and Th were measured in Assembly 38 and found to be in reasonable agreement with those measured in Assembly 24. In addition, the worth of tungsten was measured in Assembly 38 and found to be -14.8 ± 0.3 lh/kg.

4. ZPR-VI and ZPR-IX

a. Building - Sealing of the interior walls of Cells Nos. 4 and 5 in Building 315 with Carboline 305, a phenolic resin base paint has been completed. Pressure testing of the cells at 10 psig indicated that the air leakage rate had been reduced to about 8% of the cell volume per 24 hours for Cell No. 4 and 6% per 24 hours for Cell No. 5. The main leaks appeared to be around the personnel door seals and the ventilation penetrations. Epoxy resin is being used to fill the cracks around the penetrations.

b. Bed and Table Assembly - Assembly of the stainless steel matrix and the side supports on the bed and table assembly for the ZPR-VI facility has been started.

Fifty square aluminum tubes for ZPR-IX received from the supplier were found to be substandard. It may be possible to salvage the tubes by reworking them.

c. Control Console and Instrumentation - Testing of the instrumentation on the ZPR-VI and ZPR-IX control consoles has been completed. Wiring on the bed, table assembly, and other components inside of the cell has been started.

d. Miscellaneous - The loading platforms for both facilities have been installed in the two cells.

Mechanisms for holding a TV camera are being installed in the two cells.

Microswitches are being installed on all 24-in. butterfly valves to indicate the open and closed position.

Electronic components for the Rossi- α measurement experiment have been ordered from the Electronics Division and are under construction.

5. JUGGERNAUT Critical Experiments

a. Fuel Loading - The JUGGERNAUT is presently loaded with 240 fuel plates containing a total of 4040 gm of U^{235} . This maximum possible loading would provide an excess reactivity of 5.0% $\Delta k/k$ for a cold, clean reactor. The central graphite plug in the internal thermal column as well as the beam hole plugs have been removed to reduce the excess reactivity to 3.3% $\Delta k/k$. When it becomes necessary to add fuel to compensate for samarium buildup and fuel burnup, individual fuel assemblies will be removed and replaced by assemblies with a slightly higher U^{235} content (240 gm vs 202 gm).

b. Delayed Neutron Fraction Measurement - A boron-substitution experiment to determine the effective delayed neutron fraction in the JUGGERNAUT reactor has been partially completed. The absorption cross section of the boron plates in relation to fuel plates will be determined.

c. Worth of Moderator - The measured worth of the H_2O moderator is approximately 20% $\Delta k/k$. The worth of the top reflector is 0.3% $\Delta k/k$. The measured worth of H_2O to voids in the coolant lines of the lead thermal shields is -0.045% $\Delta k/k$; therefore, turning off the shield coolant pumps results in a positive period of 170 sec.

d. Power Calibration - A preliminary thermal neutron flux calibration was made at a power level of 8 watts. Within experimental error, the measurements agree with a power calibration at 100 kw obtained by measuring the temperature rise across the core. The ratio of the maximum thermal flux in the internal thermal column to the average core thermal flux is 2.5, compared to a calculated ratio of 2.32. At 250 kw the average thermal flux in the core will be $1.6 \times 10^{12} \text{ n}/(\text{cm}^2)(\text{sec})$ and the maximum thermal flux available will be $4.0 \times 10^{12} \text{ n}/(\text{cm}^2)(\text{sec})$.

B. EBR-I

1. Mark III Operation

A capsule irradiation run, performed in connection with irradiation damage studies being carried out by the Reactor Engineering Division, was completed this month after about 94 mwh of time on the reactor. Operation was at low inlet temperature and full power and marks the first extended run for some time. In the intervening period a new source had been installed in the reactor (January, 1962) and modifications had been made to the prompt scram circuits to make them entirely fail-safe on loss of power. The current run was of a routine nature; no problems were encountered, and the samples have been removed from the reactor.

2. Fabrication of Core IV Fuel

The EBR-I Core IV fuel loading is being fabricated by the Laboratory in the Fuel Fabrication Facility, Building 350. The loading will consist of: (a) 420 fuel rods, (b) 120 blanket rods, (c) 10 fuel thermocouple rods, and (d) 5 blanket thermocouple rods.

All of the 120 blanket rods have been completed, inspected and found to be acceptable. Of the 420 fuel rods needed, 358 have been completed and have passed the inspection tests. Completion of fabrication of the fuel rods has been held up many months pending receipt of additional Zircaloy-2 tubing. This has now been received and inspected and the last of the fuel rods are being loaded.

To complete the 15 thermocouple rods, 0.080 in. O.D. Zircaloy-2 instrument tubing is needed. Efforts to procure this item commercially were unsuccessful. A program has been under way since February, 1961 to produce this tubing with ANL techniques and equipment. One lot of 0.080 in. O.D. tubing for this application has been processed. Destructive and nondestructive evaluation of this material is under way. Sufficient instrument tubing may be made available from this lot of material to complete the Core IV loading. A backup effort is under way processing additional 0.080 in. tubing to provide replacements for any tubes that may be rejected for reactor service after nondestructive evaluation.

It is estimated that the Core IV loading will be completed by about the first of May, 1962.

The Metallurgy Division is providing components for breeding gain experiments to be performed in Core IV of EBR-I. Of 26 components required, stock for 23 is on hand. Stainless steel tubing stock for the three remaining items, Mark III and Mark IV size thimbles and special Mark III size blanket rod jackets, is on order from commercial suppliers.

Two methods have been developed for the jacketing of the 0.150 in. diameter by 0.005 in. thick plutonium disc samples in zirconium or aluminum foils. Figure 4 shows the tungsten inert gas shielded edge welding of 0.005 in. thick by 0.200 in. diameter discs. The heat affected zone is to extend less than 15 mils from the edge. Figure 5 shows the edge weld produced when 3 mil aluminum foils are punched and pressure edge welded, using a dual purpose die. Tooling for both methods is being readied for installation in a hood.

250X

HF-AgNO₃ Etch

Figure 4. Tungsten-inert-gas Edge Weld on 0.200 in. Diameter x 0.005 in. Thick Zircaloy-3 Breeding Gain Specimen Cover Foils.

Figure 5. Pressure Cold Weld on 0.200 in. Diameter x 0.005 in. Thick Aluminum Cover Foils for Breeding Gain Specimens. Specimen Sectioned before Trimming.

C. EBR-II

1. Reactor Plant

During the month, the parts of the fuel gripper, holddown, and transfer arm that were damaged during the initial dry heat tests were replaced. Each unit was checked out in operation, including the

establishment of force limit settings for both "up" and "down" motions. Subsequently, a complete fuel-handling cycle was checked out to ascertain proper alignment of the units at all transfer points before restarting the dry heat test.

Progress on the second test is satisfactory and no problems have developed. The tank is now at a temperature of approximately 300°F. It is anticipated that the high temperature fuel handling system testing and structural deflection measurements will be completed shortly and installation work being performed inside of the primary tank preparatory to filling the tank with sodium will be resumed.

Two mechanical design changes to the fuel transfer arm are being made. As reported in ANL-6525 (Progress Report, February, 1962), the transfer arm end socket was modified to assure vertical alignment of the suspended subassembly irrespective of a possible deviation in the horizontal attitude of the transfer arm. The modification resulted in a substitution of a spherical-seat pocket for a flat-seat pocket which suspends the subassembly as shown in Figure 6.

Figure 6. EBR-II Transfer Arm End Socket Fitting

The modification also included a change in material from Type 304 stainless steel to Inconel X to provide a hard surface for the spherical seat to accommodate the imposed sliding loads.

It was also noted in the last progress report that the straightened transfer arm shaft required a subsequent heat treatment (for stress relief). Because the lower section of the shaft, in which the bend occurred, is fabricated of Type 304 stainless steel and the upper section is made of 17-4 PH steel (see Figure 7), a compromise on the most desirable heat treatment temperature for the shaft as a whole was required.

Figure 7. EBR-II Transfer Arm

The entire assembly was stress relieved at 900°F, this temperature being selected primarily to avoid sensitization of the stainless steel. This is not the most favorable heat treatment for the 17-4 PH; 1100°F is preferred, and results in a lower hardness and increased ductility.

Difficulties have been encountered recently with 17-4 PH steel parts in water-cooled reactors. Although no comparable difficulty has arisen with this material in sodium-cooled reactors, prudence suggests that any special consideration possible should be given to its use in any reactor at this time. Accordingly, the use of this material in the fuel transfer arm shaft has been thoroughly reviewed and additional precautionary steps taken as described below.

Selection of 17-4 PH steel for the upper portion of the shaft originally was made primarily to provide hardened surfaces at certain bearing and seal locations. After the heat treatment referred to above, hardness measurements were made along the entire length of the 17-4 PH section of the shaft and found to vary over a range of hardness from about R_C 43 to 47. This is consistent with the anticipated results (R_C = 40 to 45 minimum) from the 900°F heat treatment. Because stresses in the shaft are very low and expected impact loadings are negligible, the decreased ductility is considered acceptable. In addition, since this material is susceptible to surface cracking during heat treatment, the entire 17-4 PH section of the shaft was given a thorough Zyglo inspection and no cracks were detected.

The transfer arm assembly consists of an "L" shaped shaft and arm supported by bearings at the top and guided radially along its vertical length. Approximately midway along the length of the 17-4 PH portion a backup support is provided by a support ring which makes contact with a shoulder if the assembly is dropped about one inch. (This support ring is utilized during the normal assembly procedure to support the shaft during the assembly of the upper bearing housing.) The transition between the 17-4 PH section and the Type 304 stainless steel section occurs below this point of support. The smallest section of the 17-4 shaft below this support point is $3\frac{5}{16}$ in. OD with better than a 1-in. wall thickness. The normal stresses in this section, therefore, are very low (less than 2500 psi). It is almost inconceivable that this section of the shaft would fail; nevertheless, a second point of support below the material transition point is being added. As shown in Figure 7, this consists of a new support ring fastened to the Type 304 stainless steel section of the shaft and a new support shoulder fastened to the lower guide bearing mount. This will serve to support the entire assembly (after dropping about 1 in.) should failure occur above this point of support.

Failure of the transfer arm would not represent a reactor hazard. It is radially displaced from the reactor and could not fall into the reactor or directly damage the reactor. The major consideration relates to the

secondary damage that could be caused and difficulties which might be encountered in making repairs. Of particular concern, of course, would be dropping the unit itself (which weighs some 800 lb) into the bottom of the primary tank and posing a difficult retrieval problem. With the points of support now provided on the shaft, a failure in the shaft (or support in general) would result only in the unit dropping approximately one inch from its normal operating level which would cause no secondary damage. However, it would still result in a very sizeable maintenance because the entire shield plug and shaft assembly would have to be removed for replacement of the shaft.

The conclusions reached from this review are that: (1) the probability of failure of the transfer arm shaft is extremely small; (2) the addition of a second point of support to limit damage in the remote event of failure can be accomplished simply and inexpensively; and (3) employment of the shaft, particularly in view of the new support point to be added, is fully warranted.

2. Power Plant

The 20,000 kw turbine-generator and its auxiliaries were operated with 175 lb steam from the auxiliary boilers for training purposes. The turbine-generator was also synchronized for a brief period to the NRTS loop for the first time. With generator protective devices operational, the unit was kept on the line and loaded to a maximum output of 800 kw.

As part of the training program, operating personnel participated in the startup procedures, synchronization of the generator, and operation of steam plant equipment on a continuing basis. Thus, all prospective power plant personnel have had the opportunity to familiarize themselves with synchronizing procedure.

The hydrogen system of the generator was also checked out by purging the casing of air with CO_2 and then filling it with hydrogen. After two days with hydrogen in the generator the system appeared to be tight, purity meters were calibrated, and the hydrogen was again purged with CO_2 . The generator is now in a stand-by condition with air in the casing.

Although miscellaneous minor checkouts and adjustments of components remain to be performed (some of them awaiting high-pressure, high-temperature steam), the power plant is essentially ready to go on the line.

3. Sodium Boiler Plant

All welding on the steam generator and secondary sodium system piping has been completed up to the terminal point. With testing of the pipe

and weld deposit material, cleaning of the secondary piping system, and installation of the repaired evaporator, the major portion of the corrective work has been accomplished.

As per specifications, the 175 psig pneumatic pressure test of the secondary sodium piping system has been accomplished, and helium leak testing of this system has been started. During the helium leak test a crack was detected in a 2-in. sodium-filling line. The damaged section of the pipe was replaced.

The helium leak test of the secondary sodium system is in progress. Prior to the test, all ring joint flanges were thoroughly inspected for surface finish condition and composition. All ring joint flanges and rings are required to be of Type 304 stainless steel. During the inspection a carbon steel ring was found and replaced.

4. Fuel Cycle Facility

a. Installation - Equipment for the EBR-II Fuel Cycle Facility is being installed by the J. F. Pritchard Company under the direct supervision of Argonne National Laboratory personnel. Electrical work is being done to provide services for equipment for the refabrication of fuel pins and for the transfer cell. A degassing furnace is being installed. This furnace will be used to degas process equipment items before they are placed in the Argon Cell. The installation of the pillar jib crane on the roof of the Fuel Cycle Facility is continuing. The crane will be used to move material in and out of the Argon and Air Cells through the shielded roof hatches.

Specifications for the off-gas monitoring system have been prepared and issued for bids. The system will continuously monitor the total activity in the gas discharged from the off-gas holdup tank. Gases released in melt refining and in the oxidation step of the skull reclamation process will be stored in the tank until meteorological conditions are favorable for the discharge of the gases through a 200-ft high stack. The rate of discharge of the off-gas will be established by the total activity in the off-gas and by the prevailing meteorological conditions. It is estimated that the total activity in the off-gas will vary from 10 to 10,000 curies per cubic meter.

Bus bar collector assemblies have been modified to improve their performance. The main defects needing correction were the unsatisfactory functioning of the spring-loaded contact brushes and the tendency of the guide pins to bind.

In the way of corrective work the argon cell liner has been cleaned on the ceiling and wall portions, with the floor left for a later date.

Argon cell cranes and manipulators are being dismantled, cleaned, and reassembled with new bearings being installed where required. Refinishing of the lower flange of the 6-ft diameter transfer lock was accomplished to conform with specifications.

Meanwhile, further discrepancies were noted this month. The hanging crane rail in the argon cell was installed too high and will have to be corrected. The bottom of the argon filter box is covered with a slimy mold, and the zinc coating therein has corroded. The filter retainers in the argon filter box are not installed in accordance with contract drawings. Further, the argon compressor microphones are not properly mounted. The Hastings vacuum gauge on the transfer lock between the air and argon cells has been found to have a defective meter, and the argon purification system valve actuator gland is leaking badly. These items will be corrected in turn.

b. Process and Equipment Development - Measurements have been made of temperatures reached in an air-cooled, simulated, standard sub-assembly of 91 discharged clad fuel pins. Heat was supplied electrically at rates of 1 to 11 watts per pin. The maximum level was almost equal to that calculated for two percent burnup pins about two weeks out of the reactor. Data from these tests is now being analyzed and is significant to the resolution of process and transfer problems.

The application of non-destructive test methods to the detection of defects in zirconium oxide crucibles used in melt-refining is being studied. The use of a simple vibration tester to record behavior upon impact was unsuccessful in denoting defects. Other potentially useful methods are being examined.

The first analysis of an irradiated fuel pin sample was undertaken in the remote analytical facilities. Planned analyses were carried out successfully with only minor problems; therefore, the general adequacy of the facilities and techniques is indicated.

NRRG-159 radiation-resistant grease (a product of the Standard Oil Company of California) has been found satisfactory for use on bearings in the Fuel Cycle Facility. The grease was tested by using it to lubricate a roller bearing which was then alternately irradiated and run under an 800-lb axial load at 40 rpm. The bearing operated satisfactorily after a total running time of 300 hours and an accumulated gamma dose of 4×10^9 rad. This performance is equivalent to four years of operating time in the Fuel Cycle Facility.

c. Development of Remote Control Methods and Equipment for Fuel Fabrication - Work has continued on the preparation of installation drawings, job instructions and operational manuals for the EBR-II fuel

recycle equipment. The as-built installation instructions and drawings for the two-stage argon compressors were received from the Ingersol-Rand Company on March 19th. These made minor revisions necessary in the Argonne National Laboratory prepared compressor installation drawings. The work description, job instruction and the installation packet on the compressed argon system are in final preparation.

Design work was completed in the subcell and for the argon cell auxiliary and control equipment. Except for the remote weighing balance, all design work was completed for the argon cell refabrication equipment. Completion of the melt preparation balance requires further drawings from the manufacturer of this equipment. Only the bond and level inspection equipment and subassembly construction equipment drawings must be completed in the air cell.

Both injection casting furnaces have been assembled and have been checked out. Some minor modifications were required. These have been made and the drawings have been revised to the as-built condition.

It has been necessary to relap all valve seats and surfaces in the leak detectors. Adiprene O-rings were not successful. These were replaced by quadrings made in Adiprene. A technique was developed for vapor depositing a fairly heavy aluminum surface on the valve face seals. This has replaced lead seals which were ineffective.

d. Fabrication of Supplemental EBR-II Type Fuel Elements for Testing - The manufacture of the 2000 supplemental depleted fuel rods for the EBR-II Fuel Cycle Facility startup has been completed. These were jacketed and sodium bonded by the methods employed for Core I. An additional 355 inspected bare fuel pins, 100 raw and unprocessed castings and approximately 530 Vycor molds with castings in place have been prepared for test of the various machine components.

5. Fuel Development

a. Irradiation of Refractory Alloy-clad U-5 w/o Fs Alloy - As described in an earlier Progress Report (ANL-6473, November, 1961), the post-irradiation examination of 12 refractory alloy-clad U-5 w/o Fs alloy specimens revealed that, in most of the specimens, fuel had extruded through 0.0625 in. diameter holes that had been drilled in the cladding for insertion of the NaK. The specimens had been under irradiation for periods ranging up to 13.5 months at central temperatures near 700°C. Burnup analyses obtained recently on specimens previously irradiated in the same location in CP-5 indicate that the maximum burnup achieved was near 4.1 a/o, rather than the maximum burnup of 2.5 a/o which was estimated in ANL-6473.

The internal pressures which developed in the clad pins during irradiation are of considerable interest, since the strength required in the cladding is largely a function of the pressure exerted by the fuel. Two of the specimens which were jacketed with Inconel-X had been provided with a 0.001 in. thick zirconium foil which separated the fuel from the cladding. Although this foil completely covered the holes, the restraint offered by the zirconium was insufficient to restrain the swelling of the fuel, as evidenced by the fuel protruding from the holes.

Hydrostatic rupture tests were conducted at room temperature on six zirconium foils in an attempt to estimate the pressures involved. Circular discs were cut from the 0.001 in. zirconium foil stock which had been used for the specimens. A backup plate was made having a circular hole of identical dimension to that of the specimen jackets. The foils ruptured hydrostatically over a range of 1200 to 1850 psi.

The temperature of the specimen cladding during irradiation was calculated from the thermocouples monitoring the NaK temperature adjacent to the specimen. At the calculated irradiation temperature the zirconium should have retained approximately 48% of its original tensile strength. The pressure required to rupture the foil during irradiation is therefore estimated to be near 700 psi.

This would have resulted in a hoop stress in the cladding of 6400 psi, which is far below the long term rupture strength of Inconel-X. Since the Inconel-X cladding ultimately did rupture, it is evident that the internal pressure which developed in the fuel exceeded 700 psi by a wide margin.

b. Properties of Uranium-Plutonium-Fissium Alloys - A study of the properties of uranium-plutonium-fissium alloys was initiated to obtain certain physical and mechanical properties of these alloys. The Core II reference composition, U-20 w/o Pu-10 w/o Fs, has been well evaluated. A few properties such as heat capacity, hot strength, and thermal cycling stability are expected to be determined. Little information has been obtained on high zirconium fissium alloys.

The alloy from the pyroprocessing cycle can be varied in plutonium content and in fissium content. The effect of these variations on various properties of the fuel alloy must be determined. Any adverse effects should be eliminated by adjustment of the pyroprocessing cycle. Thermal conductivity would be expected to vary considerably with composition and this property will be determined first. Five different alloys were cast with varying proportions of uranium, plutonium and fissium.

6. Process Development

a. Melt Refining Process Technology - Two small-scale experiments, each using about seven grams of highly irradiated EBR-II prototype fuel alloy, were carried out to evaluate the behavior of volatile fission products in the melt refining process. Each sample was melted in a zirconia crucible under an argon atmosphere and held for one hour at 1400°C. A Fiberfrax trap was used for the collection of the condensable fission products. The temperature gradient of the trap was between 420° and 1000°C. This is similar to the temperature gradient of the trap in the plant-scale EBR-II melt refining furnace. Information concerning the location of the fission products in terms of temperature zones along the Fiberfrax trap and the retention of the fission products by the Fiberfrax will become available upon completion of the analytical work which is now in progress.

b. Skull Reclamation Process - Studies of phase separation methods were continued (see Progress Report, February 1962, page 27). Separation of molten metal from solidified flux, a separation required after the noble metal extraction step of the skull reclamation process, was made without difficulty. In each of three experiments, 5 kg of cadmium, 1.5 kg of halide flux, and 0.375 kg of skull oxide were used. The cadmium was used as a stand-in for zinc to permit the use of stainless steel equipment. In each run, about 90 percent of the metal was transferred.

The separation of molten metal from molten flux was also studied. This separation is required after the skull oxide reduction step. Because both phases are molten, this is one of the more difficult separations encountered in the skull reclamation process. Four separations of this type were made during the month. Cadmium was again used in the place of zinc. The scale of these experiments was similar to the scale of the experiments with molten metal and solidified flux. In all runs, a clean separation of phases was obtained. The unit used for the separation is one in which it is unnecessary to locate the metal-flux interface. With this method, it is believed that transfers of over 90 percent can be readily achieved in plant operation.

A number of the molten metal and flux transfers were made to receivers placed in the open, a procedure which simulates possible plant operation. Bonded fiberglass filter units placed over the receivers have proved very effective in trapping the fumes from the transferred materials.

A phase separation common to both the skull reclamation and blanket processes is the removal of a 50 percent magnesium-zinc supernatant solution from precipitated uranium metal. On a 500-gram uranium scale, very good removals (93.5 to 97.5 percent) of the supernatant phase have been achieved without difficulty. In most of the runs, the entrainment of uranium metal was negligible.

Testing of the prototype skull oxidation furnace is continuing (see Progress Report, February 1962, ANL-6525, page 24). A run was performed to determine what effect nitridation of the skull material would have on its subsequent oxidation. About 50 percent of the skull, which contained about 420 grams of uranium, was first converted to the nitride. The skull was then oxidized using an oxygen feed rate equivalent to a conversion of about two grams of uranium per minute. The reaction was completed in five hours. This is similar to the time required for the oxidation of skulls of equal weight which have not been nitrided. No unoxidized material was found in the product and the general quality of the product appeared to be satisfactory, inasmuch as only a few agglomerates were present.

The use of 60-cycle frequency power to induce heating and mixing of the reactants encountered in the skull reclamation process is being investigated. The reduction of skull oxide on a large scale (2 kg skull oxide, 39 kg zinc, 3 kg magnesium, and 5.6 kg flux) was attempted. Good stirring was achieved, but the maximum temperature attained (700°C) was considerably lower than the desired process temperature of 800°C. Samples of the metal phase are being analyzed to determine the extent of reduction. The induction coil is being modified to permit the attainment of the desired reaction temperature.

Mechanical mixing experiments are underway to determine the controlling variables in the mixing of two liquid phases and a solid phase. The information is needed for the rational design of equipment for the skull reclamation process. Some of the variables that are being studied are impeller type and diameter, impeller speed, shaft offset from center of container, depth of the two liquid phases, shape of the container bottom, and the order of addition of phases.

c. Plutonium Recovery Process - Work on the removal of rare earth fission products from uranium-plutonium fuels by selective extraction into a molten halide flux is being continued (see Progress Report, February 1962, ANL-6525, page 28). Distribution coefficients of rare earths, uranium, and plutonium at 800°C are being determined for a two-phase system consisting of liquid magnesium-zinc alloy and molten magnesium chloride. As the magnesium concentration in the metal phase is increased, the distribution coefficients decrease sharply, reaching a minimum at about ten weight percent magnesium, then rise gradually as the magnesium content is increased to 100 percent. The distribution coefficients of several elements are listed in Table III, together with the calculated separation factors.

Table III. Distribution Coefficients and Separation Factors for Rare Earths, Uranium, and Plutonium

Flux: magnesium chloride
 Metal: zinc - 10 w/o magnesium
 Temp: 800°C

Metal, M	Distribution Coefficient, $K_d = \left(\frac{\text{w/o M in flux}}{\text{w/o M in Zn-10 w/o Mg}} \right)$	Separation	Separation
		Factor, $\frac{K_d(M)}{K_d(Pu)}$	Factor, $\frac{K_d(M)}{K_d(U)}$
Yttrium	1.9	127	950
Cerium	0.88	59	440
Praseodymium	0.6	40	300
Plutonium	1.5×10^{-2}	-	7.5
Uranium	2×10^{-3}	0.133	-

D. Fast Reactor Test Facility (FARET)

The sodium temperature coefficient for various FARET systems were investigated. The calculations were performed for plutonium-enriched uranium oxide systems and for uranium carbide systems. The investigations included a study on the effects of various structural materials, in particular niobium and stainless steel, reflector thicknesses, size, and geometry on the reactivity coefficient.

A preliminary neutron inventory indicates that niobium contributes a moderate fraction of the positive reactivity feedback. Stainless steel, although its effect is negative, contributes an almost negligible fraction. These effects are understood as follows: A decrease in sodium density results in a hardened spectrum as well as an increase in neutron leakage. The niobium capture cross section, in the neutron energy range considered, is not only much higher than that for steel, but its magnitude decreases appreciably with increasing neutron energy, in particular it decreases more rapidly than does the fuel absorption cross section. The stainless steel capture cross section on the other hand varies only slightly with energy so that its effect although small results in a negative feedback. It should be noted here that the experimentally determined cross sections for the stainless steel components are rather sparse and the effect above should be stated with some qualifications. Further detailed analyses are continuing.

III. GENERAL REACTOR TECHNOLOGY

A. Applied Reactor Physics

1. 3.0-Mev Van de Graaff

a. Experimental Studies of Neutron Scattering

(1) Elastic Neutron Scattering from Iron - The pulse shape discrimination detector was used to measure the angular distribution of neutrons elastically scattered from iron. The data were taken in 8-keV energy increments with a neutron energy spread of 23 keV covering the range from 700 to 1100 keV. Measurements have been completed for the laboratory system scattering angles of 25, 55, 90, 125, and 160 degrees. The absolute normalization of the differential cross sections was obtained by repeatedly comparing the scattering yield from iron to the scattering yield from carbon. Both the iron and carbon scattering samples were cylinders 1.27 cm diameter by 2.54 cm long. The results for scattering angles of 25, 90, and 160 degrees are shown in Figure 8. The structure and its dependence on the scattering angles is clearly evident. These data are as yet uncorrected for multiple scattering. A computer program for multiple scattering correction and for a least squares Legendre polynomial fit is available. The data described here and data from measurements in progress will be analyzed to determine the integrated elastic scattering cross sections and the Legendre polynomial coefficients for the angular distributions.

Figure 8. Elastic Neutron Scattering from Iron

(2) Fast Neutron Scattering from Nb, Zr, Cu, and Zn - Utilizing time-of-flight techniques in contrast to the method described in (1) above, careful studies of both elastic and inelastic scattering from niobium, zirconium, copper, and zinc have been completed up to an incident neutron energy of 1.2 Mev. Typical results pertaining to zirconium and niobium are shown in Figures 9 and 10. These graphs display the respective differential elastic scattering cross sections in the form:

$$\frac{d\sigma}{d\Omega} = \frac{\sigma_t(\text{elastic})}{4\pi} \left(1 + \sum_{e=1}^4 W_e P_e \right)$$

All the data shown have been fully processed including Monte Carlo corrections for the effect of multiple scattered events within the samples. Inelastically scattered neutrons were observed from both niobium and copper. These measurements are now being processed.

Figure 9. Elastic Scattering from Zr

Figure 10. Elastic Scattering from Nb

The experimental studies described in (1) and (2) above deal with materials of considerable application in fast reactor cores. In addition, niobium and zirconium are characterized by nuclear properties making them particularly amenable to theoretical study. Such theoretical interpretation of these two nuclei is now in progress.

2. Argonne Thermal Source Reactor (ATSR)

a. Resonance Integral Measurements - In the calibration of the resonance integral measuring system, it was observed that the rod response was nonlinear (Progress Report, December, 1961, ANL-6485, p. 40). During these observations, it was noted that the noise characteristic of the rod movement appeared quite peculiar; the frequency seemed to change noticeably from time to time. This effect was traced to lateral movement of the rod and has been corrected.

Utilizing the automatic control system on the ATSR, curves of net autorod deflection per unit mass vs sample thickness have been obtained for gold, molybdenum, tungsten, graphite, and Teflon. The gold data will be used to provide a reference standard for the determination of the resonance integrals of molybdenum and tungsten. At present an attempt is being made to develop an analytical fit of the data in order to obtain the zero thickness extrapolation. The graphite and Teflon data will be used to obtain an estimate of scattering and moderation effects. In addition, data have been obtained for a single lead sample, and this too will be used in estimating scattering effects.

All samples have been oscillated for a sufficient time to reduce the error in the net deflection to $\leq 4\%$. Net deflections (deflection of sample plus holder minus deflection of holder) as small as 0.3% of full scale autorod worth have been obtained, but the accuracy is greatly reduced for those samples where deflections are less than 1% of full scale (1% full scale $\approx 1.51 \times 10^{-7} \Delta k/k$). Table IV lists the number of samples of each material and the approximate time required to obtain sufficient data to give the $\leq 4\%$ accuracy. In addition to the times given in the table, from one to three hours are required to attain the necessary stability in the equipment before data can be taken. This "stability time" appears to depend somewhat on outside weather conditions and possibly on several other undetermined factors.

Table IV. Oscillation Period for Deflection Sample Testing

<u>Sample Material</u>	<u>Number of Samples</u>	<u>Time Required (hr)</u>
Gold	14	80
Tungsten	5	19
Molybdenum	5	50
Graphite	4	5
Teflon	4	16

The infinite dilution resonance integral for zinc, tin, silver, and cobalt will be determined next. These metals were selected on the basis of discrepancies in, or the complete absence of, reported values available in a form convenient for measurement.

b. Flux Perturbation Measurements - The gamma counting characteristics of the equipment used for the flux perturbation experiment were improved by the use of a foilholder which is directly attached to the crystal cap, by stabilizing the line voltage, and by carefully readjusting the circuitry. The reproducibility of counting rates is now about 0.1%. In addition, the dead time of the system was measured with the two-source method and the amount of absorption of gamma rays within the foil itself was determined by a transmission experiment.

The flux perturbation measurements were continued with the improved equipment. Foils were activated in a monitored flux on the axis of the graphite block attached to the leakage face of ATSR. Foils of thicknesses between 7.5 and 500 microns were activated in one of three ways: (a) directly in the graphite; (b) in a cubic void of 20-cm edge length; or (c) in a small rectangular cadmium tube which served to give the flux a strong angular dependence. Foils were activated both perpendicular and parallel to the leakage face.

The flux perturbation caused by the foils activated directly in the graphite behaved as expected for a foil in an isotropic flux, in spite of a flux gradient of about $0.03 \phi/\text{cm}$ in the block. It followed essentially the function $[\frac{1}{2} - E_3(t \Sigma_a)]$, the outer flux perturbation in graphite being comparatively small. It made no difference whether the foil was inserted with its edge or with its face toward the reactor. In the first measurements in the cavity, these two foil positions yielded different results, the foil with the edge towards the reactor showing stronger self-shielding. In both positions the perturbation was weaker than observed with the foils imbedded directly in the graphite. This was shown to be mainly due to the influence of the foil holding device, through additional measurements made with a foilholder consisting of only a few aluminum wires. These later measurements did not reveal any significant influence of anisotropy in the flux. The measurements in the cadmium tube showed a dependence on foil thickness and angular positioning, and the self-shielding was generally stronger. However, these results are of limited interest, since they are influenced by peculiarities of the device used.

The accuracy of the measurements is now about 1%. As a next step, measurements in larger flux gradients close to the surface of the graphite block will be performed.

3. Fast Neutron Spectrum Measurements

In previous tests, a fast neutron spectrometer equipped with two solid-state charged particle detectors provided better than 2% energy resolution (Progress Report, January, 1962, ANL-6509, p. 30). This was effected by collimating the products from the $\text{Li}^6(n,\alpha)\text{H}^3$ reaction. The solid state neutron spectrometer has been reconstructed so that its volume is now reduced to a minimum. The neutron-sensing lithium fluoride has been coated directly onto one of the detector surfaces instead of being supported by a separate thin plastic film, consequently, the entire detector assembly is sturdier and smaller and no longer need be evacuated to operate properly. The Li^6F film deposited on the detector face is thicker than the film used for earlier tests, so that the resolution is now of the order of 3%. It is expected that the entire assembly will be inserted in the ZPR-III reactor where the first spectrum measurements will be made. The machine program which will correct the experimental points for the effects of line width is now being tested.

A fourfold decrease, from 400 to 100 kev, in the lower limit of the useful range of a hydrogen recoil counter for fast neutron spectroscopy was reported previously (Progress Report, December, 1961, ANL-6485, p. 42). This was accomplished by discriminating electronically between counter pulses and pulses produced by gamma radiation.

A measurement of the neutron energy spectrum of the ATSR in a converter-cave assembly has been made with this recoil counter. The actual spectrum is not of particular interest, but this measurement has given useful information regarding statistics to be expected and the energy range over which it is possible to obtain meaningful results. Reduction of the total recoil pulse height spectrum yielded a neutron energy spectrum with $\pm 10\%$ energy resolution from 50 kev to 1 Mev. The spectrum is roughly Maxwellian and peaks near 150 kev.

An improved converter location, having less "room-scattered" background and fewer gamma rays, would be desirable to determine whether better energy resolution and an even lower energy limit might be obtained. A position on the front of the large graphite thermal column on the ATSR leakage face is currently under test.

4. High Conversion Critical Experiment

Criticality was achieved in a water-reflected core having a tightly loaded central slab flanked by two end slabs having more reactive composition. The central slab contained 2650 UO_2 fuel elements spaced on a 1.18-cm center-to-center triangular lattice. There were 53 staggered rows of 50 elements each. The atomic ratio $\text{H}:\text{U}^{238}$ was 1:1. The fuel was 3 wt-% enriched UO_2 pellets, 0.935 cm diameter, and stacked to form a column

122 cm in length. The pellets are contained in 0.96-cm ID, 1.06-cm OD, Type 6061-T6 aluminum tubes. Each end slab contained 300 fuel elements arranged in six full rows separated by vacant rows (to gain reactivity). The total of 3250 elements contained 73 kg U^{235} and 2367 kg U^{238} .

Excess reactivity, ρ , for this loading was less than 0.02%. Addition of more fuel gained less than 0.01% ρ per element. Preliminary measurements of safety rod worth indicated that the required 1.5% safety margin was provided by the two most effective of the four blade rods in the reflector. Measurement of reaction rates and flux distributions are in progress.

It is also planned to measure the effect on reactivity of changing from aluminum to stainless steel fuel tubes. It is not yet clear whether the fuel supply will be adequate to obtain criticality with a single fuel zone of unit H: U^{238} atomic ratio. Inverse multiplication plots curve as the loading of the central portion progresses, each loading producing a less favorable prediction than did the preceding increment. The extrapolations, based on loadings of 2712 to 3189 elements, predict that for the stainless steel tubes the critical loading will fall between 4500 and 5000 elements. A total of only 5574 elements are available.

5. Theoretical Physics

a. Measurement of the Doppler Coefficient in Fast Critical Assemblies - The possible error in determining the Doppler coefficient for large fast reactors from measurements made on small heated fuel samples in a room temperature critical experiment was pointed out previously (Progress Report, October, 1961, ANL-6465). There is now a considerable interest in building heated critical experiments to make these measurements.

For accurate measurement of the Doppler coefficient, fuel should be heated in a spectrum characteristic of the heated fuel itself. There are two types of changes in the spectrum which occur on heating of fuel. One is a local change in the vicinity of a resonance which affects resonance self shielding, while the other is a gross change in the low energy spectrum because of changes in average fission and capture cross sections. The former effect is believed to be considerably more important than the latter. The error in the former effect that results when a heated sample is placed in a cold reactor was previously considered. The implicit assumption was made that the heated sample was infinitely thin and did not perturb the flux of the critical assembly, which was assumed homogeneous.

In practice, however, the situation is more involved and one must take account of the finite size of the samples and of the fact that

coolant and clad materials exert a partial blocking effect on neutrons which would otherwise pass uncollided from one fuel rod to another. Accurate evaluation of the effect of this blocking is important as it decreases the flux depression at a resonance and reduces the Doppler coefficient. If the degree of heterogeneity in the experiment is too great the blocking becomes complete and the rods are effectively isolated from one another as far as the local flux changes are concerned. Fuel rods one inch in diameter, for example, appear to be too large according to this criterion.

If one is willing to forgo experimental measurements of the low energy coolant and clad effects, one can do an experiment with a large hot fuel rod in a cold reactor without encountering the error previously referred to (Progress Report, October, 1961, ANL-6465), since uncollided neutrons from other rods would be blocked off. The small effect of the gross energy spectrum change could also be missed. The principal incentive for performing the heated critical is to try to evaluate the coolant and clad effects. It is desirable in this heated critical experiment to minimize heating of any material but fuel because of the difficulty of subtracting out expansion effects. The ideal arrangement is one in which fuel is heated, probably electrically, while the structure is cooled by gas or sodium.

b. Fuel Cycle Studies - The COST code used for fuel cycle cost studies in conjunction with the fuel cycle code CYCLE is being modified. The new version of COST will compute the differential effects on fuel cycle costs produced by variations of selected input parameters. A simplified input format is being incorporated so that the user may easily indicate which parameters are to be varied. The code will yield the detailed reference fuel cycle costs and the derivatives of the total cost with respect to changes in the selected input data.

The CYCLE production series has continued with studies of the influence of higher plutonium isotopes in the feed material, single batch instead of cyclic core operation, and the effect of higher and lower recovery factors for the fuel alloy. Work on the Th-U²³³ cycle problems mentioned previously (Progress Report, February, 1962, ANL-6525) has been deferred until the desirability of using ELMOE modified cross sections for these systems is resolved.

The next series of CYCLE production problems will study the effect of lower fission product removal, higher core burnup, and higher blanket burnup.

c. Feasibility of Pu²³⁹-U²³⁵-Fueled Fast Criticals to Predict Pu²³⁹-Fueled Assembly Dimensions - Calculations of bare homogeneous Pu²³⁹-U²³⁵-fueled fast criticals to estimate properties of Pu²³⁹-fueled assemblies were described in ANL-6525 (Progress Report, February, 1962).

Reflected homogeneous cores and reflected two-region cores are also being studied.

For the two-region cores the central region contains 5% Pu²³⁹ fuel by volume and the outer core region contains 5% U²³⁵ by volume. Core dilution consisting of the fertile, structural, and coolant materials is identical in the two regions. It is estimated that construction of a reflected two-region core having the ratio of central core-region volume to total core volume of 1:10 should decrease the calculated error for predicting the critical core size of a corresponding Pu²³⁹-fueled assembly by more than a factor of 10.

B. Reactor Fuels Development

1. Corrosion Studies

a. Aluminum Powder Product Tubing - Strong tubing manufactured by Armour Research Foundation from A288 alloy powder ball milled 72 hours was permanently removed from a 290°C corrosion test after a little more than two months of exposure. One portion of the tube circumference was longitudinally converted to oxide. The same tubes had shown extensive end swelling after about one month of exposure.

The following A288 alloy powder product tubing still is in satisfactory condition and on test at 290°C:

- (a) From as-atomized powder (ARF): 3 months
- (b) From ball-milled (48 hours) powder (ARF): 3 months
- (c) From as-atomized powder-finned tubing (Torrance Brass) 2 months.

b. Zirconium Alloys for Superheated Steam - Corrosion of selected samples in steam at 540°C and 600 psi is continuing (see February Progress Report) but samples have not been inspected during this period. Additional hydrogen analyses confirm the low hydrogen pickup of samples corroded at 650°C as compared with identical samples corroded at 540°C. It is planned to use X-ray diffraction, electron microscopy, and electron diffraction in an attempt to observe the relationship, if any, between corrosion film structure and hydrogen absorption.

c. Lightweight Alloy for Liquid Mercury - As reported in the December 1961 Progress Report, the addition of zirconium to mercury retards considerably the corrosion attack on titanium at 371°C. Recent tests have indicated that the corrosion resistance of titanium can be

improved by a factor of about 16 at 454°C by the addition of 500 ppm zirconium to the mercury. Under these conditions commercially pure titanium showed a weight loss of 2.62 mg/cm² in 14 days as compared to a weight loss of 42.90 mg/cm² in 14 days in pure liquid mercury. Approximately 1 to 10 w/o zirconium was detected on the surface of the samples from the zirconium-containing tests. However, no zirconium was found on uninhibited specimens. Tests using other additives are in progress.

2. Fabrication Development

a. Fabrication of Zirconium Alloy Tubing - Zirconium alloy tubing is a desirable structural and fuel jacketing material for use in reactors. Attempts are being made to develop techniques for fabrication of defect-free tubing in size ranges not commercially available. Work to date has been limited to development of techniques for fabrication of tubing smaller than 0.5 in. OD.

Zircaloy-2 tubing of 0.299 in. OD x 0.021 in. wall has been fabricated at ANL from ANL extrusions. This work was a backup effort for a commercial tubing order to provide fuel jackets for the EBR-I Core IV. The base tube as extruded was approximately 0.400 in. OD x 0.046 in. wall. Consumable electrode, vacuum arc melted Zircaloy-2 forgings were machined into billets and 145 were extruded into base tube. Approximately 90 tubes have been "core drawn" to size, annealed and are ready for eddy current inspection for wall defects. When the commercial tubing was received the ANL material was set aside. The belt sanded surface finish on the purchased tubing was similar to that obtained on the previous commercially procured production material. Preference was given to the commercial tubing because of this surface uniformity.

Processing of the ANL fabricated tubing will continue at a later date.

Considerable improvement in the "as drawn" tube surface condition has been noted on a number of these tubes. The parameters controlling the reproducibility of this "mirror" finish are under investigation. Previous Progress Reports have mentioned that it may be associated with the lubricating properties of a layer of red-grey material that forms between zirconium and copper during the initial production operations.

b. Preparation of Plutonium Foils - Certain techniques for preparing plutonium foils are being investigated in connection with the fabrication of a target for use in the cyclotron. Work is being done on the production of thin foils (0.006 in. thickness) and on methods of covering them with 0.002 in. thick aluminum foil and bonding them to water-cooled aluminum target holders.

Various bonding methods (roll bonding and hot press bonding using silicon, germanium, tin, magnesium powder, silver foil and silver powder) have been tried. Hot press bonding, using an interfacial layer of silicon, was selected as the best method. Figures 11a and 11b show metallographic results from the pressure bonding of aluminum and plutonium. Figure 12 shows the interfacial layer developed by silicon bonding the edges of the specimen.

100X

(a)

Unetched

(b)

Unetched

Figure 11. Photomicrographs Taken at Center and Edge of Plutonium Cyclotron Specimen Pressure Bonded in Aluminum.

100X

Unetched

Figure 12. Silicon Braze Aluminum Cover to Aluminum Target Holder.

As a result of these experiments, six assemblies consisting of a water-cooled target plate, plutonium foil and aluminum cover, have been assembled and pressure bonded. The cover foil after pressure bonding is approximately 7 mils thick. This thickness will be reduced to the required 2 mils by careful etching in sodium hydroxide solution. The final operation will be to machine the composite plates to the dimensions of the cyclotron target holder.

3. Ceramic Fuels

a. Preparation of Plutonium Carbide Fuels - A 200-gram plutonium carbide sample, pulverized and screened to -200 mesh, was manufactured for another laboratory. This material was prepared by melting plutonium metal in a graphite liner which was held at a temperature of 1560°C for a period of about 40 minutes and allowed to solidify in the furnace. The as-reacted carbide billet was carefully separated from the crucible material and pulverized. It was not remelted, in order to take advantage of the primary grain and fracture structure to make pulverizing easier. Carbon analysis was 4.35 w/o (47.7 atom percent) and 4.50 w/o (48.5 atom percent). The nitrogen impurity was less than 25 ppm and the total impurity of other elements was less than 0.1 percent. Metallographic examination indicated two phases. The second phase was approximately 10 percent of the total area.

Twelve vibratory compacted specimens were made using a mixture of 80 w/o uranium carbide and 20 w/o plutonium carbide. These were compacted to a depth of 2 in. in 0.156 in. ID specimen tubes. The compaction efficiency of the specimens was between 77% and 80% of theoretical density. Eleven similar specimens were prepared from various plutonium carbide melts and with one exception, compaction efficiency was greater than 80%. The jackets were stainless steel and niobium-1 w/o zirconium. Closure welds were made in the vertical position. None of the welded beads were free of fixed contamination.

Two plutonium carbide castings have been made in tantalum-coated graphite sleeve molds. The dimensions of these molds were 0.350 in. diameter by 2.75 in. long. The first casting was porous. The second casting appears to be sound but is still under evaluation. In both cases, the structure was fine grained and the mold was machined away without fracturing the casting.

4. Nondestructive Testing

a. Ultrasonic Techniques - Some time ago preliminary development work was undertaken on a high-power, short-duration pulse generator. This equipment, when completed, will produce a very short, well shaped pulse which will provide better pulse resolution than the equipment now being used.

Construction of the pulse generator is in a fairly advanced stage, except that arcs have developed in the high voltage power supplier. These have been located and are being repaired. Pulse width and repetition rate controls are now being incorporated into the device.

b. Neutron Techniques - The photographic detection of neutron images is being investigated, with at least one long range objective being to apply neutron radiography as a useful inspection method for nondestructive testing. Because neutrons have little influence on normal photographic emulsions, it is expedient to use intermediate converter materials next to the film. These materials emit photographically detectable radiation when bombarded with neutrons, and thereby improve the photographic speed of the detection process.

Image sharpness studies are continuing. The development of a new resolution test object containing a number of drilled holes whose separation continually decreases, has contributed to a significant reduction in the effort required to determine the resolving power of many of the neutron detection methods being evaluated. The resultant radiographs can now be examined visually and the resolution can be determined directly. The present test object is now being used for many of the lower and medium resolution converter materials but it is not yet adequate for determining the resolution of the better converter materials. It is hoped that further modifications in the test piece will allow us to extend these measurements.

Exposure curves for a number of heavy metals have now been completed. The data include exposure situations for transfer techniques and for fast (scintillator methods) and slower speed direct exposure methods. The thickness variations detectable in most cases are about 1 to 2% for all the techniques except the scintillator direct exposure method. This fast method generally appears to yield contrast sensitivities of about 6% using a B-10 enriched scintillator with Type F X-ray film.

The exposure data for uranium have been applied to an inspection problem involving a pressed uranium oxide cylinder being fabricated as a pycnometer. The 1 in. diameter cylinder was inspected in green form to assist in determining accurately the location of the hole to be drilled through the wall to the internal cavity. Of the radiographic equipment available for this inspection, the neutron inspection facility presented advantages in regard to both exposure time and reduction of scatter.

c. Eddy Current Techniques - Development work has continued on the pulsed-field reflection system. This electromagnetic test method replaces the sinusoidal currents and test coils or probes of the older eddy current test system with pulsed fields emanating from small apertures in special masks. Test information about the metallic specimen being

examined is received as a series of reflections from the surface and from inside the metal. This system represents a possible new approach to the problems of electromagnetic testing.

In a recent experiment it was found that minor changes in mask design can effect at least a 10 db increase in the amplitude of the reflected signal with no penalty in resolution or equipment complexity. This improvement is so encouraging that further design changes are being made along the lines indicated by this experiment.

The pulsed-field reflection system is being used to check the bond quality of pins of plutonium-fissium alloy enclosed in jackets made from a variety of materials and bonded with sodium. The resolution of this system is good enough to provide a very sensitive test for voids in the bond layer. The bond quality of 32 more pins has been checked prior to irradiation testing of the pins. Some of these pins had been checked previously prior to additional treatment applied to them to improve the bond. This test equipment provides a convenient way to check the efficiency of various bonding procedures.

5. Irradiation Studies

a. Irradiation of Metal Fiber Reinforced $\text{ThO}_2\text{-UO}_2$ - Previous work at Argonne (Irradiation of Metal-fiber-reinforced Thoria-Urania, ANL-6397) has shown that $\text{ThO}_2\text{-UO}_2$ with a 10 w/o addition of molybdenum fibers has a better thermal conductivity than the unfibered oxide in irradiations up to 34,500 Mwd/T burnup. Above integral $kd\theta$ values of about 75 w/cm, however, melted fibers in helium-bonded specimens migrate to the center of the specimen and coalesce into a sphere. In order to determine if such migration would occur spontaneously during a reactor transient and thereby vitiate the improved thermal conductivity afforded by the fibers, transient tests were performed in the TREAT reactor.

To date, two pellets of $\text{ThO}_2\text{-10 w/o UO}_2$ and $\text{ThO}_2\text{-30 w/o UO}_2$ each with 10 w/o Mo fibers have been run. The pellets were $\frac{3}{8}$ in. in diameter and $\frac{3}{8}$ in. long. Both pellets were clad in Zircaloy-2 with a 0.001 in. annulus filled with helium. A $\frac{1}{16}$ in. helium-filled space remained above each pellet.

The $\text{ThO}_2\text{-10 w/o UO}_2$ (93% enriched) pellet received a 217 Mw-sec transient on a 98 millisecond period. The cladding diameter at the midplane of the pellet increased 5.5% and a bulge was formed in the cladding on the bottom pellet face. The cladding was sheared away and the pellet removed intact. A slight diameter increase at the midplane was noted but this could not be compared to a preirradiation measurement. One side of the pellet was considerably lighter than the remainder and contained a few surface cracks. Droplets of molybdenum were present on the top

pellet face. These are believed to have been extruded through surface cracks on contraction of the pellet with rapid cooling. Transverse and longitudinal metallographic sections showed no apparent change in the oxide structure. Random cracks were in conspicuous evidence. Incipient melting of some fibers was also evident, particularly adjacent to the top face. There was no gross movement of the fibers.

The ThO_2 -30 w/o UO_2 (19% enriched) pellet received an integrated power transient of 580 Mw-sec on a 67 millisecond period with a peak power of 1270 megawatts. Calculated peak temperature was 3000°C . The Zircaloy-2 cladding completely melted away from the lateral surfaces of the pellet. The pellet fractured into two pieces almost at the midplane with a rough fracture surface. An irregularly shaped void, about $\frac{3}{32}$ in. diameter, had formed off-center. Transverse and longitudinal metallographic sections showed extensive melting of the molybdenum fibers. Whole fibers were present only $\frac{1}{32}$ in. from the outer surface. Melted fibers tended to coalesce into large patches of irregular shape and also to flow between grains. There was no tendency for movement in one specific direction. The oxide showed equiaxial grain growth but no columnar grains. Other than the large irregular void and a few smaller void patches, the internal structure was free from porosity. Very fine dendrites of oxide were present in molybdenum matrices indicating a molten ceramic core. Another dendrite phase was also noted in the molybdenum but its nature is not yet known. The large void cavity is attributed to the release of adsorbed gases from the molten core.

6. Metallic Alloys

a. Thorium-Uranium-Plutonium System - The study of the thorium-uranium-plutonium system was initiated with the intention of developing metallic fuel materials which are suitable for operation at significantly higher temperatures than uranium-plutonium alloys. Both binary thorium-uranium alloys and the thorium-plutonium alloys have been shown to have resistance to swelling and irradiation stability superior to the presently used alloys. However, there is no information on the ternary alloys and it would be valuable to learn how the properties of the ternary alloys compare with those of the binary alloys.

As a first step a phase diagram study of the ternary system is being undertaken. Useful ternary alloys may be either solid solutions of plutonium and uranium in thorium or two-phase alloys paralleling the binary thorium-uranium system with plutonium additions of up to 20 or 30%. In view of existing discrepancies, checks on some aspects of the binary phase diagrams, particularly the thorium-plutonium system, will be necessary.

Two series of alloys have been prepared by arc melting. The thorium-plutonium series with up to 60% plutonium was made from high purity metal and will serve for the determination of the solid solubility of plutonium in thorium particularly at high temperatures, where the published diagrams differ greatly. The thorium-uranium series with up to 50% uranium has been prepared from metals of commercial purity, also by arc melting, and will be used to study the distribution of the second phase (U) in the thorium matrix under a variety of heat treating conditions.

C. Reactor Materials Development

1. Radiation Damage in Metals

a. Neutron Damage Studies - Two sets of irradiations, one in CP-5 and one in EBR-I, were described in previous Progress Reports (ANL-6485, p. 49 and ANL-6525, p. 40). The capsulated SA-212B steel specimens in CP-5 received 486 effective full-power hours of operation, and were removed from the reactor and disassembled. A small piece of the nickel monitor wire was removed from each capsule and will be counted. The impact testing machine to be used in the cave is being cleaned and checked. The collection and preparation of data to initiate detailed spectrum calculations in the high energy range are almost complete.

The copper and steel specimens in the EBR-I reactor will receive a total of about 90 full-power hours of exposure before being returned to the DuPage site for analysis.

D. Heat Engineering and Fluid Flow

1. Two-phase Nozzle Test

The two-phase nozzle test was initiated to determine the effectiveness of a converging-diverging nozzle for isentropically converting the thermal energy of a saturated liquid into kinetic energy in a two-phase mixture. The kinetic energy will be determined by changing the momentum vector of the two-phase mixture and measuring the resultant force.

Pending the installation of additional electrical power capacity in the laboratory, the nozzle and thrust measuring device were checked with cold water. With the exception of minor difficulties in the thrust measuring device, the systems operated satisfactorily. The flow orifice was calibrated and the data were found to lie within 1% of the calculated value.

2. Critical Heat Flux Survey for Boiling Systems

A criterion has been established to distinguish between hydrodynamically stable and unstable critical heat flux data, based on previous investigations at ANL and on recently published data. A reliable separation of the existing data resolves many of the conflicting trends and conclusions and should expedite a utilitarian solution of the problem.

The hydrodynamically unstable data have been correlated on the basis of a critical mixture velocity of the water coolant. The correlation is valid for unstable flow of water-steam mixtures in vertical channels and the critical heat flux values are appreciably lower than in the case of hydrodynamically stable flow. The operational variables causing the flow instability are qualitatively defined in a report now being completed.

3. Hydrodynamic Instability

The time-dependent behavior of a boiling loop is being examined by determining the frequency response of steam volume fraction to flow and power oscillations.

Mechanical work on the system components is progressing satisfactorily. The flow-oscillator drawings have been submitted for fabrication and the boiling test section has been assembled.

The tests of the onset of hydrodynamic instability in a natural circulation loop are near completion.

Effects of restrictions in the riser and in single-phase regions on a 2.38-cm ID test section and riser have been investigated at 20 atm (300 psia) and at 41 atm (600 psia). The onset of instability is now being determined for a 2.38-cm ID test section with a 1.6-cm ID riser.

4. EBWR Hydrodynamics

Two riser configurations, designed to reduce carryunder of steam voids into the downcomer at 100 Mw, have been studied in detail. One is a tapered (cone-shaped) single riser which reduces the total flow rate and provides more downcomer area at the top of the riser than the present design. The other is a double riser (two concentric cylindrical shells) which also reduces the total flow rate, but provides a relatively low resistance to flow through the central 36 fuel elements, of which 20 elements are the high power density "spike" type elements.

Both designs reduce the flow rate considerably and provide greater downcomer area than the present system. The double riser has a distinct

advantage, however, since it provides for decreased velocity without depriving the high power density region of coolant, thus maintaining the original vapor fraction value in the critical central core region.

E. Chemical Separations

1. Fluidization and Fluoride Volatility Separations Processes

a. Fluoride Separations - Study of the decomposition of plutonium hexafluoride by gamma radiation has continued (see Progress Report, January 1962, ANL-6509, p. 41). Recent experiments have shown that the molar ratio of fluorine produced to plutonium tetrafluoride formed during the decomposition is close to 1, as anticipated from the stoichiometry of the decomposition. A G value of 5.8 ± 0.8 for the decomposition has been obtained. Additional data have been obtained for the formation of plutonium hexafluoride from plutonium tetrafluoride and fluorine, between 60° and 70°C, by the action of gamma radiation. The data can be correlated with the energy absorption in plutonium tetrafluoride and a tentative G value of 0.07 has been obtained.

The preparation of mixed chlorofluorides of uranium (VI) has been attempted. Thus far no success has been achieved either by reaction of uranyl chloride with sulfur tetrafluoride at 250° to 325°C or by reaction of uranyl fluoride with chlorine or carbon tetrachloride at 600°C.

b. Fluorination of Uranium Dioxide Pellets - A series of fluorination runs is being made in the three-inch diameter finned fluid-bed reactor to determine optimum processing conditions for larger pellet charges (18-inch deep bed compared to a 6-inch deep bed). A scheme employing off-gas recycle together with addition of nitrogen as diluent was used to improve fluorination rates and efficiencies. In a run at 500°C, an 18-inch deep pellet bed charge was 96 percent fluorinated in 22 hours with 77 percent overall fluorine efficiency. This is an improvement over previous runs; however, caking occurred in this run. The fluorination rate and efficiency are considered adequate, but further study is required to eliminate caking.

c. Studies on Fluidized-Packed Beds in Mockup Systems - Measurements were made of the elutriation and mixing rates of uranium oxide fines in a fluidized-packed bed such as employed in the direct fluorination of uranium dioxide pellets. An 18-inch bed of uranium dioxide pellets was used with 9 to 24 inch depths of 60 to 200 mesh alumina grain as fluidized bed. Elutriation tests were made with column freeboards of 2 and 3 ft. The uranium oxide fines were a size of the order of 12 microns. Mixing of the fines with the alumina was observed to occur in a fraction of a minute at fluidization velocities. The fines elutriation rate increased

markedly with gas rate and lower freeboard, but was little affected by alumina bed height in the range tested. It appears likely that such fines are retained in the fluidized bed long enough to be completely fluorinated in typical operating conditions.

A brief study was made to compare gas mixing in fluidized beds with and without packing. It was shown that the gas mixing is controlled chiefly by the packing. For design purposes it is considered that gas mixing in fluidized-packed beds is similar to that in packed beds without the fluidizing medium.

d. Plutonium Pilot Plant Facility - An engineering-scale facility for fluorination process studies of plutonium-uranium oxide fuels is being built. The alpha enclosures are being installed, and a contract has been let for room modifications. Orders have been placed for more than half of the equipment and instruments. The fluorination reactor and uranium hexafluoride condenser are being fabricated. The engineering design of the panelboard has been completed and fabrication drawings have been started. The present estimate is that the project will be ready for plutonium operations early in 1963.

e. Studies of Hydrochlorination and Fluorination Steps with Down-Flow Fixed-Bed Filters - Hydrochlorination-fluorination studies on an enriched alloy fuel recovery scheme continued in the $1\frac{1}{2}$ -inch diameter fluid-bed reactor. Two runs at higher temperatures, 450° and 550°C for both reaction steps, as compared to 350° to 450°C , resulted in 2.0 and 1.6 weight percent, respectively, of the uranium charged being left in the fluid-bed portion of the system. Somewhat lower residual values are expected for the down-flow packed-bed filter section. Some indications of excessive corrosion became evident during the higher temperature hydrochlorination with the formation of particulate (believed to be nickel chloride) which plugged the down-flow filter-bed section.

f. Fluid-Bed Hydrolysis of Zirconium Tetrachloride - Experiments involving two to three hours of operation were continued pending receipt of the enlarged zirconium tetrachloride feed hopper. An electric boiler is now being used to supply steam for the reaction in place of the two-fluid water atomizing nozzle which was believed to be the source of some attrition of bed material. An increase of gas residence time in the bed (achieved by lowering the superficial gas velocity) did not appear to eliminate the apparent gas phase reaction occurring above the bed in the filter region as evidenced by excessive increases in pressure drop across the filters.

g. Process Design Work for Reprocessing Zirconium-Enriched Uranium Alloy Fuels - Drawings of major pilot plant process vessels and some auxiliaries are being given a final design check. Layout sketches for equipment placement are being made. Specifications for instrumentation

are nearly completed to allow placing of orders. Exploratory tests are being conducted in Lucite columns to observe the behavior of an up-flow filter-bed-fluidized-bed system containing simulated fuel assemblies. This consists of a lower fluid-bed reactor section and an upper enlarged diameter fluid-bed filter section.

The reaction of fluorine with charcoal, considered as a method for disposal of excess fluorine from the pilot plant, was examined briefly in a 3-inch diameter pipe section filled to a depth of 25 inches with Barnebey-Cheney type CG-2 coconut charcoal, nominal 5 mesh size. Flow conditions varied from 11.0 to 19.7 scfh of gas containing from about 8 to about 65 percent fluorine (remainder nitrogen). Only at the highest flow was there evidence of unreacted fluorine (as determined by potassium iodide-starch paper). Overall results are quite satisfactory.

2. Chemical Metallurgical Process Studies

a. Liquid Metal Solvent Studies - The solubility of sodium in liquid zinc was found to vary between 1.8 weight percent at 562°C and 0.042 weight percent at 430°C.

The zinc-tantalum system is being studied. The solubility of tantalum in zinc was found to be too small for accurate determination. Solubilities at 750°C varied from a few tenths of a ppm to 3 ppm. Samples of tantalum powder and zinc corresponding to compositions of TaZn_2 and TaZn_5 were heated for nine days at 600°C, and then ice quenched. Metallographic examination showed some free zinc in the TaZn_5 preparation, but none in the TaZn_2 preparation. X-ray examination revealed the existence of a new phase containing tantalum and zinc; this new phase is being characterized.

The solubilities of uranium in magnesium-zinc solutions were determined and are shown in Table V.

Table V. Solubility of Uranium in Magnesium-Zinc Solutions

Temperature (C)	Magnesium (w/o)	Uranium (w/o)
702	24.2	4.63
651	24.4	4.33
552	26.7	1.45
502	29.9	0.374
450	35.3	0.142

Additional measurements were made to establish more accurately the consolute temperature of the lead-zinc system. The consolute temperature is now shown to lie between $784^\circ \pm 2^\circ$ and $789 \pm 2^\circ\text{C}$.

Effusion studies of the scandium-cadmium system have confirmed the existence of the compound ScCd_3 (see Progress Report, August 1961, ANL-6409, page 56) and revealed the existence of an additional compound, ScCd , which was found to have the CsCl structure.

Additional experiments were carried out to delineate the boundaries of the epsilon uranium-zinc intermetallic compound. The zinc:uranium atom ratios on the zinc-rich side were found to be 11.73 at 500°C , 11.49 at 650°C , and 9.83 at 780°C . A 53-day annealing experiment at 650°C with alloys having zinc:uranium atom ratios from 11.03 to 8.61 indicates that the uranium-rich boundary of the epsilon phase corresponds to $\text{UZn}_{9.5} \pm 0.5$.

b. Metal Distillation Studies - With the installation of a new heating coil, operation of the large cadmium distillation unit (see Progress Report, May 1961, ANL-6374, page 27) was resumed. Over 600 kg of cadmium have been distilled in three runs. Operation of the unit has been essentially trouble-free. However, the efficiency of the new induction coil assembly is lower than that of the previous one. The reasons for this difference in efficiency are being explored.

c. Mass Transfer Studies - A dynamic loop for the study of mass transfer from a solid uranium sphere to flowing cadmium has been constructed. Calibration of flow instruments is in progress.

d. Calorimetry - A series of nine combustions of cadmium in fluorine and a series of ten combustions of magnesium in fluorine have been completed. The calculation of the heats of formation of cadmium and magnesium fluorides awaits additional calibration experiments.

Exploratory noncalorimetric experiments are being carried out to investigate the feasibility of calorimetric studies with fluorine on tantalum, niobium, graphite, and diamond.

Some preliminary experiments on the combustion of ruthenium in fluorine have been made. This work may lead to the determination of the heat of formation of ruthenium pentafluoride by the direct combination of the elements in the bomb calorimeter.

3. General Chemistry and Chemical Engineering

a. Simultaneous Reaction Studies - Increased uranium dioxide particle densities to nearly 9.0 g/cc (about 81 percent theoretical density) have been achieved in recent simultaneous reaction studies for converting uranium hexafluoride to uranium dioxide in the three-inch diameter reactor. Feed rates ranged from 19 to 27 g/min of hexafluoride feed as compared

to 10 g/min used previously where a density of 6.6 g/cc was obtained. The procedure involves intermittent feeding of the hexafluoride - 45 minutes on, 15 minutes off, but with continued steam and hydrogen flow. Low residual fluoride contents (200 to 250 ppm) were maintained at the lower hexafluoride rate. It is yet to be determined whether steady state has been reached.

b. Preparation of Uranium and Plutonium Carbides - Work has started on the preparation of enriched uranium carbide by precipitation from a 12 weight percent magnesium-zinc solution at about 750° to 800°C. Activated charcoal, previously degassed, is being used as the carbon source. Construction of facilities for the preparation of plutonium carbide is continuing.

IV. ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

A. Argonne Advanced Research Reactor (AARR)

For the past several years, Argonne National Laboratory has recognized an increasing need for a source of higher neutron fluxes to permit the logical extension of basic research in nuclear physics, nuclear chemistry and solid state science, and has carried on a program to develop design concepts for higher flux research reactors to fill this need. The latest of these concepts is the Argonne Advanced Research Reactor (AARR), a heterogeneous, light water moderated flux-trap type facility having a maximum unperturbed thermal neutron flux of $5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$. The reactor is designed primarily for studies involving the use of neutron beams, but will also have a significant number of irradiation facilities. The nominal power level of the reactor is 100 Mw, with a coolant outlet temperature of 176°F. Information developed in previous studies on AARR is contained in ANL-6451.*

Currently the Laboratory is carrying on more detailed studies of plant layout and equipment arrangement, as well as initiating research and development work, largely in the area of fuel element, critical experimentation, hydraulics and mechanical design. A subcontract has been negotiated with United Nuclear Corporation to perform a complete engineering study and cost estimate of the facility to be used as supporting information for the construction authorization request. It is anticipated that the work performed under this subcontract will materially reduce the time required for a formal A-E Title I design study and would thus expedite completion of the facility.

*D. H. Lennox, et al., "Status Report on the Argonne Advanced Research Reactor," ANL-6451 (November, 1961).

V. NUCLEAR SAFETY

A. Thermal Reactor Safety Studies

1. Fuel-Coolant Chemical Reactions

High-speed motion pictures were taken of condenser discharge runs with aluminum wires under heated water. The cause of previously reported slow pressure rises could not be determined. A series of runs with stainless steel wires has begun.

Final results of studies of the reaction between aluminum-uranium alloys and steam by the pressure-pulse method were reported. Oxide films on alloys containing up to 40 weight percent uranium were very protective at 1200°C. Reaction rates were only slightly increased over those reported with pure aluminum.

Studies of metal-steam reactions by the levitation melting method are continuing. Results of several preliminary runs with molten aluminum at temperatures between 1400° and 1900°C show no definite correlation of extent of reaction with temperature. Additional runs are planned.

Five transients were completed in TREAT on fully carbided, graphite fuel specimens in water, helium, and partial vacuum environments; the results obtained are summarized in Table VI.

Table VI. Summary of Results from TREAT Tests on Fully Carbided Graphite Fuel Specimens

	CEN TREAT Transient Number				
	83	84	81 ^a	82	85
<u>Reactor Pulse</u>					
integrated power, arbitrary units:	1.05	3.60	5.36	7.71	7.66
period, milliseconds:	141	74	54	42	40
<u>Peak Temp of Fuel Pin</u>					
°C	900	1710	2425	2570	-
<u>Xe¹³³ Released from Fuel</u>					
microcuries:	109	1680	3870	2150	1070
<u>Appearance of Fuel Pin after Transient</u>					
observations:	large fragments and particles	small fragments and fine particles	intact, no change	large fragments and a few particles	fragments and particles
d_{sv} , mean particle size, mils	86	28	500	330	66
<u>Environment:</u>	water	water	helium	helium	partial vacuum
<u>Extent of Fuel-Water Reaction</u>					
w/o reacted:	17	25	-	-	-

^aCEN-81 was a calibration experiment in which the fuel specimen was subjected to six transients of progressively higher energy; the final transient had an energy of 5.36 units.

From these data and from the previous tests on shelf-type graphite fuel pins the following conclusions can be drawn:

1. Fully carbided fuel specimens can be subjected to more energetic reactor transients (and therefore higher temperatures) without destruction relative to shelf (hydrolyzed) graphite fuel specimens; the respective threshold temperatures are about 2500°C and 1700°C.

2. Correlation of mean particle diameter of the fragmented fuel as a function of the energy of the reactor burst established a relative order of physical stability of the fuel in reactor transients as follows:

Most stable:	fully carbided fuel in helium
Medium stability:	shelf fuel in helium
Least stable:	coated fuel in water.

3. Correlation of the amount of xenon-133 released with the energy input showed that at low energy transients (temperatures up to about 1500°C) only a few percent of the xenon-133 was released, whereas at high energy pulses (2000° to 3000°C) from 50 to 80 percent of the xenon-133 was released.

4. Hydrogen and carbon monoxide are the main reaction products formed when the fuel reacted with water during the reaction pulse; 17 and 25 percent by weight were observed. Significant amounts of iodine-131 were found in the water in which the fuel was fragmented.

2. Metal Oxidation and Ignition Studies

The calculation of burning temperatures of metals in air has been completed. Calculations were previously reported for both natural and forced convection. Recent calculations were made of the reaction rates corresponding to the previously reported burning temperatures. The heat transfer coefficients involved in the combustions were also calculated in explicit form. The results are being prepared in the form of a topical report.

Mixtures of either CF_3Cl or CF_2Cl_2 with CH_3CHF_2 in a 1:1 ratio were found to be nonflammable and are good inhibitors of uranium burning in air. The mixtures did not affect the shielded ignition temperatures of uranium although they decreased the burning curve ignition temperatures somewhat.

Ignition experiments in air and oxygen have been completed on plutonium samples containing the following atomic percent additives: 0.27 Si, 1.26 Ni, 2.40 Cu, 0.40 and 1.03 Zn and nominally 2 Mg and Cr. All the additives caused limited increase of the ignition temperature, copper having the greatest effect of approximately 50 degrees.

Studies of the oxidation and ignition of uranium are continuing. Isothermal oxidation studies have shown that the oxide is somewhat protective above 450°C, while the oxidation appears to be autocatalytic between 300° and 450°C. The existence of two kinds of oxide is now thought to be responsible for the large effect of specific area on ignition temperature. One possible explanation of the specific area effect is that there is a time lag in converting the low temperature form of the oxide to the high temperature form. High specific area samples are capable of rapid temperature changes in which there may be insufficient time for protective oxide to form as the sample self-heats to temperatures above 450°C. Further tests of this theory are in progress.

B. Fast Reactor Safety Studies

1. Experimental Meltdown Program

In-pile experiments are being performed in the TREAT reactor to study the important mechanisms producing failure in fast reactor fuel elements and meltdown product motion.

a. EBR-II Elements in the Absence of Sodium - A series of four meltdown experiments (Series XXIX) of uncooled clusters of EBR-II Mark-I elements (see Progress Report, February, 1962, ANL-6525) were completed in TREAT, and the capsules opened for inspection of the samples. Each cluster consisted of seven natural enrichment EBR-II Mark-I elements on the EBR-II pitch with one central element surrounded by a ring of six. In each experiment chromel-alumel thermocouples were mounted at the center and 9 cm above the center on the inner element and on one element in the outer ring. Tantalum neutron absorbers were used to form the sample flux axially into a shape typical of that for fast reactor cores. The series was arranged to expose each successive cluster to transients of progressively increasing intensity. A summary of the results is given in Table VII.

In test No. 2 all elements but one had spots of eutectic along the fuel. The center element showed one small penetration with an underlying area of eutectic erosion. A similar penetration was located opposite this point on the contiguous surface of an outer element. The underlying uranium, however, was not covered with eutectic. This suggests that the second breach was made from the outside. In addition, the cladding of all the elements in the first two tests showed a general increase in diameter in the fuel region of approximately 0.0076 cm. This increase is attributed to internal pressure generated by the sodium.

Table VII. Series XXIX EBR-II Clusters in TREAT

Test No.	Maximum Temperature, °C				Condition
	Center Position		Top Position		
	Inner	Outer	Inner	Outer	
1	730	690	580	610	All fuel elements warped, no failures, inner element least warped.
2	920	750	710	760	All fuel elements warped, small penetrations on opposing inner and one outer element.
3	1040 (est)	940	940	990	Gross failure of all elements, 18-cm long band of uranium on wall at center, some intact pieces of fuel inside cladding.
4	1200 (est)	1100 (est)	1100 (est)	1100 (est)	Very drastic failure, fuel distributed along length of sample, most of fuel at bottom, little cladding left.

It was also noted in test No. 3 that a considerable number of unmelted pieces of uranium remained in portions of the cladding. A number of these pieces were in positions showing that they were moved bodily upward from the bottom of the fuel can. The movement amounted to several centimeters. This movement was probably caused by the sodium vaporizing inside the cladding at the bottom of the element in agreement with previous results on single samples.

Preparations to conduct the second series of cluster meltdown tests (Series XXX) have been completed. This series will be a continuation of the dry opaque tests. Six capsules, consisting of two of each of three types of sample arrays will be subjected to transients designed to induce mild and drastic failure. The experimental procedures will permit the exposure of prototype fuel element clusters using natural and enriched uranium pins in hexagonal outer cans of approximate EBR-II subassembly can wall thickness.

b. Uranium Oxide Samples in the Absence of Sodium - Tests are being performed on EBR-II size, inert gas-bonded uranium oxide fuel samples in the absence of sodium coolant. Elements from Series XXVIII, the first series of oxide samples exposed in transparent capsules, were examined. The results are noted below:

Sample 1 - The estimated maximum temperature was 1100°C. After the transient, the original nine slugs were found to have been cracked into forty-nine shorter pieces. The steel cladding was warped around the spiral spacer wire, but was intact.

Sample 2 - The estimated maximum temperature was 2550°C. The steel cladding was found to be destroyed and the original slugs were broken down into shorter fragments similar to those of Sample 1. Evidence of melting in the center of the slugs was found.

Sample 3 - The estimated maximum temperature was 2750°C. The steel cladding was destroyed and the fuel cracked into shorter fragments. Fourteen of the recovered pieces were tubular and had an annular void about 0.12 cm in diameter indicating that the centers had melted.

Sample 4 - The estimated maximum temperature was 2400°C. The tantalum cladding was intact after the test and the oxide slugs again were broken into shorter pieces. There was no apparent change in the slug centers.

Sample 5 - The estimated maximum temperature was 2750°C. The tantalum cladding was intact after test but the oxide slugs were completely granulated into pieces from about 0.1 to 0.3 cm in length. Approximately 3% to 5% of the oxide was present in the form of fines and dust.

The source of the apparent discrepancy in the production of the exposed oxide found in the steel clad and in the tantalum-clad samples is not known at present. Figure 13 shows still prints from the color motion picture films taken of the five samples.

2. Equipment Development

a. Dry Opaque Capsule for Remote Assembly - Two prototype dry opaque capsules for remote assembly have been received from the shops, and preparations are being made to perform the assembly mockup tests. In addition, the slings for the coffin have been received and the spreader bar is being constructed in the shops. Also, the coffin was tested mechanically in a handling mockup test and was found to operate satisfactorily.

A shielding arrangement was designed to permit gamma scanning of the pre-irradiated samples prior to encapsulation.

b. Large TREAT Loop - The crane and steel for the stairwell rework and loop room modification have been received and are ready for shipment to Idaho. In addition, the loop piping and fittings have been received and will be shipped shortly. The tank pit drawings have been examined and approved and construction is expected to start shortly.

The specifications for the bellows expansion joints have been established and will be ordered. The support frame for the upper shield blocks and the pipe grinder are being fabricated. The test section pressure transducer mounting has been designed and the final design for the shielding blocks has been completed. Design requirements for the loop heater wiring have been established and the final layout will be prepared.

Figure 13. Black and white prints made from the color motion pictures taken of oxide sample meltdown experiments. In each case, the sample appears at the left and its image appears in a mirror at the right. Frames "a" to "e" show Sample 3 of Series XXVII, a stainless steel clad sample. Frame "a" shows the sample before the experiment. The second picture shows a small cladding failure with emission of a small cloud (probably powdered uranium oxide). As the heating continues, stills "c," "d," and "e" show the sample reaching white hot temperatures when it collapses under the force of gravity. For comparison, print "f" shows a tantalum-clad sample exposed to approximately the same energy input (Sample 5), and which did not fail. The alumina thermocouple insulator tubes can be seen leaning against capsule walls following the melting of the platinum-platinum/rhodium thermocouple junctions which had been spot-welded to the tantalum cladding.

Installation of the new power distribution transformers required to supply power to the large sodium loop was completed. Approximately 90% of the instruments for the sodium loop have been received and installation of these instruments is about 50% complete.

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ANL-6486 FREE CONVECTION HEAT TRANSFER TO A SUPER-
CRITICAL FLUID: AN ANALYTICAL AND EXPERIMENTAL
STUDY

C. A. Fritsch and R. J. Grosh