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PROGRESS REPORT

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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:

December 1961	ANL-6485
January 1962	ANL-6509
February 1962	ANL-6525
March 1962	ANL-6544
April 1962	ANL-6565
May 1962	ANL-6573

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I. WATER COOLED REACTORS

A. EBWR

1. Preparation for High Power Operation

a. Reactor - Leads for instrumentation to be used during the approach to the high power runs were installed. About fifty leads from the core and from within the reactor vessel were brought out through three instrument nozzles and individually pressure sealed at the flanges. The instruments connected include thermocouples, ionization chambers, turbine type flowmeters, flux monitoring tubes, and pressure taps. The reactor vessel was then hydrostatically tested. The plant is currently being checked out for operation.

b. Soluble Poison Control System - The ion exchange units, pre-filters, and afterfilters in the EBWR purification-boric acid removal system were modified to provide greater radiation safety, more versatility, and simplicity. Provision was made for the selective use of either the pre-filter or the afterfilter. Thus, the best combination of cooling system pumps and filter units can be selected for use while the unused filter units are re-charged. In addition, the prefilter vessels are being fitted with baskets containing a mixed resin saturated with boric acid which will serve the dual purpose of filter and chemical cleanup.

The anion units for the removal of boric acid were modified to include one cubic foot of mixed resin at the bottom of the resin basket which was separated from the anion resin by a stainless steel screen. Eight cubic feet of anion resin were used to fill the basket. Thus, the mixed resin will remove any impurities introduced by the anion resin. (Quantities of sodium ion have been previously encountered using the anion resin.)

c. EBWR Neutron Flux Distribution - Spectrum - A total of five Inconel tubing loops were installed in the southeast quadrant of the EBWR core. An additional loop was placed in the downcomer region in the reactor vessel. Another loop was located between the reactor vessel and the reactor shield.

The loop locations within the core were chosen to give a radial and axial representation in one plane, and to give an indication of the distribution at other locations of interest in the specified quadrant. Bead chains will be used to locate activation foils containing various detection elements for characterization of the neutron distribution and energy within the various loops. Insertion and removal of foils using these loops can be accomplished during reactor operation.

As a preliminary study of neutron flux distribution in the EBWR, a number of wires were activated under cold, critical conditions and with

various concentrations of soluble poison in the reactor water. The activations and radiochemical analyses were completed and data preparation is in progress.

d. Test of Demister Sample - A sample of Inconel mesh material, similar to that which will be used in the EBWR demister (if a demister is required) was tested in a loop for two weeks under simulated EBWR operational conditions. The weight loss incurred during the test was 0.0032 gm of the original sample weight of 25.8713 gm. A slight discoloration was noticed on the upstream end of the sample mesh. This discoloration was due to corrosion products from the loop which were carried along with the moisture in the stream and deposited on the mesh. The performance was considered to be satisfactory and fabrication of the demister is proceeding. Delivery is expected within six weeks.

2. EBWR Transfer Function

a. Preliminary Noise Results - Five test runs were made for the random binary noise signal. The generated signal was fed through two different networks. One is a simple lag network and the other an approximated transfer function for EBWR, 20-Mw, 600-psi power. For the latter case, the time-noise data obtained from previous tests were added to the output signal at various noise-to-signal ratios. The noiseless record in both cases proved to have excellent accuracy. However, at high noise-to-signal ratio ($N/S = 1$), high scattering occurs at the frequency where noise power density is strong. (It is off almost 4 db at low frequency.) At $N/S = \frac{1}{4}$, the scattering is reduced by a factor of 4 in decibels. By analyzing the noise-to-signal ratio for a $\pm \frac{1}{2}$ -in. stroke at 20 Mw power, it was found to be $\frac{1}{3}$. Hence, if a ± 1 -in. stroke can be used for the binary noise input, the accuracy should be within ± 0.5 db and ± 3 degrees.

A filter (based on 20-Mw power) was built for the pre-whitening of the output data. This method will be used to see whether further improvement in accuracy can be made.

3. IAEA Participation

The first official IAEA inspection and meeting was held on June 6, 7, and 8 at the EBWR facility. Briefly, the inspection covered a general discussion, inventory status, a physical inventory, engineering status, the EBWR experimental program, the method of determining burnup and plutonium production, and the safeguard procedures. The inspection appeared to be satisfactory to all concerned.

B. BORAX-V

1. Operations and Experiments

Since the high temperature pre-operational tests had to be terminated due to heater failure earlier in the month (see item 4.a.), the experimental program has been rearranged. The new control rods will be installed, the fuel reloaded, the oscillator rod system tested and additional flux wire irradiated at room temperature while awaiting procurement and installation of new heaters.

Data taken during the cold critical experiments are still being analyzed. Axial data taken with uranium-zirconium and gold flux wires has been fitted to a chopped cosine shape by the least-squares technique. The maximum-to-average axial neutron flux values for various radial locations are shown in Table I.

Table I. Axial Neutron Flux Values

<u>Core Position</u>	<u>Radial Dist. from Core Center</u>	<u>Axial Max.-to-Avg. Flux</u>	
		<u>Gold</u>	<u>U-Zr</u>
CP54, WC13	$2\frac{7}{16}$ in.	-	1.42
CP55, WC8	$4\frac{1}{2}$ in.	1.359	-
CP57, WC8	$15\frac{5}{16}$ in.	1.333	-
CP58, WC8	$16\frac{3}{16}$ in.	1.510	-

No significant difference was noted between the cadmium ratio measurements made on flux wires irradiated in aluminum thimbles and those made in stainless steel thimbles of the same dimensions.

2. Modifications and Maintenance

Subsequent to the recent high temperature pre-operational tests, several items of plant checkout and adjustment have been completed. The trace heaters on the boron tank piping and valves have been re-worked to provide improved temperature control. The low-steam-flow-scrum D/P cell was found to have a response time of about 10 seconds. It has been returned to the vendor for modification to improve response time. The fission product monitors have been moved to the heating and ventilating building to reduce background radioactivity during reactor operation. A new area radiation monitor has been installed in the reactor pit. The "Reactor On" warning system has been revised to provide more meaningful operation through elimination of needless warnings. The air-water fuel assembly

test loop has been checked out and the thulium source re-irradiated in preparation for testing the flow and void measuring devices on the first instrumented boiling fuel assembly. The water chemistry sampling station conductivity cells were recalibrated with standard conductivity solutions and adjusted to read properly.

The control rod drive seals have been overhauled. All the Stellite seal rings that were scored, chipped, or out of tolerance have been replaced. New Delrin wipers were installed, and 17-4 PH stainless steel roll pins have been replaced with Inconel pins. The old 17-4 PH stainless steel control rod extension shafts were replaced with new shafts having the 1100°F heat treatment. Modifications have been made on the telescoping grapples for boiling fuel assemblies. A stronger worm gear and bearing housing was designed for more severe service.

The boiling core structure was removed, and the control rod channels in the core structure shroud were gauged. Only a small amount of widening of the center rod channel was required. A crane indexing system has been designed, installed, and checked out. This system will facilitate the future handling and placement of irradiated core structures and other reactor vessel internals.

To remove the magnesium oxide released by the failed immersion heaters, the reactor vessel and auxiliary piping were cleaned with 3% citric acid-ammonium citrate solution containing approximately 0.2% Triton X-100 wetting agent. The temperature was held at 160-190°F for about 28 hours, with continuous mixing only during the day shift. Removal of the cleaning solution was followed by three demineralized water flushes. During the test, the pH went down to 2.8. The iron concentration of the cleaning solution rose to 3.6 ppm, where it appeared to be leveling off. A rough calculation indicates that some 40 grams of Fe_2O_3 were removed from the stainless steel surfaces. Analyses showed that the magnesium content of the cleaning solution leveled off after a time, indicating that all the magnesium oxide had been removed from the system.

The boiling core structure and feedwater sparger were returned to the reactor vessel. Nine new control rods were installed and the dash-pot, seal, drive mechanism, etc., were reinstalled. The control rod stroke has been reduced to 25 in. by relocating the upper limit switch.

The make-up water demineralizer has not been giving theoretical capacity. Investigation revealed that about 5 cubic feet of anion resin was lost from the vessel, probably during backwash. Modifications are planned to prevent recurrence of this loss. A change in mixing procedure, regenerant level, and mode of regenerant addition is being presently investigated.

3. Procurement and Fabrication

a. Superheater Fuel - To date, 155 FPE (fully-enriched peripheral) superheater fuel plates have been inspected, with 137 accepted and 4 rejected. The remaining 14 plates are being rechecked for total thickness.

A total of 71 central superheater 4-plate fuel elements have been vacuum brazed with Coast Metals 60 alloy. The 60 elements which satisfactorily passed the water immersion leak test and dimensional inspections have been welded into 12 fuel subassemblies containing 5 elements each. Five of the remaining 11 elements have been satisfactorily rebrazed and have been welded into an additional subassembly.

b. Reactor Components - Assembly of four new T-shaped control rods was completed. After inspection, nine of the new Boral control rods canned in stainless steel were shipped to the reactor site. Long-term autoclave proof-tests are continuing on two spare control rod poison sections.

To replace the electric immersion heaters which failed in the reactor water preheater, 100 new 2000-watt Inconel-sheathed cartridge heaters have been ordered. Fourteen new stainless steel heater flanges, each containing six dry wells, are being fabricated for these heaters.

c. Experimental Components - A second instrumented boiling fuel assembly has been completed and received at the reactor site. For this assembly, Conax thermocouple packing glands and a vertical seal plate have been used as the reactor vessel pressure seal. This pressure seal satisfactorily passed a 1200 psi hydrostatic test.

A method is still being sought for salvaging the thermocouples on the first instrumented boiling fuel assembly which were damaged during the brazing operation on the pressure seal block.

The Zircaloy-II-canned Boral poison section of the oscillating rod was completed and satisfactorily passed a three-day autoclave test at 600 psig, 489°F conditions. The drive shaft and guide bearing assembly was also completed. With the delivery of the synchronous drive motor and the high pressure carbon ring rotating seal, all the components of the oscillator rod system are on hand at the reactor site. This system was assembled and test-operated in a test tank at atmospheric pressure without the seal. The oscillator operated smoothly up to a speed of 1150 RPM (19 cps), at which point a severe vibration of the drive shaft was experienced. The design maximum speed is 1200 RPM.

Development work on a Zircaloy-II to stainless steel pressure-tight joint on the in-core step-function generator thimble is continuing. A test braze, using a braze alloy consisting of 8% Ni, 8% Cr, 84% Zr on a mockup of this joint, cracked while cooling. A second test braze was made, using pure silver as a braze material. This sample is being corrosion-tested in water at 600 psig, 489°F conditions to determine the corrosion resistance of the joint.

4. Development and Testing

a. Plant Tests - The pre-operational tests at high temperature were terminated short of completion due to failure of the sheaths in a portion of the immersion-type auxiliary water heaters and the consequent release of magnesium oxide and inability to maintain adequate quality of the reactor water. Since this type of heater might well be subject to future failure in the same manner, it was decided to replace these units with dry-well-type heaters.

Certain useful information was gained prior to the ending of the pre-operational test. The new feedwater cooling system was found to be satisfactory, and the revised superheater drain system appeared to perform well at an initial test pressure of 100 psig at 335°F. A test dump of the boron addition tank indicated that because of severe water hammer the tank will have to be maintained at reactor temperature and pressure rather than the planned low temperatures. The recently installed auxiliary water system strip-type heaters performed well. The seal water control system was found to require the addition of a reset device which has since been installed.

The high temperature source drive was operated at 350°F. The drive was left in the full up position; and, as a result of thermal contraction of the lead screw when cooled, one of the graphite bearings cracked. Replacement bearings have been procured.

b. In-Core Instrumentation - A purchase order was placed for the fabrication of six developmental ultra-high temperature thermocouples, using a tungsten/26% rhenium alloy sheath, tungsten-tungsten/26% rhenium thermocouple wire, and 99.8% pure thoria as the insulation material. A test specification for testing and calibrating these thermocouples up to 5000°F has been prepared and sent out for bids.

c. Corrosion Tests - A static corrosion test on X-8001 aluminum and Type 304 stainless steel samples in a boric acid solution at a concentration of 20 g/gal is being made at 600 psig, 489°F conditions. The sample bundle is formed of alternate strips of aluminum and stainless steel coupled to test electrolytic action. Examination at 180 hours showed little evidence of corrosion.

Corrosion testing of stainless steels in superheated steam has been carried out using a variety of degassing and startup procedures. At the higher test temperature (650°C) it appears that small differences in test procedure exert important influence on corrosion behavior. In particular, as indicated last month and below, the corrosion rate appears to be very sensitive to oxygen concentration at low levels of oxygen concentration.

High purity feed water has been degassed prior to generation of superheated steam by three methods.

- a) Continuous boiling of feed water in an open tank. Oxygen concentration is in the range of 1-2 ppm. This value will be checked.
- b) Continuous helium sparging of feed water in an open tank. Oxygen concentration is 100-700 ppb.
- c) Simultaneous evacuation (resulting in boiling at room temperature) and helium sparging in a closed carboy for 10-15 minutes, followed by saturation with helium at 17 psia for duration of exposure. Oxygen content to be determined.

Oxygen content in these low ranges of interest is measured using thallium oxide. Present difficulties in technique, believed to be mainly associated with the low flow rates characteristic of these systems, cause readings of below 500 ppb to be of doubtful accuracy.

High corrosion rates for stainless steels exposed at 650°C using "degassing" method (a) have been previously reported and are repeated in Table II for convenience.

Table II. Corrosion of Stainless Steels in "Degassed by Boiling"
Superheated Steam at 650°C and 600 psi

Material	Wt. Gain, mg/dm ² after 15.8 Days (Duplicate Tests)	
304 (pickled)	1) 312	2) 312
316 (wet ground)	1) 43	2) 94
347 (wet ground)	1) 77	2) 89

After this period of exposure samples were covered with a heavy gray film and spalling had begun.

Wet ground samples of 304, 316, and 347 have been tested 650°C and 600 psi for one week using degassing method (b) and for two weeks using method (c). All samples are covered with temper films containing spots (believed to be due to surface inclusions) of a different color. After

one week, samples exposed using method (b) had weight gains of 2-3 mg/dm². Samples exposed for two weeks using method (c) had weight gains of 2-4 mg/dm².

It is believed these results demonstrate the importance of low oxygen concentration. Other test variables are being investigated.

The closed carboy system was also used to test under oxygenated (about 35 ppm) conditions to serve as a check on system and procedure. After one week, samples were for the most part covered with a dull gray film and the 304 sample showed severe spalling. Weight changes are as follows:

<u>Material</u>	<u>Wt. Change mg/dm²</u>
304	-103
316	+133
347	+ 24

These results are comparable with those obtained in previous tests.

It is of interest to note here that G.E. (Pearl et al.) does not observe spalling in oxygenated systems providing the stoichiometric quantity of hydrogen is also present. However, the amount of metal corroded, as determined by stripping technique, is not affected by the presence of hydrogen.

A section of a BORAX-V fuel assembly, of which the individual plates were stainless steel wire brushed (to facilitate brazing) has been exposed to oxygenated superheated steam at 650°C for 41 days. There was no effect of this treatment, as compared with pickled plates.

d. Recombiner Tests - The test cartridge holder for the H₂-O₂ recombiner test rig has been modified to allow placement of various catalyst-demisters without the damage or loss of catalyst previously experienced. All sub-systems except the Knallgas generator are completed and ready for testing, and the Knallgas system is nearly complete.

e. Continuous Chloride Analyzer - The continuous chloride analyzer is being calibrated and evaluated. Information to date has shown: (1) that the instrument is sensitive to temperature changes as low as ±5°F and must be kept in a controlled temperature environment; (2) that the instrument is sensitive to changes in chloride concentration as low as 0.05 ppm; and (3) that stability is fair for one-hour periods.

f. Advanced Superheater Fuel - As part of a program for testing improved superheater fuel materials, an addition was made to the original superheater fuel plate fabrication contract with Atomics International to cover fabrication development work on cermet fuel plates made of UO_2 dispersed in a matrix of Type 406 stainless steel and clad with Type 406 stainless steel. Six developmental plates, using depleted UO_2 , have been fabricated and are now being destructively tested.

II. LIQUID METAL COOLED REACTORS

A. General Research and Development

1. ZPR-III

a. Experimental - Construction of Assembly 40 has been initiated. This assembly is an 8-region cylindrical reactor. It is somewhat unique in that it contains a partial density axial beryllium reflector and a full density radial beryllium reflector beginning 9 cm from the outer core radial periphery. Region I (core) is an approximate cylinder, 28 cm long by 28 cm in diameter. A section view cut along the cylindrical axis is shown in Figure 1, along with the volume fraction compositions of the various zones.

Figure 1. ZPR-III Assembly 40

Because of uncertainties in reactivity effects associated with passing beryllium through the core (when loading a drawer into the reactor) and when passing core material through the axial beryllium zones (when withdrawing a control or safety rod), the normal experimental approach was somewhat modified. Additional uncertainties associated with reactivity effects of hydrogenous material (people working between the assembly halves) at the core interface prompted further modifications in the normal experimental approach.

(1) The Fat Man Experiment - The limited knowledge of the leakage properties of this reactor did not allow a reasonable prediction of reactivity changes associated with people working between the assembly halves. To assure personnel safety, a fat man experiment was conducted.

A 4 ft x 4 ft x 6 in.-thick slab of polyethylene was bolted to the front face of Half No. 1. Half No. 2 was loaded in increments. Each time an increment of fuel was added, the halves were run together and the

sub-critical count rates from three different counters were recorded. A plot of the inverse count rates was used to determine the reactivity effects of the hydrogenous moderator. After 20 percent more fuel was loaded in Half No. 2 than was expected in that half for the completed assembly, it was still far subcritical with the polyethylene up against the assembly face (the halves driven together). This experiment was conducted with no beryllium in the axial blanket.

(2) The Critical Approach - The reactor was initially brought critical with no beryllium in either of the axial blanket regions. The critical mass of this configuration was approximately 95 kg of U^{235} .

The reactivity effects of replacing core material with Zone V material was then determined along the core axis and also axially near the core radial periphery ($r = 11$ cm). The results were negative in all cases, showing that one drawer containing beryllium in Zone V could be safely added at a time.

Beryllium was then added in Zone V of the four drawers surrounding a central control rod. The reactivity worth as a function of position of this rod was then measured. The results indicated no net positive reactivity effects associated with passing core material through the beryllium of Zone V. This procedure was extended to include all of the control-safety rods of one half of the reactor both singly and ganged.

The situation in which all the safety rods simultaneously pass through the axial beryllium blanket, Zone V, as on a scram, was considered quite uncertain and potentially hazardous. In this situation, the enriched material of the core safety rods must pass through the partial density beryllium, momentarily forming a lattice configuration which has uncertain multiplication properties. Measurements of the reactivity while slowly bringing the rods into this configuration proved, however, that all possible systems were substantially subcritical.

b. Theoretical - Two quadratures, the modified single Gauss (MSG) and the thin region integral method (TRIM), for use with the discrete ordinate method of solution of the transport equation for thin binary slab cells were reported in ANL-6573 (Progress Report, May, 1962).

Criteria for determining a priori the suitability of double Gauss (DG), MSG, or TRIM quadratures were considered. Quadrature choice depends upon the regional thicknesses in mean free path units and upon the number of discrete angles to be employed. Whenever feasible DG or MSG quadratures are preferable as the parameters are already available for the former and easily obtainable for the latter by transformations of available single Gauss parameters. TRIM quadrature parameters must instead be specifically calculated for a given cell.

Results, for the case of a binary cell having regional thicknesses A and B, in mean free path units, indicate that MSG or DG quadrature is suitable if there is a discrete angle

$$|\mu_\lambda| \lesssim \frac{1}{2} \left(\frac{AB}{A+B} \right)$$

among the quadratures $N/2$ values of $|\mu_\lambda|$. A TRIM quadrature is otherwise indicated.

2. ZPR-VI and ZPR-IX

a. ZPR-VI - The wiring of the ZPR-VI facility was completed. All the control and interlock circuits except for those pertaining to the dual-purpose rod drives were checked out. Modifications to the piston housing of the dual-purpose rod drive mechanisms is in progress.

The hazards summary report for ZPR-VI is being printed.

b. ZPR-IX - Wiring of the ZPR-IX facility is in progress. The control rod drive mounting plate and matrix supports have been installed. Bids for the fabrication of the aluminum drawers are being considered.

c. Miscellaneous - The faulty "No BAC" bearing assemblies for the two loading platforms were reworked by the vendor and returned. They were installed in the loading platform drive mechanism and the platform appears to be working satisfactorily.

The reactor cell for ZPR-VI (Cell No. 5) was sealed and pressurized to 10 psig. At this pressure the leakage rate was found to be about 5% of the total volume per 24 hours. This appears to be about the best leakage rate attainable with this type of construction.

Installation of the emergency venting system for the two reactor cells has been started by the contractor.

Bids are now being received for the installation of the argon gas purging system.

Electrical wiring installation drawings for the control of the reactor cell ventilating system were completed. Bid invitations for this work will be sent out during July.

B. EBR-I

1. Mark III Operation

Four thimbles were installed in the reactor core for the irradiation of notched impact samples of SA212B steel being used in irradiation damage studies. These will allow contact of sample capsules with reactor coolant. These irradiations will provide quantitative data concerning property changes as a direct function of exposure in a high energy neutron flux. Irradiations were made at full power and low inlet temperature. Ten samples were irradiated for exposures varying from 2 to 50 Mwh of reactor operation. The irradiations have been completed and the specimens are being prepared for return to Argonne, Illinois, for inspection.

Two Mark III fuel rods, having the longitudinal positioning ribs sheared off and with a marked bow in the fuel sections, were removed from storage and returned to the Metallurgy Division for inspection.

Modification of the EBR-I hot cell was begun. This will provide improved ventilation for this cell and the special materials vault and will allow the installation of improved manipulators.

2. Fabrication of Core IV Fuel

Fabrication of Zircaloy-2 tubing, 0.080 in. O.D., for the EBR-I Core IV blanket thermocouple rods has been completed. The final tubing lot, consisting of 21 tubes, is ready for eddy current evaluation and final visual inspection.

C. EBR-II

1. Reactor Plant

Major activity in the reactor plant continued with completion of additional miscellaneous tasks and work in preparation for filling the primary system with sodium.

a. Helium Leak Testing - The primary heat exchanger was successfully leak tested preparatory to installation. Helium leak testing was successfully completed on the last of the cover lock mechanisms.

b. Primary Tank Heaters - The modifications to install an additional 320-kw of electrical feeder capacity for the installed heaters is continuing.

c. Flexible Seals - These two assemblies, required for secondary system piping penetrations through the reactor building, were installed and tested. Minor additional work is being done in connection with the seals including the completion of insulation and connection to secondary system piping external to the reactor plant.

d. Nuclear Instruments - Checkout using a 30-curie neutron source was completed.

e. Storage Basket - Tests were conducted to determine the degree of accuracy of shaft alignment when the basket is loaded eccentrically. It has been determined that as many as ten subassemblies loaded in such a manner as to produce maximum eccentricity of loading can be accommodated. The storage basket normally will be loaded uniformly and it is highly unlikely that eccentric loading conditions similar to these will be met in actual plant operation.

f. Primary Tank Cleanup - Cleaning inside the primary tank preparatory to sodium filling is continuing.

g. Filtering Subassemblies - The filter loading of the reactor was completed. The reactor loading for sodium filling consists essentially of a standard outer blanket loading with the exception of one special filling filter subassembly which is incorporated to filter the sodium as it flows into the outer blanket region as the tank is filled. This subassembly will be removed before any flow of the bulk sodium is started. The inner blanket and core regions will be mostly special elements. The inner blanket region is filled with fifty-one plugged dummy subassemblies and fifteen circulating filter subassemblies. The core region contains one special filling filter subassembly as in the outer blanket, twelve circulating filter subassemblies, thirty-two plugged dummy subassemblies, and sixteen natural uranium subassemblies to be used for wet critical.

h. Large Rotating Plug Baffles - Recent testing of the shield cooling system indicated that flow distribution through the various inlets at the top of the biological shield could be improved by the addition of air baffles around the periphery of the large rotating plug. This work was completed.

i. Thimble Cooling System - Parts of the filter bank isolation dampers are being reworked to provide more effective closure. This improvement will be advantageous during the changing of filters at which time the filter elements will have been removed from the system and the dampers will be required to seal off the thimble cooling system.

j. Backup Bellows - A set of bellows was added to the auxiliary pump to augment the bellows originally installed. A similar unit has been added to the outlet flowmeter. This should result in greater operational reliability of the units.

k. Sodium Surveillance Periscope - The purpose of this device is to permit observation of certain areas in the primary tank during the sodium fill period, and for a limited time thereafter when the sodium is being circulated for cleanup. This instrument was completely installed.

l. Fuel Handling System - Final electrical and mechanical check-outs of the entire system have been started. This work includes a thorough check of modifications, normal and special operations, and operation during simulated electrical and mechanical malfunctions

m. Fuel Unloading Machine - The test installation was completed of the three helically coiled cables, through which electrical connections are made to the refueling machine. Installation of the permanent wiring of the control panel, cable connectors, etc., is nearing completion also.

The transistorized synchronism-detection circuit for the gripper and sensing chains has been installed and is operating. This circuit detects loss of synchronism among the gripper and sensing chains. Operational tests of the modified gripper in sodium were performed satisfactorily.

Current work on this machine includes the final test assembly and alignment prior to shipment to the Idaho Site. Final test assembly is also being performed on the gripper assemblies, shield components, carriage, and the temporary heating and cooling loop.

n. Argon-Cooling System - Emergency circulation of argon in the argon-cooling system is provided by a battery-powered standby blower. A preliminary endurance test has been run on the battery in which it was used to power the fully loaded 5-hp blower motor. The unit was run continuously for one hour during which the battery voltage dropped from an initial value of 129 volts to 100 volts. This exceeds by an ample margin the design requirement for $\frac{1}{2}$ -hour operation. The unit will be tested again after assembly of the final argon ducting.

2. Effects of Coolant Flow Blockage in EBR-II Outer Blanket Subassemblies

An investigation was made of the effects of coolant flow blockage in the EBR-II outer blanket subassemblies. Complete flow blockage of the subassemblies was assumed, such as might occur due to plugging of the inlet coolant flow passages, with stagnant sodium filling the coolant channels.

The maximum uranium (or uranium-stainless steel interface) temperatures were determined in subassemblies in various locations in the outer blanket and in various configurations or arrangements of blocked subassemblies. The results are shown in Figure 2 with maximum temperature plotted against row number (reactor rows 8 through 16 comprise the outer blanket). Five combinations of blocked subassemblies are plotted ranging from a single subassembly to three adjacent rows of subassemblies.

Figure 2. Maximum Temperatures Resulting from Complete Blockage of Coolant Flow in EBR-II Outer Blanket Assemblies

The following assumptions were made in developing these data:

(a) The effective thermal conductivity of the blocked subassemblies is $22.3 \text{ Btu}/(\text{hr})(\text{ft})(^\circ\text{F})$, which is obtained by weighting the conductivities of the uranium, sodium, and steel constituents of the subassemblies by their respective volume fractions. (This procedure assumes that the coolant channels in the blocked subassemblies are completely filled with stagnant sodium.)

(b) Power densities, total powers, and coolant flow rates in the affected subassemblies are for full power operation of the reactor (62.5 Mwt). The values taken for each subassembly row pertain to the hottest subassemblies (those closest to the center of the reactor) in the row.

(c) The blocked subassembly configurations were simulated by equivalent cylinders or infinite slabs, whichever was relevant. The average power density in the equivalent configuration was assumed to be uniform throughout that configuration. (A study using the infinite slab geometry showed that use of this last assumption produces somewhat higher temperatures than would be produced by using the actual exponential power distribution. Such is undoubtedly also the case with the cylindrical geometry.)

(d) All heat from each blocked subassembly configuration goes into the cooled subassemblies immediately adjacent to the blocked subassemblies, thereby increasing the average coolant temperature rises in the adjacent subassemblies. The temperature rise within the blocked configuration at the hot spot, calculated using assumption (c) is added to the average of the increased average coolant temperature rises, also to the hot spot, in the adjacent cooled subassemblies, and that sum is then added to the reactor inlet temperature to obtain the temperatures given in Figure 2. This use of average coolant temperature rises appears justified because in the cooled subassemblies adjacent to the hottest side of the blocked configuration, the local coolant temperature rise near the blocked configuration is normally lower than the average coolant temperature rise. This results from the temperature gradient across the subassembly (highest at the side of the subassembly toward the center of the reactor and lowest away from the center). Thus under normal conditions (with coolant flow in all subassemblies) the lowest temperature coolant within a subassembly is adjacent to the hottest side of the adjacent subassembly.

(e) The reactor is at rated power of 62.5 Mwt.

(f) No uncertainty factors are included.

In Figure 3 six different patterns of blocked subassemblies are shown, each pattern being amenable to analysis by use of Figure 2. The maximum uranium temperature which would result from each pattern is given adjacent to the pattern, these temperatures being derived directly from Figure 2. Other patterns are possible but these are indicative of the amount of coolant flow blockage which can occur without serious consequences.

It can be seen that when the blocked configurations were laid out in the patterns of Figure 3, the configurations were kept separate by two rows of cooled subassemblies. This was done because the Figure 2 values were based on the cooled subassemblies adjacent to a blocked configuration receiving extra heat from that configuration only [assumption (d)]. If two blocked configurations were separated by only one row of cooled subassemblies, that row receives more heat than the assumptions employed in deriving Figure 2 permit and it is no longer valid. However, it should not be inferred from Figure 3 that if two blocked configurations are separated by only one row of cooled subassemblies, unreasonably high temperatures will result. For example, if alternate rows 8, 10, 12, 14, and 16, are blocked simultaneously, the row 8 and row 10 maximum temperatures are only 27°F and 22°F higher, respectively, than the comparable "one row" values given in Figure 2. (Increases to the row 12, 14, and 16, values are even smaller.) Similarly, if alternate rows, 9, 11, 13, and 15, are blocked simultaneously, the row 9 and row 11 maximum temperatures are only 19°F and 25°F higher, respectively, than the comparable "one row" values of Figure 2.

Figure 3. Maximum Temperatures Resulting from Six Different Patterns of Complete Coolant Flow Blockage in EBR-II Outer Blanket Assemblies

3. Power Plant

At the beginning of the month, the job of adjusting the pipe hangers for correct load and relocating some hangers was completed. The drain lines at the northwest corner of the building were repiped to assure proper drainage. The inspection of the motor-driven feedwater pump discharge pressure control valve, the by-pass steam stop valve, and both large and small by-pass steam control valves was completed, and these valves were re-assembled. All are in good condition.

The turbine oil receiver tank was moved into position, and the drain from the turbine oil tank was permanently piped to it.

Full-size globe valves and by-pass valves of $\frac{3}{4}$ -in. size were installed in the turbine oil cooler inlet water lines. Reach-rods to permit operation of these valves from the turbine floor are being fabricated.

The modification of the air ejector suction valves is almost complete, now awaiting only completion of the installation of a remote drive for the north valve.

4. Sodium Boiler Plant

Insulation of steam generator components and sodium piping, application of induction heater wire, coating miscellaneous ferrous surfaces, and installation of thermocouples continued. Also in progress was the painting of evaporators and piping as the insulation was applied. Electricians worked on the wiring of motor-operated drain valves, and a small amount of work was accomplished by the pipe fitters on the surge tank modification packet. Welding of the 12-in. sodium lines to the flexible seals of the reactor building was started after careful alignment and adjustment of the cold spring. A 2250 psi hydrostatic test of the steam system and a 2750 psi test of the feedwater lines were completed as per specifications, and the system was left filled with treated water containing 2000 ppm sodium chromate at pH 8.5.

5. Fuel Cycle Facility

a. Construction and Installation - Installation of process and auxiliary equipment and correction of construction deficiencies were continued in the Fuel Cycle Facility. The delay tank for storage of off-gas from melt refining was provided with steel shielding, and activated carbon was placed in the tank to increase the retention of radioactive isotopes. The inert atmosphere cell cooling systems were tested; some valves in both the cooling and gas circulating loops require repair or replacement, but system operation appears to conform with design. The hydraulic and mechanical systems for the large air-to-inert atmosphere cell transfer lock were installed and tested; operation was generally satisfactory, but minor piping and safety system modifications will be necessary.

The tank sections of all cell shielding windows have now been installed, and removable slabs were installed in four of the windows. It was found that for many of the window frames the intermediate removable slabs might be unstable, due to irregularities and slope of the ledge upon which they sit. It is proposed to correct this by providing feet on the outer edge of the slab. All but one of the eight electromechanical manipulators have now been placed in the cells; installation of the last one will be used to provide a photographic record of the remote technique. Minor mechanical and electrical difficulties continue to be encountered with these manipulators and with the in-cell cranes; corrections are being accomplished. The manipulators were used to install all 65 service sleeves in the argon cell by procedures essentially simulating those which will be used in hot operations. Complete installation of one of the melt refining furnaces and its accessories was also accomplished remotely.

Construction and installation of valve cabinets and other sub-cell equipment was continued; and that equipment required for injection casting, pin processing, and welding is almost complete. Bases for the two argon gas compressors were poured, and the compressors and storage tanks were set in place.

Equipment for determining carbon in sodium and fission was obtained, and testing and development of procedures is under way. New procedures for rhodium and rare earths are being tested. Glove boxes for analytical work were set in position in the laboratory. Study of the failure of a bronze nut on one of the junior cave shielding door lifting mechanisms was continued. The main difficulty seems to be poor workmanship in the original fabrication; the jackscrew was badly grooved on the thread lands, the threads were slightly irregular, and the screw was bowed about $\frac{1}{8}$ in. Examination of the other nut on the same door indicated it to be in fairly good condition, while threads on a nut from a second (and smaller) door showed serious deformation. Stronger aluminum and manganese bronzes were obtained, with a view to making replacement nuts for tests; and consideration is being given to replacing the jackscrews as well as the whole lifting mechanism.

Arrangements have now been completed for procurement of the major portion of fuel assembly structural components and expendable process materials which are expected to be required for initial phases of Fuel Cycle Facility operations. Significant quantities of these materials have already been received. The last order placed was for 26,000 fuel element tip castings. Other assembly hardware is being obtained in sufficient quantity to provide for about 200 complete fuel and control assemblies. About one-third of an order for 26,000 Vycor glass molds has been delivered, and graphite and zirconia crucibles, fume traps, ingot molds, and make-up alloying ingredients are also being obtained.

b. Development of Remote Control Methods and Equipment for Fuel Fabrication - Design work has been completed on all major items of in-cell refabrication equipment for the EBR-II fuel. During the reporting period the drawings and specifications were completed on a "Universal Fuel Element Assembly Machine." This machine is essentially a jig which holds all components of the fuel element in accurate alignment as the fuel rod bundle is assembled by master-slave manipulators. A sequence programmer and electrically controlled mechanical interlocks prevent incorrect assembly. After attachment of the fuel bundle to the lower blanket assembly, the outer hexagonal stainless steel tube is forced down over the fuel rods and welded to a lower adaptor by means of six automatic welding guns. This machine differs from previous machines in that it may be adapted to produce all four fuel element configurations used in the EBR-II Core I.

Design work was also completed on the furnace charge preparation table, the furnace pallet loader and unloader, sample collecting apparatus, crucible tongs and various other special tools required in the process cell.

The problem of producing noncontaminated castings in molds for test of the fuel pin processing machine was solved by use of an 89 w/o Cu-10 w/o Al-1 w/o Fe bronze. This melts at approximately the same temperature as the fissium alloy. Several hundred bronze pins were cast in Vycor molds using the furnaces to be installed in the fuel cycle cell. These were left in the molds to test the demolding, shearing, weighing, dimensioning and transfer functions of the pin process machine. The vendor supplying the pin process machines has given us a new delivery date of July 31.

The bonding and settling machines were received and are under test. In these machines, a spring suspended mass oscillates vertically at a frequency of 30 times per minute. This mass strikes a series of small hammers, 50 in the bonders and 20 in the settlers. The mass is driven by solenoid and, because the frequency varies somewhat with the change of load on the machine, an electrical escapement is required to time the solenoid pulse in proper phase with the natural frequency of the system. Two circuits have been tested; one employs a proximity sensing impedance coil, the other circuit uses a mechanical switch to sense the position of the mass. The switch circuit proved to be simpler and less subject to drift. It was chosen for the development of the final driving circuit.

Pushbutton stations and electrical controls for the injection casting furnaces, pin process machines and welding machines were completed and shipped to Idaho during the reporting period. Circuit improvements have been made in the eddy-current testers with the addition of an oscilloscope display to the strip chart record display. The scope display has been found necessary to monitor pulse shape. It has also been found effective for observation of defects.

6. Training

Twenty hours of training to provide technicians assigned to reactor operations with a detailed knowledge of reactor control, fuel handling, and power plant systems were completed.

Forty hours of classroom work were held in the Reactor Technology Course to provide the 13 new technicians with general nuclear information.

Additional technicians have reported to work in both the analytical and operating groups. This substantially completes the non-professional staffing of the Fuel Cycle Facility planned for initial operations.

7. Process Development

a. Melt Refining Process Technology - The fifth high-activity level melt refining experiment has been completed. The charge consisted of 382 grams of uranium-five weight percent fission irradiated to about two total atom percent burnup and cooled for 25 days. These conditions closely approach the maximum fuel irradiation levels expected for the first core loading of EBR-II. Information on the yield and fission product removals is not yet available.

Preparations are being made to investigate more thoroughly the behavior of fission product iodine in the melt refining process. Inactive and low-activity-level materials will be used in these studies.

In two-kilogram-scale experiments, the storage of unirradiated EBR-II fuel pins at 650°C in a five percent nitrogen-argon atmosphere was found to have a negligible effect on the melt refining yield. These storage conditions simulate the conditions which exist after fully irradiated EBR-II fuel pins have been charged to a melt refining crucible and prior to sealing the bell jar of the melt refining furnace and pumping out the trapped cell gas. The effect of radiation on the nitridation of fuel pins may be very important at this temperature since swelling of the pins (and consequently an increase in surface area) occurs as a result of the pressure generated by trapped fission product gases.

b. Skull Reclamation Process - The skull reclamation process requires the heating and mixing of molten metal-salt systems at temperature up to about 800°C. Plant-scale runs have demonstrated the feasibility of low-frequency, 60-cycle inductive heating. The feasibility of inductive mixing has not yet been demonstrated quantitatively; however, visual observation has revealed that inductive agitation of metal-salt systems does occur.

One demonstration run of the skull reclamation process in an inert atmosphere glove box has been completed. Uranium losses in waste streams amounted to about three percent. These losses are sufficiently low for the EBR-II fuel cycle since only ten percent of the total fuel will be processed by the skull reclamation process. However, it is believed that the losses can be reduced.

Initial results on fission product behavior indicate that noble metals (molybdenum and ruthenium) were removed to a lesser extent in the noble metal extraction step in the new process equipment installed in the inert atmosphere box than in previous demonstrations of the noble metal extraction step. Poor mixing of the zinc and the flux phases in the new equipment is believed to be the major reason for the decreased removals.

In the skull reclamation process, all phase transfers by pressure siphoning have been carried out without difficulty with the exception of the removal of the zinc-rich supernatant solution after precipitation of the U_2Zn_{17} intermetallic compound. Removal of this supernatant has been easily effected at 550°C but not at lower temperatures (475° or 500°C) which may be desirable for the purpose of decreasing the solubility of uranium. The reason for the difficulty at the lower temperatures is being investigated.

Design and procurement of equipment for large-scale ($2\frac{1}{2}$ kg of uranium) demonstration of the skull reclamation process is proceeding satisfactorily. A prototype transfer line fabricated of a 30 percent tungsten-70 percent molybdenum alloy has been constructed and tested. Nine transfers of molten zinc at temperatures ranging from 650° to 850°C were made without difficulty and without visible change in the transfer line. Molten zinc in 8-kg quantities at temperatures up to 750°C has been rapidly transferred into thin wall (25-mil) stainless steel containers. This operation simulates the most severe waste handling and disposal problem encountered in the skull reclamation process. No serious attack of the stainless steel containers occurred.

A beryllia crucible supported in a stainless steel secondary with the annulus filled with beryllia powder continues to perform satisfactorily as a container for the last three steps of the skull reclamation process (two precipitation steps and the retorting step). A thixotropically cast beryllia crucible has undergone no serious deterioration through two uses.

c. Blanket Processing Studies - The fifth of a series of blanket process demonstrations runs was completed. The blanket process involves dissolution of plutonium-bearing blanket material in a zinc-rich alloy, precipitation of uranium by addition of magnesium, and removal of the plutonium in the supernatant phase which is then boiled down to produce a plutonium concentrate suitable for feeding into the core cycle. The run, which provided the best results to date, resulted in a plutonium separation factor (from uranium) of 230 as compared to a previous high value of 160. A 92 percent removal of the plutonium-bearing supernatant solution was realized instead of the desired 95 percent or greater, but this desired removal should be easily achieved in large-scale operation. In a separate study of the efficiency of removal of the plutonium-bearing supernatant solutions, high removals (about 95 percent) continue to be demonstrated.

d. Plutonium Recovery Process - Distribution coefficients of uranium, plutonium and several rare earth elements between zinc-magnesium solution and molten magnesium chloride at 800°C were reported previously (Progress Reports, April and May 1962, ANL-6565 and 6573) The distribution coefficients of thorium were determined

during June for the same system. The results are given in Figure 4. The distribution coefficients were found to fall between the values for uranium and plutonium. Distribution data for praseodymium and plutonium between cadmium-magnesium alloys and equimolar lithium chloride-magnesium chloride at 650°C (Figure 5) showed a separation factor of about 33 at 100 percent magnesium concentration. This may be compared to the study of the distribution of the same elements between zinc-magnesium alloys and magnesium-chloride at 800°C (Figure 6) wherein a separation factor of 21 was obtained in pure magnesium. The difference in the values of the separation factors is attributed to a temperature effect, since variations in composition in the lithium chloride-magnesium chloride system are found to have no effect on the separation factor for plutonium and praseodymium over the range of 25 to 100 mole percent magnesium chloride. Codistribution data for uranium and cerium between magnesium-zinc alloys and equimolar potassium chloride-magnesium chloride at 800°C were also obtained. The results are shown in Figure 7. The results are in agreement with previous data obtained with the individual elements (see Progress Report, April 1962, Figure 4, page 27).

Temperature: 800°C
Mixing Rate: About 300 rpm
Crucible: Tantalum
Atmosphere: Argon

Figure 4. Distribution of Thorium Between Zinc Magnesium Alloy and Magnesium Chloride

Temperature: 650°C
Mixing Rate: About 300 rpm
Crucible: Tantalum
Atmosphere: Argon

Figure 5. Distribution of Praseodymium and Plutonium Between Cadmium-Magnesium Alloy and 50 w/o Lithium Chloride-50 w/o Magnesium Chloride Salt

Figure 6. Distribution of Praseodymium and Plutonium Between Zinc-50 w/o Magnesium and LiCl-X m/o MgCl_2 Flux

Figure 7. Codistribution of Uranium and Cerium Between Potassium Chloride-Magnesium Chloride and Magnesium-Zinc Solutions at About 800°C

e. Fused Salt Study - The spectrum of uranium pentachloride suspended in petrolatum shows the characteristic peaks in common with a species tentatively identified as uranium(V) which is formed when the higher uranium oxides are dissolved in molten chloride media.

f. Materials and Equipment Evaluation - Corrosion tests made under noble metal extraction conditions (zinc-flux medium at 800°C) for periods of 400 and 500 hours, with the corrosion specimens attached to an agitator shaft, indicated very little corrosion of tungsten. Weight changes of the samples ranged from a loss of 0.22 percent to a gain of 0.078 percent and dimensional changes were less than the detectable limit of $\frac{1}{2}$ mil. Samples of molybdenum-30 percent tungsten alloy, which were exposed under the same conditions, lost 3.8 percent in weight but suffered no detectable dimensional changes. Samples of tungsten-0.8 percent iron-0.4 percent nickel alloy were severely attacked.

g. Sodium Removal from Fuel Subassemblies - Irradiated fuel subassemblies on discharge from the EBR-II reactor will be coated with a thin film of sodium. A process for the removal of the sodium coating has been tested in the laboratory. This process depends upon the conversion of the sodium to the hydroxide by means of nitrogen gas containing small amounts of oxygen and water vapor. The hydroxide is removed by means of a water wash. The laboratory experiments have confirmed the adequacy of the method.

D. FARET

The general engineering and physics parameters of an experimental facility to test the characteristics of advanced fast reactors are being examined in detail (see also section III E 1).

One problem of interest is the design of an experimental fuel element for Doppler coefficient studies. This element should permit controlled changes in the fuel temperature with a minimum of structural side effects. Preliminary calculations indicate that a low conductivity, or thermally insulated, fuel could have such characteristics. To refine these estimates, representative sections of fuel elements are being fabricated for heat transfer measurements. The first fuel tube assembly was filled with UO_2 to approximately 65% theoretical density using argon gas in the voids. A calibrated platinum heating wire, within the fuel, was used as a line heat source along the central axis. The external surface temperature of the fuel cladding was held constant at 25°C in a water bath. After a central temperature of 980°C had been reached (corresponding to a $\int k d\theta$ of 12.5 w/cm), the heating wire burned out. A radiograph revealed that the platinum wire was off center within the fuel tube probably causing a short to the clad. More fuel tubes will be made to investigate the conductivity of low density fuel elements.

III. GENERAL REACTOR TECHNOLOGY

A. Applied Reactor Physics

1. Neutron Scattering

a. Neutron Elastic Scattering from Iron - Further results of the total neutron cross section σ_t and the total elastic neutron cross section σ_{el} from natural iron are reported. The measurements were made with the neutron-gamma discrimination detector. Only elastically scattered neutrons were detected. The absolute elastic cross section was determined by measuring the scattered neutron intensity from iron relative to the scattered neutron intensity from carbon. The carbon differential scattering cross sections of Langsdorf, et al.,¹ were used as the standard.

The results for σ_t and σ_{el} for iron for incident neutron energies between 1000-1100 keV are given in Figure 8. Strong inelastic scattering ($\sigma_{inel} = \sigma_t - \sigma_{el}$) is evident between 1000-1020 keV and between 1065 and 1095 keV. The region around 1050 keV shows little or no inelastic scattering. Angular momentum and parity consideration suggest that the region around 1050 keV is dominated by S wave scattering and that the resonances between 1000-1020 keV and between 1065-1095 keV are dominated by P and/or D wave scattering.

Figure 8. Total and Total Elastic Cross Section of Natural Iron

b. Fast Neutron Scattering from Gold - The elastic and inelastic scattering of fast neutrons from gold has been studied experimentally. The measured differential elastic scattering cross section expressed in terms of Legendre polynomials is shown in Figure 9. When account is taken of neutron captures and inelastic scattering, good agreement with

¹R. O. Lane, A. S. Langsdorf, Jr., J. E. Monahan, and A. Elwyn (Private Communication).

the measured total cross section is obtained. Concurrent with the elastic scattering measurements, studies of inelastic scattering to the three lowest excited states in gold were carried out. The results for scattering to the first level at 77 kev and the second and third levels comprising a doublet at about 270 kev are shown in Figure 10. The measured excitation function of the first level is in good agreement with calculations employing Hauser-Feshbach theory. The excitation of the 270 doublet, as measured in this work, has the same shape dependence on incident neutron energy as measurements reported in the literature. However, the absolute value of the doublet cross section measured here considerably exceeds that previously reported. The measurements are being continued in order to provide more detail over a wider incident neutron energy range.

Figure 9. Elastic Scattering from Gold

Figure 10
Inelastic Scattering from Gold

2. Lithium Target Physical Characteristics

The higher resolution elastic scattering experiments require neutron sources with narrow energy spread (2 to 10 kev). Tests have been initiated to determine what lithium-containing material and what target structure will give the highest yield and longest lifetime, yet still maintain the required resolution.

The first test was a comparison of lithium metal with lithium fluoride. The theoretical stopping power of the fluoride is 3.2 times as great as that of the metal at 1.9 Mev. However, apparently due to contamination, the thin metal targets do not give full yield.

A series of targets was prepared from each material ranging between 1.6 and 50 kev thick. The thickness of each was measured by measuring the initial rise. The total yield was measured at 2.1 Mev proton energy.

It was found that for thick targets (15-100 kev) the experimental stopping power of protons was very nearly three times as great for LiF as for lithium metal. When thinner (1-10 kev) targets were tested it was found that the neutron yield (counts/target thickness) for lithium targets dropped by a factor of two. This drop is most easily explained by contamination effects. It was also found that LiF could be used to make the thinner targets with much more reliability and reproducibility than the lithium metal. Therefore, it appears that unless the loss of yield of the lithium metal targets can be eliminated the more easily handled lithium fluoride is preferable. Other tests will be made to determine what material holds up longest under the proton beam as well as what makes the best target backing.

3. Fission Spectrum of Cf²⁴⁶

The fission fragment kinetic energy distribution was measured for Cf²⁴⁶ relative to Cf²⁵². The preliminary results are:

	<u>Cf²⁴⁶</u> <u>(Mev)</u>	<u>Cf²⁵²</u> <u>(Mev)</u>
Most probable energy of light fragment	114	107
Most probable energy of heavy fragment	84	79
Most probable total energy	198	186

4. Cross Section Measurements

Measurements of the absolute cross section of the Ni⁵⁸ (n,p) Co⁵⁸ reaction were made in the region of 1-2.7 Mev neutron energy. The absolute

neutron flux was measured with a U^{235} fission counter. Absolute yields of Co^{58} were determined by counting the 800-kev gamma ray of Co^{58} in coincidence with the associated annihilation radiation. Preliminary results are shown in Table III.

Table III. Absolute Cross Section of the
 Ni^{58} (n,p) Co^{58} Reaction

<u>E_n (Mev)</u>	<u>σ(mb)</u>	<u>E_n (Mev)</u>	<u>σ(mb)</u>
1.04	1.2	1.67	17.9
1.15	3.8	1.87	31.0
1.25	5.6	2.07	48.7
1.36	5.1	2.37	91.4
1.46	10.6	2.67	142.0
1.57	11.9		

The neutron total cross sections of dysprosium (several of the rare earth elements including dysprosium have been proposed as possible reactor control materials) have been measured in the energy range 13 to 210 kev. This work complements other work in this program which has been concerned with determination of fast capture cross sections of other rare earths and elements of interest to the fast reactor program. The total cross sections are given as a function of energy in Figure 11. The standard deviations of the cross sections relative to each other are about ± 1.5 percent. The neutron energy spreads at each point are about ± 1.5 kev.

Figure 11

Neutron Total Cross Section of Dysprosium

5. High Conversion Critical Experiment

Data were obtained for the capture cadmium ratio and initial conversion ratio in the Hi-C, 1:1, H:28 core loading (Progress Report, May, 1962, ANL-6573). An attempt is being made to determine the

thermal utilization. The data obtained earlier for the fast fission factor in this high fuel density core yielded a value of 1.127.

A procedure has been developed for weight-correcting the foil data from flux traverses, where it is impractical to get a large number of matched foils. The method will be applied to the data from the gold and indium Hi-C traverses. This procedure takes into account the perturbations due to thermal flux depression, thermal self-shielding, and resonance self-shielding. The procedure was applied to one gold traverse and showed considerable improvement over the conventionally weight-corrected data. An interesting conclusion is that the thickness of a foil need not be particularly uniform, even for resonance activation of gold or indium, provided that its mass and area are well known.

The calculations involved in the full procedure are somewhat long. An approximation is now being tested for gold, wherein the assumption is made that 1-mil gold is very thick for resonance (4.9 ev) neutrons, and very thin for thermal neutrons.

6. Argonne Thermal Source Reactor (ATSR)

a. Flux Perturbation Measurements - Flux perturbation measurements reported in ANL-6544, Progress Report, March, 1962, were extended to include other important aspects of this problem. The 20-cm cubic void in the graphite block at the ATSR leakage face was surrounded by cadmium on all sides except the one facing the reactor. Gold foils, 1 cm² in area and of different thicknesses, were activated at its center, with the foils placed both perpendicular and parallel to the leakage face. Thus, the foils were in effect irradiated by a 20 x 20 cm leakage face at 10 cm distance. The effect of the angular positioning of the foil was observed to be as large as 20%. The self-shielding was stronger or weaker than in the isotropic case, depending on the orientation of the foil.

In order to establish the influence of the aluminum covers and the foil holder now in use (an arrangement of aluminum wires), five of these were redone in a modified arrangement. Small holes were drilled in the foils, and then the bare foils were suspended on 0.0025 cm thick wire in the center of the void. None of the results obtained in this way differed by more than 1.5% from the results of measurements with the foil holder used previously.

A further series of foil measurements was performed, with the 20 cm cubic cavity filled with graphite and surrounded with Cd on five sides. Some deviations from the isotropic case were found, especially for the case when the foil is placed perpendicular to the reactor face. This produced a significant increase of self shielding.

The results of these recent measurements of flux perturbations by foils were communicated to the other workers in this field at a recent ANS meeting. It is likely that these data will be of use for testing a method of transport calculations of flux depression reported at the meeting.

Some of the measurements will be repeated in the original or a slightly modified way in order to confirm the results. The measured irregularities for foils thinner than 50 microns in a strong flux gradient in graphite were qualitatively reproducible, but the results seem to be very sensitive to the experimental arrangement, and the best way for extrapolating to zero foil thickness has yet to be found. There is much hope that these problems can be solved when it will be possible to irradiate more foils in a more efficient way in JUGGERNAUT.

b. Resonance Integral Measurements - The infinite dilution resonance absorption integrals for tungsten and molybdenum have been determined relative to a gold standard value of 1558 barns. The measurements were performed inside a 0.102 cm thick cadmium tube, having an effective cutoff energy of 0.56 ev, located in the center of ATSR. Data from measurements on teflon and carbon were used to estimate errors introduced by moderation effects of the samples. Table IV compares the values obtained in these experiments with other determinations. The tungsten value is in good agreement, but the value of the molybdenum integral is higher than that reported previously. In the case of molybdenum the correction for slowing down was of the order of 1%.

Without additional experimental study, the reasons for the difference in the molybdenum values are not apparent.

Table IV. Infinite Dilution Resonance Integrals (barns)

<u>Element</u>	<u>Measured Values</u>		<u>Calculated Values</u>
	<u>Present Work</u>	<u>Others</u>	
Gold	1558 (ref)	-	-
Molybdenum	32.1 ± 3.1	20 ± 3.8	22
Tungsten	367 ± 33	340 ± 62	353

7. Theoretical Physics

a. Reactor Physics Data Analysis, ZPR-VII - In connection with the experiments performed with BORAX-V fuel in ZPR-VII, a calculation of material buckling was carried out for a lattice which was quite similar to one of the BORAX-V lattices and which had been previously studied at Babcock and Wilcox. This lattice has been studied extensively both in critical and in exponential experiments. The buckling calculated, using

the GAMI-SOFOCATE combination, was $9.4 \times 10^{-3} \text{ cm}^{-2}$ whereas the experimental value is $8.8 \times 10^{-3} \text{ cm}^{-2}$. Thus, the calculated buckling is somewhat high as was the case for the BORAX-V square and triangular lattices. An attempt is being made to find the reasons for the discrepancy between theory and experiment. One effect under investigation is the progressive hardening of the thermal spectrum as one goes from the unit cell boundary in towards the center of the fuel rod.

b. Feasibility of Pu²³⁹-U²³⁵ Cores to Predict Pu²³⁹ Core Parameters -

In the calculational study reported in ANL-6525 (Progress Report, February, 1962), the dimensions of the core regions were varied uniformly to attain criticality. The core volume fraction of the fissile materials (both Pu²³⁹ and U²³⁵) was assumed constant at 0.05. An effect of this is that a large inventory of U²³⁵ is required both for loading of the composite Pu²³⁹-U²³⁵ system and of the corresponding U²³⁵-fueled system. In practice it might be better to retain a constant core volume and attain criticality by varying the U²³⁵ and Pu²³⁹ fissile materials relative to the U²³⁸ fertile material. A calculational study based on the latter method has been initiated for the case of a much larger plutonium fueled core having a critical mass of approximately 750 to 1000 kg of plutonium compared to 200 kg in the previous study. Again for the composite critical the central region is to be fueled with Pu²³⁹ and the outer region to be fueled with U²³⁵. The central region would contain about 100 kg Pu²³⁹.

B. Reactor Fuels Development

1. Corrosion Studies

a. Aluminum Powder Products - Relatively short creep tests have been performed on three different powder tubes at 290°C, 4000 psi. These conditions of temperature and stress were chosen as representative of a tube in an EBWR type reactor. The creep curves appear to be linear when the strain is plotted against log time. Extrapolating the plots to 1% creep gave times of 1 day for "as atomized" A288 tubing, 30 days for "as atomized" A341 tubing, and over 10,000 days for a tube manufactured from powder ball milled 48 hours. The time of the creep test (about 2 weeks) is much too short to have any confidence in the latter value but the numbers serve to illustrate the improvement. A piece of wrought alloy (A288 type) was observed to creep 5% in less than one hour under these conditions.

Some samples of the ball milled powder tubing are still in good condition after six months corrosion in water at 290°C.

b. Zirconium Alloys for Superheated Steam - Electron diffraction studies of zirconium alloy corrosion films were suspended as effort was diverted to other work. Preliminary diffraction results show some differences between patterns from films formed at 540°C and one pattern of

a film produced in a 650°C exposure. But, since these are "selected area diffraction" patterns and not necessarily representative of the films as a whole, a more extensive survey of these films is needed.

A number of electron photomicrographs were made showing the surface features of the corrosion samples used in the diffraction work.

2. Ceramic Fuels

a. UC-PuC Fuel Development - Effort is being continued on the UC-PuC program specifically on the preparation of UC. The cooperative effort with Chemical Engineering on uranium monocarbide precipitated from a liquid metal media continued. Efforts in the Ceramics Group are directed toward establishing the sintering behavior of this type of carbide and determining the minimum requirements for this type of monocarbide from which a usable fuel body could be made.

One of the most troublesome impurities which has been present in all of the carbide runs thus far is free carbon. The amount has varied from 0.5% to about 2.0%. Since uranium monocarbide contains only 4.8%, these amounts of excess carbon cannot be tolerated if the sintered product is to be the monocarbide.

The carbon powder used in the precipitation of these carbide runs was an activated charcoal with minimum particle sizes below 1 micron and maximum particle sizes up to 100 microns. To determine if the free carbon was mainly present as cores of the larger particles, run No. P6 was classified by sieving into three particle size ranges: +53 μ , -53 μ , +37 μ , and -37 μ . Approximately 80% of the carbide was <37 μ .

Table V lists the C/U atom ratio and percent oxygen of the P6 material before classification and of the coarse and fine fractions after classification.

Table V. Composition of Run No. P6 before and after Classification

	<u>C/U</u> <u>Atom Ratio</u>	<u>Percent O₂</u>
Before Classification	1.21	0.83
+53	1.40	-
-37	1.07	0.98

These results are in agreement with some metallographic examinations of large particles done in the Chemical Engineering Division where the carbon cores have been detected visually.

This would seem to indicate that the carbide is formed by diffusion of the uranium into the carbon particles.

3. Fabrication Development

a. Fabrication of Zirconium Alloy Tubing - During June a number of copper jacketed Zircaloy-2 billets were extruded to provide base stock for subsequent tubing development operations. A copper plating technique has been developed in an effort to improve the as-drawn surface finish of Zircaloy-2 tubing. A 0.283 in. O.D. tube was copper plated prior to drawing through six carbide dies. At 0.194 in. O.D. it was annealed (675°C - 45 min.), replated and drawn through six more carbide dies to 0.119 in. All the tube drawing operations were performed over a ductile core. Throughout the drawing the plating adhered well to the tube surface and prevented die seizing.

b. Preparation of SL-1 Type Specimens for Test in TREAT - Six SL-1 type aluminum-uranium alloy plate specimens have been prepared for metal-water reaction experiments in TREAT. The material was rolled to 0.050 in. from a thickness of 0.200 in. and sheared into $\frac{1}{2}$ in. x $\frac{1}{2}$ in. specimens.

4. Irradiation Studies

Postirradiation Annealing of Th-U Alloys - In 1960 a series of irradiations of cast thorium-uranium alloys was completed. The alloys contained 10, 15, 20, and 25 w/o uranium in thorium. The irradiations were made in uninstrumented capsules in the MTR. Analyzed total atom burnups in the specimens ranged from 0.90 to 4.7 percent, and calculated maximum irradiation temperatures were from 290° to 770°C. Catastrophic swelling did not occur in any of the specimens. The specimen irradiated at the highest calculated temperature, 770°C, showed a swelling rate of only 5.2 percent volume increase per atom percent burnup. At 650°C the rate of swelling was approximately 2.9 percent volume increase per atom percent burnup.

These results were so much better than had been achieved on any uranium-base alloys that it seemed advisable to determine by an independent method whether the alloys were as resistant to high temperature as the irradiation experiments indicated.

Accordingly, a high-burnup specimen from each composition was selected for postirradiation annealing. This approach has the advantage that the annealing temperatures can be measured accurately, whereas the irradiation temperatures could only be calculated. The annealing studies were started at 550°C. The specimens are being held for 100 hours at each temperature. After each anneal they are removed from the annealing furnace, chemically cleaned, and their densities measured by immersion.

Table VI shows the results that have been obtained to date. As can be seen, none of the specimens has begun to swell catastrophically, even when annealed at 700°C. The experiments are being continued to higher annealing temperatures.

Table VI. Volume Increase in Irradiated Thorium-Uranium Alloy
Specimens after Annealing for 100 Hours

<u>Specimen No.</u>	<u>Composition, w/o</u>	<u>Burnup, a/o</u>	<u>Annealing Temp, °C</u>	<u>% Δ Vol. a/o Burnup*</u>
ANL-42-7-1	Th-10 U	4.0	550	0.2
			600	1.0
			650	1.9
			700	2.0
ANL-42-1-2	Th-15 U	4.5	550	0.3
			600	0.6
			650	0.7
			700	0.9
ANL-42-1-3	Th-20 U	4.5	550	0.2
			600	0.7
			650	1.5
			700	1.5
ANL-42-5-4	Th-25 U	4.2	550	0.4
			600	1.5
			650	2.1
			700	1.8

*The volume change is based on the assumption of no weight change and was calculated from the expression $\% \Delta V = [(\rho_i/\rho_f) - 1] \times 100$ where ρ_i and ρ_f are the initial and final densities respectively.

5. Nondestructive Testing

a. Ultrasonic Techniques

(1) A Study of the β to α Transformation in Pu - The primary objective of this program is to determine the mechanism of the β to α phase transformation in plutonium. A definite conclusion as to the nature of this mechanism cannot be made on the basis of existing experimental data. The question is whether the transformation occurs as a nucleation and growth mechanism, which is essentially a diffusion controlled process, or by a martensitic process, where whole groups of atoms move cooperatively, sometimes at rates near sonic speeds. These two processes can be distinguished by means of certain techniques, one of which involves

measuring the time necessary to change from zero to fifty percent transformation at various temperatures. The plutonium is heated to some elevated temperature, one at which the high temperature phase is in stable equilibrium, after which it is suddenly quenched to a temperature at which the low temperature phase is stable. As soon after quenching as possible, the velocity of sound in the plutonium is measured. The velocity in the high temperature phase is radically different from the velocity in the low temperature phase; a difference of 60% is expected. By continuously monitoring the velocity, it should be possible to determine when the phase transformation is 50% complete. This measurement is done at several different quenching temperatures, and the result is a graph of time (to 50% transformation) versus quenching temperature.

Since the measurements are being made on plutonium, remote operation is required and the transducers must be separated from the electronic equipment. Well shaped pulses are necessary for accurate measurements so very little distortion can be tolerated in the echo train. Any operation using long cables from the low impedance generator and receiver to the high impedance transducers does give large distortion and signal loss. Thus, one of the most important problems is the construction of a satisfactory transmitting and receiving system. Other problems involve the satisfactory mounting of the transmitting and receiving crystals and of the sample itself, and the achievement of the proper impedance match between transmitter and transmission line.

(2) Imaging Studies - A new ultrasonic generator has been received which will permit extending imaging studies to frequencies up to 15 mc. Preliminary data indicate that the response of photographic film to ultrasonic radiation, using the iodine solution exposure method, is relatively independent of frequency up to that point. Resolution studies are now in progress, as a function of ultrasonic frequency. Preliminary data indicate that object-film distance will have much more influence on resolution than will the frequency of the ultrasonic radiation. Using reasonable object-film distances (on the order of $\frac{1}{16}$ in.), wires in the order of 0.030 in. have been resolved at frequencies from 0.5 to 15 mc. Similar observations using the continuously-pumped, electronically scanned ultrasonic pickup tube at Northwestern University indicate that wire sizes in the order of 0.005 to 0.007 in. can be resolved at frequencies of 2 to 5 mc.

b. Neutron Techniques - The resolution capabilities of the neutron imaging facility at CP-5 have been further examined by observing the change in resolution detected as the resolution test piece and the detector were separated by distances up to 4 in. The neutron beam is well enough defined so that the degradation of resolution, as determined by examining radiographs and resolving the images of closely spaced holes in a cadmium test object, is not very great for object-film distances up to 4 in. The change

in resolution observed was from 0.0012 in. to 0.004 in. over this distance. The introduction of a scattering medium such as graphite between the object and film, however, does result in appreciable loss in resolution. Resolution has degraded to about 0.030 in. by the time 4 in., or four scattering mean free path thicknesses, have been inserted between object and film.

c. Eddy Current Techniques

(1) Tests on Sodium-Bonded Pu-Fs Pins- 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 mask aperture assembly used with this system does not directly depend upon impedance changes for its operation, as do the probes and coils of more conventional electromagnetic test systems. Consequently, the reflection system can be used to test components over a wide range of temperatures without significant effect on the operation of the system. As an example, sodium bonds are now being checked with the sodium in molten condition. In this way only actual nonbonded areas are detected without confusion resulting from the presence of possible shrinkage voids in the solid sodium.

Recent tests on Ni-1 w/o Zr jacketed pins heated to a temperature above the melting point of sodium have shown that small voids will often disappear after the sodium melts, but tipping the pin from the vertical position, or even a sudden motion of the pin in the vertical position can cause a very large nonbonded area to appear which is permanent unless the whole procedure for rebonding the pin is undertaken. Further tests are being undertaken to study both the disappearance of shrinkage voids and the amount of motion necessary to introduce serious nonbonds in the bond layer.

(2) Development of the Pulsed-Field Reflection System- Development work has continued on the pulsed-field reflection system. This electromagnetic test method replaces the sinusoidal currents and test coils or probes of our 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.

This test method is now being adapted to the inspection of thin-walled tubing. Double frequency sinusoidal test equipment is currently being used for this purpose, but the pulsed field equipment seems to possess the potential for better sensitivity, reliability and simplicity of operation.

The double frequency system had demonstrated its usefulness in a wide variety of problems, so the changeover will be completed only when the new system has demonstrated its superiority over the old in all situations of interest.

C. Reactor Materials Development

1. Radiation Damage in Metals

a. Irradiation of Steel in EBR-I - A series of Izod specimens of SA-212B steel were encapsulated, sent to EBR-I, and irradiated. The results should generate a curve of ductile-brittle transition temperature shift as a function of exposure in EBR-I. The first such irradiations were for 95.4 Mw-hr and the specimens developed a temperature shift of about 140°C (ANL-6573, Progress Report, May, 1962). The new samples were irradiated for periods varying from 2 to about 60 Mw-hr. The irradiated samples were then shipped to Argonne, Illinois for impact testing.

A nickel monitor was included with each specimen which will be used to determine relative exposure. In addition, each specimen will be analyzed for Mn^{54} activity, the result of another high neutron threshold energy reaction.

b. Dosimetry - Two more dosimeter capsules containing a variety of bare and cadmium-covered foils were exposed in the two previously used irradiation locations in CP-5. The foils appear to be in good condition and an analysis of most of them is complete.

Similar capsules in steel cans for high temperature service in EBWR were loaded into special tubes located just inside the reactor thermal shield and only a few inches from the pressure vessel wall. Steel samples were also inserted at these locations. The reactor schedule calls for high power operation in the near future.

D. Reactor Components Development

1. Development of Manipulators for Handling Radioactive Materials

a. Control System for Master-Slave Electric Manipulators - Work is progressing on a control system for use with future master-slave electric manipulators with 50 lb load capacity. This system will probably be used first with the Mark 4 Manipulator. It will use separate power amplifiers for master and slave arms. This gives a flexible system which will permit preliminary testing of features which are being considered for future manipulators, such as motion scaling, motion interchanging and electrical counterbalancing.

The electronics for each motion will consist of the power amplifiers, mounted on separate chassis, plus a third chassis with circuits common to both master and slave such as the fail-safe circuit, the demodulator, the modulator, etc. Both the power amplifier and the common

chassis have been built in prototype models. These prototype models were tested to explore noise pickup problems resulting from the component arrangement and thermal problems resulting from the compact arrangement.

The power amplifier is a 200 watt all solid state device. Each power amplifier is forced convection cooled by its own fan. This cooling is effective enough to allow operation in ambient air of 135°F without degradation of performance. The power amplifier and the common circuit chassis both use plug-in circuits wherever possible for easy maintenance.

Further work on the electronics for the Mark 4 will be the design of the power supplies and the selection of the best position indicating device.

b. A Preliminary Study of a Radio Link for Master-Slave Electric Manipulators - Preliminary studies of the requirements for a radio link for master-slave electric manipulators indicate that considerable improvement over commonly used systems is required. The positional signal must be transmitted with a resolution of about 1 part in 5000.

One system being investigated is a multiplexing and decoding method using sonic delay lines. A magnetostriction delay line is commercially available which has been tested satisfactorily for this operation. In order to use this system, the data must be put into digital or impulse form. The conversion equipment for conventional digital codes is quite complex.

A type of modulation which results in a pulse output called delta modulation is being investigated. Essentially, a pulse rate is generated proportional to the amplitude of an analog signal. For reconversion these pulses are integrated.

Present efforts are toward using a very constant area pulse generator consisting of a saturable magnetic core ahead of the pulse integrator. This core, when switched between negative and positive saturation, switches a constant amount of flux which represents a very constant volt-second area pulse for integration. Current plans are to operate a manipulator motion using the system as a link.

E. Heat Engineering and Fluid Flow

1. Fast Reactor Test Facility Heat Transfer

A survey of available literature for information concerning thermal conductivity of unconsolidated porous media has nearly been completed. The survey is being conducted in order to obtain or formulate a generalized correlation capable of predicting conductivity for a range of temperatures and pressures in powdered ceramic fuels. Such materials may be suitable as fuels for experimental studies in FARET.

The NACA conductivity data originally used directly to assess powdered fuel performance was not as extensive as required and was observed to have considerable scatter around the correlation presented by the investigators. These data were re-evaluated using more recent values for thermal conductivity of the various powders and gaseous media but the agreement was not substantially improved. It was concluded that the relation did not describe the various heat transport mechanisms in sufficient detail to allow reliable extrapolation. Several other data sources and attempted correlations were located in the literature and reviewed. Of these, the work of Kunii and Smith³ appears to account most logically for all the heat transfer aspects involved and provides good agreement with a large amount of data. The thermal conductivity versus temperature of uranium oxide and carbide packings with different particle sizes and gas media is therefore being computed on this basis. Some information on the effect of reduced gas pressure on conductivity has been found but not yet applied to the present problem.

Some approximate calculations were made to point up problems involved in the design of the FARET heat exchange system. Off-design operating situations were computed for various assumed design conditions. Results indicate that if the design conditions are 50 Mw and high (650°C) primary sodium temperature, then about 60% more air-cooled heat transfer surface or 80% more air flow capacity must be added to maintain the design power level if the primary temperature level is reduced to 480°C. Conversely, if the system is initially sized for the lower temperature case, the surface area will be more than adequate for the high temperatures, but excessive exhaust air temperatures must be avoided.

2. Measurement of Thermal Conductivity

An experiment is being designed to measure the thermal conductivities of materials at high temperatures to determine their possible application as nuclear reactor fuels. The apparatus is based on theoretical foundations developed and verified experimentally by K. H. Bode.⁴ The method is applicable to high temperature and seems to be very promising. No absolute temperature measurements are required, and it is not necessary to evaluate the heat losses from the specimen. This is very advantageous at high temperatures because of the difficulty in evaluating the heat losses accurately. The thermal conductivity, at a given temperature, is expressed only in terms of the variation of the electrical resistance and the current of the specimen.

³Daizo Kunii and J. M. Smith, "Heat Transfer Characteristics of Porous Rocks," AICH.E. Journal 6, No. 1, 71-78 (1960).

⁴K. H. Bode, "Measurements of the Thermal Conductivity of Metals at High Temperatures," Progress in International Research on Thermodynamic and Transport Properties, ASME, New York, 1962, p. 481.

F. Chemical Separations

1. Fluidization and Volatility Separations Processes

a. Fluoride Separations - Additional experimental work (see Progress Report, April 1962, ANL-6565, page 44) has been done to evaluate the filters employed in the ventilation systems of facilities used for handling plutonium hexafluoride. The filters are being evaluated for efficiency in removing plutonium particulate matter resulting from the vapor phase hydrolysis of plutonium hexafluoride. With air containing about ten milligrams of water per liter of air entering the experimental apparatus and with three to 22 milligrams of PuF_6 released into the reaction chamber, the overall percent penetration of two AEC filters in series varied from 10^{-5} to 10^{-6} percent.

Data on the decomposition of plutonium hexafluoride by gamma radiation have been reported previously (see Progress Report, March 1962, ANL-6544, page 44). Preliminary results have been obtained on the effect of the presence of small amounts of several gases on the decomposition. The data appear to indicate that both oxygen and krypton reduce the extent of the decomposition of plutonium hexafluoride, whereas helium is without effect. Additional experiments with these and other gases, such as neon and fluorine, are being planned.

Initial results from a study of the alpha-induced decomposition of gaseous plutonium hexafluoride at room temperature have been obtained. Decomposition of plutonium hexafluoride at 100 mm Hg pressure in a 128-cc volume was found to proceed at a rate of 0.38 percent per day on standing at room temperature for 5.62 days, and at a rate of 0.36 percent per day for a 12.6-day duration. Plutonium hexafluoride at 50 mm Hg pressure in the same volume decomposed at a rate of 0.30 percent per day over a 12.6-day period. Studies of longer durations are being made.

The separation of mixtures of uranium hexafluoride and plutonium hexafluoride by selective thermal decomposition is being investigated. Gaseous mixtures of plutonium hexafluoride (10 mm Hg), uranium hexafluoride (90 mm Hg), and helium (660 mm Hg) were circulated through heated, nickel vessels which were packed with nickel wool. The gas mixtures were sampled before and after circulation for an analysis of uranium and plutonium. The uranium-to-plutonium ratio in a gas mixture increased from 12.8 to 1.94×10^4 during circulation through a nickel vessel at 200 C, indicating a removal of 99.9 percent of the initial plutonium. The uranium-to-plutonium ratio in another gas mixture increased from 13.2 to 3.63×10^3 during circulation through a nickel vessel at 300 C, indicating a removal of 99.6 percent of the initial plutonium.

b. Direct Fluorination of Uranium Dioxide Fuel - Laboratory work on the Direct Fluorination Process has been directed toward the development of fluorination techniques for the effective removal of plutonium from a solid solution of uranium dioxide and plutonium dioxide mixed with granulated Alundum. Experiments were made using an initial mild fluorination step which employed 10 volume percent fluorine, 25 volume percent oxygen, and 65 volume percent nitrogen at 450°C. This initial mild fluorination step gave improved removal of plutonium presumably as a result of reduced sintering of the solids due to local overheating. This was followed by a second fluorination at 550°C and in some experiments, at 650°C using 75 volume percent fluorine and 25 volume percent oxygen. The lowest plutonium concentration in the Alundum residue was obtained for experiments in which the second fluorination was performed at a temperature of 650°C for five hours. This procedure resulted in an average plutonium concentration in the residue of 0.016 weight percent (best previous result was 0.027 weight percent) corresponding to a removal of 98.7 percent of the plutonium.

Experiments have been carried out to determine whether any interactions occur between plutonium compounds and fission product element compounds during fluorination. Information on such interactions would be of value in indicating the mechanism for the retention of plutonium on inert solids in the presence of fission products. The initial experimental work has been concerned with the fluorination of mixtures of plutonium and fission product element oxides and fluorides (atom ratio of fission products to plutonium 1:1 or 2:1). The fluorinations were carried out for 10 hours at 450°C. Mixtures of plutonium dioxide and plutonium tetrafluoride with the oxides and fluorides of barium, zirconium, lanthanum, neodymium, praseodymium, cerium, and cesium have been studied. Analyses of the fluorination products were made by X-ray diffraction and, in some cases, by conventional chemical means. Possible compound formation has been indicated for the plutonium-barium and the plutonium-cesium systems. The experimental evidence suggests that the double compounds most probably formed are BaPuF_6 and Cs_2PuF_6 . In the plutonium-cerium and plutonium-zirconium systems in which all the components form tetrafluorides, the formation of solid solutions is possible. In the other binary systems (plutonium-lanthanum, plutonium-praseodymium and plutonium-neodymium) no interactions were observed. Only the individual fluorides were identified by X-ray diffraction.

In a recently completed series of pilot plant-scale "deep-bed" uranium dioxide pellet fluorinations, batch fluorinations of 18-in. deep pellet charges were carried out at 500°C in a 3-in. diameter fluid-bed reactor with partial off-gas recycle, a technique developed to control oxygen concentration (from 1 to 4 percent) in the fluidizing gas while obtaining the benefits of recycle through which higher fluorine efficiencies can be

gained. The results indicated satisfactory fluorine efficiency (~ 77 percent), but caking tendencies were not eliminated for uranium hexafluoride production rates higher than 20 lb uranium hexafluoride/(hr)(sq ft reactor cross-sectional area). This production rate is considerably less favorable than with 6-inch pellet beds, which achieved 50 lb uranium hexafluoride/(hr)(sq ft) under satisfactory conditions.

Process optimization studies are continuing. A new series of runs is under way which is designed to obtain specific effects of lower pellet bed heights (6 and 12 in.) and alumina fluid-bed heights (12 and 24 in.) under a range of process conditions (studying principally the effect of oxygen concentration). Pellet bed heights will be maintained at a nearly constant level by intermittent addition of pellets; in this way, steady state performance can be used to evaluate the effect of bed height.

c. Plutonium Pilot Plant Facility - Design, procurement and installation of the engineering-scale facility for fluorination process studies of plutonium-uranium oxide fuels is continuing. The order for the major process valves has been placed; because of expected long delivery time of these valves, an extension of about two months may be involved in the overall completion time. Further scheduling analysis will be made in an effort to shorten the overall project time extension.

d. Separation of Uranium from Zirconium Alloy Fuels

(1) Studies of the Chlorination and Fluorination Steps with Down-Flow Fixed-Bed Filters - Additional work on the two-step chlorination-fluorination fluid-bed volatility scheme for processing enriched uranium-zirconium alloy fuels dealt with the use of phosgene as an aid in the initial reaction (chlorination) of the alloy. A 4:1 mole ratio mixture of hydrogen chloride and phosgene was used for a period of five hours as the initial reactant gas. Phosgene (with nitrogen diluent) was then fed for an additional three hours. The second step in the process scheme, fluorination of the uranium-containing residue, was carried out for seven hours. Both chlorination and fluorination were conducted at 400°C . Preliminary results indicate that the residual uranium content of the inert Alundum bed material was slightly lower in this case than in a previous experiment where the chlorination reactants were used separately, 0.04 weight percent as compared to 0.05 weight percent after a similar fluorination time period of seven hours. The lower value is similar to that found with the use of greater excesses of hydrogen chloride (see Progress Report, May 1962, ANL-6573).

A spontaneous ignition of halogen gases in the cold traps used for uranium hexafluoride collection occurred in the run in which mixed halogens (hydrogen chloride and phosgene) were used as the initial reactant

gas. The incident occurred upon venting of the cold traps (maintained at -80°C during the run) after the fluorination step had been completed. Causes for the ignition are being sought.

2. Chemical Metallurgical Process Studies

a. Chemistry of Liquid Metals - The solubility of strontium in liquid zinc over the temperature range 506° to 727°C may be represented by the equation

$$\log (\text{atom percent strontium}) = 9.619 - 9940 T^{-1}$$

An experiment to determine the solubility of technetium in liquid zinc was completed. Analytical data are not yet available. However, evidence was found for at least two different technetium-zinc intermetallic phases.

Emf data were used to compute the activity coefficient of plutonium dissolved in liquid zinc. Values of the activity coefficient were found to range from 2.7×10^{-8} at 431°C to 2×10^{-4} at 721°C . These values are preliminary and will be refined when additional emf data are obtained.

A comparison of the data obtained by means of a recording effusion balance for the lanthanum-zinc system with the corresponding X-ray results showed good correlation for the compounds LaZn_{13} , LaZn_{11} , $\text{La}_2\text{Zn}_{17}$, LaZn_2 , and LaZn . However, some discrepancies within the solid solution ranges $\text{LaZn}_{7.4-8.0}$ and $\text{LaZn}_{4.0-6.6}$ were noted. Within these regions the diffraction patterns indicated the presence of two phases. A possible cause for this disparity may be an excessive depletion of zinc in the surface of the sample during the course of the effusion experiment. New effusion cells with smaller orifices have been fabricated in an effort to reduce this surface depletion.

b. Metal Distillation Studies - Some measurements have been made of the extent of liquid cadmium entrainment in cadmium vapor during distillation in the large cadmium distillation unit. A nonvolatile solute, tin, was dissolved in the cadmium; measurement of the tin concentration in the distillate provided a measure of entrainment. Entrainments of one to three percent occurred at a distillation rate of from 37 to 43 kg per hour. Further measurements will be made at higher distillation rates.

c. Calorimetry - Techniques for the combustion of zirconium diboride in fluorine are being developed in preparation for the determination of the heat of formation of zirconium diboride.

Preliminary combustions of ruthenium in fluorine were carried out in the program to determine the heat of formation of ruthenium pentafluoride.

Satisfactory ignition techniques have been developed, and work is now directed toward the development of an analytical technique for combustion products other than ruthenium pentafluoride.

In view of the large corrections due to oxygen contamination which were applied to the heat of formation values of zirconium hydride and zirconium deuteride in a previous series of combustions (see Progress Report, December 1961, ANL-6485, page 57), a second series of combustions was made using new samples. From this series, the following values for the heats of formation were obtained: zirconium hydride, $\Delta H_f^\circ(25\text{ C}) = -38.90 \pm 0.31$ kcal/mole; zirconium deuteride, $\Delta H_f^\circ(25\text{ C}) = -40.25 \pm 0.27$ kcal/mole.

3. General Chemistry and Chemical Engineering

a. Simultaneous Reaction Studies - Preparation of High Density Uranium Dioxide Particles - Fluid-bed studies on the preparation of high-density spheroidal uranium dioxide directly from uranium hexafluoride by simultaneous reaction with steam and hydrogen are continuing. In an attempt to confirm previous results, it was found that particles of varying densities were obtained for similar steam excesses when the extent of immersion of the inlet hexafluoride nozzle, which enters the column at the apex of the inverted cone bottom of the 3-inch diameter reactor, was altered. Two positions were tried, a relatively deep immersion ($\sim 3\frac{3}{4}$ inches) into the bed and a retracted position, ($\sim \frac{3}{4}$ -inch immersion into the bed). Using steam excesses nearly 45 percent above the stoichiometric requirement, lower densities were achieved with the retracted position, 8.0 and 8.5 g/cc as compared to 9.3 and 9.5 g/cc with greater immersion of the nozzle.

IV. PLUTONIUM RECYCLE

The initiation of a study of plutonium recycle reactor systems was noted in ANL-6565 (Progress Report, April, 1962).

It has been decided to use MUFT-SOFOCATE to determine four group cross sections for possible EBWR plutonium cores. A comparison of this calculational method with the results of homogeneous plutonium critical experiments was made previously. For heterogeneous lattices, the effect of self shielding of the Pu^{240} resonance will have to be taken into account. Using cross sections from ANL-6514 for a possible uniformly enriched (2.2% Pu^{239} , 0.2% U^{235} , 0.15% Pu^{240}) core design, it was noted that the reactivity decreased by 3% as the effective cross section of Pu^{240} was increased from 1000 b to 2000 b. With a $\text{Pu}^{240}:\text{Pu}^{239}$ ratio of 0.07 the plutonium enrichment for a given reactivity is very close to the U^{235} enrichment necessary for the same reactivity.

V. ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

A. Argonne Advanced Research Reactor (AARR)

1. Facility Design

United Nuclear Corporation has completed its Engineering Feasibility and Cost Study on AARR, (see Progress Report, April 1962, ANL-6565) and draft copies of the final report on the study have been received. A detailed evaluation of the draft report is in progress. The final issue is expected in mid-July.

2. Research and Development

a. Critical Experiment - Fuel specifications for the AARR critical assembly have been drawn up. A new core layout of hexagonal shape has been drawn and detailed. Prototype models of a simplified control rod drive are under construction for testing and demonstration. Alternate designs for the fuel assembly components are being tested, as a means to reduce the assembly time. A dummy element is being fabricated for water displacement, air entrapment, and loop flow tests.

A simple method of determining epithermal neutron spectral information is being investigated for possible use in the AARR critical assembly program. The epithermal flux from one side of the ATSR will be used in the initial testing. A beam will be formed by collimating the leakage flux; the beam will be scattered by a thin target of suitable nuclear mass and the scattering products will be detected by a detector with a strong resonance, positioned in an energy region of interest. From the resonance energy of the detector material and from the angle between the direction of the incident neutron beam and the radius vector connecting the scattering target and the detector foil, the original energy of the neutrons can be inferred. From the activation intensities of detector foils positioned at various angles with respect to the incoming beam, the relative intensities of different energy components of the leakage beam may be determined.

The scattering assembly and foil support have been designed and fabricated. Scattering materials have been chosen and the isotopes have been procured.

b. Core Physics Calculations - Reactivity changes associated with a number of perturbations were determined. In particular, the over-all temperature coefficient was found to be $-2.34 \times 10^{-5} \delta k/^{\circ}\text{C}$ and the over-all void coefficient, $-0.16 \times 10^{-2} \delta k/\% \text{ void}$. The void coefficient in the core only was found to be $-0.22 \times 10^{-2} \delta k/\% \text{ void}$.

The effect of voiding the internal thermal column (ITC) was positive up to voids of 40% where the gain over no void was 1.25%. The difference between 16% void and 40% void was one dollar; hence, it may be necessary to load the ITC with 16% inert material to avoid possible hazardous conditions. Since the sample influence was not included in these calculations, they are now being repeated with a slightly oversize sample to see if the increased sample size will modify the reactivity gains from voids in the ITC.

The reactivity feedback at 100-Mw operation was found to be proportional to the power for periods longer than 1 sec and powers less than 130 Mw. The coefficient is -4.5×10^{-5} per Mw. Above 130 Mw boiling takes place and the coefficient becomes larger in absolute value. The metal expansion accounts for about 7% of the coefficient and is prompt negative. The delay time before appreciable water heating takes place is about 12 m-sec. No cross-section effects have been included in the feedback term. These cross-section effects are now being computed but they are not expected to alter the situation materially.

B. Subcritical Reactor as a Substitute for Isotope Heat Sources

The limited supply of relatively long half-life isotopes having a reasonably high power density and the low conversion efficiencies attainable with thermoelectric devices at the present time limits the power output of isotope fueled power sources to several tens of watts. It has been proposed that a heat source consisting of a subcritical multiplying medium surrounding a strong neutron source might be attractive as an alternate to isotope heaters (Progress Report, May, 1962, ANL-6573). The neutron source would be 100 mg of Cf^{252} giving about 3×10^{11} n/sec. The subcritical multiplier would probably be a homogeneous dispersion of U^{235} in Li^7H or ZrH moderator and would have a multiplication of 500. This would give a total power for the assembly of 3750 watts. At a conversion efficiency of 5%, about 188 watts of electric power would be available.

1. Physics Calculations

A series of critical radius problems for reactors with various moderators and fueled with U^{235} were run using the RE-122 Code. Cross-section Set 201 was used for the calculations. The results, given in Table VII, gave somewhat larger critical radii than did the calculations reported in the literature,¹ and it is believed that DSN transport theory

¹ Benjamin Pinkel and George B. W. Young, "A Discussion of the Correlation of Reactors," Nuclear Sci. and Eng. 2, No. 4 (December, 1961).

calculations would give better values than the RE-122 diffusion theory calculations actually used. Nevertheless, the calculations do give an indication of the relative sizes for H_2O , ZrH , and Li^7H -moderated systems. Further work (probably DSN calculations) is necessary to establish the correct absolute values.

Table VII. Results of Critical Radius Calculations with RE-122
(Geometry: bare unreflected spheres)

Prob. No.	Fuel	Moderator	Mol Ratio, R <u>mols mod</u> <u>mols fuel</u>	Critical Radius (cm)	U^{235} Loading (kg)	Total Weight, Fuel + Moderator (kg)
40010	U^{235}	ZrH	50	21.18	11.87	244.6
40011	U^{235}	ZrH	100	22.13	6.81	282.7
40012	U^{235}	ZrH	200	24.44	4.59	365.8
40013	U^{235}	Li^7H	10	19.33	57.20	76.50
40014	U^{235}	Li^7H	50	20.12	14.04	37.72
40015	U^{235}	Li^7H	100	20.73	7.74	33.84
40016	U^{235}	Li^7H	200	22.29	4.86	37.61
40017	U^{235}	Be	10	17.53	86.75	119.9
40027	U^{235}	Be	50	23.81	51.92	151.3
40028	U^{235}	Be	100	26.24	35.64	172.1
40029	U^{235}	Be	200	27.85	27.85	184.8
40030	U^{235}	H_2O	10	17.28	26.40	46.61
40031	U^{235}	H_2O	50	17.71	5.99	28.92
40032	U^{235}	H_2O	100	18.87	3.65	31.58
40033	U^{235}	None	-	9.36	64.64	64.64
40034	Pu^{239}	None	-	5.79	16.05	16.05

2. Heat Removal Calculations

In order to have a simple space power plant, the subcritical reactor should preferably be cooled by conduction of heat to thermoelectric elements on the reactor surface. This eliminates the need for auxiliary coolant loops. Calculations show that beryllium, ZrH and U^{235} -unmoderated systems are not heat transfer limited in producing 4000 watts of power. A lithium-hydride-moderated system and the Pu^{239} -unmoderated sphere are limited to about 700 watts and 1500 watts respectively, if conduction cooled. Owing to their relatively large weights, however, ZrH -moderated and beryllium-moderated systems must produce about 23,000 watts and 15,000 watts respectively, to give a specific power of 0.10 watt/gm (Sr^{90} as $\text{SrTiO}_3 \rightarrow 0.10$ watt/gm).

C. Direct Conversion Studies

The study of a thermionic energy conversion diode using a film boiling liquid metal was noted in ANL-6573 (Progress Report, May, 1962).

The experimental diode with a liquid metal electron collector is being fitted with a larger liquid metal inlet tube for easier filling. In the meantime, the qualitative behavior of the vapor produced in subcooled film boiling is being investigated using water at atmospheric pressure film which is boiling on submerged hot metal experimental shapes.

VI. NUCLEAR SAFETY

A. Thermal Reactor Safety Studies

1. Metal Oxidation and Ignition Studies

Isothermal oxidation studies with ANL base uranium were summarized in the form of empirical equations as follows:

Temp Range (°C)	Rate Equation
300 to 450	$W^{4/5} = 2.6 \times 10^7 t \exp(-16,800/RT)$
450	$W = 840 t$
above 450	$W^{6/5} = 7.3 \times 10^7 t \exp(-14,300/RT)$

where: W is oxygen consumption in $\mu\text{g}/\text{sq cm}$
 R is gas constant, $1.987 \text{ cal}/(\text{mole})(\text{K})$
 T is temperature, Kelvin
 t is time, min.

The rate equations are now being used to calculate the course of uranium ignitions. A step-wise calculation using a simple heat transfer model which takes into account heat generated by the reaction and heat exchange with the furnace is used. The results have shown that the course of burning curve ignition experiments can be predicted almost exactly for specimens having specific areas of 0.5 and 5 sq cm/g . Experimental ignition temperatures for material having a specific area of 50 sq cm/g , however, are lower than the calculated values by about 50°C .

Studies of the ignition of spherical zirconium powder are continuing. Investigation of the effect of varied sample weight in $\frac{1}{4}$ -inch crucibles has been completed for -325 +400 and -140 +170 mesh powders. Ignition temperatures of -140 +170 mesh powder range from 750° to 500°C for sample weights from 0.01 to 0.3 gram; ignition temperatures of -325 +400 mesh powder range from 630° to 370°C for sample weights from 0.01 to 0.6 gram.

Measurements have been carried out in air of the maximum burning temperatures of uranium foils of varying widths and thicknesses (widths ranged from 0.1 to 0.7 cm, thicknesses from 0.005 to 0.031 cm). Observed temperatures decreased with increasing width and thickness. The results are in general agreement with theoretical values based on a diffusion model of uranium burning in air which neglects the thickness of the burning surface, but deviate somewhat for the thicker foils studied. Typical maximum burning temperatures (corrected to an emissivity of 0.7) range from 1810°K for 0.005-cm thick by 0.1-cm wide foil to 1300°K for a 0.031-cm thick by 0.7-cm wide foil. Corresponding theoretical temperatures were 1900°K and 1510°K .

2. Metal-water Reaction Studies

Seven in-pile metal-water experiments have been performed in TREAT. SL-1 type fuel material and zirconium-uranium alloy plates were subjected to transients. Two runs with Zircaloy-2 clad 93.5 weight percent uranium-5 weight percent zirconium-1.5 weight percent niobium alloy fuel pins were carried out in modified assemblies which allow heating of the autoclave prior to the reactor transient. The autoclave was at 285°C and at 1000 psi water pressure. The runs in the modified assemblies were made to obtain a comparison with previous runs at initially low pressures and temperatures in the autoclave (30°C and 20 psia). No data are yet available for these runs.

B. Fast Reactor Safety Studies

1. Experimental Meltdown Program

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

a. Clustered EBR-II Elements in the Absence of Sodium - The best two capsules (Nos. 5 and 6) of Series XXX (see Progress Reports for April, 1962, ANL-6565 and May, 1962, ANL-6573) were opened and examined. Each capsule of the series contained seven EBR-II type elements arranged on an EBR-II fuel pitch inside a stainless steel hexagonal tube similar to the can surrounding a complete 91-element subassembly. Tantalum neutron absorbers were included inside the capsule to shape the axial power profile in the sample into one typical of a fast reactor core. Both capsules, Nos. 5 and 6, contained a central 6% enriched fuel element surrounded by a ring of six 4.1% enriched fuel elements. Maximum recorded temperatures for the elements in the two clusters were 930°C and 1300°C, respectively. In capsule No. 5, the fuel elements were warped, but otherwise undamaged. In capsule No. 6, the elements were found to have been extensively damaged. Frozen fuel alloy was concentrated in the central, hottest portion of the axial length in the region of cladding failure. Approximately 65% to 70% of the fuel was found remaining inside the cladding fragments. No attack by fuel alloy was found on the hexagonal can. No dissolution of the hexagonal can had been observed in any of the previous four tests in Series XXX.

b. Meltdown Experiments on Pre-Irradiated EBR-II Elements - A study of the behavior of pre-irradiated EBR-II elements under transient nuclear heating in TREAT (Series XXXI) was noted in ANL-6573 (Progress Report, May, 1962). The first three samples of pre-irradiated EBR-II elements were prepared and remotely encapsulated in dry opaque containers.

The special shipping coffin, containing the three test elements and a fourth unirradiated encapsulated element for equipment and handling checkout, was received at TREAT.

c. Photographic Meltdown Experiments on Uranium Oxide Fuel Samples - Transient experiments are being run on inert gas-bonded, EBR-II size uranium oxide fuel samples as part of a program to study failure characteristics of ceramic fast reactor fuel. Series XXXII, which consists of two niobium-clad elements and two tantalum-clad elements, was prepared for shipment to TREAT. Each specimen was instrumented with fast-response cladding thermocouples of tungsten-5 wt-% rhenium versus tungsten-26 wt-% rhenium, and contained in a standard dry transparent meltdown capsule.

2. Equipment Development

a. Small (Package) Sodium Loop - A package sodium loop, sized to fit inside the 10 cm square by 244 cm long space of a TREAT fuel element, is being developed for performance of meltdown experiments on single elements or small clusters under flowing sodium.

Because of difficulties encountered in previous attempts to "fire" thermite-type chemical heat source pins under flowing sodium in the loop, as a test of loop containment, one of the molybdenic oxide-uranium pins was run in a fixture outside the loop and observed visually. A high temperature reaction was observed, and an additional pin was tested inside the loop. For the loop test, the sodium flow was held at approximately 3 cm/sec (9.75 ft/sec) at 172°C. A maximum transient sodium temperature of 196°C was registered during the firing, and the maximum heat source surface temperature recorded was 376°C. No significant sodium pressure rise was observed.

b. Large Sodium Loop - A large sodium loop, designed to be a semi-permanent installation at TREAT, is being constructed for exposure of clusters of sample elements in flowing sodium, under EBR-II full design flow conditions.

Special loop design work continued on the vapor traps for the gas transfer system. Provision was made on the loop to permit the testing of a fuel element failure detector which is designed to detect delayed neutrons emanating from fission products released into the sodium stream by a failed element.

3. Theoretical Studies

a. In-Pile Measurement of Uranium Oxide Thermal Diffusivity - A study was conducted of a proposed experiment for measuring the high temperature thermal diffusivity of uranium oxide reactor fuel. In the proposed experiment two rods of different uranium enrichment are pressed together

and sintered. When the combined rod is subjected to a short TREAT transient, each half of the rod is heated to a different temperature. By means of internal sample thermocouples imbedded in the rod at distances of $\pm L$ cm from the interface, the temperature difference δT between the two measurement points may be measured as a function of time. At a given time, t , the difference is given by

$$\delta T(t) = \delta T(0) \operatorname{erf} (L/2 \sqrt{Dt})$$

where D = thermal diffusivity.

The method was analyzed for relative influence of errors arising from uncertainties in measurement of T , L , and t , from the use of finite cylinders which are subject to edge effects and heat loss, and from the use of actual TREAT power bursts which are not infinitesimally short. Limitations were placed on sample size because of neutron flux depression in the sample, and on difference in enrichment between the two rod halves for the first trial rods. Optimum times for measurement of δT , and the optimum range of values for L were also obtained.

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