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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

SEPTEMBER 1959

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

SEPTEMBER 1959

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
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OAK RIDGE NATIONAL LABORATORY
STATUS AND PROGRESS REPORT

September 1959

This Status and Progress Report summarizes that portion of the Laboratory's work which is unclassified. Some of the topics are included every month, but the majority are reported on a bimonthly schedule.

PROGRAM 2000 – SPECIAL NUCLEAR MATERIALS

Power Reactor Fuel Reprocessing Pilot Plant. – During August and September, 5 tons of BNL fuel and several tons of irradiated depleted uranium were processed. The plant was shut down for about 30 days after BNL fuel processing for cleanout, decontamination, and equipment repair and modification. Measured uranium and plutonium losses for the BNL fuel processing totaled 2.2 and 3.6%, respectively; plant cleanout and decontamination operations accounted for 50% of the uranium and 80% of the plutonium losses. Over-all gross gamma decontamination factors across three solvent extraction cycles averaged 1.7×10^5 and 1.5×10^6, respectively, for uranium and plutonium. Both uranium and plutonium products met fission product specifications for subsequent processing.

The irradiated depleted uranium is being processed to recover plutonium containing a high percentage of Pu^{240}. Over-all gross decontamination factors across three cycles of extraction averaged 9×10^6 for uranium and 4×10^6 for plutonium. Plutonium losses averaged 0.9%. (AEC Activity 2461)

PROGRAM 4000 – REACTOR DEVELOPMENT

GAS-COOLED REACTOR PROGRAM

EGCR Construction. – Management Services, Inc., is grading the EGCR site and has started the required excavations. Approximately 120,000 yd^3, of an estimated total of 226,000 yd^3, of both rock and dirt have now been moved. Excavations for the reactor and service buildings are being made.

The Roehl Construction Company continued work on the EGCR access road, with progress reaching the 25% completion level. Stripping of the roadway has been completed. Culverts have been delivered to the site and distributed to the proper locations. Fill and compaction are now in progress in the vicinity of station 0, which is adjacent to the site. The Roehl Construction Company also continued work on the 16-in. potable water line, reaching approximately 10% completion, with about 500 ft now installed.

[redacted]

Construction work on the transmission line from the EGCR site to the ORNL substation continued. Line structures have been installed, and clearing of the right of way has been completed. This work is being done by the Tennessee Valley Authority. The H. K. Ferguson Company was selected by AEC as the cost-plus-fixed-fee contractor for construction of the EGCR. (AEC Activity 4141.1)

Physics of EGCR

Results of recent detailed studies of power distribution in the EGCR with experimental loops show that:

1. There is hardly any asymmetry in the power distribution in a fuel cluster next to an empty 5-in. loop tube (near the center of the core), and the average power is the same in the next ring of fuel elements (two channels away from the loop).

2. Over-all power flattening is essentially the same as calculated during the scoping study, that is, $(P_{max}/\overline{P})_{radial} \sim 1.28$.

3. Near the large loops in positions (12,12), the power is unsymmetrical in the outermost fuel channels; the variation from inner to outer edge of the cluster is about 6%.

4. While the familiar upswing of the flux at the core edge next to the reflector does occur, the fuel elements do not see it; essentially all the flux variation occurs in the graphite; the power in the outer row of fuel channels is symmetrical and slightly below that in the next row of fuel elements (second row in from the reflector). (AEC Activity 4141.1)

GETR Experiments. - Compressors 1 and 2 have continued to operate satisfactorily in the GETR loop, having accumulated about 200 hr of operation. Considerable difficulty is still being experienced with helium leaks through gaskets and miscellaneous seals. The loop is operating satisfactorily, however, despite the leaks. (AEC Activity 4141.1)

ORR Experiments. - Irradiation of eight capsules continues. On September 6, at the end of reactor cycle 15, the calculated burnups were as follows:

Capsule	Burnup (Mwd/metric ton)
01	1227
02	1241
03	1389
04	1578
05	1538
06	1282
07	992
08	860

(AEC Activity 4141.1)

Metallurgy. - Microscopic examination of several type 304 stainless steel specimens from creep tests has demonstrated the great effect which stress may have in accelerating diffusion processes. Sections are cut for microscopic examination from the gage length of the specimen and from the shoulder, which is under very low stress. Specimens 0.060 in. thick tested in CO at 1500° F were carburized to a depth of approximately 0.002 in. in the shoulder, whereas the gage length was carburized quite heavily through the entire thickness. A similar specimen tested in air at 1700° F was found to contain a very dense Widmanstaetten precipitate and a second-phase grain-boundary network in the gage length, whereas the shoulder was void of second-phase material. In this specimen, as well as ones tested in air at 1500° F, the gage length exhibited intergranular oxidation throughout its thickness, but the shoulder of the specimen was uniformly oxidized to a depth of approximately 0.001 in. (AEC Activity 4141.1)

Heat Transfer Experiments. - Naphthalene mass transfer in septafoil geometries is being studied as a means of estimating heat transfer. Two naphthalene mass transfer runs have been completed, using a mass transfer rod at tube position No. 4 (peripheral position) at $L/d_e = 17.9$ (1 in. upstream of the midcluster spacer) and at a Reynolds modulus of approximately 68,000. The mass transfer data obtained indicate that the surface temperature asymmetries during heat transfer experimentation are not due to phenomena associated with heat transfer (e.g., variation in physical properties with temperature) but are actually due to flow maldistribution at the entrance

to the test section. These data indicate the existence of a light left-hand helical flow through the tube cluster. The existence of this helical flow (which has a pitch of 3 to 5 ft) was confirmed by attaching fiber tufts to the inlet of the tube cluster. This swirl was found to be very unstable, and at a given flow rate it could be stopped or even reversed in direction by judiciously locating small obstructing strips in the inlet plenum chamber. (AEC Activity 4141.1)

THERMAL-BREEDER REACTOR PROGRAM

Homogeneous Reactor Program

Homogeneous Reactor Test. – Run 20 was terminated on August 30, after the reactor had accumulated 1738 Mwhr at power levels up to 5 Mw. It was necessary to end the run because of the leakage of radioactive liquid through the diaphragms in the heads of the two fuel feed pumps. Both heads failed from unknown causes after only 2000 hr of service; the expected life of these pumping units is 4000 to 8000 hr. During replacement of the pump heads, the core dump valve and fuel feed block valve were replaced also.

The inside of the core vessel was examined with optical devices and photographed. The appearance of the hole, diffuser screens, and core wall was essentially unchanged from the previous viewing period which followed run 17. Core-tank thickness measurements were taken and indicated that the previously reported corrosion of 0.040 in. at the equator was incorrect; the maximum loss of wall thickness in the upper hemisphere was only 0.010 in.

Startup operations preliminary to run 21 are in progress. (AEC Activity 4151.3)

Reactor Analysis. – Survey calculations have been made for determining the critical concentration, core-to-blanket power ratio, core-wall power density, and breeding ratio for the HRE-3 as a function of core diameter, blanket thickness, blanket uranium concentration, and moderator composition. In the range of sizes studied, the lowest core critical concentrations occurred at moderator compositions ranging from 50 to 80% D_2O. The smallest core diameter for which the core critical concentration was less than 10 g/liter was computed to be 2 ft when the blanket uranium concentration was 40 g of U^{235} per kg of Th (1-ft-thick blanket). For a 36-in.-dia core, core fuel concentrations of less than 10 g/liter could be obtained for a 6-in.-thick blanket containing no uranium. (AEC Activity 4151.1)

Homogeneous Reactor Instrumentation. – The transistorized electropneumatic converter developed for the Laboratory by the University of Virginia was redesigned to improve reliability and to provide packaging which is more adaptable to HRP instrument systems.

Life testing of the first of the 12 titanium bellows assemblies was completed. This unit withstood 39,420 cycles of 1/8 in. each in UO_2SO_4 at 285°C and at 2300 psig differential pressure applied externally. This bellows assembly was developed by Fulton Sylphon for a minimum life of 10,000 cycles under the conditions described above. (AEC Activity 4151.1)

Design. – The design of a manipulator for handling the miniature television camera during viewing operations inside the HRT core was completed.

A system was designed for disposal of gases associated with core-screen removal operations.

Several design changes were prepared to improve the operational reliability of the fuel sampler and the fuel-side primary and secondary recombiners. (AEC Activity 4151.1)

Metallurgy. – In an effort to reduce the number of welds used in fabricating a Zircaloy-2 core tank and thereby increase the reliability and reduce costs, studies are being made of forming methods. The first method studied in detail was a warm power-spinning operation, which allowed the forming of flat Zircaloy-2 disks into hemispheres. Difficulties with wrinkling and spring-back were overcome. However, contrary to the original expectations of the fabricators, it will be necessary to spin a thicker wall than desired and machine to the final dimensions, because of variations in ellipticity and thinout. One advantage of the power-spinning process is that it results in a thick area around the pole. A finished hemisphere has been hot-pressed in the thick area to form a reverse-curvature flared opening, which provides a convenient means of attaching the vessel to a pipe.

On September 10, the thickness of the HRT core vessel was surveyed. Measurements were made from inside the vessel at approximately 30° intervals longitudinally and 20° intervals latitudinally, using a newly designed rig and ultrasonic techniques. The thickness varied from 0.305 to 0.313 in. in the central and upper portions of the vessel. In the lower portion, values of 0.262 to 0.310 in. were obtained; however, the heavy scale present in this area caused difficulty. The original thickness varied from 0.303 to 0.317. The four vertical welds in the vessel were difficult to measure and appeared somewhat thicker than the parent plate, the thickness being in the range 0.308 to 0.330 in. One of the welds is in the location of the previous ultrasonic measurements made from the blanket access.

In past measurements, an ultrasonic resonance technique was employed, with the Vidigage being used for the thickness measurements. A recent study has revealed that the velocity of ultrasound in Zircaloy-2 varies as much as 3%. Certain pecularities in the Vidigage system which arise because of the use of high-order harmonic resonances cause errors due to velocity differences to be multiplied by the harmonic number. Such an error, probably arising from the presence of the weldment, is thought to explain the single measurement of 0.278 in. made on September 25, 1958. A direct pulse-echo technique was used for the new measurements. This technique measures directly the length of elapsed time between successive pulse reflections across the wall. With such a technique the maximum error can be only 4% and in practice should be much smaller. (AEC Activity 4151.1)

Laboratory Corrosion Studies. – The susceptibility of cast type 347 stainless steel to stress-corrosion cracking was examined in boiling 42% $MgCl_2$ solution and in water containing 100 ppm of chloride at temperatures of 200 and 300°C. Among the variables included in the tests were stress levels from 15,000 psi to above the yield point, and machined, mechanically polished, and electropolished surface conditions. To date, no cracks have been observed under any test conditions when the metal was stressed within the elastic limit. When stressed beyond the elastic limit so as to permanently deform metal grains, cracking was observed frequently.

A number of other cast alloys, including types 304, 309SCb, 316, 330SCb, and Misco C (29% Cr, 10% Ni, 0.22% C) stainless steels, were tested in a stressed condition in chloride-containing water at 200°C. Four twin-strip assemblies of each alloy type were used; a portion of each strip was stressed to the point of permanent deformation. Cracks were found in all specimens of types 304, 330SCb, and Misco C stainless steel after exposure periods of 500 hr or less. Two specimens of cast type 309SCb stainless steel cracked in less than 100 hr, while two companion specimens survived a 1000-hr period without cracking. All four specimens of the type 316 stainless steel have passed a 1000-hr period without cracking; these tests are being continued. (AEC Activity 4151.1)

Dynamic Solution Corrosion Tests. – The second-liquid-phase temperatures for solutions initially 1.25, 2.5, 5, and 10 times as concentrated as HRT core solution during HRT run 20 (0.03 m UO_2SO_4 + 0.018 m D_2SO_4 + 0.015 m $CuSO_4$ + 0.007 m $NiSO_4$) were found to be 323, 315, 305, and 307 ± 5°C, respectively, by circulating the solutions in a titanium loop equipped with a hydroclone which separated the heavy phase. Although samples withdrawn from the hydroclone underflow pot contained heavy phase mixed with more or less light phase, it was determined that the copper- and nickel-to-sulfate ratios were approximately the same in both the heavy and light phases, whereas the uranium-to-sulfate ratio was greater and the free-acid-to-sulfate ratio was less in the heavy phase than in the light phase. (AEC Activity 4151.1)

In-Pile Solution Corrosion Loops. – In-pile exposure of the first ORR in-pile solution loop, 0-1-25, was completed, and the loop is being examined. The loop was withdrawn from the reactor on September 7, 1959, after 2741 hr of operation. The ORR operated 36,895 Mwhr during this time, mostly at the 16- to 20-Mw level. The loop was fully inserted during 30,159 Mwhr, fully retracted during 6100 Mwhr, and partially retracted during 636 Mwhr. The estimated ratios of thermal-neutron fluxes in the fully and partially retracted positions to that in the inserted position are 0.01 and 0.60, respectively.

The solution originally charged to the loop was 0.036 m UO_2SO_4, 0.024 m H_2SO_4, and 0.0054 m $CuSO_4$ in D_2O. This solution was replaced with a fresh solution of the same composition after 2078 hr of loop operation. Temperatures of 280°C in the main stream and of 310, 305, or 295°C in the pressurizer were employed. The loop was constructed of type 347 stainless steel, and corrosion of this material, during some portions of the exposure, was greater than that encountered in prior LITR loops which employed similar solutions in H_2O. Complete evaluation of the corrosion results will be made after the loop examination. In general, the performance of the loop system was good, only minor difficulties being encountered.

Preliminary Cs^{137} analyses of solution samples from the loop during the first 800 hr show a fission power of about 650 w in the loop solution with the reactor at 20 Mw and the loop in the fully inserted position. This indicated loop power is about 30% greater than that found in similar solutions in LITR HB-2 loops at an LITR power of 3 Mw. No Zircaloy-2 sample activation analyses are available yet, but a flux map of the ORR facility made within a mockup core shows that the decrease in flux with increasing

distance from the reactor lattice is greater than that in the LITR, so that the expected difference between maximum power density in the LITR and ORR facilities is greater than the difference in average power density as indicated by cesium analyses. (AEC Activity 4151.1)

Molten-Salt Reactor Program

Chemistry: Phase Behavior. – Phase studies of the binary system BeF_2-ThF_4 were completed. A single eutectic was found at 1.5 mole % ThF_4 and 526°C.

In the ternary system NaF-BeF_2-ThF_4, the peritectic reaction β-4NaF·$ThF_4 \rightleftharpoons$ 2NaF + β-2NaF·ThF_4, which occurs at 558°C in the binary system, is found at 543°C; the lowered temperature is tentatively attributed to solubility of BeF_2 in 4NaF·ThF_4 and corresponds to the limit (at the composition 76 mole % NaF, 11.5% BeF_2, 12.5% ThF_4) of the liquidus in equilibrium with 4NaF·ThF_4.

The effect of composition and temperature on the solubility of HF in NaF-BeF_2 was studied between 50 and 75 mole % NaF. A marked increase in solubility with increasing NaF concentration was found to occur at 67% NaF and was interpreted as the effect of changing from a liquid in which the fluoride ions have Be^{++} neighbors (tetrahedral coordination of Be^{++}) to a liquid containing fluoride ions like those in NaF. The heat of solution increased by a factor of 2 (to -10 kcal/mole) in the vicinity of the change.

The use of NH_4HF_2 in the synthesis of inorganic fluorides was explored in connection with the conversion of $(NH_4)_2Cr_2O_7$ and CrO_3 to $(NH_4)_3CrF_6$ and of CeO_2 to CeF_3. All these reactions proceeded smoothly to give an ammonium complex compound. (AEC Activity 4152.1)

Chemistry: Graphite Compatibility. – The best recent performance with regard to the impermeability of graphite to molten fluoride fuels and blankets was encountered with some experimental grades of fine-grained graphite tubing produced by extrusion of a blend containing coke flour and Thermatomic furnace black. These samples showed weight losses averaging about 0.1% after exposure to fuels at 700°C under 60 psig.

Although there is little reason to expect that time is an important factor in the penetration of graphite by nonwetting liquids, such as molten fluorides, this point was checked by comparing results from 1000- and 48-hr soaking intervals under both 60 and 2 psig. No difference with time was noted. With AGOT graphite, previously shown to be relatively permeable to fuels, the permeation occurred to about 1.5% by volume at 2 psig and 9.5% at 60 psig. Chemical analyses of concentric layers from cylindrical samples, soaked without pretreatment, showed that the permeation by a blanket fuel containing no uranium occurred uniformly except for a slight increase in the Be/Th ratio above the nominal value toward the center. (AEC Activity 4152.1)

Metallurgy. – A series of grades of graphites with various bulk densities and various percentages of void volume were tested for penetration by the salt mixture 67 mole % LiF, 18.5% BeF_2, 14% ThF_4, 0.5% UF_4 at 1300°F and a pressure of 150 psig. Although the percentage of the volume of the graphite penetrated generally varied with the density and void volume of the graphite, it was observed that graphite specifically treated to obtain

low permeability was penetrated to a lesser extent than normal graphites with comparable densities and void volumes. GT-123, a high-density grade of extruded graphite, was found to be penetrated to only 0.13% of the bulk volume.

Several alloy systems are being investigated as potential braze materials for attaching metals to graphite. The Au-Ni-Ta system contains alloys that exhibit moderate flow on graphite at approximately 1300°C. The application of molybdenum to graphite surfaces was accomplished both by resistance brazing and by spraying from a plasmatron. (AEC Activity 4152.1)

Corrosion Tests. – Metallographic and chemical examinations were completed on two thermal-convection loops of INOR-8 and one of Inconel; all had operated for one year. The Inconel loop, which circulated a salt mixture of the composition 53 mole % NaF, 46% BeF_2, 1% UF_4 at a maximum salt-metal interface temperature of 1250°F, exhibited subsurface voids to a depth of 7 mils. (Attack in a similar system tested at 1350°F maximum temperature was greater by a factor of 2-1/2.) One of the INOR-8 loops circulating the same salt composition at a maximum temperature of 1350°F sustained only moderate pitting to 3/4 mil, approximately the depth of attack found in a 1250°F test. The other INOR-8 loop circulated blanket salt (58 mole % LiF, 35% BeF_2, 7% ThF_4) at 1350°F; it contained intergranular subsurface voids to a depth of 2 mils. Attack of this nature was not expected in this salt and has not occurred in this temperature range with more corrosive salts; therefore the possibility of contamination of the salt is being investigated. (AEC Activity 4152.1)

Graphite Systems. – Preliminary sketches were made for a frozen-salt seal to be formed in a narrow annulus between a 5-in.-dia graphite tube and a metal tube. It is expected that the frozen-salt seal will be approximately 5 in. in length. (AEC Activity 4152.1)

In-Pile Tests. – The fuel pipe of the second MSRP in-pile loop was sectioned in preparation for the metallographic examination. The metallographic examination of the first in-pile loop was completed; the fuel and surface deposits in the pump are now being analyzed chemically.

Design of the MSRP graphite-fuel capsule test was almost completed, and fabrication and assembly work for the bench test and in-pile experiment were begun. (AEC Activity 4152.1)

Pump Development. – Testing of the helical-grooved bearing, which began last month, was concluded after 335 hr of operation at 1200°F and 106 start-stops. In addition to operation at 1200 rpm at loads up to 500 lb, it was operated at 600 rpm with loads up to 375 lb. The test was terminated to investigate the impending seizure which was in evidence at loads from 300 to 375 lb at 600 rpm. Small wear spots were found at each end of the bearing on the loaded side. There were corresponding wear bands encircling the journal.

Bench tests were performed on the salt-lubricated thrust-bearing tester with the thrust bearing immersed in a bath of oil. A problem of ingassing was found, and arrangements for eliminating it are under investigation. (AEC Activity 4152.1)

Remote Maintenance Demonstration. – Eleven remote tools, adapters, and a tool rack for use with the General Mills manipulator were received.

The third and fourth sets of closed-circuit television were installed.

Modifications to the structural-steel support for the General Mills manipulator were completed.

Some remote-maintenance operations, including the removal and replacement of the pump, were started. (AEC Activity 4152.1)

Thermal-Breeder Evaluation

Graphite-Moderated, Gas-Cooled Breeder Reactor. – The detailed evaluation of thorium breeder reactors will proceed by examining one reactor system at a time, in sequence. The desire to examine first a solid-fuel-element reactor, and the delay entailed in preparing to handle heterogeneous reactor cell calculations, led to the selection of the graphite-moderated gas-cooled reactor as the first system for detailed examination. The following is a description of the reactor selected for parametric study.

The system selected for study consists of a cylindrical core having a diameter of 3 to 5 ft and a length of 15 to 25 ft. The permanent graphite structure frames square fuel channels and occupies about 15% of the total volume. The fuel-element assembly consists of uranium-impregnated, unclad graphite plates about 2 ft long, 6 to 8 in. wide, and 1/4 to 3/4 in. thick, supported by nonfueled graphite side plates. Fuel-element structural graphite occupies about 20% of the core volume, fuel-bearing graphite about 40%, and coolant-gas passages about 25%. The blanket region is nominally 2 ft thick and consists of an extension of the core structure filled with fertile elements having plates of ThO_2 ceramic. Surrounding the blanket are a graphite reflector and a boron-steel thermal shield layered with cooling channels. The pressure vessel has a diameter of 10 to 15 ft and operates at a pressure of 750 to 2000 psi.

The core and blanket elements are charged at and discharged from both ends of the core, with countercurrent motion of elements in adjacent channels with the reactor at full power. This fueling scheme results in a steady-state core having a nearly uniform composition in respect to fuel and fission products.

Helium enters the core at 500°F and is discharged at 1500°F. A sufficient number of cores to produce 100 Mw(electrical) at a gross thermodynamic efficiency of 40% are combined in a single plant.

Irradiated core and blanket elements are processed in an on-site Thorex plant. Fuel elements are stored for 90 days to allow the fission product activity to decay. They are then wet-ground in nitric acid leaching solution. The graphite is filtered off, washed, and stored in a disposal area; the filtrate is sent to the Thorex plant. Blanket elements are processed after 30 days of storage. However, additional storage of about 120 days is required for the aqueous raffinate from solvent extraction to allow Pa^{233} to decay. All the U^{233} produced is mixed with reprocessed fuel.

The fuel is remotely reconstituted by impregnating preassembled fuel elements with solutions of uranyl nitrate and firing at 800° C to reduce the nitrate to the oxide. Structural graphite in the elements is protected from impregnation by a coat of wax, which is volatilized during firing.

The excess fuel, consisting of an equilibrium mixture of U^{233} and higher isotopes, and contaminated with U^{232} and its products of decay, is accumulated for use in other reactors of the same kind. (AEC Activity 4160)

Protactinium Poison Limitation in One-Region Breeders.- An evaluation of the effect of protactinium losses in a breeder reactor in which all the thorium is intimately mixed with the fuel (as in a very large one-region reactor) has been carried out for a reactor for which the doubling time is specified to be 25 years. For a reactor with a 25-year doubling time in which there is no external inventory, the minimum breeding gain before subtraction of protactinium losses is about 0.10, occurring at a specific power of about 1.9 Mw per kilogram of fissionable uranium. If there is as much inventory external to the reactor as there is in the reactor core, the minimum breeding gain before taking protactinium losses into consideration is about 0.14, and this occurs at a specific power of 1.5 Mw per kilogram of fissionable uranium. It is apparent that having all the thorium intimately mixed with the fuel is a substantial handicap if a reactor with a 25-year doubling time is required. (AEC Activity 4160)

Physics - BeO-U^{233} Reactors. - Reactivity and conversion-ratio calculations are in progress for a series of homogeneous BeO-U^{233} reactors with thorium in the core. Results indicate that, with optimum Be-to-U^{233} ratios, even quite modest specific powers lead to significant losses in conversion ratio because of neutron absorption in Pa^{233}. With a Be-to-U^{233} ratio of about 2000 and a specific power of 3600 kw per kg of U^{233}, the loss is about 0.10 in conversion ratio. Because of the large resonance integral of Pa^{233}, these losses are strongly dependent on the neutron spectrum. It is likely, therefore, that even in a Th-U^{233} cycle in which breeding per se is not the principal objective (e.g., a power reactor with long fuel exposure), some separation of fissionable and fertile material may be desirable.

Multigroup calculations were performed to study the effects of core size, U^{233} and thorium concentration, blanket composition, presence of a thin buffer region between core and blanket, etc. Results to date indicate, among other things, that the achievable breeding ratio with BeO is 0.06 to 0.08 greater than with graphite moderation [mainly due to the (n,2n) effect]; and that even with a BeO core, a graphite blanket appears to be better than a BeO blanket; here it is simply absorption cross section, rather than moderating ratio, that matters.

The effect of the fast-neutron reactions [(n,2n) and (n,α)] in BeO was computed and compared with those calculated for beryllium metal by Hafele. The results are as follows, where + and - signs indicate upper and lower limits for the (n,2n) cross section. The quantities ϵ^+ and ϵ^- represent upper- and lower-limit estimates of the fast effect, ϵ, in each case.

Moderator	$P^{+}_{n,2n}$	$P^{-}_{n,2n}$	$P^{Be}_{n,\alpha}$	$P^{O}_{n,\alpha}$	ϵ^{+}	ϵ^{-}
Be	0.11845	0.09397	0.04235		1.076	1.051
BeO	0.07943	0.06034	0.02915	0.00444	1.046	1.027

(AEC Activity 4160)

NUCLEAR POWER PLANTS

Small-Size Nuclear Power Plants. - Conceptual design of the nuclear power plant was initiated by the architect-engineer, Gibbs and Hill, Inc. Preliminary drawings were reviewed by the ORNL Advisory Group, and comments were transmitted to the Oak Ridge Operations Office (AEC). Assistance was provided to OROO in the formulation of information to be included in the invitation for proposals from electrical cooperatives for participation in the program. (AEC Activity 4160)

HIGH FLUX ISOTOPE REACTOR

The preliminary design efforts have been largely concerned with nuclear optimization studies, especially efforts to maximize the thermal flux in the central island at any given reactor power but with reasonable power densities and total reactor power. If power density per unit volume is the principal practical limitation, the maximum flux is obtained in a core with uniform power distribution. The flux per megawatt can be improved a few per cent by increasing the power density near the flux trap and decreasing the power density at the outside of the core. The net result, however, is that the maximum flux significantly decreases at any given maximum power density. Design studies will be continued, aiming for uniform power distribution. The next step is to calculate reactivity, flux, and power densities as a function of fuel burnup.

Different methods of machine calculations give somewhat different results. A critical experiment is necessary to find which of the codes gives the most nearly correct results. A multiregion cylindrical aluminum tank, 3 ft high and 4 ft in diameter, is being fabricated. In the experiments the core will be filled with H_2O or H_2O-D_2O mixtures, the first annulus with an H_2O-D_2O solution of $U^{235}O_2(NO_3)_2$, and the second and third annuli with D_2O and H_2O reflectors, respectively. Provision is made for an H_2O reflector both above and below the annular regions. A map of flux and power distribution will be made as well as a determination of the ratio of peak flux to total power.

No appreciable design efforts have been expended on portions of the project outside the core region and probably will not be until an architect-engineer is selected.

The reactor core is considered to be a cylinder about 15 in. OD, 5 in. ID, and 18 in. high, capable of operating at 100 Mw. Studies of swirl-type elements show that, although higher heat transfer and burnout coefficients are obtainable on a per unit area basis, these elements provide sufficiently less heat transfer area that a lower net heat removal per unit volume results.

Three types of fuel elements are being considered.

Mark I (wedge). This type is a plate element with circumferential curved fuel plates similar to those in the MTR. The side plates are not parallel. No new developments are required before this element can be manufactured. It will be more expensive, since each plate in an element is of a different uranium content, width, and curvature. The reactor fueled with this element will not reach as high a power and flux as with Mark II or possibly Mark III.

Mark II (involute). Involute plates radiate from a central tube. Two or three concentric tubes with associated plates are required. Experimental non-fuel-bearing involute plates have been made and have been assembled into segments of a circle. To be useful, involute plates must have a variable fuel concentration across the width. Several variable fuel concentration plates have been rolled to external dimensional tolerance, but results are not available yet on the degree of control of fuel distribution. At this time the technology of Mark II fuel elements appears accessible.

Mark III (concentric cylinder). If suitable spacers and spacer arrangement can be found, this element would have the highest performance of any of the current proposals. Experiments are required for determining the effect of spacers on burnout heat fluxes before a practical fuel element can be designed. This element requires the manufacturing of fuel plates of a width not previously attempted. One fuel-bearing plate was rolled which was of sufficient width and length to make one complete cylinder. No flaws were found by visual and x-ray inspection.

A safety factor of 2 from burnout at the hottest spot in the reactor has been established as a desirable goal. An experimental program to establish burnout heat fluxes under the specific conditions to be encountered in this reactor has been established, since available data from previous investigations do not include the ranges of the variables. The first of these programs was an investigation of vortex flow tubes for which a series of 40 twisted-tape burnout tests was conducted over the following ranges of variables: tube ID = 0.136 to 0.402 in.; tape twist ratio = 2.08 to 12.03 diameters/twist; L/D = 6.6 to 88.2; tube material = Al, Cu, and "A" nickel;

exit axial velocity = 14.7 to 156 fps; exit resultant velocity = 16.6 to 195.3 fps; inlet water temperature = 48 to 138°F; exit water temperature = 91 to 350°F; pressure drop = 3 to 432 psi; burnout heat flux = 2.77×10^6 to 37.35×10^6 $Btu \cdot hr^{-1} \cdot ft^{-2}$.

The burnout results were best correlated by:

$$\left(\phi_{bo}\right)_{min} = \frac{380\left(G_r\right)^{0.65}}{\left(D_1\right)^{0.2}\left(L/D_1\right)^{0.45}} , \qquad (1)$$

where

ϕ_{bo} = burnout heat flux, $Btu \cdot hr^{-1} \cdot ft^{-2}$;

G_r = resultant mass velocity, $lb \cdot hr^{-1} \cdot ft^{-2}$;

D_1 = tube ID, ft;

L/D_1 = tube length-to-diameter ratio.

For 38 of the 40 burnout points, Eq. (1) exhibits a negative deviation of 0% and an average positive deviation of 20.2%.

These tests confirmed earlier indications that swirl-flow burnout heat flux is independent of degree of subcooling and of pressure, and that at constant pumping power in the same geometry, swirl flow yields burnout fluxes approximately double those for purely axial flow.

Vortex-flow surface heat transfer coefficients were also determined over broad ranges of the variables, and the results may be briefly, though approximately, summarized by stating that swirl-flow nonboiling coefficients are double those for straight flow in the same geometry at equal velocity and coolant bulk temperature.

Swirl-flow friction factors were correlated with a modified Reynolds number, tape twist ratio, and equivalent diameter. The j/f factors obtained indicate that even in nonboiling heat transfer, certain combinations of the geometric variables produce more efficient heat transmission than can be obtained with axial flow (greater amount of heat transferred for a given magnitude of friction loss). Work on swirl-type flow for the HFIR project has been terminated in order to obtain comparable information on parallel plates with narrow gaps.

Two burnout tests have been conducted with parallel plate assemblies; these are summarized below:

Test No.	Flow Gap (in.)	Exit Pressure (psia)	Exit Velocity (fps)	Burnout Flux ($Btu \cdot hr^{-1} \cdot ft^{-2}$)
1	0.043	560	31.8	3.55×10^6
2	0.045	465	35.4	3.84×10^6

In these tests, the plates were "A" nickel, the heated length 1 ft, and the flow-gap width 1-1/16 in. The observed burnout fluxes agreed within 7% with values predicted by the extended Bernath method (Preprint 110, Third National Heat Transfer Conference, August 9 – 12, 1959). A third test with nickel plates will be conducted, after which a series of aluminum test sections will be fabricated and tested. Friction factors and nonboiling heat transfer coefficients are also being spot-checked during the tests.

Another practical limitation on reactor performance is corrosion. A desirable goal of less than 2 mils penetration in 10 days has been established, since a fuel cycle of 10 days is contemplated. Ten-day dynamic corrosion tests have been completed at 200, 230, 260, and 290°C with types 1100, 6061-T6, and X-8001 aluminum alloys; alloys 5154-H36 and X-2219 were included in only part of the tests. The test medium was distilled water flowing at 20 to 107 fps past the test specimens. The two alloys showing the most promising short-term corrosion behavior at temperatures up to 260°C were types 1100 and X-8001 aluminum; at 290°C, type 1100 corroded severely. The table below shows the observed corrosion rates for X-8001 aluminum based only on weight loss. Results for 1100 aluminum were very similar except at 290°C. Preliminary metallographic examinations indicate some localized penetrations as deep as 3 mils in certain cases.

Velocity Range (fps)	Penetration in 10 days (mils)			
	200°C	230°C	260°C	290°C
20–31	0.7	0.5	0.2	1.7
31–44	1.9	0.5	0.6	1.6
44–67	1.7		0.8	1.5
67–95	2.6	0.8	3.7	2.8
95–107	3.8	1.4	10.8	6.0

In general, all materials showed velocity-dependent corrosion. The velocity effect was very pronounced at flow rates of 65 fps and above for all alloys tested. At 260°C the preliminary data indicated that there was no effect of cavitation on corrosion of the various alloys, since runs with overpressures of 10, 100, and 200 psi produced about the same extent of corrosion at all flow rates.

The present reactor design contemplates a hot-spot fuel plate surface temperature of 205°C plus an unknown temperature drop through the corrosion film. Previous investigations have shown that isothermal corrosion tests should be operated at the sum of these two temperatures. To find the correct temperature drop and to make a direct measurement of corrosion penetration, an electrically heated test section for an existing corrosion test loop has been designed and will be constructed. The test section can test aluminum channels of the same gap, water velocity, temperature, and purity as contemplated in the reactor design at heat inputs up to 1.5 times reactor design conditions. (AEC Activity 4410)

PACKAGE REACTOR DEVELOPMENT

Army Package Power Reactor (APPR-1). – Dimensional measurements on APPR-1 stationary and control rod fuel elements revealed that no significant changes had occurred as the result of approximately 10 Mw-years of exposure in the reactor. A slight rippling on the external plates of the stationary fuel elements was observed. Visual examination of the surfaces of various fuel plates showed no apparent evidences of cracking.

A program is in progress to determine the mechanism of boron loss during sintering of boron–stainless steel compacts, with the objective of ultimate control and prediction of such loss. Compacts of boron with the elemental metals occurring in stainless steel, or with their oxides, were sintered in high-purity argon, in a vacuum, and in wet (+70°F dew point) helium. Appreciable loss of boron was observed only in an environment which provided a source of oxygen. This suggested the following reaction sequence, with the second stage being the rate-controlling step:

$$4B(s) + 3O_2(g) \rightleftharpoons 2B_2O_3(\ell)$$

$$B_2O_3(\ell) \rightleftharpoons B_2O_3(g)$$

(AEC Activity 4630)

GENERAL REACTOR RESEARCH

Reactor Evaluation Studies. – The fuel value of plutonium (95% Pu^{239}) when used in various reactors (Dresden, GCR-II, SGR, Carolinas-Virginia, Yankee) has been calculated, taking into account the hardening of the neutron spectrum associated with large thermal cross sections. The fuel value ranged from about \$9 per g of Pu^{239} in the Yankee-type reactor to about \$18 per g of Pu^{239} in the Carolinas-Virginia type of reactor. The effect of spectral hardening caused the plutonium value to be less than if this effect were not present, with the largest decrease associated with the H_2O-moderated reactors. (AEC Activity 4160)

Reactor Safety Studies. – The pulsed-neutron technique used to measure the neutron lifetime of the BSR-II (see ORNL-2795, p 6) was also employed to measure values of reactivity in the BSR-II which were too large to measure with the positive-period inhour method. These included the shutdown k value, the worth of the individual control rods, the worth of the reactivity insertion device, the worth of a corner fuel element, and some shading effects of rod position. These values were verified by the inhour method where possible. The core was then disassembled, packed, and shipped (together with the auxiliary control equipment) to the SPERT-I facility for safety tests which will begin early in October. (AEC Activity 4171.1)

Fuel Element Catastrophe Studies. – Additional experiments on the oxidation of irradiated uranium from the Hanford plutonium production gave more consistent results on the amounts of various fission products volatilized from the specimen and plated out on the attached furnace tubing. The new results are tabulated below. In none of these experiments was any particulate

material collected on the entrainment filter, although a slight vaporization of UO_3 and a significant amount of particulate radioactivity had been observed in one previous experiment when samples were heated in air. Particles had been measured ranging from 0.3 to 1.0 μ.

Samples Heated at 1200°C in Air

Time At Temp (min)	Amount of Uranium Oxidized (%)	Fission Product Liberation (% of total)						
		Gross Gamma	I Gamma	Te Beta	Cs Gamma	Ru Gamma	Rare Gases	Sr Beta
5	43.6	0.12	31.94	8.14	18.48	0.035	97.7	0.024
8	66.2	0.24	23.16	12.74	14.53	0.163	99.2	0.028
10	68	0.7	39.9	24.8	17.10	0.215	98.7	0.596
12	77.5	3.5	36.4	23.0			99.8	0.9
15	71.9	0.55	52.8	51.6	28.6	4.3	99.8	3.1
40*	4.1	0.4	47.0	0.58	2.03	0.002	97.7	0.9

*Sample heated in helium atmosphere.

A single trial run, in which the uranium specimen was ignited in oxygen instead of air, resulted in the production of visible quantities of vaporized uranium oxide. Some of the particles as measured on the electron microscope were as small as 50 Å. The ease of vaporization of particles when uranium is burned in oxygen supports the conclusion that occasionally a sufficiently activated spot may reach vaporization temperatures when specimens are burned in air. (AEC Activity 4171.1)

Long-Range Planning Studies. — A study was made of the energy distribution of the neutron flux in different regions of research reactors. It was found that peaking of the flux in the intermediate-energy range could be attained in an iron region outside the fuel region; however, the absolute value of the intermediate-energy flux in the iron region was less than that in the fuel region.

Assistance is being given in connection with the thorium breeder reactor evaluation. (AEC Activity 4410)

Power Reactor Fuel Reprocessing: Feed Clarification Studies. — In comparative scouting tests with a 5-in.-dia centrifuge, silica was completely removed from solvent extraction feed solutions of dissolved Foreign Research Reactor fuel, Darex process solutions of APPR fuel, and Sulfex process solutions of type 304 stainless steel cladding, at the following feed rates and centrifugal fields: 400 cc/min at 800g for Foreign Research Reactor feed solution, 300 cc/min at 750g for Darex-APPR feed solution, and 100 cc/min at 1000g for Sulfex-304 SS dejacketing solution. Thus the removal of silica from Sulfex decladding solutions appears to be more difficult (because of the higher density and viscosity of the solution). Black carbonaceous material in the Sulfex solution, which is essentially impossible to filter, was removed at 100 cc/min and 350g.

Spray addition of acid wash to an equal volume of cake plus supernatant at 1.3 g/min and 30 psig completely dislodged siliceous Darex-APPR cake from the 5-in.-dia centrifuge bowl. Successive wash additions and skimming for five cycles resulted in average over-all washing efficiencies of 86 to 92% of theoretical. (AEC Activity 4441)

Power Reactor Fuel Reprocessing: Zirflex Process. — Dissolution rates of 7% U–Zr alloy were greater than 5 $mg \cdot cm^{-2} \cdot min^{-1}$ in 4 to 6 M NH_4F as long as the F/Zr mole ratio in solution was greater than 7; for a mole ratio of 6 the rate was only 0.05 mg/cm^2. At ratios below 10 the addition of 1.2 M NH_4NO_3 to ammonium fluoride solution had little effect, but at ratios above 10, ammonium nitrate accelerated the dissolution. The highest rate measured was 42 $mg \cdot cm^{-2} \cdot min^{-1}$ in 5.4 M NH_4F–1.2 M NH_4NO_3 at an F/Zr mole ratio in solution of 36.5.

In six Zirflex cyclic corrosion tests, welded specimens of Ni-o-nel and type 309SCb stainless steel corroded at maximum rates of 0.96 and 1.02 mils/month, respectively. For heat-treated specimens of Ni-o-nel, the maximum rates in two cycles were 0.74 and 0.57 mil/month for air- and water-quenched specimens, respectively. (AEC Activity 4441)

Power Reactor Fuel Reprocessing: Perflex Process. — Corrosion rates were obtained for Hastelloy C under Perflex process conditions. In five 1-hr cycles in 1 M HF–0.06 M H_2O_2, the maximum over-all corrosion rates for Hastelloy C decreased from 87 to 57 mils/month. The lower value corresponds to 7.9 mils in 100 dissolution cycles. (AEC Activity 4441)

Power Reactor Fuel Reprocessing: Sulfex Process. — Two ThO_2-UO_2 (Consolidated Edison) fuel pins irradiated to 200 Mwd/ton were declad in boiling 6 M H_2SO_4. The declad pellets remained intact during 2 hr further exposure to the boiling decladding solution; the soluble uranium loss was 0.07 to 0.44% and the thorium loss 0.04 to 0.25%. Irradiated pins as received were not passive to boiling HNO_3-free 6 M H_2SO_4, even though the initial contact was at room temperature. As little as 0.001 M HNO_3 in boiling 6 M H_2SO_4 produced passivation on stainless steel jackets for periods of 1 hr or more. Contact of the pins with Ni-o-nel did not induce passivation in the absence of HNO_3. A tenfold excess of formic acid (up to 0.02 M HNO_3 and 0.2 M HCOOH) did not remove the HNO_3 sufficiently to break passivation in 1 hr. Elimination of HNO_3 contamination from the Sulfex system required thorough, repeated water rinses of the dissolution equipment.

Addition of ferric nitrate to the dissolvent in concentrations up to 0.5 M increased the dissolution of high-density (94% of theoretical) unirradiated 95.8% ThO_2–$UO_{2.36}$ pellets in 200% excess of boiling 13 M HNO_3–0.04 M NaF to a maximum of 90% in 5 hr. Rates were lower at iron concentrations higher than 0.05 M. The amount of pellets dissolved in 1 hr in the reagent plus 0.5 M $Fe(NO_3)_3$ plus sulfate ion decreased from 40% to 2% as the sulfate was increased from 0 to 0.3 M. Uranium losses to boiling 6 M H_2SO_4 from particles larger than 100 mesh were less than 0.5% in 29 hr of digestion.

In engineering-scale tests on semicontinuous decladding of Consolidated Edison fuel, 15.5-kg samples of type 304 stainless steel were dissolved continuously in boiling 6 M H_2SO_4 at a rate directly proportional

to the dissolvent flow rate. The product concentration was maximum, 62 g of stainless steel per liter, at 4.7 M H_2SO_4; at this point the dissolution rate was 1.6 $mg \cdot cm^{-2} \cdot min^{-1}$. The mole ratio of H_2 generated to stainless steel dissolved was 1.13. These data indicate a decladding time of 4 hr for a Consolidated Edison element, assuming uniform attack.

It has been previously shown that stainless steel can be depassivated in 12 M H_2SO_4. However, a 5-min "depassivation" treatment in boiling 12 M H_2SO_4 increased the corrosion rate of Ni-o-nel from 3 to 290 mils/month. (AEC Activity 4441)

Power Reactor Fuel Reprocessing: Solvent Extraction and Radiation Damage Studies. – Laboratory experiments with an acid Interim-23 flowsheet (of possible use for Consolidated Edison fuel) with a low-acid feed gave decontamination factors from ruthenium, zirconium-niobium, and rare earths which were greater than 10^5 and a uranium loss of less than 0.01%. With an acid Thorex flowsheet, using 30% tributyl phosphate in Amsco, operation was good but thorium loss was about 0.2%. About 40% of the acid in the column was extracted with the thorium.

Simple filtration of samples of recovered TBP-Amsco solvent used in high-level pilot plant runs removed as much as 99% of the gamma-active materials, the amount removed being dependent on the pore size of the filter.

A very large decrease in gross fission product decontamination factor experienced during Purex Pilot Plant operation was apparently caused by incomplete rinsing of the Turco cleaner used in previous equipment decontaminations. In laboratory experiments the addition of 1% Turco cleaner to synthetic pilot plant feed resulted in almost complete loss in decontamination from zirconium. Pilot plant solvent presumably contaminated with the cleaner was decontaminated by factors of 10 to 100 from fission products, and the agent that was complexing the zirconium was removed, by scrubbing with 0.2 M Na_2CO_3 and washing with 0.1 M HNO_3.

TBP-Amsco solutions containing 0 to 100 wt % TBP were irradiated to the 10^{22}-ev/g level (330 to 400 whr/liter) after being saturated with water. The water had essentially no effect on gas yields at any concentration or on total acid at concentrations up to 25 wt % TBP. At higher TBP concentrations, water depressed acid formation by as much as 37%.

The solubility of zirconium dibutylphosphate in aqueous uranyl nitrate–nitric acid solutions varied from 0.35 to 159 mg/liter as the uranium concentration increased from 0 to 490 g/liter and the acidity from 0 to 4 M HNO_3. Corresponding zirconium solubilities in 30% TBP–Amsco solutions equilibrated with aqueous uranyl nitrate–nitric acid were 0.33 to 3.55 g/liter. Under Purex conditions the total concentration of zirconium formed in fission of natural uranium fuel and contained in the feed solution is 40 times less than the solubility of zirconium dibutylphosphate in the extractant. (AEC Activity 4441)

Power Reactor Fuel Reprocessing: Zircex Process. – In further tests to evaluate methods for removing uranium from the volatile $ZrCl_4$ stream, substitution of two Ni-o-nel screens (30 × 250 mesh) for the glass wool packing in the 300°C uranium chloride desublimer caused the volatile uranium loss to increase from <0.1% to 7 to 8%. Over 99.9% of the uranium was recovered in nitric acid when glass wool packing was used. (AEC Activity 4441)

Power Reactor Fuel Reprocessing: Darex Process. — The main emphasis on Darex process development was changed from high- to low-enriched uranium and thorium fuels as a consequence of the cancellation of the APPR-Darex pilot plant at Building 3019 and substitution of a Darex pilot plant for low-enrichment fuels at Building 2527. The processing facilities at Building 2527 will now comprise a main-line Darex facility with a Sulfex dissolver and auxiliary equipment.

In small-engineering-scale tests the ORNL reference flowsheet as applied to Yankee Atomic fuel was found to be applicable also to NS "Savannah" prototype fuel without modifications and to total dissolution of stainless-steel-clad 5 to 8% UO_2–ThO_2 (Consolidated Edison fuel) by adding fluoride catalyst to dissolve the mixed oxide pellets. Over-all heat transfer coefficients for chloride removal were 193 to 272 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot (°F)^{-1}$.

In tests to determine ThO_2-$UO_{2.36}$ loss to 5 M HNO_3–2 M HCl Darex decladding solution, digestion of +100-mesh oxide particles for 29 hr at boiling resulted in a uranium loss of less than 0.5%.

In more refined tests to determine whether hydrogen was evolved during the Darex dissolution step, none was detected in the dissolution of 60-g samples of type 304L stainless steel in 5 M HNO_3–2 M HCl.

In continuing Darex corrosion tests, titanium showed a maximum corrosion rate of 0.23 mil/month over 13 cycles between initial and final Darex-Thorex dissolver solutions. In middle and final Thorex solutions without Al^{+++}, the maximum rate in a 1000-hr exposure was 0.21 mil/month. In the presence of 100 ppm of chloride (no Fe^{+++}), maximum rates were 8.33, 0.44, and 0.34 mils/month in 840-hr exposures to beginning, middle, and final Thorex solutions, respectively. In beginning Thorex solutions containing 0.1 and 0.2 M Fe^{+++}, titanium corroded at maximum rates of 9.85 and 10.4 mils/month, respectively, in 168-hr exposures. (AEC Activity 4441)

Power Reactor Fuel Reprocessing: Mechanical Processing. — In preliminary testing of the hydraulic SRE decladder, the tube winder and the rolling and cutting equipment operations were satisfactory. The only malfunction was in the jackscrew drive to expel slugs from end-cropped tubes.

Plunge cutting of ferrules between the tube sheets of tubular fuel assemblies with a 1/8-in.-thick, 16-in.-dia Manhattan No. 239 abrasive-disk blade at 10,450 surface feet/min was not feasible because of blade breakage.

Ucon 150 Cp hydrolube (ethylene glycol–water solution plus additives), which appeared usable as a fire-retardant hydraulic fluid in the mechanical processing program, was tested in a radiation field to evaluate its radiation stability. At a gamma radiation dose between 10^8 and 3.4×10^8 r, the material changed from a solution to a gel.

At gamma-ray exposures of 10^{10} r, Pfaudler No. 53 cover glass became separated from Pfaudler B-5 ground coat on an Inconel, Hastelloy C, or type 304L stainless steel base but not on a carbon steel base. (AEC Activity 4441)

Power Reactor Fuel Reprocessing: Recovery of Uranium and Thorium from Graphite Fuels. — Three spherical Pebble Bed Reactor fuel samples which had been ground to -200 mesh were leached with two portions of boiling 15.8 M HNO_3. Uranium recovery was 99.9% from the two ungraphitized admixture specimens but only 98.1% from the solution-impregnated ball.

The amount of graphitized uranium-graphite fuel which disintegrated when several 37-g samples were soaked in bromine at 25°C varied from nearly zero

for a sample containing 0.7% uranium to 94% with a sample containing 9% uranium. Swelling of the fuel by the bromine did not increase the uranium recovery by nitric acid leaching of the 9% uranium-graphite fuel over that achieved with the standard grind-leach method. However, the uranium recoveries from bromine-treated 0.7% uranium, 2% uranium, and 1.5% uranium-7.2% thorium fuels which were then leached with two portions of boiling 15.8 M HNO_3 were 96, 99, and 94%, respectively. Mechanical grinding of these fuels to -4 +8 mesh and leaching by the same technique resulted in recoveries of only 80, 90, and 89%, respectively. (AEC Activity 4441)

Fluoride Volatility Processing. — The fused salt-fluoride volatility process was successfully tested on a laboratory scale with 650-g batches of fully irradiated S2W1 fuel (first "Nautilus" core) decayed 2.5 years. In two runs through the complete process, including the NaF absorption-desorption step, the product UF_6 contained essentially no activity other than that due to the Th^{231} daughter of U^{235}. Ruthenium decontamination factors in the fluorination step at 500°C were from 110 to over 10^3, in contrast to 10 or less for fused-salt fluorinations at 600°C.

In 74.5 hr of hydrofluorination of zirconium under proposed pilot plant flowsheet conditions, starting at 20 mole % ZrF_4 and 650°C and ending at 45.50 mole % ZrF_4 and 525 to 550°C, the INOR-8 vessel was corroded less than 2.5 mils. The average dissolution rate of the zirconium in six runs, with an HF flow rate of 4.0 lb/hr, was 1.4 $mg \cdot min^{-1} \cdot cm^{-2}$, compared with 2.0 $mg \cdot min^{-1} \cdot cm^{-2}$ for the 700°C flowsheet.

In low-temperature fuel dissolution studies, zirconium metal reacted vigorously with anhydrous HF containing KF, NH_4F, NO_2, or organic materials, for example, ethanol. According to the theory of solvent systems, such materials act as bases and increase the concentration of the fluoride anion. Emphasis is being placed on the NH_4F-HF system, leading to the product $(NH_4)_3ZrF_7$, which can be thermally decomposed to ZrF_4. The reaction rate of 15 mole % NH_4F in HF was found to be 90 $mg \cdot min^{-1} \cdot cm^{-2}$ at the reflux temperature of 60°C.

In dissolution tests in INOR-8 equipment a black solid formed which appeared to decrease the zirconium dissolution rate and formed objectionable deposits in the system. Metallic nickel is a major constituent of the black deposit, which is believed to be the product of a chemical reduction of the NiF_2 corrosion product. In a test in an all-copper system with a salt to which NiF_2 had been added, typical black solids were deposited. (AEC Activity 4442)

Waste Treatment and Disposal. — Evaporation to dryness and calcination of residual solids are being studied as a means of preparing high-level solvent extraction wastes for permanent disposal. The nitrate content of calcined TBP-25 waste (aluminum nitrate base) was found to be 0.84, 0.53, and 0.16% after 24 hr at 600, 800, and 1000°C, respectively. Purex waste solids were 2.7% nitrate at 400°C, 0.1 to 0.4% at 600 to 800°C, and 0.1% at 1000 to 1200°C. Thorex waste solids were 5.3% nitrate at 400°C, 1.3% at 600°C, and 0.2 to 0.5% at 800 to 1200°C.

In evaporation and calcination of acid wastes of various types in a stainless steel pot equipped with a glass condenser, 4 to 50% of the ruthenium was found in the condensates, as contrasted with 0.1 to 0.2% with neutralized wastes. About 15% of the ruthenium plated out on the calciner and condenser.

Thermogravimetric analysis of waste components indicated that both sodium and cesium are volatile as nitrates at proposed calcination temperatures but not as sulfates.

The thermal conductivity of as-formed TBP-25 waste solids of bulk density 0.40 g/cc is represented as a function of temperature by the empirical relation $k\ [\mathrm{Btu \cdot hr^{-1} \cdot ft^{-1} \cdot (^\circ F)^{-1}}] = 0.00732 + 0.000244T\ (^\circ F)$. The thermal conductivity of Darex waste solids which were granulated after calcination to a bulk density of 1.14 g/cc increased linearly with temperature from 0.100 at 200°F to 0.145 at 1500°F.

An experimental calcination run in which concentrated simulated Darex (stainless steel nitrate) waste was fed continuously to the 8 × 48 in. stainless steel calcination vessel operating at 900°C terminated in a violent rupturing of the vessel. This resulted from the carry-over of foam into the vessel vent line, where solids were deposited. Pressure measurements from the top and bottom of the vessel indicated a uniform pressure rise from 3 to over 100 psig in about 17 min, at which point the bottom weld failed and the vessel rocketed out of the calcination furnace. The vent system is being redesigned. (AEC Activity 4452)

PROGRAM 5000 — PHYSICAL RESEARCH

Reactor Operations. — The operating cycle of the ORR which ended on September 7 was the most successful to date. No unscheduled shutdown occurred during this cycle. The preliminary proposal for 30 Mw of additional cooling capacity for the ORR has been submitted to the AEC for approval.

Approval for annealing the Graphite Reactor was received on September 11. Plans are being made to install the cross ducts this fall. If the cross ducts can be installed by spring, the annealing will be done early next summer.

Laboratory Facilities. — Work is scheduled in several of the new segmenting cells in Building 3026-D.

PHYSICS

Fast-Chopper Time-of-Flight Spectrometer. — A precise determination of the total neutron cross section of B^{10} has been made from 0.018 to 0.4 ev, using boron samples highly enriched in B^{10}. The total cross section follows a 1/v law in this energy range and has a value of 4020 ± 30 barns at 0.0253 ev. Total and scattering cross-section measurements have been made on platinum foils ranging in thickness from 0.001 to 0.020 in. in order to determine the parameters of the resonances up to 320 ev. Transmission measurements were also made on a sample enriched in Pt^{195}. Of the 14 resonances assigned to Pt^{195} up to 320 ev, only two or possibly three have a spin of 0, and the others have a spin of 1. Capture cross-section measurements have been made on thin samples of Au, Pt, W, In, and Ag with a 28-in.-dia liquid scintillator tank. The data are being analyzed to give the capture cross sections of the elements in the kev energy region and parameters for the resonances at low energies. (AEC Activity 5220)

High-Voltage Experimental Program. — Since all the capture cross sections have been measured relative to indium, considerable effort has been expended in accurately determining the shape and magnitude of the indium cross section. The capture cross section of indium was measured relative to the absorption cross section of U^{235}, using $Li^7(p,n)Be^7$ neutrons from a thin lithium target in the energy range 10 to 400 kev. Neutron energies were determined from the incident proton energies, and the neutron flight times were used to distinguish capture events from background. In a second experiment, the cross section of indium as a function of energy was measured relative to the known 1/v shape of the $B^{10}(n,\alpha\gamma)Li^7$ cross section in the energy range 10 to 175 kev, using $Li^7(p,n)Be^7$ and $H^3(p,n)He^3$ neutrons from thick targets and using velocity selection to determine the neutron energies. The absolute value of the cross section was obtained by normalization to the thin-target results. In a third measurement, an indium foil was calibrated by activation with 29-kev, $Li^7(p,n)Be^7$ forward threshold neutrons, where the neutron flux was determined with a graphite-sphere detector. [Nuclear Instr. 1, 335 (1957)]. This calibrated foil was then activated, using 29-kev neutrons, while measuring its gamma-ray yield with the scintillator tank. This same pair of measurements was repeated at

63 kev, using forward threshold neutrons from the $H^3(p,n)He^3$ reaction. Finally, from the shape of σ_a of U^{235} with energy and the 175-kev value (1.77 barns) for $\sigma_a = \sigma_f(1 + \alpha)$, the absorption cross section of U^{235} has been determined from 10 to 175 kev. (AEC Activity 5220)

CHEMISTRY

Raw Materials Research and Development. — The selectivity of Primene JM for thorium over rare-earth and iron contaminants was improved markedly by addition of a long-chain alcohol to the kerosene diluent. For example, coefficients for extraction of cerium(III) and iron(III) from sulfate solutions decreased by factors of 6 and 20, respectively, on addition of 15 vol % tridecanol to 0.08 M Primene JM-kerosene, whereas thorium coefficients were virtually unaffected.

In stripping thorium from Primene JM with sodium chloride solutions, effective chloride utilization (4 to 5 lb of NaCl per lb of ThO_2) was obtained by using about 300% excess chloride for stripping and recycling about 75% of the strip solution (after precipitation of thorium) to the strip circuit. A second primary amine (1-nonyldecyl), however, showed poor response to this stripping method.

Continuous tests demonstrated the feasibility of stripping uranium from tertiary amines with ammonium sulfate solutions while adjusting the pH to 3.5 to 4 with ammonium hydroxide. The process yields an easily filtered, high-assay, sodium-free product with low reagent costs but has the disadvantage of relatively poor separations from molybdenum. (AEC Activity 5310)

Solvent Extraction Research. — Work has continued on primary amine extraction of small concentrations of uranium and plutonium from sulfuric acid dejacketing liquor that results from decladding (Sulfex process) of stainless-steel-clad fuel. It appears that a single-stage extraction (assuming 100% stage efficiency) with 0.3 M primary amine should recover 98% of the uranium and 95% of the plutonium from the sulfuric acid—stainless-steel-containing dejacketing solution. This conclusion results from calculations made with distribution coefficients obtained in multistage countercurrent extraction studies (ORNL-2795). The amount of sulfate carried into the stripping product from single-stage extraction is approximately 4.5 g per g of U. Previous countercurrent multistage extraction experiments have shown that >99.5% recovery of uranium and plutonium can be obtained with a sulfate concentration in the strip product of approximately 2.5 g per g of U. For recovery circuit application, single-stage extraction and stripping would be more economical and simpler to operate.

In thorium sulfate extraction by di-n-decylamine—sulfate—bisulfate in benzene, sulfuric acid is transferred back to the aqueous phase. Assuming that the extracted complex contains no bisulfate and that the presence of thorium does not alter the bisulfate content of the uncomplexed amine, measurements of such acid transfers indicated an average of slightly less than three amine sulfates per thorium in the complex, in good agreement with loading results.

Activity coefficients were estimated for the extraction reagents tri-n-octylphosphine oxide (TOPO) and di(2-ethylhexyl)phosphoric acid (D2EHPA) (ORNL-2737) in hydrocarbon solution by using isopiestic measurements and data from the extraction of $Fe(ClO_4)_3$ by D2EHPA. For 0.1 m D2EHPA in n-octane, $\gamma_{(D2EHPA)_2} = 0.5$ and $\gamma_{TOPO} \doteq 0.7$. The corresponding value for the isopiestic reference solute triphenylmethane (TPM) was $\gamma_{TPM} = 0.85$, which is the first estimate available of its activity coefficient in an n-alkane solution. (AEC Activity 5310)

Ion Exchange Technology. — The apparent diffusion coefficient of uranium into 1200-μ Dowex 21K resin from 0.004 M uranyl sulfate solutions was measured as 1.2×10^{-7} cm²/sec. This value is essentially the same as was measured with 0.0059 and 0.0029 M uranyl sulfate solutions. The self-diffusion coefficient of sulfate ions in uranium-loaded 1200-μ Dowex 21K resin was $(2.34 \pm 1.24) \times 10^{-7}$ cm²/sec. As was expected, this value is between the true self-diffusion coefficient of sulfate ions and the uranium self-diffusion coefficient.

Equilibrium studies of Duolite S-30, a phenol-formaldehyde resinous sorbent used commercially for decolorizing sugar solutions, show that it will sorb significant quantities of alkali ions from alkaline solution and that it sorbs cesium preferentially over sodium by a factor of about 150. The mechanism of sorption is presumably ion exchange with the weakly acid phenol group. The cation uptake from 0.1 M NaOH was 3.8 meq per gram of dry resin, and the uptake from 0.1 M CsOH was 4.7 meq/g. The distribution coefficient (weight basis) for cesium tracer in 0.01 M $NaNO_3$ increased rapidly with pH, from 750 at pH 9 to 5400 at pH 12.2. In 1.0 M $NaNO_3$ the distribution coefficient increased from 45 at pH 9 to 350 at pH 12.

Experience with Pm^{147} indicated that Dowex 50 resin was still usable after receiving a beta dose of 2400 whr/liter (4.7 whr per gram of dry resin).

Reactor coolant water studies, in connection with the NS "Savannah" program, indicated an optimum demineralization flow rate of about 30 gpm per cubic foot of resin. At higher flow rates the effluent resistivity dropped due to insufficient contact time. The drop in resistivity at lower flow rates is less easily explained but is possibly due to leaching of organic ions from the resin.

A concrete block prepared with simulated reactor demineralizer resin containing 10 mc of mixed fission products showed no detectable leaching of activity by synthetic sea water in 200 hr. (AEC Activity 5310)

Inorganic and Physical Chemistry: Thermochemistry and Calorimetry. — The low-temperature heat capacity of K_2ReBr_6 was measured from 10 to 300°K. A cooperative type of anomaly (second-order transition) in the heat capacity was found with a maximum at approximately 15°K. This anomaly, which is undoubtedly magnetic in origin, occurs at a considerably lower temperature than would be derived from the magnetic susceptibility. In addition, the compound exhibits an evolution of heat on warming at approximately 140°K and an anomalous heat capacity over the temperature range 210 to 260°K. (AEC Activity 5330)

Inorganic and Physical Chemistry: Chemistry of Technetium. — Spectrophotometric titration with $Ce(SO_4)_2$ showed that reduction of TcO_4^- in HCl

solution with zinc produces Tc(III); Tc(V) and $TcCl_6^{--}$ are produced as intermediates. At least two species of Tc(III) are obtained; one is more stable in higher acidities and has an absorption maximum at approximately 2650 Å in the ultraviolet spectrum, and the other is more stable in lower acidities and has a maximum at approximately 4200 Å. The maxima of both bands shift toward the red with increasing HCl concentration, and their height is approximately a linear function of the acidity. The Tc(III) is rapidly oxidized when exposed to air but is not readily oxidized by H^+ and can be obtained in 8 M HCl. (AEC Activity 5330)

Chemistry of Corrosion. — As part of the study of cathodic processes related to corrosion, the mechanism of reduction of oxygen on zirconium was investigated. The polarization measurements made in connection with the determination of the rate of growth of an oxide film gave good Tafel lines, and for both crystal-bar zirconium and Zircaloy-2 the cathodic process was first order with respect to the concentration of oxygen, except in certain special cases in which zero order was observed. Cathodic processes on zirconium were slow, when compared with rates on other materials. The kinetic results are consistent with a mechanism which postulates a barrier within the film. (AEC Activity 5330)

Nuclear Methods of Chemical Analysis. — The applicability of a radioactive tracer in the determination of the surface area of UO_2 powder by a surface-exchange method has been demonstrated. Preliminary results indicate that the method is rapid and easy and that it can be applied with a precision of approximately 10%. The only special requirements of the method are the availability of a suitable radioisotope tracer and radiation detection equipment. Advantages of this method are the speed and simplicity with which determinations can be made (approximately 2 hr for four samples). (AEC Activity 5330)

Fundamental Study of High-Temperature Chemical Reactions. — Completed studies of the phase equilibria in the system $CsF-ThF_4$ have shown the occurrence of seven intermediate compounds. Three compounds, $3CsF \cdot ThF_4$, $2CsF \cdot ThF_4$, and $CsF \cdot ThF_4$, melt congruently. Three others, $2CsF \cdot 3ThF_4$, $CsF \cdot 2ThF_4$, and $CsF \cdot 6ThF_4$, melt incongruently. The compound $CsF \cdot 3ThF_4$ is subsolidus at its composition, and $CsF \cdot 2ThF_4$ undergoes solid-state decomposition at low temperatures. Crystal polymorphism has been observed only in the compound $2CsF \cdot ThF_4$. Close resemblance of the systems $CsF-ThF_4$ and $RbF-UF_4$ is observed in the liquidus profiles and in the number and characteristics of compounds in the systems.

Vapor pressure measurements on $LiF-UF_4$ melts containing 10 mole % LiF were completed. The total vapor pressure of melts of this composition is given by the equation

$$\log P(\text{mm Hg}) = -(16550 \pm 150)/T - 7.046 \log T + 35.22 \pm 0.1,$$

which was obtained by a least-squares treatment of the experimental data, using an assumed ΔC_p of vaporization of -14 cal·deg^{-1}·mole^{-1}.

X-ray diffraction studies of the condensate of the vapor phase above $LiF-UF_4$ melts (10-90 mole %) have shown that the vapor consists of UF_4 and $LiF \cdot 4UF_4$. The presence of compounds in the vapor phase precludes obtaining the activities of the components of the melt from vapor pressure measurements. The vapor pressure of the melt is lowered more than would be expected from

Raoult's law, assuming UF_4 to be the only volatile species. The apparent activity coefficient is 0.93. (AEC Activity 5330)

High-Temperature Aqueous Solution Reactor Chemistry. (See AEC Activity 4151.1, ORNL-2825). — The investigation of the effect of varying the proportions of nitrate and sulfate on the liquid-liquid immiscibility temperature boundaries of the system UO_3-SO_3-N_2O_5-H_2O was extended to 0.16 m UO_2^{++} solutions and to 370°C. The data in the high-temperature range did not show the expected elevation of the immiscibility temperature and, in fact, showed a lowering of this temperature upon the initial substitution of nitrate for sulfate. An effort is being made to estimate the extent of loss of nitrogen oxides to the vapor phase; preliminary observations indicate that the lowering of the immiscibility temperature is only partially due to this loss.

The solubility information previously reported for the system UO_3-HNO_3-H_2O from 5×10^{-4} to 5 m HNO_3 and from 135 to 300°C was further substantiated by additional determinations. The ultimate goal of this investigation is to obtain solubility data accurate to approximately ±0.5% which can be used for theoretical evaluation of the system.

The solubility of CuO in HNO_3-H_2O solutions was determined at 200, 270, and 300°C from 0.02 to 0.6 m HNO_3 by direct sampling and subsequent analysis of the solution phase. For 0.2 m HNO_3 the saturation mole ratios, m_{CuO}/m_{HNO_3}, are 0.49, 0.47, and 0.46 at 200, 270, and 300°C, respectively, whereas for 0.02 m HNO_3 the corresponding saturation mole ratios are 0.44, 0.41, and 0.34.

Studies of the equilibrium $H_2O_2 + UO_2^{++} \rightleftharpoons UO_4 + 2H^+$ in aqueous solution and the associated reactions

$$H_2O_2 \rightarrow H_2O + 1/2\ O_2$$

$$UO_4 \rightarrow UO_3 + 1/2\ O_2$$

$$UO_3 + 2H^+ \rightarrow UO_2^{++} + H_2O$$

have qualitatively shown that the equilibrium constant K_c is of the order of 10^{-2}. The total peroxide concentration at the point of incipient precipitation of uranyl peroxide in acidic solutions of uranyl ion is not greater than 1×10^{-3} mole/liter in 0.03 M UO_2SO_4 and 0.015 M H_2SO_4 solution at 80°C. The rate of solution of solid UO_4 is rapid under conditions where solution can occur (i.e., before equilibrium conditions are satisfied). The total allowable peroxide concentration before precipitation of UO_4 occurs increases rapidly with increasing H^+ concentration. (AEC Activity 5330)

Chemical Separation of Isotopes. — In order to calculate equilibrium constants for isotopic exchange reactions, investigation of the vibrational frequencies of isotopic molecules was continued. The infrared spectra of gaseous $N_2{}^{15}O_4$ and $N_2{}^{14}O_4$, and the Raman spectra of these species in the supercooled liquid at -35°C were recorded. The 11 active fundamental vibrational frequencies were assigned and are presented in the table below. The data strongly support the planar N-N bonded structure for N_2O_4.

Assignment			Frequency (cm^{-1}) $N_2{}^{14}O_4$	$N_2{}^{15}O_4$
A_g	ν_1	(NO stretching)	1379.6	1360.2
	ν_2	(NO_2 deformation)	808	797
	ν_3	(NN stretching)	266	264
B_{1g}	ν_5	(NO stretching)	1712	1672
	ν_6	(NO_2 rocking)	482	478
B_{1u}	ν_7	(NO_2 wagging)	429	420
B_{2g}	ν_8	(NO_2 wagging)	672	654
B_{2u}	ν_9	(NO stretching)	1748	1707
	ν_{10}	(NO_2 rocking)	381	377
B_{3u}	ν_{11}	(NO stretching)	1262	1251
	ν_{12}	(deformation)	750	739

(u) infrared frequency; (g) Raman frequency.

(AEC Activity 5340)

METALLURGY AND SOLID STATE PHYSICS

High-Temperature Reactions. — As a part of a study of chemical species in solutions of metals in molten salts, preliminary measurements were made of the visible absorption spectrum of solutions of bismuth in molten bismuth chloride over the temperature range 260 to 360°C and at concentrations up to 0.7 M (ca. 6 at. %). Strong absorption bands were found at 560 and ca. 420 mμ. The effect of concentration on band intensity was such as to show that the dissolved bismuth was partitioned between two species and that the concentration of each changed as the over-all concentration of bismuth changed. (AEC Activity 5420)

THERMONUCLEAR PROJECT

DCX-1 Facility. — Most of the month has been devoted to the study of an anomaly that was first observed months ago; when an obstacle is introduced endwise into the trapped H^+ plasma in DCX, to the extent that it produces a 50% reduction in the signal observed on a detector looking at the charge-exchanged H^0 particles flying from the plasma, there is apparently not a concurrent reduction in the mean containment time of the plasma. One would think that the introduction of the obstacle (a carbon "paddle") would supply a new loss mechanism that would cause the plasma to live less long. The experiments have now been repeated in extenso, and the effect seems to stand. Explanations that have been suggested so far have had to be so artificial as to be almost unbelievable, and at the moment the situation is regarded as unresolved.

With the carbon arc off, and with only the residual gas in the vacuum system to dissociate the entering H_2^+ ions, mean containment times as long as 16 msec have now been observed. Under these conditions, the plasma does not spread as it does with the arc present. The spreading mechanism is therefore associated with the arc; without it the stability of the trapped H^+ plasma seems to be excellent, although the densities are still low, namely, 10^{10} to 10^{11} H^+ per cm^3. Calculations have shown that multiple Coulomb scattering of the H^+ from carbon ions in the arc is entirely inadequate to explain the extent of the plasma spread with arc present, during the mean H^+ residence times that are observed. (AEC Activity 5540)

DCX-EPA. — Experiments have been carried out with a view to studying the relative C^+, C^{++}, and C^{+++} concentrations in the carbon arc as the anode is withdrawn into the weaker-field region of the anode mirror coil and as the baffles are changed. Observations are made spectroscopically. The conclusions that have been drawn are as follows:

1. Anode baffles (which are required by the demand for differential pumping) are bad in that they reduce the mean charge state of the carbon ions.
2. A floating baffle is worse than a grounded one.
3. A grounded baffle is the least objectionable.
4. The positions of the anode remote from the center of the machine are favorable to production of the higher charge states. (AEC Activity 5540)

Ion Source Development. — Development of the von Ardenne source continues. It has been found that the anode plate, which is customarily steel (being part of the magnetic circuit), can in fact be replaced by copper without detriment to the output, so long as the nearby zwischen electrode is steel. The result is a net improvement, inasmuch as the cooling can be improved.

A total ion beam of 150 ma has been realized at an energy of 75 to 80 kev, using an arc current of 8 amp. Assuming molecular ions, the corresponding gas efficiency is 80%. Since that observation, a change in geometry indicates that 60 ma of ion current per ampere of arc current can be achieved, although it is presently impossible to see how far the total current can be increased, because of a limitation in the power supply currently available. (AEC Activity 5540)

Theory. — A complete statistical treatment of plasma particles and interacting electromagnetic radiation has led to the conclusion that the entropy of the electromagnetic field as well as the entropy of the plasma particles is a constant. The result has been used to demonstrate that only completely coherent radiation can be emitted by a plasma that obeys the Vlasov (i.e., the collisionless Boltzmann) equation.

The dissociation cross section for H_2^+ ions in the carbon arc has been calculated under the Born approximation, using exact molecular wave functions (ORNL-2766). Computations are in progress, using the analog computer, but complete results are not yet at hand.

The cross sections for charge exchange of the ions C^0 through C^{5+} have been calculated, using standard theory given by Brinkman and Kramer. The results have been distributed in the form of graphs.

The critical current required for burnout in an OGRA-like machine has been computed on the basis of theory recently derived here, using both Russian and United States cross-section values.

The derivation of the expression for the diffusion rate of plasma across a magnetic field has been generalized; the results agree with the previous calculations made at ORNL. (AEC Activity 5540)

PROGRAM 6000 — BIOLOGY AND MEDICINE

BIOLOGY

Genetic and Cytogenetic Effects of Radiation. — Complementation and genetic mapping of ad-3 mutants of Neurospora is being continued, using mutants of spontaneous, H_2O_2, diepoxybutane, ultraviolet, and x-ray origin. Since complementation tests on unselected samples of mutants showed that the most extensive complementation reactions are given by "leaky" mutants, several thousand ad-3 mutants are being screened for mutants of this type to expedite the complementation mapping. Present data indicate a minimum of ten cistrons and some suggestion of a nonlinear relationship. The analysis of intercrosses of complementing mutants has been started to determine the type of correlation that exists between complementation and genetic maps of this region. Such data should help provide a better understanding of gene-enzyme relationships, since it seems probable that the complementation map results from interaction in the cytoplasm between defective gene products at the ribonucleic acid or polypeptide level. (AEC Activity 6130)

Effects of Radiation on Paramecium. — Earlier work has shown that there is a sudden change in x-ray-induced mutation at about the midpoint of the interdivision interval in Paramecium aurelia. The amount of mutation induced by radiation is modifiable by posttreatments provided both the radiation and the posttreatment are given before the midpoint of the interval. Irradiation after the midpoint produces much less mutation than that before. Work in collaboration with T. O. Caspersson and his associates in Stockholm has shown by optical methods that synthesis of deoxyribonucleic acid (DNA) and other macromolecular components of the nucleus begins about midway through the interdivision interval and continues throughout the rest of the interval. It has now been shown by quantitative autoradiography that the uptake of tritiated thymidine into DNA likewise starts about midway through the interval and continues through the whole second half. Thus all evidence shows that the sudden change in the mutational process is associated with the onset of chromosome replication, not with its whole course. (AEC Activity 6130)

Insect Cytology and Genetics. — A method has been devised which allows for exposure of salivary gland nuclei to egg cytoplasm of various stages of development. Incubation of such nuclei (from Drosophila busckii larvae shortly before pupation) in cytoplasm from Drosophila melanogaster eggs before blastoderm formation (0 to 2 hr) resulted in the disappearance of the puffs X3 and IIL8 (approximate locations after the maps of Sirotina), while the puff in IIIL6 persisted. A very large puff in IIR22, a region which has never been found to puff in controls, could be seen in 56 out of 58 nuclei in which it was possible to identify this region positively. A smaller puff was induced at IIR17.

Incubation in cytoplasm from eggs at the peak of blastoderm formation resulted in the disappearance of X3 and IIL8 and the persisting of IIIL6. The puff at IIR22 was not induced in 16 out of 17 nuclei.

It is assumed that the puffing patterns induced correspond to a pattern of activity of loci characteristic for the respective stages in the early development of the egg. (AEC Activity 6130)

Biophysics. — The magnitudes of the radiation-induced electron spin resonances in polycrystalline glycine and in glycine molecularly adsorbed on silica gel were compared in two independent and carefully controlled experiments. The results of these experiments have corroborated previous exploratory studies indicating that the number of magnetic centers produced in the adsorbed molecule is markedly less than the number of centers produced in the pure crystalline state. This evidence supports the hypothesis that in crystalline amino acids the radiation-induced resonances do not arise entirely from true free radicals, but rather in combination with some type of lattice defect.

A calibration of the 200-curie Cs^{137}-Ba^{137} gamma-ray facility has been completed. The chronic Co^{60} gamma-ray source and the chronic polonium-beryllium fast-neutron source in Building 9743-2 have both been recalibrated. Although there have been various vague remarks in and out of the literature to the effect that the neutron decay of polonium-beryllium sources does not follow the exponential radioactive decay of polonium, the 95% confidence interval for the half life for neutron decay was found to be 138.0 ± 1.7 days over a period of ten half lives. This agrees very well with the last four precise determinations in the literature of the half life of Po^{210} as 138.4 days, with an uncertainty of no more than 0.1 day. A systematic method for the location of lost gamma-ray or neutron sources has been developed.

Escherichia coli B, after inactivation by ultraviolet irradiation in nitrogen, shows a recovery in the presence of pure oxygen immediately after irradiation. Detailed studies are in progress of the effect in strain B/r. It operates primarily at low survival levels (below 10% survival). Cells inactivated to 10^{-3} survival are brought up to 10^{-2} survival by this effect. It operates with pure oxygen, but not with air. It reaches maximum level after a few minutes of bubbling with pure oxygen. The oxygen may be applied as much as 15 min after irradiation, provided the cells are kept under nitrogen bubbling in the meantime. (AEC Activity 6130)

Radiation Protection: Living Cells. — In an effort to understand the position of the hemin block in the porphyrin-requiring mutant of *Escherichia coli* (H_7), studies were carried out in a synthetic medium. During the course of these investigations it became apparent that a pigment was being formed which had not been noticeable in the complex medium on account of the masking color of peptone. A large quantity of the pigment was obtained and concentrated. The deep-red pigment was partially purified by pyridine extraction. Further attempts to purify and characterize the pigment have been made and are contemplated.

Studies on the hydrogenlyase reaction in the porphyrinless mutant of *E. coli* showed that cells grown with hemin are not comparable with cells grown without hemin but to which the tetrapyrrole had been added in resting cell suspensions. The former had a pronounced hydrogenlyase activity, whereas the latter could not metabolize formate even in long-term experiments under conditions where adaptive formation of the enzyme has been

studied in other strains. In contrast with these findings the addition of hemin to intact resting cells causes the immediate appearance of catalase, though this activity becomes more pronounced on prolonged incubation. (AEC Activity 6230)

Mammalian Radiation Recovery. — Mouse bone marrow which had been slow-frozen in 15% glycerol and stored at various temperatures has been tested after six months of storage. Marrow stored at -30°C lost its protective ability. Marrow stored at -79 and -196°C retained its protective ability, although by four months the marrow stored at -79°C already showed signs of deterioration.

Of 157 (101 x C3H)F_1 lethally irradiated (950 r) mice given 100 mg each of isologous adult liver on days 0, 1, and 2 after exposure, 6.4% have survived beyond 30 days. This is a small but significant survival, since irradiated controls do not survive this x-ray dose.

It was found by the use of autoradiography that the radiation-protection compound S,2-aminoethylisothiourea (AET) localizes primarily in the cytoplasm of mouse bone marrow cells. Some of the chemical was also found in the cell membrane and nucleus. A protective effect of AET on recovery of intestinal esterases and on sodium and potassium equilibria of the intestines of lethally irradiated mice has been demonstrated.

Removal of solid food following a dose of 2500 r to the exteriorized gut was found to be a beneficial posttreatment. Treated mice lived about two days longer than controls. This technique should enable studies to be made of the later stages of x-ray damage to the gut and of the early stages of recovery.

When rat erythrocytes labeled with Cr^{51} are transfused into a rat-mouse chimera they have a life span somewhat shorter than that of rat red cells in the normal rat, but considerably longer than that of rat red cells in the normal mouse. However, when mouse erythrocytes are transfused into a similar chimera they usually show a normal mouse red cell life span pattern with a maximum of about 50 days. Occasional chimeras are found, however, that eliminate labeled mouse erythrocytes within a few days after transfusion. (AEC Activity 6230)

Radiation Immunology. — During the course of an investigation of the nature of antigen-antibody aggregates, a new method was devised such that the antigenic components detectable electrophoretically can be directly associated with those detectable by the serum agar diffusion method. One of the more interesting findings made with this method is that proteins having mobilities comparable with that of serum albumin are, in fact, large molecules that diffuse very slowly. Information of this nature should make possible purifications of immunochemically distinct proteins that previously appeared very difficult. (AEC Activity 6230)

PROGRAM 8000 — WORK FOR OTHERS

International Reactor Program at ORNL. — Twenty-five students currently enrolled in the foreign programs have completed 32 weeks of school since the beginning of classes in February. The nine Operations Supervision students have completed their classwork at ORSORT and are now assigned to a rotating shift at one of the Laboratory's reactors for three months. This course will terminate October 26, 1959. At this time, 16 Hazard Evaluation students will begin the final three months of study, which consists of the preparation by the student of a reactor hazard report covering one of four different types of reactor.

The Committee on Admissions met in August to select qualified applicants for the second session of foreign programs starting November 9, 1959. Twenty-six students representing 15 foreign countries have been selected and notified of their acceptance by the Division of International Affairs, AEC. (AEC Activity 8717)

ASEE-AEC Nuclear Engineering Institute. — Thirty participants in the 1959 ASEE-AEC Summer Institute, conducted by North Carolina State College, concluded the last two weeks of study with laboratory experiments at ORSORT. The ORSORT work consisted of seven experiments in radiochemistry and physics, tours, lectures, and movies. (AEC Activity 8736.3)

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