ANL-6269
Reactors - General
(TID-4500, 15th Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

November, 1960

N. Hilberry, Laboratory Director

<u>Division</u>	<u>Director</u>
Chemical Engineering	S. Lawroski
Idaho	M. Novick
Metallurgy	F. G. Foote
Reactor Engineering	B. I. Spinrad
Remote Control	R. C. Goertz

Report coordinated by R. M. Adams

Issued December 15, 1960

Operated by The University of Chicago
under

Contract W-31-109-eng-38

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

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

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.

Previous reports issued in this series:

August, 1960	ANL-6215
September, 1960	ANL-6234
October, 1960	ANL-6253

TABLE OF CONTENTS

	<u>Page</u>
I. Water Cooled Reactors (040101)	1
A. General Research and Development	1
1. Heat Transfer and Fluid Flow Studies	1
2. Ultrasonic Level Transducer Evaluation	2
3. Boiling Slurry Reactor Experiment (SLURREX)	2
B. EBWR	2
1. 100 Mw Modifications	2
2. Examination of Original Control Rods	3
3. Irradiation Specimens	4
4. Core II	4
5. Separate Superheater Concept	4
C. BORAX V	4
1. Construction	4
2. Procurement, Fabrication and Installation	5
3. Design	6
4. Development and Testing	7
II. Sodium Cooled Reactors (040103)	9
A. General Research and Development	9
1. Fast Reactor Physics	9
2. ZPR-III	10
3. ZPR-VI and IX	12
4. Juggernaut	13
B. EBR-I	13
1. Operation	13
2. Mark III Stability Studies	14
3. Mark IV Development and Fabrication	17
C. EBR-II	18
1. Construction	18
2. Installation of Equipment - Package 4	19
3. Engineering	25
4. Procurement	25
5. Component Development - Instrumentation	25
6. Component Development - Reactor	26
7. Component Development - Steam Generators	28
8. Component Development - Fuel Reprocessing Facilities	28
9. Fuel Development and Fabrication	29
10. Fuel Prototype Irradiation	30
11. Physics	30
12. Process Development	31

	<u>Page</u>
III. Studies and Evaluations (040116)	34
A. Fast Reactor Fuel Value Study	34
B. Nuclear Superheat Studies	35
C. Miscellaneous	35
IV. Reactor Safety (040117)	36
A. Thermal Reactor Safety Studies	36
1. Fuel-Coolant Chemical Reactions	36
2. Kinetics of Oxidation and Ignition of Reactor Materials	36
B. Fast Reactor Safety Studies	38
1. TREAT - Experimental	38
2. Out-of-Pile Meltdown Studies	39
3. Meltdown Configuration Studies	40
4. TREAT - Operation and Development	40
V. Nuclear Technology and General Support (040400)	41
A. Applied Nuclear and Reactor Physics	41
1. Experimental	41
2. Theoretical	46
B. Reactor Fuels Development	47
1. Ceramic Fuels	47
2. Properties of Metals and Alloys	49
3. Corrosion Studies	49
4. Irradiation Studies	50
5. Nondestructive Testing Techniques	51
C. Reactor Materials Research and Development	52
1. Irradiation Damage in Steel	52
2. Structural Properties of Steels	53
3. Measurement Techniques	53
4. Structures	54
D. Reactor Components Development	54
1. Development of Viewing Systems	54
2. Instrumentation and Control	55
3. Heat Engineering Studies	55
E. Separations Development	56
1. Fluidization and Fluoride Volatility Separations Processes	56
2. Chemical-Metallurgical Process Studies	58

	<u>Page</u>
F. Advanced Reactor Development	59
1. High Power Density Fuel Element Design Studies	59
2. Basic Radiation Effects Reactor	60
3. Direct Conversion Studies	60
4. Fast Fuel Test Reactor (FFTR)	61
5. Lunar Power Plant	61
VI. Publications	62

I. WATER COOLED REACTORS (040101)

A. General Research and Development

1. Heat Transfer and Fluid Flow Studies

a. Steam-Water Separation Studies - Studies are being continued on the problem of steam carryunder in the downcomer of a natural circulation boiling reactor. Data were obtained from a set of experiments in the air-water loop on a system having a smaller geometry than that used in the previous work. A reduction of approximately 10% in the carryunder was obtained in the smaller system for similar conditions of flow and quality. However, it is not completely clear that the reduction in carryunder rate was caused by the change in geometry. The possibility that the separator in the downcomer may have been functioning with a greater efficiency in the smaller system because of reduced throughput is currently being investigated.

b. Hydrodynamic Instability - Measurements are being planned of the power-to-void transfer function for an electrically heated test section in an out-of-pile test loop. A 1.27 m long rectangular test section is to be inserted in the small scale 135 atm (2000 psi) loop. The effect of sinusoidal power variations on steam bubble density inside the test tube at different heights will be measured as a function of oscillation frequency.

The spectrum of the thulium-170 gamma source to be used in the experiment has been measured. It has been found that most of the gamma energy emitted is due to bremsstrahlung in the energy spectrum 100 to 300 kev. Using a lead filter only has the adverse effect of cutting down the intensity, and does not help to bring out the 84-kev line of the thulium spectrum. Nevertheless, it was found that the detector reading for the bare source falls off according to a single exponential absorption constant with decreasing void fraction.

All the electronic components for the sinusoidal power control, and for the loop instrumentation, have now been built and tested separately. These components are now being put in place by the loop and connected to the primary measuring devices.

c. Burnout Tests - Experiments will be run on a test section of an EBWR fuel element to determine burnout heat flux limitations. Burnout detection is provided by a d.c. burnout detector and by a temperature burnout detector. The first instrument responds to a resistive unbalance in the test section due to a temperature excursion; the second instrument will shut off power when the temperature near the heated exit exceeds a predetermined level. The temperature profile along the test section will be scanned by a mercury-jet switch.

A second detector stage was added to the thermocouple burnout detector so that either or both of two thermocouple outputs will trip power during temperature transients. Further progress was made on temperature scanning of the test section with power applied.

2. Ultrasonic Level Transducer Evaluation

An ultrasonic level transducer was received from the AEC for evaluation. The purpose of the evaluation is to determine the ability of this transducer to detect a steam-water interface level and to determine its reliability when exposed to a high temperature, nuclear environment.

An attempt was made to measure the output impedance of the level transducer as a function of two-phase mixture density. After a relatively short period of room temperature testing, however, the transducer became inoperative. This failure appears to be the result of water entering the sensing head through the pressure seals. Tests of a second transducer should be definite evidence of leaks at the pressure seals. Both transducers will be returned to the manufacturer for modification or repair.

3. Boiling Slurry Reactor Experiment (SLURREX)

The Laboratory has been officially informed of the Commission's decision not to support SLURREX. A draft of the terminal report on the reactor engineering aspects of the project has been completed. This report, ANL-6248, will be published this coming spring as will a report on slurry fuel development.

Construction of the half-scale mockup is completed. The terminal report on the mockup experimental program will be completed by the end of the fiscal year.

B. EBWR

1. 100-Mw Modifications

The two primary reboilers from the reboiler building were removed and shipped to the vendor for repair of the leaks. With the return of the primary reboilers to the vendor, many temperature and level instruments connected to primary reboilers have been removed from the EBWR system. These instruments will require reinstallation when the reboilers are returned to the site.

Wiring of the boric acid addition system has been completed except for the indicator panel which will be finished as soon as the required instruments are received. The conductivity recorder has been converted to a 12-point recorder and new cells connected with the clean-up system have been installed and checked out. The reference leg level pot on the deaerator has been raised so that the full 100 in. of the transducer may be utilized. Recalibration will now be required.

As-built drawings for the electrical system have not been brought up to date, as was erroneously reported in the Progress Report, October, 1960, ANL-6253. The Architect-Engineer has been apprised of the necessity for updating these drawings concurrently with the completion of the work.

a. EBWR Instrumented Assemblies - The new clips that locate the in-core flux monitors between full plates of the instrumented assemblies were fabricated and autoclaved. One of the flux monitors is now installed in the fuel assembly and the other is in the process of being installed.

b. Neutron Source - A fission counter and ion chamber were installed in the EBWR core to observe the change in source level during the installation of the new source. The reactor counters indicated an increase by a factor of about 1500, and the indicator on the linear flux just about doubled.

c. Void Simulators - The rectangular tubing for experimental void simulators was received November 10, 1960. Considerable denting of the sides had occurred and will require straightening prior to assembly into the header box.

d. Storage Pit - Drawings have been completed for the new fuel storage rack. The fuel inverting fixture is being installed in the pit. The hydraulic cutter to be used for sectioning irradiated control rods was received.

e. EBWR 100-Mw Oscillator - An adjustment was made to the oscillator to permit operation at the full high speed stroke of ± 2.5 cm. The oscillator was operated at this stroke at speeds up to 9 cps with the control rod at an elevation of 25 cm prior to its removal from the reactor. The oscillator will be stored in the reactor sub-basement until needed for transfer function measurements.

f. Containment Shell Leak Rate Test - A leak rate test of the EBWR containment shell was conducted during the period October 11, 1960 - October 13, 1960. The average leak rate obtained from this test was 946 ft^3 of air at STP/24 hr, which is within the allowable rate of 1000 ft^3 .

g. Operator Training - All valves, connections for pressure gauges and temperature indicators, flow meters (except piping), etc., have been depicted on the EBWR main steam and feedwater piping display and training board.

Lectures to reactor operating personnel have continued. All operators are being given instruction and experience on the EBWR simulator.

2. Examination of Original Control Rods

When the original EBWR control rods were removed from the reactor (September, 1959), No. 9 rod was found parted at the spot-welded lap transition between its hafnium and Zircaloy-2 portions. Examination of pieces (see Progress Report, October, 1960, ANL-6253) reveals that the failure was caused by dynamic forces imposed during rod oscillation to obtain power transfer functions.

Examination of No. 4 control rod shows a partial failure of the riveted transition between its 2% boron-stainless steel and Zircaloy-2 portions. The countersunk heads on one end of over 90% of the rivets had popped off allowing the 0.16-cm thick Zircaloy-2 side sheets to open up causing a wedging action in the core shroud guides. Fortunately lateral forces imposed during control rod scram tended to close the wedge. The cause of rivet failure is being investigated.

An accumulation of radioactive crud was found in pockets at the bottom ends of No. 4 and No. 9 control rods. The crud had originated from the core region, had silted through slight opening between adjacent angle faces to the center hollow stem, and had then settled to the bottom. An analysis currently being made shows the crud includes the nuclides Co^{58} , Co^{60} , Mn^{54} , Zn^{65} , and Cs^{137} .

3. Irradiation Specimens

Three Type 304 stainless steel dosimeter capsules have been loaded into the EBWR vertical tube irradiation facilities preparatory for the forthcoming reactor operation. Each capsule is pressurized with helium at 35.0 atm and contains the following detector materials: aluminum sulfate, bare and cadmium-covered gold foil, cadmium-covered natural uranium foil, bare thorium foil, bare and cadmium-covered gold wire, bare and cadmium-covered cobalt wire, and bare and cadmium-covered nickel wire.

4. Core II

Drawings have been completed for a Core II fuel element design and are being checked. Physics and heat transfer calculation results have not yet been completed; therefore, no definite commitment can be made on enrichment, rod diameter, or pellet size.

5. Separate Superheater Concept

A study was initiated on the relative merits of separate nuclear superheat for reactor power plants. The parameters involved in the design of such systems will be investigated. A flow diagram was prepared to indicate what would be involved in adding a separate superheater to the EBWR complex.

C. BORAX V

1. Construction

Installation of the reactor vessel was delayed until November 15, 1960, primarily because of defective thermocouples for measuring reactor vessel temperatures. Before the vessel was set in place the thermocouples and stainless steel wool insulation and jacket on the vessel shell and bottom head were installed. The vessel has been checked for alignment, plumb, and elevation and is now in its final position. Tie-in of the forced-convection system piping to the vessel nozzles is proceeding.

Process piping in the east trench is essentially complete. Pipe-fitting and welding in the steamline trench, access shaft, and subreactor

room is still in progress.

After installation of the reactor vessel the roof of the reactor building was completed and the roof ventilators and wainscoting inside the building have been installed. ANL has beneficially received the reactor building heating unit for operation.

Installation of conduit and wiring trays in the reactor building is well advanced and the running of control and power and lighting wiring is proceeding.

The construction contract was approximately 88% complete on November 30, 1960.

2. Procurement, Fabrication and Installation

Twelve sample boiling fuel rods of normal UO_2 clad with stainless steel and 48 normal UO_2 pellets, have been received from the supplier. Both the pellets and the rods are being evaluated. The UO_2 density per pellet is within specifications but the total weight of UO_2 per rod is higher than specified. After a brief study of the effects of an increased loading, the fuel rod weight specifications were changed from $371.5 \pm 2.5\text{g}$ to $374 \pm 2.5\text{g}$. Some of the sample boiling fuel rods exceed the newly established weight tolerance, and some also do not meet the tolerance for straightness.

A strike at the supplier's plant made it impossible to meet the original scheduled completion date of November, 1960 for delivery of the boiling fuel. However, the strike has now been settled and it is expected that 100 rods per day will be produced beginning December 12, 1960. No enriched rods have yet been produced.

The scheduled delivery of superheater fuel plates from Atomics International remains unchanged. Evaluation plates are due December 26, 1960. Assuming a maximum of two weeks for evaluation and acceptance, delivery of developmental plates and fuel plates would commence the week of January 9, 1961, and be completed by April 10, 1961.

Components of the three core structures (boiling core, central superheater, peripheral superheater) will be fabricated by commercial vendors by the U. S. Naval Ordnance Plant, and by the Laboratory's Shops with assembly to be done by the Laboratory. The boiling core structure should be delivered by March 1, 1961, and the two superheating core structures by April 1, 1961. Planning, estimating, and procurement for other miscellaneous core items such as control rods, chimneys, holddown boxes, flood and drain manifolds, etc. has been started.

All components for the modification of the old EBWR control rod drives are on hand, and assembly and testing is proceeding. Delivery is expected in late December.

In the control building essentially all components for the reactor control console and recorder panel are on hand and installation of these

items is about 60% complete. Installation and wiring of the process and superheater temperatures reference junctions, and switching circuits in the temperature control cabinet were completed and turned over to the contractor for installation in the reactor building.

At the contractor's request the existing make-up water demineralizer system was reinstalled in the reactor building by Laboratory personnel. The water from this system will be used for hydrotests on process piping and equipment.

3. Design

Detailed design drawings of the following items have been completed and sent to the Shops for fabrication: core structure assemblies, superheated steam plenums, superheater flood and drain manifolds, thermal-sleeve shield plugs for control-rod-drive nozzles, superheated steam exit manifolds, void and water rods, void and water tubes, stainless steel poison rods and boron-stainless steel poison rods. Detailed design has also been completed on the thermopile assembly for measuring downcomer temperatures and the capsules and brackets for in-vessel irradiation of reactor vessel steel samples. Detailed design has been started on the sample panel for water chemistry control and analysis. Design work on an in-vessel underwater periscope for viewing during fuel handling operations with the reactor vessel head "on" has been started. Preliminary design of a grappler for handling the superheater fuel assemblies is complete.

A preliminary analysis has been carried out to investigate the feasibility of revising the design of the spacers between the superheater fuel plates. It is proposed that the 1/8 in.-diameter, stainless steel disc spacers be replaced with longitudinal wires. The purpose of this proposed change is to reduce the cost and the complexity of the fabrication. Calculations indicate that 1/16 in. square wire between fuel plates is preferable to round wire because the ratio of the exposed surface area in the wire to the area of the fuel plate covered by the wire is about 1:1 in the case of square wire as opposed to a ratio of 10:1 for round wire. The difference in operating temperature between the wire and fuel plate is therefore much smaller in the case of square wire, and the thermal stress at the brazed joint between square wire and plate is allowable. A 1/32 in.-diameter round wire scalloped to reduce heat loss may be used between the outer fuel plates and the insulating tube. A problem has been submitted for machine calculation to verify the temperature distribution in wires and fuel plates.

Calculations were made which establish that the required neutron source strength for BORAX V is about 400 curies. The dimensions of the Sb-Be source have been selected and a preliminary mechanical design has been completed.

As a part of the hazards analysis of BORAX V, studies have been continued on both statics and kinetics problems. A quantitative problem has been set up in which all of the heat remains in the UO_2 and the only shutdown mechanism is Doppler broadening of U-238 resonances. The kinetics code RE-129 is being modified in order to evaluate the Doppler shutdown mechanism. UNIVAC results were received on problems relating

to $1/v$ poison and analysis of these data gives an average prompt neutron lifetime of 24 microseconds. Results of perturbation theory problems to determine prompt neutron lifetimes and effective delayed neutron fractions indicate, for the various cores and conditions, that the prompt neutron lifetime is in the range of from 23 to 40 microseconds. Results of PDQ problems indicate that shutdown reactivity is adequate for all three cores in the cold condition with all control rods "in."

A preliminary experimental program for BORAX V has been prepared. The program covers tests and experiments under five general headings:

1. Plant Tests; 2. Central Superheater Experiments, 3. Peripheral Superheater Experiments; 4. Boiling Reactor Stability Experiments; and 5. Advanced Experiments.

Work is proceeding on the Final Design and Hazards Report. The major work remaining is on hazards evaluation.

4. Development and Testing

Development work on the fabrication of superheater fuel assembly is continuing. Equipment for preparation of metallographic specimens has been installed, and equipment for microscopic examination of specimens containing enriched uranium has been ordered. With the exception of a test for surface halide ion contamination, all nondestructive testing equipment has been readied. Complete evaluation of the testing procedures cannot be done until the first plates are received from Atomics International.

Metallographic examination has been performed on welded and brazed joints taken from an experimental assembly supplied by the Laboratory's Shops. In the specimens examined, a satisfactory bond was achieved by the welding method. Since the plates used in the experimental assembly are solid stainless steel, there is no positive proof that the heat affected zone will not penetrate into the core of the actual fuel plates. Examination of side plate-to-plate, and spacer-to-plate bonds has shown that effective brazing can be achieved if good contact is maintained at all the joints during the furnace brazing cycle.

Tensile tests were performed on four welded and brazed sections of this assembly. In each case, failure of the assembled sections occurred at a brazed interface. The values obtained ranged between 47,600 and 51,750 psi. Special grips are being made to permit the tensile testing of side plate-to-plate bonds.

A mockup of a complete instrumented superheater fuel assembly has been essentially completed. Drawings of a superheater fuel element redesigned to use wire spacers have been sent to the Shops for fabrication development.

Samples of two superheater fuel plate thermocouples were successfully Microbrazed at 1850°F into simulated fuel plates. Calibration of these brazed couples against standard couples up to 1500°F was within the standards set by the National Bureau of Standards.

Samples of boiling fuel rod thermocouples from two suppliers have undergone first calibration and resistance tests at the Hoskins Company in Detroit. The samples were calibrated to 4100° F but preliminary data indicate that it will not be possible to use a standard calibration curve since the curve changes above 3000° F because of decreased resistance in the insulation. The resistance tests indicate that BeO maintains its insulating properties at higher temperatures than MgO.

After modification of the air-water loop to reduce system vibrations, a dummy boiling fuel rod in a 49-rod assembly was tested for vibration. Maximum amplitude measured under all conditions up to 600 gpm and 25 percent voids was 0.002 in.

II. SODIUM COOLED REACTORS (040103)

A. General Research and Development

1. Fast Reactor Physics

a. Performance Evaluations - Critical mass, spectrum, breeding ratio, and coolant removal coefficients have been calculated for a series of large, fast breeder, power reactors using a new 16-group cross section set,* which reflects the effect of most of the recent microscopic cross section measurements. The parameters studied include reactor size, plutonium isotopic content, and type of structural material. Reactors cooled with lead-bismuth eutectic were also examined, as were those fueled with a thorium-U-233 mixture. Somewhat higher breeding ratios for U-238-plutonium fueled fast reactors are now obtained than were calculated in 1958. This effect is attributed primarily to the greater number of neutrons per fission now believed appropriate for U-238. The breeding ratios for the thorium-U-233 cycles are about the same as in 1958, falling in the range between 1.2 and 1.3 for carbide, oxide, or metallic-fueled systems having steel cladding.

The addition of higher isotopes of plutonium to the fuel raises the breeding ratio and apparently makes the sodium coefficient more negative (or less positive). However, with either steel or niobium cladding in a plutonium-fueled reactor, the sodium coefficient becomes more positive as reactor size is increased and can attain a positive value. Negative sodium coefficients were calculated for all core sizes in thorium-U-233 systems having steel cladding. Negative coolant coefficients were also calculated for plutonium-U-238-steel systems using lead-bismuth eutectic as coolant.

For plutonium-uranium metal-fueled reactors, a positive sodium coefficient was found for a niobium-clad core of approximately 400 liters volume. The coefficient became positive at 800 liters with a steel clad core. The core composition in each case was taken as 25% cladding and structure, 25% fuel alloy, and 50% sodium by volume. The threshold size for a positive sodium coefficient can be expected to increase if the volume fraction of structure or fuel alloy is reduced, with a corresponding increase in the sodium volume fraction.

Since niobium, molybdenum, and tantalum can all be expected to exhibit positive sodium reactivity coefficients in relatively small cores, there is a possibility that fuel element development in the metallic alloy field might best be aimed at other cladding materials, such as vanadium, yttrium, zirconium, or titanium. For the ceramic systems stainless steel is of interest as well.

b. Shape Factors - The shape factor is generally defined to be the ratio of the critical mass of a system in spherical geometry to the critical mass of the identical system in cylindrical or slab geometry. Experimental information at present is available only on the small high density systems studied at Los Alamos and those of EBR-II size studied at ZPR-III. Past experience has demonstrated that diffusion theory is adequate for the prediction of shape factors for EBR-II size systems; hence, theory has been used to study this effect in some much larger reactors more typical of eventual central station designs.

*Yiftah, Okrent, Maldauer, "Fast Reactor Cross Sections - A Study Leading to a 16-Group Set," Pergamon Press (to be published).

Two distinct reactor systems were investigated. One, a relatively high density uranium system, had a core volume of about 600 liters in good cylindrical geometry. The other, a relatively low density system, with core and blanket composition similar to EBR-II, had a core volume of about 900 liters, again in good cylindrical geometry. Calculations have given a shape factor of 0.99 for the high density system and 0.92 for the low density system. Further studies are needed to predict the variation of this factor as the ratio of length to diameter of the cylinder is reduced.

2. ZPR-III

Critical studies of the first of a series of oxide cores were completed and a second oxide core loading was begun.

The first oxide core loading, Assembly 29, was critical with 422.91 kg of U-235. A typical drawer loading contained three columns of uranium, six columns of aluminum oxide, four columns of 45% aluminum, and three columns of stainless steel. (Further descriptions of this assembly can be found in previous Progress Reports.) The experiments completed this month were central danger coefficient worths and central fission ratios.

This assembly, the first made with an oxide loading, revealed large discrepancies between calculated and experimental critical masses. Consequently, the experimental program was planned to give as much information as possible on the factors which might be in question. Calculation techniques in the past have been reliable for central fission ratios; therefore, these were obtained experimentally for comparison. Central reactivity coefficients also give at least a qualitative basis for comparison between different assemblies; therefore, a complete set of these values was obtained. On the other hand, experiments on the bunching and rearrangement of materials were omitted, as it is believed that their effects are of a second-order nature and can be at least temporarily neglected until the worths of materials are better understood.

In Table II the central danger coefficient measurements are compared with those obtained from Assembly 20, the clean-core mockup of the Fermi power reactor, and Assembly 24, an all-uranium core. The core volume and critical mass of the three assemblies are listed in Table I.

Table I: Core Characteristics

<u>Assembly Number</u>	<u>29</u>	<u>24</u>	<u>20</u>
Critical mass (kg, U-235)	420.7	469.6	431.5
Core Volume (liters)	454	331	378

Table II: Central Danger Coefficient Measurements

<u>Sample</u>	<u>Worth (lh/kg with respect to void)</u>		
	<u>29</u>	<u>24</u>	<u>20</u>
Be	108	-48.6	34.5
C	41.6	-35.6	6.6
Al	1.3	-13.1	-4.0
SS	-1.9	-9.1	-4.1
B ₄ C	-1640	-1148	-1275
U ⁴ (Fully enriched)	207	273	228
U (Depleted)	-6.5	-7.7	-5.5

Apparently the spectrum of Assembly 24 has a sharper peak, and fewer high energy neutrons as well as fewer lower energy neutrons than Assembly 29. The moderating materials, Be and C, are negative in Assembly 24 because they reduce the fission contributions from U-238. In the other two assemblies, the amount of U-238 present is smaller and the spectra are altered so the reduction in U-238 fissions is more than offset by the increase in U-235 fissions caused by the moderating effect of Be and C. These materials are therefore positive in Assemblies 20 and 29. B₄C, which is a poison, reflects the low energy end of the spectra of the three assemblies by increasing in negative worth as the spectrum becomes softer.

Central fission ratio measurements were made, and the results are compared with those from Assemblies 20 and 24 in Table III.

Table III: Fission Ratios

<u>Assembly Number</u>	<u>29</u>	<u>24</u>	<u>20</u>
U-238/U-235	0.0356	.0284	.0381
U-234/U-235	0.259	0.246	0.292
U-233/U-235	1.469	1.44	1.52
Pu-239/U-235	1.059	1.14	1.15
Pu-240/U-235	0.289		0.332

U-238 and U-234 ratios indicate the relative amounts of high energy neutrons in each assembly.

As pointed out previously, calculations of the critical mass for the first oxide loading were not in agreement with the experimental results. Therefore, in order to have another oxide assembly for which calculated and experimental results could be compared, it was decided to build an assembly in which the amount of oxide was reduced. The second oxide assembly, which is now being loaded into the reactor, will have only three columns of oxide per drawer rather than six as in the first oxide assembly. These three columns are being replaced by 45% aluminum so the net change is a loss of oxygen. Since this represents a loss in reactivity and the core volume is being reduced slightly, the uranium enrichment is being increased compared to the first oxide loading.

3. ZPR-VI and ZPR-IX

a. Building Construction and Procurement - The status of construction for the cells in the building is as follows:

Cell No. 4: Shoring, forms, and resteel are being placed for the upper portion of the cell in preparation for the concrete pour. Several core samples from the bottom half of the cell were taken to check the strength of the concrete. The A, C, and D doors are in place.

Cell No. 5: The concrete pour for the upper half of this cell was made on November 4, 1960. Forms and shoring have been removed and the area is being cleaned up. The A, C, and D doors are in place.

Cell No. 6: All of the brick work has been completed. The bridge crane has been installed. Work is progressing on the installation of mechanical and electrical items.

For the remainder of the building, the exterior brick work is approximately 80% complete. The interior concrete block walls in the basement and first floor are approximately 75% complete. The penthouse roof over the office portion of the building has been framed and poured. Electrical lines between Building 315 and Building 316 were installed. The installation of the electrical substation and the cooling tower is almost complete.

b. Critical Facilities - Recently a review has been undertaken to determine the advisability of deferring ZPR-IX. This delay has unavoidably slowed the pace of procurement of components for ZPR-VI also. Pending completion of this review approval of the drawings for the bed and table assembly and neutron source drive assembly is being withheld, as well as the placing of the contract for the control console and nuclear instrumentation.

The coating unit for the depleted and enriched uranium has been fabricated and is being installed, and coating tests using KEL-F will be initiated upon completion of the installation. Approximately 32 metric tons of depleted uranium blanket and "dilution" plates have been received from the vendor and inspected. As yet, none of the enriched uranium fuel plates have arrived.

A second batch of stainless steel tubes for the matrix assembly has been received and inspected. None of the 31 units met specifications, and so the total shipment was returned with a letter of explanation regarding the basis for the rejection. Further samples will be submitted before the vendor proceeds with the production.

c. Control Rod Drive - Two types of control rod drives are to be used for these reactors: (a) an expulsion type, whereby a fuel drawer will be ejected from the core; and (b) an insertion type, which will insert a boron-filled shim rod into the core.

The latest scram time measurements for the fuel drawer expulsion drive indicated that the magnet release time was much too long (98 milliseconds). This was attributed to inadequate chrome plating of the magnet armature. The magnet shell and armature have been replated to give a total chrome gap of 0.15 mm between magnet and armature, which should be

sufficient to reduce the magnet release time to a value near 10 ms. The drive will now be reassembled and timed.

Other modifications are being made to the drive simultaneously to yield a reduced over-all scram time. The time to travel the full stroke of 61 cm, not including magnet release time, is 275 ms. It is expected that the total time, including magnet release time, can be reduced to about 240-250 ms. Rod motion is almost linear with respect to time between the first 10 cm of travel and the last 5 cm.

The B¹⁰-filled blade insertion drive is completely fabricated and being assembled. A test stand similar to that used for the expulsion drive is being built and will be installed in Building 316.

All drawings for both types of drives are complete and are being checked. Final approval of each design awaits completion of performance testing.

4. Juggernaut

Phase 2 of the reactor construction (see Progress Report, September, 1960, ANL-6234, p. 12) is complete and Phase 3 has begun. After the biological shield was poured some evidence of honeycombing appeared at the upper surface. The effect does not appear to penetrate deeply but the adequacy of the shield will not be known until the system is operated at design power levels.

The 5 ft. by 5 ft. top shield has been delivered without the concrete fill so that the shim-safety rod thimbles may properly be aligned with the slot in the graphite.

The control console was moved to Building 335. Wiring of the console was completed, and the assembly of cables was begun. All of the trip channels and counting channels are now operative. The units will be tested in place to determine whether they will operate at the existing temperatures.

Five of the console wiring diagrams have been brought up to date. A panel was designed to hold the throttle valve indicators, tower fan switches, flowmeter, and relays.

The Hazards Summary Report and Operating Manual are being reviewed and final editing is in progress.

B. EBR-I

1. Operation

Reactor operation this month has been directed primarily toward the training of operators for the EBR-II reactor. This program began in October at the conclusion of the runs made in connection with the stability studies on the Mark III core.

2. Mark III Stability Studies

A final report has been prepared on the stability studies conducted on the Mark III core. The following is a summary of the results obtained:

From the results of instability studies it may be concluded that the operational characteristics of the fully-ribbed and rigid Mark III loading are governed by feedback processes which guarantee safe and stable operation under normal operating conditions. The influence of the stabilizing ribs is felt in two ways, both of which are in the direction of increased stability. The ribs provide strong radial coupling between fuel rods and between fuel rods and hex cans with the consequence that a large radial contribution to power coefficient is realized. The ribs have also eliminated, for all practical purposes, positive feedback effects arising from the inward bowing of fuel rods during power increases. Absolutely no evidence of positive reactivity effects was noted for the fully-ribbed and rigid core.

The results of logical extrapolations of full power, full flow test data indicate that the reactor could be brought into a resonant condition only at power levels exceeding 1000 megawatts. Since nominal full power operation is limited by design of 1.2 megawatts the hypothetical instability power is one of mere academic interest. The limit to which power may be increased is dictated not by matters of instability but by matters of heat removal. In a practical sense, then, the fully-ribbed Mark III loading could never be made unstable.

The strong non-linearities observed in the power coefficient pose no serious operational problems at least over the range of power associated with the tests. Except for the region between zero and 200 kw, where the power coefficient is small but still negative, the power coefficient is everywhere very strongly negative. The non-linearities are in fact relatively unimportant from the stability viewpoint. They are, however, bothersome since they greatly complicate the interpretation of test data in terms of a single unified model which may be applied over wide ranges of power, flow and temperature conditions. The origin of the non-linearities appears to be a consequence of power and temperature sensitive clearance systems existing between rods, between rods and hexes, and between hexes. Attempts to define such systems mathematically are further frustrated by differences existing in the expansion coefficients of fuel and structural materials. Obviously, the association of such extremely tenuous clearance systems with power coefficient non-linearities cannot be placed on an analytical basis. The descriptive interpretation of the effects of such clearance systems is nevertheless important, not only for an understanding of Mark III behavior but also to serve as a guide in the prediction of power coefficient non-linearities in other conceptual fast reactor designs.

As an additional complication the power coefficient has also been shown to be sensitive to the temperature of the inlet coolant. The effects of inlet temperature increases are reflected by significant increases in the magnitudes of both static and dynamic power coefficients. The time dependence of feedback has been found to be insensitive to such changes. The explanation of this effect lies unquestionably in the strong temperature dependence of the expansion coefficient for the uranium-zirconium alloy fuel material. The effects of more feedback at higher temperatures, while real,

are relatively minor. At higher inlet temperatures the reactor would be more stable at the lower frequencies but would be less stable for frequencies above 0.12 c/sec. The effect is nevertheless small and in no way invalidates general stability conclusions based on lower temperature data.

The influence of structural feedback has also been observed in Mark III, fortunately to a degree much less serious than in the earlier Mark II design. At the higher inlet temperatures approximately 13% of the static power coefficient is associated with a feedback process operating with a time constant of the order of 400 seconds. In view of the extreme slowness of action the physical process responsible for the feedback must certainly be one associated with some massive portion of the structure at some downstream location. The existence of this delayed component has little effect on stability conclusions since its effect cannot be sensed for frequencies greater than .003 c/sec. For frequencies less than this its presence actually causes the reactor to be more stable.

Attempts to enhance and to measure possible feedback effects arising in the radial breeding blanket were unsuccessful. As a consequence it must be concluded that feedback mechanisms based on the pre-heating of inlet coolant by the transfer of heat between core outlet and blanket inlet in series flow are unimportant. Since pre-heating possibilities were very nearly the same for both Mark II and Mark III it may also be concluded that strong blanket feedback mechanisms were not the source of the Mark II delayed negative component.

As the coolant flow is reduced (at constant power) three basic changes occur in the feedback. The power coefficient increases by an amount roughly inversely proportional to the change in flow rate. The time constant for the expansion of fuel, and the effective transport lag, also increase. All of these changes are in the direction of increasing instability since the result of such changes is manifested by strong increases in the phase and amplitude of feedback vectors in the important III quadrant.

Logical extrapolations of approximately one-third flow test data lead to the conclusion that the reactor would reach resonance instability in the vicinity of 10 megawatts, a power level approximately twenty times that of the designed value for one-third flow. Further reductions of flow at constant power would result in an accelerated lowering of the resonance power. Further reductions would be inadvisable not only because of the lowered resonance power level but because of the possibility of fuel rod temperatures exceeding limits set by design.

The results of removing the stabilizing ribs from fuel rods have demonstrated conclusively the existence of a strong positive effect which must logically be associated with the inward bowing of fuel rods. An extrapolation of the results obtained from shearing the ribs from approximately one-third of the fuel rods results in a value of $+2.06 \times 10^{-6} \Delta k/k/kw$ for the fully-sheared core. Since the prompt negative component associated with fuel, coolant and blanket expansion is $-2.205 \times 10^{-6} \Delta k/k/kw$ the response of the reactor even when fully-sheared would still be governed by a smaller but still negative prompt power coefficient. Under these conditions runaway excursions such as that damaging the Mark II core would be impossible.

An application of the Nyquist stability criterion to the extrapolated fully-sheared data results in the conclusion that the reactor would attain resonance instability at full flow at approximately 11 megawatts. The frequency at which resonance will occur is in the vicinity of .14 c/sec. The net result of the positive component associated with rod-bowing is a drastic lowering of the resonance power.

It has also been shown that rib-shearing results in an unexpected and unexplained increase in the magnitude of the delayed structural power coefficient component. An empirical fit of feedback data to a model describing the dynamic and static behaviors of the partially sheared core resulted in the following values for the respective prompt negative, rod-bowing, and delayed structural power coefficient components: -2.21×10^{-6} , $+0.543 \times 10^{-6}$ and $-0.873 \times 10^{-6} \Delta k/k/kw$. These values correlate in a curious manner with those empirically deduced for the prompt negative and delayed structural power coefficient component for the fully-ribbed core; namely, -2.21×10^{-6} and $-0.330 \times 10^{-6} \Delta k/k/kw$. The magnitude of the positive power coefficient component is equal to the increase in the magnitude of the structural component. Apparently rib-shearing introduces two feedback processes: rod-bowing which is prompt and positive and one of an unspecified nature that is negative and extremely delayed in time. To explain the origin of the increased delayed negative effect a postulate has been formulated which considers wall deformation in those assemblies containing sheared fuel rods. Regardless of the exact nature of the process it is clear that the strong increase in the negative structure term does not affect conclusions regarding the resonance stability of the partially or fully-sheared cores since the process acts so slowly that the reactor oscillating at frequencies as low as 0.02 c/sec cannot sense the feedback.

The over-all behavior of the Mark III core whether fully-ribbed and rigid, fully-ribbed and loose, and even partially or fully sheared, is one of extreme stability, a behavior in striking contrast with that of Mark II. To appreciate the reasons for this significant difference it is instructive to show how those features responsible for the instability of Mark II have been eliminated in Mark III through changes of design.

The most important and disturbing feature of Mark II was the existence of the prompt positive rod-bowing component. The inclusion of stabilizing ribs and a system of tightening rods in the Mark III design has eliminated or at least minimized to a point beyond detection similar effects in Mark III. It is equally significant from the viewpoint of understanding Mark II results that the Mark III tests have shown that rod-bowing is the consequence of special, and not easily obtained, conditions of fuel rod restraint.

The explanation of why rod-bowing existed in Mark II in which conditions were superficially not conducive to rod-bowing is based on an ordered arrangement of rod tips brought about by expansive action of the lower shield plate. Expansion based solely on the temperature differential across the core was evidently sufficient to force the fuel rods outward until the tips were bearing against the outer edges of the locating holes. Under these conditions inward rod-bowing is the most likely consequence of deformations accompanying power increases.

The explanation of why the prompt power coefficient of Mark II was

positive, and is indicated to be negative in Mark III even for the fully sheared condition, is found not so much in the magnitudes of the positive components as in the magnitudes of the prompt negative components. Because of the lack of radial coupling between fuel rods in Mark II the effects of radial expansion contributed little to the prompt power coefficient. The net prompt power coefficient was associated with axial fuel expansion and coolant expulsion and was insufficiently large to cancel the prompt positive component from rod-bowing. In Mark III, on the other hand, which represents a strongly coupled rod system, the prompt negative power coefficient includes contributions from both axial and radial expansions. The combined prompt negative effects are sufficiently large to override the positive component from rod-bowing and the net power coefficient is negative. The inclusion of stabilizing ribs in the Mark III design is therefore beneficial in two respects: rod-bowing is eliminated and the prompt negative power coefficient is strengthened.

The second disturbing feature noted in Mark II was the presence of a delayed negative power coefficient component which now appears to be the result of delayed expansions in the ligamental structure of the lower shield plate. Experiments carried out on a dummy shield plate have established the existence of radial movements of the shield plate ligaments which, because of their contiguity with the fuel rod extensions, effectively modify the core size following power changes. Such an effect is clearly negative and operates with a time dependence reasonably consistent with that deduced analytically from a consideration of Mark II test results. Structural features such as the shield plate system of Mark II have been eliminated in Mark III. This fact, coupled with the observation that no similar component has been observed in Mark III, supports the validity of this explanation.

In summary, those features responsible for the instability of Mark II, namely, the prompt positive and delayed negative power coefficient components, were apparently the result of design peculiarities. The elimination of the perforated shield plate system coupled with the addition of stabilizing ribs to the Mark III fuel rods has resulted in a reactor the performance of which is stable under all credible operating conditions.

3. Mark IV Development and Fabrication

Both centrifugal casting and injection casting have been tested for the production of EBR-I fuel elements. Injection casting now appears to produce satisfactory fuel elements at a higher yield and with less difficulty than centrifugal casting.

Three injection cast melts have been made. Owing to improper alignment of the crucible and mold support the first melt produced only one completely filled mold. The second and third injection castings were made with a redesigned mold support and a tantalum crucible liner. The results have been quite satisfactory, producing castings from 14 to 18 in. long. Diametral tolerances were held within ± 0.0005 in. There was some evidence of reaction between the Vycor glass mold and the material but it is believed that this can be eliminated by the proper mold coating technique and mold temperatures.

The first upsetting dies did not produce fuel pins of the proper dimensions to evaluate the die rib forming technique. The die is being re-machined to closer tolerances prior to further upsetting tests. The machining

experiments have indicated that this alloy can be machined with the use of carbide tools and without fluid coolants in the helium hood atmosphere. No problems are anticipated.

The automatic control circuits for the girth welder and the seal welding machine have been constructed. Test welds made with this equipment appeared to be satisfactory. Extensive modification work was required on the liquid metal filling chamber to adapt it to the NaK filling of the Core IV fuel elements. The equipment was rebuilt so that the seal weld can be made in the NaK filling chamber allowing closure welds to be made at a controlled subatmospheric pressure.

Development work has been completed on the 8 in. and 22 in. vinyl tube loading pouches and drawings have been submitted to a commercial fabricator for their construction. A 1 KW sealing generator has been procured and a 15 in. dielectric sealer has been designed for use with this generator. The purpose of the tube loading pouches is to allow assembly of the plutonium core slugs in the zirconium jackets without the external contamination of these jackets.

Production of plutonium 1 w/o aluminum fuel slugs has been delayed pending the receipt of plutonium for this work. Approximately 50 fuel slugs have been produced during the experimental work and will be transferred to the production inventory.

Fabrication of the Zircaloy-2 rod stock mentioned in October is now complete. This rolled and centerless ground material is ready for non-destructive ultrasonic through-transmission inspection for internal defects.

Drawing of Zircaloy-3 wire for jacket spacer ribs is approximately 35% complete. After drawing to size the wire will be eddy current inspected for soundness.

December 20 has been set as a firm shipment date for the total 2,200 foot order for Zircaloy-2 jacket tubing. The vendor of the Zircaloy-2 tubing required for the Mark IV thermocouple assemblies has indicated uncertainty as to his ability to deliver this item. Therefore, development of techniques for fabrication of this item is under way at the Laboratory.

Evaluation of eddy current traces from the jacket tubing inspection equipment is continuing.

C. EBR-II

1. Construction

The following tabulation presents the status of construction contracts as of November 22, 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)	**62	**December, 1960
Fuel Cycle Facility (Package 3)	65	May, 1961

*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.

**Construction delayed; revised schedule not yet available.

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, Sanitary System and Warehouse.

2. Installation of Equipment - Package 4

a. Reactor Plant - It has been possible to accelerate significantly the rate of installation of equipment by the Package 4 contractor during the present period because of the greatly reduced concurrent activity of the Package 2 contractor within the building. Work is in progress on about one-half of the "packets" involved in this plant.

One major obstacle was encountered in the Reactor Plant Package 4 installation work of this period. In the course of field preparation for installation of the reactor vessel grid assembly, inspection disclosed the presence of substantial amounts of fabrication debris (chips, turnings, bits of cloth, etc.) within the assembly. In respect to the need for cleanliness, this assembly is perhaps the most critical component in the EBR-II system. The coolant passes directly from the grid plenums to the fuel subassemblies. Debris from the plenums conceivably could lodge in a fuel subassembly and cause flow perturbations or, in an extreme case, flow stoppage. Because of the critical nature of the assembly, the cleaning and inspection operations performed in the field to assure an adequate final state of cleanliness are here reported in detail.

To aid in visualizing these operations, the design of the grid assembly is presented pictorially in the isometric cutaway of Figure 1.

Most of the debris noted was located in the high pressure (center) plenum and the high pressure inlet annulus, although some was found in the low pressure plenum. Preliminary efforts made to remove these contaminants showed clearly that a major program of inspection and cleaning of this component would be required to assure achievement of satisfactory cleanliness. The greatest difficulty encountered in implementing this program stemmed from the lack of adequate access to the critical internal regions. The assembly is comprised of a cylindrical, box-type structure having a complicated internal arrangement and only two large-sized (11 in. diameter) openings to the exterior. With the exception of the bottom cover, the removal

of which does not provide access to the most complicated region (the high pressure plenum), the assembly is of welded construction. Thus, access to the center plenum is limited to the 2 in. diameter subassembly adapter holes through the grid plates, and access to the inlet annulus is limited to the two large sodium inlet nozzles.

The first cleaning method pursued was to remove the bottom cover plate to permit access to the region beneath the low pressure plenum, then to vacuum the interior chambers as far as possible using, in addition to the standard vacuum cleaning attachments, a special attachment made from copper tubing to reach as many corners and crevices as possible. Figure 2 shows the assembly with bottom plate removed. It was also noted during this operation that the lower edge of the upper grid plate had not been deburred at the center plenum region holes. A special hand tool was fabricated to remove these burrs using a short piece of emery stone and/or a piece of a file to cut the burrs loose. Approximately one pint of chips were generated and removed by this procedure.

While the above work was being performed, a blowing tool was fabricated to permit a blast of air to be directed radially through the spaces between the blanket region tubes in the coolant plenum region between the two grid plates. This tool was an 0.75 in. copper tube with a right angle discharge port at the end to permit insertion through a center plenum hole to blast air outward, and was also inserted in the outer annulus to blow air downward. This cleaning procedure resulted in the recovery of approximately one additional pint of chips. The cleaning operations up to this point successfully removed the major portion of the chips encountered, (see Figure 3).

Temporary blind flanges for the nozzles were next made to permit flushing and rinsing operations with water and trisodium-phosphate solution. Gasketed plates were also bolted over the center plenum to permit filling of the high pressure plenum with the grid standing on its side. The grid was then removed from the building onto the loading dock at the freight door to proceed with the various flushing operations.

The first flushing attempt made was by attaching a fire hose at the center of the grid, then flushing out through one inlet nozzle using full flow from a fire hydrant. The supply hose was next moved to the opposite nozzle to flush through the full diameter of the grid as nearly as possible. This cleaning method resulted in the recovery of only a minor quantity of additional chips; therefore, it was abandoned after spending approximately three and one-half shifts on the process.

The nozzle originally used as a drain was then plugged, the grid filled approximately two-thirds full of water, and 30 quarts of trisodium-phosphate solution added. This mixture was agitated in the grid by blowing air through a copper tube having a series of holes drilled along its length to give thorough agitation throughout the grid. This process was more successful in recovering chips than the straight flushing operation, although not as successful as the air blowing method.

The above process was repeated substituting steam flow for the air agitation to produce a hot solution to remove any oil or grease in the grid.

This was followed by a water rinse and further air blowing.

The grid was next brought back into the building and rotated 180 degrees, then returned to the loading dock to repeat the flushing and rinsing operations on the other half of the grid.

With the grid thus degreased and flushed with water, the air blowing operations were repeated on a systematic basis in an attempt to blow through each channel formed by the blanket hole tubes before making a final inspection for chips. This method was continued until no further debris was found after covering all channels thoroughly.

The next operation was to make as complete an inspection of the grid interior as possible. The grid hole pattern is hexagonal, and the spacer tubes in the blanket region form straight rows in three directions. This arrangement provides clear, straight paths on both sides of each row of tubes; thus it is possible theoretically, to see six sides of any tube. The design is such that these paths form clear channels approximately 1/16 in. wide between the upper and lower plates of the grid in the high pressure region where the chips were most likely to lodge. It should be noted that some of these channels run from the annulus to the central plenum region and some run from one point to another in the annulus without intersecting the central plenum. Inspection methods permitting viewing both from the central plenum and through the inlet nozzles had to be used to provide inspection of as many channels as possible. Owing to the inaccessibility of many of the channels that did not intersect the central plenum, it was not possible to inspect completely all tubes on six sides. However, most tubes were inspected on at least four sides and all tubes were inspected on at least two sides. As the inspection progressed, each path inspected was marked by a strip of masking tape on the face of the grid along the line of sight as a record of the inspection coverage.

The method used for inspection of the channels intersecting the central plenum was to provide a light source in the annulus at the channel to be inspected, then insert a small 1 in. by 3 in. mirror swivel mounted on a long handle through a center region hole to view the lighted channel. This method is illustrated in Figure 4 with the inspection instruments being held in their proper relative positions but outside the assembly so that they may be seen.

To inspect the channels not intersecting the central plenum, basically the same technique was used. Since all viewing had to be performed through an inlet nozzle, and because the circular baffle plate interfered with many of the channels, it was not possible to make a complete inspection of all of these channels. The most successful light source for these inspections consisted of a series of six small (15-watt) bulbs mounted in a row which permitted completely lighting a single channel without lighting any others. This made for ease of locating the desired channel with the inspection mirror. By traversing the mirror lengthwise along a channel, the entire channel could thus be inspected for chips.

A flexible borescope was used to get a close-up view of all areas which could be reached by it. It was not successful, however, for inspecting a full channel through the blanket region. This inspection was continued

until the point was reached where only an incomplete survey of a channel could be made, and that only after considerable effort. Because of the difficult locations and angles as much as two hours was spent in attempting to inspect a single channel with only partial success.

During the course of the above inspections, a significant amount of additional debris was located. This was removed by use of probes and air blowing techniques, as appropriate. All chips so located were recovered and are shown in Figure 5. In only one region were chips found that could not be completely recovered. These were found jammed into the crevices where the high pressure system flow baffle fits into the lower grid plate. It appears that assembly was made with some chips wedged in these joints. By scraping in the cracks with a thin stainless plate, the bulk of these chips was removed; however, some chips broke near the surface of the joints and could not be completely removed. One such chip remaining is shown in Figure 6.

After inspection, complete cleaning with acetone and alcohol was accomplished.

The above cleaning and inspection work consumed approximately 500 man-hours of effort by the contractor and probably about the same amount of effort by ANL personnel. At the conclusion of this work, it was considered that every reasonably practicable step had been taken to assure the requisite cleanliness prior to installation, and the grid assembly was therefore installed within the primary tank.

Further inspection will be performed with the grid in place. A special-purpose periscope which will permit detailed observations from the central region outward toward the periphery has been developed and shipped to the field. Inspection with this device has been temporarily postponed because of conflict with installation work in and around the reactor vessel which cannot be delayed without jeopardizing construction schedule objectives. This final inspection can be performed at any time in the future and will therefore be scheduled to minimize interference with other construction activities. In the event that additional contamination is found, past experience has shown that with sufficient ingenuity it can be removed with the grid in place. However, it is not anticipated that any contamination of significance will be disclosed.

To provide further assurance of adequate cleanliness, the primary sodium coolant system will be operated initially with no subassemblies in the reactor and the reactor vessel cover in the "up" position. This will permit any debris which might dislodge from the plenums to flow through the reactor directly into the bulk sodium of the primary tank. It is confidently believed that any such debris within the primary tank can cause no problem because of the very large settling area exhibited, the absence of appreciable circulating current, and the considerable physical separation between pump inlets and either the outlet from the reactor or the bottom of the tank. In addition, the possible advisability of preliminary operation with removable dummy subassemblies equipped with filters inserted in the core region of the reactor is being examined. By removal of these filter subassemblies after such operation, any debris collected would be removed permanently from the system.

Figure 1
Details of EBR-II Grid Assembly

Figure 2
Grid Assembly with
Bottom Plate Removed

Figure 3
Chips Removed from Grid Assembly

Figure 4
Method of Inspecting the
Grid Assembly Channels

Figure 5
Additional Debris Removed from Grid Assembly

Figure 6
Inaccessible Broken Chip

Figure 7
Test Setup for EBR-II Reactor Vessel Cover
Lock Mechanism and Plug: Upper End

Figure 8
Test Setup for EBR-II Reactor Vessel Cover
Lock Mechanism and Plug: Lower End

b. Power Plant - The machine shop in the Power Plant is essentially complete and has been put into operation. Appreciable progress has been made on the electrical packets for the continuous power supply, instrumentation, and primary sodium pump controls. Checkout is being made on the rod position indicators, gamma detectors, diesel emergency transfer system, and general nuclear instrumentation.

3. Engineering

The Architect-Engineer (H. K. Ferguson Company) has completed the maintenance manuals for the Reactor Plant, Power Plant, Sodium Boiler Plant, and Fuel Cycle Facility, and the Operating Manuals for the plant services systems. These represent all the manuals to be prepared by them. All of these manuals, and the originals of all officially issued Ferguson drawings for EBR-II, have now been received by the Laboratory.

Work has been started at the Laboratory on the Hazard Report Addendum and the Operating Manual.

4. Procurement

The procurement of equipment by the Laboratory is satisfactory except for the superheaters (see below), the sodium-to-sodium heat exchanger, and a few minor items. The AEC is assisting the Laboratory in expediting delivery by the vendor of the sodium-to-sodium heat exchanger. If the present promised delivery date is realized, no delay in the installation schedule (Package 4) will result.

5. Component Development - Instrumentation

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

The design of the final control unit has been completed and it is now being fabricated. Four individual units will be interconnected in the automatic control system. They are: power supply, amplifier, power demand potentiometer positioner, and Δk limiter. The power supply is a commercial unit. The remaining three units are being fabricated according to specifications.

b. Control Rod Reactivity Generator - A sinusoidal reactivity generator and wave analyzer are being designed to measure the transfer function of EBR-II. The improved nulling circuit of the transfer function measurement unit has been tested. In the absence of noise, the system has a sensitivity of ± 0.2 decibels in amplitude and ± 0.5 degree in phase shift. Techniques are being investigated for achieving a null in a minimum number of cycles. This is especially important at 0.005 cps where 200 sec elapse for each cycle. If ten cycles are required to reach a null, approximately one-half hour will have to be devoted to a single data point at this frequency.

The internal reactor component drawings on the reactivity generator mechanism have been checked and corrected. The preliminary layout of the drive mechanism is approximately 75% completed.

c. Start-up Instrumentation - Special instrumentation is being provided for EBR-II startup. Two movable thimbles will be installed in control rod locations. High temperature fission chambers capable of operating at 455°C will be used. Two separate channels of temporary electronic equipment, high voltage supplies, pre-amplifiers, log count rate amplifiers, period amplifiers, scalars, and audio count rate circuits will be installed.

A dual high voltage supply manufactured by General Electric Company and two scaling units made by Eldorado Electronics were received and tested for performance and stability. A bread-board model of an audio circuit for the scalars was constructed and tested. Except for the high temperature fission chambers, all components are on hand and assembly has commenced.

d. Fuel Cladding Failure Detector - Estimates have been made for the efficiency of the delayed neutron detection scheme (Progress Report, September, 1960, ANL-6234, p. 25) on the basis of certain plausible assumptions about the nature of cladding failures and about the site of the equipment and its performance characteristics. An experimental (as well as theoretical) effort to design the moderator part of the detection equipment is in progress. It is expected that a useful instrument, with capability of reliable detection of those failure events in which a relatively large hole in the cladding develops, will be ready in January, 1961. Further improvements are feasible in the direction of fission chambers of higher efficiency as well as electronic circuitry of improved stability, reliability and speed of response. Such improvements may be made at a later time, if operating experience indicates the requirement.

6. Component Development - Reactor

a. Reactor Vessel Cover Lifting Mechanism - The vessel cover lifting mechanism suspends the cover from the small rotating plug mounted in the primary tank support structure, and raises or lowers the cover by means of two 4-inch diameter circular columns which extend through the rotating plug and engage a pair of jackscrews driven by a single motor.

As previously reported (October, 1960), fabrication and assembly of this mechanism was completed at the Watertown Arsenal, and an initial, full load, operational test was conducted which showed the unit to be satisfactory in every respect except that some chattering of the traveling nuts on the lifting screws occurred. Attempts were made to relieve the latter condition through realignment of the mechanism subassemblies, use of various types of lubricants, and working-in of the nuts on the screws, all without significant result.

During the present period, the threaded portions of the traveling nuts, initially made of aluminum-bronze, were replaced with inserts of Meehanite hardened to Rockwell C 30-35, and the mechanism retested. In this retest the unit performed completely satisfactorily, with no chattering. The unit is now being disassembled and prepared for shipment to the construction site.

b. Reactor Vessel Cover Lock Mechanism - During reactor operation the reactor vessel cover is held clamped to the top flange of the reactor vessel by means of three equally spaced "cover locks." These locks are actuated remotely to permit unclamping of the cover (preparatory to the

lifting of the cover) during the fuel transfer process. The locking mechanisms extend from the bottom of the cover up through the bulk sodium and biological shield to the top of the primary tank support structure. The drive units are electrical, with manual override, and are located at the top ends of the mechanisms.

As reported previously (October, 1960), two of the three lock mechanisms required were given a no-load operational test to check smoothness of operation and correctness of wiring connections. During the present period, the third mechanism was similarly tested with satisfactory results. All three units now have been shipped to the construction site.

To further check the suitability of these units for application in the EBR-II, an identical unit was completely tested some months ago under full-load conditions equivalent to those anticipated during reactor operation. Figure 7 shows the upper portion of this unit in the "up" position mounted in a nozzle similar to that to be used in the actual installation. Note the operator on top of the unit with both manual and electrical controls installed. The loading device consisted of a beam lever arm loaded with lead brick as shown in Figure 8.

The test consisted of electrically lowering the unit until the "holddown" column exerted a force slightly greater than the design load of 30,000 lb. (as evidenced by a lifting of the dead load on the longer lever arm.) At that load a limit switch was arranged to trip the motorized operator and prevent further downward motion. This procedure was repeated for several cycles using the motorized operator. It then was repeated using only the manual operator. The unit performed completely satisfactorily during both the motorized and the manual operations.

The results of this test and the recent no-load tests on each of the three mechanisms shipped for installation are considered to assure proper operation in the EBR-II.

c. Gripper and Holddown Mechanism - The function of the gripper and holddown mechanism is to insert and extract subassemblies from the reactor vessel. This mechanism performs its operations remotely. It incorporates several design features which lend added reliability to the fuel transfer process, such as "sensing" devices, "force-limiting" devices, and holddown of adjacent fuel subassemblies to prevent their simultaneous removal. The mechanism is necessarily complex and requires extremely careful alignment and adjustment of many interconnected assemblies. Accordingly, it has seemed advisable to assemble and test this unit at the Laboratory prior to shipment to the construction site.

Within the present period, assembly has been completed, including all internal wiring, and testing of certain subassemblies started. Assembly was accomplished with fewer difficulties than had been anticipated; although a few minor interferences were encountered, no problems of significance occurred. The gripper drive motor has been actuated and vertical travel of the gripper mechanism checked out satisfactorily. Similar testing of the holddown subassembly was attempted, but operation of this drive motor was unsatisfactory. This difficulty apparently arises from improper adjustment of the vendor-supplied motor controller which is now being readjusted by a vendor's representative. A temporary power and control console is being

wired up to enable complete operational testing of the mechanism as a whole. It is anticipated that final testing will be completed and shipment of the unit made in the next three to four weeks.

7. Component Development - Steam Generators

Three superheaters are now complete up to the welding of the tubes to the sodium tube sheets. One sodium tube sheet girth weld has been completed on Unit 4, the last required for the EBR-II plant, and on Unit 5, a spare. No tube to sodium tube sheet welding has as yet been done on any of the production units because of a welding problem which has been encountered. The problem consists of overpenetration of the duplex tube wall which ranges from penetration up to the nickel phosphorus bond line in most cases, up to complete penetration of both walls in a few cases. This problem has incited intensive work on the welding procedure. A large fraction of the welds now being made on facsimile samples are acceptable, and penetration through both tube walls in a given duplex has been eliminated. The verification of good welds can be done presently only by destructive methods. Therefore, welding development is being continued on facsimiles until either a method incapable of penetrating the bond layer is developed, or a nondestructive test is available for determining that the layer has been violated. Production of the first three superheaters will not go forward until either or both of these last conditions have been met.

To protect against the possibility that welding of the nickel-phosphorus bonded duplex tubes to the sodium tube sheet cannot be done within a reasonable time period, the possibility of using single wall tubes of the same wall thickness as the total wall of the duplex tubes is being investigated.

8. Component Development - Fuel Reprocessing Facilities

All of the structural steel for the Fuel Cycle Facility has been erected. The contractor is now erecting roof decking and aluminum siding, and installing exhaust fans, refrigeration equipment, motor control centers, lighting panels, and general electrical equipment. Enclosure of the building should be completed during December.

Design and testing of equipment for the Fuel Cycle Facility continues. The window shutter design is being examined to eliminate possible conflicts with other equipment. A mockup of a portion of the electrical and thermocouple service plug is almost completed. If tests with this unit are successful, final design of all service sleeve plugs for the Argon Cell will follow. The drawings for the small transfer locks are almost completed.

The specifications for the transfer coffins have been sent out for bids. The three large shielded units will be used to transfer highly radioactive materials between the reactor and the Fuel Cycle Facility, and between the Fuel Cycle Facility and storage or waste disposal sites.

The manipulator and crane control equipment was inspected and tested in the manufacturer's shop. The manufacturer was advised of the necessary corrections based upon the tests.

No significant effects resulted from the continuation of irradiation of aluminum paint samples, MI cable sealants, and asbestos wire insulation

during the month.

Three types of radiation resistant greases have been irradiated and then tested in roller-type bearings. Radiation doses ranged as high as 5×10^9 rad and test periods as high as 300 hours. While further testing is necessary, the results are encouraging. Tests of unlubricated bearings showed that lubrication is necessary even under light loads.

The efficiency of the Fiberfrax fume traps was investigated as a function of capacity. The efficiency is 98 to 100 percent for sodium loadings up to 80 grams, but decreases at higher loadings. It has been estimated that the trap in plant operations will be loaded with a total of 80 grams of condensible material.

The special viewing and remote maintenance problems of the Air and Argon Cells of the Fuel Cycle Facility caused the selection of 1000 watt mercury vapor lamps for illumination in these cells. A series of tests over a two-year period showed that irradiation would not appreciably darken the lamps. Subsequently, the lamp manufacturers modified the type of glass used in the lamp and radiation darkening of the new lamps now poses a serious problem. The manufacturer is being queried concerning the possibility of supplying lamps having the same composition as those which were not darkened by irradiation.

Work is continuing on the design of the component devices required to decan the individual fuel elements. The detailing of Mark II Spiral Decanner and the Fuel Element Magazine are nearing completion. The other parts of the complete machine are in layout design.

The design is being reviewed for feasibility of fabrication procedures and for possible ways of reducing fabrication costs. Particular attention is being given to the adequacy of the specifications and tolerances in order to assure the remote replacement of subassemblies.

The design of refabrication equipment for irradiated EBR-II fuel is continuing and currently is estimated to be 45% complete. The design of shielded cell feed-through fittings and injection casting furnaces is advanced to the point where the specifications governing their procurement are being written and the drawings checked. Performance tests to define the sensitivity of the leak detection station have been completed and the comparison between the pressure decay method and the helium mass spectrometer method is being developed from these tests. The radiation testing of the quartz crystal piezoelectric transducer has been discontinued because of its instability in radiation fields. In comparison, the strain gauge type transducer is still functioning satisfactorily after a total dose exceeding 5×10^8 roentgen and in radiation fields of 10^6 roentgens per hour.

9. Fuel Development and Fabrication

The assembly of EBR-II fuel elements was continued during the month. Fourteen hundred new elements were bonded along with 575 reloads. Over 1600 elements were accepted to bring the total ready for subassembly loading to 7400. (9350 finished elements are required)

The operations of fuel pin casting, pin processing and inspection have been essentially completed except for replacement of any fuel pins destroyed or rejected in subsequent operations. Fuel element assembly was started during the reporting period.

During the month of November, an additional 3,434 inner and outer blanket rods were assembled. Of these, 61 (1.8%), were rejected due to leaky welds, faulty sodium bond or other defects.

A total of 4,139 inner and outer blanket rods were eddy current inspected. Initially, 6 rods were rejected for low sodium level, 3 for high sodium level and 125 for nonbond. The latter group was recycled in the bonding furnace and 111 of these rods were eventually acceptable.

The total of assembled inner and outer blanket rods is 15,329 to date. The total accepted figure is 14,626. This is roughly 3,500 blanket rods in excess of initial reactor requirements. The extra rods will be placed in storage.

The following is a summary of blanket element status:

	<u>Assembled</u>	<u>Requested</u>
Inner blanket elements	72	72
Outer blanket elements	0	548
Upper blanket elements	46	79
Lower blanket elements	46	79

Reclaiming of uranium from rejects has been resumed.

10. Fuel Prototype Irradiation

The fuel irradiation specimen CP-5-19 was removed from CP-5 and examined after removal from the irradiation capsule. The computed burnup for this specimen was slightly more than 2% of total atoms. Also, the maximum central metal temperature was in the vicinity of 1100°F for approximately the final fifth of the irradiation capsule. However, upon visual examination it was immediately apparent that severe damage had been suffered by the element. It was noted that the top three and one-half inches containing the restrainer was bent. In addition, the cladding was penetrated just below the top of the uranium with a small quantity of eutectic present. Also, the cladding was split longitudinally for several inches starting about two inches from the bottom and extending to approximately the mid-section of the element. The external diameter of the cladding was measured at a maximum of 0.208 in. at the split section while above and below it was at the normal 0.173 - 0.174 in. It was not possible to remove the cladding from the uranium since it adhered tenaciously. A gamma scan was made of the element prior to further examination.

11. Physics

a. Long-Term Effects - Calculations of the reactivity worth of plutonium built up in the blanket of EBR-II indicate that for full power reactor operation at 80% plant factor for one year the reactor core size will be reduced

by three or four fuel subassemblies, assuming that enrichment of the fuel alloy is held constant. By this time a maximum burnup of about 0.5 atom-percent would have resulted in the innermost subassemblies of the inner blanket.

After about three years of full power operation, approximately 1 atom-percent of plutonium should be built up in the inner portions of the outer blanket, at which time the burnup in the same region should be less than 0.1 atom-percent. This plutonium buildup would double the power generation in these regions and may present a heat removal problem.

b. Power Distribution - In the past differences have existed between the theoretical predictions of power distribution and the measurements made on the EBR-II mockup in ZPR-III. The few group two-dimensional calculations predicted a higher peak density than that measured. A multi-group, two-dimensional calculation has recently been completed and gives somewhat better agreement with experiment but the data are not sufficiently detailed nor precise to resolve the discrepancy. Hence, there is something like a 7% uncertainty in the determination of maximum local core power, given the total power output of the reactor.

c. Possible Hazard from Fuel Element Failure in EBR-II - Results from TREAT experiments suggest that if an EBR-II fuel pin contained and bonded inside the stainless steel clad is rapidly heated to above 1015°C , cladding failure could take place. The design temperature on outside fuel alloy clad is about 500°C . The axial temperature distribution in the reactor suggests that the clad would fail near the core midplane and most of the molten fuel might be ejected at this point including that from the pin extremities. Ejection would be caused by the presence of vaporized bond sodium at high pressure. Such ejection would result in an increase in fuel alloy density near the core midplane, yielding a reactivity increase, the magnitude of which would depend on the number of failed fuel elements.

Diffusion theory calculations in two-dimensional geometry suggest that averaged over the core, such failure brings an increase of 1.2 inhours per failed fuel pin. Near the core centerline a single pin failure would result in a 2.0 inhour reactivity increase. It is quite apparent that if a large number of pins were to fail simultaneously, this could be the source of a large reactivity insertion rate. However, the experiments basic to this conclusion were performed with a single EBR-II fuel element in an inert atmosphere with a flat axial flux. A realistic evaluation must await similar TREAT experiments conducted on EBR-II fuel elements suspended in flowing sodium and with a buckled axial flux distribution.

12. Process Development

a. Melt Refining Process Technology - Final analytical results have been obtained from the first melt refining experiment using highly irradiated (0.5 total atom percent burnup) enriched uranium-fissium alloy. A 388-gram charge of 40-day cooled EBR-II type fuel pins was melted and held in a lime-stabilized zirconia crucible in an argon atmosphere for three hours at 1400°C . Fission product decontamination factors were as follows: rare earths and yttrium, 123; cesium, 197; tellurium, 21.5; barium-strontium 2.25×10^5 ; iodine, 678; zirconium, 1.1. Further high-level melt refining experiments are awaiting the irradiation of additional fuel in CP-5 and MTR.

The data from two fission gas release experiments using uranium-five percent fissium pins irradiated to a total atom burnup of 0.6 percent have been analyzed. On heating the pins, the bulk of the krypton-85 and xenon-133 was released over the temperature range of 750° to 1025°C. The temperatures at which gas evolution occurred were not affected significantly by doubling the heating rate. In both experiments the rapid release of gas was accompanied by an increase of about 50 percent in pin diameter. A comparison of the amount of krypton-85 evolved during aqueous dissolution of a sample of the irradiated fuel showed that 98.5 percent of the krypton-85 present in the pin was released in the heating experiment.

Further studies were conducted on the nitridation of unirradiated uranium and uranium-five percent fissium pins, both uncoated and coated with sodium. Although significant, the rates of nitridation in an argon atmosphere containing four percent nitrogen were much lower at 197°C than in previous experiments at 308° and 504°C. In all cases, the rates of nitridation of sodium-coated pins were higher than those of uncoated pins. Calculated values of the nitride film thicknesses after 24-hour exposures at 197°C and 308°C varied from 0.5 to 1.0 micron in the case of the uncoated pins, and from 0.9 to 2.5 microns in the case of the sodium-coated pins.

A second high yield (96.5 percent) was obtained in a melt refining run employing sodium-coated fissium pins. (Results of the first high-yield run were reported in the September Progress Report, ANL-6234.) Work has been resumed on development and testing of procedures for counteracting the effects of formation of crusts on melts and of the formation of straw-like pin shells which result from oxidation or nitridation of fissium pins.

b. Processing of Melt Refining Skulls - The dragout process for recovery of the skull material remaining in zirconia crucibles after a melt refining operation involves oxidation of the skull material to form a free flowing oxide powder, reduction of the oxide by magnesium in liquid metal media, purification by utilizing solubility differences between uranium and impurities in various liquid metal solutions, and retorting to isolate a uranium metal product.

Equipment for study of behavior and methods of handling volatile oxides formed during skull oxidation has been constructed. Material has been prepared for irradiation in order to make runs with radioactive material.

The demonstration run mentioned in the October Progress Report (ANL-6253) involving reduction of a cerium-fissium skull oxide by a 5 percent magnesium-zinc solution was terminated because of incomplete, although continuing, reduction (81 percent complete after five hours).

Some of the future demonstration runs of the dragout process will employ a molten chloride flux which, if properly chosen, not only accelerates the reduction of skull oxide but also extracts the magnesium oxide reaction product. A skull oxide reduction run was made in the presence of a molten salt in order to gain experience in such reductions.

The extent of magnesium oxide transfer to a molten salt flux (sodium chloride - potassium chloride - sodium fluoride) was determined and values of 91 to 96 percent were obtained. The flux, which had been chosen to facilitate analytical work, was found to be unstable under process conditions, and will not be used further. It is believed that magnesium oxide transfers can be made quantitative by use of a more suitable flux, and measurements which are expected to substantiate this belief are now being made. A systematic investigation is in progress to establish an optimum flux composition for (1) the promotion of the reduction reaction (2) the removal of solid magnesium oxide and (3) the possible removal of certain fission products.

c. Engineering Development for Liquid Metal Processes - In small scale experiments (about 20 g fissium skull oxide) filtration was shown to be an effective method of separating from the liquid metal supernatant solutions the uranium-zinc intermetallic precipitated out of a zinc-rich solution and the uranium metal resulting from decomposition of this intermetallic by addition of sufficient magnesium. Two experiments, which in effect constituted small-scale demonstration runs of the dragout process, indicated that unavoidable losses of uranium in the flux and metal filtrates would be less than 2 percent of the charged amount.

In two small scale experiments (about 340-gram charges of metal), 94 and 99 percent of magnesium oxide was removed from liquid magnesium-zinc solutions by filtration. The scale of the experiment was too small to encounter filter plugging which had occurred previously in this separation. In the present runs, the filtration was carried out with about 50 grams of metal per square centimeter of filter area using a grade 40 graphite frit in the first run and a grade 60 graphite frit in the second run. However, in both runs large metal holdups were experienced which, in plant operation, would require a metal washing operation to remove the contained uranium.

d. Blanket Processing - Work has been started on the direct low-temperature interaction of blanket uranium and zinc to form a uranium-zinc intermetallic compound (epsilon phase). On subsequent decomposition of this compound with magnesium, the plutonium would be dissolved in the magnesium-rich phase in which the uranium is insoluble. At the temperatures suitable for this direct reaction (about 450°C), reaction rates have been low, requiring about 80 hours for complete reaction of a 3/8-inch diameter uranium rod. Because of the bulk of the uranium rods, the use of ordinary stirring methods to speed up the reaction is difficult and, perhaps, may even be ineffective. An ultrasonic stirring method which might be applicable to the system under study has been utilized by Aeroprojects, Inc., to speed up the dissolution of uranium in zinc.

e. Materials Evaluation Studies - A program is under way to test the corrosion resistance of refractory metals, graphite, and ceramics to molten metal systems of interest. Equipment is being assembled for the testing of the corrosion resistance of various container materials to molten salts.

III. STUDIES AND EVALUATIONS (040116)

A. Fast Reactor Fuel Value Study

The Laboratory's efforts have been almost completely in the physics studies.

The reactor studies involved systems generating 800 Mwt from the core. The metal fueled cores operated at a power density of 1 Mwt/liter and had the composition 6% Fe, 14% Nb, 2% Mo, 28% U-238 + Pu and 50% Na. The oxide fueled cores operated at a power density of 0.533 Mwt/liter and had the composition 25% Fe, 25% PuO₂-UO₂ (density = 8.5 gms/cm³) and 50% Na. In all cases, a 45 cm thick blanket having 20% Fe, 60% natural U and 20% Na was used. The allowable burnup assigned to the metal cores was 5% and to the oxide cores was 10% of the original fertile plus fissionable material. The results of the sixteen cases studied, using an assumed plant life of 30 years and an 80% plant factor, are presented in Table IV. In the uranium cores, the composition changed continually. In the plutonium cores, equilibrium was reached in about 10 years. The critical mass includes fissionable isotopes only, U-235 Pu-239, Pu-241.

Table IV. Fast Reactor Fuel Value Study Results

A. Metal Core									
Feed Composition				Equilibrium Composition				Critical Mass	Breeding Ratio
U-235		U-236		U-235		U-236		kg	
1.0		0.0		0.672		0.328		705	1.22
Pu - 239	240	241	242	Pu - 239	240	241	242		
1.0	0.0	0.0	0.0	0.676	0.277	0.036	0.010	420	1.80
0.4	0.1	0.25	0.25	0.544	0.260	0.068	0.129	396	1.87
0.95	0.05	0.0	0.0	0.682	0.269	0.038	0.010	420	1.82
0.752	0.195	0.049	0.004	0.624	0.313	0.048	0.015	412	1.85
0.53	0.301	0.143	0.026	0.546	0.361	0.064	0.029	396	1.91
0.313	0.336	0.250	0.101	0.532	0.326	0.078	0.066	394	1.91
0.416	0.246	0.136	0.202	0.502	0.337	0.058	0.103	396	1.90
B. Oxide Core									
Feed Composition				Equilibrium Composition				Critical Mass	Breeding Ratio
U-235		U-236		U-235		U-236			
1.0		0.0		0.592		0.408		724	1.15
Pu - 239	240	241	242	Pu - 239	240	241	242		
1.0	0.0	0.0	0.0	0.590	0.335	0.052	0.023	472	1.66
0.4	0.1	0.25	0.25	0.339	0.282	0.108	0.270	407	1.79
0.95	0.05	0.0	0.0	0.568	0.351	0.056	0.025	471	1.69
0.752	0.195	0.049	0.004	0.459	0.425	0.078	0.038	446	1.77
0.53	0.301	0.143	0.026	0.357	0.469	0.106	0.068	421	1.84
0.313	0.336	0.250	0.101	0.285	0.444	0.129	0.142	393	1.91
0.416	0.246	0.136	0.202	0.389	0.314	0.092	0.206	407	1.86

Contacts have been established with the Hanford group. It appears that some of the Hanford codes under development will be worthwhile for the fast reactor work. This is particularly true for the economics phase of the over-all effort. The Hanford group has agreed to furnish the codes as soon as they are ready.

B. Nuclear Superheat Studies

A survey of reactor concepts which may be useful as external superheaters has started in order to determine if one group of reactors has outstanding potential for advanced superheat systems. Three basic reactor types are being investigated:

(a) Reactors whose fuel elements are cooled by direct contact with steam.

(b) Reactors whose fuel is dispersed or lumped within a solid matrix with steam cooling in separate passages.

(c) High temperature reactors that would superheat steam in an indirect cycle external to the reactor.

The reactor selection criteria are first, the technical feasibility for the given concept to meet the desired operational conditions, and second, the economic promise of the concept. In general, however, the projected cost advantage attributed to superheating is small so that uncertainties in cost estimating for nuclear plants and fuel cycles can make a significant change in the incremental savings over the power costs from advanced reactors producing saturated steam. Also, the steam conditions that correspond to the minimum power cost point vary from one reactor concept to the next depending upon individual degrees of optimism with regard to factors that are not considered current technology. Consequently, to establish ground rules for this study, it was arbitrarily decided that those reactors which could realize modern steam plant conditions and also reach high thermal efficiencies before nuclear or materials limitations were imposed would ultimately show the best economic gain and be able to profit most from future turbine improvements or an increased need for fuel conservation. The operational limits selected are that the nuclear superheater will take saturated steam at 69.8 atm (1000 psi) and superheat the steam to 458°C (1060°F).

C. Miscellaneous

Comments on Volumes 1, 3 and 4 of the Nuclear Power Plant Cost Evaluations Handbook were prepared and transmitted to AEC-Washington.

An evaluation of a proposal for the study of a high temperature liquid metal-cooled reactor was started in the latter part of the month.

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.

Additional runs with 60-mil zirconium wires in the high pressure condenser discharge apparatus with the pressure range extended to 900 psi indicate that the reaction is unaffected by pressure in the range from slightly above atmospheric to 900 psi.

The aluminum-water reaction is being studied by the pressure-pulse method. Apparatus for operation with one atmosphere of steam was completed and tested. Runs at 1000°C showed that approximately 0.1 standard cc of hydrogen per sq. cm. of metal surface was formed during a contact period of one second.

Particle size measurements were made on uranium wire which had been melted and dispersed under water by the TREAT reactor bursts. The particles of most frequent occurrence have diameters in the range 4 to 8 mils; this is to be compared with the original diameter of 64 mils. However, the presence of larger globules resulted in Sauter mean diameters (6 x volume/surface area) ranging from 17 to 78 mils. A correlation was observed between the extent of uranium water reaction and the change in specific surface area; as the ratio of final area to initial area increased from 1.0 to 5.4, the percent of metal-water reaction increased from 0.4 to 17.2 percent. Preparations were made for the tenth series of in-pile experiments; ceramic and stainless steel-uranium dioxide cermet fuel pins will be studied.

The differential thermal analysis apparatus for the study of the reaction of aluminum with uranium oxides has been completed.

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

Additional studies of the ignition temperature of uranium powders by the burning curve method showed ignition temperatures in air to be about the same as those in oxygen. Ignition temperatures were found to increase with heating rate in accordance with theoretical predictions. Ignition temperature values are in good agreement with theoretical values calculated by the theories of Frank-Kamenetskii and Murray, Buddery, and Taylor.

A theoretical model of burning propagation along metal foils in air has been constructed. It was assumed that reaction rate is determined by the rate of diffusion of oxygen through the nitrogen of the air. Preliminary comparison of calculated rates of propagation to experimental values were within 30 percent for uranium foils. Calculated values for zirconium burning, however, were far lower than experimental values.

Recent data show no significant effect of surrounding temperature on burning propagation velocities of uranium up to ambient temperatures of 300°C. With 5-mil zirconium foil a 40 percent rate increase occurred at an ambient temperature of 400°C. No significant increase occurred, however, when 1-mil foil was studied in a 400°C environment.

Because the presence in air of certain halogenated hydrocarbons has prevented burning or reduced the rate and burning temperature of single specimens, a program was undertaken to determine the practicality of utilizing halocarbons as a protective atmosphere for storing and shipping zirconium scrap. A few experiments on one-gallon pails of swarf zirconium, although lacking rigid controls of uniformity of scrap and of non-standard fires, indicated that it was much more difficult to ignite scrap in a can containing dibromomethane vapor, and that the intensity of burning of the scrap was greatly reduced when dibromomethane was added.

Experiments on ignition of plutonium alloys in air contaminated with certain halocarbons have been completed. It appears that the advantageous effect of halocarbon contaminants in air to prevent or decrease the burning propagation must be accompanied by the disadvantage of an exothermic reaction at a lower temperature than the ignition temperature in pure air. This reaction results in a sample self-heating of 50 to 100 degrees in contrast to 300 to 400 degrees observed upon ignition.

B. Fast Reactor Safety Studies

1. TREAT - Experimental

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

a. EBR-II, Mark I - Dry Samples - In the Progress Report for August, 1960, (ANL-6215, p. 36) experiments were reported in which dry prototype EBR-II fuel elements were subjected to flattened power bursts of about 2 sec duration. The results showed only a slight qualitative difference in sample failure when compared against the results of earlier tests using short clipped reactor transients. The clipped bursts were of the same integrated power (about 110 Mw-sec, sufficient to produce extensive failure) and were of 0.2 sec duration. Cladding was observed to have failed, relatively uniformly, along lengths of 2-5 cm, with essentially all the alloy ejected from the cladding by internal pressure. The sodium originally present to produce a thermal bond between fuel and cladding is vaporized causing internal pressures to develop. However, the violence of expulsion appeared to be somewhat less than that noted for clipped power bursts of the same energy release.

Six meltdown experiments (Series XV) were performed in which the reactor was programmed for flattened power transients of durations in the range of 5-22.5 sec. Total integrated power values were set so that the sample energy inputs, minus the heat energy transferred to the walls of the meltdown capsule during the transients, would be sufficient to produce extensive failure if introduced during a short clipped transient. Sample temperature records from fast-response cladding thermocouples indicated that some fuel melting occurred during the transients, but gave no clear evidence of sudden, rapid failure. All six elements are now being shipped back to the Lemont site for post-mortem examination.

The remaining two niobium-clad, natural enrichment EBR-II elements in Series XIII have been run at TREAT. All transient experiments thus far on refractory metal-clad EBR-II elements have employed short (of 0.4 sec duration or less) temperature-limited reactor transients. In Series XIII, total energy inputs were as much as 70% greater than that necessary to cause fuel melting in such elements.

b. Oxide Fuel Meltdown Experimentation - Meltdown experiments on uranium oxide fuel samples are being planned as the first step in extension of the Fast Reactor Safety Program to non-metallic fuels. Oxide samples totaling about 2160 linear centimeters will be fabricated by the Metallurgy Division. Approximately 1780 cm consist of oxide pellets, with the remainder to be powder compacted inside cladding tubes. Stainless steel, tantalum and niobium cladding tubes are being prepared for the powder-compacted samples. All oxide now ordered is to be used inside EBR-II, Mark-I size cladding. Ten percent enriched uranium will be used for both types (pelleted and compacted oxide) of oxide. In addition, some 3% enriched pellets are included in the 1780 cm for use in experiments on the behavior of elements under conditions in which the central temperature is relatively

moderate. Two density ranges of pellets have been specified: 65-75% theoretical density, and 85-90% theoretical density. Both ranges lie in the regions of interest for proposed high burnup fuel elements, and both provide possibilities for appreciable oxide density changes due to internal sintering, melting, etc.

c. Experiments on Meltdown in Stagnant Sodium - Four stainless steel clad EBR-II elements, contained in stagnant sodium, are scheduled for the next series of meltdown experiments. This series will be the first survey into meltdown phenomena of elements submerged in sodium. The first capsule, which has been assembled, contains a pressure transducer for measurement of vapor pressure of the stagnant sodium bath. This capsule will be used for four mild "check" transients prior to the three meltdown experiments proper, each of which will have a separate capsule equipped with a thermocouple embedded in the sample instead of the pressure transducer. The internal sample thermocouples are sheathed with tantalum tubing and extend into a hole drilled axially in the fuel cylinder at its top. Assembly of the remaining three capsules is now under way. All four capsules are equipped with electric heaters to permit preheating the capsules and sodium to operating temperature. In the case of the first series, the sodium bath is specified to be preheated to 260°C.

d. Internal Sample Temperature Measurement - Development of an acceptable internal thermocouple for instrumenting EBR-II sample elements would afford a method of measuring internal sample temperatures in both dry and sodium-cooled experiments. The relatively simple thermocouple arrangement under development for the first stagnant sodium experiments is bonded with sodium to the sample element, hence it is primarily a device for measuring temperatures prior to "incipient failure." A total of six natural enrichment elements with tantalum-molybdenum sheathed in 0.114 cm tantalum thermocouples have been received, checked, and are ready for auxiliary brazing operations.

2. Out-of-Pile Meltdown Studies

Work continued on out-of-pile meltdowns of EBR-II, Mark-I elements, using an electric furnace and a peak temperature of 900°C. Failure by penetration of the cladding occurred in 300 to 600 seconds. These results were consistent with those of runs reported earlier. Cladding failures were the size of pin holes, located over large spots of eutectic. Experiments are being planned to study the effects of enforced contact between fuel and cladding.

Attempts were made to follow the formation of eutectic prior to cladding failure by using non-destructive testing techniques. An element was run in the furnace, removed, cooled, and the bond was quenched. The sample was then scanned with a "Cyclograph" eddy current coil. Although regions of eutectic were found to produce changes in the Cyclograph record, the techniques used were not sufficiently sensitive to permit a sample to be exposed in the furnace for several runs, with positive non-destructive inspections being performed between runs. Modification of the cyclograph inspection for this particular application has been postponed because the instrument is being used full time for inspection of the sodium bonds in EBR-II elements now being fabricated.

3. Meltdown Configuration Studies

One aspect of the theoretical work in Fast Reactor Safety deals with the possible formation of meltdown configurations which initially are autocatalytic in the disassembly associated with an explosion. The possibility of using two-dimensional diffusion theory for the prediction of the behavior of such assemblies has been studied by comparing the results of PDQ calculations with the experimental data of ZPR-III for assemblies Nos. 27 and 28. These two mockups of Fermi meltdown configurations themselves exhibit no autocatalytic character. However, if theory and experiment were reasonably close for these two assemblies, theory could then be used for studying other more hazardous configurations. Excellent agreement on both power and danger coefficient distributions were obtained for assembly No. 27. For assembly No. 28 fairly good agreement on power distribution was obtained and reasonably good agreement for danger coefficient measurements in the middle of the dense core region. However, near the interface of the dense core region and the void region, where the flux gradients are the largest and the diffusion theory approximation the worst, theory gave not only the wrong magnitude but the wrong shape. Hence, while the diffusion theory method has some applicability in such studies, caution will be required in evaluating its results.

4. TREAT - Operation and Development

The 3rd and 4th pins of the GEANP series were run and returned to the ANP site for examination and analysis. Following the irradiation of the capsule containing pin 3, radioactive contamination was detected in the reactor building. This was cleared by means of the reactor ventilating system and natural decay. Building background level is now essentially normal. Inspection of the capsule subsequent to the irradiation indicated that the leak probably occurred in the final evacuation tube seal weld. Future capsules will contain a helium atmosphere and will be mass spectrometer leak tested after final welding.

A d.c. integrating-type circuit using an ion chamber as a detector was connected in the reactor shutdown system in lieu of one integrating fission-counter circuit. A second circuit is now being built to replace the other fission-counter circuit. Extensive testing of the current-integrating circuit prior to using it in the reactor shutdown system proved that it was superior to the counter-type circuits which have a limited dynamic range.

The mockup for the transparent capsule assembly was tried in the reactor core. Insertion and removal of the container was found possible after minor modification.

V. NUCLEAR TECHNOLOGY AND GENERAL SUPPORT (040400)

A. Applied Nuclear and Reactor Physics

1. Experimental

a. Neutron Induced Fission Cross Sections of Pu-241 and Am-243 -

The neutron induced fission cross sections of Pu-241 and Am-243 have been measured from 0.030 to 1.8 Mev and from 0.3 to 1.6 Mev, respectively. Both measurements were made relative to the fission cross section of U-235 using a back-to-back gas scintillation counter. The masses of the plutonium and americium samples were determined by alpha counting. The mass of the U-235 sample was determined by quantitative electrodeposition. Neutrons were produced with a Li-7(p,n)Be-7 target using protons from a 3-Mev electrostatic generator. The Pu-241 cross section was found to have an energy dependence similar to that of U-235 above 200 kev and a value of 1.8 barns at 1.0 Mev. The Am-243 cross section was found to rise from a value of 0.01 barns at 0.3 Mev to a plateau of 1.2 barns above 1.3 Mev.

The work will continue in order to provide the more accurate knowledge of the neutron induced fission cross section necessary for optimum reactor design.

b. Fast Neutron Reactions - Tungsten is a unique substance possessing metallurgical properties that make it suitable for applications demanding structural strength at very high temperatures. For this reason the scattering of fast neutrons from tungsten is being studied. The experimental technique utilizes the fast time of flight method to give good energy resolution and sensitivity.

The differential elastic scattering cross section of tungsten over a range of incident neutron energies is graphically illustrated in Figure 9. The differential elastic cross section is shown as a function of angle and incident neutron energy.

The experimental work has resulted in a detailed understanding of inelastic scattering from tungsten. Figure 10 summarizes the results. Natural tungsten is characterized by nuclear levels at 120 kev, about 350 kev, 630 kev, 750 kev, 930 kev, 1030 kev, and 1130 kev. In Figure 10 the differential cross section for the excitation of each of these levels by inelastic neutron scattering is given as a function of incident neutron energy. The angular distribution of the inelastically scattered neutrons is essentially isotropic. The solid curves in the figure are the result of theoretical calculation based upon the Hauser-Feshbach compound nucleus average cross section formula. The agreement between theory and experiment is remarkably good.

c. High Conversion Critical Experiment - Several additional calculations required for the hazards summary report have been completed, as a consequence of the availability of the BORAX V boiling region fuel elements and the possibility that a few measurements would be made on a superheat region similar to the BORAX V design. The BORAX V boiling region fuel offers an opportunity to obtain measurements consistent with the Hi-C program at an additional enrichment. A mockup of the superheat

Figure 9

Differential Elastic Scattering Cross Section
of Tungsten as a Function of Angle and
Incident Neutron Energy

Figure 10

Differential Inelastic Scattering Cross Section of Tungsten

region in BORAX V would provide some information needed urgently in regard to the power production patterns and the reactivity effect of flooding.

All parts of the core structure required for the conversion of the ZPR-VII facility have been received, and installation is progressing. The facility is ready for loading of fuel, consequently the time schedule for startup will be determined by the processing and approval of the hazards summary report.

d. Critical Experiment Techniques - It has been proposed that a measurement be made of the temperature coefficient of reactivity of the present EBWR core loading by the introduction of a quantity of cold water into the system while at a temperature moderately above ambient. To test the feasibility, the EBWR system was brought to equilibrium at a temperature of approximately 50°C and 1900 liters of cold water were added. Measurements of the temperature distribution in the core during the heating and the addition of cold water indicated the need for a more thorough mixing if a uniform temperature distribution is to be established by this method in a reasonable length of time.

Measurements made previously of the cadmium ratio as a function of height in the original EBWR core at operating conditions showed that the axial distribution of void content could be deduced from the variation in cadmium ratio. An investigation of the adaptability of this technique to the determination of growth in the central thin enriched fuel elements during the proposed operation of EBWR at high power levels is under way. The effect of such growth would be to decrease the width of the water channel with an attendant reduction in the amount of moderation in the immediate vicinity. This decreased moderation would be enhanced further by the increased void content resulting from decreased flow of coolant in the vicinity of a fuel plate which had experienced growth under irradiation. Two very small U-235 fission counters are available for possible insertion at appropriate locations in the core in the hope of detecting by a change in cadmium ratio any growth and impending failure of the fuel elements.

e. Macroscopic Pulsed Neutron Studies - During the month the time dependent decay of the thermal flux in ZrH_2 assemblies (rectangular parallel pipes of packed ZrH_2) having bucklings ranging from 0.03 to 0.1 cm^{-2} was measured. A series of $1/V$ transmission measurements utilizing the leakage flux from a ZrH_2 assembly, having a $B^2 = 0.03 \text{ cm}^{-2}$ were completed. The data are now being analyzed. These studies of the macroscopic properties of the neutron-molecular reaction in ZrH_2 are of particular interest owing to the unusual structure of this material which particularly lends itself to the study of the reaction.

f. Thermal Reactor Spectra - Emphasis has been placed recently on the investigation of two interrelated problems which represent major obstacles for the successful determination of thermal reactor spectra with the Selove mechanical chopper. Effort has been made to reduce the background counting rate and to improve the efficiency of neutron detection in order to achieve a signal-to-noise ratio sufficiently high to warrant a continuation of these measurements.

The addition of a large amount of shielding around both ZPR-IV'

reactor (which is being used as the neutron source for the preliminary experiments designed to check out the equipment) and the neutron beam detector has reduced the background counting rate to 1/60 that which existed previously. By blocking the entrance to the counter shield with paraffin backed by cadmium, the counting rate was reduced an additional factor of 100, indicating that gamma rays do not account for an appreciable portion of the background. To date it has not been possible to detect with certainty a difference in the counting rate with the chopper opened and closed. Much of the remaining background seems to be coming from the reactor itself in the vicinity of the chopper, and so it may be necessary to shield the entire flight path with a combination of paraffin and cadmium. In all likelihood, the neutron background from Juggernaut (where the proposed experiments will actually be performed) will be less severe.

An attempt has been made to produce a slow neutron scintillator composed of a lithium fluoride-zinc sulphide mixture in lucite. Hope for improved efficiency and added discrimination against gamma radiation by such a combination is based on results reported recently at Chalk River. The first scintillator produced contained natural lithium and had a sensitivity about half that of a commercially available boron-zinc sulphide unit. It is anticipated that the sensitivity will be improved as experience is acquired in producing discs with fewer visible irregularities, and by corrugating the discs for better light coupling with the photomultiplier tube. Use of 90% enriched Li-6 and the above improvements should produce a detector with an efficiency for thermal neutrons of about 20%.

In addition to the problems related to the achievement of an adequate signal-to-noise ratio, there exists a question of whether the beam extracted from a moderating medium is representative of the spectrum existing in that medium. Calculations to date on a six-energy-group basis offer promise of providing information on the distortion of the spectra by a beam hole. Because of several uncertainties in the calculation and the limited availability of computer time, preparations for an experimental determination have been undertaken. A right circular cylindrical tank of 75-cm diameter will be used with a range of beam holes from 2.5-20 cm diameter. The emergent neutrons will be collimated and a foil changer will allow the introduction of gold foils between the source of the beam and a BF₃ counter. These measurements will be made with light water using the Argonne Thermal Source Reactor (ZPR-IV). While it will be possible to deduce spectra from such measurements, the desired information regarding a dependence of spectrum on beam hole size might be obtained just from the transmission measurements.

g. Measurement of ν - Current effort to determine the average number of thermal neutrons per U-235 fission absolutely to 1% precision are concentrated on preparations for preliminary experiments to be performed at the CP-5 reactor during the month of December. These included plating of uranium for the foil to be inserted in the fission counter, assembly and testing of the manganese bath tank, preparation of the shield tank, and construction of shielding materials required for use at the thermal column of CP-5 and improved collimating elements to replace existing thermal column plugs. The fission counter and Hornyak-button type of fast neutron detector and their associated circuitry will be available

for these preliminary measurements, although the components for coincidence counting will not. It is anticipated that these measurements will determine the neutron fission rate and fast neutron detector efficiency, the background rates, the induced activity per unit volume of the manganese bath solution, the degree of suppression of precipitation of the manganese, and the effectiveness of the collimator shielding and shield tank. These results will serve as a basis for making final plans for the actual measurement.

A circuit designed to discriminate against narrow background pulses while retaining rise time information for ZnS(Ag) decays has been tested. It appears to have nearly the same discrimination capabilities as long integration time constant circuits, yet provides the important timing information. Further improvements by fine adjustment, by use of uni-tunnel diodes, and by use of better delay cable may improve the discrimination characteristics.

One further attempt to reduce background from the Hornyak button was unsuccessful, although additional information was obtained which leads to better understanding of the problem. It was found with a high intensity α -source that fluorescence of lucite (and other materials) was an important mechanism contributing to background. Filters were used in the vain hope that the emission spectra would be in wave lengths shorter than that of ZnS(Ag) , but apparently the emission is characteristic of impurities which have emission bands in the blue.

h. Calibration of ZPR-IV' - The renovated ZPR-IV' reactor, currently referred to as the Argonne Thermal Source Reactor, was restored to operation last May. However, additional control rod calibrations and reactivity effects of some loading changes desirable to achieve versatility in the use of this system have only recently been determined. The reactivity effect of replacing the heavy water thermal column by 9 cm of wood backed by 10 cm of lead has been established. The worth of fuel in the two core grid positions used for the auto-rod and the reactivity measurement neutron detector has been determined, and the differential calibration of the two control rods has also been measured. These measurements on ATSR provide a set of reference data for estimating the reactivity effects of changes which may be desirable in future use of this facility. The reactivity effect of voids in the central grid position when filled with water had been determined previously. The effect of adding a cadmium liner to the void tube in the central position (a change required if the facility is to be used for resonance integral determinations) remains to be investigated.

i. Pile Noise Analyzer - Because of widespread interest in the pile noise method for determining prompt neutron lifetime, an improvement of the circuitry used has been undertaken.

An a.c. powered pile noise preamplifier has been constructed and found to perform satisfactorily. A novel integrator which will provide linearity and stability within a few tenths of a percent has been assembled and tested. It uses a phantastron circuit which produces a square pulse, the duration of which is proportional to the input signal voltage. The phantastron is triggered periodically, and its output pulse opens a gate which passes a train of constant frequency pulses into a scaler which serves to accumulate

the integral.

Other aspects of the pile noise analyzer circuitry used previously will also be improved. The new design eliminates batteries, improves the tuned circuit performance at low frequency, provides an appropriate signal for the phantastron integrator, increases the range of the attenuator and makes ample use of feedback to improve stability and linearity. The entire improved analyzer has been assembled in breadboard fashion and tests indicate that all of the objectives have been attained.

2. Theoretical

a. Reactor Physics Constants Center - The preparation of the revision to Reactor Physics Constants (ANL-5800) continues to consume most of the effort of the Theoretical Physics staff.

b. Theory of Average Nuclear Reaction Cross Sections - In the past explicit formulas for average nuclear reaction cross sections have been derived under the assumption that the average channel widths $\langle \Gamma_c \rangle$ are very small compared to the average level spacing D . This assumption often breaks down for neutron channels at optical model maxima even at kilovolt energies. It has now been shown that the average compound nucleus cross section for entrance channel s and exit channel t with total angular momentum J is given $\pi \lambda^2 g_{Js} \langle T_s T_t / T \rangle [1 + \text{higher order terms}]$, where the average value of the resonance parameter $T_s = 2\pi \Gamma_s D J^{-1}$ is related to the optical model transmission factor T_s by $T_s = \langle T_s \rangle - \langle T_s^2 \rangle / 4$. This latter relation is shown to be valid if the reduced width amplitudes have random signs and if $\langle T_s \rangle < 2$. Since $\langle T_s \rangle \rightarrow 2$ as $T_s \rightarrow 1$, the effect is to enhance strength function peaks compared to transmission factor peaks. The higher order terms may, in the continuum region, approach $(2\pi)^{-1} \langle T_s + T_t \rangle$ but can there be taken into account by applying the Thomas correction factor to $\langle \Gamma_s \rangle$ and $\langle \Gamma_t \rangle$. The effect of these considerations on neutron capture and on compound elastic and inelastic scattering cross sections has been examined.

c. Mathematical Numerical Methods Analyses - An Algol procedure published by Forsythe⁽¹⁾ for solving systems of linear equations was corrected and extended to include the computation of the determinant. Since there was pressing need for a determinant evaluator for reactor design calculations, the routine was coded for the LGP-30 computer, and was found to work satisfactorily. A write-up and certification of the revised algorithm have been prepared for publication.

In continuing the investigation of new applications for the square root operation, the utility of approximations of the form $f^*(x) = a + b\sqrt{c+x}$ has been studied. Customary methods of forming approximations were unsatisfactory, since it is not convenient to interpolate with this form of function. However, it was found that specification of a trial approximation by specifying the locations of the two interior maxima of the error curve, and requiring that their sum be zero, led to a relatively easy expression for the parameters a , b , and c , and required the calculation of only four

(1) Communications of the Association for Computing Machinery, Vol. 3, p. 507 (1960).

points on the error curve, instead of the customary fifty or more. This saving reduced the computing time per iteration cycle to under two minutes on the LGP-30 compared to approximately 30 minutes for the interpolation methods. With this program, approximations to $\log_2 x$, $\arctan x$, $\arccos x$, and $\arcsin x$ have been computed in less than one hour for which the maximum errors agree in magnitude to 4 or 5 significant figures. Unfortunately, the square root approximation form does not give appreciably better accuracy for these functions than conventional rational approximations. The device of specifying approximations by the locations of the extrema of the error curve is more generally applicable, however, and can produce a significant saving of effort in producing polynomial, and some rational approximations.

B. Reactor Fuels Development

1. Ceramic Fuels

a. Refractories for Reactor Use - This program is concerned with the development and testing of ceramic fuels containing UO_2 in solid solution with various oxide derivatives with the objective of improving high temperature stability.

Investigation of the lanthana-urania system has been continued. Fourteen previously calcined compositions containing from 0 to 85 w/o urania were pressed isostatically at 50,000 psi. Compacts were sintered in air at 1675°C for 5 hours and in hydrogen at 1600°C for 5 hours.

X-ray and microstructure investigations indicated the cubic fluorite solubility limits were between 30-80 and 35-100 w/o urania for the air and hydrogen sintered compacts respectively. Sintered densities were, in general, higher for air-sintered than for hydrogen-sintered compacts. Accurate measurement of the lattice parameter of oxidized and reduced compositions is being made to determine its variation with stoichiometry.

b. Fabrication Studies of UC-PuC - In the development of a good production procedure for UC for use in UC-PuC, efforts to achieve large surface contacts between UO_2 and carbon were hampered by the lack of a high surface area UO_2 . This material is on order and is expected to be delivered momentarily. In the meantime the UO_2 on hand was mixed with petroleum pitch and reacted with very good results. UC content of better than 95% has been obtained thus far.

The following general procedure was followed: a calculated stoichiometric composition of UO_2 and petroleum pitch were ball milled with ethyl alcohol. The resulting slurry was air dried and granulated. The granulated material was pressed into pellets.

These pellets were fired in an argon atmosphere in a period of four hours. Maximum temperature reached was 1700°C with a soak for one hour at 1670°C to 1700°C . The fired pellets were cracked but had the gray metallic luster of the carbide. The X-ray diffraction powder pattern showed the material to be UC with a very faint UO_2 pattern showing.

c. Development of Methods for Melting and Casting UC-PuC - The purpose of this program is to produce prescribed shapes of plutonium-carbide

or uranium-carbide or combinations of both for detailed irradiation studies and to evaluate production methods for carbide melting.

Temperatures near 2100°C have been achieved in the high temperature furnace without difficulties. However, further heating has always resulted in arcing and weld amperage fluctuations. Resistance readings between the electrodes and ground were found to be less than one ohm when the furnace was operating at 2100°C as compared to 40,000 ohms at room temperature. Several steps have been taken to increase this resistance but none has been completely satisfactory. The internal graphite parts are being remachined and flame coated with alumina in the attempt to decrease high temperature leakage and arcing. The refabrication of the internal furnace parts and revised electrical connections is approximately 85% complete.

d. Urania-Thoria Bodies - In past months work on urania-thoria bodies has been concentrated on determining the temperature of solid solution formation in $\text{U}_3\text{O}_8\text{-ThO}_2$ compacts, the effects of composition and temperature on the densification and stoichiometry of air-sintered $\text{U}_3\text{O}_8\text{-ThO}_2$ compacts, and the oxidation behavior of air-sintered $\text{U}_3\text{O}_8\text{-ThO}_2$ solid solutions during thermal cycling. A report on some phases of this work appeared in the Journal of the American Ceramic Society, 43, 520-524 (1960).

Present efforts are directed toward the evaluation of X-ray lattice parameters, density and microstructures resulting from oxidation and reduction of air-sintered $\text{U}_3\text{O}_8\text{-ThO}_2$ solid solutions.

e. Uranium-Thorium Sulfide - Chemical analyses have been obtained which permit the evaluation of the systematic study run on US material homogenized and sintered in vacuum. As the homogenization temperature for one hour was raised from 1500°C to 1900°C , the amount of secondary phase (predominantly UO_2) dropped from 9.4 to 2.0 w/o while the S/U ratio of the US phase dropped from 1.029 to 0.970. Raising the firing temperature of the 1800°C homogenized material from 1700°C to 2000°C caused the secondary phase to decrease further from 2.4 to 1.3 w/o (1.3 w/o of UO_2 represents 0.1 w/o O_2 contamination). Since US fired at 1900°C - 2000°C is much more brittle and only slightly more dense than that fired at 1800°C it appears that the best US bodies should be obtained by an extended homogenization at 1800°C - 1900°C followed by sintering at around 1800°C .

US No. 15 material homogenized in both argon and vacuum has been fired to 1800°C for one hour in hydrogen, argon and vacuum. Firings in hydrogen resulted in melted and deformed pellets. The results of the argon and vacuum firings are best shown in tabular form:

Table V. Densification as a Function of Preparation and Firing Conditions

<u>Treatment</u>	<u>Density (% Theor.)</u>	<u>Open Porosity (%)</u>	<u>Closed Porosity (%)</u>	<u>Resistivity (microhm-cm)</u>
Argon homogenization				
Argon firing	91.6	0.2	8.2	172
Vacuum firing	87.8	0.6	11.6	161
Vacuum homogenization				
Argon firing	88.7	0.4	10.9	258
Vacuum firing	84.9	14.8	0.3	428

It can be seen that argon homogenization produces a 3% denser body than does vacuum homogenization for the same firing conditions, while argon firing produces a 4% denser body than does vacuum firing for the same homogenized material.

An apparatus has been set up to measure the room temperature electrical resistivity (see Table V) of the 0.250 inch diameter pellets used in the sintering studies. These compare with a value of 100 ± 30 microhm-cm obtained for a very pure, dense sample of US. The apparent wide range of 100 to more than 400 microhm-cm indicates that resistivity measurements might be used as a measure of physical and chemical variations in the bodies. The values in the table, for example, reflect open porosity rather than density and may indicate compositional variations as well.

2. Properties of Metals and Alloys

a. Properties of Pu-Fs Alloys- A program is in progress to determine the thermal properties of non-plutonium materials as a preliminary step in developing equipment to determine thermal properties of plutonium-containing materials.

Using U-Fs alloys (uranium as a stand-in for plutonium) it has been found that the isothermal drop calorimeter (see Progress Reports for September, 1960, ANL-6234; October, 1960, ANL-6253) gives a precision of $\pm 1\%$ or better to temperatures up to 450°C . However, the formation of steam adjacent to the hot specimen results in poor precision above 500°C in this type of apparatus. For this reason a copper block calorimeter is being built which will eliminate the steam problem. It is designed to yield enthalpy data with a precision of better than 1% at temperatures up to 1200°C .

b. Thermal Measurements of U-Fs Alloys - Under subcontract to the Laboratory, the University of Denver is determining the enthalpy of 3, 5 and 8 w/o U-Fs alloys (see Progress Report, September, 1960, ANL-6234). Calorimetric measurements have been made on pure uranium up to the first transformation temperature, and one sample has been run up to 900°C . Work is now being done on 5 w/o fissium alloys and determinations have been made up to 550°C .

c. Properties of Pu-1 w/o Al Alloy - Hot deformation studies under simulated reactor load conditions are in progress as part of an engineering evaluation of cast plutonium-1 w/o aluminum alloy slugs for the Mark IV loading of EBR-I. A specimen (1.5 in. x 0.205 in. diameter) contained in a Zircaloy can (0.225 in. ID) slumped in excess of 0.100 in. during fifty hours of testing in the temperature range 400 - 600°C , under a static load of the order of 30 psi. This specimen buckled, touching the can in several places. Another specimen slumped in excess of 0.200 in. in a similar test under a static load of the order of 40 psi. This specimen completely filled the 10 mil annulus resulting in intimate contact with the can. The data are being evaluated to determine the amount of slumping at 400° , 500° and 600°C as a function of time for loads ranging from 5 to 40 psi.

3. Corrosion Studies

a. Corrosion Resistant Aluminum Alloys - The development of a

mechanically strong, corrosion resistant alloy is continuing. Of the strengthened alloys based on A288 (1 w/o Ni, 0.5 w/o Fe, 0.1 w/o Ti, low Si), one containing 0.3 w/o Cr and 0.2 w/o Zr is the most promising to date. After 8 days soaking at 350°C, its ultimate tensile strength at 315°C is about 9000 psi. Initial results show about the same corrosion resistance as A288.

Small capsules made from the new alloy having the dimensions currently specified for EBWR Core II fuel tubes collapsed during a few days exposure to helium at 260°C, 600 psi. The creep rate is apparently high at about 40-50% of the ultimate tensile strength. New attempts to incorporate AlPO₄ particles into the wrought alloy have been started. To evaluate materials, a small test apparatus is being used to measure the time for 5% creep at 4000 psi and 290°C.

b. Lightweight Alloys for Liquid Mercury - The Lunar Reactor Plant Study (see Progress Report, September, 1960, ANL-6234, p. 69) requires the use of a lightweight tubing material to resist liquid mercury at elevated temperatures.

Tests have shown that commercially pure titanium suffered an unevenly severe attack in mercury at 370°C. The average rate during 30 days exposure was 11 mg/cm²-day. Titanium alloy 110-T (5 w/o Al, 2.5 w/o Sn) was partially attacked on the polished surfaces, showing adherent black powdery coatings. X-ray diffraction indicated the presence of HgTi₃.

4. Irradiation Studies

a. Development of High Absorbing Materials - An ideal control material would combine the following properties: large thermal macroscopic absorption cross section, significant epithermal and resonance properties, plus a chain of neutron absorbing isotopes, ease of fabrication, room and elevated temperature strength and ductility, irradiation stability, corrosion resistance, lightweight for rapid movement with minimum inertial effects, good heat transfer properties, and availability at low cost.

Materials now used extensively as the absorber material in reactor control rods include boron, hafnium, and cadmium. None of these materials is ideal. The main disadvantage of boron in the B¹⁰ (n,α)Li⁷ reaction which introduces two new atoms into the lattice. Each atom is larger and possesses different properties than the parent boron atom. For high flux, long life cores, radiation damage in boron-containing control materials can be expected to be significant and thus the suitability of boron materials for such applications is questionable. The use of hafnium produces a rod possessing desirable nuclear and mechanical properties and long life. However, hafnium is not readily available and is expensive. Cadmium and its alloys have low mechanical strengths and do not have adequate absorption properties in the epithermal region.

The lanthanons (rare earths) appear to offer greatest promise as a basis for development of a class of control materials that will not be subject to the disadvantages of present materials. Work is in progress to incorporate dispersions of these elements into various matrix materials. After physical and mechanical property determinations, the more promising combinations will be evaluated by long term irradiations at elevated temperatures.

At present, both 25 w/o and 38 w/o Dy_2O_3 have been dispersed in plates of X8001 aluminum alloy. Two tensile specimens will be machined from each plate.

A composite plate has been fabricated with 30 w/o Sm_2O_3 in the outer layers and an inner core containing a 20 w/o Gd-Sm-Eu oxide mixture plus 10 w/o Er_2O_3 . The matrix material in both the inner and outer layers is cobalt.

Specimens containing the lanthanon oxides in a $\text{CaO-Al}_2\text{O}_3\text{-SiO}_2$ glass matrix are in development. Two compositions are under development, 80 w/o Dy_2O_3 and a 80 w/o Gd-Sm-Eu oxide mixture. The fabrication technique is being varied in order to provide both single phase and two phase structures. Also, specimens containing 15, 50 and 90 w/o B_4C in $\text{CaO-Al}_2\text{O}_3\text{-SiO}_2$ glass will be produced.

5. Nondestructive Testing Techniques

a. Ultrasonic Techniques - In a study to determine the usefulness of Lamb waves for detecting unbonded areas, the following experiment was carried out. A plate 4 in. x 3 in. x 0.033 in. was cut from an 1100 aluminum block 8 in. x 3 in. x 3 in. The 4 in. plate was glued back into position in the 8 in. block. This arrangement is representative of a nonbond area while that over the solid block is representative of a good bond area. Lamb waves are generated over the nonbond area which acts as a 0.033 in. plate. As the block is scanned, the modes should disappear (no longer be generated) over the good bond area. It was found, however, that although the modes underwent a decrease in amplitude of 80 to 90 percent they did not entirely disappear when the sender and receiver were over the solid block. Theoretically, the frequency thickness curves extend to infinity but experimentally the curves were thought to be limited to thin plates. These results show that Lamb waves can be generated in "plates" of 4 in. thickness although their amplitude is very small relative to that in a plate of a few mils thickness. These tests over the nonbond area give no idea of the resolution to be expected using Lamb waves. The resolution problem will be investigated later.

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. All the information previously reported was obtained using a monochromatic (1.05 Å) neutron beam from the neutron spectrometer at CP-5. Further studies have now been made using a Van de Graaff particle accelerator as a neutron source. The advantages of using this neutron source include the yield of a uniform neutron flux over a larger inspection area and the opportunity to investigate (1) the effects of a fast neutron beam and (2) the effects of a "white" thermal neutron beam (one containing many neutron energies).

The initial efforts using the Van de Graaff did produce several neutron radiographs. However, difficulties were encountered in the problem of low effective neutron flux, the effects of gamma radiation in the beam, and poor image sharpness. The effective neutron flux was determined by film methods by comparison with neutron characteristic curves obtained using the neutron beam at CP-5. The neutron flux as determined in this manner (after subtracting gamma effects) was lower than that presently available at CP-5 by

factors of 5 and more for different experimental conditions. Highest flux values were obtained immediately adjacent to moderating material (paraffin) but radiographs could not be obtained under such conditions because of the poor imaging geometry. The best imaging results were obtained using the fast neutron beam (unmoderated) produced by the $\text{Li}^7(p,n)\text{Be}^7$ reaction. Under these conditions the effective neutron flux was about $10^4 \text{ n/cm}^2\text{-sec}$ at 12 inches from the target. Using this experimental arrangement, radiographs were taken at target-film distances of 6 in., 12 in., 24 in., and 36 in. The inverse square law seems to hold reasonably well over this distance, except that the effective flux indicated at 36 in. was higher than anticipated. This fact seems to indicate that improved backscatter elimination may be required. The best image sharpness was obtained at the 6 in. distance, indicating that air scatter of the fast beam may be causing some of the resolution difficulties.

C. Reactor Materials Research and Development

1. Irradiation Damage in Steel

a. SA-212B Steel - Studies on SA-212B steels irradiated in MTR have continued. As part of these studies, examinations of SA-212B specimens irradiated in the MTR to a total nvt in excess of 100×10^{20} (thermal) were started in the Building 301 caves. The results confirm those obtained from lower dose irradiations in that hardness and embrittlement properties were found to saturate. In addition, the samples irradiated to a high total dose were found to respond to post-irradiation heat treatments at 700°F - 1000°F in a manner similar to samples irradiated to a lower total dose.

New irradiation effects include:

(1) Anomalous re-embrittlement phenomena in post-irradiation thermally heat treated materials. Whereas all specimens irradiated to lower dosages were re-embrittled at longer holding times at the annealing temperatures of 800° , 900° , and 1000°F , some bars tested showed superior resistance to impact (over unirradiated impact specimens) at the longer holding times at the same annealing temperatures.

(2) Bars (6 of 14 tested) which displayed marked improvements to impact also displayed a new failure phenomenon; i.e., the bars bent instead of fracturing at the notch. Resistance to failure by impact fracture in unirradiated bars was not observed in the tests conducted about 3 years ago. This behavior will be checked using specimens prepared from spare bars of material available.

A review of the data in three previous groups of irradiated specimens failed to uncover this type of behavior (failure to break in the notch) for the four annealing temperatures selected. Furthermore, only two irradiated bars exhibited enhanced impact properties. These two samples were heat treated at 1150°F for 1 hr in an argon atmosphere which is a significantly higher annealing temperature than that reached in the rest of the tests.

Under a cooperative program with the U. S. Steel Corporation, small samples of the irradiated SA-212B steel have been mounted and shipped to the Nuclear Science and Engineering Corporation (Pittsburgh). Non-active

plastic replicas will be prepared at the latter installation for examination under an electron microscope.

c. Magnetic Property Changes as a Function of Irradiation - A program has been initiated to study irradiation effects on the magnetic properties of materials. Both unirradiated and irradiated steels, SA-212B with various heat treatments, wrought Type 304 stainless steel, and wrought boron-stainless steels, are available. Magnetic measurements such as induction-permeability properties and hysteresis loops are expected to yield a better understanding of the radiation damage mechanism, inasmuch as magnetic properties are greatly influenced by alloying, thermal heat treatment, and mechanical work of deformation. A review of the published data derived from the conventional destructive tests (tensile, impact and hardness) reveals that they are insensitive indicators of irradiation damage; however, they do show that irradiation has a marked influence on the structural behavior of materials. This points toward magnetic measurements since magnetic properties are strongly influenced by both heat treatment and/or mechanical work.

Work was started in procuring and assembling electrical and electronic equipment for preliminary comparison measurements on a variety of irradiated, heat treated materials, including SA-212B, Type 304 stainless steel, and boron-containing Type 304 stainless steel. Some of the measurements contemplated at various frequencies include measurements of μ , H_e , M_{sat} and eddy current losses.

2. Structural Properties of Steels

a. Long-term Stress Effects - The availability of SA-212B steel removed from the EBWR pressure vessel affords an opportunity to ascertain the long-term (approximately 10,000 hr) influence of bi-axial stress and temperature, and the effects of fabrication on the impact properties of the steel, factors which are almost unknown. The results from studies of these materials are expected to establish the validity of the assumptions of reactor safety associated with requirements for the fine-grained, high impact strength (at low temperatures), steels currently specified for nuclear reactor pressure vessels. A reference block of SA-212B steel in the form of 4 in. plate (part of the lower EBWR head) is available for the comparison study. The reference plate is a part of the same heat of steel used in the EBWR shell.

3. Measurement Techniques

A technical note was prepared on diffraction grating strain gages. The study of this new type of strain gage was undertaken to examine its application in the testing of radioactive materials. This optical method consists of observing the behavior of spectral images of a ruled or impressed diffraction grating undergoing strain. The study outlines advantages and some few disadvantages. Principal advantages are: (a) strain range of a single gage is 10^{-1} to 10^{-6} ; (b) temperature limitations are those of the melting point of the specimen; (c) very short gage lengths; (d) calibration in situ and non-drifting zero; (e) no electrical contacts necessary. The only possible disadvantages appear to be that a high intensity light source is needed and there is a small signal-to-noise ratio. In principal, any static or dynamic application requiring the measurement of strain, particularly

large strain over long times, is possible. Some suggested specific uses are: (a) wave propagation in irradiated materials; (b) long-term strain relaxation; and (c) explanation and prediction of radiation damage processes.

4. Structures

a. Concrete - Work has begun on the problem of performing structural and tightness tests of the heavy concrete test cells of Building 315 housing the critical facilities ZPR-VI and IX. Methods for recording the strains have been developed. A preliminary structural analysis for reference to experiment has been made. Instruments for the structural test are being procured and some preliminary proposals for procedures to be followed on the leak and leak rate tests have been made.

D. Reactor Components Development

1. Development of Viewing Systems

a. Radiation Protected Glass Studies - Several series of experimental glasses of the barium silicate and barium lead silicate type, in the density range of 3.0 to 3.5 g/cm³ and containing about 1.8% by wt. of CeO₂, have been irradiated to about 10⁹r. The objective is to develop a glass for shielding windows that can be used in high-level research and process facilities for longer periods of time than existing commercial glasses. The basic studies on these compositions without cerium (July, 1960 Progress Report, p. 49) were used in planning this series.

It has been found, in general, that the intrinsic light absorption of cerium-containing glasses is increased by increasing the concentration of glass modifiers, such as BaO, PbO and ZnO, which introduce pairs of single bonded oxygen ions which interrupt the covalent bonded network of the glass. However, in some compositions, increases in the BaO and ZnO were found to decrease the irradiation-induced absorption. A range of compositions will be explored in more detail, centered about a barium lead silicate which has particularly low intrinsic and induced light absorption.

b. Ultraviolet Transmitting Glass - During the course of the study of irradiation effects on "Cabal" type glasses, it has been noted that a number of composition changes, as well as melting under reducing conditions, tended to improve the ultraviolet transmittance of these glasses. A glass of molar composition, 1.0 K₂O: 4.5 B₂O₃: 1.0 Al₂O₃ has the best transmittance of the compositions tested. The transmittance for 1 mm. thick samples, including surface reflections, are compared below to a high purity fused silica of superior light transmitting properties.

<u>Glass</u>	<u>Wave Length in Millimicrons</u>	
	<u>200</u>	<u>254</u>
Fused Silica (Supersil)	89.2	91.2
Argonne #412 (reducing conditions)	76.9	89.3
Argonne #412 (oxidizing conditions)	50.0	63.0
Phosphate glass (Glaze, N.B.S.)	65.0	79.5

The glass has good durability because of its aluminum content, and

the workability of the glass should allow it to be used as a substitute for the expensive and high melting fused silica.

2. Instrumentation and Control

a. Analog-to-Digital Techniques for Auto and Cross-Correlation of Reactor Dynamics Data - Statistical theory has been applied to the field of reactor dynamics. Cross-correlation techniques can be utilized to improve experimental measurements of reactor transfer functions and stability predictions with poor signal-to-noise conditions. The same techniques can be used to obtain data from inherent noise without the application of a specific input signal.

Special equipment has been ordered which converts analog information into digital form and stores the data on magnetic tape in IBM-704 computer format. The digital system was completely checked out and is now working satisfactorily. The frequency response of the Beckman FITGO amplifiers was measured and large discrepancies were found with respect to the specifications. They do not appear to be acceptable in their present condition and alternatives are being considered for replacing them.

3. Heat Engineering Studies

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

The design of the test loop is undergoing a critical review in an effort to reduce the possibility of damaging and time-consuming delays during operation, and to improve the precision of measurements required for accurate determination of vapor volume fractions and quality. The review has produced several important design alterations, although these changes will not affect the experimental schedule substantially.

There is a possibility that instability can occur at the power input station, but this situation can be avoided if the transformer characteristics dictate an essentially constant voltage input (small variation of regulation).

A simple liquid level device has been developed for condenser operation. This indicator has no moving parts or contacts in the sodium and, as such, cannot become inoperative by oxide fouling.

b. Heat Transfer and Fluid Flow in Packed Beds - A short study of several fluids (H_2 , He, H_2O , air) for cooling packed bed fuel elements has been completed. The results show that on the mass flow basis molecular hydrogen has the highest heat removal capacity, then follows helium, water (with vaporization taking place in the bed) and air. For example, if the temperature of the coolant were raised from 0 to $2000^{\circ}C$, hydrogen flowing at a mass flow rate of 1 kg/sec would remove 31.7 Mw of heat, while water would remove only 7.3 Mw. On the mole flow basis, water has the highest heat removal capacity, then follows air, hydrogen and helium.

In order to investigate the flow and temperature stability of a heat

generating packed bed, pressure drop for a one-dimensional steady flow through porous medium was studied. An equation relating pressure drop with various parameters entering the problem (porosity, bed thickness, particle diameter, coolant temperature gradient, etc.), was derived for the case when the friction factor is given by a polynomial of the mass flow rate, and the temperature of the coolant increases linearly with the distance along the bed (this is the case of constant heat generation). Calculations of pressure drop, for the range of parameters of interest in the packed bed fuel element design, are planned.

E. Separations Development

1. Fluidization and Fluoride Volatility Separations Processes

a. Direct Fluorination of Uranium Dioxide Fuel - In continuing studies of the direct fluorination process applied to uranium dioxide fuel, initial tests were made of the complete batch process cycle of decladding and fluorination. The Zircaloy decladding was accomplished by reacting with a 10 percent hydrogen chloride-90 percent hydrogen fluoride mixture. Adequate heat transfer was achieved, but caking resulted in lower fluorination rates than expected. Several methods of minimizing caking are being considered.

Studies of the reaction at 500°C of fluorine with uranium dioxide containing radioactive ruthenium tracer indicate that 99 percent of the ruthenium, probably in the form of ruthenium pentafluoride, was distilled away from the reaction zone within twenty minutes. At 400°C, 73 percent of the ruthenium activity was distilled from the reaction zone in thirty minutes. At both temperatures all of the uranium was converted to the hexafluoride.

A test of pellet fluorination carried to completion is in progress with a forced air-cooled reactor. The early stages of the reaction of the six-inch bed with 20 percent fluorine at 500°C showed fluorine efficiencies of 100 percent. Uranium hexafluoride collections were low at this stage because of intermediate fluorine compounds. Temperature control and gas analysis procedures worked very well.

Initial tests of the efficiency of a ventilation air scrubber in removing both hydrogen fluoride and elemental fluorine showed very high efficiency for hydrogen fluoride (> 99%) and substantially lower efficiency for fluorine (~60%).

b. Cl₂-F₂ and Cl₂-HF Studies on Low Uranium-Zirconium Alloy - Zirconium has been reacted with mixtures of hydrogen fluoride and chlorine at 500°C. A maximum reaction rate is obtained at about 30 weight percent chlorine. This is similar to the results obtained previously with hydrogen fluoride-hydrogen chloride mixtures except that the maximum rate is lower for the hydrogen fluoride-chlorine mixtures.

Mixtures of fluorine and chlorine were also reacted with zirconium. At 300°C the extent of the reaction of fluorine with zirconium was increased five times by the addition of ten volume percent chlorine. However, at 50

volume percent an increase in the extent of the reaction was observed instead of the decrease observed with the previously studied mixtures. In this case the product produced was predominately zirconium tetrachloride as evidenced by its distillation away from the reaction zone to cooler portions of the apparatus.

c. Processing of Stainless Steel-Clad Fuel Elements - The reaction of stainless steel and chlorine is being investigated in a two-zone fluid-bed reactor for application to decladding fuel elements. Chlorine is used as the primary (lower zone) reactant with hydrogen fluoride being admitted in the upper zone to convert the volatile ferric chloride to the solid ferric fluoride. Penetration rates of 9.0 and 10.2 mils/hour have been achieved at temperatures near 580°C on type 304 stainless steel test specimens, 1/2-inch segments of 5/8-inch diameter welded tubing. In contrast, a penetration rate of only one mil/hr was achieved in a tube-furnace experiment on type 304 stainless steel given a heat treatment of 4.7 hours at temperatures between 500° and 650°C which was expected to effect carbide precipitation at the grain boundaries and leave the material subject to greater attack by substances such as chlorine.

A carbon tetrachloride-water vapor mixture at 550°C and bromine at 600°C proved to be of little value for dissolution of stainless steel in the horizontal furnace-tube reactor.

Corrosion studies of chlorine on materials considered for reactor construction, nickel, Inconel, and Hastelloy-C, gave comparable corrosion rates of about 20, 20 and 12 mils/month penetration, respectively, based on extrapolation of 2- to 19-hour tests performed at 600°C in a tube-furnace reactor.

d. Conversion of Uranium Hexafluoride to Uranium Dioxide - Recent fluid-bed studies on the first step (steam pyrohydrolysis to uranyl fluoride) of the process for converting uranium hexafluoride to uranium dioxide have investigated the effect of a recycle powder feed stream on particle size control. A total of 50 kg of uranyl fluoride was prepared in ten hours. Results indicate that a recycle rate equivalent to about 15 percent of the hexafluoride feed rate should be adequate for the present system; 100 g/min hexafluoride feed rate, 18-inch deep bed, 200°C , and steam excess near 250 percent.

e. Plutonium Fluoride Studies - Further work on the kinetics of plutonium hexafluoride decomposition in the range of initial pressure of 600 to 1400 mm at 161°C , indicates that the reaction is first order with respect to plutonium hexafluoride pressure under these conditions. Some of the earlier work done at lower pressures gave results which apparently indicated the reaction was zero order with respect to plutonium hexafluoride pressure. However, the earlier results may have been obtained over reaction times which were too short to yield valid results.

Experiments performed at about 80°C indicate that bromine can be used for the reduction of plutonium hexafluoride to plutonium tetrafluoride.

Further experiments on plutonium hexafluoride transport indicate that amounts in the range of 20 to 50 grams can be transferred from one container to another through nickel lines and valves with greater than 99 percent

recovery (making corrections for alpha radiation decomposition) by vacuum distillation, or by transpiration using fluorine or helium as a carrier gas.

A sample, 37.9 grams, of plutonium hexafluoride was held at 60°C (mp PuF_6 51.6°C) for 25.5 hours. The amount of plutonium hexafluoride decomposed did not exceed the calculated amount which would decompose by alpha decomposition in both the gas and condensed phase, 0.35 and 2.0 percent per day, respectively.

2. Chemical-Metallurgical Process Studies

a. Liquid Metal Solvent Studies - The solubility of erbium and nickel in liquid cadmium may be represented by the empirical equations:

$$\text{Erbium (330}^\circ\text{--590}^\circ\text{C): } \log (\text{atom percent Er}) = 6.309 - 6644 T^{-1} + 1.485 \times 10^6 T^{-2}$$

$$\text{Nickel (330}^\circ\text{--509}^\circ\text{C): } \log (\text{atom percent Ni}) = 6.722 - 5628 T^{-1} + 0.9045 \times 10^6 T^{-2}$$

$$\text{Nickel (509}^\circ\text{--650}^\circ\text{C): } \log (\text{atom percent Ni}) = 1.954 - 7408 T^{-1}$$

In the nickel-cadmium system, the peritectic point was found at 509.3°C.

Values of the coprecipitation coefficient, λ , for several elements (lithium, sodium, potassium, lanthanum, strontium, and zirconium) when carried by the cerium-cadmium intermetallic compound CeCd_{11} were reported in the October Progress Report (ANL-6253). The following additional values of λ have been obtained: yttrium, 0; uranium, 0.13; and europium, 0.099.

The coprecipitation coefficient of cerium when carried by the uranium-zinc intermetallic compound from a 20 weight percent magnesium-zinc solution was found to be 0.67. This value does not differ from the value previously reported for the 10 weight percent magnesium-zinc solution or from the value found for pure zinc.

The distribution coefficient of cerium between the immiscible liquids lead and zinc has been measured at several temperatures. The values of the coefficient (weight percent of cerium in zinc/weight percent of cerium in lead) were 24 at 650°C, 11.7 at 700°C, and 5.7 at 730°C. Over the same temperature interval, the coefficient for uranium in the same system changes from 25 to 200. From the distribution data, together with the data on solute solubilities and the mutual solubilities of the solvents, it is concluded that 650°C would be the most favorable working temperature should the lead-zinc system prove to have process applicability,

Additional studies of vapor pressure vs. composition in the cerium-zinc system have shown the existence of two intermetallic phases with extensive solubility ranges: a phase extending from a zinc-cerium ratio of 7.5 to 7 and a phase extending from 5.2 to 3.8. The study of the system is continuing and will include the determination of phase relationships for compositions below and including a zinc-cerium ratio of two. Data relating vapor pressure and composition indicate that compound formation at an atomic ratio of two is likely.

Samples of the epsilon phase in the uranium-zinc system have been separated from excess zinc by an electrolytic etching method. Chemical analysis shows a zinc to uranium ratio of 11.5 ± 0.1 , which is in good agreement with the results of effusion studies.

b. Calorimetry - Calorimetric combustions of uranium mononitride in oxygen and of uranium in fluorine have been started.

Preliminary work is in progress on methods for determining the heats of formation of titanium, hafnium, and thorium tetrafluorides.

Results of recent experiments indicate that zirconium dihydride is completely converted to zirconium dioxide upon burning in oxygen at pressures above 10 atmospheres.

A method for the calorimetric study of substances that react spontaneously with fluorine is being developed. Equipment has been fabricated which will allow the reactants to mix at the appropriate time. The first use of the new technique will be applied to the study of the combustion of boron nitride in fluorine.

Work on the study of the heat of formation of tungsten disulfide is nearly completed.

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 test loop can be operated at pressures up to 10 atm with water flow rates as high as 1900 liters/min. No temperature control is provided although an equilibrium temperature of 146°C is reached after 12 hr. of operation due to the heat input from pump friction.

The first flow distribution tests were made with an aluminum fuel element containing twenty-four 510×60 mm plates with the spacing between plates the same as their thickness (1.25 mm). After continuous operation for 21 hr. at 146°C , an abrupt change in flow distribution between channels was detected as the flow rate was being increased to 1900 liters/min. Impact tubes within interior channels of the assembly had shown a maximum difference in velocity between channels of 8%. Afterwards this jumped to 40%.

Subsequent inspection of the element revealed that the fuel plates had buckled longitudinally and, in addition, transverse bowing had occurred at the inlet end of the assembly.

During the test, the frictional pressure drop at the higher flow velocities lowered the discharge pressure from the test section to a value close to saturation at the loop temperature. Under this condition, a momentary increase in the friction drop in one channel may have occurred permitting

the water in that channel to flash and create a severe pressure gradient between channels. Examination of the test section showed that small pieces of unidentified material had been trapped in several channels indicating a possible source of pressure instability.

Additional fuel assemblies are being prepared to continue the flow tests.

b. Theoretical Studies - A comparative examination was afforded a report on the Russian "SM" high-flux reactor. The heat transfer data given were found to be consistent with standard methods of calculations. The relatively small maximum-to-average heat flux indicates a reduction of fuel concentration adjoining the internal thermal column to about half of the average fuel concentration. The "SM" fuel plates are superior structurally to the conventional MTR type for two reasons. First, the "SM" fuel plates are nickel rather than aluminum. Second, the "SM" fuel plates are only 3.10 cm wide while, for example, the AHFR fuel plates are 5.36 cm wide. As a result, the critical coolant velocity for failure under simple hydrodynamic loads is a factor of 1.4 higher in the "SM".

c. Physical Behavior - In studying high power density cores it is apparent that technology is not in good shape for predicting behavior of fuel plates and coolant flow outside ranges of experience. In pushing on to higher velocities and thinner plates, analytical (and to some extent experimental) treatments are still very shaky. To effect improvement, an attempt has been completed to determine analytically the dynamic behavior of parallel fuel plates, periodically supported, including loading by fluid acceleration. In this analysis, the plate deflection was assumed to be constant across the width of the plate and to vary along its length. No divergence is possible for this type of motion, so critical speeds can be associated only with deflection patterns that involve variations across the plate width. It appears that the presence of the fluid leads to traveling waves in the plate superimposed on the natural vibration modes.

2. Basic Radiation Effects Reactor

As a supplement to the BRER studies, exploratory calculations are directed toward the design of a converter plate which can be used in CP-5 to produce a high ratio of fast-to-thermal flux for radiation damage studies.

The results of the DSN transport code calculations for the converter element configuration of Muehlhause (NBS) have been completed. Briefly, the NBS element consists of three heavily loaded concentric uranium-aluminum cylinders (about 4.0 in. OD) surrounding an inner sleeve of boral. The assembly is placed in a multiplying heavy water medium. In the simplest form the element is just self-supporting, neutronwise, i.e., removal or insertion in the multiplying medium causes no over-all reactivity change to the medium.

3. Direct Conversion Studies

Construction of the auxiliary heater mentioned last month is completed. The calibration, which is in progress, allows one to infer the radiation power loss in an experimental cell with parallel plate geometry. This loss is the major factor reducing efficiency of power conversion.

Theoretical studies concerning work functions and electron emissivities for metals and semi-conductors are in progress to select proper emitter materials.

4. Fast Fuel Test Reactor (FFTR)

An information meeting on this reactor concept, sponsored by the AEC, was held at the Laboratory on November 1, 1960. Attendees included potential users of such a facility.

5. Lunar Power Plant

A first draft of the terminal report on LP-1 has been prepared and sent to the AEC-NASA office. Future participation in space power problems was discussed. A separate draft report on analysis of the LP-1 mercury-condensing radiator has been completed.

VI. PUBLICATIONSPapers

TELEMETERING RADIATION DATA BY FREQUENCY VARIATION

Hans Kohn Richards

Electronics, November 11, 1960

COMMENT ON "VACUUM APPLICATIONS OF INDUSTRIAL GLASS PIPE"

C. E. Cohn

Rev. Sci. Inst. 31, 1165, (1960)(letter)

SCATTERING OF FAST NEUTRONS FROM TUNGSTEN

A. B. Smith and P. Moldauer

Bull. Am. Phys. Soc., Series II, Vol. 5, No. 6 (1960)

IDENTIFICATION OF THE DIFFUSING SPECIES OF URANIUM OXIDATION

J. G. Schnizlein, J. D. Woods, J. D. Bingle and R. C. Vogel

J. Electrochem. Soc. 107, 783-785 (1960)

THE DETERMINATION OF URANIUM AND PLUTONIUM IN URANIUM-PLUTONIUM-FISSIUM ELEMENT ALLOYS

R. P. Larsen and C. A. Seils

Anal. Chem. 32, 1863-1866 (1960)ANL Reports

- ANL-6010 SUMMARY REPORT ON IRRADIATION OF PROTOTYPE
EBR-II FUEL ELEMENTS
J. H. Monaweck and E. S. Sowa
- ANL-6174 THE REACTOR KINETICS OF THE TRANSIENT REACTOR
TEST FACILITY (TREAT)
D. Okrent, C. E. Dickerman, J. Gasidlo, D. M.
O'Shea, and D. F. Schoeberle
- ANL-6193 STUDY OF THE PACKED-BED FUEL ELEMENT CONCEPT
M. B. Rodin
- ANL-6224 THERMAL STRESSES IN CONCENTRICALLY HEATED
HOLLOW CYLINDERS
J. C. Heap
- ANL-6226 MULTIGROUP DIFFUSION THEORY FORMULATION OF
THE CALCULATION OF THE MEAN SQUARE SLOWING
DOWN DISTANCE IN AN INFINITE MEDIUM
B. J. Toppel

ANL Reports

- ANL-6227 SPECIAL LECTURES ON THE PHYSICAL FOUNDATION OF
REACTOR ANALYSIS
Paul J. Persiani
- ANL-6243 STUDIES OF CORROSION BY MOLTEN ZINC AND CADMIUM
SYSTEMS
J. P. DeKany, H. W. Lavendel, L. Burris, Jr.
- ANL-6244 ON THE PROBLEM OF LIQUID ENTRAINMENT
Gordon C. K. Yeh and Novak Zuber
- ANL-6246 THE THERMODYNAMIC AND TRANSPORT PROPERTIES OF
SODIUM AND SODIUM VAPOR
E. L. Dunning
- ANL-6247 THE FISSION ENERGETICS OF Th^{232}
A. B. Smith, R. G. Nobles and A. M. Friedman