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I. WATER COOLED REACTORS (040101)

A. General Research and Development

1. Heat Transfer and Fluid Flow Studies

a. Analytical Studies of Two-Phase Flow - Further studies were made to investigate the general problem of steam carryover and entrainment in the downcomer of a natural circulation boiling reactor. In order to predict flow rates and reactor power for a given value of reactor void fraction, it is necessary to know the density of the water in the downcomer. This month's work covered an analysis of the condensing rate of a moving steam bubble in subcooled water.

A manuscript entitled, "On The Collapse of Vapor Bubble in a Uniformly Subcooled Quiescent Liquid," is currently being prepared. The theoretical results are to be presented in graphical form to facilitate its use and also to enable one to see readily the influence of system pressure, degree of subcooling, initial bubble radius, and fluid properties on the collapse rate.

A second problem is the evaluation of void fraction during boiling. Present practice is to rely upon experimentally measured slip ratios. Curves of slip ratio vs. steam quality, mass velocity and system pressure are available for use in boiling reactor design; however, extrapolation from these curves leads to questionable results.

Hence, several attempts were made to predict theoretically the void fraction and slip ratio in turbulent two-phase flow in a vertical duct under wide ranges of quality, liquid phase velocity, system pressure, etc., but without success. Available literature on the subject has been briefly reviewed and, to date, it seems that Levy's momentum transfer theory (ASME Paper 59-HT-15) offers the best promise. However, a large discrepancy exists when Levy's prediction is compared with Petrick's data on air-water mixture (ANL-5787). On the other hand, analysis of the flow model in which the liquid phase is continuous but the gaseous phase is discrete spherical bubbles, shows good agreement between theory and experiment for low quality mixtures. These studies confirmed the fact that the over-all problem of two-phase flow has to be considered separately according to the flow pattern. Basic information necessary to the understanding of many complex problems in boiling systems, but which is not available in the literature, concerns the motion of bubbles in a turbulent stream.

2. Boiling Slurry Reactor Experiment (SLURREX)

SLURREX is a boiling aqueous slurry reactor proposed for construction at the National Reactor Testing Station. The reactor consists of a natural circulation boiling core system plus functional subsystems such as fuel handling, gas handling, heat dissipation, and water distribution. The design power level is 5 Mwt and the operating pressure is 10 atmospheres.

The preliminary design of the fuel handling system is completed. Components were designed and piping arrangements were selected to accomplish the functional requirements and satisfy the design criteria established for the fuel handling system. The operations performed in the fuel handling system were divided into four types: (a) slurry storage, (b) water storage, (c) concentration or dilution of slurry within the reactor, and (d) transfer of material to or from the reactor.

Slurry storage is provided in a coil of 4 in. pipe. A coil is used to provide both a non-critical geometry and a means of maintaining the slurry in suspension. Suspension of the slurry is maintained by a 200 gpm pump which circulates the slurry through the coil.

Reactor slurry concentration will be changed during critical operation by bleeding slurry from, or feeding slurry to, the reactor and adding more or less feedwater. All material transfer operation will be accomplished by a static pressure differential driving force.

Brief operational procedures were prepared to illustrate the conditions and steps necessary to accomplish the system functions.

Work has been initiated on the design of the gas handling system with the determination of the functions and design criteria for this system. The functions of the gas handling system are: (a) to receive and store the noncondensable gases carried with the steam, (b) to provide the void in the core to produce natural circulation during noncritical conditions, and (c) to provide the pressure differential for material transfer.

The preliminary design of a charcoal bed absorber was completed. The primary function of the bed is to absorb xenon and krypton from the gas system. About 4 cu. ft. of activated charcoal are required to absorb the gaseous fission products produced in 1000 hours of 5 Mw operation.

The preliminary design of a full flow catalytic recombiner was completed. It is estimated that a catalytic bed of platinum coated wire mesh 2 ft. in diameter by 4 ft. long will be required. Recombination of the radiolytic hydrogen and oxygen will result (for a G value of 1.3) in 30°F of superheat to the full flow of reactor steam. A correlation was prepared showing the degree of superheat available at 5 Mw operation for a range of G values from 0.8 to 1.8.

Calculations were made to determine the critical concentrations of U^{235} . The fixed parameters used in the study were: (a) power level of 5 Mw, (b) operating temperature of 185°C, (c) average void fraction of 0.32, and (d) core diameter of 59.7 cm. The core length to diameter ratios were varied from 1.25 to 1.5. The thoria concentration was also a variable. Using a length to diameter ratio of 1.5, the critical concentration of U^{235} was 34.5 gms per liter with 600 gms per liter of ThO_2 , and 27.5 gms per liter with 200 gms per liter of ThO_2 .

B. EBWR

1. 100 Mw Modifications

a. Modifications Within the Containment Shell (Package III-A). -

All contract work on equipment in this package was completed on August 31. Future work includes paint and insulation touchup and modifications to equipment which might arise in debugging procedures. This will be performed by Laboratory personnel. All mechanical equipment has been checked out satisfactorily with the exception of the new reactor feedwater pumps. Control of the shaft-bearing forced lubrication system does not function according to specifications on this item. The difficulty is being resolved with the vendor. Electrical work is also complete within the same stipulations. Limit switches on new control valves appear to require further adjustment and it may be necessary to do this during reactor operation. Some cabling and wiring discrepancies have been found and corrected and work is continuing on checking out the instrumentation and calibrating of new instruments.

b. Work Outside the Containment Shell (Package III-B). - Very appreciable delay has occurred in completion of the Package. This is due almost exclusively to setbacks in the date of delivery of the primary reboilers which are, of course, the major components of the 100 Mw system. The latest vendor's shipping date for these reboilers is September 7 for one, and September 15 for the other. However, these dates are considered to be optimistic.

Periodic visits to the vendor's plant for inspection of other heat exchangers needed for EBWR has convinced Laboratory personnel that only a token effort is being made on these reboilers.

Other items which may cause a delay in the project are the flash condensate rate unit and the secondary steam separator condensate return unit. The latter item is an addition to the original scope of work. The pumps for the unit are long-delivery items requiring 8 to 10 weeks.

Electrical work on this Package is complete except for equipment which cannot be installed before delivery of heat exchangers.

c. Fabrication of EBWR Mark I-A Spike Fuel Elements - The fuel rods for element number thirty-five were completed, bundled, and loaded into the frame assembly. These rods, containing Al-U₃O₈ core material clad with Zircaloy-2 tubing were passed through an induction coil to zone-melt and diffusion bond the core to the cladding. An inert-gas shield surrounding the rods prevented surface oxidation as they were heated to approximately 700°C. The over-all length of the bonded rods was found to be 0.25 in. shorter after the bonding process than before; however, subsequent repeated heating of a bonded rod showed that no unusual thermal expansion was occurring at temperatures up to 260°C. Processing of the bonded rods proceeded in the usual manner. Fabrication of fuel elements for the EBWR Mark I-A is now completed.

d. EBWR 100 Mw Oscillator - A mechanism utilizing electrical clutches has been designed to allow remote gear changing of the EBWR oscillator rod. This, in turn, will permit the running of sequential transfer function experiments without requiring personnel to enter the containment shell between runs. The oscillator can operate at a mean-rod level either 25 or 100 cm withdrawal, and the amplitude of oscillation is ± 12.7 cm between 0.001 and 0.1 cycle per second; and ± 2.5 cm up to 8 cycles per second.

e. EBWR Demister Basket - An attempt to utilize a demister to reduce entry of water into the external steam dryer units is being made. For this purpose a demister basket is now being fabricated.

f. In-Core Instrumentation - Two instrumented fuel assemblies have been built for the purpose of measuring some of the limiting hydrodynamics parameters. They will be placed in core positions such that the radial power distribution will not be significantly disturbed.

The instrumented assemblies are standard elements consisting of thick 1.44% enriched fuel plates and modified inlet and outlet fittings which accept special turbine type flowmeters that measure the inlet water velocity and the exit liquid velocity. The exit steam void fraction can be calculated from the two liquid velocity measurements if flow continuity between the inlet and outlet is assured. One assembly, designated as Unit III, has the perforated side plates sealed to provide this flow continuity. The second assembly, designated as Unit II, has the original perforated side plates. An attempt will be made with this assembly to study the effects of fluid coupling between adjacent fuel elements.

Both instrumented fuel elements have thermocouples attached at the inlet of the assemblies which will measure the reactor subcooling. The water recirculation flow rate can be calculated from the measured subcooling. The subcooling will be measured directly by connecting the inlet thermocouples in series with the thermocouples installed in the steam zone, thus forming a thermopile. Absolute inlet temperature will also be measured.

Unit III will have, in addition to the above instrumentation, two miniature in-core flux monitors, one of which is cadmium covered, located in diagonally opposite corners of the element. The chambers will measure both the neutron flux and the cadmium ratio at a point 66 cm. above the lower edge of the fuel plates. The steam void fraction can be calculated from the measured cadmium ratios.

g. Chemical Control - Core I-A has been subjected to extensive physics analysis (see Monthly Report, July, 1960). As a result of this analysis, it appears that the recommended core will be subcritical cold with nine rods fully inserted. However, in order to provide enough reactivity to be able to reach power levels in the 100 Mw range and still have a shutdown margin, it will be necessary to use boric acid control for shutdown, and it is not impossible that this control will also be used as during operation.

An auxiliary boric acid addition system is now being provided. Construction is expected to be complete by September 15. The system provides for metered addition of boric acid to the reactor via (a) the low pressure boric acid ring, (b) the three-inch feedwater line, or (c) the condenser hotwell.

The ion exchange system is being modified to maintain proper boric acid concentration within the system and prevent its inadvertent removal. The concentration of boric acid will be determined by titration, mass balance, and continuous monitoring of neutron attenuation in a flowing reactor water sample.

h. Thermal Hydraulic Analysis - Final hydrodynamic and heat transfer calculations were completed for the EBWR Core I-A based upon the power distribution obtained in the recommended loading in the latest physics review of the core. The significant results of the calculations are given in Table I.

Table I. Hydrodynamic Data for EBWR Core I-A

Total coolant flow	11.8×10^6 lb/hr
Average reactor quality	2.6%
Recirculation rate	38.5 lb H ₂ O/lb steam
Inlet subcooling	8.91 Btu/lb
Average core inlet velocity	7.4 fps
Average voids in reactor core	19.7%
Maximum heat flux	346,000 BTU/hr ft ² (central thin enriched elements)
Maximum power density	218 kw/L (spike elements)
Maximum centerline temperature	
Thin enriched elements	665°F
Thick enriched elements	587°F
Spike elements	2000°F

These power densities and heat fluxes predicted for the reactor are considerably below limiting values determined experimentally from hydrodynamically similar channels and hence the possibility of a burnout or a hydrodynamic instability in core I-A at 100 Mw is believed to be remote.

i. Reactor Operator Training Program - A reactor operating training program was started last June to familiarize the operators with the modified plant. Fifteen lectures were given by the water section staff covering various phases of the plant. The lectures are being continued bi-weekly by the members of the Reactor Operations Division. Periodic examinations covering the more important points will be given.

A series of simplified flow schematics are being prepared that will depict the entire flow scheme associated with the 100 Mw design. These schematics will show all major valves in their relative location as well as location of piping entering various equipment components. Considerably more time will be required to complete the layout inasmuch as previous system changes were not kept current.

2. Core II Studies

A draft report on the physics of possible future cores in EBWR has been completed. Studies were made of a number of seed blanket arrangements involving both U^{235} and Pu-fueled seeds and variable blanket enrichments. In addition, uniform cores and 2-region cores were studied.

Tentative conclusions pertinent to future loadings of EBWR are as follows:

a. Since EBWR has no provision for channeling the flow into the seeds in seed blanket arrangements, it is necessary to keep maximum seed power densities within the same range as maximum blanket power densities. Because of this, a relatively low concentration of fissionable material is necessary in the seed.

b. In consequence of this, and of the fact that the most attractive use of seeds insofar as neutron economy is concerned occurs in arrangements such that the seed power density is appreciably greater than that of the blanket, the seed blanket configuration is not useful in EBWR at high power ratings. This conclusion may be softened if the reactor is operated at less than 100 Mw, but the studies of this variable have not been made yet.

c. For a combination of long fuel life at high power in EBWR, a uniform core enriched to about 3% of U^{235} would be specified. A more elegant system would be a two region core with the outside at higher enrichment and the center at lower enrichment. This would allow a relatively rapid approach to a recommended clad zone irradiation scheme in which fuel is introduced from the outside and moved inward as radiation progresses.

d. A major problem in both reactivity and conversion is presented by the fact that with the high core powers in a system without the radial flattening provided by an outside-in clad loading, the voidage in the upper part of the hottest radial portion is so great as to suppress the chain reaction in this zone and reduce conversion. A possible solution would be to shorten the core in these hot zones still more and perhaps to include axial blanket sections.

C. BORAX V

1. Construction, Design and Procurement

On August 26th, 1960, the over-all construction project was 67% complete vs. 82.5% scheduled.

The electrical contractor completed the external electrical work with the exception of the line between the turbine and reactor building. Progress on interior electrical work is limited by the fact that piping to the major components is incomplete.

The heating and ventilating contractor has begun installation of equipment. The reactor building and heating and ventilating building are nearing completion. Two sections of the reactor building roof are left open to allow for installation of the reactor pressure vessel.

Of the 75 items on the process instrumentation and control equipment, 16 remain to be shipped. All valves and transmitters required to be installed by the contractor have been shipped. Testing of the transmitters requiring high performance dynamic response was witnessed. Some corrections in the readings of steam and feedwater flow will have to be made for high-frequency reactor oscillator tests.

Installation by ANL of the new steam separator in the turbine building was completed. The cleanup of the condenser was completed and leak-testing was begun.

Machining on the reactor pressure vessel is proceeding at a rate slower than anticipated. Shipment of the reactor vessel is now estimated at September 15. A Laboratory representative is stationed at the Nooter Corporation Plant to expedite decisions on any questions that arise.

The 10,000 gpm forced-convection pump was received and installation was begun during the last week of this month. Final nontemperature tests were witnessed and approved prior to shipment. The piping contractor is now able to resume work to make connections of the forced-convection piping.

Negotiations with the low bidder for furnishing superheat fuel plates were concluded with the bidder refusing to accept the terms of the contract. Negotiations with the second lowest bidder have begun.

2. Development Studies

Vibration tests were begun using a mockup boiling fuel assembly in the air-water test loop. Vibration of the loop was sufficient to interfere with the measuring of the true deflection of the fuel rods. Additional bracing has been added and expansion joints have been ordered to reduce the vibration of the loop. Preparations are in progress for testing the Potter turbine flowmeter received this month.

The redesign on the core holddown Belleville spring and washer assembly and the forced-convection baffle have been completed. A 60,000 lb. spring will be used for forced-convection operation and a 20,000 lb. spring for natural convection. Large piston rings were incorporated in the design of the forced-convection baffle as a means of assuring low leakage around this baffle. Core structure design is completed. Final review of drawings is in progress preparatory to releasing them for manufacture.

Reducing the thickness of the control rods was considered as a means of increasing the clearance between the control rod and the shroud. A machine problem calculation was made to determine the change in effectiveness caused by reducing the thickness from 0.375 to 0.25 in. The results are being analyzed. Studies preparatory to further hazard analysis are in progress.

Preliminary design of the boiling fuel elements handling tools has been completed. Design of a test stand to check the workability of remote underwater handling devices has begun.

The design of the modification of the air ejector exhaust system is continuing.

The hollow metal O-ring with retaining rings manufactured by Advanced Projects Company has been selected for the superheat fuel element seal. Preparation for a final life test is in progress. Inconel X has been selected as the material for the holddown springs.

The superheat fuel plates which will be obtained commercially for BORAX V are a stainless steel-uranium oxide dispersion type containing four different weights of UO_2 content depending on plate location in the reactor core. A total of 840 of these plates containing highly enriched UO_2 are to be supplied. In addition, 208 similar plates containing depleted or normal UO_2 will be fabricated before the enriched plates are fabricated. These depleted or normal plates will be used for evaluation and practice

assemblies. Twenty-four of these will be evaluated by the supplier and twenty-four by the Laboratory for adherence to specifications.

Assembly of equipment for nondestructive testing at the Laboratory is continuing. Specified destructive tests, e.g., the Strauss sensitization test, chemical analysis for core homogeneity, and measurement of core and cladding thickness, are being performed on sample plates.

An experimental program has been initiated in order to continuously monitor the rate of irradiation damage to the BORAX V pressure vessel.

The vessel is made from SA-212-B steel clad on the inside with Type 304 stainless steel. The nominal wall thickness of the vessel is $2\frac{1}{16}$ in., of which $\frac{3}{16}$ in. is stainless steel. The operating temperature and pressure of the reactor will be 255°C and 600 psig, respectively.

Although it is well known that steels of the SA-212-B type are subject to deleterious changes resulting from fast neutrons, the present status of knowledge does not permit an accurate prediction to be made of the magnitude of the changes for a given reactor exposure. Therefore, in common with the approach being used at several other installations, monitor specimens will be used to determine the changes occurring in the vessel itself. The specimens will be cut from the plates used to fabricate the vessel. They will include parent metal, weld metal, and metal from heat-affected zones.

Specimens will be encapsulated and placed near the inside wall of the vessel at the reactor midplane. Specimens will be withdrawn from the reactor at intervals during the life of the vessel. The impact and tensile properties of the irradiated specimens will be compared to the properties of identical specimens which were not irradiated but which were held at the same temperature for the same periods of time. The irradiated specimens will also be compared to identical control specimens which have been held at room temperature.

A capsule of typical specimens was irradiated in CP-5 in order to determine the temperature rise in the specimens to be expected in BORAX V as a result of gamma heating under the most extreme conditions (40 Mw). A temperature rise of 80°C was measured. Efforts are in progress to use more efficient packing to decrease the temperature rise.

Four 11 in. x 12 in. x $2\frac{1}{16}$ in. pieces of plate have been received from the vessel fabricator and are presently being machined into test specimens. From 500 to 600 specimens will be made, depending upon cutting losses.

II. SODIUM COOLED REACTORS (040103)

A. General Research and Development

1. ZPR-III

Critical studies of the EBR-II reactor have continued during the last month. The purposes of the experiments performed have been to aid in the design of an oscillator and to provide sodium worth data for calculation of the power coefficient.

It is planned to perform oscillation studies on EBR-II by inserting an enriched boron carbide rod in one of the control rod locations and rotating it eccentrically. Control rods are located in the corners and at the flats of the hexagonal core. The oscillator can also be operated in a fully or partially inserted position.

These arrangements were simulated in the ZPR-III mockup using an oscillator in which the quantity of boron carbide could be varied. A loading was found by trial to give the desired reactivity amplitude (about 28 inhours fully inserted). Fourier analysis of the reactivity curve showed the deviation from a pure sinusoidal reactivity variation. The second, third, and fourth harmonics were 1.45, 0.35, and 0.20 percent, respectively, of the fundamental.

The worth of sodium as a function of position has previously been measured in this assembly, but the results were suspiciously high at the center of the core (40.9 inhours/kg). These previous measurements were made by substituting sodium for void in local regions of a core in which the majority of the coolant was simulated by aluminum. It was thought that the presence of the aluminum might have contributed to the unexpected sodium worth. The experiment was therefore rerun in a core from which all aluminum was removed and replaced with sodium cans. In this loading, sodium had a worth of 25.4 ih/kg at the core center, although values at the edge agreed quite well with those obtained previously in the aluminum core. The results raise some doubts about the validity of sodium substitutions in cores whose coolant has been simulated by aluminum. Corroborating experiments are being devised.

2. ZPR-VI and ZPR-IX

a. Construction - Over-all, construction of the building is 30% complete. The first and second floor slabs have been poured. Some of the second floor columns have been poured. The remaining columns and beams are being formed.

The cell for ZPR-VI (#4) has the bottom layer of reinforcing steel for the floor mat in place. In cell #5 (ZPR-IX) the fuel handling tubes to the vault, and the access tubes to the control room have been positioned.

The floor has been poured for Cell #6, and columns and beams to the second floor level have been constructed.

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

An insertion type control rod drive is being designed to insert a shim rod 90.17 cm into the core from a point 91.44 cm from the core center. The first 87.63 cm will be traversed in 100 milliseconds, and the remaining 1.27 cm in an additional 25 milliseconds. Bounce is not to exceed 1.27 cm.

3. Juggernaut

The Juggernaut building (335) is now 95% complete and should be ready for limited occupancy by September 1, 1960.

Assembly of some of the reactor components began August 1, but because the building was not available, control rod drives, the beam hole shutter mechanism, and thermal column mechanism were preassembled in another area.

Design is completed except for minor corrections and normal completion work. The instrumentation used on the Geneva ARGONAUT reactor has been redesigned for use on Juggernaut. The system drawings were changed to include several new circuits. Modifications to the panels are in progress.

The fuel plates have been inspected and approved and the manufacturer notified to proceed with fabrication.

Preparation of the Hazards Report and Operating Manual are in progress.

4. Fast Fuel Test Reactor Design

a. Over-all Design - A report, ANL-6194, on the FFTR concept is in the final editing state. This reactor is intended for irradiating fast reactor fuels and structural components. The operating power level is 200 Mwt although a power level of 250 Mwt may be reached. The core contains

sufficient beryllium to degrade the flux to the hard intermediate region. The design appears capable of producing specific powers in samples up to 1.5 kw/gm Pu²³⁹. The preliminary cost estimate for the over-all facility is \$36,000,000.

The suggested fuel element is a plate type containing UO₂ - stainless steel cermet clad with stainless steel. An alternate fuel design is of the same type except that UN is substituted for the UO₂.

The minimum core volume that would accommodate the desired facilities and make possible the desired specific power is about 300 liters. The reactor design is of a preliminary nature and was carried only far enough to make possible reasonable cost estimates. It is intended that the report be a first step toward eventual construction of such a reactor. This would permit the testing of fuel assemblies under operating conditions impossible to obtain in any facility now existing or under construction.

b. Heat Transfer - In view of the number of assumptions made in the sizing of the FFTR intermediate heat exchangers, it was felt that further study on the subject was warranted. It became apparent during the initial study of the exchanger that there was a serious lack of shell side heat transfer information applicable to large liquid metal exchangers, and that the surface area requirements are very sensitive to the shell side coefficient employed. The relation resulting from experiments performed at Mine Safety Appliances (MSA) Co. on a small baffled heat exchanger was employed, with reservations, to calculate coefficients. Recently published data obtained from tests of the SRE intermediate exchanger indicate that, for this particular configuration, the MSA equation yields results which are conservative by approximately a factor of two. Apparently the factor in the MSA equation relating heat exchanger geometry with Nusselt number cannot be extrapolated to either SRE or FFTR conditions. However, if SRE data are extrapolated to FFTR Peclet numbers, approximately the same Nusselt numbers used in the FFTR design are obtained. Since future crossflow exchanger designs will presumably place more emphasis on reducing by-pass flow area than was the case for SRE, it appears that the FFTR design numbers may, fortuitously, be somewhat conservative.

5. Fast Reactor Fuels - General

a. Fast Reactor Fuel Jacket Development - The use of stainless steel as a jacketing material for fast reactor fuel operating at elevated temperatures has decided limitations. Among these limitations are: (a) the low melting uranium-iron eutectic and the even lower melting plutonium-iron eutectic, (b) high diffusion rates with U-Pu-Fs alloy, and (c) the moderately low yield and creep strengths of these steels at elevated temperatures. These limitations become even more severe if the jacket is to

be relied upon to restrict the swelling of metallic fuels induced by irradiation. Elevated temperature irradiation tests indicate that the swelling temperature of U-Pu-Fs alloy is below EBR-II design operating temperature. The limitation imposed by the low melting eutectics and high diffusion rates can possibly be overcome with barrier foils which are compatible with both fuel and stainless steel jacket. To lessen the restriction imposed by low yield and creep strength at elevated temperatures, however, new alloys must be developed. The desired jacket material, of course, is one which combines compatibility with the fuel with high strength and creep resistance for suppressing swelling. Additional requirements of these alloys are compatibility with sodium, good fabricability and weldability and good thermal conductivity.

At the present time, attempts are being made to secure refractory alloys for fabrication development work and samples of good quality high strength alloy tubing for testing prototype fuel elements and for fuel meltdown tests in the TREAT reactor. Niobium base alloys are being supplied by Battelle Memorial Institute and vanadium base alloys are being supplied by Armour Research Foundation for fabrication development work at the Laboratory (see Monthly Report, July, 1960). Both of these elements rank high in possession of the above desirable properties. Commercial suppliers have been contacted for samples of high strength alloy tubing.

The V-5 w/o Ti-20 w/o Nb alloy tube, as reported in July, was further reduced to an O.D. of 0.239 in. by sinking. A section of the weld area (approximately 1 in. long) cracked in the process. The surface area in general was good; however, several laminations appeared which will have to be removed prior to further drawing operations.

Lengths of welded tubing of the other two vanadium base alloys, (V-10 w/o Ti-3 w/o Nb and V-10 w/o Ti-3 w/o Ta), measuring approximately 0.250 in. O.D. were cold worked in an effort to improve existing fabrication techniques. Again folds, laminations and flaking occurred. Some of the material is in the process of being cleaned and remelted with the hope that the cast material can be fabricated into lamination-free sheet for the purpose of making sound welded tubing.

b. Compatibility of U-20 w/o Pu-10 w/o Fs Alloy with Potential Canning Materials - Several potential canning and barrier materials have been tested for compatibility with U-20 w/o Pu-10 w/o Fs, the reference alloy for the first plutonium-containing fuel for EBR-II.

Diffusion penetration of the jacket can lead to serious consequences. Complete penetration means that the reactor coolant will become contaminated with long-lived fission products as well as plutonium. Potential penetration can be expected to reduce the jacket strength. Reduction in

strength is of special concern where the jacket is being relied upon to restrain swelling of the fuel core under reactor conditions. Solid phase bonding will also make more difficult separation of the jacket from the core required for reprocessing of fuel.

Tests of the following diffusion couples have been made at temperatures up to 800°C for times up to 60 days: Nb-5.3 w/o V vs. Y and U-20 w/o Pu-10 w/o Fs vs. Ti, Mo, V, Y, Nb, Ta, Zr, Zircaloy-2, Nb-5.3 w/o V, and Types 430 and 347 stainless steels. The following conclusions can be drawn from work to date:

1. Ti, Mo, and V show relatively little interdiffusion with U-20 w/o Pu-10 w/o Fs. For example, V will penetrate one mil in approximately 50 days at 600°C and 20 days at 700°C (two mils would take 200 days at 600°C and 80 days at 700°C).
2. Nb, Zr, and Zircaloy-2 show relatively extensive diffusion penetration in temperature ranges of interest.
3. The stainless steels show extensive penetration at 500°C (with 347 stainless, one mil in approximately 9 days at 550°C), and a liquid phase is present above 600°C.
4. Injection casting of U-20 w/o Pu-10 w/o Fs alloy results in a cored, nonequilibrium structure. Heating to 800°C results in extensive grain boundary melting. (It has been shown that the equilibrium solidus for the U-20 w/o Pu-10 w/o Fs alloy is 810°C.) This cored, cast fuel alloy appears to react with Ta, V, and Nb at temperatures as low as 700°C to form a liquid phase. Under vacuum, the liquid phase was found to wet and travel along the jacket material surface. For example, the liquid from a U-20 w/o Pu-10 w/o Fs vs. Nb-5.3 w/o V couple travelled along the Nb-V until it reached a 302 stainless steel specimen with disastrous results. Holding at 800°C for 10 days almost completely dissolved the grain boundary material. Further work, however, is necessary to clarify these observations.

c. Properties of Plutonium-1 w/o Aluminum Alloy - An engineering evaluation of certain physical properties of the Pu-1 w/o Al fuel alloy has been started. An alloy of this nominal composition will be used in a ZPR-type assembly, and is also currently being considered for the Mark IV fuel loading of EBR-I. Six alloy compositions ranging from 0.3 to 3.4 w/o aluminum have been made by induction melting and casting. Dilation experiments between 100°C and 600°C are in progress to obtain expansion coefficients and to determine evidence of transformations.

d. Properties of PuC - Plutonium carbide is under study as a potential high temperature fuel for fast reactors. The carbide is being considered as an alternate for plutonium bearing metallic fuels of the fissium type to overcome the problem of irradiation damage (swelling) without undue sacrifice in thermal conductivity and density. Since PuC is isomorphic with UC, it is expected to exhibit properties similar to those of UC. The higher thermal conductivity and density determined for UC with equivalent resistance to irradiation damage make the carbide more attractive for this application.

Very little is known about the properties of plutonium carbide with suitable compositions for use as reactor fuels. The objective of this program is to characterize arc melted and cast alloys near the plutonium monocarbide composition by determining such properties as thermal expansion, thermal conductivity, and compatibility with jacket materials. Some heat treatments will be conducted to clarify the above-mentioned properties. Arc melting and casting techniques have been developed and property measurements are in progress.

Thermal expansion data of several carbide compositions were reported last month. The expansion coefficient of the 50 a/o alloy was 10×10^{-6} per °C. Working of this alloy produced a phase transformation with a large thermal expansion at about 700°C. This nonequilibrium phase annealed out on cycling to 900°C.

B. EBR-I

1. Mark III Operation and Studies

Routine operation of the reactor on the irradiation run continued on a 3-shift basis for the first twelve days of the month. At that time an increase in the activity of cooling air exhaust was noted over a 12-hour period. Tests of safety rod operation showed that Rod No. 2 was sticking and it was decided to inspect the outer blanket cup. Inspection revealed that in two locations a brick was seized to the brick directly beneath it. The cladding of one of these bricks was ruptured on a lower inside corner. In the other case, although no cladding rupture had occurred, there was a gap between the two seized bricks. Very little oxide was present. The four bricks were replaced and the reactor returned to full power on August 16. Normal operation continued for the remainder of the month.

Analysis of the recent stability tests on the Mark III core has continued. Measurements of the steady-state power coefficient indicated a discrepancy between the static coefficient and an extrapolation to zero frequency of the dynamic power coefficient as measured by oscillator experiments. This was especially true in measurements taken at the normal, high-inlet coolant temperature. A possible explanation is the

existence of a negative power coefficient with a time constant of the order of several minutes. Such a coefficient might not be effective in dynamic measurements with frequencies as low as 0.02 cycles per second, the lowest used in the oscillator tests. However, this negative coefficient would certainly influence the steady-state power coefficient.

Attempts to demonstrate the existence of a delayed power coefficient were made through studies of power changes covering a period of many minutes following a step decrease in reactivity (rod drop experiments). Following a reactivity loss, the reactor power decreased rapidly to a value consistent with the dynamically measured power coefficient. After several minutes, the effect of the postulated delayed term is felt and the power begins to rise slowly over a period of many minutes to a value less than the initial power, but 20-40 percent higher than the power minimum. Such results may eventually explain the 13 percent discrepancy noted between static and dynamic power coefficient values measured for the fully-ribbed, rigid core, and the 24 percent discrepancy noted between corresponding values for the partially-sheared core. More definitive studies of the delayed effect are in progress.

2. Analysis of Mark I and Mark II

Concurrently, analysis of EBR-I, Marks I and II has continued. Attempts to detect thermally-induced motions in the ligament structure of a dummy shield plate from Marks I and II have been encouraging. Alternate bursts of hot and cold water were passed through the ligament structure to simulate the effects of an oscillating outlet coolant temperature. Strain gauges were used to register the amount of deflection. While the results are as yet preliminary, there are strong indications that expansion effects in this region of the structure were not negligible, as previously supposed. These effects may well constitute the source of the negative power coefficient component which has been postulated for Marks I and II, but which has not to date had a physical explanation.

3. Mark IV Development

Methods are being developed for the fabrication of the plutonium-1 w/o aluminum alloy which constitutes the fuel for the Mark IV core.

To date, 13 test melts have been made by the centrifugal casting method. These have produced a series of samples for metallography, physical property measurement and analytical study. A number of fuel slugs have been made and stored for periods of over one month in a can containing a helium atmosphere and a desiccant. They have shown only slight surface discoloration.

C. EBR-II

1. Construction

A summary of the status of EBR-II building construction contracts in progress is given below:

<u>Building</u>	<u>% Completion</u>	<u>Estimated Completion Date</u>
Lab and Service Building (Pkg. 1-A)	100	August 1960
Power Plant and Cooling Tower (Pkg. 2)	99	October 1960
Reactor Plant (Pkg. 2)	98	October 1960
Sodium Boiler Plant (Pkg. 3)	37	November 1960*
Fuel Cycle Facility (Pkg. 3)	37	May 1961

*Does not reflect steam generator fabrication and installation delays.

In addition to the above facilities under construction, the following buildings have already been accepted and occupied; Fire House, Guard House, Pump House No. 1, Fuel Storage and Pump House, Sanitary System, Electrical Substation, and Warehouse.

a. Lab and Service Building - Construction of the Lab and Service Building was completed during the month of August and accepted. The Laboratory has now taken beneficial occupancy of the entire building and the operating staff is preparing the experimental equipment of operation. The Laboratory has had partial occupancy of this building for many months and it has been used as the headquarters for the construction engineering staff.

b. Power Plant and Cooling Tower - The construction of the Power Plant is essentially completed except for installation of the water demineralizing equipment (which was late in delivery), and final checkout of equipment and systems and associated wiring and control connections. Beneficial occupancy has been taken of parts of the Power Plant including the auxiliary boiler wing, office wings, and part of the first floor area. The Cooling Tower was completed and accepted during August and the Laboratory has taken beneficial occupancy of this facility.

c. Reactor Plant - The Reactor Plant is also nearing completion with final checkout of various systems and equipment items in progress. The final leak rate test on the Containment Vessel was completed on August 9, 1960, and although the analysis of the experimental data has not been completed, it is certain that the leak rate was well below the specified limits. After the analysis of the data is completed, a topical report will be prepared including the results as well as a description of the test method.

d. Sodium Boiler Plant - Construction of the building structure is essentially complete and equipment installation is in progress. The secondary sodium pump is installed and the sodium tanks piping and purification system installations are in progress. Installation of control panels and power supply equipment is also in progress. The eight evaporator units are installed and piping to them is in progress. The superheaters are not yet delivered owing to difficulties encountered by the Laboratory in fabrication (see Section IIC6). It is estimated that final delivery of the superheaters will not be completed until November. The piping contractor has experienced some difficulty in obtaining the Schedule 20 piping for the secondary sodium system, which is delaying that area of construction. Because of these difficulties, the present estimated completion date of November 1960 appears unrealistic; however, it is not anticipated that these construction delays will affect system operation.

e. Fuel Cycle Facility - Structural work on the building and cell are in progress. The circular argon cell is of greatest interest since it is steel lined and must be constructed to close tolerances and provide a gas tight closure. Erection of the steel liner and associated appurtenances is proceeding reasonably well to close tolerances. The heavy concrete walls are being poured concurrent with the liner erection. Installation of cell equipment will not begin until construction is completed.

2. Installation of Equipment

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

Package 4 installation in the Lab and Service Building is nearing completion and has included the windows, manipulators, and experimental equipment in the Junior Caves, experimental machine shop, counting room equipment, and mass spectrometer, etc.

During late August, beneficial occupancy was obtained of the machine shop area in the Power Plant, and installation of machine tools is now in progress in that area. Plans are in progress for a similar arrangement in the Reactor Plant to permit equipment installation work to begin on a limited basis early in September to proceed concurrent with construction. It is expected that the Package 4 installation work will be gradually increased in scope until it can be performed on an unrestricted basis sometime in October. At that time, essentially the entire Package 4 effort will be concentrated in the Reactor Plant and Power Plant, and scheduled to achieve dry critical operation of the reactor at the earliest possible date.

Barring unforeseen difficulties and assuming that the above time schedule is realized, it is estimated that the installation work prior to dry critical operation will be completed by about the end of February, 1961. At that time, the major portion of Package 4 installation work will be shifted to the secondary sodium system and sodium filling operations preparatory to sodium filling and wet critical operations. Following power operation of the reactor and power system, the Package 4 installation work will be shifted to the Fuel Cycle Facility.

3. Engineering

Construction engineering for the plant is essentially completed. The H. K. Ferguson Company, the Architect-Engineer, is preparing final instrumentation and control schematics, checking supplier drawings, preparing maintenance and operating manuals, and miscellaneous completion work. This work is scheduled for completion by the end of September. During October all final record documents will be transferred to the Laboratory.

Equipment design and installation design and planning are proceeding concurrent with construction and to a schedule matched to estimated construction completion dates. Equipment design in the reactor and power plants is essentially complete with but a few components still in the design phase. Equipment design for the fuel cycle is in progress including both prototype equipment and final equipment for the Fuel Cycle Facility. The prototype equipment is under test in various mockup areas and in Building 6 where the first fuel loading is under fabrication. Experience obtained from these operations is being incorporated into the design of the equipment to be installed in the EBR-II Fuel Cycle Facility.

Installation design and planning is in progress for the various equipment items and systems in the reactor plant and power plant. This work is subdivided into discrete work units (installation packets) and released to the field in this manner. Approximately 75% of the packets for the reactor plant and power plant are now completed.

4. Procurement

The Laboratory is procuring approximately \$14,000,000 worth of equipment for installation in the EBR-II facility. This includes equipment which was procured prior to the completion of construction design to permit delivery at an early date, and equipment which is critical in nature and over which the Laboratory required direct control. Some of this equipment is installed by the construction contractors (Schedule X), and the remainder is installed by the Package 4 installation contractor. All of the Schedule X equipment for the Package 2 construction contractor has been delivered and installed. A very large part of the Schedule X equipment

for the Package 3 contractor has been delivered; some of it has been installed; the remainder is in storage. No difficulty is anticipated in the delivery of the remaining equipment, with the exception of the superheater units mentioned earlier.

The equipment to be installed by the Package 4 contractor is being delivered in accordance with the installation requirements. No difficulties are anticipated at present for providing this equipment when needed.

5. Component Development - Reactor

a. Reactor Coolant Flow Distribution - Achievement of optimum coolant flow distribution through the reactor core is of critical importance. Any deviation from optimum distribution will result in reduced reactor power level, overheating (and shortened life) of the fuel elements, reduced steam temperature, reduced plant efficiency, and other undesirable effects.

Extensive work has been accomplished to date, both analytically and experimentally, to assure realization of proper flow distribution. Basically, the objectives have been to determine correctly the flow rate required through each "subassembly" (or cluster of fuel elements), to effect a system design which could be expected to provide such flow rates, and then to establish with reasonable certainty that these flow rates will in fact be achieved. The first two of these steps were completed some time ago. The last step, the "proof" that prediction of actual flow distribution will be sufficiently accurate, is still in progress. This experimental work is being carried out by Franklin Institute under contract to, and under the supervision of, the Laboratory.

Some of the questions expected to be resolved by this work are:

(a) Using the orifices as presently designed, what is the average flow rate per subassembly within each row of subassemblies in the reactor? (b) What is the maximum deviation from average flow rate per subassembly within each row? and, (c) Is the flow pattern within the upper (outlet) plenum radial according to design? Additional questions of lesser importance are: (a) How effective is mixing within the upper plenum? and, (b) What is the transient temperature behavior within the upper plenum?

Briefly, the experimental work required to answer these questions has involved the following equipment and procedure. A 0.6 scale model of the reactor vessel, including both inlet plenums (one for the Core and Inner Blanket flow and one for the Outer Blanket flow), the single outlet plenum, and a complete complement of simulated subassemblies was constructed. Also, a small test loop useable for calibrating flow rate versus pressure drop through the 0.6 scale simulated subassemblies was provided. Finally, a larger test loop for calibrating flow rate versus pressure drop through full scale, accurately modeled, subassemblies was erected.

Employing water as the simulated coolant in all units, instrumented flow tests were conducted to enable prediction of EBR-II sodium flow rates through use of analytical techniques based on principles of geometric and dynamic similarity.

Although flow tests are not yet completed, the first preliminary results are now available. These show that the average flow rate per subassembly is very close to that desired, except in rows 4 and 5 (the peripheral rows of the reactor core). The flow in rows 4 and 5 tentatively appear to be 6 or 7 percent lower than expected. The results also indicate that the maximum deviation from average flow rate per subassembly within a given row is acceptably small, and that the flow pattern within the upper plenum is almost perfectly radial.

It is expected that tests now under way will determine with precision the accuracy to which we can predict the flow rate through each subassembly within EBR-II. Future work to be accomplished in the next few months will also give the information desired on mixing and transient behavior.

b. Coolant Flow Distribution Within Core Subassemblies - An additional facet of the general flow distribution problem which is of much importance is that of achieving a proper coolant flow distribution inside each subassembly. Even if total flow rate through a subassembly is correct, any internal maldistribution of flow will result in reduction of achievable reactor power and increase in operating fuel element temperature. Accordingly, it is necessary to be able to predict with accuracy the coolant velocity profile, or mass flow rate associated with each of the 91 clustered fuel elements, within the subassembly.

A model subassembly has been constructed, complete with 91 fuel elements, with dimensions approximately three times full scale. This model is being tested in a water loop. Extensive measurements of internal flow distribution are being accomplished with hypodermic-sized pitot tubes, making complete traverses from wall to wall in various directions and at multiple elevations through the fuel element cluster. These studies, while not yet complete, tentatively indicate that the velocity profile across the subassembly is almost uniform; the average coolant velocity for any one element appears to be not more than about 2 percent lower than the average, or nominal. This is a highly desirable condition and represents a profile superior to that anticipated from theory. Tests are being continued to verify these preliminary results.

c. Fuel Element Stability - In any fast reactor (including the EBR-II), because of the need for achieving very high power density, the fuel elements are of extremely small cross-sectional dimensions. Assurance that such elements can exhibit the mechanical strength required

under the severe operation conditions imposed represents a difficult problem. A major portion of the problem is the achievement of sufficient element strength to resist warpage when subjected to the complicated hydraulic forces induced by the high coolant flow rates employed.

The question of possible fuel element warpage cannot be answered by analysis alone. Experimental demonstration is required. For EBR-II, a prototype subassembly has been operated in a sodium test loop at EBR-II full flow (about 157 gpm) and operating coolant temperature (about 950°F) for approximately 600 hours, equivalent to several months of initial reactor operation. At the conclusion of the test, all 91 fuel elements were found to be in identically the same shape as before the test. There was no evidence of warping nor of vibration. These results indicate clearly that the fuel elements as designed are completely satisfactory in respect to mechanical stability. No further work on this problem is contemplated.

d. Control Rod Drive Testing - Before the reactor may safely be placed into operation, it is essential to know that the control rod drives will operate reliably and that they will exhibit the time-displacement characteristic planned. Because of the extreme importance of the control rod function, experimental verification is required in addition to careful design and analysis. This can be obtained only through extensive testing.

The rod drives for EBR-II have been thoroughly analyzed and designed over a period of years. The major mechanisms have been tested repeatedly, in some cases for many thousands of cycles. Complete drives (actual prototypes except for minor deviations) have successfully undergone rigorous tests in both air and sodium and with simulated control rods attached. The final pre-installation test employing a true prototype operating under actual reactor conditions, will be made.

Equipment for this final test is now being set up. One of the actual fourteen control rod drives fabricated for the reactor (of which thirteen have been sent to the construction site) will be tested with a prototype control rod attached and operating in sodium at 580 - 950°F. This control rod system is identical in every way to those to be installed in the reactor. Testing will include measurement of rod release time, scram time-displacement characteristic, dash pot pressures, and all other possible criteria of performance. In addition, reliability and endurance will be verified by imposing more than 1000 operating cycles.

e. Automatic Flux Control - An automatic flux control system is being designed for EBR-II which will control reactor power plus or minus 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. This will not be a load-following device, but only an automatic flux control to maintain the reactor power at a pre-set level irrespective of load.

A simplified control circuit utilizing an operational amplifier and dual vacuum tube differential amplifier was designed. This particular circuit is based on linear potentiometer characteristics and does not require a custom shaped potentiometer characteristic such as that used in the CP-5 control system. The components for this control circuit were assembled and test data will be taken to evaluate performance.

f. Neutron Source - An antimony-beryllium neutron source is being designed based on requirements established in the EBR-II critical experiments performed in the ZPR-III. To avoid the need for an external neutron shield the neutron source is being designed in two parts. The beryllium section will be placed in the core and will resemble a control rod thimble. The antimony section will resemble a control rod and will be assembled into the beryllium within the reactor core proper. The antimony will be irradiated in the MTR prior to insertion into the reactor core.

g. Startup Instrumentation - Special in-core instrumentation is being provided for initial EBR-II startup; dry critical, wet critical, and the early steps of approach to power operation. The normal operating instrumentation for EBR-II is located external to the core and is unaffected by fuel handling operations. The in-core instrumentation must be moved during core loading operations because it must enter the core through the reactor cover which moves with the fuel gripper. The in-core instruments will, therefore, be installed in two movable thimbles in normal control rod locations. They can then be retracted to permit movement of the fuel gripper. Fission chambers capable of operation at 455°C will be used in order to permit in-core measurement in high temperature sodium. Two separate channels of electronic equipment, high voltage supplies, preamplifiers, log count rate amplifiers, period amplifiers, scalars, and audible count rate circuits will be installed during the experimental startup program.

h. Control Rod Reactivity Generator - A sinusoidal reactivity generator and wave analyzer are being designed to measure the transfer function of EBR-II. The reactivity generator will utilize a specially designed rotary mechanism which will fit into a control rod location. A velocity servo accurate to 1%, over a 10 to 1 speed ratio, will be used in combination with gear changes to cover the frequency range from 0.01 to 100 radians/second. The wave analyzer used to measure amplitude and phase shift will be similar to the one used on EBWR.

A Servo-Tek speed regulator was measured to determine the accuracy of speed control over a 100 to 1 speed range. At full speed (3,600 rpm), the speed variation is $\pm 0.4\%$; at 10% speed, the variation is $\pm 1\%$; and at 1% speed, the variation is $\pm 40\%$. An inertia load (7.6 cm diameter, 1.9 cm thick brass disc) was then added to the output shaft and the

test repeated. The variation was reduced (100% speed $\pm$.125%, 10% speed $\pm$.3%, and 1% speed $\pm$ 5%). From the above results, it can be concluded that the drive is only usable over 10 to 100% speed with a 1% control accuracy.

6. Component Development - Steam Generators

As was reported in the Monthly Report for July, 1960, considerable difficulty was experienced in the fabrication of the last few evaporator units for EBR-II. This difficulty, which arose after fabrication of the units was well along and apparently proceeding in an orderly manner, was associated with a different heat of welding wire for the tube-to-tube sheet welds than that with which the weld procedure had been developed. After considerable time and effort it was determined that satisfactory welds could not be produced with this material. This necessitated a very thorough re-examination of the welding procedures, the welding wire requirements, and the metallurgical processes. Both the automatic welding operation and the metallurgical processes are very complex and involve many variables. Although a complete understanding of the entire process has not yet been established, it is now thought to be sufficiently well understood to complete the fabrication of the superheater units.

The problem is concerned apparently with very high temperature oxidation kinetics during the welding process (metal temperatures in the arc approach 4000°K), and stems from the fact that a small quantity of oxygen must be present in order to maintain the arc. Furthermore, controlled oxidation of the weld material is required to produce some FeO in the melt to produce the necessary fluidity and wettability of the metal. Consequently, excessive oxide is formed which must then be reduced prior to solidification. This reduction of the greater-than-normal quantities of oxide present requires weld rod analysis which is different than that used by the more common welding procedures. Some reduction is effected by carbon in the alloy, however, carbon reduction alone produces porosity in the welds (production of CO). Reduction is also accomplished by silicon and manganese in the alloy. This reduction process forms a very complex slag containing SiO, MnO, and FeO, the properties of which must be correct to avoid slag inclusion and resulting severe cracking in the last weld material to solidify.

After considerable investigation, it has been found that alloy wire containing both high carbon (0.12 - 0.15 percent) and high silicon (0.60 - 1.00 percent) will produce satisfactory welds. Wire of the proper composition has been obtained recently and has been employed in the fabrication of the last evaporator unit. Wire from the same batch is now being redrawn for use on the superheaters. It is believed that this will permit an orderly completion of the superheaters and an estimated schedule for completion on this basis has been prepared:

Estimated Completion Schedule for
EBR-II Superheaters

(Shipping Dates)

Unit No. 1	October 18, 1960
Unit No. 2	November 1, 1960
Unit No. 3	November 15, 1960
Unit No. 4	November 29, 1960

With the exception of the bonded tubing required, all of the components necessary for the four superheaters plus one spare are now on hand. Bonded tubing is continuing to be received for bond inspection at the Laboratory, and enough is now on hand for three units.

7. Component Development - Fuel Reprocessing Facilities

A remotely operated fuel reprocessing-refabrication plant is being built for the direct recycle of fuel in the EBR-II reactor system. Apparatus and components are being developed and tested in a full-scale mockup of a section of the Argon cell.

As reported in July, a gamma dose of 4.0×10^9 rad appreciably decreased insulation resistance of MI cable samples sealed with Temporell #741. The samples were then baked in an oven at 100°C and the original insulation resistance of greater than 200 megohms was restored. Evidently, moisture pickup caused the insulation resistance to decrease.

Samples of reinforced mylar-dacron and mylar-nylon plastic sheet, proposed for disposable manipulator booting, are being irradiated. After 1.9×10^8 rad gamma dose the samples were brittle, some spalled and cracked on flexing.

The production of enriched 10 kg uranium-fissium ingots for the manufacture of EBR-II fuel was completed with the production of ingots 75 through 80. Thirty kilograms of enriched scrap pins from the fuel pin casting step were remelted into 3 ingots.

In melt refining, metallic vapors will be evolved during the heating cycle. A Fiberfrax disposable fume trap was proposed for absorption of this material. However, in test runs with a molded Fiberfrax absorber, excessive dross formation led to melt pouring failures. These pouring difficulties encountered when using the uranium melting and pouring furnace have been resolved by proper degassing of the traps prior to use.

Design of refabrication equipment for the Fuel Cycle Plant continued. An analysis of the gamma heating effect in various arrays of fuel pins was

completed and indicated several areas where trouble may be expected from this source. Experimental determination of the heating effect in the depalleting hopper will be necessary. Designs were completed on casting furnaces.

8. Fabrication of Initial Fuel Loading

a. Fuel Pins and Fuel Elements - The status of production of EBR-II fuel elements is shown in Table II. A total of 9,350 fuel elements are required.

Table II. EBR-II Fuel Element Production

<u>Operation</u>	<u>Cumulative Nos. of Fuel Elements</u>	
	<u>July 29</u>	<u>Aug. 20</u>
(1) Fuel Pins Cast	5,462	7,117
(2) Fuel Pins Processed, Inspected & Accepted	3,655	5,845
(3) Fuel Elements Assembled & Welded	3,335	4,989
(4) Fuel Elements Bonded, Inspected and Accepted	1,291	3,028

A manually operated inspection line was set up to supplement the remotely controlled inspection of fuel pins. The cumulative inspection-rejection of fuel pins as of August 20, from all causes has been 806 of 6,651 inspected, or about 12 percent, compared to 17 percent as of July 29.

b. Blanket Rods - A total of 4,526 inner and outer blanket rods have been loaded, bonded, inspected, and accepted for reactor use as of August 20. This represents about 40% of the number required for the reactor loading. The required loading schedule for inner and outer blanket rods of 1,080 per week is currently being maintained. Nondestructive eddy current inspections are used to determine bond integrity and sodium fill height in each blanket rod. Equipment is being assembled for salvaging the uranium slugs from damaged and rejected blanket rods.

c. Stainless Steel Hardware - The procurement of stainless steel parts for the EBR-II core is continuing. Approximately 67% of some 60 items has been completed and is stored. The remaining items average about 30% complete.

The assembly and welding of the stainless steel components for fuel and blanket rods is continuing. The assembly and welding of stainless steel components for 7,500 upper and lower blanket rods has been finished. This phase of the program is now completed. The stainless steel components for 10,051 inner and outer blanket rods have been assembled, welded and delivered to the canning line. The total quantity of stainless components

for inner and outer blanket rods to be produced is 16,000. A total of 12,645 stainless steel components for fuel rods has been produced and delivered for fuel rod assembly. Stainless steel components for 21,000 fuel rods are to be fabricated. Although no additional components were finished during this period, 3,700 were assembled for welding. The totals shown above for components to be produced are in excess of those required for a single reactor loading.

9. Process Development

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

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

a. Melt Refining Process Technology - The first of a series of melt refining demonstration runs using 400 gram charges of irradiated uranium-fissium fuel alloy has been completed. Uranium-fissium fuel pins of the type to be used in the initial core loading of EBR-II were irradiated to a total atom burnup of about 0.96 percent and cooled approximately 30 days prior to the experiment. The uranium was ten percent enriched in U^{235} . Although some difficulty was encountered in pouring the molten metal after three hours of melt refining at 1400°C , a yield of 72.4 percent was obtained. This yield was considered adequate for the first run of the series and for the small size of the melt. Analytical results from the run are not yet available.

An experiment was also performed on the evolution of xenon and krypton from a single irradiated fuel pin as it was heated to 1120°C at a rate of 5°C per minute. The results have not yet been analyzed. No unusual or unforeseen behavior of the pin was evident during the experiment.

A melt refining run was made using conditions selected to approximate the most unfavorable conditions which may prevail in the Argon Cell of the Fuel Cycle Facility after the pins have been decanned. Two kg of sodium coated pins which had previously been stored for 66 hours at 300°C in a 5 percent nitrogen - 95 percent argon atmosphere were melt-refined

at 1400°C for 3 hours in a zirconia crucible. The 70 percent yield obtained represents about a 10 percent decrease, based on total charge, as compared to similar runs made without the storage period.

b. Processing of Melt-Refining Skulls - In the oxidation of melt refining skulls, the presence of cerium appears to inhibit the volatilization of molybdenum and tellurium oxides. This effect is believed to result from the formation of complex oxides or solid solutions of oxides. Ruthenium oxide volatilization appears to be unaffected by the presence of cerium. The behavior of technetium has not been determined.

c. Engineering Development for Liquid Metals Processes - The previously completed dragout process demonstration runs (June Monthly Report) show excessive uranium losses in the crucible residue after the uranium-zinc solution transfer and in the zinc waste ingot. It is expected that crucible reuse and zinc recycle combined with better phase separations will decrease this loss.

III. STUDIES AND EVALUATIONS (040116)

Three reviews have been carried out at the request of the Commission:

a. A review of the Martin Company's proposal for a 1500 Kwe reactor for McMurdo Sound, Antarctica, was made. Written comments were furnished to the Commission. However, since the proposal contained proprietary information, the comments are not included in this progress report.

b. A review of two volumes of the modification, updating and additions to the AEC's 1959 ten-year program for boiling water reactors was completed. The work in these volumes was performed by Allis-Chalmers. The first of the two volumes reviewed was titled, "Economic and Technical Status and Potential Small Boiling Water Power Reactors." The second volume was titled, "Modification and Updating of 1959 Study for AEC Ten-Year Program - Boiling Water Reactors." Of the comments offered on these reports, the most significant was that the magnitude of the research and development program seemed unprofitably large.

c. Two APDA-prepared reports, "Addendum to TID-8518, Book 1" and "Addendum to TID-8517, WASH-1025," were reviewed. A written list of specific comments was submitted to the AEC.

In addition to the above, the following form reports or proposals are in the process of being reviewed, also at the request of the Commission.

a. A variable moderator reactor (VMR) proposal.

b. A proposal for the development of high temperature thermocouples.

c. A proposal for development of a fuel failure detection system.

d. Two review reports on the SGR concept. One of the reports is an updating of the 1959 SGR status report; the other deals with designs of two small reactors, 12 and 44 Mwe.

IV. REACTOR SAFETY (040117)

A. Thermal Reactor Safety Studies

1. Fuel-Coolant Chemical Reactions

The program is directed toward determining the nature and extent of chemical reactions that may occur in pressurized water or steam. 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 residue indicates the surface area exposed to reaction. A second method consists of heating the metal inductively and then subjecting it to a steam pulse to induce a metal-steam reaction.

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

Regular operation of the high pressure condenser discharge apparatus is under way. A series of runs with 60-mil zirconium wires will be initiated to provide detailed information on the effect of pressure on the zirconium-water reaction.

Condenser discharge runs reacting room temperature water with uranium resulted in the liberation of hydrogen gas; only negligible quantities (~1%) of hydrogen are dissolved in the metal residue.

Studies of the aluminum-water reaction by the pressure-pulse method have been resumed in a heated apparatus in the near atmospheric pressure range. Four runs have been completed.

The eighth series of metal-water experiments in TREAT is under way. Runs will be made with 30-mil, fully enriched uranium wires to provide a comparison with results obtained by the condenser discharge experiment. Two transients will be conducted on unclad 90 Al-10 U₃O₈ cermet pins to study directly the aluminum-water reaction.

2. Kinetics of Oxidation and Ignition of Reactor Metals

Studies are being made of the oxidation and ignition kinetics of the metals uranium, zirconium, plutonium and thorium, in order to provide information leading to an understanding of the reactions. This knowledge should make it possible to minimize the hazards associated with handling these nuclear reactor materials. Isothermal oxidation on microscope stage, shielded ignition, burning curves, rate of propagation of burning

foil and burning temperatures are the techniques being used. In the continuing study of ignition and burning of uranium, zirconium, and plutonium, more emphasis is being placed on the burning process. Burning propagation rate studies provide a useful tool to observe the effects of many variables. Addition of halogenated hydrocarbons to air in which uranium or zirconium specimens are burning has been observed to lower markedly the burning propagation velocity. This effect is being investigated.

In the continued study of the ignition of uranium, it has been shown that an aggregate or array of pieces of the same specific area ignites at a considerably lower temperature than an individual piece. The condition of decreased heat loss from an interior piece surrounded by other pieces which are also generating heat by reaction may be the explanation of uranium fires which have occurred.

Continued studies of the effects of halogenated hydrocarbons on the burning propagation of uranium and zirconium foils indicate the possibility of preventing continued combustion of zirconium foils or turnings thicker than 0.5 mm (2 mil). This may include the entire practical range of turning thicknesses.

The effect of specific area on the ignition temperatures of spherical uranium powders has been examined over the range from 2 through 70 sq. cm. per gram. The Frank-Kamenetskii theory of thermal explosions appears to be applicable by considering the effective thermal conductivity of the powder which changes with particle size.

B. Fast Reactor Safety Studies

1. TREAT - Experimental

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

a. EBR-II Mark I - Dry Samples - Three ranges of meltdown behavior of dry EBR-II Mark I fuel samples subjected to short (about 0.2 to 0.5 sec.) power bursts had been determined by experimentation prior to the report period as follows:

(a) Samples with maximum cladding temperatures less than about 950°C were below the failure threshold, although some loss of sodium bond and localized alloying between the uranium fuel alloy and the stainless steel cladding were observed.

(b) Maximum sample cladding temperatures in the range 950°C to about 1015°C marked incipient failure. Loss of bond and fuel cladding alloying became extensive, and mild failures sometimes occurred. The clad failures were localized, and there was a tendency for essentially all the fuel alloy to flow from the cladding through the points of failure.

(c) Maximum cladding temperatures greater than 1015°C marked the region of extensive failure, in which cladding was observed to have failed relatively uniformly over element lengths of about 2-5 cm, with essentially all the fuel alloy being expelled vigorously from the cladding under the influence of the vapor pressure of bond sodium, to form annuli of "frozen" fuel-steel alloy on the walls of the graphite meltdown crucibles.

Seven samples which had been run under different conditions and which were examined during the month of August, disclosed no significant change in the patterns observed previously. The test conditions included:

(a) 0.2 second duration transients in which the sample energy inputs were more severe than previous ones, and up to about 75% greater than necessary to produce sample failure;

(b) flattened power bursts of relatively constant power (within about $\pm 15\%$), of durations in the range of 0.8 - 2 seconds.

Qualitative changes were noted, however. The results indicate that the more severe the "short" power pulse, the greater the total amount of cladding dissolved and the greater the tendency for failure to occur along the entire length of the element, rather than at one or two places located randomly. Further, the failures observed for the samples given constant power bursts appeared to show somewhat less vigorous expulsion of fuel alloy than that noted for samples given the same total energy in shorter reactor transients.

Future experiments are being planned to explore the effects of higher sample energy inputs and longer, lower constant power bursts, in attempts to determine if such experiments can produce new modes of failure, or more pronounced modifications of those already observed.

b. Fermi I - Dry Samples - Meltdown behavior observed for dry Fermi I fuel samples given short power bursts may be summarized as follows: It was found that elements could be raised above the fuel melting point without producing failure. Considerable warpage was noted in such cases. Mild failure consisted of a small amount of liquid fuel alloy escaping from a small flaw in the cladding and flowing down the cladding under the influence of gravity. As the total sample energy input increased, the amount of material escaping and the amount of failed cladding both increased.

Work is still under way on sections of Fermi I samples to determine the compositions of the various material layers in the fuel cladding interface region, to shed light on the nature of the failure initiating mechanisms.

c. EBR-II Refractory Metal Clad - Refractory-metal clad EBR-II fuel elements have been studied also in TREAT. The results indicate that the temperature of samples can be raised 150°C above the melting point of the fuel for short times without failure. Studies of the clad-fuel interaction, and the evidence for movement of molten fuel inside the cladding during the transients are still under way.

d. EBR-II Mark I Stagnant Sodium Samples - Experiments on meltdown of fuel samples inside stagnant sodium offer an opportunity to observe meltdown characteristics produced by the presence of a liquid metal coolant at the surface of samples, as well as characteristics of meltdown product movement introduced by the coolant. Work is continuing in preparation for the first meltdown experiments in stagnant sodium. Preliminary energy input reactor calibration runs have been performed in TREAT with a stagnant sodium capsule containing a 3% enriched sample.

2. TREAT - Reactor and Component Development

a. Kinetics of the TREAT Reactor - During the shakedown period following the first approach to criticality for TREAT in February 1959, a series of physics experiments were performed on the reactor to provide experimenters with empirical data on its performance characteristics to be used in planning experiments, and to provide information on a relatively simple configuration for use by theoreticians in checking their calculations. These measurements, plus some more recent additional data on TREAT performance, have now been correlated and analyzed, and the results are being assembled in two reports, ANL-6173 on the results of experiment, and ANL-6174 on the comparison of theory and experiment. Both reports are now in the final draft stage.

In general, the reactor has performed very much as predicted in early design calculations. The design flux integral has been exceeded, so that at a maximum hot spot fuel temperature of 400°C, the integrated thermal flux exceeds that of any other pulsed reactor by nearly an order of magnitude. This number can be increased by an additional factor of 2 or 3 before reaching an operating condition where corrosion of the cladding might be serious. The reactor can provide twice the energy required to melt a natural uranium sample without exceeding its conservative temperature limit.

Agreement between theory and experiment has been quite good on general kinetic behavior. The gas model of energy interchange between

graphite and neutrons has been found to give a 25% higher temperature coefficient than observed experimentally, but gave better agreement on prompt neutron lifetime. Agreement between spectra measurements, i.e., fission and cadmium ratios, and calculation has also been reasonably good.

b. Control Rods - Two control rods which had been filled with boron carbide powder suspended in epoxy resin were inspected by X-ray after two weeks' normal operation in the reactor and 50 scrams. Since there was no noticeable change in the material as shown by this inspection, nor apparent change in the reactivity worth of these rods, all of the control rods will be replaced with rods of this type to prevent reactivity worth changes associated with the movement of the boron carbide powder in the original control rods.

c. Gamma Camera - Extensive testing, in TREAT, has disclosed that the Mark I design of the γ -ray camera system being developed by the Naval Research Laboratory under subcontract to the Laboratory, was not acceptable because of low signal strength, high background from γ -ray "glow" of container materials, and high general background. A series of tests at TREAT on a Mark II scanning device were completed by the end of August and are now being evaluated by NRL personnel prior to subsequent ANL review of the system capabilities.

d. Temperature Measurement - It would be highly desirable to be able to measure the internal temperatures of meltdown samples by a technique requiring no foreign body inside the sample. An ultrasonics device is being developed in an attempt to measure an "averaged" internal sample temperature by measuring the time between the introduction of a sound pulse at one end of the sample and the return of the echo. The output recording system has been modified to give greater stability, yet retain a response time of the order of one millisecond. Before and after tests on an ultrasonics transducer crystal being given a continuing series of irradiations in CP-5 indicate that it can receive a total dose equivalent to that of the design TREAT burst without suffering any appreciable damage.

V. NUCLEAR TECHNOLOGY AND GENERAL SUPPORT (040400)

A. Reactor Fuels and Materials Development

1. Ceramic Fuels

a. Refractories for Reactor Use - The success of UO_2 and $\text{ThO}_2\text{-UO}_2$ as reactor fuels has stimulated interest in other oxide systems for nuclear power applications. Volatilization of UO_2 at elevated temperatures, however, restricts the use of these fuels. This program is concerned with the development and testing of ceramic fuels containing UO_2 in solid solution with various oxide additives with the objective of improving high temperature stability. The phase relationships, stability, and testing of cubic fluorite structured compositions are of prime interest.

X-ray powder patterns of sintered La_2O_3 compositions, previously described in July, were re-examined. X-ray lines reported as simple cubic were found to match the hexagonal structure of LaAlO_3 . Spectrochemical analyses needed to determine Al_2O_3 impurity content are still in progress. The comminution of powders using alumina pebbles is thought to be the source of contamination.

Thirteen $\text{La}_2\text{O}_3\text{-UO}_2$ compositions are now in progress. Fabrication of high density bodies using a one-step sintering procedure will be attempted to reduce mill contamination.

b. Fabrication Studies of UC-PuC - The current interest in UC-PuC as a fast reactor fuel makes necessary the development of fabrication techniques and procedures for this material. Three methods which are employed in the making of carbides are the reaction of finely divided metal powder with a gaseous hydrocarbon, arc-melting the metal with graphite, and the reaction of UO_2 with carbon. The investigation of preparation techniques by ceramic methods ($\text{UO}_2\text{-carbon}$), the fabrication of UC-PuC into various shapes, and the investigation of physical properties and characteristics of various UC-PuC bodies are the purposes of this work. The immediate objective is to produce samples of UC-PuC of acceptable density and analysis.

PuC and 80 w/o UC-20 w/o PuC pellets were polished and examined microscopically. This examination indicated the presence of a single phase in both materials. X-ray diffraction patterns of ground 80 w/o UC-20 w/o PuC samples show a solid solution of UC and PuC plus a slight contamination of U_2C_3 .

c. Urania-Thoria Bodies - Solid solutions of urania-thoria have been proven of value as reactor fuels. For the most part interest in these fuels has been concentrated on compositions of low urania content although

continuous solid solution between UO_2 - ThO_2 is possible with suitable sintering procedure. The solid solution system is somewhat unique in that a continuous, stoichiometric solid solution series is formed when UO_2 - ThO_2 compositions are sintered in hydrogen, while a limited, nonstoichiometric solid solution series occurs on sintering U_3O_8 - ThO_2 compositions in air. There has been some controversy as to whether the deviation from stoichiometry is desirable in this fuel system. This program has been concerned with the investigation of both stoichiometric and nonstoichiometric UO_2 - ThO_2 solid solutions of interest as nuclear fuels. The study has been directed toward, (a) the determination of the stoichiometry of the solid solution and its relationship to stability of the material, (b) the densification behavior of urania-thoria compacts, (c) physical property measurements, and (d) microstructure evaluation of solid solutions of high density.

Progress in the past month has been limited to preparation and treatment of samples for microstructure studies. Urania-thoria solid solutions formed by both air and hydrogen sintering at various temperatures are being made ready for microscopic study. The materials to be studied will represent stoichiometric and nonstoichiometric solid solutions as sintered and after subsequent oxidation of elevated temperature.

The samples above will also be used for measurements of various physical properties. The sintered densities possible in urania-thoria solid solutions and coefficients of thermal expansion will be determined.

d. Urania Glasses - The use of glasses as reactor fuels has received almost no attention, although glasses have characteristics which make them interesting as ceramic fuel materials. Glasses are inorganic liquids which have such a high viscosity at room temperature that they are, for all practical purposes, solids. At elevated temperatures, glasses exhibit no sharp melting point but a gradual decrease in viscosity, with resultant softening, until a fluid condition is reached. The stresses set up in glass at low temperatures may be annealed out by heat treatment. The object of this study is to make a preliminary investigation of two possible applications of glass in nuclear fuel materials: (a) glass as a solvent for the dissolution of urania fuel, and (b) glass as the matrix for a dispersion of UO_2 , metallic particles, or other fuel compounds.

During the month of August, X-ray patterns have been made of the devitrification products of Na_2O - K_2O - UO_2 - SiO_2 and PbO - UO_2 - SiO_2 glasses. Identification of the phases present is in progress.

A preliminary test setup has been made for evaluation of the fabrication parameters involved in hot pressing of glass-crystalline mixtures. A permanent apparatus will be built on the basis of the information obtained.

Urania bearing glass has been cast into stainless steel tubing for compatibility test samples. Extended time tests at elevated temperature will be carried out to evaluate stainless steel as a clad material for irradiation samples.

e. Uranium-Thorium Sulfide - Although uranium monosulfide has potential as a reactor fuel because of its high uranium atom density coupled with a high melting point and semi-metallic properties, no systematic study has been run on sintered material. This program is directed toward the development of pure, dense US bodies and includes measurement of basic physical and chemical properties. US-ThS solid solutions are also under investigation.

An attempt was made to homogenize and sinter uranium monosulfide in one step instead of the two separate steps presently used. Pellets of US (low temperature mixture, Batch No. 14), pressed at 10,000 to 50,000 psi and vacuum fired for one hour at 1800°C remained intact, but the highest density obtained was only 66% of theoretical.

Other portions of the US (Batch No. 14) were homogenized at 1500°C to 1900°C, ground, pelletized and sintered. A theoretical density of 82.3 to 85.8% was obtained by sintering for one hour at 1900°C, and 80.2 to 85.1% by sintering for one hour at 2000°C. The fired density increased with homogenization temperature, and the weight loss correspondingly decreased. The observed weight loss was considerably less than previously experienced and may indicate more pure US. Chemical analyses are in progress. Polished sections of the pellets fired at 2000°C showed none of the unidentified secondary phase which has appeared in varying degrees on the grain boundaries of previously sintered material. X-ray diffraction pattern of the homogenized US (Batch No. 14) showed in all cases the presence of a phase, possibly UOS, in very low concentration. The lattice parameter (a) increased from 5.4892 Å at 1500°C to 5.4926 Å at 1900°C and is believed to indicate an approach to stoichiometry as explained in the Monthly Report for May, 1960.

The modulus of rupture was determined for some pellets of varying density; however, it does not appear to be related directly to the density.

2. Corrosion Studies

a. Corrosion Resistant Aluminum Alloys - The mechanical properties of wrought Al-Ni-Fe corrosion resistant alloys, particularly their poor elevated temperature creep resistance, prevents their use in certain reactor applications. Laboratory-size extrusions fabricated from milled aluminum alloy powder have shown the same excellent corrosion resistance as the better wrought alloys and at the same time vastly improved hot creep and tensile properties.

An effort is being made to have a commercial source (ALCOA) fabricate powder tubing suitable for cladding a UO_2 pellet loading of the EBR. The first attempt was unsuccessful from the standpoint of corrosion resistance. Dimensional tolerance and mechanical strength appeared adequate. ALCOA is trying a second time, utilizing a powder obtained from another source. The extrusion conditions are also being varied in an attempt to improve the corrosion resistance.

Concurrently, an attempt is being made to improve the mechanical properties of wrought Al-Ni-Fe alloys by adding other elements in low concentration (0.1 to 0.2%) and by adding dispersed solids through chemical reactions in the molten metal. An improvement of the order of 35% in the ultimate tensile strength of X8001 at 315°C has been accomplished by the latter technique using $\text{Ni}_3(\text{PO}_4)_2$ or $\text{H}_4\text{P}_2\text{O}_7$, presumably giving AlPO_4 suspended in the melt.

3. Irradiation Studies

a. Irradiation of UC-PuC and PuC - The usefulness of metallic fuels is, as is well-known, severely limited at high temperatures by the phenomenon of swelling. Ceramic fuels are less subject to swelling and are therefore being actively developed for high temperature applications. Fast reactors require fuels with not only good high temperature properties but also high fissile atom densities. Thus, UC, with a uranium density of 13.0 gm/cm^3 , is more attractive than UO_2 , with a uranium density of 10.5 gm/cm^3 . Furthermore, for the temperatures of interest, the thermal conductivity of UC is an order of magnitude higher than that of UO_2 . Of even greater interest is PuC, since fast reactors will operate most efficiently on plutonium fuels.

For fast breeders the plutonium fuel should be diluted with as much fertile uranium as possible in order to enhance internal breeding. A typical carbide fuel composition for a medium-sized reactor would be UC-20 w/o PuC.

For compact fast reactors, where high power densities are sought, PuC is a fuel of great interest. Its favorable plutonium atom density (13.0 gm/cm^3), relatively high melting point (about 1600°C), and a thermal conductivity estimated to be comparable to UC, provide a potentially large advance in the design of small high temperature fast reactors.

Three EBR-I fuel rods have been prepared with carbide fuel and are under irradiation (see Monthly Report, May, 1960). Two of the rods contain alternating 1 in. long specimens of PuC and UC-20 w/o PuC. The specimens in the two rods were prepared by arc casting. The third rod contains UC-20 w/o PuC specimens made by pressing and sintering. This

rod is thermocoupled. The most recent readings at EBR-I show a central temperature of 365°C. This is close to the temperature calculated assuming that the fuel has a thermal conductivity close to that of pure UC.

Since the uranium in the carbide specimens is fully enriched, the UC-PuC specimens are fissioning at approximately the same rate as the PuC specimens. The metal atom burnup achieved to date is estimated to be near 0.1%.

4. Nondestructive Testing Techniques

a. Ultrasonic Techniques - Ultrasonic waves are useful for measuring material properties and the nondestructive evaluation of reactor materials. The ultrasonic techniques program at the Laboratory includes three different problems: (a) the measurement of elastic constants; (b) Lamb wave propagation studies; and, (c) ultrasonic attenuation studies. The elastic constants program is primarily concerned with the measurement of the elastic constants of single crystals of zirconium and uranium. The elastic constants are calculated from measured velocities in different directions in the crystal.

Lamb waves are propagated in thin specimens. The velocity and mode of propagation of such waves depends upon the thickness of the material and the frequency of the wave. Lamb waves are being investigated in detail to determine the usefulness for evaluating material properties.

Ultrasonic velocity and attenuation measurements as a function of temperature are useful in determining what happens to engineering materials at high temperatures.

The pulsed ultrasonic equipment for measuring the elastic constants of single crystals has been completed. The technique is based on the single crystal technique developed at Bell Telephone Laboratories. Modifications have been made on the equipment in order to achieve better results and stability.

The majority of the measurements of the elastic constants of alpha zirconium at 20°C have been completed. These include the determination of C_{11} , C_{33} , C_{12} , C_{44} and C_{66} . Cross checks have shown that these data are probably accurate to within 0.1% even before the completion of the various small corrections which will be made later. Most of the measurements have been made using ultrasonic frequencies in the neighborhood of 50 mc/sec. Measurements are now being made at higher frequencies to check on possible diffraction effects. These effects seem to be negligible based on the data obtained so far.

In addition, an analysis of the effect of the buffer-seal-specimen interface is being undertaken to determine the exact corrections necessary to account for phase shift across these boundaries. The main experimental difficulty has been the maintenance of the specimen temperature to within 0.2°C regardless of the wide variation in ambient temperature. Because of the precision of this method of measuring the elastic constants, variations in specimen temperature of this magnitude are observable in the data.

Upon completion of this set of measurements, the specimens will be degassed and a known amount of hydrogen will be added. The measurements will then be repeated to determine the effect of hydrogen in the lattice on the elastic constants of alpha zirconium.

To date all the experimental work on Lamb waves has been carried out with the transmitter and receiver on the same side of the specimen. A through transmission technique is now being used for the detection of Lamb waves. Using the through transmission technique the phase velocity versus frequency-thickness product was verified over a wide range of frequency-thickness values. The separation of Lamb waves from other signals that are present when using this technique is not experimentally a simple task. Consequently, the through transmission technique appears to have no advantage over the reflection technique.

An attempt is being made to use the concept of "leaky waves," that is, loosely bound waves, to explain the mechanism by which Lamb waves are radiated from a specimen.

An investigation of the effect of stresses on Lamb wave propagation is being initiated. The best shape for the fatigue specimens has been determined. Three thin aluminum plates have been fatigued with a stress of about 10^{15} lbs/in². All three of these plates were checked before and after stressing. No difference in velocity of the various Lamb modes could be determined. If such an effect does occur it is probably small and can not be detected with the present experimental setup. Further refinements of the experimental setup are in progress.

Preliminary experimental work has begun for measuring the ultrasonic velocity and attenuation of specimens at high temperatures. A prototype specimen has been made to check the technique at room temperature. The specimen selected is an aluminum bar 10 in. long x 1 in. diameter. The quartz crystal transducer is mounted on one end of the specimen. Five groups of echoes returning from the opposite end and from the sides of the specimen have been found. The interpretation of the various echoes has not been fully explained.

A vacuum furnace for use in this study has been designed. The furnace will contain an evacuated heating chamber and a water jacket for cooling the specimen where it enters the air. The requirements for the necessary vacuum system to be used in these experiments have been determined.

Work is continuing in an effort to determine the best medium for bonding the quartz to the specimen. It has been found that an aluminum foil can be bonded to quartz crystal satisfactorily with household cement.

b. Neutron Techniques - This investigation is concerned with the detection of neutron images with the long range objective of being able to apply neutron radiography as a useful inspection method. Neutrons have little influence on normal photographic emulsions so that it is necessary to use intermediate layers which emit photographically detectable radiation when bombarded with neutrons.

Intermediate metallic materials which have been investigated thus far include cadmium, silver, gold, indium, and rhodium. All except cadmium are primarily beta emitters, while the photographic effect produced by cadmium screens results from emitted gamma radiation. Optimum screen thickness information has now been obtained for cadmium, silver, and indium. A detailed comparison of the merits of these various screen materials has not yet been made, but preliminary data indicate that, in the neutron flux presently being used (10^5 n/cm²/sec), rhodium screens produce the best film density for a given neutron exposure, followed by indium, cadmium, silver and gold in that order.

The photographic response of a boron-loaded, ZnS neutron scintillator material has been tested for both neutron and gamma radiation exposure with several types of X-ray film. For a given neutron exposure, type F film yields the best film density among the film types tested. These include Kodak X-ray film types KK, K, No Screen and F. Type KK film has the best response, however, if the relative photographic effect of both neutrons and gamma radiation is considered.

c. Eddy Current Techniques - The eddy current program (high frequency electromagnetic waves) is concerned with two major problems; the development of new and better eddy current techniques and the application of such techniques to the inspection of reactor components such as tubing.

An eddy current instrument is being developed for inspecting welded stainless steel tubing. In order to achieve a fast rate of inspection an encircling coil was first investigated. This coil, incorporated in a differential type of eddy current system, gave poor results. The experiments exhibited a rapidly varying "background" signal which was difficult

to interpret. A dual frequency type point probe instrument developed by the nondestructive testing group is being tried on the stainless steel. Preliminary results have been encouraging.

When it became apparent that the pulsed-field through transmission system offered substantial advantages over conventional eddy current equipment for some applications, an effort was made to modify the original concept to allow operation from one side of the test specimen only and still retain the high resolution of the new method. This pulsed-field echo concept was shown to be practical in May of 1960. The electronic instrumentation necessary to make this new approach useful for practical test problems is nearly complete, and preliminary tests have been encouraging. For some applications, notably the detection of nonmetallic inclusions in good conductors, it definitely produces better results than the older equipment. Additional tests are being undertaken to find other areas in which it might produce superior results.

B. Reactor Components Development

1. Development of Viewing Systems

a. Glass Scintillator for Neutron Detection - An improved glass scintillator for use in a neutron detector has been developed which can be used as one of the main components of a slow-neutron time-of-flight spectrometer. The glass scintillator previously used was a "Cabal" type glass containing about 55 mole percent B_2O_3 and 3.6 mole percent Ce. Experience developed in melting this type of glass in an inert atmosphere, in connection with studies on radiation induced coloration, was applied to the problem of preparing a series of cerium-activated glasses with a very high boron content. After detailed adjustment of the composition, and perfection of the melting procedure under highly reducing conditions, a sodium alumino-borate glass containing 73 mole percent B_2O_3 and 8.0 mole percent Ce was melted. This glass is water-white with very high light transmittance for wave lengths greater than 3800 Å. This glass has a neutron detection efficiency 25-30 percent greater than the previous glass in the energy range of 10^2 to 10^4 ev. and is considerably better than any other type of detector. It is planned in the near future to prepare a number of larger samples of this glass (about 2.5 cm dia. by 1 cm thick) for use with the Argonne fast chopper.

This systematic study has shown that the neutron pulse height increases with increasing cerium content up to a maximum and then decreases with further increases in the cerium content. It was found that the molar content of cerium required to obtain a maximum pulse height for each B_2O_3 concentration increases linearly with the increase in the B_2O_3 content. However, the maximum pulse height decreased linearly with the logarithm

of the B_2O_3 content, from 17.0 arbitrary units at 55 mole percent to 6.5 at 73 mole percent. This lower pulse height is still considerably above the photomultiplier noise at room temperature.

2. Development of Manipulators for Handling Radioactive Materials

a. Electro-Hydraulic Servo Valve - Studies have continued on the development of an electro-hydraulic servo valve for use in a force-reflecting servo for a master-slave manipulator having a load capacity of 100 pounds and a maximum velocity of 36 inches per second. An hydraulically powered system will have less inertia than an electric system, along with other advantages, such as simplified electronics, elimination of reduction gearing, and a very significant reduction in power requirements, which is important for future radio-controlled operation.

The manipulator will be required to handle loads down to small fraction of the maximum load. This imposes the requirement for a signal threshold significantly lower than is tolerable in the usual servo. In addition, it must be capable of smooth and reliable operation with contaminated fluid, in order to facilitate remote servicing of the hydraulic system. Also, the design effort has emphasized the development of valves with pressure control characteristics which eliminate the need for force sensors in a force-reflecting system. This type of valve functions in a manner similar to an electric servo motor.

b. Electronically Controlled Master-Slave Manipulator Mark V - Conceptual design studies are continuing on an electronically controlled master-slave manipulator (30-50 lb. capacity) for use in alpha-gamma research facilities which can be operated for long periods of time without the need for personnel to enter the contaminated areas of the cave to service the manipulators. The design of the Mark V manipulator will emphasize remote repairability, dependability, and economy of fabrication.

Two types of configurations for the manipulator body and arm are being studied. One of these configurations utilizes telescoping tubes for the arm with the motor packages mounted to a support framework which is used to position the manipulator arm by switch controlled motor drives. This particular configuration has the advantage of allowing the master arm of the manipulator to operate close to the shielding window. The second configuration utilizes a hinged arm with the motor packages mounted to the support framework in a manner similar to the telescoping configuration. Although this second configuration can be made compact, difficulties arise in counter-weighting the arm as well as in positional resolution between master and slave arm.

The design of the Mark V manipulator is being closely integrated with the design of a compact overhead mounting system. It is planned to mount two arms on a turret attached to a special bridge and crane. It is desired to keep the bridge width within 40 inches.

A servo drive unit has been designed for the Mark V manipulator and test units are being fabricated. This unit does not include the position transducer, of which several types are being investigated. The drive unit is powered by four 25-frame Diehl motors. The spur gearing has been arranged so that mechanical take-offs on the output shaft can be made on either side of the drive unit. This permits a more compact arrangement of the drive units on the manipulator.

Special attention was given to keeping the inertia of the gear train as low as possible. An improved method was developed for attaching the pinion to the motor shaft. The pinion is designed with a thin-walled split hub over which is pressed a light aluminum collar with a locking tapered bore.

3. 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 equipment has been received and is being assembled and readied for operation.

4. Heat Engineering Studies

a. Boiling Liquid Metal Studies - These studies are efforts in the field of basic heat transfer. The initial loop now under construction utilizes 0.5 in. stainless steel pipe. The first fluid to be used in the test will be sodium boiling at atmospheric pressure. The original design called for a sodium manometer for measuring the pressure differential across the venturi and test section. This idea has been abandoned in favor of a differential pressure transducer.

C. Separations Development

1. Fluidization and Fluoride Volatility Separations Processes

Volatility separations are being considered for reprocessing power reactor fuels. Emphasis is now on the development of the direct fluorination volatility process applicable to the processing of oxide fuels. The fuel for the Dresden reactor (UO_2 clad in zirconium) is a typical example of one of this type.

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

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

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

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

a. Direct Fluorination of Uranium Dioxide Fuel - Preparations have been completed for pellet fluorination runs in a fluid-bed reactor.

b. Hydrochlorination - Hydrofluorination Studies on Uranium-Zirconium Alloys - Additional studies of the direct hydrofluorination of low uranium-zirconium alloy fuels were made in the 6-inch diameter fluid-bed reactor using beds of calcium fluoride as the heat transfer medium. Essentially complete reaction of a ten plate, 1.5 weight percent uranium-zirconium alloy subassembly has been achieved with the two-zone scheme where anhydrous hydrogen chloride diluted with nitrogen contacts the zirconium alloy in the lower portion of the reactor and the volatile zirconium tetrachloride is converted to the solid tetrafluoride in the upper zone by contact with hydrogen fluoride. The two zones are separated by a conical baffle which limits the backmixing of hydrogen fluoride into the lower zone.

c. Processing of Stainless Steel-Clad - Uranium Dioxide Fuel Elements - A 1.5-inch diameter fluid-bed reactor system has been installed to study the processing of stainless steel-clad - uranium dioxide fuel elements. Initial experiments will be directed toward decladding by direct gas-metal reactions using gaseous reagents such as chlorine, hydrogen chloride, fluorine and hydrogen fluoride. A literature search has indicated appreciable corrosion of stainless steel may be expected using these reagents at temperatures near 550°C. Preliminary experiments will attempt to verify these data.

d. Conversion of UF_6 to UO_2 - An experiment at the highest feed rate to date, 142 g/min uranium hexafluoride (equivalent to 247 lb uranium/(hr)(sq ft)), was made in the 3-inch diameter Monel fluid-bed column studying the conversion of uranium hexafluoride to uranium dioxide with simultaneous steam-hydrogen reactions. The run was made at 650°C and lasted 1.9 hours. A batch of uranium dioxide prepared via the 2-step route (pyrohydrolysis with steam at 140°C to form uranyl fluoride followed by reduction with hydrogen at 650°C) is being used to supply industrial companies that wish to evaluate the material for nuclear fuel purposes.

e. Fused Salt Processing of Zirconium Matrix Fuels - A dissolution of a synthetic 10-plate Zircaloy-2 fuel element was made at 700°C in the graphite dissolver. The reaction was one-half as fast as with pure zirconium, being 90 percent dissolved in 10.7 hours and 99 percent dissolved in 13.4 hours. This compares with 90 percent in 4.95 hours and 99 percent in 6.75 hours for the previous dissolution of a pure zirconium element. The Zircaloy-2 alloy constituents of tin, iron, chromium and nickel were added to the fused salt prior to the dissolution at concentrations expected for continuous processing.

f. Plutonium Fluoride Studies - The reaction of plutonium hexafluoride with sulfur tetrafluoride has been studied further. It had previously been established that plutonium tetrafluoride was the solid reaction product. It has not been verified that sulfur hexafluoride is the gaseous product of the reaction.

g. Metal Fluorine Reactions - The mechanism of the nickel-fluorine reaction, in which fluorine migrates through the nickel fluoride film to react at the metal fluoride interface, had been demonstrated in previous work. In order to substantiate the experimental method and the conclusions drawn from it, the same technique employed to study the nickel-fluorine reaction has been applied to the nickel-oxygen system. Kinetic studies, by other workers, had indicated that the oxidation of nickel occurred via the migration of nickel through cationic vacancies in the oxide to the gas-solid interface where the reaction between metal and oxygen took place. This mechanism has been substantiated employing the nickel-63 tracer, autoradiographic technique.

2. Chemical-Metallurgical Process Studies

Liquid metal solvent systems are being investigated with the objective of providing new pyrometallurgical methods for reprocessing nuclear reactor fuels. The studies include determination of various physicochemical data on liquid metal systems and thermochemical data on materials of interest to high-temperature chemistry.

a. Liquid Metal Solvent Studies - Preliminary experiments showed that the solubility of samarium in cadmium is about 6% at 600°C, 2.9% at 500°C, and 0.85% at 400°C. A peritectic at 430°C in the cadmium-samarium system was revealed by metallographic examination and solubility data.

Emf data were obtained on a uranium-thallium cell from 370° to 670°C. These data were used to determine the standard free energy of formation of UTl_3 :

$$\Delta F^\circ = -13,370 + 12.00 T$$

The distribution coefficient of strontium between the two partially miscible liquid metals zinc and bismuth is 8×10^{-4} (wt % strontium in zinc-rich phase/wt % strontium in bismuth-rich phase) at 550°C.

The partition coefficient (wt % uranium in zinc phase/wt % uranium in bismuth phase) of uranium between the partially miscible liquid metals zinc and bismuth changed from a value of 0.08 at 450°C to 0.16 at 550°C.

Additional evidence of a peritectic in the zinc-uranium system at about 830° to 850°C was obtained.

b. Calorimetry - Combustions of tungsten and sulfur in oxygen are being carried out as an auxiliary measurement for the determination of the heat of formation of tungsten disulfide. Preliminary trial combustions in oxygen of the mono- and di-nitrides of uranium and the di-boride of zirconium have been carried out.

A few additional calorimetric combustions of boron and molybdenum are being carried out to supplement the data already obtained. The determinations of the heat of formation of MoF_6 and BF_3 are in their final stages. The determination of the heat of formation of ZrF_4 is considered complete.

D. Advanced Reactor Development

1. Argonne High Flux Reactor (AHFR)

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

a. Conceptual Design

(1) Shielding - Since californium-252 produced in the AHFR creates the most severe handling problems, a special study is being made. The transfer coffin must provide both gamma and neutron protection. Calculations indicate that a mixture of iron-shot and paraffin gives a minimum shield weight. A coffin with an 85 cm thick wall will limit the surface dose rate from 70 milligrams of Cf^{252} to 10 mr/hr. The gross weight of such a coffin would be around 21,000 kg.

(2) Hazards Analysis - A survey of potential hazards in the AHFR system indicates that one of the more serious accidents would be loss of primary coolant through a ruptured beam tube. The resultant pressure rise within the containment building would be approximately 0.1 atmosphere. Methods of using the canal water as a heat sink to reduce the over-pressure from primary coolant release are being studied.

(3) Shutdown Cooling - The decay time, required before convective cooling (with boiling) is adequate, has been estimated to be two hours. For free (non-boiling) convection to take over, the decay time is extended to approximately 60 hours. As the permissible decay time for spent fuel can be an important component of the required down time for refueling, a single channel heat transfer experiment is under way to define the convection cooling limits.

(4) Physics Analysis - In an effort to reduce the peak power density in the AHFR core, a non-uniform fuel loading was investigated. With the 77.72 liter core divided into six regions, each with a different U^{235} loading, it was possible to reduce the ratio of peak power density to maximum flux by 6% compared with a uniform fuel loading. At the same time, however, the total reactor power increased from 100 Mw to 120 Mw in order to maintain the flux level required in the central thimble. The net result is that a slight reduction in local heat transfer demands can be obtained at the expense of fuel uniformity and gross power level.

Additional PDQ problems are in process to see if a sizable reduction in power density is possible with non-uniform fuel loadings and also with different central thimble geometry.

b. Experimental Program

(1) Critical Experiment - The data for rod worth by the ramp input reactivity method yielded a total worth for fine rod #1 of 0.46% reactivity. The data on the coarse rod were inconclusive and are being re-run. Coarse rod #1 has been calibrated by the rising period technique yielding a total worth of 1.97% reactivity. In order to find the differential curve for the whole rod, it was necessary to insert a fractional void in the flux trap to increase the excess reactivity.

A detailed study has been made of the void worth in the flux trap. Measurements have been made with central air voids, with distributed teflon voids, and with distributed aluminum voids all over a range of void percentage. The void coefficient for the flux trap was found to be essentially constant at + .09% per percent void up to a point where the flux trap was about 40% voids. Here it began dropping off, going negative at roughly 70% void. The void coefficient (teflon) for the fuel zone is approximately .0004% per cm^3 .

Detailed flux maps have been taken in the fuel region with U-Al foils. The flux has also been mapped in the fine rod channels both with the rod out and inserted.

The absolute measurement of absolute thermal flux by both Au and U^{235} foils is being repeated because of inconsistencies in the U^{235} data. The ratio of peak sub-cadmium flux in the flux trap to that in the fuel zone is about 8, using an uncorrected cross section for gold of 96 barns.

Other measurements made with the data still to be processed; worth of different materials (lucite, beryllium, Cd) in the beam tube; effect of aluminum liner on flux shape in the flux trap; flux traverses have been made through the beam tube and into the core with indium foils; and measurement of the non-equivalence of teflon and air as a void in large quantity.

(2) Heat Transfer - Heat transfer tests have been conducted on a 0.114 cm (0.045 in.) ID x 11.4 cm (4.50 in.) long vertical circular 304 SS tube in the forced convection and local boiling regions up to the critical heat flux occurrence. These tests were made to establish the validity of the standard Dittus-Boelter equation for the force convection heat transfer coefficient and the Jens and Lottes equation for wall temperature superheat in the local boiling condition for the heat flux ranges and geometrical sizes of the proposed AHFR core design.

Full evaluation of the test data is still incomplete, but the following conclusions have been reached:

(a) The forced convection heat transfer values given by the Dittus-Boelter equation are approximately 15% higher than the test data. (This spread is within the prescribed range of the equation.)

(b) The heat transfer surface temperatures given by the Jens and Lottes wall superheat equation at a system pressure of 13.8 bars (200 psia) are generally somewhat lower than the test data values, the difference increasing with increasing heat flux; the test data falls generally within the spread of the data upon which the Jens-Lottes equation is based.

(c) The net result of (a) and (b) is that the incidence of local boiling is at a slightly lower bulk coolant temperature than that predicted by the use of the Dittus-Boelter and Jens-Lottes equations. The test data indicate a transition from the forced convection to local boiling condition occurring over a range of bulk coolant temperature. There are indications of resulting flow disturbances which are believed to be aggravated, if not caused by, the characteristics of the circulating pump in the test loop.

(d) Based on a preliminary evaluation of a relatively narrow range of the pertinent variables, it is tentatively concluded that the AHFR core thermal design is adequately safe regarding the critical heat flux.

(3) Fuel Element Test Loop - The loop for hydraulic testing of experimental fuel elements is being modified to improve the accuracy of pressure drop and flow measurements. Fibrous material from gaskets in the loop deposited on the element during the first test and changed the effective flow area between plates. Flexitallic gaskets are now being substituted for the asbestos ones. Shock-mounts are being placed under the pump frame to decrease vibration transmission to the pressure gauges. A venturi-tube has been installed to check the calibration of the orifice-plate flow meter.

2. Basic Radiation Effects Reactor

The BRER is a small (1 Mwt) fast research reactor designed to meet the particular needs of solid state science experiments. These experiments require flux levels from 10^{11} to 2×10^{13} n/(cm²)(sec), flux energies as uniform as possible from fission spectrum to 10-50 kev, and experimental operating temperatures from 4° to about 600°K. In the addition the flux should be quite constant both in space and time. All experimental facilities are in the external blanket. The most probable blanket material is lead.

Most proposed experiments for the reactor require carefully regulated cooling. As a result, it is important to minimize gamma heating. The results of a five-region, thirteen group physics calculation were used to calculate gamma production rates throughout the system resulting from inelastic scattering and radiative capture. These rates, in addition to data on prompt fission gammas and fission product decay gammas, were used to calculate heating rates in the reactor vessel, inner reflector (control rod region), and outer reflector. The results are given in Table III.

Table III. Heating Rates in the BRER

<u>Region</u>	<u>Heating Rate (Kw)</u>
Reactor Vessel	4.8
Inner Reflector	57.4
Outer Reflector	<u>34.6</u>
Total Heat Outside Core	96.8

In addition, volumetric specific heating rates for various materials at different points in the outer reflector were calculated. The results are given in Table IV.

Table IV. Specific Heating Rates for Various Materials in the BRER

<u>Material</u>	<u>Radius (cm)</u>	<u>Energy Absorbed (cal/sec gm)</u>
Pb	52.5	3.5×10^{-4}
Fe	52.5	2.1×10^{-4}
Al	52.5	1.8×10^{-4}
Pb	81.0	$.88 \times 10^{-4}$
Fe	81.0	$.53 \times 10^{-4}$
Al	81.0	$.43 \times 10^{-4}$

A first stage draft report on this concept is in process. The basic development work is complete.

3. Coupled Fast-Thermal Steam Superheater

This is a coupled fast-thermal reactor concept with the thermal zone water cooled and the fast zone steam cooled. Of the total reactor power of 216 Mwt, 163 Mwt comes from the fast zone. The heat absorbed in the thermal zone coolant is used as a portion of the preheat for the turbine condensate. Part of the superheated steam from the fast zone is desuperheated with the preheated turbine condensate. The desuperheated steam is then compressed by the amount representing the pressure drop across the reactor and then is fed back as coolant for the fast zone of the reactor. The steam from the reactor is at 95 atmospheres and about 450°C. The breeding ratio with plutonium fuel is 1.4 which is comparable to similar systems using sodium as the coolant.

This type reactor represents a relatively complex design owing to the necessity to keep the two relatively high pressure coolant streams separated. This separation is necessary to prevent flooding the fast zone. The relatively poor properties of steam as a coolant places stringent requirements on the fuel materials. The core must operate at the highest possible temperature in order to achieve a profitably high power density in the core. This particular reactor concept requires more evaluation to establish its potential.

A first draft report was completed and is being reviewed.

4. Packed Bed Fuel Elements

The study of packed bed fuels results mainly from the requirements for reactor performance of the highest power density. This may be for applications such as a research reactor requiring the highest possible flux or a power application requiring a maximum power to weight ratio. In addition such fuel might be used where power density is of minor importance but where the coolant temperature must make the closest approach to the fuel temperature limitations. The more probable fuels for use in the packed bed elements are micron sized, unclad ceramics.

A survey and concept analysis report is in the publication stage. This report is ANL-6193, "Study of the Packed Bed Fuel Element."

Additional literature and survey work on heat transfer and pressure drop has continued. The data and correlations of pertinent studies were critically examined and a list of equations for use in packed bed fuel element design compiled. Currently available information appears adequate for the analysis of such designs, and out-of-pile tests do not seem to be required. An analysis of such a fuel element using particles of about 200 microns diameter, having packing density approaching the maximum

and operating at a power density of about 10 Mw/liter, would have a temperature drop of about 30°C across the particles with a coolant (helium) flow rate of less than 30 meters per sec.

In-pile experiments are warranted for this fuel concept to study both the heat transfer properties with high internal heat generation and the effect of fissioning on the particle attraction, or agglomeration, and bed-packing characteristics. Preliminary design calculations have started for a CP-5 in-pile loop. Plans call for a self-containing, compact loop using steam as the packed bed coolant.

5. Lunar Power Plant

This is a study to produce a design of a fully automatic packaged power plant for use under conditions such as those found on the moon. To fulfill this purpose, the following objectives were established:

- (a) An electrical output of 1 Mw per unit
- (b) Two year life with unattended automatic operation
- (c) Minimum specific weight
- (d) Rankine cycle with radiative heat rejection
- (e) Vacuum environment
- (f) Withstand departure and arrival launch conditions
(accelerations, vibrations and shock)
- (g) Minimum radiological hazard at lunar site .

The reactor system proposed is a direct cycle, boiling mercury-cooled fast reactor. Large radiative coolers are used to condense the mercury vapor. The physics and thermal analyses are complete, as is most of the design. Since the radiative cooling system is of necessity large, all possible means of reducing the weight of the radiators are being examined.

The vapor-liquid separator for the power plant is an eight segment radial flow design. A full size model of one of the eight segments has been constructed and will be tested using air-water mixtures.

PUBLICATIONSPapers

A GAMMA-RAY INDUCED ABSORPTION BANK IN SOME LEAD BORATE GLASSES

Adli M. Bishay

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COLORIMETRIC DETERMINATION OF PALLADIUM IN URANIUM-FISSION ELEMENT ALLOYS

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