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REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

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I. WATER COOLED REACTORS (040101)

A. General Research and Development

1. Heat Transfer and Fluid Flow Studies

Studies of the behavior of two-phase mixtures characteristic of boiling water reactor systems are continuing in an effort to obtain a quantitative understanding of mechanisms involved. These studies are carried out for the most part in four different laboratory loops: (a) a large scale, 135 atm (2000 psi) loop; (b) a small scale, 135 atm loop; (c) a natural circulation boiling loop (Armadilla) operating at 40 atm (600 psi); and (d) an air-water loop operating at atmospheric pressure.

a. Steam-Water Separation Studies - Studies are being made to investigate the problem of steam carryover and entrainment in the downcomer of a natural circulation boiling reactor. In order to predict flow rates and reactor power for a given value of reactor void fraction, it is necessary to know the natural circulation driving head, which depends on the density of the water in the downcomer. This problem is attacked by determining bubble size distribution and comparing the curve of bubble velocity versus bubble size in downflow with analytical predictions.

A temperature effect on the size of bubbles carried under was investigated on the air-water loop. Using a Polaroid Land Camera and a 1/1500-sec strobe light, photographs were taken in both the riser and downcomer sections. The data from these tests indicate a possible 5% increase in bubble size as temperature is increased from 20°C to 40°C.

A separate investigation, conducted at a constant temperature of 30°C, indicated that when the water level above the riser was varied there was little effect on bubble size.

b. Hydrodynamic Instability - Flow oscillations in a boiling reactor may cause power oscillations through the feedback effects of flow on voids, voids on power, and power on flow. Measurements are therefore being planned of the power-to-void transfer function for an electrically heated test section in an out-of-pile test loop.

A 1.2 m long rectangular test section is to be inserted in the small scale 135 atm (2000 psi) loop. The effect of sinusoidal power variations on steam bubble density inside the test tube at different heights will be measured as a function of oscillation frequency.

The test element heating power is to be varied electronically in the range 0.01-10 cps by building up a sine wave using 10 line segments.

The test section itself and a smaller piece to be used for calibration purposes are being finished by Central Shops.

With the exception of a shield container, all components for the void detector system, including the detector carriage and traversing arrangement, are now on hand. Final assembly work will not be started until the test section has been put in place in the loop.

Work has continued on the theoretical prediction of the power-to-void transfer function. The aim is to develop a detailed theory describing the physical processes in the specific test element to be used, rather than general formulas for use in reactor design.

The test element corresponds to a rectangular plate fuel element with no cladding. It has been shown that although the test element wall is thin (0.89 mm) and the heat transfer good, the delay due to the heat capacity of the wall cannot be neglected. A one time constant delay gives an excellent representation for the upper part of the channel where boiling takes place. In the lower part however, fuel plate delays cannot be separated from water-flow transport delays, and the combined expression becomes rather involved. Work is being carried on to try to simplify the result without the introduction of significant errors.

c. Computer Program for Hydrodynamic Calculation of Reactor Cores - Calculation of flow by natural circulation in a boiling channel is a lengthy iterative computation. A computer program (CHOP) was written three years ago to permit computation (on the IBM-650 computer) of hydrodynamic characteristics in a single channel, with uniform heat input, and with the coolant circulating by virtue of natural circulation. This program was revised last year to permit it to be used for certain multichannel reactor core calculations and to put it on the IBM-704 computer to take advantage of an excellent water properties program that was available. Even with this revised program (RECHOP RE-160) a considerable number of iterative hand calculations were required to complete the hydraulic analysis of a reactor core.

Several core configurations have been proposed recently which vary geometrically in the axial direction. The available computer program cannot calculate this type of configuration. In addition, the magnitude of the effect of the axial power distribution on the axial void distribution has often been requested. To analyze these more comprehensive problems a new computer program is being written. This new program will be quite general in that it will be able to compute natural circulation conditions in up to ten different types of fuel elements, each of which may have up to three geometrical changes in the axial direction. Any radial or axial power distribution may be specified and the axial power distribution may be made a function of radial position.

Three types of risers may be considered: (a) individual risers which are associated with each individual fuel element; (b) compound risers which serve a number of fuel elements; and (c) common risers which receive the flow from the entire core. Each type of riser may have as many as three geometry changes.

The program will also permit calculation of the effects of steam carryunder into the downcomer and the effects of up to five flow restrictions, such as grid plates and spacers, in both the boiling and nonboiling regions of the core. Input and output of the program will be in metric units.

The progress on this project to date is that all equations and most of the correlations have been derived. They are being assembled in proper sequence and will be presented shortly to the Applied Mathematics Division

for programming.

2. Boiling Slurry Reactor Experiment (SLURREX)

SLURREX is an aqueous boiling slurry reactor experiment proposed for construction at the National Reactor Testing Station. It is looked upon as a BORAX-type experiment to determine the potential of the concept. The experiment consists of a natural circulation boiling core plus functional sub-systems for fuel handling, gas handling, heat dissipation, and water distribution. The design power level is 5 Mwt and the operating pressure is 10 atm. Pending approval by the AEC, a limited experimental and preliminary design program is under way.

A half-scale, atmospheric pressure, air-water mockup of the primary system is being built to study hydraulics problems. Air and water piping for the mockup of the primary vessel is being constructed and placed.

Two preliminary plant layout schemes for the full-scale SLURREX system have been completed. One consists of two flooded tanks for radioactive components, the other of three tanks. In the latter scheme, the primary vessel has been separated from most of the other fuel-containing components in the hope of simplifying maintenance on those components. The relative merits of the two schemes are being evaluated, especially in light of a detailed maintenance philosophy which is being developed concurrently.

Work on the design of the gas handling system was continued. A gas absorption column was studied as a possible alternate method to charcoal absorption as a means for removing xenon and krypton. Such a column could operate at higher temperature than a charcoal bed and with no concern about water vapor content. The major disadvantages are spent solvent storage and the possibility of solvent carryover into the reactor where it could cause foaming.

Parametric studies were continued on the effect of core L/D ratio, and thorium oxide concentration, on the critical concentration of U-235 required for operation at 185°C and 5 Mw (i.e., average voids in the core equal to 0.32) in a core with a radius of 29.8 cm. Based on the results of these studies, the two parameters were fixed as follows for future calculations: L/D equal to 1.25; uranium (93% U-235) to thorium ratio 10:90. For this L/D and uranium to thorium ratio, the minimum slurry concentrations to obtain criticality for various average void fractions at the operating pressure of 185°C were calculated. The results are as follows:

<u>Average Void Fraction</u>	<u>Solid Concentration, gm/kg of water</u>
0	321
0.12	372
0.18	435
0.25	520
0.32	673

The solid particles are assumed to contain UO_2 and ThO_2 . A void fraction of 0.12 corresponds to 0.5 Mw and is the proposed condition

for startup; i.e., during reactor fill it is proposed to inject gas into the core to maintain an average void fraction of 0.12; and the transition from gas to steam voids will be made at that void fraction, although at a lower pressure. The system will then be raised to operating pressure before increasing the power level.

B. EBWR

1. 100 Mw Modifications

The EBWR is being prepared for a new series of experiments in which an attempt will be made to determine the maximum power obtainable from the system. The design power limitation of the system is being raised to 100 Mwt. Changes have been made in the pressure vessel, and equipment for the new heat removal system has been added both inside and outside the containment shell. The reactor loading (Core 1-A) will use the old fuel elements and 32 additional "spike" fuel assemblies containing enriched uranium. The major modifications are almost complete and startup is possible with the old heat removal system.

a. Construction and Equipment - The primary reboilers were finally delivered to the Laboratory. Installation of these units is in progress and construction work by the contractor has resumed. Work had come to a virtual standstill prior to delivery of the reboilers inasmuch as other work, such as closing the building, etc., depended on having the reboilers in the building. Now that the reboilers have been delivered, electrical work associated with these units has been resumed.

Items which may cause the contractor a delay in completing his work, but which will not necessarily affect the operation of EBWR, are the delivery of the flash condensate return unit and the secondary steam separator condensate unit. These items have been ordered and delivery is still expected in October.

Small miscellaneous jobs within the containment vessel are still in progress and should be completed early next month. This includes minor piping modifications, touch-up insulation and painting.

During the continuing plant checkout, difficulty with the steam flow transducer was encountered. It has been returned twice to the factory for repairs.

The auxiliary boric acid addition system was essentially complete as of September 15. Two flowmeters and portions of the alarm system, scheduled for delivery on October 15, are required to complete the system.

The facility to cut the original EBWR control rods was completed and is being assembled in the fuel storage pit. At least one boron stainless steel and one hafnium control rod will be cut into lengths of about 15 cm for examination. The short lengths will be stored in the fuel storage pit and removed one at a time to Bldg. 301 caves for analysis of changes in control rod materials caused by irradiation. Effects of operation on the structural integrity of the control rods will also be investigated.

b. EBWR 100 Mw Oscillator - The control rod oscillator has been operating successfully throughout its full range on a shutdown EBWR. Observation of the control rod and shroud during the test revealed minimal vibration. The rod edge slides along the channel, as expected, but even at an elevation of 100 cm (the maximum obtainable), a speed of 50 radians/sec, and a ± 19 mm stroke, no slap is detectable. Operation at speeds below 6 radians/sec is exceptionally smooth up to and including the maximum low speed stroke of ± 127 mm.

The oscillator was not run at its maximum high-speed stroke of ± 25.4 mm because a spring adjustment necessitated by the lighter weight of the hafnium rod had not been made prior to the start of the test. Smoothness of operation at the ± 19 mm stroke is expected to be maintained at the maximum stroke also.

c. EBWR Instrumented Assemblies - Single-phase and two-phase pressure drop across the turbine-type flowmeters being used on the EBWR instrumented assemblies has been determined in order to be able to predict accurately the hydrodynamic performance of these fuel elements. It has been found, as a result of these measurements, that the single-phase pressure drop across these flowmeters is slightly lower than the value that was obtained from the manufacturer's data. Tentative two-phase measurements indicate a slightly larger pressure drop than the value that was theoretically predicted. More two-phase data are needed, however, to substantiate this latter conclusion. Because of these differences in pressure drop values, a recalculation of the hydrodynamic performance of the instrumented assemblies will have to be made.

Instrumentation for the series of performance studies of the EBWR fuel element test section is currently under construction and is being tested. Primary components are a temperature scanner and a d-c burnout detector. The temperature scanner will monitor 50 chromel-alumel thermocouples welded to the test section. Display of the temperature profile will be accomplished by high-speed switching of the thermocouples and information display on a high-gain, differential-input oscilloscope.

d. Hazards Report and Operating Manual - An addendum to the original hazards summary report (ANL-5781 Addendum) was submitted to the Commission in July, 1959. After review by the ACRS and HEB approval was granted to proceed with experimental operation leading ultimately to operation at power levels up to 100 Mw. However, since that time a number of modifications have been made which makes it desirable to revise the hazards report addendum. The most important modification is the proposed use of boric acid chemical control (see Progress Report, August, 1960). Accordingly, the hazards summary report addendum has been revised and will be submitted to the Commission in October.

A completely new operating manual for the 100 Mw plant has been written in draft form and is now being reviewed and edited. It is expected to be ready for submission in November.

2. EBWR Core-2 Hydrodynamic Calculations

During the past two years various core concepts and configurations

have been analyzed for the purpose of ultimately arriving at a suitable design for a proposed second core for the EBWR. An analysis of hydrodynamic characteristics during operation at 100 Mw has been completed on a core that is 152.4 cm high and consisting of the following types of fuel elements:

(a) Seventy-nine (79) rod-type elements composed of forty-nine (49) fuel rods (1.002 cm OD) in a 1.252 cm square lattice, and located on the periphery of the core.

(b) Sixty-eight (68) rod-type elements, of which the lower 76.2 cm consists of forty-nine (49) fuel rods (1.002 cm OD) in a 1.252 cm square lattice and the upper 76.2 cm consists of thirty-six (36) fuel rods (1.002 cm OD) in a 1.503 cm square lattice. These are to be located in the central region of the core.

This particular core design resulted from previous analyses of a core composed entirely of the type of fuel elements described under (a) above and a core composed entirely of the type of fuel elements described under (b) above. These analyses revealed that in the first type of core, the coolant velocities in the central, high power region were too low, while in the second type of core, the coolant velocities on the periphery of the core were too high. This core is a compromise of the first two in that the "non-uniform" fuel element, which provided more flow area in the upper (high void fraction) region of the core, was retained in the central portion of the core to give adequate coolant velocities in this high power region, while the "uniform" fuel element was incorporated on the periphery of the core to reduce the velocity of the coolant passing through this region. The geometric factors for the 36-rod and the 49-rod portions are:

	<u>36-Rod Element</u>	<u>49-Rod Element</u>
Equivalent Diameter	1.87 cm	0.99 cm
Flow Area per Element	57.3 cm ²	47.1 cm ²

For the calculations the core was divided into three radial and two axial regions and two simplifying assumptions were made:

(a) An "average channel" from each of the three radial regions adequately represented those regions.

(b) Each half (upper and lower) of a radial region operated with a uniform axial power distribution.

The driving head associated with the top 30.48 cm of the common riser was arbitrarily neglected in an effort to compensate for carryunder of steam into the downcomer.

Table I lists the results of the calculations for the various regions of the core.

Table I. Hydrodynamic Characteristics of a Proposed EBWR Core II

	Power Mw	Average Inlet Velocity m/sec	Average Void Fraction*	Average Boiling Length cm
Central Region (36 fuel elements)				
Lower 1/2 (49-rod zone)	21.3	2.1	0.15	107
Upper 1/2 (36-rod zone)	6.4	---	0.35	---
Upper Reflector	----	---	0.45	---
Intermediate Region (32 fuel elements)				
Lower 1/2 (49-rod zone)	22.7	2.1	0.10	100
Upper 1/2 (36-rod zone)	8.0	---	0.31	---
Upper Reflector	----	---	0.40	---
Peripheral Region (79 fuel elements)				
Lower 1/2 (49-rod zone)	30.2	1.8	0.08	92
Upper 1/2 (49-rod zone)	11.7	---	0.26	---
Upper Reflector	----	---	0.40	---

*The void fractions shown are the "average void in the fuel cell" and not the void fractions in the fuel element. The average void fraction for the core was calculated to be 0.198.

From the results obtained the following conclusions may be drawn:

(a) The velocity distribution in this core design is a considerable improvement over the velocity distribution obtained in the two previous designs.

(b) The higher recirculation velocities and the "opening up" of the upper half of the high power region should alleviate hydrodynamic instability and subsequent burnouts.

(c) Adjustment of the velocity distribution has produced a more desirable radial void distribution.

C. BORAX V

1. Construction, Design and Procurement

The heating and ventilating equipment in the heating and ventilating building has been installed. The piping contractor continues the installation of piping and valves in the steam and water trenches and the equipment pit. The electrical contractor is continuing to install conduit and wiring. The construction was 79.5% complete on September 30 vs. the scheduled completion of 93%.

The reactor vessel is scheduled to be shipped October 4. Delays were experienced due to problems in meeting dimensional tolerances and in passing ferroxy tests. Also hydrostatic testing revealed some leakage in the main head gaskets, which required correction.

Fabrication of the boiling fuel rods, being made by Westinghouse, is being delayed owing to the late delivery of Inconel X springs. Partial deliveries will begin in October. Assembly and inspection will be performed at the Laboratory and should be completed by mid-December.

The contractor originally selected for fabrication of the superheater core has withdrawn and contract negotiations with an alternate fabricator are awaiting Commission approval.

Checking of core support structure drawings is nearly completed and they are expected to be released for manufacture on September 30. Drawings of the control rod drawing hold-down boxes and chimney will be released at the same time.

The control rod thickness and boron content has been established. Increasing the boron concentration from 35 to 50 w/o boron carbide with a minimum thickness of 0.147 inches will provide more boron than the original design called for. The over-all thickness, including cladding, will be 0.312 inches.

Machine problems have been prepared to study the reactivity worths in the reference design core. Additional machine problems are in preparation for determining the temperature distributions in the superheater.

2. Development Studies

The BORAX V superheat fuel plates (stainless steel-uranium oxide dispersion) are to be evaluated by the Laboratory for adherence to specifications. A fuel plate similar to the superheater plates has been machined into a suitable specimen for preliminary evaluation of a clad thickness measuring technique. Encouraging results have been obtained using an absolute coil eddy current bridge device. However, the available specimen represents the least difficult type to work with as it contains the greatest amount of UO_2 that will be present in the BORAX V fuel elements. Some elements will contain roughly one-third as much UO_2 by weight. The UO_2 content is an important factor as this determines the difference in electrical resistivity between the clad and the fuel matrix. With the minimum amount of UO_2 the resistivity difference between the two layers may be so small as to be undetectable by an eddy current technique. For this reason, it is of some importance to obtain a fuel element containing the minimum amount of UO_2 for future work.

Still under development is the method for assembly of fuel plates into fuel assemblies by bonding stainless steel spacers (0.125 in. dia. x 0.125 in. long) to adjacent plates at predetermined locations. Metallographic examination of experimental resistance spot welds has shown that the weld area penetrates too deeply into the plate in most cases, and frequently UO_2 particles are dispersed toward the outer surface of the plate. Microbrazing 50 and GE 81 commercial brazing alloys are being investigated as possible bonding agents. To date, GE 81 has shown the most promise of producing a satisfactory bond, although several variables, e.g., amount of alloy and temperature of braze, must still be optimized to produce the best possible results. A method is being developed whereby

strength tests can be performed on experimental assemblies to evaluate the bond between plates and spacers.

On the air-water flow test stand, 41 test runs were made to study the turbine-type flowmeter. Water flows in the range of 100 to 500 gpm with void ratios from 11 to 55% were attained. Agreement between the Potter flowmeter and the gamma attenuator was good to within 10%. A similar agreement was found with calculated data for voids in the range of 20 to 30%. The method of using instruments on this low pressure loop test has revealed the importance of knowing the pressure at the point where void measurements are made. Readings must be corrected as much as 50% when comparing the gamma attenuator with the Potter meter. The added pressure drop due to the presence of the Potter meter is 25% of the total drop in a 49 rod element at 500 gpm with 21% voids.

Redesign and testing of the control rod drive have been completed.

II. SODIUM COOLED REACTORS (040103)

A. General Research and Development

1. ZPR-III

Critical studies of the EBR-II reactor were completed this month. The last experiments were concerned with the validity of substituting aluminum for sodium coolant in fast reactor cores. The worth of sodium as a function of position was measured for a second time. Sodium was substituted for void in local regions of the core and fine blanket with all of the remaining coolant represented by aluminum. An all-sodium-coolant core of identical size had been used to make previous measurements in which the worth of sodium was found to be 25.4 lh/kg at the center. The worth of sodium in the core containing aluminum was measured as 34.5 lh/kg at the center. Values checked at the edge of the core agreed quite well in both cases.

The conclusion is that local sodium substitutions at the center of a core in which the coolant is simulated by aluminum will be in error, but integrated sodium worths over the entire core will not be affected greatly because the volume of the central part of the core is small compared to the total volume of the core.

The next series of studies on ZPR-III will consist of surveys of the critical properties of oxide and carbide uranium compositions which offer the possibility of overcoming low burnup limitations of metallic fuels. Although designs are not yet firm for fuel elements using oxides and carbides, enough of the probable characteristics can be anticipated to plan reasonable, clean, reactor mockups in ZPR-III.

The first ZPR-III loading designed for the survey series (assembly 29) has an oxide core with an estimated size of 370 liters. The core is a cylinder, 28 inches long and about 32 inches in diameter, blanketed with 12 inches of depleted uranium. The core contains 26 v/o UO_2 , 24 v/o Al metal (which represents 48 v/o sodium), and 26.9 v/o stainless steel. The first core loading with each drawer containing one column of fully enriched uranium and two columns of depleted uranium did not go critical in the estimated core size. Fuel enrichment is now being increased to make the reactor critical.

A Russian set of multigroup fast neutron cross sections has been used to calculate the critical masses and U-238/U-235 fission detector responses for a series of representative fast assemblies. Results show that the set predicts these quantities about as well as do sets currently used in ZPR-III analyses. In general the trend for the Russian set is to underpredict critical masses.

Central danger coefficients for U-235, U-238, iron, aluminum, graphite, U-233 and Pu-239 have been calculated by the method of perturbation theory for the ZPR-III assemblies 6F, 9A, 22(11), 24 and 25 having U-238 to U-235 core atomic ratios in the range of one to more than ten. Comparisons with corresponding reported experimental values are in most cases better than 20%.

2. ZPR-VI and ZPR-IX

A building (D-315) is under construction to house two moveable half type critical machines suitable for studies of fast and mixed fast-thermal reactor cores. Almost all the contracts for major components have been placed. Major effort is now directed to preparation of the hazard report, solution of problems as they arise in connection with the manufacture of components, and initial preparations for the experimental program.

a. Construction and Procurement - Construction of the building is estimated to be approximately 40% complete. All columns, beams, and the roof slab for the second floor have been constructed.

In Cell No. 4 work is proceeding on door frames and door installation. All of the conduit which runs through the cell floor has been installed.

For Cell No. 5, all of the reinforcing steel is in place. The forms for the east, south, and west walls have been installed and work has started on the form for the north wall. A fixture is being built to align the anchor bolts which are embedded in the cell floor to hold down the ZPR-IX machine bed.

All columns, beams, and the roof slab for Cell No. 6 have been constructed. The steel pit covers are in place.

Contract negotiations were concluded with the lowest bidder for the control console and nuclear instrumentation for both facilities. The contract calls for the design, fabrication and installation of two sets of control consoles and nuclear instrumentation in accordance with the Laboratory's specifications and interlock circuit requirements.

An inspection of the matrix support castings for ZPR-VI was made. Flaws in the castings were found to be minor, hence the castings are acceptable. The representative of the matrix tube supplier indicated that the first shipment of tubes would be made about the last week in September. The Laboratory will inspect these tubes thoroughly to see that they meet specifications before production is continued.

b. Development - Two types of control rod drives are to be used for these reactors: (a) an expulsion type, whereby a fuel drawer will be ejected from the core; and (b) an insertion type which will insert a boron-filled shim rod into the core. The complete expulsion-type drive is being installed on the test rig and testing will begin at once. General operation of the drive is thus far satisfactory, but a final evaluation must await conclusion of the endurance test. Drawings of the drive are being revised and are expected to be completed by October 10, at which time they will be submitted for bid. Fabrication of the insertion-type drive is continuing and should be completed by October 10. All purchased parts for this drive are on hand.

Tests were started to compare the coating qualities of FC-144, a fluorochemical, and KEL-F, both manufactured by 3M Company for use with the depleted and enriched uranium.

Shielding calculations for the hazards report on the reactor cell concrete wall have been started. For hazards evaluation purposes the

conditions for penetration of the 120-cm reinforced concrete cell wall of the building were analyzed. A superprompt critical excursion was hypothesized and pieces of uranium blanket material were assumed to be accelerated as missiles. It is estimated that a 5 cm cube of uranium would have to be traveling at the rate of 10^5 cm/sec, in order to penetrate the cell wall.

Studies have also continued on the radioactive fission product dispersal problem after a reactor accident for the hazards report.

3. Juggernaut

The Juggernaut is a water-cooled and graphite-moderated reactor of the ARGONAUT type. It is to operate at power levels up to 250 kw and provide a versatile facility for nuclear research and component development.

a. Construction - The building (D-335) was accepted on September 6 and construction of the reactor began September 12.

The reactor is to be constructed in three phases:

(1) Setting of the graphite with its steel encasement, the steel forming for the test cave and universal facility, the vertical and horizontal beam tubes and the thermal column gate housing.

(2) Pouring of the monolithic shield and electrical wiring of motors, interlocks, etc. Phase 2 is to be done by a contractor.

(3) Setting in place the 5 x 5 ft. top shield and rotating plug; general cleanup and installation of minor parts.

Each phase, on the average, will take one month. The pipe fitting and electronic work will be concurrent with Phases 1 and 2.

Phase 1 is in progress. The graphite is placed and covered with steel. Eight of the ten vertical beam tubes are in place. Both thermal column gate housings are in place. Phase 1 is to be completed by October 7.

The pump room wiring diagram has been prepared in collaboration with the Plant Engineering Division and the wiring is expected to start shortly.

b. Instrumentation - The electrical schematics for the control console have been drawn up and the detailed wiring diagram will now be checked. The control console will be moved to the new building by the first week of November and all the changes and additions in the console are to be completed before that.

The control rod indicators are being put together and will be mounted on the center panel as soon as the fine control rod indicator is assembled.

The count rate meter, the pulse amplifier, and the linear trips are expected to be ready by the last week of October. The circuit of the count rate meter had to be changed because of the failure of most of the transistors at 40°C . The transistor types were also changed.

The rod drives for the shim and safety rods have already been assembled in Building 335 and drop time tests are being run. The average drop time is 0.32 sec and is in close agreement with the drop time observed in CP-5.

No thermocouples will be installed in the thermal shields; instead, the flow rate for the shield cooling pumps will be determined only by the indication of the temperature of the outlet water and by operating experience in the future. If necessary, thermocouples can be installed at a later date.

4. Fuel Fabrication Facility - Operations Development

The two-stage helium compressor (Progress Report July, 1960), now equipped with glass reinforced Teflon packings, was inspected after 805 hours operation. The rings appeared to have worn between 0.003 and 0.005 in. compared to approximately 0.300 in. per 400 hours for the carbon rings previously employed. Tests of the helium purifying system under room temperature operation and refrigerated operation have shown the rate of adsorption to be increased by a factor of three and one-half by refrigeration to -35°F . Moisture has been maintained at less than 100 ppm and O_2 at less than 0.1 percent.

5 Fast Reactor Fuels - General

a. Fast Reactor Fuel Jacket Development - The Fast Reactor Program requires a fuel jacket material which is compatible with the reactor coolant, (sodium), and fuel materials, (uranium and/or plutonium alloys), and which will have a combination of high strength and low creep at elevated temperatures. These requirements can best be met with certain refractory metal base alloys. Unalloyed refractory metals, even though they can be more readily fabricated and have high melting points, do not have optimum structural properties for this service. Most development work to date has been limited to available vanadium, tantalum and niobium base alloys.

Efforts are still being made to procure additional refractory alloys for fabrication development work. When available, tubing made of these alloys will be used to make prototype fuel elements for irradiation testing in MTR and for meltdown tests in the TREAT reactor. In the meanwhile, work on existing alloys continues.

The V-5 w/o Ti-20 w/o Nb alloy tube, as reported in August, was further cold worked after removing surface defects by grinding and removing of the cracked weld section. The tube was drawn over a 0.163 in. diameter mandrel to an O.D. of 0.217 in. Checking in the weld area and flaking became evident as drawing progressed.

Several Ta-0.1 w/o W alloy thimbles, measuring 0.430 in. O.D. x 0.029 in. wall thickness x 8 in. length, have been received. One thimble, with the closed end removed, has been cold drawn to 0.260 in. O.D. x 0.220 in. I.D. The surface area is good; however, the tube parted circumferentially twice during the drawing operation. The past history of the material is not known. It is felt that the material must be annealed prior to any further cold working.

A plate sample of Nb-1 w/o Zr alloy has been readily cold rolled from

0.127 in to 0.016 in. and formed into a tube. The tube was then butt welded by inert-gas shielded arc and then cold drawn. The weld failed completely. No defects occurred in the parent metal.

b. Structures and Properties of Uranium-Fissium Alloys - A program is in progress to determine the thermal properties of nonplutonium materials as a preliminary step in developing equipment to determine thermal properties of plutonium-containing materials.

A prototype of an isothermal drop calorimeter has been completed. Tests have been performed to evaluate assumptions made to account for errors in temperature measurement. These assumptions include the effects of variables such as lengths of flat temperature zones in a furnace, thermal conductivity, radiation differences, and ambient temperature fluctuations. This equipment has now been moved into the Fuels Technology Center and calibration of the calorimeter was begun after its reassembly.

Approval has finally been received for the contract with the University of Denver to determine the enthalpy of 3, 5 and 8 w/o Fs-U alloys. Specimens prepared and encapsulated at the Laboratory several months ago (see May, 1960 Progress Report) are being shipped to the Denver Research Institute.

c. Properties of Plutonium-1 w/o Aluminum Alloy - The purpose and scope of this engineering evaluation of certain physical properties of the Pu-1 w/o Al alloy were explained in the July, 1960 Progress Report.

Density and hardness values have been obtained for as-cast specimens of five plutonium-aluminum alloy compositions in the range 0.23 w/o Al to 1.16 w/o Al.

Two dilation runs have been completed for the alloy composition Pu-0.68 w/o Al. Interpretation of the results of the first dilation run was precluded by the incipient melting of the upper end of the specimen at 550°C. Results obtained in the second dilation run indicate a net shrinkage in specimen length of 0.002 in. which is presumably the manifestation of a tendency toward phase equilibration.

Metallographic specimens of the above-mentioned alloy compositions are being inspected. Included are alloy specimens in the as-cast condition, as-cast and equilibrated for 113 hours at 426°C, and cold worked and equilibrated for 113 hours at 426°C.

B EBR-I

1. Mark III Operation and Studies

In May, 1960, after the experimental portion of the stability studies had been completed, the reactor loading was rearranged (see Progress Report, May, 1960) and the following experimental rods were loaded into the reactor: (a) 10 rods containing various isotopes for which measurements of α were to be made; (b) 4 plutonium Mark IV prototype fuel rods; (c) 3 plutonium carbide fuel rods.

The continuous 800 Mwh irradiation run on these samples has now been

completed. The α measurement samples have been removed and prepared for shipment. Two of the Mark IV plutonium prototype rods, and one of the PuC rods, were also removed for shipment. These three rods showed no apparent change in the jackets after irradiation.

At the conclusion of the irradiation run a set of dimensional measurements was performed on the Mark III fuel rods. Total reactor time was 2680 Mwh. Results of these measurements are given below. A more complete comparison of the current measurements with three previous sets is to be presented in a full-length report.

- (a) The maximum jacket diameter increase was 0.034 inches in a sheared rod taken from Hex D at a distance of 2.1 inches from the core centerline. The total reactor time on this rod is 2680 Mwh, with 1270 Mwh as a sheared rod (unrestrained radially in the hex).
- (b) A comparison of diameter measurements made on one sheared and one non-sheared rod, with identical reactor times and identical radial distances from the reactor centerline, showed a diameter difference of 0.026 inches, the sheared rod being the larger of the two. Length measurements made on these rods show a difference of 0.180 inches, the sheared rod being the shortest.
- (c) A comparison of length measurements made on non-sheared rods with approximately the same reactor times, but at different radial distances from the reactor centerline, indicates that the length decrease varies with radial distance. The shortest rods are toward the center and the longest rods toward the outer edge of the core. A typical difference between a center rod and an outer rod is 0.142 inches. The inner rod which shrunk the most was 2.35 inches from the outer rod.
- (d) The rods in Hex D have been sheared for 212 Mwh more than those in Hex C. A comparison of diameter measurements on one similar rod from each hex shows a difference of 0.008 inches, the rod from Hex D being the larger of the two. Also the rod from Hex D was found to be 0.063 inches shorter than the one in Hex C.
- (e) A maximum diameter growth of 0.034 inches for a sheared rod compared to a maximum growth of 0.017 inches for a ribbed rod would indicate that lack of radial restraint within the hex has considerable accelerating effect on rod growth.

Dimensional changes in the unsheared monitor rods from the central subassembly after the 800 Mwh run were not proportionately as large as they were in these rods for previous operation. This may be due to the lower temperature and flux in the central subassembly during this run because of increased coolant passage area provided by the plutonium sample elements.

The stripped rib subassemblies (Hexes C and D) have been restored to the configuration that was in effect at the end of the stability studies last May. The uranium outer blanket has been replaced by the lead outer blanket. Additional power coefficient measurements are being made at varying inlet temperatures to gather additional data.

2. Mark IV Development

Methods are being developed for the fabrication of the plutonium-1 w/o aluminum alloy which constitutes the fuel for the Mark IV core. Fabrication will be done in the Fuel Fabrication Facility (Bldg. 350) and is the first production job to be carried out therein. Pending clarification of the budget situation actual production has been held in abeyance, and work has been limited to development and procurement.

Primary emphasis has been placed on the development of centrifugal casting methods. A number of test casts have been made to produce samples for various measurements. On the basis of these casts, the distributor was redesigned and thoria-coated graphite crucibles are now being used. Although it is felt that the centrifugal casting process is now good enough to be used for actual production, some additional work is being done on the possibility of injection casting. A furnace has been built, Vycor molds obtained, and some test casts were made with non-radioactive metal.

Upsetting dies have been designed and are in manufacture for the production of centering ribs on the fuel to eliminate the need for spacer clips. Methods for drilling deep holes are under investigation for production of fuel slugs with thermocouple holes. Experiments to date on delta plutonium are not promising. Cored casting methods are under consideration.

Tests of the girth welding of the Zircaloy coupler to the fuel tube have indicated that no unusual problems will be encountered. A capacitor discharge welding machine for sealing the NaK fill hole has been designed and is in construction. The seal welder was incorporated in the NaK fill chamber.

An ultrasonic through transmission technique has been developed and set up for inspecting Zircaloy rod to be used as end plugs in the EBR-I Mark IV fuel. The September delivery date for commercially procured tubular components has been extended to December, 1960.

Cost estimates and preliminary criticality safety procedures for the actual production have been prepared.

C. EBR-II

1. Construction

The following tabulation presents the status of the construction contracts as of September 17, 1960:

<u>Building</u>	<u>% Completion</u>	<u>Estimated Completion Date</u>
Power Plant (Package 2)	99.4	October, 1960
Reactor Plant (Package 2)	98.6	October, 1960
Sodium Boiler Plant (Package 3)	40.3	November, 1960*
Fuel Cycle Facility (Package 3)	44.9	May, 1961

*Construction delayed; revised schedule not yet available.

The following buildings have been completed and accepted: Lab and Service Building; Guard House; Fuel Storage and Pump House; Electrical Substation; Cooling Tower; Fire House; Pump House No 1; Sanitary System; and the Warehouse

a. Power Plant - The water demineralizer, which was the last major item of equipment, has been installed in the Power Plant. The final check-out of equipment and systems, including the electrical and the control systems, is proceeding and should be complete by early October

b. Reactor Plant - The construction of the Reactor Plant is essentially complete. The remaining work for the construction contractor consists largely of checkout of the heating and ventilating systems, and electrical and instrumentation systems. The Package 4 installation contractor has begun work on a restricted basis in the Reactor Plant concurrent with the completion of the construction work.

c. Sodium Boiler Plant - The construction of the Sodium Boiler Plant is now approximately 40% complete. Although it is scheduled for completion in November, 1960, construction delays will require an extension of the completion date. A revised construction schedule has not yet been prepared. The delays are due primarily to difficulties the Laboratory has experienced in the construction of the steam generator units and difficulties the piping subcontractor has experienced in procurement and fabrication of piping.

The evaporator units have been erected and installed and most of the steam piping to the drum and evaporators has been hung in place. The piping subcontractor elected to fabricate all of the steam piping in his shops to the nominal piping dimensions shown on the construction drawings (disregarding suggestion that allowances be made to accommodate construction tolerances on the supporting structure and system components). The subcontractor is now experiencing some difficulty in adjusting piping to provide the required cold-springing which involves very close tolerances. Although some difficulty in achieving the required installation is anticipated, it does not appear that the expected delays in the construction of this plant will adversely affect the completion of the entire plant or its operation.

d. Fuel Cycle Facility - The construction of the Fuel Cycle Facility is approximately 45% complete with major emphasis on the erection of the shielding cell. Erection of the gas-tight steel liner in the argon cell is continuing and has been completed up to the level of the ceiling of the cell.

and work has started on the erection of the ceiling plates. Each lift of heavy concrete is poured after the liner section has been completely inspected and tested.

Conduit and duct work are being installed in the basement, structural steel erection for the building wall and roof structure is essentially complete, and work has started on the masonry and panel outside building walls.

2. Installation of Equipment

The critical equipment in the plant such as the reactor and primary system, primary heat exchanger, primary sodium pumps, fuel handling equipment, reactor control systems, fuel process and fabrication equipment, etc., are to be installed after the completion of building construction. This work will be performed by an installation contractor (Package 4) under the direction of the Laboratory staff.

During September, the Package 4 contractor established two-shift operations. The second shift effort is being concentrated primarily in the Reactor Plant where the total work effort is limited both by the confined work area available and restricted operations due to the activities of the construction contractor who is still completing his work in the building.

Package 4 work in the Lab and Service Building is essentially completed and should be finished in October. The small amount of work remaining is being accomplished on a "fill-in" basis to adjust work loads to maintain a constant work level for the various craftsmen.

The machine tools have been set up in the Power Plant machine shop area and this shop will be activated during October. It will be used by the Package 4 contractor for fabrication of parts and modifications required during equipment installation. Remaining Package 4 work in the Power Plant must await completion of work by the construction contractor which is anticipated early in October.

Package 4 work in the Reactor Plant has been confined to the primary tank area and the preparation of components to be installed in the primary tank. The primary tank has been thoroughly cleaned and washed with acetone, and critical surfaces and areas are protected from future contamination. Scaffolding, lighting, ventilation, etc., required for work within the primary tank has been installed. The permanent magnet for the d-c auxiliary pump and the intermediate heat exchanger shell have been installed in the primary tank. Preparations are underway for the installation of the reactor grid which is the first step in the installation of the reactor and vessel assembly in the primary tank. The rotating shield plugs and the primary sodium pump shield plugs have been filled with shielding (steel balls in their lower compartments and heavy concrete in their upper compartments). Installation of the bearing rings, bearings, and other equipment for the rotating plugs has been started. After curing of the shielding concrete, the plugs will be installed for alignment of the reactor components. Assembly of the primary sodium pumps is also beginning in order that these can be installed after the shielding concrete has cured in the plugs. Work in the Reactor Plant has been restricted by

the construction contractor's activities, particularly his checkout of the thimble and shield cooling systems which have limited the access to the top of the primary tank. It is estimated that the construction contractor will complete his work by about mid-October, after which this work can proceed on an unrestricted basis.

3. Engineering

As reported in the August Progress Report, the Architect-Engineer (H. K. Ferguson Company) has completed the construction design including the final instrumentation and control schematics and the plant systems maintenance and operating manuals. All final record documents will be transferred to the Laboratory during October.

A final review is being made of all panel board detail drawings and wiring diagrams indicating corrections and additions to be made by the Package 4 contractor. The first issue of the instrumentation drawings for panel boards in the Power Plant has been completed. This issue includes the data logger, electrical panels E-1 through E-7, the utility panel, and the trend recorder racks.

Electrical design of the physical installation of the conduit, cable trays, junction boxes, etc., associated with the reactor system components is essentially completed. This includes the special pre-wired raceways to be installed around the upper primary tank cavity. The distribution of cabling from the terminal points provided by the construction contractor to the various pieces of equipment to be installed in Package 4 is in progress. This is being coordinated with mechanical requirements of the equipment to provide maximum utilization of the depressed area above the primary tank. This work is being coordinated with the equipment installation schedule and the electrical installation work will commence shortly after the construction contractor has completed his work.

4. Procurement

Procurement of equipment by the Laboratory is proceeding satisfactorily for the most part. In addition to delays incurred in the superheater fabrication (described in the August report) there is some concern about the fabrication of the intermediate sodium-to-sodium heat exchanger. Assembly of the unit has started; however, it is obvious that the promised delivery date of September will not be met. Based on past experience, it will be necessary to continue to expedite this unit to avoid future delay. It is not anticipated that the Package 4 work will be delayed unless the suppliers' performance falls short of revised estimates. Fortunately, this is one of the last pieces of equipment scheduled for installation.

With the exception of the possible delay in delivery of the heat exchanger, the equipment to be installed by the Package 4 contractor is being delivered in accordance with the installation requirements.

5. Component Development - Reactor

a. Reactor Coolant Flow Distribution - Studies of the flow relationships in the 0.6 scale EBR-II reactor model using water as the test fluid

continued at the Franklin Institute. One of the major objectives of this program is to establish with reasonable certainty that predicted flow rates through each subassembly will in fact be achieved.

Previous results from this test program indicated that except for Rows 4 and 5 (the peripheral rows of the reactor core), the average flow rate per subassembly is very close to that desired. The flow rates in Rows 4 and 5 appear to be 6% or 7% lower than expected.

In an effort to determine the modifications to the present orificing which will be necessary to achieve flow rates in Rows 4 and 5 closer to that desired, nine sets of data have been obtained with various combinations of flow conditions. These combinations involve restricting the flow in selected reactor rows to determine the effect of plenum flow and flow to adjacent reactor rows on these selected rows, and in particular, Rows 4 and 5.

These recent tests show an impedance to flow into subassembly inlet holes when an adjacent subassembly row also has high velocities in those of its inlet holes which lie in the same plane. A mathematical relationship has been established and its several constants are now being evaluated from results of these flow tests. This relationship will permit the reactor pressure drop relationships for each reactor row to be calculated. With the experimentally determined EBR-II subassembly hydraulic characteristics, the desired reactor flow values can be obtained. This relationship is being established for application to future core loadings as well as the first loading. An additional experimental test to prove the relationship at partial flow rates may be undertaken.

b. Fuel Element Stability - In any fast reactor (including the EBR-II) the need for achieving very high power density necessitates fuel elements of extremely small cross-sectional dimensions. Assurance that such elements can exhibit the mechanical strength required under the severe operation conditions requires experimental demonstration.

As reported in the August Progress Report, a test of an EBR-II prototype subassembly under full flow (about 9.9 ℓ /sec - 157 gpm) and operating coolant temperature (about 510°C - 950°F) for approximately 600 hr demonstrated that the fuel element as designed is completely satisfactory with respect to mechanical stability, as there was no evidence of warping or of vibration.

Additional visual examination of the fuel elements upon conclusion of the mechanical stability test revealed that approximately 10 of the 91 elements had areas of pitting which appeared as pin pricks on the outside wall of the element tubing. Very slight traces of this pitting condition were also observed on five additional elements. The pitting did not occur on adjacent tubing areas and no relationship between the pitting and the flow pattern could be established. The location of the pitting areas was also random, occurring on both the upper and middle portion of the fuel element tubing. Two of these fuel element tubes displaying the most extreme examples of pitting were examined in greater detail by the Metallurgy Division.

The results of the metallurgical examination of the two tubes

indicated that the pits were a result of intergranular attack or cracking resulting in the loss of complete grains from the outer surface of the tubes. Carbide precipitation at grain boundaries ranging from discontinuous to continuous envelopment of grains was observed. These carbon-rich grain boundaries were attacked owing to the relatively high sodium oxide content of the fluid sodium (no effort had been made to control oxide concentration in the hydraulic test loop).

Chemical analysis showed that the pitted fuel element tubes contained 0.09% carbon. This test was conducted with fuel elements fabricated from an old stock of tubing procured early in the experimental program when the primary emphasis was placed on dimensional tolerances, straightness, etc. Subsequently, the specifications for the production lot of tubing were established to require 0.04% maximum carbon. Certified chemical analyses have been received on all of the stainless steel employed in the fabrication of the first EBR-II fuel loading and, in addition, the Laboratory has performed random verification analyses on these parts.

It is also true that a lower sodium oxide concentration would have reduced the pitting on the test element tubes. The oxygen content in the EBR-II primary system sodium will be less than 10 ppm

Because of these controls, the pitting phenomenon observed in this test is not expected to occur in the EBR-II. The results of this test (which were entirely secondary to the basic purpose and objectives of the test) verify the importance of these controls.

The hydraulic stability test is being continued for a 1000-hour test on the 91-element fuel bundle. The fuel elements with pitted tubes have been replaced and the remainder returned to the test loop. Upon the conclusion of these tests, it is planned to re-examine all the fuel elements for evidence of pitting and also to analyze some of the previously unpitted tubes for carbon content

c. Fuel Transfer Arm Testing - Fuel handling in the EBR-II reactor will be performed remotely by a number of electro-mechanical devices. The fuel handling system, much of which is submerged below the surface of sodium in the primary tank, includes an unloading and refueling machine, transfer port, transfer arm, storage basket and a gripper and holddown mechanism.

The fuel transfer arm mechanism forms the link between the fuel unloading machine, the storage basket, and the gripper and holddown mechanism. Its function is to transfer fuel subassemblies between the gripper mechanism, storage basket and the transfer port location. This device effects the fuel transfer remotely by means of a horizontal arm (rigidly connected to a vertical shaft extending upward through the top biological shield) which travels between these positions, moving through a circular arc during the transfer operation. The fuel subassembly remains entirely submerged in sodium in the primary tank during this transfer operation. The fuel transfer arm mechanism is used for both loading fuel into the reactor and subsequent removal of this fuel.

The fuel transfer arm mechanism, approximately 22 ft in over-all length, was completely assembled and operationally tested on an elevated mockup structure. The various design features of the unit, such as load sensing devices, interlocks, elevation (location) adjuster, etc., were tested and slight improvements incorporated. The unit was then given a series of final operational tests and performed satisfactorily. The entire unit was then helium leak tested to ensure its leak tightness. The unit was disassembled and crated, and is now ready for shipment to the EBR-II site.

d. Festoon Cable Test for EBR-II - The purpose of the festoon cable test is to provide information concerning the reliability and mechanical integrity of the 200-conductor cable which will supply power, instrumentation, and control to the various electrical components located on and within the small rotating shield plug. The rotation of this shield plug during refueling operations requires a means of taking-up and paying-out this cable and prevent kinking or snarling and subsequent damage to the conductors. Therefore, the cable is carried over large rollers, or pulleys, from the rotating plug to the wall of the reactor building where it is hung in festoons on trolleys running on a rail.

This unique arrangement for handling such a large diameter cable required that a test be carried out to develop methods of supporting and moving the cable and to determine if the severe bending and twisting to which it is subjected would damage the insulation or conductors. Hence, the complete system, including a simulated rotating shield plug, was assembled in a full-scale mockup.

It was believed that the most likely cause of cable failure would be the breaking of one or more conductors, rather than insulation failure. Consequently, in order to monitor the continuity of the cable conductors, a circuit was designed in which the coils of several relays are connected to a 120-volt a-c supply, each through a series connected group of cable conductors. The contacts of these relays are so connected as to stop the cable drive and actuate an alarm in the event of conductor breakage. In addition, the conductor shields are maintained at ground potential, and are connected to a circuit which detects a short circuit between any conductor and a shield.

After approximately 1600 cycles of operation a spot check of insulation integrity was made on a 20% sample of the conductors. This test consisted of the application of a 100-volt, 60-cycle potential between adjacent conductors. Certain of the conductors were also tested at 1450 volts, 60 cycles. No dielectric failure occurred.

With concurrence of the cable supplier, it was deemed advisable to test the cable under the same conditions of conductor heating as those expected in the final installation. Accordingly, on September 21, the electrical connections to the test cable were revised so that the central three conductors in each of the twelve subcables carry 5 amperes at 460 volts, three-phase. This closely approximates the actual steady-state reactor load supplied through the cable. The balance of the conductors remain connected as before.

In addition to the above test, the mockup has provided information on the mechanical operation of the apparatus which permitted modification of certain components and increased reliability of the system.

The mockup was completed and operation commenced the first week in June, 1960, and has continued 8 hr per day since that time, except for several days when modifications were being performed. No shutdowns occurred as a result of damage to the cable or mechanical components in the system.

As of this date, the cable has completed 2250 complete cycles, one cycle consisting of a full travel of the cable from one end of the overhead rail to the other end and back, or the rotating plug making two revolutions in one direction and two revolutions in the opposite direction. There is no evidence of wear or deterioration on the outer jacket and in all respects it appears the same as when installed.

The test will be continued until about the second week in October when it will be necessary to dismantle the system for the shipment to Idaho of those parts which will be used in the system installed in the reactor plant.

On the basis of information gained in this test, the components were designed and parts are on order for the festoon cable arrangement to service the large rotating plug. This cable, although it contains only 131 conductors, will be identical in size to the small rotating plug cable and associated equipment will be patterned after that used in the test.

e. EBR-II Rotating Plug Freeze Seals - The purpose of these seals is to prevent gases from leaking into or out of the primary tank around the peripheries of the rotating plugs. Each seal consists primarily of a vertical circular "blade" attached to the periphery of each plug, and of a mating trough attached to each plug stationary support. The blade moves within the trough, which contains "Cerrotru" (58% bismuth - 42% tin alloy, 281° F melting point) in which the blade is immersed. When the reactor is operating, the plugs are not rotated, and the Cerrotru is frozen to permit the seals to resist a pressure differential. A mockup of the seal has successfully sustained pressure differentials up to 5 psi (although the normal pressure differential will be only a few inches of water). When it is necessary to rotate the plugs with the reactor shut down, the Cerrotru is melted, at which time it will be capable of withstanding a pressure differential of approximately 0.5 psi.

Preliminary experiments indicated that upon freezing, the Cerrotru separated from blade and trough, permitting gas leakage. Accordingly, a system in which only the upper half of the alloy in the trough was frozen was tested and found to be satisfactory.

Using these results, the seals were designed to incorporate the following features:

(1) "Fire Rod" heaters for melting the Cerrotru are located at two different levels within the blade; those at the lower level, which

are in direct contact with the Cerrotru, are used for maintaining both the half molten-half frozen condition and the completely molten condition, while those at the upper level are used to aid in making the transition from the half molten-half frozen condition to the completely molten condition.

(2) At each level the heaters are spaced circumferentially about 7 in. apart; heaters at the upper level are spaced halfway between the lower level heaters. The lower level heaters are connected by a copper ring to reduce circumferential temperature gradients.

(3) The seals are cooled by circulation of shield cooling air around one side of each trough.

(4) Thermocouples are provided for measuring Cerrotru temperatures at four different levels. At each level the thermocouples are spaced circumferentially about 15 in. apart. The first (top) level of thermocouples must indicate that the Cerrotru is safely above the melting point before plug rotation is permitted. The fourth (bottom) level thermocouples provide overheat protection. The third level thermocouples provide complete molten control, and the half molten-half frozen interlock. The second level thermocouples provide half molten-half frozen control, and half molten-half frozen interlock.

To test the operation of this seal design, a model seal, full-scale in vertical and radial directions and about 2 ft long in the circumferential direction was constructed and operated. Some of the information gained from the model and from extensive supplementary analyses and analogue studies is listed below:

(1) The heaters can maintain any condition desired using only a small fraction of their rated capacity. This should insure long heater life.

(2) In the half molten-half frozen condition, the temperature difference from bottom to top of the alloy was only about 35°F a situation tending to aggravate control problems. Analysis of the design indicated that the addition of cooling fins to the upper half of the trough exposed to the cooling air should increase the Δt to 50°F. Fins are provided on the EBR-II seal troughs.

(3) Changing the seal from the half molten-half frozen condition to the completely molten condition takes about one (1) hour. The seal returns to the half molten-half frozen condition from the completely molten condition (all heaters off) in about one and one-half hours.

(4) In the half molten-half frozen condition, the third and fourth (bottom) level thermocouples, which are in the molten region, indicate very little change in temperature as long as any part of the seal is frozen. The second level thermocouples do indicate the condition of the seal and are employed to control the half molten-half frozen condition.

(5) Setting the completely molten condition control point at 350°F will insure that it is impossible for frozen alloy (281°F) to exist anywhere between the indicating thermocouples.

f. EBR-II Automatic Flux Control - An automatic flux control system is being designed for EBR-II which will control reactor power to $\pm 1\%$ from 10% to 100% of full power. An on-off control system will be used in order to utilize the control motors and power contactors provided for manual operation.

Test data were taken on the simplified control circuit and operational amplifier. The gain and linearity of the unit was satisfactory but the 1-volt rms noise level (60 cps hum) and 3 volts per day drift rate (under optimum conditions of 100-hour break-in and filament voltage regulation) was not acceptable. The original bread-board design was partially the cause of the poor hum and drift characteristics. A new and improved chassis model was constructed which will provide better shielding and temperature characteristics. Test data are being taken on the new assembly.

g. EBR-II Control Rod Reactivity Generator - A sinusoidal reactivity generator and wave analyzer are being designed to measure the transfer function of EBR-II. The reactivity generator will utilize a specially designed rotary mechanism which will fit into a control rod location.

Various methods have been investigated for measuring and indicating the reactor transfer function phase shift. Conventional one-to-one synchro transmitter-receiver pairs were tested and found to have about a one-degree accuracy. The use of gear trains to improve accuracy is being investigated. Work is being conducted on drive layouts and final checking of reactor component details is in progress. Some redesign has also been done to satisfy physics requirements.

h. EBR-II Startup Instrumentation - Special instrumentation is being provided for EBR-II startup. Two moveable thimbles will be installed in control rod locations. High temperature fission chambers capable of operating at 455°C will be used. A purchase order has been issued to Westinghouse Electric Corporation for four such high temperature fission counters and for eight high temperature cables.

Two separate channels of temporary electronic equipment, high voltage supplies, pre-amplifiers, log count rate amplifiers, period amplifiers, scalars, and audio count rate circuits will be installed. A partial shipment of this equipment, ordered from General Electric, has been received. Tests are in progress to check the performance specifications of the log count rate period meter and preamplifiers.

A safety circuit, in the form of a memory circuit, has been designed to indicate the existence of an empty hole in the core and guard against reactor start-up under this condition. During the refueling operation the circuit can follow the existence of two empty holes and will sound an alarm if an attempt is made to empty a third.

i. Detector for Fuel Element Failure - A device is being designed which will detect the delayed neutrons from fission products as a means of

detecting a fuel element failure. The operating environment is extremely difficult because of the high radiation background.

The effect of a high background of gamma radiation from Na-24 on the fission counter and associated circuitry proposed for use as a detector of fuel element failure in a fast reactor has been investigated. An appreciable increase in background counting rate was observed when a gamma field of approximately 3×10^4 r/hr existed at the detector location. It was found that this pronounced increase in background counting rate was due to neutrons originating through (γ, n) reactions involving the sodium gamma rays and the small amount of deuterium present in the paraffin moderator. A shift to graphite moderator restored the background level to that existing when no gamma source was present. However, the appreciably larger slowing down area in graphite reduced the sensitivity of the counter and associated moderator to a fraction of that obtained previously with paraffin.

Based on the observations with samples irradiated in CP-5, it is estimated that a fuel element leak can be detected in which fission products are entering the coolant at a rate equivalent to 50 mg of U-235 per minute, provided that the background conditions simulated in these tests at CP-5 are not exceeded. The background condition and observation of delayed neutron emission one minute after the material leaves the reactor can probably be satisfied in the sodium cold trap cell of EBR-II. It should be possible to detect the failure of a single fuel pin, provided that the corrosion rate is such that the fuel pin would be dissolved in approximately one day. In all likelihood, initial failure of the cladding would be accompanied by a sharp increase in neutron counting rate, but it is felt that the delayed neutron moderator developed in these tests should be supplemented by a confirming device which would detect longer-lived fission product activities in the blanket gas. Further studies will be performed to optimize the moderator and shielding in order to bring up the efficiency of the device.

6. Component Development - Steam Generators

The ninth and final evaporator unit, a spare, is now completed. Final disposition of this unit has not yet been determined, but it is expected that it will be stored in Lemont. Significant progress toward the solution of the welding problem referred to in the August report has been achieved, thus making possible the completion of the spare unit. The welding problem which involved severe cracking in the craters and overlap areas in the tube to tube-sheet welds is believed to be related to insufficient deoxidation of the weld deposit in the short time interval between deposition and solidification.

All of the components necessary for the four required superheaters plus one spare are on hand. Assembly of the first superheater, however, has not yet started. The delay has been caused by the inability of an outside processor to upset-forged the ends of the bonded tubes, an operation which is required to permit the Laboratory's Shops to prepare the tube ends properly for welding. Laboratory personnel are working closely with the processor in order to solve this problem to permit unit assembly to begin.

The ultrasonic through transmission inspection of the superheater duplex tubes for bond quality has been completed. A total of 291 superheater

duplex units were tested. Of the 291 duplexes tested, three were rejected and subsequently evaluated by metallography. Metallographic examination revealed that two of the defective specimens contained inclusion in the wall near the bond line while the other rejected tube had dissimilar grain structure in several areas. This latter difference in metallurgical structure was probably a result of improper heat treatment.

7. Component Development - Fuel Reprocessing Facilities

a. Fuel Cycle Facility - Equipment - Preliminary designs of service sleeve plugs required for the Argon Cell of the Fuel Cycle Facility have been completed. Fabrication of transfer cell equipment is nearly complete.

The five-ton cranes built by PACECO have been shipped to the Idaho site. Fabrication of the operating manipulators for the Argon Cell has started.

A 54-conductor rubber covered cable will be used to feed power to the manipulators in the Argon Cell. Irradiation hardens the cable sheath and it breaks under stress. Bending and twisting tests have been performed on irradiated samples of the cable. The maximum useful life under anticipated cell conditions is estimated to be six years. A new design of the 1/10-HP Howard motor for use in the manipulator arm was test run in a dry argon atmosphere. Projected brush life was about 500 hours operating time, twice that of the old design motor.

b. Fuel Disassembly Equipment - The reactor core subassemblies will be dismantled in the air cell of the Fuel Cycle Facility. The fuel and blanket elements from these subassemblies will be inspected and transferred to the Argon Cell for the operations of decanning and reduction to short lengths which are preparatory to the melt refining operations.

A preliminary layout of the dismantling and inspection equipment in the air cell has been made. Preliminary design of the dismantling device is proceeding. The first step in the dismantling operation is to cut the hex tubing sheath on the subassembly. This has been done on several test fuel subassemblies by driving two opposing sets of rolling cutters held in a yoke across the opposite flats of the hex tubing. However, this technique required very high forces to part the cut, since it was not complete at the corners, and the high force produced by the cutters normal to the hex tubing deformed the inner structure of the subassembly. An alternate technique has been checked experimentally and a device is being fabricated for loading one or three cutters in contact with the hex tubing as it is rotated.

Conceptual design studies are being carried out on devices for removing the fuel elements from the support grid.

An inspection device for detecting excessively distorted elements and elements with damaged support tips has been completed in layout design.

Work is continuing on the design of the component devices required to decan the individual fuel elements. A full scale wooden model of the Mark II Spiral Decanner has been built and most of the design features of this re-engineered version of the machine (being tested in Bldg. 6) have

been completed. The preparation of detailed fabrication drawings has been started.

The spiral decanner in Bldg. 6 has decanned about 2000 reject fuel elements with the same tool, drive rolls and bearings, and no apparent change in performance has occurred.

A wear test has been performed on a pneumatic actuating cylinder which had a chromium plated sleeve bore and cast iron piston rings. The dew point in glove box was maintained at -50°F or less, and no lubrication was used. The cylinder was operable after 52,000 cycles with a moderate increase in leakage.

c. Equipment and Methods for Remotely Controlled Fuel Refabrication - Equipment and methods are being developed for the remote refabrication of irradiated EBR-II fuel. A class H-insulated synchronous motor, a strain gauge type pressure transducer, a microswitch and an unlubricated cylinder assembly have been exposed to gamma radiation to a total dose of about 10^9 roentgen (see Progress Report, July, 1960). The motor was operable after this exposure although observable changes had taken place in the insulation. The pressure transducer showed signs of being slightly influenced by radiation level transients. These components are employed in the design of the fuel pin processing and inspection equipment.

Designs were completed on the injection casting equipment for the argon cell and are nearing completion in the subcell injection casting equipment. Design of control equipment was started for the injection casting stations. Designs were completed and entered for construction of a new leak detection station which may combine in one air cell apparatus the functions of pressure reduction and mass spectrometer leak detection.

8. Fabrication of Initial Fuel Loading

a. Fuel Pins and Fuel Elements - Sixteen-hundred fuel pins were loaded and bonded during the month. In addition 550 rejects were stripped and reloaded. The total of accepted enriched elements for EBR-II is now 4400 elements. About 600 additional elements are ready for the cyclograph bond examination. Assuming the present rate of production, the entire core loading for EBR-II should be ready toward the end of 1960.

b. Blanket Rods - Sodium is used as a bonding agent between depleted unalloyed uranium slugs and stainless steel jacket tubes in the EBR-II blanket rods. Nondestructive eddy current inspection is used to determine bond integrity and sodium fill height in each blanket rod. Non-bonded rods are recycled through a bonding furnace resulting in a low ultimate percentage of rejects.

A total of 10,075 inner and outer blanket rods have been loaded, bonded, inspected and accepted as of September 20, 1960. This represents 84% of the number required for the first reactor loading. Fourteen inner blanket subassemblies have been assembled.

Equipment for salvaging the uranium from damaged or otherwise non-acceptable blanket rods has been successfully tested and this phase is

proceeding satisfactorily. Seventy-five of these rods have been disassembled to date.

c. Stainless Steel Hardware - Sufficient upper-lower blanket tube components have been produced to complete this phase of the work.

15,733 of a required 21,000 fuel tube components have been produced and delivered to the assembly operation. Enough tips and tubes are on hand to produce a total of approximately 22,500.

12,084 of the required 16,000 inner-outer blanket tube components have been produced and delivered to the canning line. Further progress in this phase is being held up pending delivery of tubing which had been returned to the vendor to be reworked to the specified dimensional tolerance.

d. Bonding Supporting Investigations: Cyclograph - A furnace was constructed to allow the investigation of void detection by eddy current methods of the sodium bond at elevated temperature ($\sim 140^{\circ}\text{C}$). Investigation is based upon the assumption that the bond sodium will reside in the annulus in a molten condition as a consequence of the gamma heating contributed by residual fission products in the fuel alloy. This condition would render unnecessary the gas quenching operation but would impose a higher operating temperature upon the testing equipment. The furnace constructed permits an element to be uniformly heated along its length to the required temperature after which the element can be drawn downward through the cyclograph coil.

As observed in the previous investigations, the drastic change in the electrical properties of the bond during phase transition are reflected in a pronounced change in the sensitivity of signal detection. Because of this, the range of the instrument was found to be inadequate for effective recording of the trace. Future efforts will be directed toward the modification of the instrument range. It is of interest to note that the early investigations with the "hot cyclograph" demonstrated the appearance of shrinkage voids upon fraying of the bond sodium. It was the observation of this effect that led to the inclusion of quenching as a step in the fuel element production.

e. Leak Detection - In an effort to recheck the effectiveness of the current methods of production line leak detection operations, additional tests are being performed. The fuel elements rejected from the production line are disassembled with the top weld and approximately 3 cm left intact. These "heads" will then be subjected to various cleaning operations intended to remove sodium or sodium oxide which may be plugging the flaw. After cleanup, the segments will be rechecked to determine if the readings are consistent.

f. Fuel Irradiation Studies - Two EBR-II fuel pins (#19 and #20) are still in the CP-5 reactor and are accumulating burnup at low temperature (about 370°C). Number 19 has a burnup of $\sim 1.75\%$ and 20 about 0.75% . A possibility exists of an increase of CP-5 power from 2 to 5 Mw within the next month. If this condition is actually fulfilled, temperatures in the fuel specimens should approach design temperatures (in the vicinity of 650°C). Under these conditions irradiation of these elements would be continued to a burnup of 2% or better before discharging from the reactor.

9. Process Development

The initial core loading of EBR-II, a 50 percent enriched uranium-fissium alloy, will be reprocessed by melt refining. A supplementary (dragout) process involving a liquid metal system will be used on the skull residues to recover uranium and plutonium and to maintain the proper concentration of noble metal fission products in the recycled fuel. A second liquid metal process will be used to concentrate the plutonium made in the reactor blanket so that it can be used for replenishing the reactivity of the core alloy.

Process flowsheets based on pyrometallurgical methods are being developed. The program includes studies of the chemical and physical variables and parameters; development of engineering operations into workable plant-scale techniques; and demonstration of the processes under levels of radioactivity approaching those expected at the plant.

a. Melt Refining Process Technology - The first experiment on the evolution of fission product krypton and xenon from an irradiated EBR-II type fuel pin was completed. The fuel pin, consisting of 10 percent enriched uranium and five percent fissium elements, was irradiated to a total atom burnup of 0.96 percent and cooled 49 days prior to the experiment. On heating at a rate of about 5°C per minute, the diameter of the pin increased from 0.14 to 0.20 in. within the temperature range of 650°C to 800°C. Rapid evolution of krypton-85 and xenon-133 began during this period, and stopped when the pin reached a temperature of about 1020°C. At this temperature the pin began to lose its cylindrical shape, and at 1080°C, melting appeared to be complete. The patterns of evolution of krypton-85 and xenon-133 as a function of time and temperature were identical.

A 97 percent yield was obtained on melt refining 2 kg of sodium-coated pins. This is a considerably higher yield than obtained in previous runs. The improvement is attributed to elimination from the apparatus of all materials which might contribute atmosphere impurities, and to several modifications in procedure designed either to improve degassing or to eliminate possible contamination.

Studies on the nitridation of unirradiated uranium and uranium-fissium pins show that sodium coatings increase the rate of uranium nitride formation on the surface in a static argon atmosphere containing about five volume percent nitrogen, the maximum concentration anticipated in the Argon Cell of the EBR-II Fuel Cycle Facility. At a pin temperature of about 300°C the nitridation rate is significant, suggesting that measures may be necessary to prevent "pinshell" formation during storage of the fuel prior to melt refining. Further experiments are planned to determine the effects of temperature, nitrogen concentration, and irradiation.

In tests of alternate materials for use in a melt refining furnace, a fibrous potassium titanate grain retainer was found to be a very effective heat insulator, but it possesses less strength than rigid Fiberfrax retainers. Carbon wool was not as effective an insulating material as silicon carbide, graphite, or carbon black powders.

Further information regarding the first series of melt refining demonstration runs using 400-gram charges of irradiated uranium-fissium fuel alloy will be reported when the analytical results for the runs become available.

b. Processing of Melt Refining Skulls - Preliminary results on the reduction of oxidized melt refining skulls by magnesium at 800°C are encouraging, with yields exceeding 90 percent. In processes for the recovery of uranium from melt refining skulls, this reduction represents a means of introducing the uranium into liquid metal solutions for subsequent fission product separations.

Calcium alloys and molten salts are being examined as potential media for the extraction of rare earth fission products from plutonium-uranium fuels.

Three essentially quantitative reductions of fissium skull oxide were made during the past month. Two of these reductions were effected at 800°C in a 10 percent magnesium-zinc solution; the third was effected at 800°C in a magnesium-0.5 percent sodium system. A Dow-230 fused chloride flux was present during one of the reductions in the magnesium-zinc alloy. At least part of the difficulty in showing high reductions previously was traced to the use of Vycor sampling tubes. Generally low results were obtained with these tubes, indicating some type of reaction with the melt or sample. Tantalum sample tubes with graphite frit filters are currently being employed and are proving to be very satisfactory.

Yields of skull oxide from oxidation of fissium-cerium skulls in plant size crucibles were greater than 99.5 percent in two runs and 98.4 percent in a third. Material in the latter run did not dump out as well as in other runs. Analyses of particle size fractions showed cerium to be concentrated in the fine particles (-325 mesh) and molybdenum and ruthenium in the coarser particles.

c. Engineering Development for Liquid Metal Processes - Work has been started on the physical separation of magnesium oxide from liquid metal phases. In the first preliminary run, a separation factor ($\text{U: MgO product/U: MgO in starting material}$) of 14 was obtained. This is not considered sufficient for a process, but should be improvable.

Work on the separation of uranium-zinc intermetallic from zinc mother liquor by filtration has been resumed. This does not appear to be a difficult separation by either filtration or decantation.

Several additional retorting runs were performed in tantalum for recovery of uranium from zinc systems. Problems were encountered with sticking of the uranium product to the tantalum crucible when the charge was prepared in the retorting crucible. Apparently some wetting resulted by this procedure. This behavior in tantalum crucibles contrasts with previous results obtained in ceramic-coated graphite crucibles in which no problems of product removal were encountered.

d. Materials Evaluation Studies - An extensive corrosion and stability testing program is under way to obtain information on materials and fabrication

techniques for liquid metal processes. A number of corrosion tests have been made and a solution stability run is in progress. However, interpretations and analyses are not sufficiently complete for reporting

10. EBR-II Analysis

a. Model for Analog Studies - A mathematical model of the thermal, mechanical and nuclear properties of an EBR-II fuel element is being formulated for analog studies relating to reactor dynamic behavior. The effect on the feedback of the uranium phase change in the freely floating pin was found to be significant and markedly non-linear. This means that the phase change effect cannot be neglected to simplify the mathematical model and hence it is being incorporated

b. Coupled EBR-II Concept - Exploratory work is being done on conceptual designs for a possible coupled loading for EBR-II.

The breeding ratio of the coupled fast-thermal system of the design being considered is 1.06 for metallic uranium fuel. This low value, compared to 1.23 for a more ideal system, occurs largely because only 10.5 cm of outer blanket is provided in order to use a nearly optimum beryllium thickness. It is felt that calculations can satisfactorily determine the breeding ratio which would occur if more space had been available.

The results of loss-of-sodium accidents for the current coupled concept have been calculated. They are as follows:

<u>Region of Sodium Loss</u>	<u>Change in Reactivity, % $\Delta k/k$</u>
Core and control rod region only	-3.08
Both hex can rows of inner blanket only	-0.48
Outer hex can row of inner blanket only	-0.20
Inner hex can row of inner blanket only	-0.28
Beryllium moderator region only	+0.355

III. STUDIES AND EVALUATIONS (040116)

A considerable effort is devoted to carrying out studies and evaluations at the request of the Commission. The status of the nine evaluations under way is as follows:

Completed:

- a. Evaluation of a Variable Moderator Reactor proposal.
- b. Evaluation of a proposal for Development of High Temperature Sensors.
- c. Evaluation of a proposal for the development of a pressure transducer utilizing a directly deposited strain gauge.
- d. Review of a proposal for a high efficiency nuclear power plant utilizing a free piston engine.

Still in progress:

- e. A realistic evaluation of current and potential power generation costs appropriate to fast reactor systems with 300, 150, 44 and 12.5 Mwe output.
- f. Evaluation of a proposal for the development of a fuel failure detection system.
- g. The updating of the SGR 10-year program reports.

Initiated during the month:

- h. Evaluation of a proposal for a direct cycle diphenyl reactor.
- i. A further extension of the BWR status reports.

IV. REACTOR SAFETY (040117)

A. Thermal Reactor Safety Studies

1. Fuel-Coolant Chemical Reactions

Knowledge of the nature and extent of chemical reactions with nuclear reactor core metals that may occur in pressurized water or steam is essential to safe operation of reactors. The principal laboratory procedure uses a condenser discharge to provide almost instantaneous heating and melting of metal wire in water or steam. The energy input to the wire indicates reaction temperature; the transient pressure measures reaction rate; light emission indicates time-temperature behavior; hydrogen generated gives extent of reaction; and particle size of the residue indicates the surface area exposed to reaction. A second method consists of heating the metal inductively and then subjecting it to a steam pulse to induce a metal-steam reaction.

Studies of the kinetics of metal-water reactions under reactor incident conditions are being made in the TREAT reactor.

A series of runs with 60 mil zirconium wires was carried out using the new high pressure condenser discharge apparatus. Several runs were made at an initial metal temperature of 1840°C (the melting point) with the metal remaining solid, and several were made with fully liquefied metal also at the melting point.

The results indicate that the over-all reaction (4 to 9 percent solid metal reacted) depends strongly on the pressure of water vapor up to some point between 0.5 and 5 psia. A further increase in pressure causes no additional reaction. This limit suggests that the reaction becomes controlled by the transport of water vapor through a hydrogen film. The transport would be expected to be relatively independent of pressure.

The eighth series of CEN metal-water experiments was completed in TREAT. A 98-megawatt-second reactor burst on a 96-millisecond period converted a 93 percent enriched, 34 mil diameter uranium wire in a helium atmosphere into particles and a fine dust. A 47-megawatt-second burst (273 calories per gram uranium estimated energy input) on a 96-millisecond period melted a 34 mil wire into particles. This gave 28.3 percent uranium-water reaction as measured by the hydrogen production. This is the highest extent of metal-water reaction observed to date in the in-pile experiments.

Two experiments on 90 weight percent aluminum-10 weight percent U_3O_8 , unjacketed, cermet pins, 93 percent enriched, resulted in complete melting of the pins with bursts of 235 and 440 megawatt-seconds. The unjacketed cermet fuel pins did not break up. It is concluded, therefore, by comparison with previous experiments with Zircaloy-2 jacketed pins, that the formation of particles is probably caused by the cermet spraying out through a hole in the jacket. Two pressure spikes were also observed for the first time during the current series. The two cermet pins gave sharp jumps in the pressure-time oscillograph record of 90 and 300 psi with 0.7 and 3.8 percent aluminum-water reaction. The possibility of a thermite reaction of aluminum with the U_3O_8 in the case of the cermet pins is being investigated.

Figures 1 and 2 show typical results observed on clad and unclad cermet pins after reaction. Figure 3 shows a typical jacketed oxide core pin after reaction.

A 315 megawatt-second burst (413 cal/g core calculated) on an oxide core fuel pin gave extensive cracking of the core with no changes to the stainless steel jacket.

Preparations were completed for the ninth series of transients. Uranium wires will be reacted at various reactor periods while holding the burst energy constant to determine the influence of the rate of energy input on the extent of metal-water reaction.

2. Kinetics of Oxidation and Ignition of Reactor Metals

Studies are being made of the oxidation and ignition kinetics of the metals uranium, zirconium, plutonium and thorium in order to provide information leading to an understanding of the reactions. This knowledge should make it possible to minimize the hazards associated with handling these nuclear reactor materials. Isothermal oxidation on microscope stage, shielded ignition, burning curves, rate of propagation of burning foil and burning temperatures are the techniques being used. In the continuing study of ignition and burning of uranium, zirconium, and plutonium more emphasis is being placed on the burning process. Burning propagation rate studies provide a useful tool to observe the effects of many variables. Addition of halogenated hydrocarbons to air in which uranium or zirconium specimens are burning has been observed to lower markedly the burning propagation velocity. This effect is being investigated.

Previous studies of ignition of uranium cubes in oxygen (ANL-5974) have demonstrated that additions of copper increase the ignition temperature while aluminum and silicon decrease the ignition temperature. The concentration dependence of copper, silicon, and aluminum has now been extended beyond that previously established. Although both low-level silicon and aluminum additions (100 to 600 ppm) decrease the ignition temperature of uranium markedly, in the region of 5000 ppm, only aluminum causes this decrease in ignition temperature. Copper concentrations greater than approximately 1000 ppm increase the ignition temperature about 200°C, while an intermediate concentration causes an intermediate increase.

The use of nickel sulfate or zinc chloride solutions to decrease the pyrophoricity of uranium powder, as reported in France, has been attempted. No evidence of increasing the ignition temperature of spherical or irregular 200 mesh powder or 0.13 mm (5 mil) foil was observed from burning curve experiments after treatment with either solution.

The relationship between ignition temperature and specific area (or particle size) of uranium powder has been extended to include particles as small as 12.5 microns mean diameter. Ignition temperatures in oxygen varied from 240°C for 12.5 micron particles to 260°C for 37 micron particles.

Preliminary studies of the effects of various gas contaminants on

Fig. 1

Typical appearance of Zr-2 clad cermet pin after reaction.

Conditions: Zr-2 clad, cermet core
(90 Al - 10 U_3O_8)
525 Mw-sec burst
63 m sec period

Fig. 2
Typical appearance of unclad cermet pin both before and after reaction.

Conditions: Unclad cermet
(90 Al - 10 U_3O_8)
440 Mw-sec burst
65 m sec period

Fig. 3

Typical appearance of Al clad oxide pin after reaction.

Conditions: Al clad, oxide core
490 Mw-sec burst
65 m sec period

the burning propagation rates of a plutonium alloy indicate that dibromomethane (CH_2Br_2) and bromochloromethane (CH_2BrCl) markedly decrease the burning rate.

B. Fast Reactor Safety Studies

1. TREAT - Experimental

In-pile meltdown experiments in the TREAT reactor are being performed in order to survey types of fast reactor fuel element failure and meltdown product distribution, as well as to determine the mechanisms operative in meltdown and meltdown product movement phenomena.

a. EBR-II, Mark I - Dry Samples - Previous experiments on un-cooled EBR-II prototype fuel elements disclosed that failure of cladding does not take place in short-term (0.2 to 0.5 sec.) bursts in which cladding temperatures do not exceed 950°C . The tenth series of the dry capsule TREAT experiments consisted of three EBR-II prototype elements exposed to "flat top" transients for intervals of time up to 10 sec. in length. The objective was to determine if the elements would fail if sustained at temperatures approaching 950°C for longer periods of time than the early experiments. The results of the three experiments are tabulated below:

<u>No.</u>	<u>Temperature $^\circ\text{C}$ (approximate)</u>	<u>Condition</u>
1	960	No damage sustained by cladding.
2	840	No damage.
3	900	Sodium blew out through penetration approximately 10 cm from bottom.

The results indicated that the failure threshold for EBR-II elements is not lowered appreciably for such longer time-at-temperature exposures. None of the three samples suffered appreciable attack of the steel cladding by the uranium-iron eutectic reaction, although one sample showed a small local cladding failure, through which most of the sodium from the element had been expelled. Hence, the failure observed for that sample is consistent with that to be expected due to a small flaw in the cladding being opened by internal pressure. It is the first failure observed for an EBR-II, Mark I element in which uranium-iron eutectic formation did not play the dominant role, and the first for which material flaws could be hypothesized as the cause.

A series of six experiments designed to explore the effects of higher sample energy upon failure of EBR-II, Mark I samples is under way.

b. Fermi I - Dry Samples - Results of meltdown experiments on dry Fermi I samples form an interesting contrast to those of the EBR-II type elements. Inspections of sections cut perpendicular to the axis of the control sample, and a sample carried to the threshold of failure, were augmented by "electron microsoundings" examinations to identify positively the metals present in various zones in the fuel-clad interface region. Based on the electron microsoundings results it is concluded that the mechanism initiating failure was the opening of cracks in the cladding,

rather than significant dissolution of cladding by the molten uranium alloy core.

Four experiments on Fermi I samples were run at TREAT during the report period. All four were subjected to flat top transients of relatively constant power in an attempt to check the hypothesis (based on results of CP-5 irradiations of Fermi I elements in which no melting occurred, but sharp temperature changes took place) that the cladding cracks were due to stresses arising from the short power duration transients employed previously. Post-mortem examinations have not yet been performed on these four Fermi I samples.

2. Out-of-Pile Furnace Tests

The experiments described in (a) above can be extended in duration in out-of-pile furnace tests. Several preliminary tests were conducted in the high temperature fuel element furnace to determine its effectiveness for such tests. It was found that a minimum time of 240 sec. was required to reach the maximum temperature with the furnace in the vicinity of 1000°C. Following this, three tests were conducted in which the elements were taken to 980°C for progressively increasing periods of time. The results are as follows:

No.	Temperature °C	Time at Temperature sec.	Cooling Time to 700°C, sec.	Comments
1	980	50	40	Warped, eutectic formed
2	980	50	40	Terminated T.C. failure
3	980	75	40	Broad eutectic

After 50 seconds the element was found to have a definite bow to its structure with large patches of eutectic beginning to form along zones of enforced contact. Some small erosion pinholes were observed in the fuel. After 75 sec., the warped element was very largely covered with eutectic with deep and large areas of erosion wherever the eutectic was present. The eutectic appeared to have flowed downward and against the inner cladding wall. No penetration of the cladding took place in either instance. These results, though preliminary, lead to the conclusion that failure can be expected at 980°C somewhere in the vicinity of 100 sec. Further tests are being planned to cover varied times at both lower and higher temperatures.

3. TREAT - Component Development

a. Internal Thermocouple - Development of an acceptable internal thermocouple for instrumenting EBR-II sample elements would afford a method of measuring internal sample temperatures in both dry and sodium-cooled experiments. Internal temperatures must now be inferred from calculations and from cladding temperature measurements. Acceptable direct cladding temperature measurement appears to be unattainable for experiments on meltdown in sodium. One bonded and clad EBR-II, Mark I fuel element, internally instrumented with a tantalum-sheathed, tantalum-molybdenum thermocouple has been produced through a joint effort by

Advanced Technology Laboratories and ANL. Calibration of the element is under way in preparation for future encapsulation in a dry opaque TREAT meltdown capsule for a "proof test" experiment in which the performance of the internal thermocouple will be compared against that of standard cladding thermocouples. It is anticipated that the internal thermocouple design developed will display a time response comparable with that of the cladding thermocouples, i.e., no appreciable lag for TREAT experiments.

b. Gamma-Ray Pinhole Camera - In-pile experimentation on core meltdown phenomena requires special instrumentation for measurement of sample temperature, pressure, and movement.

A series of tests at TREAT by Naval Research Laboratory personnel, designed to display characteristics of a proposed Mark II gamma-ray camera, was noted in the August Progress Report. Based on an evaluation of those tests, as well as earlier tests, it was concluded that the camera is at best a marginal device, even with the improvements incorporated into the Mark II design.

Briefly, the expected performance of a complete Mark II camera system may be summarized as follows:

(1) In order to obtain a signal-to-background ratio sufficient to have the signal from a meltdown sample stand out above the background with a statistical confidence level $\sim 99\%$, a space resolution of about 0.38 cm horizontal by 1.88 cm vertical is necessary.

(2) An EBR-II element of 6% enrichment (appropriate for TREAT experiments) stands out clearly above the capsule background, even in the "worst" case of a stagnant sodium capsule set inside the standard graphite and Zircaloy experimental TREAT dummy fuel element. In this case, the signal from the sample was approximately equal to that from the container.

(3) EBR-II, Mark I sample failure produces a marked drop in camera signal, owing to the expulsion of fuel from the element cladding. This characteristic, which had been predicted on the basis of previous steady-state measurements, was verified experimentally during a meltdown experiment on a 6% enriched sample.

(4) Because of general problems of background and resolution, effective use of the Mark II camera would seem to be limited to experiments on single elements.

c. Miscellaneous Experimental Equipment - A welder was assembled for the purpose of investigating the effectiveness of remote closure of the TREAT capsules for exposure of irradiated elements. The assembly consists of a d-c operated, modified pipe threading machine coupled with a heliarc torch and R.F. arc initiation. Several promising welds were made on simulated capsule sections to demonstrate that this method is entirely feasible.

A conceptual layout was made of the fuel element handling and shipment flask for the TREAT capsules containing irradiated elements.

This flask is designed to handle and contain four capsules at one time and can be separated into two sections for loading flexibility.

V. NUCLEAR TECHNOLOGY AND GENERAL SUPPORT (040400)

A. Applied Nuclear and Reactor Physics

1. Experimental

a. Fission Physics

(1) Delayed Neutron Emission Resulting from the Thermal Neutron Induced Fission of Pu-241 - The experimental measurement of the time dependence and yield of delayed neutrons resulting from the thermal neutron induced fission of Pu-241 was completed. All measurements of the delayed neutron activity were made following a saturated irradiation in the thermal beam of the CP-5 research reactor. A comparison of the delayed fission neutron yields of Pu-241 + n and U-235 + n showed that the Pu-241 delayed neutron activity decayed somewhat faster than that of U-235. The largest differences in the decay curves occurred in the region of the 6-second and 54-second delayed neutron groups. In both regions the Pu-241 yield was suppressed relative to the U-235 yield. The absolute Pu-241 delayed neutron yield per fission was measured relative to that of U-235. The number of delayed neutrons per Pu-241 fission was somewhat smaller than for U-235.

The analysis of the Pu-241 delayed neutron data is continuing and certain systematic trends in the delayed neutron process are becoming evident. This is illustrated in Fig. 4, which shows the time dependence of delayed neutron emission from various isotopes compared to that from U-235 fission. In the figure the ratio of delayed neutron emission from various isotopes compared to that from U-235 is normalized to unity at time zero. The data shown are a summary of measurements at this and other laboratories. Certainly systematic shifts in delayed neutron time dependence are evident but the relation of these shifts to such parameters as the Z and A of the fissile nucleus is not clear. The analysis is continuing with the objective of establishing such an empirical relation thereby enabling the reactor physicist to predict accurately the delayed neutron properties of any fissile nucleus or combination thereof.

(2) Prompt Fission Neutron Spectrum of Pu-241 - An experimental measurement has been completed of the "prompt" fission neutron velocity spectrum resulting from the neutron induced fission of Pu-241. All of the velocity measurements are relative to the U-235 fission neutron spectrum. The results are being combined with n-p emulsion studies to give a complete description of the fission neutron spectrum of Pu-241 from 0.3 to 10.0 Mev. Partially completed analysis indicates that there is little difference between the fission neutron spectrum of U-235 and that of Pu-241. The average fission neutron energies of the two isotopes must differ by less than 100 kev.

Although the experiment was not designed to determine ν (Pu-241) a value for the quantity was obtained. Preliminary estimates give ν (Pu-241)_{th} = 2.95 ± 0.10 assuming ν (U-235)_{th} = 2.43 as a standard. Certainly this value of ν and the energy distribution of fission neutrons are basic "input data" to the reactor physicist considering long term fast reactive systems.

Fig. 4

Time dependence of delayed neutron emission from various isotopes compared to that from U^{235} fission.

(3) The Dependence of Fission Mass Asymmetry on the Rotational State at the Saddle Point - A fragment kinetic study of Th-232 + n fission at incident neutron energies of 1.45 and 1.60 Mev has been concluded. It was presumed that fission at these two incident neutron energies occurs principally from different rotational saddle point bands.* No fragment energetic or mass shifts were detected in going from bombarding energies of 1.45 Mev to 1.60 Mev. Therefore, the conclusion is that fragment mass and energy are not measurably affected by the rotational state at the saddle point as is the fragment angular anisotropy. As a result of this experimental conclusion, further investigation of the collective model is being abandoned as it applies to the fission process.

b. Fast Neutron Cross Sections

(1) The Inelastic Scattering of Fast Neutrons - The fast timing equipment has been utilized, in a routine manner, to measure the inelastic scattering cross sections of thorium, uranium (natural), tungsten, zirconium, and iron. The incident neutron energy remains, as previously, 0.3 to 1.5 Mev. The status of the measurements extends from essentially completed, in the case of thorium to well under way for iron. Typical results in the form of cross sections for scattering to specific residual levels are given in Figs. 5 through 9. These distributions were measured at 90 ± 30 degrees, a range over which the cross sections were measured to be essentially isotropic.

Some time was spent attempting to examine the inelastic scattering to the first level in W-186. The results were not good owing to: (a) the small (12 gm) sample of the separated isotope; (b) the chemical form of the sample (WO_3). The contribution of elastic scattering from oxygen was particularly troublesome.

The millimicrosecond bunching magnet has been received and is being set up. It should increase burst intensity by about one order of magnitude, thereby greatly facilitating the inelastic scattering work.

c. Instrumentation - The basic tools of the experimental physicist are his instruments. His understanding and utilization of them are fundamental to experimental success. During the month the following instrumental work was under way or completed:

(1) Accelerating Tube - The new accelerating tube was installed in the Van de Graaff. It has been in use for about 200 hr. during which the machine operation was remarkably stable. Beam intensities have increased to a total of more than 110 microamps of analyzed protons on a target 4 m from the analyzing magnet. At 3 mev this represents an intense beam of more than 0.3 kw in a circle of 0.125 in. diameter.

(2) Van de Graaff Beam Exit Assembly - The new assembly is in use and has proved to be a very considerable improvement over the old unit. The resulting ease of adjustment and stability of the ion optic

* L. Willets and D. Chase, Phys. Rev. 103, 1296 (1956)

Fig. 5

Fig. 6

Fig. 7

Fig. 8

Fig. 9

system is outstanding. Associated with the new unit is an improved high vacuum system. Using it, pressures of less than 2×10^{-6} mm of mercury in the accelerating tube have been obtained. Moreover, the over-all pumping speed has been improved. As a result of the improved vacuum system the consumption of liquid air has been reduced by a factor of five.

(3) Gas Handling System for the Van de Graaff - Parts of an insulating gas system for rapid transfer and optimum drying of accelerator gas have been installed. Work is proceeding in a satisfactory manner.

(4) Neutron Response Selection in Organic Scintillators by the Pulse Decay Method - The discrimination between the neutron and gamma-ray response of organic scintillators by choosing the proper scintillator time decay component is under study. Using liquid scintillators very nice separation between gamma ray and neutron responses is obtained down to a neutron energy of about 1 mev. An effort is being made to reduce this neutron energy limit. An alternative approach is the selection of gamma rays followed by anti-coincidence rejection of unwanted gamma rays from the neutron distribution being studied. If properly perfected the technique will be extremely useful in many fast neutron experiments including time of flight spectroscopy and ν (neutrons per fission) determinations.

(5) Silicon Surface Barrier Charged Particle Detectors - These devices have very excellent alpha particle resolution ($< 0.5\%$). If this quality can be obtained with a large enough detector area the opportunities are great for use of the detectors for charged particle detection in fission counting and neutron spectrometer work.

(6) Fast Time Circuitry - A fully transistorized fast timing system equivalent in performance to the existing vacuum tube circuitry is under development. This unit will have the advantages of reliability and low power consumption (current vacuum tube units use up to amps of fully regulated B+ voltage).

d. General Reactor Physics Experiments

(1) Measurement of Prompt Neutron Lifetime by Pile Noise Analysis - In the August Progress Report, the equivalence, for a reflected reactor, of values for the prompt neutron lifetime determined from, (a) the uniform $1/v$ poisoning method, and (b) from the transfer function rolloff, was mentioned. It has now been determined that these values deviate when the neutron lifetimes in the core and reflector differ by a large factor. This provides an explanation for the discrepancy observed in the lifetime values as determined by these methods for the coupled critical experiments done in ZPR-III about two years ago. At that time a value of 16.5 microseconds was obtained from both the pile noise and the Rossi alpha methods, whereas a value of approximately 12 to 13 microseconds was indicated by experimental $1/v$ poisoning and also by theoretical considerations.

A method by D. P. Gamble* allows the effect of the reflector to

* D. P. Gamble, Trans. Am. Nucl. Soc. 3, No. 1, 122 (1960)

be considered mathematically equivalent to an additional delayed neutron group. This yields results consistent with those indicated by the kinetics equations derived for one energy group in a reflected system. Thus, the Gamble method has the merit of allowing kinetics calculations for reflected reactors to be done using existing bare reactor codes. Treatment of the experimental data by this method reduces the value of neutron lifetime determined by the noise method for the coupled critical experiments to 12.74 microseconds, in good agreement with the 12.7 value obtained from four group-one dimensional calculations and 12.3 from the poison experiment.

(2) Thermal Reactor Spectra - Preparations for the measurement of the equilibrium neutron spectrum at and near thermal energies in media containing various amounts of absorber and scattering material have been concentrated on the assembly and testing of the equipment to be used when the Juggernaut reactor becomes available. In regard to the problem of a sensitive neutron detector with good gamma discrimination, an assembly of counters lined with boron-10 was found to be superior to any of the scintillators used to date, but failed to provide adequate neutron discrimination in the presence of large gamma fields. On the other hand, a group of four BF_3 chambers has demonstrated adequate gamma discrimination and signal-to-noise ratio, but the detection efficiency is relatively low. It is planned to procure additional BF_3 chambers to achieve increased efficiency. The tank to contain the moderating material for the spectrum study has now progressed through a preliminary design, and several changes are now being made in that design prior to the submission of the drawings for construction.

(3) Measurement of ν of U-235 - A crucial requirement for the experimental effort to determine the number of neutrons emitted per thermal fission of U-235 is the development of detectors of very high efficiency and rapid response, in order both to reduce statistical problems and to achieve a very high ratio of true to accidental coincidences. In the current month significant progress has been made on three phases of this experiment. The fission-fast neutron coincidence circuitry has now been specified and construction has been started by the Electronics Division. Progress has been made toward achieving a significant increase in the sensitivity for beta-gamma coincidence counting of the Mn-56. This results from Cerenkov radiation which provides a means for bypassing the limitations due to the short range of the beta radiation and hence allows an increase in the size of the samples counted. Efforts are now directed toward establishing the transparency of the manganese solution to the Cerenkov radiation and selecting a photo-multiplier the sensitivity of which is maximized for the peak of the transmitted radiation. The final drawings of the manganese bath tank have been submitted to the Shops for construction.

(4) Fission Neutron Age in Water - The results of the investigation of the geometrical corrections for the Fermi age experiment are now available. Curves for an exact geometry correction to the age data are available and have been used to replace the method based on an upper and lower limit to the true correction. This results in a change of the experimental value for the age from the previous value of $27.87 \pm 0.38 \text{ cm}^2$ to the new value of $27.67 \pm 0.10 \text{ cm}^2$. A paper on the age experiments

has been submitted to the Journal of Nuclear Science & Engineering for publication, and a separate paper describing the geometrical corrections is being prepared.

2. Theoretical Reactor Physics

a. Scattering Law for Moderators - Two computer programs, necessary for the processing of the results from the Scattering Law for Moderators experiment, have been constructed. One of the programs takes the counts found in the time channels of the counters, subtracts background counts, corrects for the efficiency of the counters, and produces a normalized cross section $\sigma(E, E', \theta)$, the scattering cross section for energy transfer from E to E' at an angle θ . This program is operational and produces the input necessary for a subsequent program, the multiple scattering correction to the data.

The multiple scattering program is now in the final debugging stages. This program computes, by the Monte Carlo technique, the contribution to the cross section due to the neutron being scattered up to three times before it ends up in a channel in a counter. This program is sufficiently general to be applicable to other experiments with planar scattering sample. For a given matrix of cross section $\sigma(E, E', \theta)$ it handles up to 30 values of incident energy E, 270 values of final energy E', and 30 angular positions θ .

B. Reactor Fuels and Materials Development

1. Ceramic Fuels

a. Refractories for Reactor Use - Results on impurity levels in $\text{La}_2\text{O}_3\text{-UO}_2$ compositions were received on the compositions which were reported during July and August. These compositions are among those expected to achieve high temperature stability. Spectrochemical analysis indicated aluminum contamination of 0.5 to 2% for compositions prepared by a calcination, grinding and sintering procedure. Compositions subjected only to calcination without grinding contained less than 0.01% aluminum. The use of alumina balls for grinding of calcined powders has therefore been discontinued in order to avoid such contamination.

Several techniques for preparing sinterable lanthana-urania powders are under investigation. The calcination - grinding - sintering procedure is being modified by the use of steel balls. Acid leaching will be used to remove the iron. A one-step sintering procedure is also under investigation.

Five $\text{La}_2\text{O}_3\text{-UO}_2$ compositions containing 10 to 71 w/o UO_2 were prepared by calcining $\text{U}_3\text{O}_8\text{-La}_2\text{O}_3$ mixtures in air at 1650°C . X-ray examination indicated that cubic fluorite type solid solution occurred for compositions containing more than 30 w/o UO_2 . Investigation of powder preparation by air calcination is continuing.

b. Fabrication Studies of UC-PuC - In the attempt to provide material of good density, it has been reported that the average density obtained by vacuum sintering 80 w/o UC-20 w/o PuC was slightly under 8 gm/cc. In an effort to increase this density, UC and PuC were ground

together in the same 80-20 ratio, cold-compacted and fired under vacuum to 1600°C. The UC-PuC solid solution thus formed was ground to a powder, fabricated into right cylinders and sintered under vacuum at 1700°C. The average density of these pellets was 11.5 gm/cc. This is 83.9% of theoretical and represents a considerable improvement. Microstructure examination of this material is under way.

c. Urania-Thoria Bodies - Property and stability studies of urania-thoria solid solution compositions, as described in the August Progress Report, have been continued. Sample preparation has been completed for a wide range of solid solutions of greater than 90% theoretical density fabricated as pellets. Reoxidation tests of stoichiometric solid solutions have been completed and the results are being evaluated.

d. Uranium-Thorium Sulfide - In the continuing effort to obtain US bodies of good density, the sintering study of Batch No. 14, reported in August, showed 85.3% theoretical density, and about 1% weight loss for 1800°C firing temperature. This compares with about the same density but over 4 w/o loss for the 2000°C firing temperature reported last month.

As for pellets made of the same powder, firings at 2045°C showed a density of 90.8% of theoretical, but also exhibited a 24% weight loss.

X-ray diffraction studies made of material which condensed on the crucible cover during the 1800°C, 1900°C and 2000°C firings showed almost pure US. The lattice parameter, a , increased from 5.4736 Å at 1900°C to 5.4924 Å at 2000°C indicating an approach to stoichiometric US in this range.

Thorium sulfide has been found to sinter readily into strong, fine-grained bodies. Densities of 85.2, 98.0 and 97.6% theoretical were obtained by vacuum firing to 1900°C, 2000°C and 2100°C for one hour, respectively. Chemical analysis of the material fired to 1900°C showed 96.5% ThS_{0.976} with the remainder probably ThO₂.

2. Corrosion Studies

a. Corrosion Resistant Aluminum Alloys - Adequate corrosion resistance and mechanical strength at elevated temperatures are required in an aluminum alloy to make it acceptable for fuel cladding in a power reactor.

At this time it is believed that the most suitable alloy, from the point of view of corrosion resistance and absorption cross section, is Argonne Alloy A288, containing 1 w/o Ni, 0.5 w/o Fe, 0.1 w/o Ti 0.002 w/o (max) Si. One characteristic is an unusual dependence of corrosion rate on temperature. As the temperature is increased, the corrosion rate in static tests increases, just as for the older X8001 alloy, until at some temperature above about 300°C, it decreases. At 350°C the rate is quite low.

Recent work has probed the dependence of this behavior on the silicon content of the alloy. From 10 through 150 ppm Si, the corrosion rate is observed to decrease as the temperature is increased (test

temperatures 315°, 337°, 350°C). At each of the three temperatures, corrosion rate increases with increasing silicon content. It appears that all of this series of alloys are approaching a limiting rate at some high temperature, and that those containing the least silicon are dropping toward this value at the lowest temperatures. The importance of quite low silicon seems greater at moderate temperatures (such as 315°C) than at the highest temperatures (350° - 360°C).

Another characteristic of the corrosion of A288 aluminum seems to be superior resistance to corrosion at high water flow rates. When X8001 is corroded under typical loop operating conditions at 315°C, the amount of corrosion experienced is sharply dependent upon the number of specimens present. The best results (2 mils per year rate) have been obtained by using a large area of the alloy immediately upstream of the specimens. In a test which has run 4 months to date, the corrosion of alloy A288 has been the same as these best results, even though the number of specimens has been small. It is possible that corrosion behavior of this alloy is not sensitive to the sample surface area in the test loop.

An effort is being made to add strength to the basic alloy by adding alloying constituents. Elements tried include those previously observed not to decrease corrosion resistance. The simultaneous addition of 0.1 w/o each of Cr, Cu, Nb, Ti, Zr and V to A288 alloy increased the ultimate tensile strength at 315°C from 3300 psi (alloy X8001) to about 8000 psi. The further addition of iron phosphate to the melt, to provide some dispersion-hardening by the formation of AlPO_4 , brought the tensile strength to about 9000 psi. Preliminary results indicate normal corrosion resistance for the strengthened alloys, although testing will need to be extended.

Studies of new aluminum alloys have continued. The aqueous corrosion of the alloy containing 1 w/o Fe + 1 w/o Zr has been dependent on its mechanical and thermal history. Arc-melted buttons have been nonhomogeneous, and this might be at least part of the explanation. Another alloy showing promise of corrosion resistance during early stages of development contains 1.5 w/o Ti plus 0.5 w/o Fe.

b. Corrosion Resistant Thorium Alloys - Thorium is a potentially valuable source material for power-producing reactors. Its dimensional stability is good. Unfortunately, its corrosion resistance is poor. This program has the objective of developing alloys with good resistance to corrosion by water at elevated temperatures. The ultimate addition of some U-235 or plutonium is anticipated.

Carbon has given promise as an alloying constituent, although binary alloys have had quite a short lifetime in water at 260°C. A ternary (1 w/o C + 1 w/o Ti) has shown an encouraging corrosion rate of only 4 mg/cm²-day during 12 days in water at 260°C.

c. Corrosion Resistant Zirconium Alloys in Superheated Steam - It is logical to look to alloys of zirconium for jacketing for superheated steam fuel elements. The plan is to try to develop alloys with adequate corrosion resistance, and then to consider ways of strengthening the material. In pursuit of these objectives the corrosion resistance in steam at 540°C and

600 psi is being evaluated. Most tests have run less than two months. Alloying constituents shown to be of value include Ni, Fe, Cu, Mo, Pt and Pd.

Nickel: Corrosion rates decrease as nickel content increases from 1 to 3 w/o. The lowest corrosion rate is exhibited by a ternary containing 3 w/o Ni and 0.5 w/o Fe, although there has been some variability between arc-melted buttons. The weight gain slope is about $1.7 \text{ mg/dm}^2\text{-day}$, corresponding to about 1 mil per year penetration. The 2 w/o Ni + 0.5 w/o Fe alloy is almost as good.

Iron: Corrosion rates decrease as iron is increased from 0.5 to 2 w/o. The rate of the 2 w/o Fe alloy is slightly greater than that of the ternary mentioned above. A ternary alloy containing 0.5 w/o Fe + 0.01 w/o Pt has shown a corrosion rate of 5 mdd.

Copper: 0.5 w/o and 1 w/o Cu alloys corrode at about the same rate ($\sim 6 \text{ mdd}$). A ternary containing 0.5 w/o Cu + 0.5 w/o Fe seems better with a corrosion rate of 3 mdd.

Molybdenum and Palladium: These elements are useful in providing corrosion resistance. A 1 w/o Mo alloy corroded at a rate of 10 mdd; the rate for a 0.5 w/o Pd alloy was 12 mdd.

Systematic study of the hydrogen uptake of these alloys must be made before they can be evaluated for possible usefulness.

d. Corrosion of Miscellaneous Materials - As a structural (or cladding) material, carbon steel or a low alloy steel would be preferable to stainless steel because of lower cost and somewhat lower cross section. It has generally been felt that mild steels corrode too rapidly to be used in water where there is risk of the presence of dissolved oxygen. However, there have been indications that it is only a low concentration of oxygen which is harmful; greater amounts might provide inhibition.

Accordingly, the corrosion of electrolytic iron has been examined in water at two different oxygen concentrations (about 45 and 530 ppm) over the temperature range 100 through 260°C . A carbon steel has been studied only at the highest temperature, and it was demonstrated that its behavior was the same as that of the iron.

At the higher oxygen content surfaces acquired interference colored oxide films, and corrosion rates were quite low over the entire temperature range investigated. With the lower oxygen content the behavior was the same as that for the 530 ppm solution at 260°C only. At 200°C , a rust-colored coating formed over the thin film with a corresponding increase in corrosion rate. At 150°C , corrosion was substantially faster.

It thus appears possible to inhibit the corrosion of iron or steel to essentially negligible rates, at least for short times, by maintaining a sufficient concentration of dissolved oxygen in neutral water. At high temperatures the concentration of oxygen required is small. As the temperature decreases, the required oxygen concentration increases. These statements might provide the basis for work leading to different

and more desirable reactor operating conditions.

3. Irradiation Studies

a. Irradiation of X8001 Aluminum Alloy Clad Al-18 w/o U Alloy - Aluminum-uranium alloy fuel clad with corrosion resistant aluminum alloy is the preferred fuel element for most research reactors. It is also becoming of increasing interest for use in power reactors, as evidenced by its successful use in ALPR, a 300 psi boiling reactor. Fuel elements of this type are being irradiated in the ANL-2 high pressure water loop at MTR to determine experimentally the ultimate operating limits of aluminum alloy clad aluminum-uranium alloy. This information is necessary in order to evaluate the proposed uses of this type of fuel in future high flux research reactors and in boiling reactors operating at pressures higher than 300 psi.

The first test of this type of fuel in the ANL-2 loop was designated W-11. In this test an Al-18 w/o U alloy plate clad with X8001 alloy was operated under nucleate boiling conditions with a peak heat flux near 700,000 BTU/hr-ft² to a peak burnup of 1.31 a/o. The maximum fuel temperature was near 550°C. A faulty ion exchange column caused the coolant pH to rise to 9, thereby terminating the test because of a resulting fission break.

A second test, designated W-12, is now in progress on a duplicate plate. Heat transfer and temperature conditions are essentially identical to the W-11 test. Water quality control however, has been greatly improved over the W-11 test. An interim examination was made of the W-12 plate at the MTR site after it had achieved 0.84 a/o burnup. The plate appeared to be in good condition, although variable amounts of scale deposit were on the heat transfer surfaces. The plate will be returned to the Laboratory for removal of the scale film, after which it will be reinserted in the loop.

b. Metallurgical Evaluation of Failed BORAX IV Fuel Plates - The fuel elements for BORAX IV consisted of ThO₂-6.4 w/o UO₂ pellets bonded with lead in tube plates of X8001 aluminum alloy. Approximately two inches of free space was left above the pellets in each tube to facilitate fabrication of the final closure and to act as a reservoir for fission gas accumulation. Operation of the reactor was terminated when a series of cladding defects developed in many of the fuel elements.

Two failed fuel elements were subjected to a detailed metallurgical and hot lab examination in order to locate the defects and to determine the cause of failure.

The initial examination of the tube plates showed that the void space in all the plates had collapsed to some extent. Only one side of each tube had deformed, and this was always on the same side for any given tube plate.

Location of gross defects was done by immersing each plate in a water-filled glass tube and evacuating the space above the water. Provided fission gases or air remained in the tube, the defects could be

located by an emerging stream of bubbles. Of the seven defects located in this manner, five were in the area where the tubing had collapsed. The remaining two defects were on bottom edges, and appeared to have been caused by damage during handling and shipment.

Metallographic sections were prepared to determine the characteristics of the known defects and to investigate the soundness of the silicon-bonded end closures. It was found that the defects resulted from cracks due to overstressing of the tube walls in the collapsed areas. The silicon-bonded areas were sound, although free silicon was found at the interfaces examined.

Prior to the metallographic investigation, 39 tubes were sampled for measurements of total gas content. Fission gas release rates were also determined where possible. By the use of the bubble tests described above, large defects had been found in 7 of the 39 tubes sampled. Twenty additional tubes evidently contained small defects, as evidenced by their negligible fission gas content. The remaining tubes contained approximately the amount of gas expected from the original fabrication operation. In addition, they contained from 1.5 to 6.0 w/o of the theoretical yield of fission gas from the fuel pellets.

Based on the results of this examination, it is concluded that the defects in the BORAX IV fuel elements resulted from cracking due to collapsing of the aluminum tube walls.

4. Nondestructive Testing Techniques

a. Ultrasonic Techniques - Ultrasonic waves are useful for measuring material properties and for the nondestructive evaluation of reactor materials. The ultrasonic techniques program at the Laboratory includes Lamb wave propagation, ultrasonic attenuation, ultrasonic inspection of Zircaloy rod and ultrasonic bond testing.

(1) Lamb Waves - Lamb waves are being investigated to determine their utility for evaluating material properties.

The January, 1960 Progress Report discussed two fuel plates - half ribbed and half machined - which were scanned over a wide range of incidence angles and frequencies by the through-transmission method. Some previous data on these plates, which showed that the built-in flaws were not detected at all angles, raised the question as to whether or not this angle dependence was due to Lamb wave modes. Although it is possible to detect Lamb waves by through-transmission, it does not appear that the results on these fuel plates are due to this. A second series of runs on these plates was not consistent with the first set. The indication is that in neither case was there accurate alignment of the transducer relative to the plates for, on realignment, flaws showed up at angles at which they did not previously show up. Also, flaws showed up at the same angle regardless of the frequency; yet, for a given thickness, Lamb wave modes are dependent on frequency.

Two factors contribute to the general results on the fuel plates; beam spread and orientation of the flaws. These would be very critical

at large angles of incidence where the area of the flaw presented to the sender would be much greater. This fact, coupled with beam spread, would result in part of the beam always hitting a flaw area, and in reducing or completely cutting out the transmitted beam. At large angles of incidence, there was very little or no transmission. In 90 percent of the cases, the flaws showed up well on the machined and ribbed area only between angles of incidence of 13° to 20° . This is due to the fact that the flaws are oriented so as to make these angles best for transmission. Evidence of the orientation of the flaws is given by the lack of good indication of flaws when transmitting normally or at small angles of incidence.

A short wave receiver was used to measure the frequencies of the Lamb wave modes on stressed and unstressed aluminum plates. With this equipment it is possible to measure frequency differences of 0.001 megacycle. There was, however, no substantial difference in the frequency measurements made on the stressed and unstressed plates.

(2) Ultrasonic Attenuation - The work on measuring the velocity and attenuation of engineering materials at high temperatures is continuing. Preliminary experiments show that Eastman 910 adhesive can be used as a coupling medium between a transducer and a specimen and as a cement for attaching a thin aluminum material. Detailed shop drawings have been made for the construction of a vacuum furnace for the velocity and attenuation of the metal specimens to be measured.

Work on prototype specimens at room temperature is now being carried out to determine the proper technique for measuring velocity and attenuation. The cylindrical shape of the specimen causes the sound to be propagated in a "guided wave" mode. Different modes of vibration occur in the specimen and many of these modes are detected by the receiver. These signals tend to obscure the echo pattern and lead to a confusion between echoes reflected from the end of the specimen and those of other vibration modes. The apparent attenuation of the sample can change drastically with length owing to the interference of these different modes at the receiver. At certain frequencies, radiation scattered by the grains in the polycrystalline specimen can obscure the echo pattern. A possible way of eliminating this interference effect is to force the transmitting crystal to vibrate in only one of these modes. This may be done by proper shaping of the electrode on the crystal.

b. Neutron Techniques - This investigation is concerned with the photographic detection of neutron images with at least one long range objective being to apply neutron radiography as a useful inspection method. Because neutrons have little influence on normal photographic emulsions, it is necessary to use intermediate materials next to the film. These materials emit photographically detectable radiation when bombarded with neutrons.

The intermediate materials used thus far include a B-10 loaded ZnS phosphor (alpha emission plus light), cadmium (gamma emission), rhodium, indium, silver and gold (primarily beta emitters). These materials have been rated as to neutron photographic speed and, at least in a preliminary manner, as to image resolution qualities and to relative response to neutrons and gamma radiation. In regard to the

speed comparison, all the materials have been rated against silver screens which have arbitrarily been assigned a rating of 1.0. On this basis, the scintillator speed rating is 6.61, rhodium 1.93, indium 1.46, cadmium 1.32, and gold 0.37. These comparisons indicate that in the neutron flux used (approximately 10^5 neutrons/cm²-sec), a similar film density could be obtained by exposing silver screens and film for a time T and one of the other intensifier materials and film for a time T/R, where R is the relative speed rating for that material. On this basis then, the use of the scintillator permits the shortest exposure time.

Work is still in progress concerning resolution qualities for these intensifier materials. At this point it appears that rhodium, indium, silver and gold will all yield useful image resolution. Cadmium and the scintillator presently being used are both very poor as far as resolution is concerned. In the case of cadmium, the hard gamma radiation from the (n, γ) reaction is probably contributing appreciable image diffuseness. The hydro-genous plastic backing material on the scintillator package is undoubtedly causing scatter and thereby accounting for the poor resolution obtained with this material.

The response of these intensifier materials and film-to-gamma radiation has also been investigated in order to choose the best material for use in a mixed radiation beam. Neutron radiography is potentially useful because the relative neutron absorption of materials is different from that of gamma radiation. Since gamma radiation is almost always associated with a neutron beam, it is desirable to employ a material and film combination having a low response to gamma radiation, in order that the desired neutron image not be masked by the gamma image. Using this standard, rhodium appears very good for use in such mixed radiation beams. Silver, cadmium, and indium are grouped next, with gold and the scintillator being the least favorable. Cobalt-60 gamma radiation was used for these tests.

c. Radiographic Techniques - The analysis by gamma ray spectroscopy for U-235 enrichment in 69 derby metal sawdust samples for the Piqua reactor project has been completed. It was necessary to modify existing techniques for this analysis. Because of the geometry of the sawdust samples it was necessary to dissolve the material and then determine U-235 in the liquid sample. Further development of this technique for liquid samples is in progress.

d. Eddy Current Techniques - The eddy current program (high frequency electromagnetic waves) is concerned with two major problems: the development of new and better eddy current techniques, the application of such techniques for the inspection of reactor fuel elements and other reactor components.

Development work is continuing on the high resolution pulsed field electromagnetic test system described in the August Progress Report. Electronic circuitry is now being designed to isolate surface reflections from those due to discontinuities within the specimen.

One application of eddy current methods is the determination of the sodium level in the EBR-II fuel elements. This job is complicated

by the fact that there is a stainless steel restrainer in the upper end of the fuel element and the sodium level varies around the periphery of the fuel element. It was decided to construct a four channel eddy current system to detect the sodium level. A four channel pulsed eddy current system with differential point probes has been constructed. In constructing the four point probes it was necessary to use care in positioning each set of differential pickup coils symmetrically with respect to the driving coil. There are four identical pulse generating and detecting circuits. The detecting circuit rectifies the differential voltage. If the differential coil is precisely balanced there is no signal when the coil is put on uniform material. In making such a differential system it is not possible to attain these ideal conditions. In order to get the necessary balance conditions, adjusting resistors were inserted in the signal input circuits.

It was observed during experiments that phase differences as well as amplitude differences appeared between the two pulses induced in the differential coil. Since this phase effect causes complication in the interpretation of the detected signals, it is desirable to minimize it. For minimizing the phase effect, the circuit elements were chosen so as to rectify the initial parts (first cycle of the damped oscillating pulse) of the pulses on each repetition. In other words, rectifier build-up time was chosen to be less than half the pulse oscillation period.

Experiments have been conducted with this instrument to determine the position of the sodium level in a number of fuel elements. These fuel elements have also been radiographed. They will be stripped to determine the accuracy of the sodium level measurement.

5. Liquid Metal Slurry Studies

The Laboratory is devoting a modest effort to the study of liquid metal slurry systems. This effort has centered primarily on loop studies observing the behavior of uranium oxide in sodium potassium alloy. It has been demonstrated that a $\text{UO}_2\text{-NaK}$ slurry could be pumped and suspended at moderately high temperatures up to 600°C at moderately low flows - 1.5 ft/sec. Some problems were encountered, predominantly a little understood phenomenon which results in the precipitation of the UO_2 from the slurry when the temperature of the system exceeds approximately 450°C . Furthermore, the process is reversed once the temperature is dropped below this value. It is suspected that oxygen (as oxides) is responsible for this action since it was possible to stabilize the slurry and to pump it as high as 600°C when a small quantity of uranium powder was added to the system to scavenge the oxygen.

Another phase of the slurry effort involved an in-pile slurry loop for the irradiation of a $\text{UO}_2\text{-NaK}$ slurry. This loop was placed in the CP-5 reactor at Argonne and operated for a period of several months. During this period the operating temperature was 350°C and the burnup of the uranium amounted to 0.07% of the total atoms. The loop has since been dismantled, examined, and chemical analyses made to determine the distribution of fission products. A second loop, designed for operation at higher temperatures to a greater burnup, was later placed in CP-5 but failed after one week. Another loop, Mark III, is being fabricated and is now about two-thirds completed. The instruments for this unit have

been calibrated and a new control panel assembled. This loop is an improved version of the Mark II loop and is also designed for operation to a higher burnup at higher temperatures.

Engineering studies on the behavior of the UO_2 -NaK slurry are being carried out (under subcontract) at the Mine Safety Appliance Corporation. These studies involve the determination of pressure drops, fallout velocities, and ease of separation or addition of UO_2 to the slurry in various types of piping configurations.

An 8.33 kg charge of micronized UO_2 was reduced by hydrogen and shipped to the M.S.A. Research Corporation. This quantity of UO_2 is calculated to bring the slurry concentration up to 10 vol-percent in the engineering studies loop.

M.S.A. requested, and was granted, a time extension to November 15, 1960, to permit completion of the present phase of work.

C. Reactor Components Development

1. Development of Shielded Facilities

a. Air Filter Changing System - The current general practice in changing air filters in shielded facilities does not keep the filter within a containment enclosure. As a consequence a potential health hazard exists when highly toxic materials are involved and suitable protective clothing and respiratory equipment must be worn. A push-through system which keeps the filter within an enclosure at all times and maintains complete filtration without interruption during the filter change operation was previously described. (TID-7513, pp. 69-73). The equipment for carrying out the filter change operation while completely shielded was described at the Fifth Hot Laboratories and Equipment Conference, Pergamon Press pp. 277-292.

An improved version of this system, for use when gamma shielding is not required, has been developed and will be used in connection with the Low-Level and Alpha-Gamma Caves of the Fuels Technology Center. In this system personnel are safeguarded from the hazardous particulate material during the filter changing operation because the contaminated filter is contained at all times at a pressure below atmospheric. Complete filtration is maintained without interruption during the filter change operation. The filter is sealed both in the filter housing and in the disposal container by urethane foam gaskets mounted to the unmodified frame of the filter.

A paper, "Simplified Filter Change Without an Interrupted or Unfilter Exhaust," has been submitted for the Eighth Hot Laboratory and Equipment Conference, San Francisco, December, 1960.

b. Alpha-Gamma Containment Box - A particle-tight plastic enclosure was developed about a year and a half ago for carrying out chemical separation operations on heavy elements. (See Progress Report for May, 1959) These boxes are sealed, and their construction and handling will permit the introduction of master slave manipulators into

boots which are part of the box. Manipulators will have full freedom of motion and will be able to discard jaws and be removed from the box while a complete seal is maintained. Utilities and ventilation duct connections are made using manipulators. Transfer locks for the in-and-out passage of material will be provided. The box is designed so that containment is achieved while the box is remotely collapsed for disposal after the work is completed. The box is currently planned for use in the ANL Chemistry Hot Laboratory. A paper, "Alpha-Gamma Containment Box," has been submitted for the Eighth Hot Laboratory and Equipment Conference, San Francisco, December, 1960.

2. Development of Viewing Systems

A series of Cabal (calcium-boron-aluminum) glasses have been studied in an effort to interpret the gamma ray induced coloration in terms of structural concepts. A resolution of the observed spectra shows that three gaussian type absorption bands are induced with their maxima at about 2.3, 3.5 and 5.0 eV (550, 350 and 250 mμ). The intensity of the 2.3 eV band does not show a regular relationship with the changing structure of the glass. On the other hand, the 3.5 eV band decreases gradually with increasing mole percent of CaO and increases with increasing Al₂O₃. The 5.0 eV band shows an abrupt increase in intensity which corresponds to the appearance of nonbridging oxygens in the network. Replacing Ca⁺⁺ by Mg⁺⁺, Sr⁺⁺ or Ba⁺⁺ indicates that glasses containing large ions of low field strength (low polarizing power) show less induced coloration than glasses containing small ions of high field strength (high polarizing power). The replacement of Li⁺ by Na⁺ or K⁺ supports this view. Glasses containing potassium as the network modifier show the least induced absorption as compared to those containing Li⁺, Na⁺, Mg⁺⁺, Ca⁺⁺, Sr⁺⁺ or Ba⁺⁺.

Melting a potassium alumina borate glass under reducing conditions gives a considerably higher transmission in the U.V. part of the spectrum (before irradiation) as compared to the same glass melted under normal conditions. The gamma-induced absorption of these two glasses indicates that reducing conditions decrease the intensity of the 2.3 and 3.5 eV bands, while it causes an increase in the far U.V. induced absorption.

These studies of the gamma-induced coloration of glasses have been prepared as a paper for presentation at the fall meeting of the Glass Division of the American Ceramic Society.

Another paper summarizing the work on glass scintillators for neutron detectors (see Progress Report, August, 1960) has also been prepared for presentation at the same meeting.

3. Development of Manipulators for Handling Radioactive Materials

a. Position Transducers - The investigation of positional data systems has continued with the objective of developing a system with improved performance for use in electronically controlled master-slave manipulators. A system is desired which has the necessary signals to operate a safety circuit which will apply mechanical brakes to each manipulator motion in the event of any electrical malfunction or loss of

electrical power. Also, it is desirable to minimize the number of electrical conductors to the slave arm.

Two systems under investigation appear to have these features. One system uses synchros as phase shifters and the d.c. system uses precision potentiometers. In the first case, by applying three phase power to the stator of a synchro, the rotor output will change in phase as the position is changed. The signals from the master and slave synchros would be fed to phase discriminators.

The d.c. system using a conductive plastic potentiometer has been tested, along with a transistorized modulator that has been developed. Tests on an irradiated potentiometer (5×10^7 r exposure) have been continued in a dry box to a total of 35 million traversals with only slight changes in resistance.

b. Electro-Hydraulic Servo Valves - The design objectives of a servo valve suitable for use in a master-slave manipulator were outlined in the August Progress Report. In addition to the two stage valve (#12) reported last month, a single stage flapper valve (#13) which utilizes a venturi system to amplify the output has been operated in a simulated force-reflecting system. This valve is an improvement in some respects over the #12 valve. It has improved stability, and greater tolerance to dirty oil since it has no sliding parts. However, it has considerably greater reverse force effects which cause the system to perform as if there were excessive viscous friction. Modifications are being investigated to improve the system in this respect.

c. Electronically Controlled Master-Slave Manipulator Mark E5 - Design work on the Mark E5 configuration has been continued. A half-scale model has been built incorporating telescoping tubes for the master-slave portion of the manipulator as an alternate to the hinged arm configuration (see August Progress Report). A preliminary counterweighting system for the arm has been worked out. The supporting framework, consisting of two links which will be positioned by switch controlled motor drives, has also been modeled.

A modified version of the 300 watt transistor power amplifier reported in July has been combined with a d.c. positional servo system to give an all transistorized force-reflecting servo system. The system was stable at the gain required to give suitable stiffness at a load capacity of 50 lbs. Circuitry to achieve fixed field switching and force ratio changing will be investigated next.

4. Instrumentation and Control

a. Analog-to-Digital Techniques for Auto- and Cross-Correlation of Reactor Dynamics Data - Cross-correlation techniques can be utilized to improve experimental measurements of reactor transfer functions and stability predictions with poor signal to noise conditions. Special equipment has been ordered which converts analog information into digital form and stores the data on magnetic tape in IBM-704 computer format.

All components of the analog-to-digital data conversion and storage

system have been repaired, returned and re-installed in the system. A Packard-Bell Computer Corporation design engineer worked with the system at the Laboratory but encountered difficulties with the Ampex magnetic tape recorder.

b. Transformers for In-Pile Work - The new whiskerless anodized aluminum transformer was operated for 500 hr. at 450°C. At the end of this period it was noted that a high resistance had begun to build up between the nickel bus bars and aluminum terminals owing to oxidation at this point. These joints have now been cleaned and welded using a 95% aluminum-5% nickel alloy. The unit is now ready for operation.

5. Heat Engineering Studies

a. Boiling Liquid Metal Studies - A small scale test loop is being designed for studies of boiling liquid metals. Initial tests will be aimed at measuring vapor volume fractions and two-phase frictional losses. The design of the condenser and the holdup tanks has been completed. The condenser will be a tube-type condenser with condensation outside of the tubes. The material of construction will be Type 316 stainless steel. The fabrication of a venturi meter for flow measurement has commenced and design of the measuring section has started. Methods of measuring liquid level are being investigated. This is a major problem as the loop will have to be drained to calibrate the gamma attenuation equipment for measuring vapor volume fractions, and a meter that would be reliable after high temperature operation is needed. The possibilities of operation with other liquid metals in the same facility is also being explored.

D. Separations Development

1. Fluidization and Fluoride Volatility Separations Processes

Volatility separations are being considered for reprocessing power reactor fuels. Emphasis is now on the development of the Direct Fluorination Volatility Process applicable to the processing of oxide fuels. The fuel for the Dresden reactor (UO_2 clad in zirconium) is a typical example.

The direct fluorination of high-density, sintered UO_2 pellets in an inert bed of fluidized solids is being investigated to study problems of heat removal, reaction rate, fluorine utilization efficiency, and general behavior of the system. Decladding of Zircaloy-clad fuel by direct hydrofluorination and combinations of hydrofluorination-hydrochlorination reactions using fluidized beds of inert solids as heat-transfer media is being investigated.

Processes for recovering uranium from other enriched uranium fuel alloys using similar techniques and direct fluorination are also being studied. Reactions of stainless steel with various gaseous reagents are being studied in fluidized beds with application to decladding of stainless steel-clad and stainless steel-matrix fuels.

The fluid-bed conversion of uranium hexafluoride into uranium dioxide is being studied. The process involves either a single-step pyrohydrolysis and reduction using steam and hydrogen simultaneously, or a

two-step process using alternatively steam or hydrogen, followed by the other reactant.

Dissolution of zirconium-uranium alloys in molten fluoride salt is being studied as an initial step in a fluoride volatility process.

a. Direct Fluorination of Uranium Dioxide Fuel - A nine-inch bed of uranium dioxide reactor fuel pellets covered with a 34-inch bed of magnesium fluoride was reacted with 50 percent fluorine-nitrogen mixture. At 500°C the limiting heat transfer was reached in the pellet bed rather than on the exterior reactor cooling surfaces. No sign of plugging in either the bed or filters was indicated.

Mockup tests of heat transfer from the surface of fixed packing submerged in a fluid bed were made to determine the effects of bed height and gas velocity. The bed of 0.25 in. nickel balls and -140 + 200 mesh copper shot gave heat transfer coefficients in the range of 81 to 125 Btu/(hr) (sq ft) (°F) for gas rates for which the finer material was fluidized. The heat transfer coefficients were 5 to 8 times higher than those obtained with the fixed packing alone and 0.5 to 0.9 of those for fluidized material without packing.

Two decladding runs were made to investigate the effect of close packing of fuel elements on operation of the fluid bed reactor. One-half inch Zircaloy tubes containing uranium dioxide pellets were reacted at 500°C with 10 percent hydrogen chloride-90 percent hydrogen fluoride. Fourteen-inch lengths of these 30 mil wall tubing elements were submerged vertically in a calcium fluoride fluidized bed. Decladding was successful in both runs. For a near-maximum packing in which the cross-sectional area of the tubes was 55 percent of the reactor area, formation of a firm cake raised the pressure differential across the bed and prevented free draining of the bed at the end of the run. When the area of the tubes was reduced to 28 percent, there was no caking or pressure increase and the bed was free-flowing. No significant pellet degradation or fines production was observed in these runs.

b. Hydrochlorination-Hydrofluorination Studies on Uranium-Zirconium Alloys - Two additional runs were made in the six-inch diameter fluid-bed reactor for the direct hydrofluorination of low uranium-zirconium alloy fuels using the two-zone scheme described in the August Progress Report. In the first run a 1.0 kg charge of 1.4 percent uranium alloy (a short 10-plate assembly) was 75 percent dissolved in 4.7 hr. at 360°C using a lower zone gas stream of 90 volume percent hydrogen chloride in nitrogen. In the second run a 3.36 kg charge (15-plate assembly) was 41 percent dissolved in 4.8 hr. at 385°C using 88 volume percent hydrogen chloride in nitrogen. The results of runs to date indicate that low uranium-zirconium alloys may be dissolved at a rate to give greater than 90 percent dissolution in less than 12 hours at conditions of 360°C and a lower zone gas stream of 60 volume percent hydrogen chloride in nitrogen. Hydrogen fluoride was used to convert the volatile zirconium tetrachloride into the solid tetrafluoride in the upper zone.

Two investigations supplementary to the Zircaloy hydrogen fluoride-hydrogen chloride reaction have been made. These are corrosion

of nickel by mixtures of hydrogen fluoride and hydrogen chloride and the solubility of hydrogen chloride in liquid hydrogen fluoride. Values for the corrosion of nickel in 90 volume percent hydrogen fluoride-10 volume percent hydrogen chloride at 450°C and 500°C are 4.6×10^{-2} and 13.7×10^{-2} mg fluorine/(sq cm)(day). The solubility of hydrogen chloride in liquid hydrogen fluoride is 0.7 wt percent at -38°C and 2.3 wt percent at -78°C.

c. Processing of Stainless Steel-Clad or Matrix Fuel Elements -

The reaction of stainless steel with chlorine, proposed as a decladding step for stainless steel-clad fuels, was studied in a series of preliminary experiments conducted in a one-inch horizontal tube furnace preparatory to fluid-bed work. Under the conditions used, about 570°C, constant chlorine flow rate (2.0 g moles/hr) and variable nitrogen rate (0.6 to 7.9 g moles/hr), the penetration rate appears to become constant at about five mils/hr above 50 mole percent chlorine. An all-nickel fluid-bed system is being installed replacing the existing Monel unit since the copper in Monel was found to be severely leached by chlorine.

d. Conversion of Uranium Hexafluoride to Uranium Dioxide -

The most recent simultaneous-reaction run, steam and hydrogen at a high temperature (650°C) and a relatively high feed rate of uranium hexafluoride (218 lb. uranium/(hr)(sq ft)) showed only about 40 percent conversion to dioxide, probably because of the short residence time.

e. Fused Salt Processing of Zirconium Matrix Fuels -

Analysis of graphite dissolver performance and examination of components has continued following the completion of zirconium alloy-fused salt dissolutions. Examination of the walls of the graphite dissolver after transfer of about 500 lb. of molten salt out of the dissolver (total salt handled to date about 1500 lb.) showed the expected presence of fused salt in the middle of the lamp black insulating zone. About 30 lb. of salt was deposited in the walls. This represents the accumulated amount from all runs to date. Another 30 lb. was left on the bottom of the dissolver following the transfer.

A fused salt charge was allowed to freeze in the dissolver in order to see what effect this would have on dissolver components. This charge was then remelted in the dissolver. Extensive damage to internal graphite downlines and heaters was found. The fuel element support plate (graphite) was eroded. There was no evidence of chemical or mechanical damage to the graphite crucible.

Solids were removed from the dissolver off-gas by an absorber bed packed with 3/16-inch sodium fluoride pellets followed by grade 60 porous graphite filters. The carbon filters successfully removed all fines that passed the absorber bed. Filter fines, extending into the submicron size and totalling 311 g, were collected for 11 dissolution runs. The absorber bed collected 3750 g of additional fines.

Semi-continuous flowsheet (initial NaF/ZrF₄ ratio maintained by NaF additions) dissolutions of Zircaloy elements in fused sodium fluoride-zirconium fluoride mixtures were made at 700°C to examine the behavior of minor constituents. The concentrations of uranium, tin, iron, and

chromium in the salt phase were found to be unchanged while the concentration of nickel was reduced threefold or more in the course of a dissolution. A deposit, presumably nickel, was observed on the undissolved Zircaloy fuel element.

f. Fluorination Studies - The reaction products from the two-zone hydrochlorination-hydrofluorination reaction system were fluorinated in a fluid bed at 600°C to achieve separation of uranium. After two hours a 500 gram charge initially containing 0.06 weight percent uranium showed only 0.009 percent residual uranium. A low-temperature (400°C) fluorination was less effective.

g. Plutonium Fluoride Studies - A tenfold increase of the ratio of reactor surface to volume did not significantly increase the rate of decomposition of plutonium hexafluoride vapor. The rate of decomposition of plutonium hexafluoride vapor has been observed at several temperatures (105° to 160°C) and the activation energy for the reaction is 23 kcal/mole if the reaction is assumed to be of zero order.

The rate of radiation decomposition of plutonium hexafluoride in a mixture with uranium hexafluoride is quite similar to the previously observed rate of radiation decomposition of pure plutonium hexafluoride (about 2 percent per day).

Preparations are being made to test the recovery of plutonium hexafluoride from the exit lines of a fluorination reactor. The apparatus has been satisfactorily tested with uranium hexafluoride. Fifty grams of plutonium hexafluoride were produced.

h. Hydrofluorination-Hydrochlorination Studies on Uranium Dioxide Pellets - Experiments have been run to determine the effect of 90 percent hydrogen fluoride-10 percent hydrogen chloride gas mixtures on uranium dioxide in pellet and powder form. The rate of fluorination is affected by the method of manufacture.

2. Chemical-Metallurgical Process Studies

Investigations of liquid metal solvent systems are being directed toward providing new pyrometallurgical methods for reprocessing nuclear reactor fuels. Related physicochemical data on liquid metal systems and thermochemical data on materials of interest to high-temperature chemistry are being obtained.

a. Liquid Metal Solvent Studies - The solubilities of several elements in liquid cadmium have been determined and may be represented by the following equations:

$$\text{Titanium (425}^{\circ}\text{--650}^{\circ}\text{C): } \log (\text{at } \% \text{ Ti}) = 3.237 - \frac{5333}{T} + 1.468 \frac{10^6}{T^2}$$

$$\text{Palladium (380}^{\circ}\text{--577}^{\circ}\text{C): } \log (\text{wt } \% \text{ Pd}) = 4.4948 - \frac{3080}{T}$$

$$\text{Yttrium (325}^{\circ}\text{--602}^{\circ}\text{C): } \log (\text{wt } \% \text{ Y}) = 4.8539 - \frac{4687.5}{T} + 0.7520 \frac{10^6}{T^2}$$

$$\text{Samarium (325}^{\circ}\text{--420}^{\circ}\text{C): } \log (\text{at } \% \text{ Sm}) = 7.0729 - \frac{4878.2}{T}$$

$$\text{Samarium (420}^{\circ}\text{--600}^{\circ}\text{C): } \log (\text{at } \% \text{ Sm}) = 3.0911 - \frac{2116.1}{T}$$

$$\text{Europium (328}^{\circ}\text{--403}^{\circ}\text{C): } \log (\text{wt } \% \text{ Eu}) = 6.466 - \frac{4388}{T}$$

$$\text{Gadolinium (324}^{\circ}\text{--500}^{\circ}\text{C): } \log (\text{at } \% \text{ Gd}) = 3.217 - \frac{2379}{T}$$

The separations which can be achieved by recrystallization from liquid metal solutions are markedly affected by the extent to which impurity metals coprecipitate. A systematic study of the influence of atomic radius, valence and electronegativity on coprecipitation is under way. The coprecipitation of zirconium by the cerium-cadmium intermetallic compound, CeCd_{11} , has been measured. The coprecipitation coefficient is 0.04 ± 0.01 .

The distribution of fissile elements and fission product elements between pairs of immiscible liquid metals is being measured to investigate the possibilities for new pyrometallurgical processes. The distribution coefficient of uranium between liquid zinc and lead (wt % U in Zn phase/wt % U in Pb phase) is 270 at 647°C , 73.5 at 700°C , and 31.9 at 734°C . The mutual solubilities of aluminum and cadmium over the temperature range 682° to 722°C may be represented by the equations:

$$\log (\text{at } \% \text{ Al in Cd}) = 5.479 - \frac{2945}{T}$$

$$\log (\text{at } \% \text{ Cd in Al}) = 5.702 - \frac{2815}{T}$$

Measurements of the thermodynamic properties of intermetallic phases and liquid alloys are being done by two methods; a vapor pressure method and a galvanic cell method. Further study of the uranium-zinc system by the vapor pressure method indicates that one of the two intermetallic phases (U Zn_{11+}) has an extensive range of composition. Studies of the cerium-cadmium system by the galvanic cell method yield the following equation for the free energy of formation of the intermetallic phase CeCd_{11} :

$$F_f^{\circ} = -71,225 + 45.619T$$

Measurements of the α - β and β - α transition temperatures for uranium metal saturated with cadmium gave results which were not significantly different from those obtained for pure uranium. Appreciable solubility of cadmium in solid α or β uranium is not likely. This is in agreement with previous data obtained in galvanic cells. The eutectic temperature in the uranium-cadmium system is 0.26 degree below the melting point of pure cadmium.

b. Calorimetry - The determination of the heat of formation of zirconium tetrafluoride by combustion of zirconium in fluorine has been

completed. For the reaction $\text{Zr (c)} + 2\text{F}_2 \text{ (gas)} \longrightarrow \text{ZrF}_4 \text{ (c, } \beta \text{)}$ at 25°C , the following data have been determined: the energy of formation, $\Delta E_{\text{f}}^{\circ} = -455.71 \text{ kcal/mole}$; the standard heat of formation, $\Delta H_{\text{f}}^{\circ} = -456.89 \text{ kcal/mole}$; and the standard free energy of formation, $\Delta F_{\text{f}}^{\circ} = -432.63 \text{ kcal/mole}$.

The experimental work on the determination of the heat of formation of molybdenum hexafluoride by combustion of molybdenum in fluorine has been completed and has yielded a preliminary value for the standard heat of formation, $\Delta H_{\text{f}}^{\circ} (\text{MoF}_6 \text{ gas})$, of $-372.44 \text{ kcal/mole}$.

Calorimetric combustions of boron in fluorine are being continued, and exploratory combustions with magnesium, aluminum, zinc, and cadmium are being carried out.

The study of the heat of formation of tungsten disulfide is being continued. Combustions of tungsten and sulfur in oxygen are being carried out to obtain auxiliary data to complement the data already obtained for combustions of tungsten disulfide in oxygen.

Trial combustions of two uranium nitrides and zirconium dihydride have been made preparatory to calorimetric determinations of the heats of formation of these compounds.

E. Advanced Reactor Development

1. Argonne High Flux Reactor (AHFR)

The AHFR is to be a light water-cooled, flux trap reactor for basic research. Beryllium is used as the external reflector. At a power level of 100 Mw the expected maximum unperturbed thermal flux in the flux trap is to approach $5 \times 10^{15} \text{ n/(cm}^2\text{)(sec)}$.

a. Conceptual Design

(1) Shielding - Calculations are being made to recheck the shielding requirements around various components of the primary coolant system. It is now considered to be more desirable to design the shielding on the basis of a fuel element rupture with a subsequent release of fission products than merely to provide shielding against N^{16} . Therefore, the use of a holdup tank is no longer considered necessary.

(2) Physics Analysis - Calculations have been made on the 6.5 kg Mark I core to determine the static properties of a fresh core, its history in the course of constant power operation at 100 Mw, and considerations affecting its storage.

(3) System Operation - The environmental influence upon the operation of the cooling tower and its effect upon reactor operation is being investigated. It appears that a more efficient cooling tower and heat exchangers will permit higher power operations by reducing the temperature of the inlet primary water and keeping the primary water exit temperature at a constant value.

(4) Reactor Plant Layout - The number of beam facilities has been increased from 12 to 14. This includes six radial and six tangential holes on the horizontal core midplane and two through holes. These revisions have been made as a result of meetings held with prospective users.

As a consequence of the increased number of beam holes, the size of the holes has to be decreased to balance the amount of reflector removed by the beam tubes, and a new pressure vessel geometry is required to accommodate the beam tube nozzles and their radiation shields. The beam tubes will be elliptical in shape. The pressure vessel will have a spherical shape around the core, with top and bottom cylindrical sections to accommodate the water inlet and exit lines and control rod drives. The addition of a large amount of water moderator around the core owing to the spherical section of the pressure vessel may possibly preclude the requirement of an internal thermal shield. This possibility is now being investigated.

(5) Control Rod Drive Design - Layouts were made of control rod drive arrangements based on both eight and twelve-drive systems. A layout was also made of a suitable drive compact enough for use in the limited space under the core. However based on the variety of control conditions desired the proposed drive may be lacking in versatility. More specific drive requirements are to be developed to permit further design work.

b. Experimental Program

(1) Critical Experiments - The program of measurements planned for the cold clean form of the AHFR reference design was completed, although analysis and interpretation of the results will continue for several months. This limited program of experiments was undertaken to provide those concerned with the design of AHFR with the minimum of information necessary to permit an adequate discussion of the operating characteristics and potential hazards associated with its operation. It is anticipated that this work will be followed by a more general set of design experiments in the renovated ZPR-V cell. The facility is now being prepared for this work and instrumentation is being procured.

The differential void coefficient of reactivity for the flux trap has been determined up to approximately 80% void. The void coefficient was found to be nearly constant at about $+0.08\% \Delta \rho/\% \text{ void}$ up to a volume fraction of approximately 40%. Its magnitude then decreases but does not become negative until approximately 70% of the water has been removed from the flux trap. Thus a reactivity gain of approximately 3% is possible due to void formation in the trap. The need for a liner was apparent from the observed 0.75% increase in reactivity associated with tipping a fuel subassembly across the flux trap. An 0.25 in. aluminum liner was introduced around the periphery of the flux trap, and the resulting more sharp peaking of the flux raises the question of optimum size of the trap itself. This is one item which would be explored in any resumption of the flux trap type measurements.

Flux mapping with detectors of different neutron energy response was performed and an absolute fission rate was determined. It was found that the gold-cadmium ratio in the flux trap was approximately six times that in the fuel region. This information, when properly treated for the

non-1/v character of the detectors, will be used to determine the flux level attainable for a given operating power level.

The temperature coefficient was measured as approximately $-4 \times 10^{-4} \Delta \rho / ^\circ\text{C}$ in the range of 22° to 65°C for the clean reference design. This should be compared with a value of -1.4×10^{-4} observed previously for the core simulating the system at the end of its useful life, and also compared with a value of $-1.2 \times 10^{-5} \Delta \rho / ^\circ\text{C}$ obtained from calculations.

An occasional reactivity disturbance as large as 0.2% reactivity is observed when the 0.75 horsepower mixing device used to equalize temperatures in the reactor vessel is started. This reactivity effect is usually positive, but on at least one occasion was opposite in sign. The system recovers, usually within one minute, but sometimes unsteady operation continues through the run. This effect is very definitely associated with the mixer, although the mechanism for this reactivity change is not yet understood. It is probably related to the high degree of sensitivity of this system to small geometrical changes.

Calibration of control rods and differential worth for the top reflector have been established by both the step and ramp input methods. The results are in moderately good agreement and are probably within the errors introduced by the known limitations of these methods.

A pulsed source has been operated successfully in this system to show that adequate sensitivity exists for reactivity measurements by this technique, however, the bursts of neutrons resulted in trip signals from the period-sensitive circuits. Photoneutrons from the beryllium reflector provide some additional safety margin for the operation of a reactor of this type. Their contribution is observed to be about 16% of that due to delayed neutrons.

(2) AHFR Heat Flux and Heat Transfer Tests - Data obtained from critical heat flux tests in the small scale 135 atm (2000 psi) loop on a 0.114 cm (0.045 in.) ID by 11.4 cm (4.50 in.) long vertical Type 304 stainless steel tube are being analysed. The critical heat flux test data shows that at 150% rated power a minimum safety factor of 2.5 exists at rated flow and system pressure, assuming hydrodynamic similarity between the test section and the AHFR core design. Demonstration of such similarity is admittedly difficult, but it is believed that the hydrodynamic advantages lie with the current core design.

The above is significant because in all cases the critical heat flux occurrence was preceded by, and was caused by, flow failure and/or hydrodynamic instability detected and recorded by a differential pressure transducer.

An experimental test section for atmospheric pressure pool boiling burnout studies has been completed and testing is expected to begin in October.

(3) AHFR Hydraulic Test Loop - A theoretically adiabatic forced circulation test loop has been constructed to facilitate investigation of the hydraulic problems encountered in the fuel element designs proposed for

the AHFR. The loop is designed to operate at a pressure of 10 atm and can circulate water at 1700 ℓ /min at that pressure. With the exception of the test section and holder, the loop is constructed of stainless steel; flow studies are carried out in a horizontal plane.

The loop has been dismantled, cleaned and deburred, and reassembled with "Flexitallic" gaskets. At the same time the loop was remounted on rollers so that it could be tightly fixed to its supports while still allowing for 1 cm expansion noticed in the original shakedown runs. The pumps have been remounted on wood and rubber skids in an attempt to provide additional vibration insulation; vibration absorbers for the pump-to-piping points have not been received as yet. A venturi calibrated to 1% accuracy has been installed in the test section area and will be used as the standard in a series of runs to determine the sources of error in the flow metering system originally installed in the loop. A one-day filtering run was made after the loop was reassembled in order to remove any foreign particles still remaining in the system.

2. Basic Radiation Effects Reactor

The objective of this effort is the preparation of a conceptual design for a fast reactor which gives a maximum ratio of fast neutrons to gammas and thermal neutrons for irradiation experiments. The reactor is to be used for certain basic radiation damage studies and will permit experiments at temperatures ranging from 4°K to 300-400°C. Fast fluxes ranging from 2×10^{13} to 2×10^{11} are a requirement as is a variable neutron energy spectrum ranging from a compact distribution at the highest possible energies to a compact distribution at about 10-50 kev.

The write-up of a terminal report on the conceptual design is about 75% complete. During this month, design details on various parts of the reactor have been completed as follows:

(1) Control Rod Drives - The control rod drive design is mainly influenced by the small space available for the drive resulting from the close spacing of the control rods. The design which has been adopted utilizes two parallel lead screws driving a magnetic latch which attaches to the control rod. The lead screws are mounted between two plates which form the top and base plate of the drive. The latch is guided by struts extending between the plates which also act as the drive frame. An air dashpot at the bottom of the drive decelerates the rod at the end of the stroke. The over-all length of the drive is 96 cm (38 in.) There is a spring assist on scram which imparts acceleration to the control rod when the latch is released (de-energized). The scram time required for full control rod travel from start of motion to the dashpot is approximately 300 milliseconds.

(2) Containment - The design basis for the BRER containment shell is based on the release of one-third of the system inventory of NaK, or 450 kg (1000 lb). Complete reaction of this quantity of NaK results in air energy release of 1.1×10^6 Kcal. The resulting temperature of the air in the shell is 310°C (590°F) and the pressure is 0.95 atm (14 psig). The required shell thickness, assuming a steel working stress of 1050 kgm/sq cm (15,000 psi) is about 1 cm.

(3) Materials - The calculated nvt to which the BRER vessel will be exposed is about 4×10^{20} for 1 yr of operation at 1 Mw for neutrons with energy 1.353 Mev and greater. The limiting nvt for carbon steel usage is generally taken as 10^{18} to 10^{19} for neutrons above 1 Mev in energy; hence, stainless steel is specified for the reactor vessel.

(4) Biological Shield - Calculations to determine the biological shielding requirements are in progress. The IBM-704 Code RE-34 is being employed to obtain the thermal neutron flux distribution through Type LS high density concrete. It appears that 100 cm of concrete will be adequate.

(5) Cost - The reactor cost estimate has been reviewed and revised upward to \$2.5 million (from \$2.3 million).

3. Coupled Fast Thermal Reactor

A feasibility study is being conducted on a coupled fast-thermal reactor which can achieve breeding and generates superheated steam for electric power production. About 75% of the total power is produced in the steam-cooled fast core, its radial buffer and its axial blankets. The other 25% is produced in the pressurized, water-cooled thermal core and its blankets which surround the steam-cooled zones. A draft report on this concept has been completed and is being reviewed.

A presentation of the concept is being prepared for an AEC Nuclear Superheat Meeting which will be held October 13 and 14 at the Allis-Chalmers Greendale Laboratory.

The advantages in the concept appear to be marginal. The system is believed to be feasible to construct and could achieve an initial breeding ratio of about 1.4, but the power capability and over-all efficiency are somewhat unattractive at this time.

4. Packed Bed Fuel Elements

The use of packed bed fuel elements in a number of conceptual designs to produce ultra high neutron fluxes and high temperature effluent are under study. The use of such fluxes to produce certain transplutonium isotopes resulting from successive neutron captures is being evaluated with respect to time and fuel consumption. At this time, the effort is directed towards establishing the possible operating limits and size of the systems.

An investigation of techniques for conducting a low cost in-pile test of a 200 micron particle packed bed fuel assembly has been undertaken. Such a test would serve as a preliminary investigation of the behavior of gas-cooled packed beds at high temperatures and high power density. It is also expected that useful experience would be obtained on the problems of preparing, containing, and instrumenting such beds.

Several concepts for a test in CP-5 are being considered. All of these are based on the requirement that the coolant discharging from the bed be contained within the reactor shielding. One method would provide for a relatively long-term test. The bed would be cooled by steam which would be condensed, pumped back to inlet pressure, re-evaporated, and

recirculated through the bed. All equipment would be contained within the V-50 or V-53 experimental hole of CP-5.

Two other concepts allowing only very short-term irradiations would not recirculate the coolant and could be simpler in design and control. In one, high pressure helium from a supply external to the reactor would be the coolant. The discharge gas would be stored within the experimental hole. Operating time would be severely limited; for example, the available volume of the V-50 and V-53 hole would permit approximately two minutes of operation for a 2 kw experiment (with a helium temperature rise of 780°C ($\sim 1400^{\circ}\text{F}$) if the coolant were stored at 68 atm (1000 psig) and 870°C ($\sim 1600^{\circ}\text{F}$). A variation of this method would use steam from an external supply to cool the bed. Superheated steam discharging from the bed would be condensed and collected in a container below the experiment. Possibly two hours of operation under the conditions noted above would be permitted.

Temperature distributions in a heat-generating packed bed were studied. General equations governing the fluid flow and heat transfer were formulated for a bed composed of small particles. Since the energy transfer by thermal radiation could not be formulated in a rigorous fashion because of the complex geometrical arrangement of the particles the contribution due to thermal radiation was approximated by introducing an effective thermal conductivity. A general solution of the resulting system of non-linear partial differential equations could not be obtained analytically.

Based on the study, the following conclusions were drawn:

(1) Calculations, based on the heat transfer correlations now available, indicate that the power density of a packed bed fuel element will be limited by considerations other than heat transfer.

(2) Intraparticle heat conduction is negligible compared to heat transfer by convection.

(3) Calculations show that the performance of a packed bed fuel element is slightly improved if the particles are non-spherical.

(4) The equilibrium between the various degrees of freedom of the coolant molecule is established at the exit from the bed.

5. Lunar Power Plant

The purpose of this work is to design an ultra-compact reactor power plant which could be sent to the moon by rocket power to provide electrical energy for research on the moon. The concept is that of a fast reactor with mercury vapor coolant and radiative cooling.

The design of the power package has been revised to incorporate changes to improve flow characteristics to reduce the Hg inventory (and thus reduce weight), and, finally, to improve the assembly sequence.

A fold-out radiator design has been successfully completed and includes the necessary inlet and outlet ducting and associated solenoid valves. A method of supporting the power package prior to opening the

radiator has been designed. An alternate wrap-around type of radiator is also being considered.

Dynamics studies have been completed to the stage where the analog may be set up and operated. This includes consideration of that portion of the feedback arising from a disturbance of the thermodynamic equilibrium of boiling mercury.

One-dimensional physics calculations carried out so far indicate that the reactivity in voids in the boiling mercury is probably 4 to 5%. Since this is a fast reactor it will be necessary to insure that fluctuations due to boiling cannot allow the reactivity to exceed prompt critical. The reactivity control available from the moving reflector appears to be about 10% in k , which would be just about enough to take care of reactivity in voids, temperature coefficient burnup, and shutdown reactivity requirements.

The critical U-235 concentration of LP-1 has been found to be about 18% by volume for the core volume of 55 liters corresponding to a critical mass of 185 kg. The reactor is thus comparable to EBR-II in core size and critical mass.

6. Biogeonuclear Reactor

Present methods for botanical prospecting for ore deposits include spectrographic and colorimetric analysis of plants. These methods have been found adaptable for field use, particularly in the USSR. Because of the uptake of minerals by plants which may be representative of the blanketing soil or rock cover, plant analysis aids the prospector in his search for hidden ore deposits.

Since there exists a need to develop better and cheaper methods for plant analysis, it has been suggested that neutron activation in a portable reactor could be useful.

A short study toward the merits of such an application has been initiated.

7. Plasma Thermocouple

The construction of a new plasma thermocouple unit has been completed. Field coils for magnetic deflection are incorporated and thermocouples are provided to measure the temperatures of the collector and guard ring. Experiments with a zirconium-uranium emitter in vacuum have been performed to investigate the electron emission for a short circuit current and the effect of magnetic fields on the current. Owing to a leak in the tantalum diaphragm these investigations had to be discontinued. A new unit is under construction.

A new design for a plasma thermocouple with a cylindrical geometry has been devised. Structural problems for tungsten cylinders and welding problems will have to be solved.

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