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FOREWORD

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

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

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I. WATER COOLED REACTORS (040101)

A. General Research and Development

1. Reactor Decontamination

The reactor decontamination program has two principal objectives: 1) to determine experimentally in pilot plant equipment the degree of contamination to be expected in the steam phases of a boiling water reactor system in the event of fuel cladding failure, and 2) to find suitable methods of removing deposited activity from internal surfaces, should it be necessary because of excessive radiation levels. The pilot plant equipment used for the first objective consists of a stainless steel type 304 loop that simulates the action of a boiling water reactor. Fuel ruptures are simulated by inserting tracer activities or irradiated metallic uranium. In a parallel effort, studies are being made in the laboratory to find suitable means of removing deposited activities.

In laboratory studies, scouting experiments indicated that oxalic or dilute sulfuric acids in conjunction with hydrogen peroxide have promising decontamination capabilities on stainless steel type 304 surfaces (decontamination factor > 25). Modifications of the second step of the alkaline-permanganate-citrate decontamination procedure to reduce the corrosion on low alloy steel did not impair the ability to decontaminate stainless steel. The modifications consisted of raising the pH or adding inhibitors. Citric acid alone was not very effective as a decontaminant (decontamination factor ~ 8); peroxide additions improved the results obtained (decontamination factor ~ 30). The addition of Delchem 2128A (Pennsylvania Salt Product) as a third step in conjunction with the alkaline-permanganate-citrate procedure yielded somewhat improved decontamination over the conventional two-step procedure (decontamination factor of 150 vs 100).

B. EBWR

1. 100-Mw Modifications

a. EBWR 100-Mw Plant - Completion of the reboiler plant is still being delayed pending return of the primary reboilers which are now being repaired at the vendor's shop. To date, repair of the units has not been successful even though several different procedures have been attempted by the vendor. Difficulties encountered in the welding of the stainless steel tubes to the tube sheet have been the main problem. The repairs which have been made have resulted in more leaks than were observed originally. Cracking of tubes adjacent to the welds and faulty welds are being encountered.

It now appears that it may be necessary to remove and reclad the tube sheet if successful repairs are to be made. This would involve removal of all the tubes and preparation of the weld areas for proper welding, a procedure which would be costly and time consuming. If this method should be necessary, the completion of the reboiler plant will be delayed until at least March, 1961.

The reactor containment system (turbine plant) has undergone successful hydrostatic tests. As of the end of December the plant was ready for operation at power levels up to about 20 Mwt under approvals now in effect based on ANL-5781-Addendum.* No part of the new heat removal system, which lies outside the containment shell, nor any boric acid hold-down control, is required.

b. Boric Acid Monitoring System - The equipment for the manual monitor method for determining boric acid concentration in EBWR water has been completed and has been tested using a plutonium-beryllium source. Preliminary calibration showed that the limit of sensitivity is about 0.1 gm of boric acid per gallon of water.

Delays in the delivery of neutron sources and other components have prevented completion of the continuous monitoring equipment and the associated alarm system.

The scintillation analyzer and other radiochemistry equipment to be used for studies of the EBWR is being installed. The availability of this equipment close to the reactor will be a great advantage for studies where relatively short-lived isotopes are involved.

c. EBWR Instrumented Fuel Assemblies - The clips that locate in-core flux monitors between fuel plates of the instrumented assemblies are still presenting problems. A defective weld on one of the clips was found during the assembly process. Attempts to repair the weld were unsuccessful and a new clip is now being fabricated.

d. Void Simulators - These units were completed and have been delivered to the reactor building. Calibration of these units will be attempted in the storage pit during periods when the plant is not operating.

e. EBWR Transfer Functions - The transfer function of EBWR will be measured by inserting a sinusoidal reactivity with the control rod oscillator. An analyzer will be used to measure the amplitude ratio and phase shift between various pairs of signals, some from in-core sensing devices. The transfer function data will be used to determine reactor stability at various power levels up to 100 Mw.

Pressure transducers are to be installed within the EBWR vessel to determine the pressure vessel steam dynamics transfer function. Three pressure transducers were tested for this application. Room temperature calibrations were performed using a dead weight tester as the pressure standard. The calibrations verified the data supplied by the manufacturer. The insulation resistance between the outer metal lead conduit and any bridge connection was measured at 800 megohms during and after these calibrations.

To evaluate the zero signal shift with temperature, two of the transducers, designated No. 1 and No. 2, were immersed in ice water. The zero pressure output signal of transducer No. 2 immediately became erratic and finally the transducer became completely inoperative. Investigation revealed

*E. Wimunc, et al, "Hazards Evaluation Report Associated with the Operation of EBWR at 100-Mw," ANL-5781-Addendum (1959).

that the insulation resistance of transducer No. 2 had dropped from 800 megohms to 20,000 ohms. The insulation resistance of transducer No. 1, which was still operative, had dropped from 800 megohms to 147 megohms. These changes in insulation resistance are attributed to water entering the transducer through openings between the transducer body and the diaphragm or the transducer body and the metal lead conduit which is welded to the body.

In an attempt to evaluate the performance of these transducers in a high temperature, high pressure environment, transducers No. 1 and No. 3, with insulation resistances of 147 and 800 megohms respectively, were installed in a small pressurized system. The complete transducer bodies were enclosed by the vessel and only the metal lead conduit penetrated the system. To check the entire system for leaks, the vessel was pressurized hydrostatically to 20.4 atm (300 psig), at room temperature. Almost immediately transducer No. 3 showed the same type of erratic behavior as transducer No. 2 did in the ice bath. The insulation resistance of transducer No. 3 had dropped to 1,500 ohms and transducer No. 1 had dropped to 15 megohms. Transducer No. 1 was still operative at this time but further testing was suspended.

It now appears that leaks of varying magnitudes exist in all three transducers. The transducers will have to be returned to the manufacturer, however, to verify this finding. It should be noted that these transducers have not met the original design specifications, whatever the cause of the malfunction. Steps will be taken to obtain acceptable pressure transducers.

2. Measurement of Water Channel Gaps

Some calculations indicate that the maximum fuel plate temperatures during 100-Mw operation may be 600°C or higher, depending on the thickness and condition of the boehmite scale deposits. Thus, it is important to know something of the quantity and characteristics of the surface scale.

A description of the physical condition of fuel elements in the EBWR after approximately 0.4 a/o burnup, and information on the nature and extent of the fuel plate surface scale, are given in ANL-6091 and ANL-6136. Measurements have now been made of the thermal conductivity of the scale and of the swelling characteristics of the irradiated elements. The thermal conductivity has been measured to be 0.00186 ± 0.004 cal/cm-sec°C. The annealing studies show that at the 0.4 a/o burnup level significant volume increases occur within 45 hours at 600°C. The low thermal conductivity of the scale, coupled with the swelling data, indicate an area of possible difficulty during 100-Mw operation. This difficulty, if it should occur, would be manifest by decreasing the water channel between fuel plates.

Because of this possible problem, a device was developed to measure the water channel gaps on the EBWR fuel elements. An eddy current technique is applied in measuring the channel gap. A commercial unit, normally used for detecting flaws in materials, was modified for the present application. It utilizes a special single wound coil probe potted in a waterproof holder. With twenty feet of flexible cable connected to the holder, the probe is mounted in a spring loaded positioner which positions the probe within a channel, holding a constant probe to plate spacing on one face and measuring the variation of magnetic inductance as a function of the distance from the opposite plate face. The readings of the channel gap are recorded continuously.

The unit has been assembled and tested in a mock-up using an unirradiated fuel element submerged in water. The unit has been shown to give excellent duplication of readings within a channel. Work is now under way to calibrate the probe so that the thickness of the channel gaps can be read directly from the strip chart. Also, tests are under way to determine the response of the probe to changes in bulk water temperature. After completing the calibration runs, water channel measurements will be made on fuel elements in the EBWR which are representative of the maximum burnup and temperature conditions in the core.

The fuel elements will be checked periodically throughout the remainder of the life of the core as additional burnup and elevated temperature operation are obtained. This procedure will determine if, and to what extent, swelling is occurring and the water channel dimensions decreasing.

3. Core II

Drawings of proposed fuel elements have been completed for Core II and the long lead item requirements have been determined. The fuel in the Core II reference design is composed of 2.7% enriched UO_2 pellets clad in Zircaloy-2. The fuel elements are completely shrouded with Zircaloy-2, and are symmetrical end for end. They have a removable locating end fitting and are adaptable for handling by the existing EBWR fuel handling tools and equipment.

Calculated values of power density, velocity, and steam void volume fraction distribution show that the core has satisfactory circulation and quite uniform mass flow throughout. At least one more iteration of physics and hydrodynamics is planned in order to improve the accuracy of distributions. The best available burnout data indicate that the maximum core heat flux is less than the burnout flux by approximately 50%. Core II design data are given in Table I.

Table I. Core II Design Data

Clad diameter, OD	1.067 cm	0.420 in.
Clad thickness	0.0635 cm	0.025 in.
Pellet diameter, OD	0.927 cm	0.365 in.
Pellet-clad gap (radial) nominal	0.00635 cm	0.0025 in.
Active fuel length	121.9 cm	48 in.
Rods per element	36	
Heat flux		
average	57.03 w/cm ²	180,770 Btu/(hr)(ft ²)
maximum	149.52 w/cm ²	474,000 Btu/(hr)(ft ²)
Element flow area	55.55 sq cm	8.61 sq in.
D_e , equivalent diameter, typical channel	1.405 cm	0.553 in.
Water-to-metal ratio in core	1.7	
Fuel inventory (UO_2)	~4500 kg	

C. BORAX V

1. Construction

With a few minor exceptions, construction work was completed by the contractor in the turbine building and the building has been beneficially accepted by the Laboratory. Installation, cold springing, and welding of the large forced-convection piping in the subreactor room is complete. Installation of smaller piping in the subreactor room and the tie-in of lines to the reactor vessel shell nozzles continues. The installation of process system controls and instruments, with associated tubing and wiring, is proceeding. The 10-ton bridge crane in the reactor building has been installed but is not yet operational.

The multiconductor reactor control and instrument cables covering the half mile distance between the reactor and turbine building have been laid.

On December 31, 1960, the construction contract was 93.5% complete.

2. Procurement, Fabrication and Installation

As a result of a design change in the boiling fuel rods (see below) delivery has been delayed. Westinghouse now proposes to ship enriched rods on the following schedule: 200 on January 6, 1961, 400 on January 13, 500 on January 20, and 500 per week thereafter with completion on February 24, 1961.

The 24 sample superheater fuel plates were shipped by Atomics International on schedule and are now being evaluated.

Fabrication of the boiling fuel assembly boxes is under way at the Laboratory's Shops. With the exception of the X-8001 aluminum alloy, the stock for the thick stainless steel support plates, and the 17-4 PH stainless steel for the Belleville springs, most of the material for the three core structures and accessories is on hand, and fabrication has started at both the Laboratory's Shops and vendors' shops. Orders have been placed for the boral and X-9001 aluminum for the control rods.

Installation of reactor and process system controls and instruments in the control console and recorder panel in the control building is now about 70% complete. Specifications and drawings were completed, and bids have been invited for the in-vessel differential thermopile for measuring reactor water subcooling and for the superheater-fuel-plate thermocouples.

A tank containing a portion of a simulated core structure and reactor vessel top-head nozzles has been fabricated and installed in the turbine building basement. This tank will be used to test fuel handling tools, viewers, and procedures.

The make-up water demineralizer, installed in the reactor building by Laboratory personnel for the use of the construction contractor, has been placed in operation. A new steel grating has been installed on the main floor of the turbine building to replace a wooden deck. Work was started

on the modification of a semi-trailer as a field chemistry laboratory for the BORAX facility.

3. Design

Detailed design drawings of the flexible superheated steam exit manifolds and the Sb-Be source were completed and sent to the Laboratory's Shops for fabrication. Detailed design of the sampling panel for the water chemistry sampling station has been completed and design of the hood for this station is under way. Final revision of the superheater fuel assembly drawings for production has started. Design was completed on a new reactor-water demineralizer tank to replace the four existing demineralizer tanks in the water storage pit.

The study to determine the feasibility of replacing the 1/8-in. diameter spacers between the superheater fuel plates with 0.062-in. square wires has been completed. Results indicate that the temperature difference between the center of the fuel under the wire (the hottest spot) and the coldest spot on the spacer is only 30°F. The wires will therefore be used, since they will result in lower fabrication costs for the fuel element.

A series of RECHOP problem results were received which cover the expected range of operation with and without orifices in the boiling fuel assemblies in both natural and forced circulation. These results are now being analyzed.

The hazards analysis studies of BORAX V continued on both statics and kinetics problems. Further four-group perturbation theory problems indicate that the prompt neutron lifetime for the various cores ranges from 20 to 40 microseconds. These problems also reveal that the group structure is too coarse to yield good values of effective delayed neutron fraction. Since about 3% of the fissions occur in U-238, an effective delayed neutron fraction of 0.0067 has been assumed. With this delayed neutron fraction and these lifetimes, a series of problems have been run by the space-independent reactor kinetics code RE-129C and a special modification of this code. These problems involve the assumptions that all the heat generated in an excursion remains in the oxide and that the only shutdown mechanism is Doppler broadening of U-238 resonances. Results of these problems are being analyzed. The assumption that all the heat generated remains in the oxide is being examined with the aid of the heat transfer code RE-147.

Work involving use of the one-dimensional transport code SNG and the two-dimensional diffusion code PDG is under way to determine: (a) the flux shape across a homogenized boiling assembly and adjacent aluminum and water regions, (b) the effect on the calculated control rod worth of increasing the extrapolation distance for thermal neutron flux at the surface of the control rod, (c) the effects of the presence of control rods in upper reflector, and (d) the worth of control rods that might be stuck in an "out" position.

Evaluation of radiation hazards to surrounding population in the event of severe accident is under way. Studies of stainless steel-water

reactions using stainless steel- UO_2 cermet pellets are in progress in cooperation with TREAT and Chemical Engineering Division personnel.

Preparation and editing of the final Design and Hazards Summary Report is continuing.

Preliminary planning and scheduling has been completed on the work to be done by Laboratory personnel to place the reactor, process systems, and buildings in operation after the completion of the construction contract.

4. Development and Testing

a. Boiling Fuel - Destructive and non-destructive testing of the 12 sample boiling fuel rods has been carried out. The only serious defect was found by radiography which indicated that the UO_2 pellets were separated in a rod which had been accidentally dropped on its top. This indicated that the 3-pound design force on the Inconel-X internal spring was inadequate to maintain the pellets in position. One of the sample rods was modified to give a 7-pound spring force by means of a 1/2 in. long tubular spacer inserted between the spring and pellets. Retesting indicated that the new spring force is adequate, and the fuel rod design has been revised to incorporate this change.

Inspection of the 12 sample boiling fuel rods by the ultrasonic resonance technique showed that the wall thickness of the tubes was within specifications. The wall integrity was inspected using a shear wave technique; no defects were found.

Five of the rods were corrosion tested in water at 316°C for one week, with no evidence of corrosion. They were then removed from the autoclave and immersed in oil at 300°C . No distortion or leaking was indicated. The pellets in the tested rods were examined and found to be within specifications.

b. Superheater Fuel - Preparations are being made for pressure drop and fuel plate vibration tests on a mockup of an instrumented superheater fuel assembly. Fabrication development on a brazed four-plate fuel element of the new design using wire spacers is being continued.

The experimental program to determine whether distortions occur in the superheater element due to the temperature gradients occurring during operation was completed. Results have been reported previously of the initial, and most severe, test carried out last summer in which a mockup fuel element was heated to 550°F and then one side plate dipped into liquid nitrogen at -320°F without distortion. A two-plate mockup was fabricated and tested this month. The two plates were radiantly heated while air was passed through the coolant channel between them. Temperature gradients were limited to those corresponding to reactor operation, and only negligible distortions of the coolant channel width were observed.

c. Control Rod Drive - The final testing of drives has been essentially completed. High pitched noise from the lead screw and nut on one unit was found to be due to wear of the nut. The threads were relieved on the nut and this eliminated the noise. The extension shafts made of Armco 17-4 PH

stainless steel are to undergo ultrasonic scanning. This is being done because problems have arisen at Dresden due to the behavior of 17-4. Shipment of the drives to the Idaho site is expected to take place in early January, 1961.

d. Irradiation Monitoring of BORAX-V Pressure Vessel - The original program for monitoring the rate of irradiation damage to the BORAX-V pressure vessel steel (described in ANL-6215) has been modified. Under the modified program, 112 impact specimens, 45 tensile specimens and 36 bend test specimens are being made from the coupons of SA-212-B steel. These specimens will be divided into four sets. One set will be un-irradiated controls. Three sets will be loaded into three capsules for irradiation in the BORAX-V reactor in a location adjacent to the inside of the pressure vessel wall. Each of the capsules will be provided with a thermocouple for measurement of the internal temperature. Preparation of the specimens is almost complete.

Initially, it was planned to remove a set of specimens from the reactor at periodic intervals with a view to obtaining a history of the changes occurring in the steel. Under the current plan, the reactor will be operated for an extended period of time before the first capsule is removed.

e. Miscellaneous - Casting samples of A-132 aluminum alloy proposed for use in the nozzles on the boiling fuel assembly boxes proved unsatisfactory in corrosion test. These nozzles will be made of extruded and machined X-8001 aluminum alloy.

Testing of the high temperature, tantalum-sheathed, tungsten-tungsten, 26% rhenium alloy thermocouples was completed at the Hoskins Company. In addition to the results reported last month, it was found that the tantalum sheath was severely embrittled in the vacuum furnace at temperatures up to 4100°F. This indicates that high temperature calibration cannot be made on the actual thermocouples used in the fuel rods but must be made on sample thermocouples made from the same wire.

A turbine-type boiling-fuel-assembly-exit flowmeter and a repaired coil for the entrance flowmeter have been received from the Potter Aeronautical Company. An alternate drag-disk type flowmeter has been received from Physical Sciences Corporation. These meters will be tested in the air-water loop.

Preparations are being made to perform critical experiments using the BORAX V boiling fuel rods. It is now also planned to use the actual 5-element superheater fuel assemblies without riser and nozzle instead of attempting to mock up a superheater.

II. SODIUM COOLED REACTORS (040103)

A. General Research and Development

1. Fast Reactor Physics

a. Critical Parameters of Fast Coupled Systems - A theoretical study to determine the suitability of composite Pu-239 - U-235 fueled fast critical assemblies to predict properties of Pu-239-fueled systems is nearing completion. Briefly, it may be said that experimental knowledge of the critical mass of a composite system containing about 0.005 V_f (volume fraction) Pu-239 and 0.045 V_f of U-235 allows the prediction of the critical mass of an equivalent all plutonium-fueled system by an error of about 1/20 that which would be present if only the critical mass of a similar U-235-fueled system were known.

b. Coupled EBR-II System - A problem in the design of a coupled reactor loading for EBR-II is the method of varying the loading to reach criticality. The radius of the fast core is specified within narrow limits by the position of the control rods and also by the desire to locate the inner blanket in the two rows of hexagonal cans having high sodium velocities. These are in the locations corresponding to the first two blanket rows of the EBR-II fast core. The nature of the inner blanket is largely determined by a desire to fix the relative core and blanket power distributions, and also to fix the coupling characteristics of the system.

A possible solution to determine the critical loading can be obtained by using the EBR-II fast core fuel enrichment but permitting a choice of a lower enrichment fuel to be used in the first two to three rows at the center of the core (seven or nineteen hexagonal cans). Calculations indicate that if only two rows of hexagonal cans are varied in enrichment the enrichment of these cans must be reduced by 26%, while if three rows are varied a reduction of only 10% is needed. Otherwise a lower enrichment than that for EBR-II would be required throughout the core.

A study of the long-term behavior of the inner blanket as a breeder of plutonium as well as a burner of plutonium and U-235 has been started. The changes of the inner blanket and the outer blanket have been calculated for a burnup of 0.5% of the uranium in the inner row of cans of the inner blanket. Because of the large gradient in the burnup rate at the outer edge of the inner blanket it was necessary to divide the outer row into two segments for analysis. The changes in volume fractions are given below:

Table II. Initial and Final Volume Fractions of Fuel and Fertile Material

	<u>U-235</u> <u>initial</u>	<u>U-235</u> <u>final</u>	<u>U-238</u> <u>initial</u>	<u>U-238</u> <u>final</u>	<u>Pu-239</u> <u>final</u>
Inner Blanket Row 1 24.2-29.5 cm	0.0043	0.0032	0.5957	0.5879	0.0061

	<u>U-235</u> <u>initial</u>	<u>U-235</u> <u>final</u>	<u>U-238</u> <u>initial</u>	<u>U-238</u> <u>final</u>	<u>Pu-239</u> <u>final</u>
Inner Blanket Row 2 29.5-33.0 cm	0.0032	0.0012	0.4463	0.4409	0.0047
Inner Blanket Row 2 33.0-34.87 cm	0.0032	0.0012	0.4463	0.4403	0.0025
Outer Blanket 67.5-69.0 cm	0.0012	0.0009	0.5988	0.5979	0.00083
Outer Blanket 69.0-72.0 cm	0.0012	0.0010	0.5988	0.5983	0.00047
Outer Blanket 72.0-78.1 cm	0.0012	0.0012	0.5988	0.5987	0.0001

2. ZPR-III

a. Operations - Assembly 29, the first oxide core built in ZPR-III, demonstrated that the critical mass of oxide cores could not be calculated using the same techniques used for metallic cores. The calculated critical mass of Assembly 29 was low by 25-40%, depending on what assumptions were made in regard to heterogeneity, shape factor, and cross section set.

In order to clarify this point, Assembly 30 was planned with a 50% lower oxygen content. Other materials remained the same as for a typical fast power reactor composition, and were essentially the same as Assembly 29 except for a necessary increase in the average enrichment to keep the size of the reactor within feasible limits. Thus the core composition of Assembly 30 was 24.6% stainless steel by volume, 23.3% aluminum, 14.95% uranium enriched 39.5%, and approximately one oxygen atom per uranium atom. A blanket of about 12 in. of depleted uranium surrounded the core. This composition has a calculated critical mass of about 290 kg U-235 (S_4 , set 58, homogeneous reflected sphere). The experimental critical mass was 395 kg in a 356-liter core volume. Thus a discrepancy between calculations and experiment was again observed.

In Assembly 30 the arrangement of materials within the drawers is such that there are five columns of fuel uniformly distributed over a four-drawer cell. The other materials are distributed more or less uniformly between the fuel columns. Thus some fuel columns are adjacent to depleted uranium, others to aluminum oxide, and other to aluminum. In this respect the core is more homogeneous than Assembly 29, where the U-235, U-238, and Al_2O_3 were clustered together simulating the meat of an oxide fuel, the meat being surrounded by stainless steel and spaced by aluminum.

The large discrepancy between calculated and experimental critical masses in both Assemblies 29 and 30 has prompted experiments on heterogeneity. A wedge-shaped sector of the cylindrical core was chosen for the experiment. In this sector the 1/8-in. fuel columns were first divided into evenly-spaced,

1/16-in. fuel columns, and then were clustered in 1/4-in. columns. The first rearrangement lost 13 inhours while the second gained 34.5 inhours. In metallic fuel cores, extrapolation of this sort of bunching result from a sector of the core to the full core has revealed at least the magnitude of the bunching effect in the full core. The bunching into 1/4-in. columns from 1/8-in. columns was worth about 1% in metallic cores with volumes from 77 to 436 liters. In the oxide core of Assembly 30, the same bunching indicates a 1.5% reactivity increase when extrapolated to the full core in the same manner. Note that this correction increases, rather than decreases, the discrepancy between experimental and calculated critical mass.

The worth of various materials distributed throughout the core and at the core center was determined. In determining the distributed worths, the question was raised as to whether the measurement would be affected by the proximity of the nearest fuel column. Therefore columns of depleted uranium and of aluminum were evaluated both adjacent to and separated from the fuel. All these reactivity results are given in Table III (central) and Table IV (distributed).

Fission ratio measurements are shown in Table V.

Table III. Central Reactivity Coefficients, Assembly 30

(Worths and weights of stainless steel of canned materials have been accounted for)

<u>Material</u>	<u>Weight of Sample kg</u>	<u>Sample Worth Ih</u>	<u>Material Worth Ih/kg</u>
Enriched uranium (93.4%)	0.2878	+33.14	+115.14
Depleted U	0.2965	-0.85	-2.87
Pu-239 mixture*	0.1860	+39.35	+211.55
U-233 mixture*	0.2212	+52.55	+237.56
Depleted U	2.46	-10.48	-4.26
Aluminum	0.3506	+0.60	+1.71
Stainless Steel	1.017	-1.92	-.189
Fe	1.028	-2.06	-2.00
C	0.1964	+8.45	+43.02
Na	0.0912	+1.10	+12.04
Ph I	0.232	-3.17	-13.66
Ph II	0.140	-2.68	-19.14

* In these mixtures the named isotope predominates.

Table IV. Distributed Worths of Materials in Assembly 30

<u>Material</u>	<u>Region</u>	<u>Worth, lh/kg</u>
Al	Wedge	6.9 ± 0.2
Al ₂ O ₃	Wedge	12.5 ± 0.3
O (diff)	Wedge	18.7 ± 0.3
SS	Wedge	2.35 ± 0.1
Al (a)	Half 2	7.3 ± 0.3
Al (b)	Half 2	6.9 ± 0.3
Dep. (a)	Half 2	0.50 ± 0.05
Dep. (b)	Half 2	0.32 ± 0.05

(a) Adjacent to U-235

(b) Away from U-235

Note: The listed uncertainties are based on an uncertainty of about one inhour in the measured reactivity changes.

Table V. Central Fission Ratios, Assembly 30

	<u>Calculated (set 58)</u>	<u>Experimental (absolute counters)</u>
$\sigma_{f28}/\sigma_{f25}$	0.0428	0.0427
$\sigma_{f23}/\sigma_{f25}$	1.475	1.488
$\sigma_{f24}/\sigma_{f25}$	0.312	0.300
$\sigma_{f26}/\sigma_{f25}$	0.100	0.098
$\sigma_{f49}/\sigma_{f25}$	1.139	1.124
$\sigma_{f40}/\sigma_{f25}$	0.298	0.328

b. Analysis - In an effort to understand the reasons for the high order approximations required in the ZPR-III heterogeneity calculations*, a set of comparison studies was undertaken. The comparison study of convergence rates of the flux in thin region slab cells by various transport methods of solution has been extended by the use of a one-group program which allows a choice of discrete directions and weights in the quadrature formula. Distribution of the discrete directions so as to give better characterization of the angular region close to the tangential direction (parallel to the interface) resulted in greatly improved convergence. Reasonable a priori choice of the discrete directions may be made on the basis of a single collision distribution assuming a flat source in the region having the predominant source. Subsequent collisions function largely to increase the isotropic component, but do not affect greatly the angular spread about $\mu = 0$ in which the flux anisotropy occurs. Indications are that it may be possible to estimate heterogeneity effects by hand calculations in those cases where subsequent collisions only increase the isotropic component, and, in addition, if but few nearest neighbor cells affect the shape of the flux distribution.

*D. Meneghetti and M. F. Loomis, "Calculation of Heterogeneity Effects in ZPR-III Fast Assemblies Using the DSN Program", ANL-6218 (November, 1960).

3. ZPR-VI and ZPR-IX

a. Building Construction - The upper portion of cell No. 4 has been poured, the rails for the 5-ton crane in cells Nos. 4 and 5 have been installed, and the freight door in cell No. 6 is in place.

With the completion of the individual cells, work has been started on the portion of the building in between. The first floor slab for the vault and vault workroom has been poured. Also, the walls of the first floor vault have been poured. The forms for the second floor vault walls have been placed, and the stairway between the basement and the second floor is finished. The operating piston and guides for the hydraulic elevator have been installed. Over-all construction of the building is approximately 70% complete. Calibration of strain gauge equipment to be used to check the design and construction of the cells is under way.

b. Critical Facilities - After a review of the ZPR-VI and ZPR-IX situation, it has been decided to proceed with the construction and installation of both facilities. The supplier for the bed and table assemblies was notified that the final drawings for these devices had been approved, and was given permission to proceed with construction. It is now estimated that delivery of the first assembly can be made during February, provided performance tests are satisfactory. The second assembly should follow about one month later. These will be installed in Building 315 under the supervision of a field engineer from the supplier. Approval of the neutron source drive assemblies, which had also been withheld, was given and these devices have now been received.

The contract for the ZPR-VI and ZPR-IX consoles, nuclear instrumentation, and installation has been awarded. Meetings have been held with the contractor's representatives to discuss the details of the instrumentation and interlock circuitry. It was learned that installation of the first set of instrumentation and its console would not be completed until mid-October. This is a direct consequence of the three-month delay in awarding the contract resulting from a need to decide whether one or two critical assembly systems were to be constructed. This item represents the major delay anticipated in startup of the ZPR-VI facility.

Procurement of other components for these facilities is proceeding with no major difficulties except in regard to the stainless steel matrix tubes for the ZPR-VI system. The supplier continues to have trouble in fabricating tubes to meet the specifications, and the latest difficulty is a tendency for bell-mouthing in the tubing. This difficulty is expected to be overcome and the first shipment of tubes is scheduled to arrive in January, 1961.

c. Control Rod Drive - By chrome plating to a thickness of 0.075 in. in both the magnet armature and the face of the magnet shell on the fuel drawer expulsion drive, the magnet release time has been reduced to 15 milliseconds. The total time to travel 15 cm. is now 105 milliseconds. High-speed motion pictures have been taken of the drive to provide a time-displacement plot. A two-piece floating armature has been developed which allows a reduction in the latching magnet current from 0.9 amp to 0.75 amp. This armature is also being chrome plated.

The blade insertion drive failed to survive its first scram test. The impact upon stopping in the dashpot was sufficient to loosen the mounting structure, thereby causing severe misalignment. Steps are being taken to make the drive framework more rigid.

d. Experimental Equipment - Tests of the Perlow counter and associated circuitry for possible use in spectra measurements on the fast and coupled critical assemblies continue, with a large amount of reworking of the electronics required to achieve satisfactory performance. In the several years which have passed since this device was first used for a reactor spectra measurement on ZPR-III, improved methods for neutron detection and energy determination have been developed. There is now under way a study of the feasibility of two alternate techniques for fast neutron spectrometry in order to determine whether these offer more promise than the procedure used previously.

The first involves scattering of a collimated beam from the gas in a recoil counter, and observing the delayed coincidence recoil between the pulse and the detection of the scattered neutron. A knowledge of the angle of scattering and the time-of-flight of the scattered neutron provides a means to establish the energy of the incident neutron. The second technique involves a neutron detector sensitive to a narrow energy band which is used to survey the angular distribution of neutrons scattered from an hydrogenous target. Both of these procedures require collimated beams and do not, therefore, eliminate one of the major problems encountered previously in the use of the Perlow spectrometer. These devices, however, should provide better energy resolution below 100 kev than the Perlow spectrometer. For dilute fast systems, the spectrum in the energy range from a few kev up to a few tenths Mev is of particular interest.

e. Hazards Evaluation - For the purpose of discussion of the meltdown behavior of ZPR-VI two severe accidents have been calculated. In one case a loading representative of a power reactor of 1200 liters core size was considered to melt and collapse under a sequence of pessimistic assumptions. In the other case a non-typical core which was particularly dangerous because of a large void fraction was considered. Both of these cases were evaluated by means of the AX-I neutronics-hydrodynamics code. While the behavior of the two systems was quite different during the course of the burst, the integrated energy release turned out to be nearly the same, about 2.5×10^{20} fissions.

4. 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 - Phase 3 of the construction of the reactor (see Progress Report, September, 1960, ANL-6234, page 12) is progressing. The 5 ft. by 5 ft. shield box (without concrete) has been placed in position and centered with the reactor vessel. The shim-safety rod sheaths were then positioned to line up with the slit in the graphite reflector. The seven control rods run through the shield into the graphite close to the core. Proper alignment is essential. After removal of the 5 ft. by 5 ft. top

shield, it may be replaced to its original position guaranteeing realignment of the shim-safety rod access holes. Other access holes for the source, etc., have been placed and the shield is ready for the concrete fill. The top shield which covers the control rod mechanisms is being readied for concrete fill.

Cutting and fitting of the floor plates covering the holes on the top of the reactor and fitting the various shield plugs was carried out. The plumbing for the reactor is about 90% complete.

b. Instrumentation - The instrumentation used on the Geneva ARGONAUT reactor is being redesigned for use on Juggernaut. The shim rod indicators were calibrated with a rod at an inclination of 5 degrees to the vertical and the limit switches were adjusted accordingly. The wiring of the spare panel which holds the throttle valve indicators, tower fan switches, and the flowmeter is now completed and final drawings are in preparation.

Rod withdrawal and drop times were tested with the control console. The same shim rod was used each time. Drop time as measured on an oscilloscope was 0.415 sec; withdrawal time, 3 min.

Indicator lights for presence of overflow and absence of overflow in the reactor coolant water return line will be incorporated into the control console.

c. Startup Preparations - The first draft of an operating manual has been written.

A procedure for calibration of the Juggernaut control system has been formulated. Both rod-drop and period techniques are to be used. An experiment to determine the effective delayed neutron fraction for the Juggernaut by a poison substitution method has also been planned.

5. Fast Reactor Fuels - General

a. Fast Reactor Fuel Jacket Development - The use of stainless steel as a jacketing material for future fast reactors operating at elevated temperatures has decided limitations. These include low yield and stress-rupture properties as well as the possibility of forming a low melting eutectic with uranium or plutonium fuels. The desired higher operating temperatures and increasing fuel burnup rates require the development of new high strength alloys to cope with the situation. Additional requirements of these alloys are compatibility with sodium, compatibility with plutonium-containing alloys, good fabricability and weldability, and if possible, good thermal conductivity. These requirements can best be met with certain refractory metal base alloys. Most development work to date has been limited to available vanadium, niobium and tantalum base alloys.

Attempts are being made to secure refractory alloys for fabrication development work, and samples of good quality high strength alloy tubing for testing prototype fuel elements and for fuel meltdown tests in the TREAT reactor. Niobium base alloys are being developed for the Laboratory by Battelle Memorial Institute and vanadium base alloys have been developed

by Armour Research Foundation for fabrication development work at the Laboratory. Commercial suppliers have been contacted to supply tubing of the above high strength alloys. The fabrication methods being used for these test specimens will not be used in production. The specimens are being used for material tests only, and sound material is essential so that testing will not be compromised by defects.

The V-5 w/o Ti-20 w/o Nb alloy tube mentioned in the September Progress Report (ANL-6234, p 13) was further cold drawn to 0.185 in. O.D. x 0.150 in. I.D. x 3 in. long. Further drawing was discontinued as the surface was poor and laminations were evident.

An as-received 0.430 in. O.D. x 0.371 in. I.D. Ta-0.1 w/o W alloy thimble was annealed for 1.5 hours at 1250°C in vacuo. The hardness of the thimble decreased from R_p 81-85 to R_p 33-35. The closed end of the thimble was then swaged and drilled along its axis. Following this, the tube was coated with a lubricant, (Krylon copper paint), prior to cold sinking and drawing over a 0.321 in. diameter mandrel. After reducing the tube 44%, the mandrel was removed and the tube was further sunk to 0.299 in. O.D. x 0.259 in. I.D. The tube was annealed as before in vacuo and the hardness decreased from R_p 91 to R_p 30-34.

The tube was lubricated again, sunk and drawn over 0.225 in. diameter and 0.205 in. diameter mandrels. Following the draw through the 0.231 in. diameter die, a major portion of the tube was collapsed due to the malfunctioning of the draw bench when attempting to remove the 0.205 in. diameter mandrel. The hardness of a small length of the tube which was salvaged was about R_p 28. This section of tubing was further cold worked to size (0.176 in. O.D. x 0.156 in. I.D.) The three inch long tube was leak checked with a helium mass spectrometer and no defects were found.

A 0.600 in. diameter bar of the above alloy is currently being fabricated into rod for end plug material. All work is being done cold and intermediate anneals and hardness data are basically identical to those mentioned above for the tube.

Two strips of Nb-1 w/o Zr alloy sheet measuring 0.020 in. x 5/8 in. x 3.5 in. were prepared for resistance weld studies. Eight different welds were made by changing the variables - time, pressure and current - one variable at a time.

Prior to welding, a 0.0003 in. thick foil of Zr was sandwiched between the two Nb alloy sheets. The foil was used with the hope that the Zr foil at the interface would diffuse into the Nb alloy. The welds were examined metallographically and only one of the welds showed complete diffusion of the foil and a recrystallized area about 0.020 in. wide.

B. EBR-I

1. Operation

The majority of reactor operating time this month was devoted to training of personnel.

Toward the end of the month a series of power coefficient measurements was begun to investigate more fully some possible irregularities noted at low powers. Analysis of these data is proceeding, although the runs are not complete.

2. Mark IV Development and Fabrication

The Mark IV plutonium loading will consist of 555 assemblies. This number includes 420 plutonium fuel assemblies, 120 blanket assemblies, 10 fuel thermocouple assemblies and 5 blanket thermocouple assemblies. A 30% excess will be fabricated to provide material for evaluation, replacement for process rejects and replacement for assemblies which may not pass final nondestructive tests.

Recent test casts produced an 88% yield of useable fuel slugs. Both graphite and Y_2O_3 mold washes were used with fair success. Thoria has not been used extensively in the Fuel Fabrication Facility because it cannot be coated on the molds in the open without contamination of the area. Injection casting tests were discontinued because (a) it was felt that an acceptable process has been developed and (b) delays in the delivery of the EBR-I plutonium forced the conservation of the Laboratory's basic plutonium allotment.

Sufficient sample hardware was obtained to make 4 fuel rods. Two of these were assembled, NaK filled, welded and submitted for nondestructive testing. The other two were retained for loading after the result of the bond and level tests were obtained. The experimental loading pouch design was tested and approved and an order was placed for the required number.

Evaluation of results of the eddy current inspection of samples of the Zircaloy-2 jacket tube is continuing. Development of techniques for fabrication of thermocouple tubing is continuing.

A punch and die has been fabricated that raises four circumferential spacers at each end of beta treated depleted uranium slugs. Blanket slugs upset in this die set are being evaluated.

Samples have been received representing initial attempts to drill deep thermocouple holes in depleted uranium employing electrical discharge machining methods.

The Laboratory was notified December 20, 1960, that the jacket tubing vendor did not have a process for making the 0.257 in. tubing ordered. Every effort possible is being made to deliver good tubing but all previously quoted delivery dates must be disregarded.

As of December 22, neither plutonium nor Zircaloy tubing had been received for fabrication of the fuel.

C. EBR-II

1. Construction

The following tabulation presents the status of construction contracts as of December 16, 1960:

<u>Building</u>	<u>% Completion</u>	<u>Estimated Completion Date</u>
Power Plant (Package 2)	100	--
Reactor Plant (Package 2)	100	--
Sodium Boiler Plant (Package 3)	70	--
Fuel Cycle Facility (Package 3)	70	May, 1961

The Power Plant and the Reactor Plant had been previously accepted from the construction contractor on an Interim Facility Transfer. Progress has been made by the contractor on completing "punch list" work items, but some items yet remain to be finished.

The completion of the Sodium Boiler Plant has been delayed by the problem of field fitting of piping which interconnects the steam drum evaporators and the superheaters. The eight evaporators were delivered in the period from June through August, 1960, but the contractor has experienced difficulty with fitting the interconnecting piping properly. The superheaters have not been delivered but this problem has been resolved and is discussed under Component Development. The Commission has directed the contractor to proceed with fitting the piping properly and the work appears to be proceeding in an orderly fashion.

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

2. Installation of Equipment - Package 4

a. Reactor Plant - Installation work within the Reactor Plant falls into two general categories: (a) mechanical, comprising approximately 70% of the total; and (b) electrical, approximately 30%. For convenience in scheduling and supervising the work in the field, the mechanical work is divided into 43 major jobs, or packets, and the electrical into 5 packets. Work on 32 of the mechanical packets and all 5 of the electrical has been started. To date, approximately 40% of the total mechanical work and 10% of the total electrical work has been completed.

Some of the completed mechanical work is briefly described below. Since most of the components involved are located inside the primary tank, Figure 1 is presented to aid in the discussion.

Both the large rotating plug and the small rotating plug have earlier been field assembled, filled with shielding material (concrete and/or steel balls), and temporarily located in position within the primary tank support structure central ring to check alignment, clearances, and performance of associated drive equipment. Figure 2 shows the large plug prior to this temporary positioning. Figure 3 shows both large and small rotating plug during this positioning. These plugs have since been removed from the primary support structure to provide necessary access to the interior of the primary tank for subsequent installation work, and to enable preliminary mounting on the small plug of the numerous drive mechanisms associated with it.

FIG. 1

MAJOR REACTOR COMPONENTS
AND REACTOR PRIMARY TANK

Fig. 2

View of the Large Rotating Plug for EBR-II.

Fig. 3

Theodolite Set-up Being Used in Lining Up the EBR-II Reactor Grid Template.

Fig. 4

Photo Looking Up the Temporary Access Ladder Located Inside the Heat Exchanger Well Casing in the Primary Tank of the Reactor Building.

Fig. 5

Primary Sodium Pump Assembly in the Reactor Building
Being Moved to the Primary Tank for Installation.

Fig. 6

Primary Tank on the Operating Floor of the Reactor Building,
Showing the Reactor Vessel and Grid in Place. Photo Taken
from the Building Crane Rail Looking Down.

Fig. 7

Primary Tank in the Reactor Building, Showing Both Rows of Neutron Shield Cans in Place in the Reactor Vessel. Photo Taken Looking Down.

Fig. 8

Rod Drive Mechanism

The heat exchanger well-casing with its borated stainless steel shielding attached has been permanently welded to the bottom of the primary tank cover as shown in Figure 4. The inlet nozzle of the casing, to which the reactor outlet pipe will later be attached, may be seen on the left side of the diffuser ring at the top of the casing. All personnel access to the primary tank is by means of a ladder extending through the well casing, as shown in Figure 4.

Both primary system pumps have been field assembled and mounted on the primary tank. Figure 5 shows one of the pumps after completion of assembly except for the drive motor. The ball-and-socket disconnect may be seen at the bottom end of the unit. The integral neutron-gamma shielding plug appears at the top. In Figure 6, the drive motors for the two pumps are apparent after installation of the pumps in the primary tank.

The reactor grid assembly and the reactor vessel have been permanently mounted in the primary tank. All of the vessel "internals" have been incorporated, including thermal baffles, internal shielding can supports, shielding cans, flow baffles, and temperature and pressure sensing devices. In Figure 7 may be seen the top surface of the grid assembly and the shielding cans. All of the primary system inlet piping from the pumps to the reactor, including the throttle valves for the outer blanket flow and the electromagnetic and orifice flowmeters, has been installed. The inlet piping from one of the pumps is also shown in Figure 7. Also, note in this figure the extensive use of pliofilm over the entire bottom of the primary tank, the grid, and other areas to minimize contamination of previously cleaned surfaces.

The safety rod drive mechanism installation has been completed. One of the two drive shafts extending into the primary tank is visible in Figure 7; a portion of the drive mechanism mounted on the primary tank support structure is visible (under the man's arms) in Figure 6.

The control rod drive lifting platform, all twelve control rod drive mechanisms, and the reactor vessel cover lifting mechanism have been field assembled and temporarily mounted on the small rotating plug. These mountings were accomplished subsequent to removal of the plug from the primary support structure (after temporary positioning), the purpose being to check out the integrated mounting so as to avoid any possible fitting problems during final installation. The lifting platform and control rod drive mechanisms, after completion of their temporary mounting, are shown in Figure 8. These mechanisms will be removed before the plug is finally installed, and then remounted.

In addition to the above, installation of the auxiliary pump magnet, the radial neutron shield inner liner, a substantial amount of instrumentation located in the primary tank, and the equipment air lock dolly has been completed.

In the preceding report (November, 1960), a detailed program of inspection and cleaning of the reactor vessel grid assembly carried out in the field was described. It was also indicated that a special - purpose periscope had been developed which would permit additional inspection of the assembly from its central region outward toward the periphery, but that use of this device was being temporarily postponed in order to preclude

interference with other scheduled installation work. This inspection work has now been completed, with no further contamination found. It is now confidently believed that an appropriate state of cleanliness of this high-precision assembly has been achieved.

3. Engineering

The engineering work on Package 4, or the installation, is nearing completion.

Work pertinent to the EBR-II Hazards Summary Report Addendum is continuing. Efforts have concerned themselves with defining the design parameters and conditions which have been changed from those described in ANL-5719 (EBR-II Hazards Summary Report, May, 1957). In addition, considerable amounts of information have been developed from ZPR-III, EBR-I, TREAT, and irradiation studies. This information is being reviewed particularly for implications bearing on reactor safety.

Results from ZPR-III suggest that in all cases rates of reactivity insertion by control rods, safety rod, and fuel handling mechanisms are appreciably smaller than those postulated in ANL-5719. ZPR-III experiments also suggest that the major features of the isothermal temperature coefficient are as given in the same report.

4. Procurement

With the resolution of the superheater fabrication and delivery (see below), the remaining major item of equipment to be delivered is the sodium-to-sodium heat exchanger. The Commission and the Laboratory are both expediting delivery but, because of fabrication difficulties in the vendor's shop, definite delivery dates have not been established. It will not be necessary to install the sodium-to-sodium heat exchanger prior to conducting the dry critical experiments. However, considerable time, effort and expense can be saved if this equipment is available while the contractor still has his complete crew available. In addition, considerable time will be saved if it is not necessary to interrupt the experimental program with additional major installations in the Reactor Plant.

Equipment for the Fuel Handling Center which is being furnished by the Datex Corporation is proceeding on schedule. Drawings are being submitted for approval and it appears that delivery will be as scheduled. The balance of the equipment for Package 4 installation has been shipped or satisfactory delivery dates have been established.

5. Component Development - Instrumentation

a. Automatic Flux Control - The final control unit for the automatic flux control system is being fabricated. The power demand potentiometer positioner has been completed, all front panels have been painted and lettered, and wiring of the amplifier chassis is being completed.

b. Control Rod Reactivity Generator - A sinusoidal reactivity generator and wave analyzer are being designed to measure the transfer of EBR-II. Further investigation has been done on the nulling circuit of the transfer

function measurement unit. The circuit components and mechanical parts have been calibrated and some were replaced in order to reduce the noise induced by the component itself. The experiments were performed in absence of noise for various values of $\Delta n/n_0$, different working signal level. Values of $\Delta n/n_0$ in the range of 0.01 to 0.2 were used and the accuracy was found to be within $\pm 0.5\%$ in amplitude and ± 0.05 degree in phase.

The capability of rejecting second harmonics was measured by injecting various amounts of second harmonic with the signal. The ratio of second to first harmonic in the range of 0.02 to 1 was investigated. The results showed that the accuracy of the measurements in amplitude and phase vary as a function of the phase difference between the second harmonic and the signal. The accuracy of the method was measured within $\pm 5\%$, $\pm 1^\circ$, $\pm 2.7\%$, $\pm 0.3^\circ$; and $\pm 1\%$, $\pm 0.1^\circ$ for ratios of second to first harmonic of 1, 0.5 and 0.22, respectively. A more general noise analysis will be made by combining random noise with the signal.

Initial cost estimates of the internal reactor components are being made. Preliminary reactivity generator drive arrangement layouts have been completed. Assembly drawings and details have been started.

c. Start-Up Instrumentation - Special instrumentation is being provided for EBR-II start-up. An audio pulse output unit was designed and fabricated. The timer control circuit was incorporated into the same chassis. The mechanical drawings of the instrument thimbles have been sent to the shop for initial cost estimates. Drawings of the source are being revised and rechecked to satisfy handling requirements.

6. Component Development - Reactor

a. Gripper and Holddown Mechanism - The function of the gripper and holddown mechanism has been previously described. In the preceding Progress Report (ANL-6269, November, 1960), the setup at the Laboratory for testing this unit by use of a temporary power and control console was discussed, and it was stated that at that time only certain subassemblies of the mechanism had undergone testing and that some difficulty with the holddown drive motor had been encountered. During the present period, the drive motor trouble has been eliminated and the entire unit tested completely satisfactorily. Dismantling prior to shipment is now under way.

b. Storage Rack Mechanism - The function of the storage rack mechanism is to provide a means for temporary hold-up of new and spent fuel (and blanket) subassemblies. The rack is suspended from the primary tank cover, and is fully submerged in the tank sodium at all times to permit natural convection cooling of spent subassemblies within it. The rack is cylindrical, with a capacity of 75 subassemblies. The associated drive mechanism provides both vertical and rotational movement of the rack to permit either insertion or removal of a subassembly from the rack by the transfer arm. Because this unit is relatively complex and requires careful alignment and adjustment, it is being assembled for testing at the Laboratory prior to shipment to the construction site. Assembly is about half finished and testing is expected to be completed in time to assure delivery commensurate with the scheduled installation.

7. Component Development - Steam Generators

Despite the intensive development work that has been carried out, problems in reliably welding the bonded duplex tubes to the sodium tube sheets without the introduction of real or potential defects still persist. The welding development has included a complete re-examination of all the welding parameters including welding current wire speed, voltage, welding speed, bead size, and preheat temperature. Modifications have also been made to the welding equipment.

Among these is a nozzle which has been fitted to the welding gun. The nozzle enables a spray of water to be directed within the inner tube of the duplex directly beneath the point where the weld metal is being deposited. This, plus the substitution of dip transfer for spray transfer as the mode of metal deposition, has practically eliminated weld penetration into the inner tube. Despite these changes however, occasional penetration into the nickel phosphorus bond line still occurs.

Water chilling of the tube has exaggerated localized non-wetting of the tube by the weld metal, particularly at the start of the weld, a condition which could allow sodium to leak into the air space between the tube sheets during service. In summary, progress has been made toward the solution of the subject welding problem. It is felt, however, that the welding procedure requires additional development before it is sufficiently insensitive to minor condition changes to permit its use on the three units now ready for tube to sodium tube sheet welding.

Added stringency relative to the reliability of the welding procedure has been placed upon it by the absence of a reliable nondestructive testing technique. Attempts to develop such technique are being vigorously pressed, but as yet, have not been successful. When the tubes are welded into the tube sheet penetration sometimes occurs into the nickel bonding layer. An investigation is under way to find a means to determine the depth of weld penetration. The dual pulse eddy current unit using a special I.D. probe is being evaluated for this application. Because Croloy is a magnetic material there is a large effect due to magnetic permeability. It appears that the eddy current unit is determining differences in magnetic permeability rather than penetration.

In the event that all attempts to solve the duplex tube to sodium tube sheet welding problem fail, two alternatives are being considered. The first involves substituting single walled tubing for the bonded duplex tubes. This will be dimensionally equivalent to the duplex material. Invitations to bid on tubing have been mailed and two of the vendors have submitted proposals. Unfortunately, the delivery promises are very disappointing; e.g., ten weeks or longer (instead of the three to four weeks which were originally quoted on an unofficial basis.) Thus, the principal argument for the use of the single walled tube, namely, fast delivery, has not been substantiated.

The other alternative is the use of mechanically bonded duplex tubing dimensionally identical to existing metallurgical bonded tubing except for the absence of the metallurgical bond. Second priority consideration has been given to this alternative because (1) it was expected

that its delivery would be slow, and (2) until the adoption of dip-transfer welding, serious faults were developed in the tube during welding. Developments which have taken place during the last month indicate that welds which do not fault the tubes can reliably be made, and that the delivery of either type tube used as an alternative will be comparable.

In order that the completion of the superheaters does not delay the construction of the Sodium Boiler Plant (Package 3 contract), their installation has been deleted from the construction contract. These units will be installed on Package 4.

8. Component Development - Fuel Reprocessing Facilities

Construction of the Fuel Cycle Facility is about 70% complete. Closure of the building has been completed and inside work will continue during the winter.

Design, testing, and fabrication of equipment for the Air and Argon Cells of the Fuel Cycle Facility is being continued. Installation drawings and instructions for the melt refining furnace auxiliaries are being prepared. These cover the required piping, instrumentation and wiring diagrams for the system. Studies of designs of window shutters and their drives continue. The shutters will cover the inside of the shielding windows when they are not used for viewing and will reduce the gamma dose on the window glass. The fabrication of the top works of the electrical and thermocouple plug was completed. The function of the plug is to form connections through the shielding floor of the Argon Cell. Tests of the unit in the cell mockup revealed the need for some design changes to facilitate remote operations. Work on other items such as the transfer locks, degassing furnace, and wall periscope is progressing.

Eight operating manipulators are being built for the Argon Cell. The bridges of these manipulators are now being assembled. The control equipment for these manipulators and for the five-ton cranes has been shipped to Idaho.

Irradiation tests of mercury vapor lamps having a hard borosilicate glass outer envelope have been completed. These lamps are much more susceptible to gamma-ray darkening than the soft lime glass lamps tested previously. At 1×10^8 rad gamma dose, the soft glass lamps maintained a light output of 97 percent of original, whereas hard glass lamps maintained light outputs of between 30 and 65 percent of original.

On permanent installations of electrical leads in the Argon Cell, MI (mineral insulated) cable with hermetic ceramic end seals will be used. An economical substitute for this seal has been sought for use when replacement is comparatively simple. Such a seal (using Temporell No. 741 as the potting compound) has been successfully tested after irradiation to 6×10^9 rad. Leads with greater flexibility than MI cable are also necessary for some applications. Leads with several types of asbestos insulation have been irradiated and tested. One type has proven satisfactory after radiation doses to 1.4×10^{10} rad.

Previous tests have shown that organic grease lubricants are neces-

sary for use with roller type bearings used in the process cells despite the inherent susceptibility of these greases to radiation damage. Samples of three radiation resistant greases (NRRG-159, NRRG-235, and NRRG-335; manufactured by the Standard Oil Company of California) have been irradiated and then tested in a typical roller bearing. On the basis of these tests all the greases look promising. However, the possibility exists that excessive softening of the grease might occur at radiation levels less than those used for the bearing tests, and this possibility is being investigated.

Seven more ingots have been made in the melt refining furnace mockup from enriched fuel pin scrap. The scrap is generated in the injection casting step for fabrication of EBR-II fuel pins. The scrap is melted and recast into 10-kilogram ingots for recharging to the injection casting furnace.

The component devices which comprise a complete machine for decanning the individual fuel elements are in various stages of final design. One of these devices, the Fuel Element Magazine which is used to transfer elements into the argon cell from the air cell and to feed the elements into the decanning machine, has been redesigned and a test model built and operated successfully. The new design can accommodate a bowed element that will pass through a 1.5 in. I.D. cylinder and it has a more positive mechanism for releasing the elements. The control circuits for the entire spiral decanning machine have been designed conceptually and most control components and drive motors have been selected. A time delay, variable from 20 to 400 milliseconds, has been designed to enable the operator to adjust the position at which the decanning tool is withdrawn to allow passage of the fuel restrainer. The individual controls have been grouped into three sequences. The operator initiates each sequence after visual observation through the window assures that the machine has functioned properly. The operation, of course, can be stopped at any instant that malfunction is observed.

Design of the EBR-II Fuel Cycle Facility refabrication equipment is estimated to be 55 percent complete. Specifications have been written and detailed drawings completed on the gas pressure injection casting machine and process cell feed through ducts. Assembly drawings were completed on the fuel pin processing and inspection machine and details were started. A bid by the Minneapolis-Honeywell Company was accepted to design and build the data processing equipment to go with this machine. The function of the data processor will be to keep a record of each pin as it is processed. It will accumulate, integrate and print a data record showing length, diameter, volume, density and internal porosity of each pin in the process flow.

Voids in the sodium bond continue to be a problem, resulting in repeated recycle of many fuel elements through the bonding operation. It seemed quite evident that many of the bond voids were the result of shrinkage on solidification. Hot pin eddy current studies on elements with a liquid sodium bond were completed which showed that a majority of voids do indeed appear as the sodium solidifies and that these tend to be eliminated when the element is heated to the liquid sodium temperature. As a by-product of these tests, it was also found that liquid sodium presents a level which is more easily read by the eddy current technique than the irregular surface of the solidified sodium.

The studies are continuing in order to determine whether the Fuel Cycle Facility bond tester should be modified to determine bond with liquid sodium. Tests were also conducted which indicated that a low frequency jolting of the pins was more effective in void elimination than 30 or 60 cycle Syntron vibration. The Syntron even appears to create voids in some elements by a "cocktail shaker" action.

Tests were completed on piezoelectric and resistance strain gauge pressure transducers in the radiation facility. The piezoelectric transducer failed, as indicated in the November Progress Report (ANL-6269). The strain gauge transducer was still operative without significant change in characteristic at 10^9 roentgen total dose when it was removed from the radiation field.

9. Fuel Development and Fabrication

a. Fabrication and Assembly of Fuel - The production of fuel pins, the assembly of fuel rods and the construction of fuel elements during the past month is shown in Table VI.

Table VI. Fabrication and Assembly of Fuel Elements

<u>Operation</u>	<u>Cumulative Numbers of Fuel Elements</u>		<u>Production for Period</u>
	<u>Nov. 19</u>	<u>Dec. 19</u>	
a. Fuel Pins Cast	10,496	11,083	587
b. Fuel Pins Processed Inspected & Accepted	9,236	9,948	712
c. Fuel Rods Assembled and Welded	8,988	9,788	800
d. Fuel Rods Bonded Inspected & Accepted	7,455	8,454	999
e. Fuel Rods Assembled into Fuel Elements	1,183	4,732	3,549
Number of Elements	13	52	39

Each fuel element contains 91 fuel rods.

A survey of scrap, residues from previous production and unused material from the Core I fuel allotment indicates that enough useable fuel alloy can be salvaged from these sources to make 700 to 1000 more fuel rods. Since these can be obtained at low cost with a comparatively large cost reduction over chemical recovery, the fabrication operation was extended for another six weeks.

The 55 enriched elements necessary for the first loading were completed on December 21, using simulated remote control techniques. The remote control fuel element assembly demonstration was thus completed. Blanket fuel elements, control rods, safety rods and other special fuel elements, of which 54 are required, are being bench-assembled by manual operation.

b. Blanket Fabrication - The blanket rod loading phase of the EBR-II program is now nearing completion. The last batch of inner and outer blanket rods was run through the bonding furnace and eddy current inspected during the week of December 19, 1960. To date 15,722 blanket rods have been assembled and 15,019 have been accepted.

Shop assembly of blanket rods into blanket elements is continuing as the necessary hardware becomes available. The following tabulation shows a summary of production to date:

	<u>Produced</u>	<u>Required</u>
Inner blanket elements	72	72
Outer blanket elements	16	548
Upper blanket elements	81	79
Lower blanket elements	81	79

The effort on outer blanket elements has been hampered by slow delivery of bottom adapters.

10. Fuel Prototype Irradiation

A new EBR-II, Mark I fuel specimen is being encapsuled for irradiation in CP-5. The capsule has been prepared to attain a maximum central metal temperature of 625°C to 650°C. Again the goal will be to achieve a burnup of 2% of total atoms at reactor operating temperature.

11. Physics

a. Shield Design - An estimate was made of the gamma-ray radiation levels at two instrument positions. The estimates were based on calculations using the final configuration which includes a neutron shield inside the vessel region. The value for a radial position of 145 cm is 5×10^5 r/hr. The value at 180 cm is 1×10^5 r/hr.

b. Dynamics - The dynamics of EBR-II was revived and the old thermal model checked. The response of a pure thermal model will be determined for the latest set of values of the reactivity coefficients. When work is complete on the thermal model the metal phase transformation effects will be integrated into the model and the response determined. When the response of the thermal model with phase transformation is complete, an attempt will be made to integrate the effect of structural features into the model.

12. Process Development

a. Melt Refining Process Technology - Additional uranium-five percent fission alloy fuel pins are undergoing irradiation in CP-5 and MTR to provide material for further melt refining experiments at high activity levels. The pins, which contain ten percent enriched uranium, will be irradiated to a burnup of about one percent of the total atoms present.

Nitridation rates of uranium and uranium-fission alloy with and without sodium coatings were measured at 197°C, 308°C and 504°C in an

argon atmosphere containing five volume percent nitrogen. The purpose of the study is to provide information needed to evaluate the probable extent of nitride formation on fuel pins during their handling and storage in the EBR-II Fuel Cycle Facility. In all instances the reaction followed a parabolic rate law, with experimental activation energies in the vicinity of 15 kcal/mole. Uranium metal and the fissium alloy reacted at approximately the same rate, and in both cases the presence of sodium coatings increased the rate by a factor of about two.

b. Processing of Melt Refining Skulls - Work has continued on the development of a process for recovery and suitable purification of fissionable material contained in crucible skull material remaining in zirconia crucibles after melt refining operations. The process under consideration involves conversion of the material to oxides (to permit removal from the crucible), reduction in liquid metal solutions, two purification precipitations of the uranium, and final retorting to isolate a uranium metal product.

Reductions of skull oxide by undiluted magnesium or by high magnesium-zinc alloys appear to be less satisfactory than reductions by zinc containing about five weight percent magnesium. In either case, the use of a fused halide flux is beneficial, since it increases the reduction rate and removes the magnesium oxide by-product of the reaction from the metal phase.

A systematic survey has been conducted on the effect of flux composition on the rate and extent of uranium oxide reduction by zinc-five percent magnesium. Definite trends appeared among the alkali-alkaline earth chloride fluxes used in the study, the fastest and most complete reductions being achieved with those fluxes containing alkali and alkaline earth metal cations of the lowest atomic weight. In the better fluxes, the reduction of uranium oxide was quantitative within 30 minutes or less.

Quantitative reductions of skull oxides were obtained within two hours in experiments in which 5 percent magnesium-zinc was used with the following molten halide fluxes: (1) magnesium chloride, sodium chloride, and magnesium fluoride in the respective mole percentages of 41, 54, and 5; and (2) magnesium chloride, calcium chloride, and magnesium fluoride in the respective mole percentages of 47.5, 47.5 and 5. Process demonstration runs employing salt fluxes are being started. A small-scale (15 grams of uranium) demonstration run showed good removals of cerium, molybdenum, and zirconium, thus providing encouragement that the process will provide adequate decontamination.

The extraction of magnesium oxide (produced during skull reduction) by several different molten salt fluxes was nearly quantitative. This method of removing the magnesium oxide is considered to be much superior to other methods such as filtration and liquation of the metal to induce separation of the magnesium oxide followed by withdrawal of the metal from below the surface layer containing the oxide.

The chemical factors involved in the use of fluxes for oxide reductions are being investigated. The solubility of uranium oxide in various

flux systems is being determined, and an attempt is being made to identify the chemical species present as suspended solids and in solution.

Plutonium dioxide was reduced to the metal by 5 percent magnesium-zinc, using a calcium chloride-magnesium chloride-magnesium fluoride flux. The reduction was complete in less than 30 minutes.

The skull reclamation process requires a retorting step to separate a zinc-magnesium liquid metal phase from a precipitated uranium solid phase. In two retorting runs in which the uranium-zinc intermetallic compounds were decomposed with magnesium in tantalum crucibles, it was found that a considerable portion of the product was in the form of large (up to about 1-inch diameter) coalesced masses of uranium. Because this form is very easily handled, the formation of these masses will be encouraged. The factors involved in the formation of these masses are not yet known, but it is believed that stirring may be an important factor in their formation.

The first of a series of active runs for the study of the behavior and of methods of handling volatiles produced during skull oxidation was completed without difficulty. Iodine proved to be the major volatilized activity and it was largely removed or deposited on a Fiberfrax plug. Detection of the possible presence of other volatile species was masked by the strong iodine activity.

c. Blanket Processing - A possible simplification of the process for the isolation of plutonium bred in the uranium blanket of the EBR-II reactor involves the direct low temperature (450°C) reaction of uranium with zinc in slight excess of the stoichiometric amount needed for the intermetallic, $\text{UZn}_{1.5}$. Treatment with magnesium decomposes the intermetallic and dissolves the plutonium. Two additional runs were completed with full size (0.4-inch diameter) EBR-II blanket rods in which nearly complete formation of the intermetallic was achieved in 100 hours. This reaction time is not considered excessive in view of the process simplification which has been achieved by this procedure.

III. STUDIES AND EVALUATIONS (040116)

A. CANEL Evaluation

A review of a proposal to the Commission by Pratt and Whitney based on CANEL work was conducted on a classified basis at the request of the Chicago Operations Office.

B. Nuclear Superheat Studies

A survey is being conducted to determine the most promising economic reactor concept for a separate steam superheater. The unit will be supplied with 1000 psia saturated steam which is to be superheated to 1050°F. The gross electric power is set at 575 Mw for the study.

1. Heat Transfer

A comparison was made of two steam superheating concepts to show the steam temperature incentive of the indirect cycle with a sodium heat exchanger when compared with a direct steam-cooled reactor for the case of equal heat transfer surface temperatures. The two systems studied were a light-water-moderated, direct steam-cooled reactor, and a graphite-moderated, sodium-cooled reactor.

It was concluded that:

(a) The indirect cycle has a 100°F greater potential than a direct steam-cooled system for a surface temperature limit of 1250°F.

(b) A sodium graphite reactor with UO₂ fuel and a surface temperature limit of 1250°F could not produce 1050°F steam without drastically affecting the economics of producing power.

(c) With UC fuel the SGR system is essentially surface temperature limited.

(d) If UC fuel is developed, the SGR will be a close competitor of other power producing systems.

2. Turbine Cycles

The H₂O-moderated pressure vessel type superheater design is not adaptable to reheating either internally or externally. Internal reheating introduces a great deal of mechanical complexity to the pressure vessel design, and external reheat to initial throttle conditions (1050°F) requires that the reactor steam flow be about doubled. The outlet steam nozzles are already as large as feasible, thus external reheating is ruled out. Exceptions to this conclusion on external reheat are encountered in some PWR and BWR dual cycle saturated systems, where reheating of wet steam is done at a reduced pressure by condensing primary steam.

The cycle selected, therefore, is a 1000-psi, 1050°F, regenerative, non-reheat type. Five stages of feedwater heating are utilized, giving a boiler return enthalpy of 80% of the maximum. The gross cycle efficiency is

39.2%; with 6% power for auxiliaries this reduces to a net of 36.8%. With a 1000-Mw boiler and the 475-Mw superheater, this combination gives a net electric output of 540 Mw. The improvements in net plant heat rate that are available by going to reheat, and from 1000 and 2400 psi, are about 5% and 6%, respectively. The combined effect of these improvements will increase the net cycle efficiency from 36.8% to 41.4%. This latter efficiency is the one used for the 2400 psi 1050^oF/1050^oF cycle for the SGR.

The net heat rate for the BWR-H₂O superheater cycle is 9270 Btu/kwh; the SGR cycle is 8240 Btu/kwh.

3. Economic Studies

Comparative net power cost figures were prepared for present and 1970 plants for each of the cases listed in Table VII.

Table VII. Tabulation of Total Power Costs - 540 Mwe Plants

	<u>All BWR</u>		<u>Separate Superheat</u>		<u>Internal Superheat</u>		<u>Coal</u>	
	<u>Now</u>	<u>1970</u>	<u>Now</u>	<u>1970</u>	<u>Now</u>	<u>1970</u>	<u>Now</u>	<u>1970</u>
Capital Charges	5.07	5.07	4.71	4.71	4.67	4.67	3.33	3.33
Fuel	3.00	2.08	2.69	1.86	2.69	1.86	3.28	4.26 *
Maintenance	0.70	0.70	0.70	0.70	0.70	0.70	0.40	0.40
Total Power Cost (mills/kwh)	8.77	7.85	8.10	7.27	8.06	7.23	7.01	7.99
Incentive (all BWR base)			0.67	0.58	0.71	0.62	1.76	-0.14

* 30% fuel cost increase.

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.

The reaction of aluminum with uranium oxides is being studied by use of differential thermal analysis.

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

The series of runs with zirconium wires in the high-pressure condenser discharge apparatus is continuing. Pressure range has been extended to 1500 psi. The data from zirconium runs with water at room temperature and with heated and pressurized water show different extents of reaction as follows: (1) reaction of metal with water at room temperature increases gradually reaching 20% of total metal reacted at an initial metal temperature of 2600°C and (2) reaction of metal with water heated to 90-310°C (vapor pressure 10 to 1500 psi) reaches over 50% of total metal reacted for fully melted metal at the melting point (1840°C).

Pressure traces from zirconium runs in the high-pressure region are similar to those from uranium in room temperature water in that pressure-rise rates increase regularly. Runs with solid zirconium at the melting point required several tenths of a second to reach a final value. Runs with fully liquefied metal at 2200°C were essentially completed in several milliseconds.

Studies of the aluminum-water reaction by the pressure-pulse method are continuing. Runs at 1000°C with contact times of 0.11, 1.0 and 10 seconds showed that the film formed is protective in that the reaction rate decreases markedly with increasing contact time.

The tenth series of in-pile, metal-water experiments was completed in the TREAT reactor. In this series, meltdown transients were performed on fuel specimens containing urania and 304L-stainless steel. Both mixed oxide core (ceramic) and cermet core fuel pins were run (composition given in table). The burst energies ranged from 290 to 648 megawatt-seconds with the corresponding thermal neutron doses varying from 4.88×10^{14} to 9.10×10^{14} nvt. The reactor periods were between 49 and 72 milliseconds. Each of the cermet core fuel pins melted with extensive formation of fine particles. The cermet fuel pins attained temperatures greater than 1500°C; no pressure

spikes were observed. The peak pressure exhibited 1- to 2-second delays relative to the time of the power peak of the burst. The maximum pressure pulses recorded for the unclad, cermet fuel runs were 100 and 900 psi; it is likely, however, that at least one-third of this pressure is the radiation influence on the unbonded strain gage pressure transducer. Data obtained on the extent of metal-water reactions for these experiments are given in Table VIII.

Table VIII. Results of Series X Metal-Water Reaction Experiments

Fuel Pin		Burst Energy (Mw-sec)	% Reaction of Metal
Core	Clad		
Ceramic (w/o): ZrO_2 , 81.5; CaO , 9.1 U_3O_8 (93% enriched), 8.7; Al_2O_3 , 0.7.	Al	290	0.3
Ceramic (w/o): ZrO_2 , 81.5; CaO , 9.1; U_3O_8 (93% enriched), 8.7; Al_2O_3 , 0.7.	SS-304	460	4.2
Ceramic (w/o): ZrO_2 , 81.5; CaO , 9.1; U_3O_8 (93% enriched), 8.7; Al_2O_3 , 0.7.	Zr-2	648	24.0
Cermet (w/o): SS 304L, 90; UO_2 cermet (93% enriched), 10.	none	510	9.6
Cermet (w/o): SS 304L, 90; UO_2 cermet (93% enriched), 10	none	435	6.6
Cermet (w/o): SS 304L, 90; UO_2 cermet (93% enriched), 10.	SS-304	512	10.2

2. Kinetics of Oxidation and Ignition of Reactor Materials

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.

Studies of the ignition temperature of uranium powders by the burning curve method are continuing. It was anticipated on the basis of Frank-Kamenetskii theory of explosions that the critical height values of a given powder size should decrease with heating rate, if the use of critical height values for the significant geometrical dimension of the container were to account for the ignition behavior of uranium powder. For heating rates of 6° to $40^\circ\text{C}/\text{minute}$ the critical heights of -80, +100 mesh spherical uranium powder decreased from 3.8 to 2.5 mm and the ignition temperature increased from 250° to 280°C . Values of the activation energy of the reaction were obtained using measured critical heights and limiting ignition temperature values. At heating rates of 18° and $40^\circ\text{C}/\text{minute}$ the calculated energy of activation was 16.9 kcal/mole for -80, +100 mesh spherical uranium powder.

A theoretical model of burning propagation along metal foils in air was shown to describe burning of uranium but not that of zirconium. In order to determine whether this property of zirconium was unique, the propagation rates of iron, magnesium, and titanium have been determined and theoretical rates are being calculated for comparison with experimental results.

Experiments have been carried out to determine the extent to which halogenated hydrocarbons, which lower the burning propagation rate of uranium and zirconium, operate simply by removing heat from the burning metal. No effect of the agents was observed on the power required to maintain an electrically heated platinum foil at constant temperatures of 300°, 700° and 1000°C in air. The possibility of an effect on the effective diffusion of oxygen to the metal is currently being examined.

B. Fast Reactor Safety Studies

1. Meltdown Studies

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

a. EBR-II, Mark I - Dry Samples - During the report period, the remaining samples (Series XII; see Progress Report for October, 1960, ANL-6253, page 30 for description) were examined from a series of six experiments on the effects of high total sample energy input on failure resulting from short (duration about 0.3 sec) clipped TREAT power bursts. All six transients were initiated with a period of about 110 milliseconds, and the series covered an energy input range from about 100%-250% greater than that necessary for failure. The results of post-mortem examinations on all six samples were consistent with the previous results on high energy input samples, with the fuel alloy being sprayed out laterally, relatively uniformly along the entire length of the element, and very little cladding remaining.

All four natural enrichment, niobium-clad EBR-II elements of Series XIII were examined. Two samples, which reached maximum temperatures estimated to be about 1250°C and 1420°C, respectively, showed no failure. Neither did an element which attained a temperature of 1380°C. The remaining element which recorded a maximum cladding temperature of 1480°C, was found to be failed in three places. Appearance of the holes in the cladding was characteristic of failure due to internal pressure, and essentially all the fuel alloy had been expelled. The results of Series XIII thus confirmed the three previous experiments on such elements, in which failure for short duration transients came in the temperature range of 1300°C-1500°C, and clad bursting from internal pressure was the principle mechanism of failure. Samples have been taken for metallographic studies of fuel-clad interactions.

b. EBR-II, Mark I - Samples in Molten Sodium - Experimentation entered the area of samples contained in stagnant sodium, as two capsules each containing a single 3% enriched stainless steel-clad EBR-II element inside a sodium bath were run at TREAT. Original plans had called for the initial series on meltdown in stagnant sodium to include four capsules: one for a check transient, and three additional capsules for meltdown experiments proper. However, the instrumentation of two capsules was damaged: one in shipment and the other in the final assembly. The remaining two samples were run in order to obtain two energy input "check

points." The first sample was given a check burst of 40 Mws, which raised the sodium bath container about 40°C , and then a more severe transient of 140 Mws was run, which produced a container temperature rise of about 127°C . The second capsule received a reactor power burst of 180 Mws, which produced a sodium container temperature rise of 178°C . These samples are being shipped back to the Lemont site for examinations.

c. Fermi-I - Dry Samples - Previous testing had shown a difference in the response of dry Fermi-I samples to short (about 2-second duration) transients and flattened power transients (Progress Report, October, 1960, ANL-6235, pp. 30-31). Five samples are being encapsulated in the standard dry meltdown containers for a series of meltdown experiments investigating the effects of flattened power bursts of higher total energy inputs than previously employed, and the effects of flattened power bursts with higher heating rates.

d. Sample Temperature Calculations - A report has been issued on a heat transfer code, RE-147 ("Cyclops"), developed to calculate temperature distributions as a function of time for a cylindrical, bonded and clad fuel element immersed in coolant.* Power generation in the element is specified as a function of time, radius, and axial position. Provision is made in the code for phase changes in the fuel, bond and clad (lumped together as one region) and coolant. Options are available for flowing or stagnant coolant. The code has been in use for several months for calculating sample temperatures for TREAT experiments and for studying the effects of various experimental settings.

2. TREAT - Operation and Development

A series of oscillator tests were made to compare the characteristics of Argonne-designed compensated and uncompensated boron coated parallel plate ion chambers. Measurements of the reactor transfer function were also made with detectors in the core and reflector to investigate the effect of detector location on the magnitude and phase of the transfer function. Preliminary analysis of these data indicate that there is no difference in the response of the compensated and uncompensated chambers at frequencies above 0.1 cps. Further analysis of these data is in progress and more complete results will be reported.

Two steady state runs were made on EBR-II type fuel pins in special containers in which tantalum end-pieces were used as an absorber to shape the flux along the pin. U-235 foils were spaced along the pin to measure the resulting flux shape.

The fifth GE-ANL test sample was irradiated. Post-transient examination of the capsule at the TREAT site revealed a hole in the side of the containment capsule. The capsule was immediately sealed in a steel tube and returned to GE for critical examination. No significant permanent contamination of the reactor or reactor building resulted from the release of fission products from the test capsule.

* J. Heestand, D. F. Schoeberle, and L. B. Miller, "The Calculation of Transient Temperature Distributions in a Solid Cylindrical Pin, Cooled on the Surface - IBM-704 Program 524/RE-147," ANL-6337 (October, 1960).

Work on the TREAT Mark II control rod is complete. Drawings and graphs have been made and a report is being written.

A zirconium control rod section used in one of the transient tests was damaged in Idaho and had to be returned to Argonne for reworking. The work has been done and the rod returned.

A special magnet coil is being fabricated for test with a new release circuit. This circuit is intended further to reduce magnet delay in a special proposed experiment.

Handling tests were made at TREAT using a mockup of the transparent meltdown facility, a mockup of the handling coffin, and extension rails to support the equipment while it was inserted into the reactor. One complete assembly of the prototype transparent facility has been delivered by the shops, and is being given assembly tests. An order for fabrication of the handling coffin was placed with the shops after completion of the handling tests.

A preliminary design study has been made for a large flowing sodium loop for use in the TREAT reactor. Such a facility would permit simulation of reactor conditions of heat removal which are not possible in capsule-type tests.

V. NUCLEAR TECHNOLOGY AND GENERAL SUPPORT (040400)

A. Applied Nuclear and Reactor Physics

1. Experimental Physics

a. Fast Neutron Cross Sections - Some of the previous measurements of inelastic scattering from thorium were subject to ambiguous theoretical interpretations. This was particularly true for scattering leaving the residual thorium nucleus with 700-900 kev of excitation energy. This energy region has been examined with considerable care. The work was very tedious owing to the necessary long flight paths and very thin targets. From these measurements it was evident that there are three levels in the thorium nucleus between 760-850 kev. These levels can be related to the systematics of even-even nuclei. The resulting spin and parity assignments are utilized in the theoretical interpretation of scattering from this important isotope.

b. Diffusion Parameters of Room Temperature ZrH₂ - The time-dependent neutron distribution (see Progress Report, November, 1960, ANL-6269, page 43) in ZrH₂ is described by the exponential decay constant

$$\lambda = \overline{v\Sigma_a} + D_0 B^2 + cB^4 + dB^6$$

Essentially complete experimental measurements indicate that the B^6 term of the expansion is required to fit the experimental results. Preliminary analysis of the data yields the values:

$$\overline{v\Sigma} = 3,830 \text{ sec}^{-1}$$

$$D_0 = 59000 \text{ cm}^2 \text{ sec}^{-1}$$

$$c = 105000 \text{ cm}^4 \text{ sec}^{-1}$$

$$d = 143600 \text{ cm}^6 \text{ sec}^{-1}$$

The experiment value for $\overline{v\Sigma}$ is about 4% larger than the value calculated from the chemical analysis. However, since this does not include any trace elements, and considering the accuracy of the analysis, the agreement can be called good. The above preliminary results do not include about two-thirds of the data taken at $B^2 > 0.2 \text{ cm}^{-2}$. The inclusion of these points may cause a change in the coefficients $\underline{c}$ and $\underline{d}$ but should have little effect on $\overline{v\Sigma}$ and D_0 .

Time dependent transmission measurements have been made at three values of B^2 in order to estimate the neutron thermalization time constant. The value obtained is $\sim 120 \mu \text{ sec}$ at $B^2 = 0$.

c. Accelerator Instrumentation - During the month a considerable overhaul of the accelerator vacuum system was accomplished. This work has been indicated for some time due to the ever-increasing amount of instrumentation kept under very high vacuums. Simultaneously, a second electrostatic analyzer was installed. This new unit now permits precise energy control of the accelerator system when operating on a second target unit. Precision measurements were previously confined to only one target assembly.

Assembly and test of the fission chamber and neutron detection apparatus to be used for the determination of β of thorium and of U-238 was completed satisfactorily. The apparatus is now being installed in suitable shielding. In conjunction with this experiment use will be made of a digital timing system which has been developed. This electronic device utilizes solid state components throughout. As a result, it is hoped that the unit will be very reliable.

d. High Conversion Critical Experiment - The addendum to the Hazards Summary Report for the ZPR-VII system, which describes the Hi-C program of critical experiments, is being prepared for submission to the Commission for authorization to begin the program. Meanwhile, the assembly of the core structure for the new program has been completed. Rod drop tests have shown that their performance is satisfactory. The instrumentation has been subjected to a thorough testing and a revised operating manual has been prepared.

Preparations for the experiments to determine the various nuclear parameters are under way. Several new techniques will be tried. One is the use of gamma-gamma coincidence counting for the detection of U-238 captures (a procedure suggested by R. Sher¹). The use of lutetium as an indicator of the degree of thermalization of the neutrons will be tried, following a procedure reported by L. C. Schmid and W. P. Stinson.² A noise method for the determination of the effective delayed neutron fraction, β_{eff} , based on a technique suggested by the study of fluctuations in neutron population in reactive assemblies, will be investigated. Standard techniques for the determination of other criticality parameters, as employed in the THUD program, will be relied upon for an understanding of the nature of the high metal-to-water ratio systems to be assembled in the Hi-C program.

e. Thermal Reactor Spectra - Additional effort has been devoted during the current reporting period to the improvement of the shielding around the reactor (ZPR-IV) and detector in order to increase the signal-to-noise ratio. Such changes reduced the background counting rate at the detectors to about 11 counts per minute which represents an improvement by a factor of about 50 over the best value achieved heretofore. To accomplish this, it was necessary to shield and evacuate the flight path (since it was discovered that a large portion of the background was due to air scattering) and also to shield the entire chopper from the high flux of scattered neutrons present in the ATSR reactor cell.

1. R. Sher, " γ - γ Coincidence Method for Measuring Resonance Escape Probability in U-238 Lattices," Nuc. Sci. & Eng. 7, 479-480 (May, 1960).

2. L. C. Schmid and W. O. Stinson, "Nuclear Physics Research Quarterly Report, January to March, 1960," HW-64866, 126-136 (1960).

With this shielding arrangement counting rates were observed with the chopper opened (not rotating), rotating and closed. Comparison of the rates revealed no significant difference between that with the chopper rotating and with it closed. The rate with the chopper in the open position yielded a counting rate difference of 2.5 ± 1.1 counts/min. Thus it appears that the background still has not been reduced to the point where meaningful spectra can be measured, while shielding improvements are approaching the point of diminishing returns with the ATSR system. It also appears that flux from this reactor is too low for the Selove chopper.

A foil activation experiment indicates that the thermal neutron flux at the entrance to the chopper collimator is approximately 2×10^6 , with a cadmium ratio of about 4.3 for a power level of about 0.5 kw. At this level, a running time of approximately 500 hr would be required to attain a spectrum with a 2% statistical precision per channel if a high signal-to-background ratio were achieved. Operation at between 10 and 100 times the present power might be feasible, if shielding were added to the ASTR and if the flight path shielding were sufficiently improved. The possibility of adding a boron compound to the shield water and adding cadmium sheets to the face of the shield tank is being considered. However, it would be advantageous to move these preliminary experiments to the Juggernaut when it becomes available in order to make performance measurements under the actual conditions anticipated for this experiment.

f. Measurement $\overline{\nu}_{th}$ for U-235 - Preliminary experiments are being performed at one of the thermal columns of the CP-5 reactor. The main measurements made are an investigation of the manganese to hydrogen absorption ratio, the absorption per fission ratio, and a rough value for the number of neutrons per fission. Information being gained is mainly that of counting rates, beam strength, and background. The data have not as yet been analyzed.

In preparation for these measurements a quantitative technique for the electroplating of uranium onto backings of aluminum, platinum, and stainless steel has been developed. Thicknesses from 30 to 1000 μ g/sq cm have been obtained which are sufficiently uniform and adherent for the purposes of the $\overline{\nu}$ measurement. Such foils were incorporated in a methane flow-type fission chamber and used in CP-5 experiments. Tests with a Hornyak button neutron detector were performed in a new shielding arrangement and large improvements in the signal-to-background rates resulted. It was apparent that the source of residual neutrons was primarily leakage down the beam hole and scattering up into the detector, and so it is now known how to design the final shield arrangement for the fission counter calibration by fission-fast neutron coincidence counting.

The manganese bath system was received from the vendor, assembled, and installed at the thermal column. In the process of doing these preliminary experiments, various procedures involving the system have been tested including mixing, pumping, circulation, cleaning and emptying. No serious mechanical problems have been encountered, although the need for a number of minor improvements is apparent.

After interpretation of the observations obtained during the preliminary measurements, necessary modifications to the experimental apparatus will be made. At that time additional experiments using the manganese bath equip-

ment will be initiated at the CP-5 thermal column. About two months later the fission-fast neutron coincidence circuitry should be delivered and then the final measurements at the thermal column can begin. It will be desirable to make comparison measurements with Cf-252 to determine independently the importance of certain corrections necessary for the U-235 experiment.

g. Calibration of ATSR - In preparation for measurements of resonance integrals using the central thimble and pneumatic sample changer in the Argonne Thermal Source Reactor (ZPR-IV'), a number of measurements have been performed. A vertical plot of the flux in a 5.1 cm diameter void tube located at the center of ATSR indicated that the control rods located at nearby positions in the core lattice caused no serious flux depressions in this direction. Foils placed in a horizontal plane at the vertical midpoint of the core, however, indicated the presence of some degree of asymmetry. This will be investigated further and an attempt made to provide a uniform loading in order to eliminate this flux gradient.

It will first be necessary to establish the size of void tube that will be used for the resonance integral measurements. In this regard three important effects will have to be considered: (1) the desirability of having a zero temperature coefficient; (2) the size of the cadmium-wrapped void tube for which it will be possible to attain criticality for the ATSR; and (3) the effect the void tube size on the spectrum inside the tube. Some previous measurements exist for the temperature coefficient as a function of void tube size, and measurements have also been made with cadmium tubes of 2.5 and 4.4 cm diameter. Studies of the spectra will involve both thermal and resonance detectors. Measurements with the resonance detectors are of particular importance in order to verify the $1/E$ behavior normally assumed for the epithermal spectrum. Au-197, As-75, Co-59, and Mo-98 will be used to sample the flux at 5, 47, 132, and 480 ev, respectively.

h. Dynamic Behavior of Ion Chamber Current Amplifiers - Studies were made of the dynamic behavior of feedback-type current amplifiers used with ionization chambers. It was found that the response time is affected not only by the capacitance of the input circuit, but also by the stray capacitances from input to output. However, it was found that the time constant due to those capacitances could be eliminated by a simple method which essentially amounts to cancellation of the capacitor charging currents in the input circuit. It was also found that large time constant input circuits do not work well with amplifiers having large time constants of their own (such as chopper and vibrating-reed types which require low-pass filters to eliminate the chopping frequency), the combination resulting in under-damped oscillatory response.

2. Theoretical Physics

a. Reactor Physics Constants Center - The revision of ANL-5800, Reactor Physics Constants, continues to consume a large fraction of the time of reactor physics personnel. In the Reactor Engineering Division, the Theoretical Physics Section is heavily involved. In addition, large demands have been placed upon members of the Experimental Physics, Applied Nuclear Physics, Instrumentation and Control, and Fast Reactor Safety Sections.

First draft copy of the entire volume has now been prepared and chapter editors have begun work on final revisions.

b. Resonance Integral and Age Compilation - A center is being set up for the collection and tabulation of resonance integral and neutron age data. Published and unpublished values are sought and the center will publish periodic reports of the available data.

A preliminary draft of the integral compilation has been completed. Reports on theoretical and experimental determinations of neutron ages are being collected and the values of interest listed.

B. Reactor Fuels Development

1. Ceramic Fuels

a. Refractories for Reactor Use - Investigation of the lanthana-urania system was continued. Several compositions containing from 30 to 85 weight percent urania were placed in Vycor capillaries in the presence of uranium metal chips and vacuum sealed. The capillaries with the samples were annealed at 900°C for 200 hours. X-ray diffraction patterns of annealed compositions indicate that cubic fluorite solubility existed between 45 w/o urania and pure UO_2 . Lattice parameter measurements suggest the uranium content of these compositions has been reduced by the uranium chip anneal and probably approaches the +4 state. Several powdered compositions were found to be pyrophoric when exposed to air.

Chemical analyses of oxidized and hydrogen reduced compositions have been received and used to calculate stoichiometry of solid solutions. The results on oxidized compositions indicate that the uranium content of high lanthana solid solutions (between 30-50 w/o urania) is predominantly in the +6 state. At uranium contents greater than 50 w/o urania the average uranium valence was reduced to the +5 state at approximately 85 w/o urania. Limited data on hydrogen reduced compositions suggest a maximum uranium valence of +5.0 for high lanthana solid solutions. The average uranium valence was lower at higher uranium concentrations.

Air sintered compacts containing from 30 to 85 w/o urania are being subjected to heat treatments in air at 1000° , 1450° , and 1700°C for periods in excess of 100 hours. Phase stability, weight loss due to volatility, and microstructural changes are of interest.

A high temperature thermobalance has been constructed and will be used to determine stoichiometry of solid solutions at elevated temperatures.

b. Urania-Thoria Bodies - The urania-thoria program has been concerned primarily with the oxidation of air sintered $\text{U}_3\text{O}_8\text{-ThO}_2$ mixtures and hydrogen sintered $\text{UO}_2\text{-ThO}_2$ mixtures. Oxidation studies by weight loss methods have been used as a method of evaluating the stoichiometry of the solid solution following sintering and also as an indication of the stability of the solid solution in use.

During the past month solid solutions in the range of 2 to 70 mole percent UO_2 have been prepared by air sintering $\text{U}_3\text{O}_8\text{-ThO}_2$ compositions.

Weight change determinations have been made for these materials after reduction in H_2 and oxidation in O_2 for 100 hours at $1000^\circ C$. Stability and stoichiometry evaluation will be made from these data. The change in X-ray lattice parameters resulting from these treatments is also being evaluated.

c. Uranium-Thorium Sulfide - The effect of an H_2 atmosphere on the sintering of US must be redetermined in a more pure atmosphere. The X-ray of the last H_2 -fired US showed lines for UO_2 and UOS of almost equal intensity with no hydride apparent.

The electrical resistivity apparatus has been refined to the point where results are quite reproducible. A ThS sample had a resistivity of 16 micro-ohm-cm, about a tenth that of US and about the same as that of Ta.

d. Development of Methods for Melting and Casting UC-PuC - The purpose of the program is to develop methods for melting and casting fuel shapes from PuC, UC, and (Pu-U)C.

The current leakage and high temperature arcing problems with the high temperature carbon resistor furnace have been partly remedied by (a) increasing spacing between elements of unlike polarity, (b) filleting sharp edges, (c) coating conductive surfaces with plasma-gun sprayed alumina, (d) added mica insulation to the bottom of the furnace and (e) increasing the separation of the heat shields from the graphite conductors. Tests of this equipment have indicated that it will serve satisfactorily in the PuC melting range and over much of the (Pu-U)C melting range to about $2200^\circ C$. Current leakage may prevent the effective use of this furnace above $2200^\circ C$. A combination mold and crucible pour-hole plug, designed to be actuated by a crucible lift mechanism, was installed.

e. Properties of Plutonium Carbide - Uranium monocarbide has shown considerable promise as a high temperature, high burnup reactor fuel. By analogy, plutonium monocarbide appears promising enough as a potential fast reactor fuel to warrant this study of pertinent properties and methods of fabrication.

Arc melting and casting techniques have been developed and specimens near the monocarbide composition have been made and are under irradiation in EBR-I. The plutonium-carbon phase diagram is being checked in the region of interest and measurements of several properties of these alloys are in progress.

Microstructures of alloys containing from 42.5 to 60 a/o C have been studied in comparison with the chemical analyses. Alloys containing 45 a/o C to less than 50 a/o C have a cored dendritic structure of PuC. The composition of PuC is stoichiometric at its melting point and broadens rapidly with decreasing temperature from about 44 a/o C to about 50 a/o C at the melting point of plutonium. Alloys of 47.5 a/o C to 60 a/o C have dendrites of Pu_2C_3 in a PuC matrix. The 60 a/o C alloy is single phase (Pu_2C_3) and melts congruently.

The liquid composition in the peritectic reaction, $L + Pu_2C_3 \rightarrow PuC$ has been determined as 46 a/o C by graphical extrapolation of ratio of inten-

sities of the (510), (431) diffraction lines of Pu_2C_3 and the (311) diffraction line of PuC against the composition. This is in agreement with the microstructure of the 47.5 a/o C and the 45 a/o C alloys.

Previous thermal expansion measurements on the 45 a/o C alloy have indicated the existence of a phase transformation from 450°C to 560°C . This transformation has been confirmed by metallographic examination and work is in progress to determine the equilibrium phases present at lower temperatures. This is probably a transformation of PuC to zeta carbide. Since there is a large expansion associated with this transformation the exact phase boundary is of interest and alloys from 42.5 a/o C to 50 a/o C are being investigated.

Confirmation has been obtained for the previously reported formation of a phase with a large thermal expansion by working the 50 a/o C alloy. This phase has been investigated by X-ray diffraction and metallography. Metallographically it is present as a network through the grains and is extremely brittle and reactive. A quantity sufficient to be detected by X-ray diffraction has not yet been produced. This phase can also be produced by very rapid quenching from 700°C .

Heat treating at 1000°C for 210 hours did not produce any change in the cored PuC microstructure. Apparently much higher temperatures are needed to homogenize these alloys. The cored structure of the 45 a/o C thermal expansion specimen was altered by repeated cycles through the lower temperature transformation.

Evidence in support of a high vapor pressure of plutonium monocarbide has been demonstrated by a deposit on the inside of an evacuated sealed Vycor capsule used to heat treat several PuC specimens at 1000°C . High temperature furnaces capable of vapor pressure measurements and other elevated temperature studies on high melting point fuels are being developed.

Compatibility tests of tantalum and stainless steel with PuC are in progress.

2. Properties of Metals and Alloys

a. Thermal Properties of Uranium-Fissium Alloys - The enthalpy of uranium-5 w/o fissium alloy has been determined in an isothermal drop calorimeter described in the previous monthly reports. Table IX gives the enthalpy data.

Table IX. Enthalpy of Uranium-5 w/o Fissium Alloy

Temp. (°C)	Enthalpy- U-5 w/o Fs ⁽¹⁾	Enthalpy ⁽²⁾ Uranium	Remarks
25	0	0	
35	107	67.0	
50	266		
100	574	514.2	
150	873		
200	1260	1232.0	Error to here is approx. 2.5%
300	2050	2006.7	
350	2400		
400	2700	2847.8	
450	2840		
500	2870	3766.2	
550	2910		
600	3320	4786.5	Extrapolated capsule value; error may be larger than 3%
650	4260		
700	4750	6340.0	
750	4850		3.5% error

(1) calculated gram-atomic-weight = 222.41 g; calories per gaw.

(2) Cummings, D. C., Corruccini, R. J., J. Research Nat'l. Bur. of Standards, 39, 309 (1947).

The over-all accuracy is $\pm 3\%$, as determined by comparison with NBS measurements on vitreous silica. The precision is a maximum of $\pm 1.5\%$, and the average is less than half that value, as determined by duplicate drops.

Since the calorimeter is a drop type, the drop is an effective quench of the alloy. Thus any sluggish transformation affects the final heat output of the specimen, and appears to lower the enthalpy.

3. Corrosion Studies

a. Aluminum Alloys - The possible use of aluminum for the cladding of a free standing oxide pellet fuel in EBWR would require tubing with low creep rates at applied stresses of about 4000 psi at 290°C or below in temperature. Three wrought alloys were tested and all were found to creep 5% at these conditions within a few days. The best, X2219, a special high temperature alloy developed for the aircraft and missile industries, lasted less than 6 days.

As previously described, some aluminum powder products have had sufficient creep resistance to withstand these conditions but it has been difficult to secure commercial tubing combining strength and corrosion resistance. Briefly, the tubing prepared from "as atomized" powder has been corrosion resistant but not strong enough, while the tubing prepared from "ball milled" powder has been strong enough but not corrosion resistant. A third approach, utilizing "as atomized" powder made from a stronger alloy (Argonne 341:Al-1 w/o Ni-0.5 w/o Fe-0.3 w/o Cr-0.2 w/o Zr-0.1 w/o Ti) is now being evaluated. At present a small experimental powder product extrusion is being creep and corrosion tested. Initial results are encouraging.

It has previously been reported that a new aluminum alloy containing 1.5 w/o titanium and 0.5 w/o iron showed good corrosion resistance. Further modifications are being made to this type of alloy. A quaternary -- 1.5 w/o Ti, 0.5 w/o Fe, 0.1 w/o Nb -- has shown at 360°C an initial corrosion very similar to that of A-288 and the ternary mentioned above. In addition, it has a more finely dispersed second phase, and it is hoped that it will show good long time corrosion behavior.

b. Zirconium Alloys for Superheated Steam - Corrosion testing of the alloys mentioned in the Progress Report for September, 1960 (ANL-6234), page 49, has continued, with no appreciable change in rates. However, after extended periods of exposure, slight spalling of oxide has occurred at sharp edges.

A Zr-2.8 w/o Ni sample contained 723 ppm hydrogen (66% of calculated corrosion product hydrogen) after 235 days in superheated steam at 540°C and 600 psi. However, there was no increase in hardness. In fact, no alloy has increased in hardness as a result of exposure to superheated steam. Some have become softer and it has been demonstrated that some alloys are annealed at 540°C.

4. Nondestructive Testing Techniques

a. Ultrasonic Techniques - When sound is sent along the axis of a solid cylinder, it is propagated in several different modes. After eliminating sources of electrical reflections, one is certain that the groups of signals in the echo patterns are really due to these modes returning at different times, and are not generated in the circuitry. It has been often stated in the literature that a rough surface on the cylinder wall will eliminate unwanted modes. If a screw thread is put on the wall, it is said, mode conversions at the wall are broken up, and only the straight line, fastest mode will be propagated. The experiments which have been made at the Laboratory tend to disprove this idea. Photos were taken of the echo pattern on the same aluminum rod with the same crystal and at the same frequency for three different conditions of surface smoothness. In the first condition, the rod was machined smooth; in the second condition thread of 32 per inch, 1/32 inch deep was put on the rod; in the third condition a thread of 8 per inch, 1/8 inch deep was put on the rod. In the first case the sound wavelength was larger than the surface variation; in the second case, they were about the same size. In the third case, the wavelength was shorter than the surface variations. In all the cases, the different modes were still visible, the only things changing were relative amplitudes of different modes.

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

The thicknesses of converter screens which will yield the greatest film density when the film and screens are exposed together to a constant neutron exposure, have previously been determined for cadmium (10-20), silver (20-20) and indium (20-30). The screen thicknesses for optimum film density using rhodium screens have now been determined to be (10-10) where the numbers are the thicknesses of the front and back screens respectively, in mils. As in the cases previously reported (except cadmium) the optimum screen thickness is in the order of the range of the beta particles for that material.

Initial tests with some small samples of gadolinium as screen materials indicate a relative photographic speed about equal to that of silver for (1-2) gadolinium screens. The photographically effective reaction in the case of gadolinium appears to be the high cross section (n, γ) reactions in Gd-155 and Gd-157, since little if any increase in film density has been obtained by permitting the gadolinium to decay on the film after the exposure.

Resolution studies of the various neutron image detectors show that improved resolution can be obtained by the transfer method (in which a radioactive, image-carrying foil is transferred to photographic film) over the direct exposure method (film and screen exposed to the neutron image together). Exceptions to this general statement involve direct exposure techniques using gadolinium and the B-10 loaded scintillator screen. The image resolution obtained using these two direct exposure methods compares favorably with that obtained by the transfer method using both indium and gold.

c. Eddy Current Techniques - The dual frequency eddy current system which has been successfully used to test quantities of various types of tubing has never been developed to its full potential. For this reason, more effort is being put into modifying this system to make it faster and more sensitive.

The dual frequency system used to test the Zircaloy-2 tubing used in core I-A of the EBWR has been modified to increase its speed and sensitivity to small cracks starting from the I.D. It is now possible to detect defects reliably at a rotational speed of 280 rpm which would have been missed by the older equipment operating at a speed of 187 rpm.

C. Reactor Materials Research and Development

1. Irradiation Damage in Steel

a. SA-212B Steel - Eight additional samples of SA-212B (from the Group-IV MTR irradiation) were mounted and readied for shipment to the Nuclear Science and Engineering Corporation (see page 52 of ANL-6269,

Progress Report, November, 1960) for the preparation of replicas for electron microscopic examination. Activity levels are much higher (to 500 mr/hr at one foot: 1-Mev gammas) than any of the 24 specimens shipped previously under the cooperative program with the U. S. Steel Corporation. The Applied Research Laboratory of the U. S. Steel Corporation reported that earlier difficulties with the preparation of replicas were solved and work is now progressing satisfactorily. The few remaining irradiated impact specimens from the Group-IV irradiation are stored for future reference pending results from the electron microscopic examinations.

b. Studies of Magnetic Properties- Exploratory inductance coils consisting of one thousand turns of No. 26 gauge glass-insulated magnet wire, 1 in. axial length, were assembled into a simple a-c inductance bridge for a preliminary evaluation of the suitability of following the changes in structure magnetically. Search coil geometry was arbitrarily fixed to the geometry of available irradiated steel specimens (0.204 in. diameter by 15/16 in. long). Information measurements covering the frequency spectrum (between 20 cps and 20,000 cps), with two unirradiated steels (SA-212B, annealed, and hardened to 44 Rockwell C; and drill rod, annealed, and hardened to about 60 Rockwell C) revealed that the inductance bridge method was capable of resolving the magnetic differences between the annealed and hardened structures adequately. Similar trials with the short (0.204 in. diameter by 3/4 in. long) austenitic stainless steel specimens showed that the simple bridge was not able to distinguish this alloy from air. The result is not surprising inasmuch as the magnetic permeability of the austenitic steels is in the range of 1.02 to 1.10 while for air it is 1.0. An additional qualitative measurement with 3/16 in. by 2 in. long annealed and hardened drill rod also confirmed that the "end effects" associated with short ferro-magnetic specimens had a very large influence on the magnetic circuit - at least equal to that associated with the differences in structure of the soft and hardened varieties of the same steel.

A mathematical analysis of the simple bridge circuit taking into consideration ohmic resistance, inductive reactance, and distributed capacity reactance indicated that the resolution of the bridge could be improved by increasing the inductance of the search coils or compensating the inductance bridge.

Despite these problems, the magnetic analysis technique appears promising enough so that loadings of the EBWR pressure vessel steel irradiation surveillance capsules were modified to include unnotched cylindrical magnet bars 0.204 in. diameter by 3.75 in. long. This substitution sacrifices two multi-notch (three notches per bar) impact bars from the EBWR surveillance program described below.

c. EBWR Surveillance - Approximately 400 SA-212B tensile, impact, hardness, and magnet bar specimens will be prepared. Eighteen helium-filled capsules will be loaded into "bare" and "shielded" thimbles installed within the EBWR reflector region. Withdrawals of pairs of capsules, one each from the "bare" and "shielded" thimbles, are planned for suitable intervals to cover the exposure range of available experimental data.

Each capsule will contain 18 or 19 specimens of the following types:

<u>No.</u>	<u>Bar Type</u>	<u>Orientation*</u>
1	hardness	either L or T
2	magnet	1 L, 1 T
4	tensile	2 L, 2 T
12+	impact	6 L, 6 T

* Note: L - Longitudinal rolling direction.

T - Transverse to final rolling direction.

SA-212B plate is cross-rolled fire-box quality material manufactured to meet the ASTM A-300 specifications for low temperature service.

2. Structural Properties of Steels

a. Samples from the EBWR Vessel - The investigation of long-term stress effects (see page 53, ANL-6269, Progress Report, November, 1960) in SA-212B steel removed from the EBWR pressure vessel has started. Sectioning of these mildly radioactive steel plugs has begun in accordance with previously prepared cutting diagrams. Longitudinal and transverse direction tensile and Charpy V-notch specimens will be prepared for tests.

b. Studies of Thick Welded Sections - A draft of a report describing the development of procedures for the welding of HY-80 plate material has been prepared. The alloy content of the weld metal is below that of the parent plate; the composition of weld metal is substantially different from that of the base metal. A block, approximately 24 in. x 24 in. x 6 in., containing the alloy weld was returned to the Laboratory in accordance with prior agreements. The block will be sectioned, at a future date, into irradiation specimens for studies of: (1) alloy effects; and, (2) performance of quenched and tempered structures.

c. Structures - Preparations for the pressure test for the two cells of Building 315 are proceeding. Some specific tasks have been accomplished. The shell structure was analyzed as longitudinal and transverse concrete frames for internally uniformly distributed loads. The object was to determine what safe pressures, if any, could give detectable, finite strains. This step was necessary in view of the fact that the original design of this structure was based primarily on its adequacy to withstand blasts. The subject frames were analyzed for moments, shears and stresses for internal pressures of 2, 4, 6, 8, and 10 psi. It is intended to compare theoretical strains as obtained from the structural analysis with the experimental strains to be recorded on SR-4 gages during the tests.

Preliminary strain gage work has been initiated to determine the locations, techniques, and calibrations required for indicating stress in the pressurized concrete cells.

D. Reactor Components Development

1. Development of Viewing Systems

a. Basic Glass Studies - Energy Storage - It has been found that the charge displacement induced by gamma irradiation and subsequently released by heating is produced principally by secondary electrons (see Progress Report, October, 1960, ANL-6253, page 48). This conclusion is based on the absorption characteristics of metal absorbers having different atomic numbers. The analysis of absorption curves formed by placing aluminum and lead filters between the 0.125 inch thick glass sample and the Co-60 source showed a component of the radiation which has a tenth-value-layer of 0.21 g/cm² for aluminum and 0.08 g/cm² for lead. The ratio of these values indicates that this component is electronic rather than electromagnetic radiation. In the case of an aluminum filter, 1 mm thick, an induced charge displacement remained which is relatively independent of increased absorber thickness and is about 7% of the unfiltered response. In the case of a lead filter, 0.1 mm thick, a charge displacement remains which is about 33% of the unfiltered response.

b. Basic Glass Studies - Anomalous Glass Absorption - An anomalous light absorption induced by gamma radiation has been observed during the course of a series of studies on the role of cerium in decreasing the light absorption normally induced in glass by intense exposure to gamma radiation. The induced absorption in non-cerium containing glasses usually increases proportional to the logarithm of the exposure with a gradual approach to a constant value in the exposure range of 10⁶ to 10⁷ r. In glasses containing several percent of cerium and melted in air, very little absorption in the visible region occurs before 10⁶ r, and there is a gradual increase with increasing exposure. In the anomalous cases observed (Cabal-type glasses containing several percent cerium and melted under reducing conditions in an inert atmosphere), the absorption in the visible region increases rapidly to a maximum after about 3 x 10⁵ r, and then decreases in an oscillatory manner with increasing exposure. The maximum absorption is much higher than that for a Cabal glass containing a similar amount of cerium and melted in air. In a similar glass an increase in absorption in the ultraviolet region was observed which oscillated with the maxima and minima occurring at the same exposure as those of the absorption in the visible region for the glass described above. No explanation of this unique phenomenon is known at this time.

2. Heat Engineering Studies

a. Hydrodynamic Instability- Armadilla instability work has been delayed by a malfunction in the grid timing system contained in the optical recording galvanometer and redesign of a circuit was necessary. Calibrations of pressure gauges and of the steam flow orifice were made.

Two methods of analysis are being attempted to develop quantitative understanding of hydraulic instability. One employs analysis in terms of a fundamental periodic function and random noise. The other employs auto- and cross-correlation of observed signals and driving functions.

The model proposed for the latter investigation developed a discon-

tinuity at one frequency when the power function of the steam void was determined by means of velocity measurements. In this model, it was assumed that the cross-correlation terms between steam void fraction and water velocity are small compared to the autocorrelation terms. To avoid the appearance of the discontinuity point and to get a more realistic model, it has been decided to retain the cross-correlation terms in the analysis and assume approximate forms for them when solving for the power spectrum of the void fraction. It may be possible that the simpler model gives reasonably good results at high frequency regions where the effect of the cross-correlation terms becomes small.

The transfer function of the Potter frequency converter was checked and the break frequency was found to be about 10 cps, which means good enough response in the frequency range of interest. This value is in accordance with the result obtained by an approximate mathematical analysis of the converter circuit.

b. Packed Beds - A feasibility study on supplying power to a high power-density, packed-bed heat transfer experiment was initiated. Test samples are to be heated by induction heating equipment to find the optimum heating conditions for obtaining high heat flux.

c. Boiling Liquid Metal Experiment - A small-scale loop is being designed for studies of boiling liquid metals. Initial tests will be with sodium and will be aimed at measuring vapor volume fractions and two-phase frictional losses in a flowing system.

The design of the major loop components is complete. Design of instrumentation specifically pertinent to sodium handling and the measurement of sodium temperatures and flow is nearly complete and instruments will be ordered as soon as possible.

The power input transformer has been tested and found to be an essentially constant voltage device. This small value of voltage regulation indicates that no instability or power excursion should occur at the boiling section of the loop.

E. 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 typical Zircaloy-clad oxide fuels.

Studies are being made on a 50 to 100 gram scale of the fluorination, transportation, separation and decomposition of plutonium hexafluoride.

The direct fluorination of high-density, sintered uranium dioxide 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. Methods of decladding of Zircaloy-clad fuel are being investigated. Two of the approaches being used are direct hydro-

fluorination and combinations of hydrofluorination-hydrochlorination reactions. Fluidized beds of inert solids are being used as heat-transfer media.

Processes for recovering uranium from other enriched uranium fuel alloys using similar techniques and direct fluorination are also being studied. These include reaction of HCl-HF , $\text{Cl}_2\text{-HF}$, and other halogen gas mixtures with zirconium fuels. Reactions of stainless steel with various gaseous reagents are being studied in fluidized beds with application to decladding of stainless steel-clad and processing of 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.

a. Direct Fluorination of Uranium Dioxide Fuel - A study of the direct fluorination process applied to uranium dioxide fuel is directed toward demonstration of process control of typical pellet fuel in a three-inch diameter air-cooled reactor. An inert fluid bed (MgF_2) is used to improve heat transfer. Fluorination of a six-inch deep bed of 0.5-cm pellets (of an inert-fired type) was carried to completion in 6.2 hours at 500°C with inlet gas rate of 0.5 feet/second with the inlet gas containing 20 percent fluorine. The average uranium hexafluoride collection rate of 13 kg UF_6 /(hr) (sq ft of reactor cross-sectional area) amounted to 72 percent of the maximum rate based on fluorine introduction. Low collection rates were obtained in the first hour owing to formation of intermediate fluorides. Temperature control was good throughout. A decladding run with 10 percent hydrogen chloride-90 percent hydrogen fluoride was successfully carried to completion at 500°C ; the charge was 5 Zircaloy-clad fuel elements. Some temperature control problems were encountered during fluorination of another type of pellet (hydrogen fired). A modification of the pellet bed support (changed from nickel balls to a 1/4-inch thick disk perforated with 3/16-inch holes) eliminated this problem by providing additional fluid-bed heat transfer at the bed inlet. A successful run at 400°C with 20 percent inlet fluorine and with a three-inch bed gave an average initial rate of product collection of 3.6 kg UF_6 /(hr)(sq ft of reactor cross section). This type of pellet was fluorinated with very little formation of intermediates.

b. Heat Transfer in Packed-Fluidized Beds - Heat transfer tests are being continued in mockup systems similar to those encountered in uranium dioxide pellet fluorinations, that is, with a fluidized medium in the voids of a nonfixed bed of heated pellets.

Film coefficients for heat transfer from a single heater to the fluidized medium in the pellet voids, obtained for 3/8-inch pellets, were 75 percent of those for fluidization without pellets (165 vs 225) at a superficial gas velocity of 1.2 to 1.6 ft/sec. Heat transfer for this case appeared as good as or better than the heat transfer in a similar system with 1/2-inch pellets. Radial temperature gradients were found to be negligible under fluidized conditions. Experiments are in progress to determine the effective thermal conductivity of the pellets and the fluidized-bed system.

c. Reaction of Zirconium with Binary Halogen Gas Mixtures - It has previously been noted that hydrogen chloride has a positive catalytic effect

on the hydrofluorination of Zircaloy at 400°C over a limited concentration range. Above thirty weight percent hydrogen chloride the reaction is inhibited. Similar results have been obtained with zirconium at 500°C . Experiments have now been performed in which zirconium has been reacted with the following binary gas mixtures of various concentrations: hydrogen fluoride-chlorine, fluorine-chlorine, and fluorine-hydrogen chloride. The reaction of zirconium with hydrogen fluoride-chlorine and fluorine-chlorine mixtures proceeds to a lesser extent than the zirconium-hydrogen fluoride-hydrogen chloride reaction. However, the catalytic effect of the chlorine is similar to that of hydrogen chloride. With fluorine-hydrogen chloride mixtures there is no enhanced reaction.

d. Processing of Stainless Steel-Clad Fuel Elements - Additional experiments in the 1.5 inch diameter two-zone fluid-bed reactor investigating the decladding reaction of chlorine with stainless steel (followed by hydrofluorination) achieved about 75 percent dissolution of a 50-mil thick tubular specimen (1/2-inch long by 5/8-inch diameter) in three hours at 570°C . Results at higher temperatures (to 590°C) indicate complete dissolution would be expected in an equivalent period.

The use of a submerged nozzle with ammonia introduced in the center and the off-gas introduced through the annular space of a two-fluid nozzle has proved very efficient in a water scrubber for the chlorine-hydrogen fluoride off-gas stream. The procedure has eliminated the corrosion incurred previously in a nickel scrubber using a recirculating caustic solution.

The reaction of chlorine and carbon tetrachloride (separately and mixed) with sintered uranium dioxide pellets has been investigated in a series of tube-furnace experiments over the temperature range 500° to 600°C . This work is directed toward a possible chlorination-fluorination recovery scheme for power reactor fuels, making it possible to use HF, a comparatively inexpensive source of fluorine, for fluorinating uranium to UF_4 . Reaction rates (based on weight loss) were as follows:

<u>Gas Mixture</u> <u>(Volume Fraction)</u>	<u>Temperature Range (C)</u>	<u>Reaction Rate</u> <u>(mg/(sq cm)(hr))</u>
1/3 Cl_2 , 1/3 CCl_4 , 1/3 N_2	500 to 600	400 to 700
1/2 CCl_4 , 1/2 N_2	550 and 600	300 and 600
1/2 Cl_2 , 1/2 N_2	600	200

e. Conversion of Uranyl Fluoride to Uranium Dioxide - Recent batch, deep-bed (20-inch) experiments in a three-inch diameter fluid-bed reactor on the hydrogen reduction of uranyl fluoride to uranium dioxide achieved unexpected results: only "surface" reduction of the pellets occurred. Microscopic examination showed a black crust and yellow core (the original color of uranyl fluoride).

f. Plutonium Fluoride Studies - One aim of the current experimental program is to demonstrate steps in the direct fluorination process preliminary to pilot-plant scale operations. A mixture of plutonium and uranium tetrafluorides is to be fluorinated in the second fluorination step of the fluoride separations process. The resulting volatile hexafluorides are to be trans-

ported in the effluent stream to a condenser. Plutonium tetrafluoride has been reacted with fluorine at 550°C. The resulting plutonium hexafluoride in the exit gas stream along with the excess fluorine was passed through a heated vessel held at 250°C in one run and 145°C in a second run. In a separate experiment, pure plutonium hexafluoride vapor at a constant pressure of 19 mm was circulated through the heated vessel at 250°C. The amount of decomposition of plutonium hexafluoride that occurred in the first two experiments was 27 percent in 24 hr at 250°C and 0.4 percent in 24 hr at 145°C.

Data have been obtained on fast neutron emission and beta-gamma dose rates from samples of plutonium tetrafluoride. A typical dose is 49 n/(sec)(sq cm) at 8 inches for a 77-gram sample in a polyethylene bottle enclosed in a 20 mil polyvinyl chloride bag, 150 mr/hr hard and soft beta at 2 inches, and 60 mr/hr hard gamma at 2 inches.

g. Separation of Ruthenium from Uranium in the Direct Fluorination Volatility Process - A program is in progress to obtain information concerning the separation of fission products from uranium and plutonium in the Direct Fluorination Volatility Process. The relative rate of fluorination of ruthenium with respect to uranium dioxide under process fluorination conditions is under investigation. At both 400°C and 500°C ruthenium was fluorinated and volatilized out of the reaction mixture with the uranium. The same results were obtained when calcium fluoride and zirconium tetrafluoride were added to the mixture.

2. Chemical-Metallurgical Process Studies

Continued progress is being made in the development of new pyrometallurgical methods for reprocessing nuclear reactor fuels. Supporting studies are providing physicochemical data on liquid metal systems and thermochemical data on materials of interest to high-temperature chemistry.

a. Liquid Metal Solvent Studies - The solubilities of the rare earths holmium and ytterbium in cadmium may be represented by the empirical equations

$$\text{Holmium (325}^{\circ}\text{ to 628}^{\circ}\text{C)} : \log (\text{atom percent Ho}) = 3.409 - 2494 T^{-1}$$

$$\text{Ytterbium (325}^{\circ}\text{ to 504}^{\circ}\text{C)} : \log (\text{atom percent Yb}) = 2.992 - 1957 T^{-1}$$

Additional data for the solubility of cobalt in liquid cadmium over the range from 334°C to 653°C have been obtained. The solubilities ranged from 1.67×10^{-3} weight percent at 334°C to 1.80×10^{-2} weight percent at 653°C. These results are in good agreement with values previously reported.

Additional points on the liquidus surfaces of the zinc-magnesium-uranium ternary system have been determined. A peak solubility of uranium was found to be 22.1 weight percent at 825°C at a zinc to magnesium weight ratio of 9.41. The peak solubility is shifted to higher zinc to magnesium ratios at higher temperatures and to lower ratios at lower temperatures. X-ray diffraction examinations of ingots from these experiments showed that the zinc-uranium intermetallic phase which crystallized was the delta phase (tentatively considered to be $\text{UZn}_{8.5}$) rather than the epsilon phase ($\text{UZn}_{11.5}$).

The coprecipitation coefficient, λ , for several elements (sodium, lithium, potassium, yttrium, lanthanum, uranium, strontium, europium, and zirconium) when carried by the intermetallic compound CeCd_{11} have been previously reported in the Progress Reports for October (ANL-6253) and November (ANL-6269). The following additional values of λ have been obtained: samarium, 0.17; scandium, 0.05.

The influence of bismuth on the distribution of uranium between liquid zinc and lead has been measured. Values of the distribution coefficient at 659°C are as follows:

<u>Distribution Coefficient</u>	<u>Concentration of Bismuth</u>
$= \frac{\text{(w/o U in Zn-rich phase)}}{\text{(w/o U in Pb-rich phase)}}$	$\frac{\text{in Lead-rich Phase}}{\text{(weight percent)}}$
208	0
36.8	8.9
5.4	22.7
2.1	29.7

The influence of magnesium on the distribution of uranium between liquid zinc and lead was measured. Upon the first addition of 6.5 weight percent magnesium at 650°C the system became completely miscible.

Measurements of the magnetic susceptibility, X , of the neodymium-cadmium intermetallic compound NdCd_{11} showed that the Curie-Weiss equation

$$X = \frac{C}{T - \Delta}$$

is followed with $\Delta = -17.5$ K and $C = 1.74$. The effective Bohr magneton number for neodymium in this compound is 3.73 which is in good agreement with the theoretical value of 3.68 for the Nd^{+3} ion in a $^4J_{9/2}$ ground state.

b. Calorimetry - A preliminary value of -71.28 ± 0.26 kcal/mole has been obtained by means of oxygen bomb calorimetry for the heat of formation of tungsten disulfide.

A preliminary value for the heat of formation of boron trifluoride has been determined. The value obtained, -269.63 ± 0.32 kcal/mole, agrees with the value of -270 ± 2 kcal/mole (NBS Report No. 6252) which heretofore had been considered questionable.

Exploratory combustions of boron nitride, uranium, and thorium have been made using the fluorine bomb calorimeter, and a series of combustions of uranium mononitride have been carried out in the oxygen bomb calorimeter.

F. Advanced Reactor Development

1. High Power Density Fuel Element Design Studies

a. Hydraulic Test Loop - As part of a general investigation of high power density fuel elements, parallel plate fuel assemblies are undergoing flow tests to establish the hydraulic limits for this type of element.

The first prototype fuel element to be tested contained 1.25 mm thick plates made from aluminum alloy 1100-H14. This alloy was selected in order to approximate the "hot" mechanical properties of a fuel plate without the use of a heat source in the loop. To increase flow in the corners of the rectangular coolant channels and to equalize the static pressures between channels, the conventional fuel box side plates were eliminated. Instead, the fuel plates were fastened together with discrete keys to provide support at the edges.

After an average flow velocity of 11.89 m/sec was reached in the test box, failure occurred by buckling of several plates. This initial experiment seems to confirm a calculated velocity of 12.59 m/sec for critical divergence with no cross flow or pressure equalization; however, the coolant conditions were close to boiling (140°C at 24.46 bars absolute) so that local flow blockage may have contributed to the failure. To help pinpoint the mechanism causing failure, the next series of tests will be run at lower coolant temperatures to eliminate the possibility of local boiling. In addition, the test section is being fitted with a window to permit continual visual monitoring of flow within the fuel element.

b. Atmospheric Pressure Pool Boiling Studies - Grid-mapping of the voltage distribution in the vicinity of the power terminals attached to the vertical core plates used for atmospheric-pressure pool boiling burn-out tests has indicated a current density distribution ratio of 7 to 1. Heat generation in the plates is in the same maximum-to-average ratio, but the occurrence of edgewise cooling and the thermal sink effect of the power terminals reduces the maximum-to-average heat transfer to an estimated 4 to 1 ratio. These values were arrived at by use of a graphical orthogonal squares technique, employing iso-voltage parameters from the voltage grid measurements. The maximum-to-average heat transfer ratio estimate is based primarily on the approximated current density in the known location of the burnout occurrence.

c. Fuel Element Design Studies - The results of the first flow tests indicate that the approach to higher fuel element performance through further subdivision of the fuel (thinner plates) is reaching a limit. If open-sided fuel boxes do not prove successful, the critical divergence velocity can be raised by a factor of four by cutting the unsupported width of the fuel plate in half. One technique considered for strengthening the box is the extrusion of longitudinal support ridges in the fuel plate cladding. Heat transfer calculations are under way to determine the hot spot effect of various plate support structures.

d. Analysis of System Behavior - A study was made of some general conditions for minimizing the maximum wall and maximum fuel centerline temperatures in a heterogeneous reactor core channel with no coolant phase

change. An argument was developed which shows the requirement for constant wall and fuel centerline temperatures even under the general condition of the heat transfer coefficient as some arbitrary but reciprocally integrable function of the bulk fluid temperature. Comparisons were made with the previously known conditions for an assumed constant heat transfer coefficient. An explicit expression was developed for the "idealized" surface heat flux distribution to minimize the maximum wall temperature for given inlet and outlet coolant temperatures. A possible "measure of performance" was also indicated for comparison purposes. For the case of minimizing the maximum fuel centerline temperatures under the same conditions an implicit expression was obtained and it is shown certainly that both minimizations cannot be obtained at the same time. An upper bound was also described for the difference between the fuel centerline and surface temperatures under the "idealized conditions."

The dynamic loop test failure of the AHFR 50-mil fuel box was examined. During the test, the fuel box may have been incorrectly mounted in the test holder and local boiling may have occurred. Thus the test was inconclusive. A new fuel element design with the width cut in half was suggested.

2. AHFR Critical

The treatment of the observations made during the final months of the AHFR critical experiments is essentially complete. The results of the temperature coefficient measurement for the second core yielded a value of $-0.01 \times 10^{-2} \text{ } \rho/^{\circ}\text{C}$, but the observations did exhibit some unexplained shifts. It was found that the worth of the coarse control rods decreased by about 0.5-1% of the total value for each $^{\circ}\text{C}$ in temperature over the range from 20-60 $^{\circ}\text{C}$. The replacement of the 90% beryllium-10% lucite reflector by light water caused a reactivity loss of 15.5%. This should be compared with a preliminary value of 12% given in the Progress Report for October, 1960 (ANL-6253). Type 304 stainless steel in the core had an average poisoning effect of $0.38 \times 10^{-5} \text{ } \rho/\text{gm}$.

Critical experiments measured the worth of the beryllium reflector in the fully loaded and poisoned core. Replacement of the beryllium by a light water reflector resulted in a decrease of 18% reactivity to the system. The reactivity added by filling the flux trap with fuel elements is about 12%. Thus a core which is close to critical for a flux trap reactor design could safely be assembled in a water tank as long as safeguards are observed which prohibit addition of fuel to the flux trap region, especially if the core elements are not fully poisoned with stainless steel. The total worth of stainless steel poison strips can amount to as much as 13% reactivity.

3. Fast Reactor Core Test Facility

A study has been started to prepare a conceptual design for a universal test facility in which advanced fast reactor cores would be evaluated experimentally. Ideally, the facility is to provide the flexibility of the typical critical experiment and at the same time permit operation at high power in order to cover a wide range of reactor design parameters. A heat sink capacity of 50 Mw in conjunction with a core vessel suitable for a variety of lattice configurations is visualized. A survey of advanced fast reactor concepts is under way to establish operational requirements for the test facility.

4. Meteorological Application

In conjunction with the design and application of high power density fuel elements, the possibility of the use of nuclear power for weather modification was considered. One study concerned the use of an air-borne, nuclear heat source to modify cloud flight patterns through buoyancy control in order to suppress local rainfall for delivery to a more desirable site. Preliminary estimates indicate that a compact, nuclear heat source may be economically attractive for applications such as drouth relief.

5. Lunar Power Plant

A conceptual design of a power plant using sodium vapor as the working fluid has been started. In addition, some work is being done on trying to estimate the composition of the lunar atmosphere.

The general category of compact reactors is being studied in order to determine a minimum weight and a minimum volume power course. Such studies could also be useful in designing future space, land-mobile, and marine craft. General considerations are being studied for power plant heat removal in a space environment.

6. Direct Conversion Studies

The cesium cell which previously suffered a leaky tantalum diaphragm has been put back into operation. A preliminary estimate of radiation loss from the UC-ZrC cathode material to the copper anode gives results which are reasonable using the method described recently.

A new cell was assembled and the experimental work is in progress. The emitter contains UC:ZrC in a wt-% ratio of 40:60 i.e., the uranium has a higher concentration than was used previously.

In the first test a retarding electrical field was applied to determine the temperature and results are being compared with those of pyrometer readings. The experimental work will include determination of Richardson's constants, and application of magnetic fields in vacuum and plasma.

VI. PUBLICATIONSPapers

GLASS COMBUSTION BOMB

R. Nuttal, M. A. Frisch, W. N. Hubbard

Review of Sci. Inst. 31, 461 (1960)ENERGY DIFFERENCE BETWEEN THE BOAT AND CHAIR FORMS OF
CYCLOHEXANE

W. S. Johnson, J. L. Margrave, V. J. Bauer, M. A. Frisch,

L. H. Dreger and W. N. Hubbard

J. Am. Chem. Soc. 82, 1255 (1960)ALLOYS OF TANTALUM DIBORIDE WITH IRON, COBALT, AND
NICKEL

H. W. Lavendal

Plansee Berichte Anniversary Issue (in press)

TRENDS IN NUCLEAR REACTOR CONTROL

W. C. Lipinski and J. M. Harrer

IRE Transactions on Nuclear Science, Vol. NS-7, No. 4,
December, 1960

TRANSIENT THERMAL STRESSES IN A CIRCULAR CYLINDER

C. Youngdahl and Eli Sternberg

(Presented at the ASME Winter Meeting in New York,
November 28, 1960)

The following papers were presented at the winter meeting of the American Nuclear Society in San Francisco, California, December 12-14, 1960:

STUDY OF FAST REACTOR FUEL ELEMENT BEHAVIOR UNDER TRANSIENT
CONDITIONS IN TREAT

C. E. Dickerman, E. Sowa and D. Okrent

DESIGN FEATURES OF THE JUGGERNAUT REACTOR

J. Folkrod

OPERATING TESTS OF THE EBWR VAPOR RECOVERY SYSTEM

R. J. Gariboldi and D. R. Jacobson

EBR-II SHIELD DESIGN

M. Grotenhuis, A. E. McArthy and A. D. Rossin

EBWR SPIKED CORE FUEL ELEMENTS

V. M. Kolba

EBWR 100 MW PLANT

J. F. Matousek

A DESIGN CONCEPT OF A FAST FUEL TEST REACTOR (FFTR)

A. Smaardyk, R. Brubaker, H. H. Hummel, J. H. Kittel, and

A. McArthy

Papers (cont.)

A LUNAR POWER PLANT

J. C. Carter, R. H. Armstrong, M. J. Janicke, H. H. Hummel
and J. F. Marchaterre

NUMERICAL SOLUTION OF THE REACTOR KINETICS EQUATIONS

J. Kaganov

PRELIMINARY DESIGN OF A BOILING SLURRY REACTOR EXPERIMENT

G. A. Freund, J. D. Lokay, J. F. Marchaterre, G. C. Milak
and M. Petrick

RADIATION DAMAGE IN STEEL: CONSIDERATIONS INVOLVING
NEUTRON SPECTRA

A. D. Rossin

MEAN SQUARE INSTABILITY IN BOILING REACTOR

A. Z. Akcasu

CALCULATION OF HETEROGENEITY EFFECTS IN FAST ASSEMBLIES

D. Meneghetti

POSSIBLE EFFECTS OF SCALE DEPOSITS ON FUEL PLATES OF EBWR
CORE DESIGN

H. P. Iskenderian, C. F. Reinke, and I. Charak

COUPLED FAST-THERMAL CORE FOR EBR-II REACTOR

G. Fischer

REMOTE ASSEMBLY OF REPROCESSED FUEL SUBASSEMBLIES FOR
EBR-II

R. Olp

ANL ReportsANL-6136 WATER CHEMISTRY AND FUEL ELEMENT SCALE
IN EBWR

C. R. Breden, I. Charak and R. H. Leyse

ANL-6141 THE DRIP CASTING OF ZIRCONIUM METAL

R. J. Dunworth and R. E. Macherey

ANL-6218 CALCULATION OF HETEROGENEITY EFFECTS IN
ZPR-III FAST ASSEMBLIES USING THE DSN PROGRAM

D. M. Meneghetti and M. F. Loomis

ANL-6221 THEORETICAL FEEDBACK ANALYSIS IN BOILING
WATER REACTORS

A. Ziya Akcasu

ANL-6237 THE CALCULATION OF TRANSIENT TEMPERATURE
DISTRIBUTIONS IN A SOLID CYLINDRICAL PIN,
COOLED ON THE SURFACE - IBM-704 PROGRAM
524/RE-147

J. Heestand, D. F. Schoeberle and L. B. Miller

ANL Reports (cont.)

ANL-6267 DESIGN AND MASS TRANSFER STUDY FOR NEW MULTI-STAGE FLUIDIZATION REACTOR
K. D. Williamson

The following papers appeared in TID-7599, Proceedings of the Eighth Conference on Hot Laboratories and Equipment, San Francisco, California, December 13-15, 1960:

ALPHA-GAMMA CONTAINMENT BOX
C. H. Youngquist, J. F. Lindberg, W. C. Mohr and R. B. Wehrle

SIMPLIFIED FILTER CHANGE WITHOUT AN INTERRUPTED OR UN-FILTERED EXHAUST
R. A. Blomgren