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QUARTERLY REPORT
ON REACTOR DEVELOPMENT

July 31, 1954

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ARGONNE NATIONAL LABORATORY
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QUARTERLY PROGRESS REPORT ON
REACTOR DEVELOPMENT

4000 PROGRAM

July 31, 1954

Compiled by members of the Laboratory Staff
and Stuart McLain, Coordinator

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ARGONNE NATIONAL LABORATORY
QUARTERLY REPORT ON REACTOR DEVELOPMENT
July 31, 1954

I. POWER REACTOR PROGRAM, 4105

A. Boiling Experimental Reactor, BER

According to present plans, the Boiling Experimental Reactor facility will consist of a reactor and power plant building and a control and service area. The reactor building will consist of a gas-tight steel shell having a gross volume of about 400,000 ft³, see Figure 1. The reactor with its control drives, foundation and shielding, and the power plant, including the turbine and condensers, will be built in the steel shell. The control room, offices, and service areas required will be located in an attached conventional structure. The reactor building will be designed to maintain tightness against an internal pressure of 15 psig. It is expected that the architect engineer will work out many details of the process as well as the design of the equipment and buildings.

1. Reactor Core

The present core design is 4 ft diam by 4 ft high, containing 116 basic fuel assemblies in a lattice with an H₂O/U volume ratio of about 2.5/1. Each fuel assembly is 3-7/8 in. x 3-3/4 in. in cross section and occupies a 4 in. x 4 in. cell. One type of assembly contains six natural U plates. The other basic type contains seven enriched U plates. These natural and enriched U assemblies are completely interchangeable in the lattice to permit wide flexibility in the loading. For example, studies can be made of the effect of deliberate variations in the power distribution in the core on the operating characteristics. Changes in reactivity with time can also be compensated for by moving the enriched fuel relative to the natural U.

The meat of the fully enriched U plates is an alloy of Zr with 5% by weight of enriched U. Since this meat itself is highly corrosion resistant, only 10 mils of cladding are specified. Each natural U plate is 0.25 in. thick, including the 0.020 in. Zircaloy-II cladding. Development is proceeding on a number of alloys as discussed later under fuel element development.

Twelve, blade type, shim safety rods move vertically through the core. The absorbing portion of each rod is about 7 in. wide and 42 in. long. The drive mechanisms for these rods are located in a room underneath the reactor. Suitable grid structures above and below the core hold the fuel assemblies and safety rods in a fixed geometry

FIG. 1
CUTAWAY PERSPECTIVE OF
BER POWER PLANT

The pattern of enriched fuel assemblies to be used in the initial loading of the BER core has not been settled. Calculations show that if enriched elements are used only in a central zone 3 ft diam and 2 ft 8 in. high, about 8 kg of U^{235} are required for criticality. The over-all maximum to average power density is about 4.0 and about 30% of the total power is generated in the enriched elements. If enriched assemblies are located in a uniform pattern over the entire core, the amount of U^{235} for criticality approximately doubles, but the maximum to average power density decreases to about 2.7. The percentage of power in the enriched elements increases about 2%.

For the present cylindrical design of 4 ft high by 4 ft diam core with uniformly distributed enriched fuel assemblies, the fraction of enriched fuel elements is 0.36 with an initial operating critical mass of 14.3 kg of enriched U. Thirty-four per cent of the total power is produced in the enriched fuel and the maximum to average radial flux is 1.88. Compared with the 3 ft diam core, the reactivity difference from room to operating temperature increases while the sensitivity to voids increases only slightly.

2. Pressure Vessel and Shield

The reactor pressure vessel will operate at an internal pressure of 600 psig. It is now conceived of as a carbon steel tank lined with stainless steel with an ID of 6 ft. Calculations have shown that it will be necessary to use a borated stainless steel liner 1/2 in. thick just inside the pressure vessel to reduce the thermal stresses in the pressure vessel by absorption of the thermal neutrons and a fraction of the gamma rays. Alternative steels are being considered in an effort to find a design that will not require a liner.

Loading and unloading will be carried out through eccentric shielding plugs in the top of the pressure vessel. Irradiated fuel is stored in a water-filled concrete well adjacent to the reactor.

The biological shield for the reactor will be ordinary concrete about 8 ft thick. This will be augmented by dense concrete or other material of increased shielding effectiveness in certain areas. Cooling coils will be provided in the biological shield to remove the heat generated due to absorption of nuclear radiation and the heat transferred by normal thermal processes from the insulated pressure vessel.

3. Reactor Auxiliaries

A boric acid solution is stored under pressure for injection into the reactor through a nozzle spray system if safety rods and other shutdown mechanisms fail to function.

Water from the demineralized water storage tank flows by gravity into the pressure vessel through the nozzle spray system to cool the fuel assemblies in case the water is lost from the reactor core. Boric acid can be added to this water if there is any concern over making the reactor critical by introducing water.

To increase the efficiency of the steam formation process and, hence, the efficiency of the automatic shutdown feature, an interlock system prevents withdrawing the safety rods unless the water in the reactor is about 100°C. The heating of this water is accomplished in a bypass loop containing a pump and heat exchanger. Plant steam is used on the shell side of the exchanger for heating the reactor water. This heat exchanger is also used for cooling the reactor water after shutdown. In this case, water from the cooling tower circuit is circulated through the shell side of heat exchanger. In case of electric power failure, natural convection circulation of the reactor water and cooling tower water will be sufficient to remove the heat generated by fission products during shutdown.

About 10 gpm of reactor water will be circulated through an ion exchanger cleanup loop. This system should keep the impurities in the reactor water below 1 ppm, thereby minimizing the deposition of residue on boiling heat transfer surfaces and the carryover of radioactive dissolved and suspended material with the steam to the turbine.

4. Power Plant

In many respects, the major components of the steam system are identical with a conventional plant. Noteworthy deviations from standard practice are:

a. Since the steam is radioactive, provisions must be made for shielding some components. Leakage must also be minimized and the materials which do leak out must be collected and monitored before disposition to the outside. Corrosion rates and radioactivation cross sections are taken into consideration in the choice of materials for use in contact with the steam or condensate in the external circuit.

b. The reactivity of the reactor itself is a function of the volume of steam bubbles in the core and, therefore, a function of the reactor pressure for any given power level. Steam will normally be generated in the reactor at 600 psig pressure. However, the design will be such that experimental operation at lower pressures will be possible. The steam will pass with a slight pressure drop through a steam accumulator which will be constructed to serve as a moisture separator as well as a pressure transient damping device. Such an accumulator is required since reactivity changes produced in the reactor, due to pressure changes, are opposite to those desired. Under very rapid change conditions, "chugging" of the reactor might occur.

The amount of the steam which passes through the turbine is determined by the governor-regulated valve, whose function is to maintain constant turbine speed. When the electric load decreases, the governor valve reduces the quantity of steam entering the turbine to match the power demand. This rejection of steam would cause a pressure buildup in the reactor. Bypass valves will send all excess steam up to full steam load to the condenser, keeping the reactor pressure constant.

c. An effort is being made to obtain a turbine which will be leakproof. It appears that such a turbine may be obtained without excessive cost. The 5000 kw electrical power generated at 20,000 kw heat output will be fed into the existing utility network at Argonne.

d. Heat from the condenser is to be rejected to a cooling tower. The elevations will be such that natural circulation will occur if the pumps fail. Air is withdrawn continuously from the condenser by means of a vacuum pump. After having the moisture extracted, this air goes to holdup tanks. Mechanically coupled, cycling valves on these holdup tanks guarantee that the tank which is receiving air from the pump at a particular moment is isolated from the building stack. A ten-minute holdup allows time for stopping the cycling operation and sealing the system if abnormal gaseous activity is detected.

The condensate is pumped through full flow filters to remove suspended corrosion and erosion products. The flow rate of the feed water to the reactor is governed by the steam flow in the main steam line plus a water level measuring instrument in the reactor. The mass feed water flow rate is normally exactly equal to the mass steam flow rate. If, due to imperfect equilibration, the water level in the reactor should rise or fall outside predetermined limits, the water level indicator will override the steam flow to demand more or less feed water to correct the water level in the reactor.

5. Entrainment

The entrainment of liquid in steam as affected by the geometry and operation of a boiling vessel is being studied. The evaluation of external entrainment separators is planned.

Preliminary experiments were made by boiling water in a single vertical tube with varying conditions of water feed rate (9 and 18 lbs per hr), power input (0.5 to 1.5 kw), and pressures (0, 20 and 40 psig). A tentative evaluation of the results indicates correlation between entrainment and the volumetric flow product of steam and liquid. It is planned to extend the range of experimental conditions.

6. Reactivity Changes

For the cold reactor, the void reactivity change is estimated to be 0.06% k for the first 12% of voids and 0.48% k for the next 12% of voids.

The reactivity change from room to operating temperature of 253°C, excluding Xe and Sm, has been calculated as -1.75%. However, this reactivity change may vary during the life of the reactor.

One of the contributing factors to the thermal reactor temperature coefficient is that due to the different temperature dependencies of fission and non-fission cross sections. The temperature coefficient for the BER has been studied to estimate its value initially and after equilibrium Pu^{239} has been attained.

While the data are somewhat inconsistent, it was indicated that the neutron "temperature" effect may be positive under equilibrium Pu conditions, being opposed by density changes. More precise measurements of the capture to fission cross sections of Pu^{239} , particularly through the 0.3 ev resonance, will probably be necessary before the temperature coefficient can be calculated with confidence.

B. Borax Experiments

The Borax-I reactor was modified and reconditioned by installation of improved control rod drive mechanisms, operating circuitry, and water handling system, including the addition of a high capacity deionizer for the feed water. Three groups of tests were then run as discussed below.

1. Transient Experiments

A series of transient experiments were run in which excess reactivity up to 1.3% k_{eff} was introduced suddenly, the reactor being allowed to rise exponentially until the fraction of steam in the reactor reduced the reactivity below the point of criticality. Data relating the energy released versus the amount of subcooling from room temperature to the boiling point were obtained. This extended the data obtained during the summer of 1953.

2. Fuze Tests

Three types of reactor fuzes were tested as furnished by ANL, ANP, and KAPL. The ANL fuze consisted of a link of enriched U with a soldered connection as illustrated in Figure 2. A power surge in the reactor results in rapid heating of the U with melting of the solder and the insertion of the Cd sleeve into the reactor by the spring. Each of the three fuzes tested shut down the reactor, whose power was rising on a 20 millisecond period, before it was shut down by steam formation. The second type

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FIG. 2
SKETCH OF ANL FUSE PACKAGE 1954

of fuze, supplied by GE ANP, consisted of a plug carrying U soldered over an opening in a gas pressure chamber. While reactor shutdown was not successfully demonstrated with these fuzes, it was shown in about a dozen cases that the gas was released by unsoldering the plug. The third type of fuze furnished by KAPL consisted of an explosive with U mixed with it. The explosive has the special characteristic of being resistant to detonation to about 200°C. For all tests, the explosive fired at a temperature estimated to be about the one specified and also in a time shorter than the time required for steam to shut down the reactor in a 20 millisecond period.

3. Rapid Excursion

After completion of the fuze tests a new, stronger control rod was installed in the reactor for use in producing a power excursion of very short period. Calibration of the rod established its worth as 4.1% k_{eff} . Further instrument calibrations were made and the reactor prepared for the short period power excursion.

It was expected that the short period excursion would melt some of the fuel plates of the reactor, and that the accompanying steam pressure would distort all the remaining plates beyond the point of usefulness and would perhaps break the reactor tank. It had been established, however, that inasmuch as the reactor was to be replaced immediately by the Borax-II reactor, the information to be gained from the excursion would make the experiment worth more than the cost. All precautions were taken against possible hazard. An adequate staff of AEC Health Physics personnel was present, all personnel were stationed at least half a mile from the reactor, and a favorable wind direction was chosen. The proper wind prevailed on the morning of July 22, and the excursion was made at 8:20 a.m. on that date.

The power excursion was made by ejecting the 4.1% control rod rapidly from the reactor core under the force of accelerating springs plus gravity. Preliminary analysis of the data shows that the power was increasing on an exponential period of about 2.6 milliseconds just before termination of the power rise. The effects of the power excursion were rather violent. Water and parts of the reactor were thrown as much as 100 ft into the air; the reactor tank was ruptured; large fragments tore holes in the surrounding shield tank; and most of the fuel plates disappeared from the reactor.

The instrumental data are being analyzed. The principal problem is to determine, if possible, the precise value of the nuclear energy generated. The following ways of estimating this are being evaluated:

a. Four ionization chambers gave power traces showing a maximum. The difficulty in interpreting these is connected with the fact that in calibration all of the moderator water was present. During an excursion an unknown amount of water is replaced by steam, thus increasing the neutron flux on the chambers.

b. Gold foils were installed at a calibrated position outside the reactor. These have been recovered and counted. The same criticism which applies to the ionization chambers applies to the foils. The best foil measurement at hand now indicates 135 megawatt seconds.

c. Cobalt foils were installed inside the core. These foils are not subject to as large corrections as the outside exterior foils or the ionization chambers. These cobalt foils have been found but have not been evaluated carefully. An initial measurement shows 120 megawatt seconds.

d. A fuel plate, which was furnished with a thermocouple, at the edge of the reactor might give a reasonable estimate of nuclear energy. The thermocouple record is complete but should be corrected for the time response of the thermocouple and its detector; uncorrected it gives 167 megawatt seconds. The nature of the correction to be applied has not yet been worked out. This fuel plate which was installed for the excursion is also being analyzed for total fissions.

Accepting temporarily the 135 megawatt second figure, it is clear that much of the reactor core melted. If all of the energy remained in the fuel plates and if all the plates melted, the temperature would have risen to 1300°C. Since the maximum to average flux ratio was 1.8, it is not likely that much of the core, if any, vaporized. It is suggested that any conclusions concerning chemical reaction be deferred until further results on nuclear energy have been obtained.

The pressure transducer which was installed in the reactor and which had a full scale range of 1,000 psi apparently went off scale and, in any case, when found was thoroughly mechanically injured. The general area of the reactor has decayed to a tolerable level. One area, 60 yards square and 150 feet from the reactor, shows sizable activity, most of which is associated with droplets of aluminum. In the immediate reactor area fragments of fuel elements have been found; it is clear that the remainder of the fuel charge is to be divided between what lies in the hot area and what went up in the cloud. An attempt is being made to estimate how much for each of these.

All moving picture cameras yielded pictures of the excursion despite the fact that the shock wave cut off electricity to them. A bright flash of light is seen and 8 milliseconds after the flash, the first debris starts out of the reactor. The vertical velocity is about 500 ft/sec.

The initial spurt carried the cloud up a distance of approximately 120 feet. Fragments of burning aluminum can be seen going up and then falling down in the smoke.

4. Borax II

Borax II will be constructed on a site near Borax I. A concrete shield will be provided and a weather shelter will be placed over it to permit the work to continue except in periods of the most severe temperature conditions.

C. BER Fuel Alloys

1. Alloy Development

The development of U alloys for fuel element use in the BER has been directed toward two main objectives: (1) to develop by a direct method of approach a heat treatment for the alloys of interest, U-Zr and U-Zr-Nb, which will provide a degree of irradiation stability at least equal to that of the "castings" of these alloys; and (2) to provide back-up information in the form of constitution, metallography, heat treatability, and other physical properties for the alloy systems in general.

a. Heat Treatment and Irradiation Stability

The heat treatment characteristics of 2 and 3 w/o Zr and 5 w/o Zr-1 1/2 w/o Nb alloys have been studied in detail. In all cases the heat treatments involved transformation of the gamma phase under various conditions of cooling, ranging from slow furnace cooling to rapid water quenching. From microscopic observations and hardness data it was concluded that the "chill castings" that had been irradiated to date did not correspond to structures resulting from rapid quenching from the gamma, as had been supposed originally; rather, they corresponded more closely to slowly cooled alloys. By controlled manipulation of the cooling rate, it was possible to develop microstructures that closely approximated the "as cast" condition, and representative samples of these, along with those from several other heat treatments, have been submitted for a 2-week irradiation in the MTR (approximate burnup, 0.05%). It is hoped that these results, which are due back the latter part of August, will provide a clue as to relationship between heat treatment and irradiation stability.

b. Constitution and Properties

The physical and mechanical properties of 2 and 3 w/o Zr alloys have been shown to be subject to considerable variation by heat treatment; hardness values, for example, may range from 250 to 500 VHN, etc. Several features of the presently available phase diagrams have been

found to be in error; for example, the solubility of Zr in α -U has been found to be very low, of the order of 0.2 w/o at 640°C, as contrasted with an estimated value of about 2 w/o as shown in published diagrams. Also, preliminary X-ray diffraction work on several alloys in the 30-50 w/o range indicate a high temperature (550-600°C), two-phase region consisting of the two terminal solid solutions, which apparently transform peritectoidally to form the epsilon phase at about 53 w/o Zr. The structure of the epsilon phase is under study.

In the U-Zr-Nb system, it has been found that the "martensitic" structure obtained on rapid quenching is a transformation product of a reaction occurring between 370 and 450°C, the exact temperature varying with the cooling rate. Above about 450°C, the reaction is eutectoidal, following the usual pattern of a typical time-temperature relationship of a "C" curve. Studies on these kinetics are continuing.

2. Clad Plates

The manufacture of the Zircaloy-II clad enriched U-Zircaloy-II alloy plates will be carried out by regular techniques of hot rolling the alloy in a picture frame of the clad material.

Development of techniques for manufacture of the Zircaloy-II clad, natural U plates is proceeding along several lines. First, rolling of U cast in Zircaloy molds is being attempted. Billets of circular cross section are being made by pouring the 5% Zr, 1-1/2% Nb, U alloy into Zircaloy-II molds; billets of rectangular cross section are being made in the same way; and billets are being made by casting between Zircaloy-II side plates.

In the casting into circular molds, small inside mold diameters up to 0.37 in. were used. With properly heated molds good results were obtained. Castings made in rectangular Zircaloy-II molds 0.84 x 3-1/2 x 13-3/4 in. have given unsatisfactory bonds possibly because the proper conditions of pouring have not been found. In the third type of effort, pouring between Zircaloy-II side plates in a graphite crucible has been more successful. In this procedure, the bond between the face plates, core, and side plates is obtained by subsequent rolling. This procedure eliminates the necessity for close thickness tolerances for control of camber in rolling, and tensile stresses between core and side plates due to differential contraction of core and face plates, and the corner mold stresses.

Problems which must be solved in any casting process such as the above three processes include: (1) mold support; (2) proper mold heating; (3) prevention of bond fracture due to differential contraction of core and clad; (4) elimination of porosity; and (5) elimination of pipe. Work is progressing on all these problems.

In the few cases in which satisfactory bonded ingots have been obtained, subsequent rolling has been satisfactorily producing good fuel elements, except for the clad thickness variation on the leading edge of the fuel element. At the leading edge the rolling produces thicker clad at the expense of the core. Otherwise, plates with good edge bonding and uniform cladding have been produced. The cast bonds remain intact during rolling.

The end closure problem has not been solved. Cropping, etching out the core, and welding a wire in the crack have produced good closures.

3. Corrosion of U Alloys

a. Non-Irradiated

Argonne has prepared and tested a number of corrosion resistant U alloys in an attempt to obtain a fuel element that will not fail catastrophically in case of a cladding defect. It is desired that the fuel elements may be left in the reactor at full power for several days after cladding failure. Alloys of Nb with Zr and Sn additions have been given most attention, and the Mo alloys have received most attention at WAPD. The Argonne alloys of interest are 3 w/o Nb, 6 w/o Nb, 5 w/o Zr - 1-1/2 w/o Nb, and 3 w/o Nb - 0.75 w/o Sn, the remainder being U in all cases.

All of these alloys are prepared by quenching from the gamma phase, from 800°C or above. The gamma phase is not retained, but rather there is a Martensitic type of transformation. The alloys will age at high temperatures, around 375°C, with precipitation of normal crystalline alpha U.

Tests of aged alloys resulted in rapid disintegration in water at 350°C. Apparently hydrogen increases the rate of disintegration, and Zr cladding decreases the hydrogen effect since the Zircaloy-II is cathodic to the U alloy.

3.8 w/o or 25 atom % Si with U forms a stable epsilon phase alloy which should be stable to aging.

Results of static corrosion tests in autoclaves at 260°C indicate that the alloys (heat treated by quenching from above 800°C) are still intact after the times indicated:

6 % Nb	6600 hours
3 % Nb	4300
5 % Zr, 1.5 % Nb	4900
3.8 % Si	4400 (epsilonized at 650° for 7 days)

In tests at 288°C a 6 % Nb alloy (aged 1 month at 316°C) disintegrated in 166 hours; one 5 % Zr, 1-1/2 % Nb sample (not aged) stood up for over 3,000 hours and a second aged sample is still under test and in good shape after 300 hours.

Samples of the same alloys were placed, unbonded, in small stainless steel capsules containing small holes and corrosion tested. At 260°C all samples, except the 6 % Nb, corroded more rapidly than in the unconfined conditions. The 6 % Nb alloy sample showed no change in over 4,000 hours. At 316 and 360°C, 6 % Nb samples failed after 1068 and 375 hours respectively.

b. Irradiated

Small test rods of 3 % Nb-U alloy which had been irradiated in a NaK capsule for 0.12 to 0.15 % burnout in the MTR disintegrated in less than 1 hour at 260°C when corrosion tested statically in an autoclave. These tests were of a preliminary nature and are being repeated.

c. Corrosion of Zircaloy-II Clad Fuel Plates

In long-term corrosion tests, binary alloy plates, 3 and 6 % Nb, apparently without defects have accumulated as much as 2000 hours - in some cases without sign of failure. Tests on the plates have been run, first at 260°C statically and then at 260°C dynamically or 288°C statically. Other binary alloy plates with originally suspected or known defects have gone up to 1400 hours at 260°C since exposure of the core without catastrophic failure, but with blistering and an occasional crack.

In deliberate (drilled hole) defect corrosion tests, binary alloy plates with defects in soundly bonded areas have stood up thus far for 500 hours at 260°C without change in appearance; defects into unbonded areas result in a tendency to blister after one-week tests. Ternary alloy plates have shown some serious rupturing after one-week tests; preliminary results of a follow-up program indicate the possibility of stressing or even cracking of the core during fabrication, resulting in decreased corrosion resistance.

Thermal cycling to 260°C has no adverse effect on the corrosion resistance of sound plates. One plate was cycled 50 times to 640°C; visual inspection indicated no signs of cracking or failure after thermal cycling, but the plate virtually disintegrated in a subsequent one week, 260°C corrosion test.

Two each Zircaloy-II clad fuel plates with 3% Nb, 6% Nb, and 5% Zr, 1-1/2 Nb, cores are under test in the MTR loop at 260°C. Comparative samples are being run in the same loop but located external to the reactor.

d. High Temperature Vapor Phase Corrosion Tests on Zr Alloys

Samples of Zircaloy-II were exposed to high temperature 850 and 1000°C atmospheres of super-heated steam, dry air, and moist air to simulate loss of coolant from the fuel elements of a water reactor. Zr-clad plates of U were tested in super-heated steam at 800 and 850°C. The Zircaloy-II samples showed undesirable swelling and warping and the Zr cladding ruptured on the clad plates in about four hours of testing. There was, however, no undue reaction even with the exposed U cores. The moist air appeared more detrimental than either dry air or super-saturated steam.

4. Eddy Current Thickness Gage

In the pulsed eddy current method of determining the thickness of cladding, an investigation is in progress to make the readings of the probe much less dependent upon the distance between the probe and the surface of the metal. Some success has been achieved in that certain points in the output pulse wave have been found whose amplitudes are nearly independent of the distance of the probe from the metal, but do vary with the thickness of the clad metal.

Another investigation concerned with this same problem of the distance between the probe and the metal has been started. This investigation deals with the use of sinusoidal waves of one frequency and has as the indicator a phase measuring device.

An ultrasonic drill for cutting out small inclusions in U has been built and tested.

D. Reactor for Remote Locations

The design of small reactors for power and heating of remote installations has continued with emphasis on a boiling type of an H₂O moderated reactor. The reactor, as illustrated in Figures 3 and 4, is designed to operate at 5.5 Mw with a steam pressure of 600 psig.

The reactor is located in a horizontal tank and controlled by horizontal shim-safety rods. The core of the reactor is comprised of vertical fuel assemblies of the MTR type, but similar in dimensions to the Borax-II fuel elements. Zr-U²³⁵ alloy, Zircaloy-II clad fuel plates similar to the STR fuel plates will be used.

FIG. 3
HORIZONTAL BOILING REACTOR

FIG. 4
SMALL BOILING REACTOR
POWER PLANT

The reactor tank is divided into two compartments by means of a wall or weir. This weir controls the maximum water level in the reactor compartment, and the minimum water level is fixed by the height of the water inlet since water may flow backwards through this pipe in case of stoppage of the feed water pumps. The feed water is fed through a water injector and may inject up to 3 parts of water from the steam separation section for each part of feed water.

The steam in leaving the reactor will carry water with it and this water will be returned by the feed water injector. In case of a slight power rise, more steam will form, forcing more water over the weir; in case of a low power surge, less steam will be present in the core and less water would flow over the weir. Thus, the reactor will be held at constant power. No fine or running control rods will be provided.

This type of reactor control system has not been demonstrated, but it is a logical extension of Borax-I design.

The water circulation in the reactor will be by natural convection and about 60 times the feed water flow.

A total of 8 to 9 kg of U will be required, including the fuel for 1 yr operation at 75% of full power. The core will consist of 72 fuel elements.

The power and heating equipment layouts are shown in Figures 4 and 5. As designed at present, the condenser will operate at 20 psia, producing hot water at 215°F for circulation through the heating system. Since it is understood that steam at 35 psig is required for heating remote Army installations, the design will be revised to produce 35 psig steam by heat exchange from turbine bleed, with the remainder of the steam going to a vacuum condenser.

The normal operating density of the moderator is 14% lower than the non-boiling saturated water surrounding the core. This corresponds to approximately -2% k.

Shielding of the reactor and turbine room is indicated in Figures 4 and 5. The reactor tank is protected by several inches of a dense material which serves both as a thermal shield and insulation layer. Powdered limonite is a possible material. The biological shield will be water, concrete, and earth.

The entire plant is designed to fit into a Quonset Hut, 36 ft x 60 ft, and to require no excavations. Exchange of fuel is required yearly. To reload, the reactor is opened, the shielding plug containing the eccentric fuel loading hole is raised and supported on a ball bearing race (not shown in the sketches), and the fuel elements removed one at a time to the coffin for removal to storage.

FIG. 5
BUILDING & EQUIPMENT
 (PLAN VIEW)

II. FAST BREEDER REACTOR PROGRAM, 4110

A. EBR-II

Recently, the major part of the engineering design effort has been applied to the quarter to half scale working model of EBR-II. The general arrangement of the model, including the primary Na tank and the primary system components is shown in Figure 6. It is hoped that this represents a satisfactory arrangement of equipment for EBR-II. In Figure 6, the model is shown in the fuel unloading position. The pressure head has been rotated to the left to clear the reactor top for unloading and is shown in front of the U-shaped DC electromagnetic pump. A model fuel element is shown partially removed from the core prior to delivery to the rotary transfer arm. This arm, which operates as a hinge, is shown in front of the primary heat exchanger and delivers the fuel element to the storage rack shown to the right of the reactor at the bottom of the tank. The detailed designs of each of the major components are now being completed and are being released for construction. Both dry and Na circulation and handling experiments will be conducted.

The pipe connection between the pump and the reactor, and between the reactor and the heat exchanger, are to employ slip-fit type joints with a spring loaded seating arrangement. An experimental test apparatus is being constructed to evaluate different seat configurations and material combinations. Leakage rates will be determined as a function of Na temperature, Na pressure, seat spring pressure, and misalignment.

Drawings have been completed and manufacture is about 30% complete on a homopolar generator and a 10,000 gpm electromagnetic pump. Delays are being encountered due to procurement difficulties. The motor and pump will be tested as full sized components of EBR-II and a full scale breeder reactor.

Current work is being carried out on the design of the EBR-II mockup building. Details of the mockup will not be completed until after tests have been run on the scale mockup.

B. Physics Calculations

The multigroup diffusion theory equations in spherical geometry have been programmed and coded for solution on the UNIVAC at the AEC computing facility at New York University. The details of the code are similar to that of the cylindrical geometry setup discussed in the previous report. A large number of additional problems have now been solved on the UNIVAC using these two setups. These problems pertain to details of the power distribution with central blanket, analysis of a series of Los Alamos fast criticals, calculations of neutron generation time for various reflected experiments in EBR-I or in the fast critical facility.

FIG. 6
CUTAWAY DRAWING OF EBR-II

Among the calculations being performed, which are of a different nature, is included a fundamental mode reactor transfer function analysis for several neutron lifetimes and a range of conditions on the absolute and relative delayed neutron abundances. From the analysis thus far, there appears to be some possibility of determining total delayed neutron abundance from a measurement of amplitude response, as well as getting a check on relative abundances assigned from other experiments.

C. Exponential Program

The neutron distributions within enriched assemblies containing U^{235} , U^{238} , Al and Fe are being measured with the fast exponential. In each case the Al is present as 14.5% by volume. Measurements on the assembly containing 3 atoms of U^{238} to 1 atom of U^{235} have been completed and the bucklings are being calculated. Measurements are being made on the assembly containing 7 U^{238} atoms per U^{235} atom.

The most enriched, or 3/1, assembly was limited in size to 21 x 12 x 24 in. The 21 x 12 in. side was placed against the pedestal face and the 12 x 24 in. side rested upon the reflector surface. This reflector is 12 in. thick and is composed in turn of 100 % natural U, Pb, and Fe. The reflector savings for these materials are thus obtained. Flux measurements are made with miniature fission chambers which are moved through the assembly.

The three-dimensional flux distribution is studied with U^{235} , U^{238} , Np, and Pu detectors. Relative values of $U^{235} : U^{238}$ and $U^{235} : Np$ fission ratios are obtained. Within the 12 in. long region of spectrum equilibrium, the vertical and transverse neutron distributions fit the theoretical cosine curve very well. The experimental points deviate less than 1% from the cosine curve and this deviation includes counting statistics, numerical loss in fitting, etc.

The reflector savings are interpreted in terms of extrapolation lengths of the respective distributions into the reflectors. Careful fitting to the cosine function allows a good determination of these values. At present, an exact evaluation of the experimental results in terms of bucklings and reflector savings is dependent on the least squares fitting of the experimental points. The system used formerly has been shown to contain some basic errors and is being replaced by a revised program. This program is being coded for use of the AVIDAC. This approach will eliminate the previous numerical errors in the fitting and will provide a better interpretation of the accuracy of the measured quantities. While this procedure has delayed analysis of the data, it will shorten the total time consumed.

1. Fission Ratios

A procedure for determining the $U^{235} : U^{238}$ fission ratio in the Snell Experiment and in the various enriched exponentials has been worked out to furnish some knowledge of the neutron energy distribution. Two miniature fission chambers, one 96% enriched and the second natural U, are calibrated in a given thermal flux. Immediately after calibration, they are both irradiated in the spectrum to be measured. The desired ratio is then simply obtained. The Snell Experiment appears to yield a value of $360 \pm 8\%$ for this ratio, and the 3:1 enriched exponential a value of $20 \pm 3\%$. This latter figure is in good agreement with the theoretically predicted result. The value for the Snell Experiment, however, is some 30% higher than theoretically expected. A previous experiment (CF 1627) has set this value at 330, which is not in disagreement with the above. It is hoped that it will be possible in the future to obtain ratios in each assembly between all of the fissionable detectors available, i.e., U^{235} , U^{236} , U^{238} , Pu and Np.

2. Foil Measurements

Determination of U^{238} capture in the Snell block by observation of the 100 kev activity in U^{238} foils was tried, but examination of the spectrum before and after irradiation showed a barely perceptible difference in the region of 100 kev. The σ_{nvt} necessary for observable counting rates was then determined, from an irradiation performed in CP-5, to be much higher than could be obtained in the exponential with the present shielding.

Next, radiochemical techniques were explored for use at lower power levels. This method involves chemical separation and counting of Mo^{99} and Np^{239} , respectively, to determine the fission to capture ratio. The preliminary measurements indicate that this method is promising, but the accuracy obtained to date needs improvement.

3. Shielding

The last series of experiments with enriched lattices have been made with the source reactor power limited to a maximum of 25 watts in order to adhere to the biological tolerance level for fast neutrons. This power is sufficient for buckling measurements, but is inadequate for certain foil and spectra measurements.

Extensive surveys of the neutron field around the Exponential Building, Building 316, were made with and without various shadow shields in the reactor cell. The net result of these measurements and corresponding calculations is that the reactor or cell must be shielded with additional concrete to permit operation at the higher power levels.

4. Analysis of Experiments

Fundamental mode calculations for the core material of the enriched fast exponential experiment are continuing. It appears from studies made so far that variation of inelastic scattering parameters over what seems like a reasonable range does not produce any pronounced effect on the buckling. The effect on the energy spectrum of this variation, while significant, is not as sharp as might be supposed, apparently because of a stabilizing influence exerted on the spectrum by elastic moderation. Changes in the gross behavior of the spectrum are small for changes in the moderating power of the medium of the order of 10%. Large fluctuations in $\xi \Sigma_s$ over a narrow energy range tend to produce corresponding variations in flux proportional to $\frac{1}{\xi \Sigma_s}$, as would be expected.

The principal uncertainties in predicting the buckling are the transport cross sections and, particularly at higher dilutions, the U^{235} and U^{238} capture cross sections. It is expected that both theoretical and experimental information about transport cross sections will become available in the next few months.

The AVIDAC group is coding a slab geometry setup on the UNIVAC that will make possible multi-group diffusion theory calculations for the reflected exponential. These are expected to give satisfactory results except for the high energy flux in the reflector.

5. Analysis of Fast Critical Assemblies

A fresh start is being made on a program of comparing Serber-Wilson, diffusion theory, and integral theory calculations. A set of three group cross sections has been prepared, and solutions using them in spherical geometry will be obtained for several assemblies. The Serber-Wilson calculations will be done on the AVIDAC, the diffusion theory calculations on the UNIVAC, and the integral theory calculations on the IBM 701 at Los Alamos, using Bengt Carlson's S-4 setup described in LA-1599.

6. Measurement of Neutron Spectra in Reactors

Results obtained from Li^6 speck loaded emulsions exposed to monoenergetic neutrons from the van de Graaff generator indicate that the energy resolution of this technique is about 200 kev at 270, 400, and 600 kev. At 100 kev the resolution is somewhat better.

The speck loaded plates exposed to the mock spectrum from the van de Graaff generator have been analyzed, and the results were normalized to and compared with the spectrum calculated from the $Li^7(p,n)Be^7$ cross section. The measured spectrum of about 170 kev (3 energy groups

from 170 to 900 kev) agreed with the calculated spectrum within the statistical errors ($\sim \pm 10\%$). However, the measured result for the 50-170 kev energy interval falls about 50% above the calculated result. An effort is being made to locate the source of this discrepancy.

The following results are the latest available from a proton recoil plate exposed for 10 minutes at 8 in. from the pedestal on the axis of the fast exponential (Snell experiment) with the source reactor at one watt:

<u>Energy (Mev)</u>	<u>σnvt (neutrons/cm²-Mev)</u>
0.8	$8.35 \pm 1.1 \times 10^7$
1.0	$2.3 \pm 0.8 \times 10^7$
1.2	$1.5 \pm 0.9 \times 10^7$
1.4	$0.6 \pm 0.5 \times 10^7$

D. ZPR-III

A tentative critical assembly program has been proposed. It envisages an initial six-month period of U²³⁵ assemblies, during which information would be obtained on shape effect, streaming, and blanket design in particular, as well as some mockup data for EBR-II. The second phase, of nine month duration, would consist of a series of moderate sized Pu assemblies to provide badly needed information on dilute Pu reactors. The third phase, lasting about a year, would be a detailed mockup of EBR-II. On completion of these programs, the work would be shifted into large scale Pu fueled assemblies to give direct data on the full scale PBR.

Construction of the building to house ZPR-III is under way and is expected to be complete by November. Design work on the assembly machine and associated equipment is progressing.

E. EBR-I

1. Pu Fuel Charge

Based on information supplied by Los Alamos, the fuel slugs for the EBR-I Pu loading will be an alloy of Pu containing 9 to 10 at/o Al to stabilize the delta phase. Present plans are to use 0.384 in. diam slugs, 3 3/4 in. long, sealed in unbonded 2S Al jackets with Na as the heat transfer media. Rocky Flats has agreed to fabricate and can these fuel elements with the canning procedure a joint development.

Cold pressure welding of the cans has been attempted without more than spotty success. Work has been started on the development of percussion and rotating arc welds.

Pu-Al samples have been supplied by Rocky Flats for measurements of thermal expansion, thermal conductivity, density, and resistance to attack by NaK. Depending on the amount of information needed for this program, these specimens may also be used for thermal analysis, metallographic, heat treatability, fabricability, X-ray diffraction, and Pu-Al diffusion studies.

Dilatometer studies have been made, and tentatively the following coefficients of expansion can be reported.

$$\alpha, 27 - 319^{\circ}\text{C} = 1.03 \times 10^{-5}/^{\circ}\text{C}$$

$$\alpha, 27 - 549^{\circ}\text{C} = 1.92 \times 10^{-5}/^{\circ}\text{C}$$

Some tarnishing of the specimens and permanent increase in length throws some doubt on the accuracy of these results.

An analysis of castings from which these specimens were obtained showed an average of 1.02 w/o Al (8.37 a/o) which is somewhat lower than requested. Densities calculated from dimensions and weights range from 14.9 to 15.2 grams/cm³, whereas a density of 14.7 grams/cm³ was obtained by loss of weight in 1, 2, 3,-tribromopropane.

Hardness values of 25.5, 30.0, and 31.7 R_F were obtained on one specimen. Rocky Flats reports 136° Diamond Pyramid Hardness values of 68.5 and 65.8 using a Tukon micro-hardness tester and 5 kg load. This is slightly harder than 3S Al which has a DPH of 59.5 using a 10 kg load.

2. Operation

The EBR-I was operated to obtain further neutron spectrum data and to measure fission ratios and breeding gain. A beam was extracted from the center of the reactor and a cloud chamber was used to study the spectrum. Fission ratios between the various fissionable isotopes were measured by means of fission counters which are small enough to go into the core through a standard fuel rod hole. Traverses across the core have been made with U²³⁵, U²³⁸, and Pu foils. Measurements of the breeding ratio by foil techniques have been started.

F. U-Pu Alloys

The 5 and 10 wt % Pu-enriched U fuel slugs mentioned in the last report, and which were exposed to NaK at 500°C for 6 days and then thermal cycled 300 times between 50 and 500°C, appeared sound enough to justify testing in the EBR. They were remachined to 0.364 in. diam cylinders, introduced into the fuel section of an EBR fuel jacket, and shipped to

Arco. The 15 w/o Pu-enriched U fuel slug had numerous deep fissures after the thermal treatment in NaK and was not considered satisfactory for irradiation testing.

The first phase of the preparation of normal U-Pu alloys for irradiation studies in the MTR has been completed. Twelve 1/8 in. diam x 3/4 in. long cast and annealed pins were prepared and introduced into Kittel capsules (Zr) with NaK as a thermal bond. Castings 3/8 in. diam. x 1-5/8 in. long are now being made to be rolled to 1/8 in. diam rod to give a total of 60 pins. Also twelve 1/4 in. diam x 1 in. long specimens containing a center hole 1/16 in. diam are being cast of 10 w/o Pu-U alloy.

Pins were cast in triplicate containing 0, 5, 10, and 15 w/o Pu, balance normal U. All alloys had good casting properties and no difficulty was encountered in top pouring into a Cu mold. They were heat treated in vacuo at 500°C for 70 minutes and then finish machined.

The average Rockwell A hardnesses of the pins at various stages of processing indicate that the 5, 10, and 15 w/o Pu pins were appreciably harder than the pins containing no Pu, but that within this composition range, there was no appreciable difference in hardness. Also, the 500°C heat treatment and the machining had little, if any, influence on the hardness of the U-Pu alloys.

Casting No.	w/o Pu	Rockwell A Hardness		
		as Cast	as Heat Treat.	as Machined
2	0	53.2	55.4	57.5
4	5	61.8	61.8	61.2
6	10	58.4	58.4	60.5
7	15	61.9	60.3	60.9
14	30	72.5	70.5	-

In general, no difficulties were encountered in machining these alloys. Qualitatively the 0 and 5 w/o Pu alloys and the 10 and 15 w/o Pu alloys were alike. Although the reverse was anticipated, it was found that the ease of machining increased with increasing Pu content.

The densities of the various Pu pins were in the range 18.81 to 18.92 grams/cm³. The 15 w/o Pu pins were slightly less dense with a range of 18.72 to 18.78 grams/cm³.

A 30 w/o Pu-U alloy was cast in the form of 1/8 in. diam x 1 in. length pins. These were extremely brittle, as evidenced by cracking during hardness testing, handling, and even during storage. Although as-cast surfaces were tarnish resistant, freshly opened surfaces, such as from

cracking, disintegrated rapidly to a black powder. The alloy was appreciably harder than lower Pu alloys, as shown in the table. The alloy was easier to machine than the lower Pu alloys.

G. Engineering Development

A complete fuel section of a fuel subassembly has been constructed to the reference design specifications for EBR-II. The alloy rods will be 0.164 in. OD by 14.5 in. long with a stainless steel tube OD of 0.188 in. The fuel elements will be separated by spirals 0.065 in. high. The individual elements are assembled into subassemblies of hexagonal cross section. This first assembly has been made substituting steel pins in place of the fuel pins since it is intended for a series of engineering tests. The assembly is to be incorporated in a water loop to determine pressure drop and the flow stability of this geometry at various flow velocities. Of particular interest is any indication of vibration or chafing of the fuel elements.

In the operation of the dynamic corrosion loop, it was found that the oxide concentration failed to stabilize during an eight-hour period of operation. Two extended runs were conducted in a loop that has inherent cold trapping characteristics. It was found that an equilibrium oxide concentration is not achieved until the loop has been in operation long enough for a redistribution of oxide to occur or 40 to 50 hours of continuous operation. These experiments were conducted with the cold trap bypass circuit closed, and the cold trapping referred to occurred at discontinuities in the loop proper which operate at a lower temperature than the main loop stream.

H. Fuel Fabrication Facility

The Architect Engineer, Shaw, Metz and Dolio, of Chicago, has completed Phase I of the design, including layouts and cost estimates. The plan of the operating floor is shown in Figure 7. The facility will include hood lines for the facilities necessary to cast, form, and clad U-Pu alloys. The building will be 72 ft wide by 244 ft 3 in. long.

A sample hood has been built using extruded Al stringer pieces, Al corner castings welded to the various stringers, stainless steel bottom, and CR-39 plastic sides and top. The appearance of the hood is good. Tests for gas tightness have not yet been run.

FIG. 7

III. NAVAL REACTOR STUDIES, 4310

A. SAR

The design study of the Submarine Advanced Reactor has been completed, and a topical report, ANL-5217, has been issued.

Since one of the basic requirements of the SAR design was a high degree of reliability, a detailed study of the effects of possible malfunctions, operator errors, and sabotage actions on the reactor plant has been initiated. This study is expected to provide a clearer picture of the inherent safety of the design.

Studies of the design of the SAR pressure vessel, as well as tests on the mechanical properties and irradiation stability of "Carilloy T-1" steel, are continuing. This alloy has been selected as the most promising material for construction of the SAR vessel.

B. NAR

The study of a flashing water reactor, as a part of the Naval Advanced Reactor Program, is continuing.

C. STR Fuel Recovery Process

Zircaloy-II subassemblies were successfully dissolved in HF in the pilot plant. As indicated by the laboratory studies, it was necessary to add a small amount of HNO_3 in order to dissolve the Sn. Specimens of possible construction materials (Carpenter 20, Hastelloy C, Hastelloy F, Illium R, Monel) were exposed in the dissolver during dissolution; they showed increased corrosion due to the HNO_3 addition. After the dissolver solutions were complexed with $\text{Al}(\text{NO}_3)_3$, the distribution ratio of U between 10% tributyl phosphate in Amsco and the feed solution was 13. This is considered adequate. The stability of feed and dissolver solutions followed the pattern found in earlier laboratory studies.

IV. ADVANCED ENGINEERING AND DEVELOPMENT, 4500

A. Reactor Physics and Critical Assembly

1. Shielding Studies

Using the Monte Carlo method about 20,000 histories of neutrons have been computed, using a simple formulation for study of neutrons in water up to 20 cm thick. Reliable results have been obtained. With thicknesses of water over 20 cm, the use of an exponential transformation leads to very wild statistical behavior.

2. Discrepancies Between Bucklings Measured in Exponential and Critical Experiments

The theory of asymmetric diffusion was developed to the point where it is clear that such effects are not great enough to account for the observed discrepancies.

The effect has been cleared up by further experiments carried out at Savannah River where the discrepancies observed were greater than the one previously noted at Argonne. It is found that the exponential measurements were somewhat distorted by the superposition of a distribution of the photoneutrons produced in the D_2O by gamma rays coming from the source reactors. This intended to give relaxation lengths in the axial direction that were too great. When these neutrons were measured separately to give a background to be subtracted off from the measurement of all neutrons, the distribution in the differences gave a buckling in excellent agreement with that obtained from critical experiments. The background distribution was obtained after insertion of a Boral plate to absorb all thermal neutrons coming from the source reactor.

3. Reactivity Changes of Irradiated NRX Fuel Rods

Preliminary analyses based on the previously reported measurements of sections of NRX slugs and on more recent measurements of assorted enriched, depleted, and poisoned calibrating material gave the following preliminary results for the variation of η of natural U produced by irradiation in the NRX reactor.

<u>Irradiation MWD/T</u>	<u>$\delta\eta/\eta$ (apart from effect of Sm)</u>
0	0
50	0.002
300	0.008
1200	0.003
1400	0.002
2000	-0.003
2960	-0.019

The results will be recalculated and reported in full after analyses of the calibrating slugs have been completed.

4. Computers

a. Auxiliary Memories

The first tape drive auxiliary memory has been completed, thoroughly tested, and shipped to Oak Ridge for installation with the ORACLE. The second one, for use at Argonne, is approaching completion.

b. AVIDAC

The AVIDAC has been in regular operation. There has been some trouble with tube failures, the frequency having gone up in the hot weather. More drastic cooling appears to be needed.

Many sub-routines have been checked. A multi-group reactor problem previously worked on with the IBM CPC has been coded for the AVIDAC and checked. Other CPC problems have been or are in the process of being transferred with the expectation that the CPC will be given up in the next few months. The coding for reduction of the data of the exponential experiment by the AVIDAC is nearly complete.

c. GEORGE

Construction of additional units to be used with the magnetic core memory ordered from the Telemeter Corporation to make a new computer has been authorized. GEORGE is the name so far most used for it. Its arithmetic unit will be similar to that of the ORACLE. The control unit will have to be considerably different because of the different characteristics of the memory. Also, a few further changes in orders have been made in the light of experience gained in using the AVIDAC, ORACLE, and UNIVAC.

B. Fuel Elements, Fuel Media, and Producer Elements, 4530

1. Development of Viewing Systems

a. Optical Glass Study

The radiation stability of six different lens cements was tested. No change in two of these cements, Balsam and n-butyl-methacrylate plus Pentalyn resin, was observed with a magnification of 90 after a 10^8 r exposure to radiocobalt gamma rays. The other samples showed varying degrees of bubbling and separation of the cement to glass bond.

b. Shielding Glass Study

The non-browning property of a modified barium silicate glass, extensively produced by the Pittsburgh Plate Glass Company, was studied with respect to changes in the alkali oxide content. This glass contains nearly equal molar percentages of K_2O and Na_2O . When the same total molar content of alkali oxides is added as K_2O , the coloration for a 5×10^7 r exposure is increased by a factor of 2.0 relative to the production glass. When the same alkali oxide content is added as Na_2O , the coloration factor is 2.4. Decreasing the K_2O content lowers the coloration sharply but does not improve on the production glass. The tests indicate that the degree of radiation protection of this glass is closely dependent upon the kind and amount of alkali oxide modifiers added.

c. Window Design Study

Gamma heating of the interior of an all glass window by high intensity radiation has been calculated. It is concluded that forced air cooling of the window surface toward the source is desirable for incident intensities above 5×10^5 r/hr. For a 10^6 r/hr incident intensity from a fission product source, it is estimated that the temperature rise at 18 in. into the window will be $85^\circ C$. Exposure periods longer than ten hours at higher intensities will require internal cooling of the window section toward the source as well as its outer surface to reduce the maximum temperature and hence to lower the thermal stress in the glass.

2. Development of Manipulators for Handling Radioactive Materials

a. Electronic Master-Slave Manipulator, Model 2

Several breadboard amplifiers have been constructed and tested using the new 10-watt servomotors. The performance of the force reflecting servo is good with the exception that, at low speeds and at

intermediate speed, a negative mechanical impedance is observed. The effect may not be large enough to be significant. Further tests will be made with a newly constructed dynamometer.

The mechanical components including the electric motors and synchros are nearly completed in assembly, and some mechanical testing has been started. Amplifiers for one arm are well along in construction.

b. Master-Slave Manipulator, Model 8

This mechanical manipulator, in the form of a lower case "h," has been completed in design and one arm has been completed except for minor revisions. The manipulator has a load capacity of about 25 lb, can be easily removed from a horizontal hole in the shielding wall, has remotely removable tongs and can be booted. The over-all performance of the manipulator including its geometry, low friction, strength and stiffness, indicates it to be the best high level, mechanical master-slave manipulator yet developed.

3. Graphite Experiment Regulator Servo

An instrument servo system has been designed for use in a series of electro-chemical experiments for studying the effects of partial oxidation on the structure of graphite. The energy level of the graphite, determined by measuring the potential of the graphite relative to a reference electrode, is altered by passing current through the graphite.

The control system consists of two servo loops. One servo indicates the potential of a cell formed by the graphite sample and a reference electrode immersed in H_2SO_4 . The other servo controls the current flowing between the sample and a Pt electrode to maintain a present value of the graphite potential to an accuracy of $\pm 1/2\%$ over a period of several days.

The potential measuring servo has operated satisfactorily in the experiment and the current regulating servo is being constructed.

4. Portable X-ray Unit

Two units with reduced size and weight were built with machinable tungsten shielding. A study was made of portable shield designs for handling up to 100 curies of thulium-170 as a source of low energy gamma radiation. Safe operation of the designs is furthered by including a shuttering device with a spring return and pointer indicating whether the source is shielded or exposed. This shuttering device can be actuated remotely and can be locked.

C. Separations Systems, 4581

1. Engineering Studies Involving Solvent Extraction

The lifetime of single drops for a number of simple liquid pairs can be correlated by the equation:

$$\tau = \frac{580 \eta_c^{0.5}}{\Delta\rho^{0.4} \gamma_i^{0.1} g^{0.6}}$$

where τ = drop lifetime, sec

η_c = viscosity of continuous phase, g/cm sec

$\Delta\rho$ = density difference between liquids, g/cm³

γ_i = interfacial tension, dynes/cm

g = acceleration of gravity, cm/sec²

The equation does not apply to systems containing surface active agents.

Studies of the settling of gross dispersions have resulted in a tentative correlation of settling time with physical properties.

2. Fluoride Volatility Systems Process Research

The use of volatilization techniques for recovery of U from Zr-U alloy (STR fuel) and Al-U alloy (NP fuel) is complicated by the difficulty of reacting the bulk constituent with a suitable fluorinating agent such as BrF₃. Experiments were conducted showing that it is possible to remove the Al or Zr by vapor phase chlorination or bromination followed by fractional sublimation of the resulting chlorides or bromides from UF₄ or UBr₄. The U would then be fluorinated to UF₆ and decontaminated by distillation. As the vapor phase reaction presents problems in heat dissipation, the chlorination or bromination will be attempted in fused salt media. Preliminary experiments on the bromination of Zr-U STR alloy in refluxing molten AlBr₃ have resulted in complete reaction.

Laboratory dissolvings of 100-gram slices of U in BrF₃ under constant pressure-constant temperature conditions are rapid and easily controlled; it is necessary to remove product to maintain constant pressure. Dissolvings under conditions of constant volume-constant temperature are not recommended because the initial reaction rates are low and because there is sometimes a pressure surge which is difficult to control.

U ignitions have been observed in BrF_3 , $\text{BrF}_3\text{-UF}_6$, $\text{BrF}_3\text{-Br}_2$, BrF_5 , ClF_3 , and F_2 . U submerged in liquid BrF_3 reacts smoothly at temperatures appreciably higher than those where ignition occurs in the vapor. For example, U dissolves smoothly in BrF_3 liquid at 210°C , but ignites in BrF_3 vapor at 150°C .

Corrosion studies at liquid vapor and interface conditions extending for periods of 4 to 11 weeks were completed. The results indicate that Ni, Monel, K-Monel, and Inconel are excellent materials of construction for interhalogen use. Duranickel, Incoloy, Al, and possibly Mg appear to have some limited application. Ni, Monel, K-Monel and Inconel were exposed to 14 cycles of repeated exposure to BrF_3 and aqueous solutions as might be encountered by a plant dissolver. Corrosion of all samples was approximately the same, ~ 0.005 mils per cycle.

The 75°C isotherm of the $\text{Br}_2\text{-BrF}_3$ systems has been determined. It indicates that some species other than the two originating components is formed in the system. Vapor densities of BrF_3 and BrF_5 were measured at 75°C and the vapor densities were used to compute the apparent molecular weights. The results indicate the probability of some molecular association in BrF_3 vapor. Preliminary experiments at 75°C indicate that BrF_3 reacts quantitatively with F_2 to form BrF_5 ; it is probable that Br_2 reacts similarly.

Calibration and shakedown tests of the pilot plant have continued. The fractionating columns were operated with water, methanol, and the two binary systems methylcyclohexane - normal heptane and methylcyclohexane - iso-octane. Performance for both total and partial reflux operation was consistent with published data for the packings used. The cell gas stream scrubber was tested for absorption of BrF_3 , BrF_5 , and I_2 from air by 10% KOH solution spray.

3. Decontamination of U and Pu by Solvent Extraction from Chloride Systems

Rapid, smooth, and complete dissolution of U was effected by immersing the metal in 0.5 M HCl solution and bubbling Cl_2 through the solution. The reaction is catalyzed by 0.01 M FeCl_3 or 0.01 M HI. In the dissolution, U is oxidized to uranyl ion. Trivalent Pu is produced, requiring subsequent oxidation to the tetravalent state before extraction.

This activity has been terminated and the personnel transferred to more urgent programs. A summary report is being written.

4. Elevated Temperature Decontamination Studies

The slagging technique for decontamination of irradiated U was investigated further. When a carbon (graphite) rod was dipped into the melt, decontamination from Zr was enhanced. The bulk of the Zr was found on the surface of the rod. This technique makes it possible to use simultaneously graphite and an oxide crucible for removal of the rare earths and alkali metals via oxide slag formations. Preliminary investigations were made of the effects of Si and N₂ additions. Slagging runs were made in BeO and CaF₂ crucibles with substantially the same results as reported previously with Al₂O₃ or urania crucibles. The slag formed in CaF₂ crucibles appeared to be the same as with oxide crucibles, apparently due to residual O₂. Experiments with U alloy containing Zr, Mo, Ru, Pd, and Ce in concentrations approximately five times that expected in 20,000 g/t, 135-day irradiated, 15-day cooled fuel indicate that slagging has similar qualitative effects on the individual elements as it has on the same elements at Oak Ridge level (40 g/t, 100-day cooled).

Zone melting studies have shown that Zr moves counter to the direction of zone travel and that Pd, Mo, and Ru move with the molten zone. A 30-kw 10,000 cycle induction heater was installed.

Fission product spectra were calculated for fast fission of U²³⁵ and Pu²³⁹ under EBR-II operating conditions and for Pu²³⁹ under PBR conditions. The fission elements which predominate in amount are Zr, Mo, Ru, Pa, Xe, Cs, Ce, and Nd.

The high level cave, Model 1, being designed for the high level decontamination studies, consists of two high level cells designed for activities up to 10,000 curies of one Mev gamma radiation and a low level cell for activities up to one curie. The design will incorporate special contamination control facilities for handling Pu. These facilities include a device for transferring small pieces of equipment into the cells from the operating face and an arrangement for removing large pieces of equipment in a sealed container into the adjacent isolation rooms. Studies are continuing on methods for removing large pieces of equipment from the cave with adequate control of contamination. Final layout subassemblies have been completed covering the arrangement of the operating face, ventilation, and the transfer device for the operating face. Detailed design is in progress on the plumbing, electrical services, and lighting.

5. Fluidization Studies

A pilot scale stainless steel fluidized bed reactor for denitration of uranyl nitrate solution was constructed and installed. It consists of a 6 in. diam gas dispenser and reaction chamber, and a 12 in. diam disengaging section. The walls of the reactor are heated with electrical

resistance strip heaters. The off-gases are removed through porous stainless steel bayonet type filters placed in the disengaging section. Product UO_3 is removed from the bottom of the fluidized bed.

A study has been started of the dynamics of the system UO_3 fluidized by air. Correlations were observed between air velocity and bed expansion.

D. Gamma Irradiation Facility, 4584

Design has been completed for the 10^6 to 10^7 r/hr gamma radiation facility to be situated in Building 310. A tentative design has been made for an intersite carrier to be used in shipping the spent MTR fuel assemblies which will be the radiation source.

E. Processing of Laboratory Wastes, 4791

The removal of mixed fission products from tap water by cation exchange resins is increased by using iron and activated alumina in conjunction with the resin bed. This effect is noted only to the point of hardness breakthrough.

The concentration of aqueous waste by electrodeionization through permselective membranes is made more effective by filling the deionizing cell with granular ion exchange resins. This reduces the resistivity of the cell, making it possible to use low voltages for producing high purity water.

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