[redacted]Y

ORNL-2825
Progress

Cy. 207

OAK RIDGE NATIONAL LABORATORY
STATUS AND PROGRESS REPORT
AUGUST 1959

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

Downgrade to Unclassified After September 4, 1960

[redacted]

[redacted]

ORNL-2825

Copy No. 207

Contract No. W-7405-eng-26

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

AUGUST 1959

DATE ISSUED

SEP 4 1959

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

[redacted]

CONTENTS

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

August 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

Dissolver Solution Analyses. — The anion exchange separation procedure for plutonium (ORNL-2781) caused difficulty. Chemical degradation of the resin particles by nitric acid and subsequent leakage of the degradation products into the electrolyte solution caused the titration time to be lengthened and results to be low. Considerable effort was made to circumvent this difficulty, but without complete success.

The effects of fission products and stainless steel additives on the triisooctylamine-extraction coulometric-titration procedure for uranium were investigated. The procedure was altered to include adding ascorbic acid to the final aqueous solution to be titrated and adjusting the pH to approximately 4.0. Preliminary data show that accuracy and precision deviation is less than 1% when fission products and additives are present.

An anion exchange separation procedure for uranium is being developed which consists in adsorbing U(VI) from 6 M HCl onto 3 ml of Dowex 1-X8 (50–100 mesh), washing the column with 50 ml of the acid, drawing excess acid off the column by vacuum, and eluting the uranium with 15 ml of H_2O. The solution which comes off the column is approximately 2 M in chloride, a concentration which causes interference in the coulometric titration. The amount of chloride which can be tolerated and means of eliminating the interference will be studied. (AEC Activity 2724).

PROGRAM 4000 — REACTOR DEVELOPMENT

GAS-COOLED REACTOR PROGRAM

Experimental Gas-Cooled Reactor (EGCR) Construction. — The Advisory Committee on Reactor Safeguards has concurred in the location of the EGCR at the Gallaher Bend site, and authorization to proceed with construction has been granted. The Roehl Construction Company was awarded a contract for the site work and the road to the site. The sanitary water line to the site is being installed, and the clearing of the roadway has been started. (AEC Activity 4141.1)

Reactor Physics. — Calculations of the power distribution in EGCR fuel clusters in a plane perpendicular to the fuel axis are essentially complete for the case of fresh fuel in a symmetrical lattice. The effect of plutonium buildup remains to be checked, and the possibility of unsymmetrical distributions in the neighborhood of control rods and loops requires further study.

A study of heat transfer characteristics and power densities attainable in BeO- and graphite-moderated gas-cooled reactor cores was completed. Optimum fuel concentrations, specific powers, and other parameters of ceramic cores were determined. The results indicate that, in general, average core power densities in the range 50 to 100 kw/liter can be achieved with BeO if tubes or plates no thinner than 0.1 in. are used. With graphite tubes or plates 1/4 to 1/2 in. thick, power densities in the range 100 to 400 kw/liter can be achieved with moderate void fractions and modest pumping power. The complex relationships among the several variables describing the heat-transfer and heat-transport characteristics of these systems were compared in graphic form. (AEC Activity 4141.1)

EGCR Hazards Evaluations. — The pressure specifications for the EGCR containment vessel and the experimental loop cell were examined. It was determined that a pressure of 12 psig in the experimental loop cell would permit reasonable flexibility for future experiments. Calculations of the containment vessel pressure, based on a helium outlet temperature of 1050°F, yielded a value of 7 psig. No credit was taken in these calculations for the cooling by the steam generator and the heat capacity of the vessel and its contents. (AEC Activity 4141.1)

Heat Transfer Experiment. — Experimental heat transfer measurements were made on a system consisting of seven 0.800-in.-OD simulated EGCR fuel tubes arranged on 1.372-in. center-to-center spacings within a 3.835-in.-ID glass channel and constrained at the mid-plane to design positions by a spacer. Circumferential temperature profile data were obtained for an axial position 1 in. upstream of the mid-cluster spacer. In these experiments the over-all mean heat flux was 6500 $Btu \cdot hr^{-1} \cdot ft^{-2}$ with a flow rate of 2060 lb/hr (giving a Reynolds number of 68,000), an air inlet temperature of 108°F, and an average axial temperature rise of 68.7°F. The tube surface temperature profiles obtained were in qualitative agreement with those reported for the earlier resistance-heated-tube experiments in which no mid-plane spacer was used, and the circumferential asymmetry noted previously was again evident. The minimum tube surface temperature varied from 353 to 360°F, and the mean tube surface temperature varied from 364 to 375°F. The largest circumferential temperature difference was 34.3°F above the minimum surface temperature of peripheral tube No. 5.

An axial traverse was made of tube No. 4 under the somewhat different conditions of an over-all mean heat flux of 6700 $Btu \cdot hr^{-1} \cdot ft^{-2}$, an air flow rate of 2055 lb/hr (Reynolds number 67,600), an air inlet temperature of 118°F, and an average axial air temperature rise of 69°F. If the axial distance L is expressed in terms of the equivalent diameter d_e, the mean tube surface temperature increased from 248°F at the $L/d_e = 0$ position to 446°F at $L/d_e = 38.0$. At $L/d_e = 12.0$ (the mid-plane spacer is located at $L/d_e = 17.9$), the tube temperature was 367°F, while at $L/d_e = 21.0$ it was 313°F. (AEC Activity 4141.1)

Fuel Capsule Irradiations in the ORR. – Eight capsules have been irradiated continuously in ORR reactor cycles X through XIV. During cycle XIV, when the reactor was operating steadily at 16 Mw, subspecification temperatures were noted for capsules 6, 7, and 8. Investigations were started on capsule 7, which indicated two periods of unusual temperature oscillations, one of 140 min duration and the other of 220 min duration. Both temperature deviations started and stopped abruptly. Indications are that a shift of the inner tube by as much as 0.5 to 1 mil could cause such temperature changes. Further, the magnitude and duration of the temperature excursion could be affected by the proximity of the thermocouples to the inner tube spacer wire. It is currently postulated that a mechanical vibration or a thermal disturbance may have initiated movement of the inner tube with respect to the outer tube. This would alter the width of the thermal barrier gap and thus upset the thermal equilibrium of the inner tube. The alternate heating and cooling of opposing sides as the thermal barrier gas annulus widened and narrowed might have had an effect somewhat like that of a bimetallic strip. Such an action might be self-perpetuating until it was affected by another mechanical or thermal effect (for example, wedging). Efforts to induce the phenomenon by thermal cycling have been fruitless. A check for activity in the NaK blanket gas indicates that no leakage has occurred. (AEC Activity 4141.1)

Fuel Capsule Irradiations in the ETR. – Seven experimental fuel capsules are being irradiated in the ETR. The irradiation surface temperatures vary from 713 to 1305°F. (AEC Activity 4141.1)

Mechanical Properties of Type 304 Stainless Steel. – Creep tests are being conducted to determine the behavior of type 304 stainless steel in carbon monoxide, carbon dioxide, oxygen, argon, hydrogen, and air at 1300, 1500, and 1700°F. The data obtained thus far indicate that the creep rate is decelerated in air and accelerated in hydrogen at 300 psi and 1500°F. Under these conditions, CO and CO_2 appear to strengthen the specimens, with CO_2 having the greater effect. Argon and oxygen yield comparable, intermediate creep rates. Helium would have the same effect on the creep rate as that observed with argon. (AEC Activity 4141.1)

Beryllium Tubing Evaluation. – Approximately 28 ft of beryllium tubing obtained from England is being evaluated by nondestructive methods, which include resonance and pulse-echo ultrasonics, radiography, and fluorescent penetrants. The resonance ultrasonic technique has revealed wall thickness variations that fall within a range of between 0.035 and 0.045 in., with occasional variations in a short length in the range of 0.040 to 0.050 in.

Pulse-echo ultrasonic inspection has revealed the presence of several flaws, which have not yet been thoroughly explored. Radiography has detected numerous high-density particles and several pits, gouges, and longitudinal grooves. A few of the pits on the inner surface have diameters up to about 0.025 in., and they penetrate the wall as much as 75%. Thus far the radiographic technique seems to be the most definitive for determining the quality of the tubing. (AEC Activity 4141.1)

THERMAL-BREEDER REACTOR PROGRAM

Homogeneous Reactor Program

Homogeneous Reactor Test (HRT). — During August the HRT operated without interruption at pressures of 1150, 1250, 1600, and 1750 psig, at temperatures of 240 and 260°C, and at power levels as high as 5 Mw. More than 700 Mwhr were accumulated during the month, bringing the total for run 20 to ~1500 Mwhr. On the basis of approximately 30 hr at 4 to 5 Mw, it appears that a low system pressure (<1400 psig) is a solution to the instability difficulties heretofore experienced.

The investigations so far have been in the nature of scouting runs with the purpose of blocking out the areas where power operation results in loss of uranium. These investigations are complicated by several variables which are not completely under the operator's control. Nevertheless, the following can be concluded. There was a striking difference between certain of the system pressures selected for exploration: at 1150 and 1250 psig there were no indications of fuel instability at power levels up to 5 Mw; at pressures of 1600 and 1750 psig, however, there were rapid losses in reactivity at the higher power levels. There appeared to be little difference in the power levels at which fuel loss occurred at the two temperatures (240 and 260°C) investigated. The threshold for loss at the high pressures at either temperature was about 1.5 Mw. There were a few power excursions but none were greater than 13% of the operating power. (AEC Activity 4151.3)

HRT Chemical Pilot Plant. — Equilibrium removal rates established approximately 500 hr after the beginning of HRT run 20 were 60% higher with the 13-unit multiclone system than with the single hydroclone. The expected improvement by a factor of 3 was not achieved because of lower than expected hydroclone efficiencies. These efficiencies are very sensitive to particle size, decreasing by a factor of 2 in going from the 1-μ-size particle expected to the 0.8-μ particles obtained. Further improvement is being sought in present operation by recycling 70% of the hydroclone overflow stream back to the feed stream. (AEC Activity 4151.1)

Fuel Processing. — Development of procedures to remove nickel from solutions simulating concentrated HRT fuel solutions (80 g of U per liter, 15 g of Cu per liter, 1 M SO_4^{--}) was completed by test runs with both H_2O and D_2O solvent. Reduction of the platinum anode area increased the steady-state concentration of uranium reduced to U(IV). The amalgam was loaded to a maximum of 4.7 wt % Cu + Ni after the final run with H_2O. In the run with D_2O, the current efficiency for nickel removal was 1% and for copper was initially 57%, decreasing to a low value as the removal of this metal was completed. (AEC Activity 4151.1)

Thorium Oxide Slurry Development. — Review of radiolytic gas recombination data showed (1) an apparent activation energy of about 11 kcal for gas recombination in-pile in a heavy-water slurry of 650°C-fired Th-U oxide as opposed to 16 kcal for the light-water system; (2) a slightly greater dependence on temperature in the out-of-pile runs, indicating an apparent activation energy of 30 kcal for the heavy-water system; and (3) slightly higher gas recombination rates in heavy- than in light-water slurries.

A new palladium catalyst was prepared by refluxing 1600°C-fired ThO_2 in a $Pd(NO_3)_2$ solution (230 parts of palladium per million of thorium), reducing with H_2, and firing the resulting solid at 800°C. Under some conditions this catalyst had more than 10 times the specific activity of the Westinghouse preparation.

Initial gas recombination experiments with an apparatus for injecting measured quantities of hydrogen and oxygen into a slurry bomb at temperature were very successful. Water added by the recombination of hydrogen and oxygen was removed at operating temperature, and an essentially 100% material balance was demonstrated after more than 100 recombination experiments. In further experiments on cooling of a simulated slurry irradiation bomb by pumping water through a capillary wound around the bomb, temperature excursions of 40°C at 250°C and of 20°C at 300°C were controlled, but larger excursions were not.

Flame denitration of $Th(NO_3)_4$-$Al(NO_3)_3$ in methanol solutions with reflector temperatures of 1500°C gave 95 wt % ThO_2—5 wt % Al_2O_3 particles with a high degree of sphericity. The surface area of 1.3 m^2/g, the yield stress of 0.026 psf, and the mean particle diameter of 1.0 μ for unclassified product are a very promising combination for a blanket slurry. (AEC Activity 4151.1)

Heterogeneous Equilibria in Aqueous Systems (See AEC Activity 5330, ORNL-2795). — Concentrating a synthetic HRE-2 fuel solution lowered the temperature of formation of two liquid phases from 332°C for the unconcentrated fuel to a minimum of 305°C in the concentration range of 6 to 16 times that of the present fuel on a molal basis. Information obtained from recent HRE-2 operation suggests that with a fuel pressurizer temperature somewhat below 305°C it may have been impossible for concentration by boiling to have caused the separation of two liquid phases and the resulting aggravation of the fuel instability situation. In laboratory tests the minimum temperature for formation of two liquid phases by concentration was raised to above 324°C when the free acid concentration in the synthetic HRE-2 fuel solution was raised to 0.03 M H_2SO_4.

The two-liquid-phase temperatures of radioactive fuel samples taken directly from the HRE-2 were determined and showed little difference from those of nonradioactive solutions of the same concentrations.

The distribution of Ni^{++} and of Ni^{++} and Cu^{++} together between the light and heavy liquid phases of the system UO_3-SO_3-H_2O was investigated. Preliminary evaluation of these data indicates a distribution of both Ni^{++} and Cu^{++} in approximately the same relative mole ratio as uranium between the light and heavy liquid phases.

Preliminary investigations from 200 to 300°C of several separate portions of the system UO_3-CuO-NiO-N_2O_5-H_2O indicated that oxides rather than metal nitrates or basic nitrates are the stable solid phases. The solubility data obtained permit a preliminary evaluation of the usefulness of the nitrate system as a high-temperature aqueous homogeneous reactor fuel. (AEC Activity 4151.1)

Reactions in Aqueous Solutions. — The effectiveness of the cupric ion as a homogeneous catalyst in the UO_3-HNO_3-H_2O system was investigated in a brief survey. The activation energy for the process was found to be approximately the same as that observed in the UO_3-H_2SO_4-H_2O system. The catalytic activity of cupric ion in the nitrate system is only 40% as great as the activity in the sulfate system under similar conditions. Calculations show that a nitrate system containing 0.02 m $Cu(NO_3)_2$, 0.04 m $UO_2(NO_3)_2$, and 0.080 m HNO_3 can recombine the radiolytic hydrogen produced at a power density of 21.5 w/cc at a core nuclear average temperature of 280°C with a pressurizer pressure of 1430 psia and 100 psi of oxygen overpressure. This corresponds to approximately 10 Mw of power in HRE-2. (AEC Activity 4151.1)

Analytical Chemistry. — In the development of methods for the determination of osmium, two improved spectrophotometric methods have been devised. In one method, suitable for milligram quantities, the absorbancy of the tetroxide after extraction with chloroform is measured at any one of the five absorption maxima in the ultraviolet region. In the other method, for concentrations of a much lower order (0.5 to 30 μg/ml), the absorbancy of an intensely colored osmium-diphenylcarbazide complex is measured at a wavelength of 560 mμ. In the latter method, the complex can be formed either directly in chloroform or in an aqueous solution from which the complex can be subsequently extracted and estimated. The diphenylcarbazide method is the more sensitive, but fewer interferences are encountered with the tetroxide procedure. (AEC Activity 4151.1)

Slurry Circulation Experiments. — Circulation of a thoria slurry containing 8% uranium in the 200B loop at 280°C indicated the good particle stability found in previous tests with this type of slurry at 150 to 225°C. The initial mean particle size of 2.3 μ was virtually unchanged after 600 hr of circulation at a concentration near 500 g of Th per liter of slurry. The uranium dissolution from the slurry was no greater than that in the previous tests: approximately 2 ppm dissolved uranium was contained in the slurry supernate and less than 40 ppm was found in the pressurizer liquid. The heat transfer coefficients obtained with this slurry at 280°C and concentrations up to 700 g of Th per liter of slurry were very close to those measured with water. (AEC Activity 4151.1)

Slurry System Development. — After completion of water tests with the new 300-SM low-pressure system, a dump test was carried out to determine the system behavior. For this initial test the concentration in the high-pressure loop was first decreased to approximately 1 g of Th per liter by letting down slurry to the low-pressure system through a capillary choke while adding water with the feed pumps. In general the dump test was successful, with most of the system components performing as designed. The primary dump line, which drained from the top of the loop pipe, permitted transfer of the dilute slurry without difficulty. A secondary dump line, which drained from the bottom of the loop pipe, did not permit transfer of the dilute slurry because of settled slurry in the line between the loop pipe and the dump valve.

Preliminary slurry tests with the low-pressure system showed that slurry which had settled for one week in the conical-bottom 30-in.-dia storage tank could be resuspended by being boiled with jacket heating on the bottom of the tank. With a boiling rate of about 100 lb of steam per hour and with a slurry containing about 150 g of Th per liter in the tank, the concentration within the tank was uniform except near the bottom of the conical section. Surface heat transfer coefficients were approximately equal to those obtained with water. When the slurry in the tank was concentrated to about 850 g of Th per liter of slurry, the concentration decreased linearly with height, from 865 g/liter at the bottom of the tank to 800 g/liter at the slurry level. Transfer from the low-pressure system to the high-pressure system was effected without difficulty after minor changes in the feed pump suction line.

Slurry distribution in the 30-in. slurry core vessel is now being studied at 200 to 280°C and slurry concentrations up to 1000 g of Th per liter of slurry. (AEC Activity 4151.1)

Core and Blanket Vessel Development. - Tests with the HRE-2 core vessel model showed that solids which simulated those created by the underwater Heliarc screen cutting operation could be removed by flushing with water at 100 gpm. Solids which simulated those formed by the alternative electrolytic dissolution method for removing the screens could not be flushed through the lower screens.

In operation of the 30-in. slurry core vessel with slurry containing 800 to 1200 g of ThO_2 per kg of H_2O, uniform slurry concentrations were found at flow rates as low as 80 gpm at 25°C. Surface heat transfer rates on the conical surface of the vessel were approximately 20% lower than those measured with water at the same flow rates.

A spherical sampler probe containing seven radial openings gave sample concentrations with less than 2% variation for all orientations of the probe to the slurry flow direction. A heated thermocouple probe was found useful for detecting stagnant slurry regions within a vessel. (AEC Activity 4151.1)

Component Development. - A floating-bushing shaft seal, with nitrided titanium wear surfaces, was installed in the 300A slurry pump and operated successfully in high-temperature water. The test system was then loaded with ThO_2 slurry to about 1000 g of ThO_2 per kg of H_2O and was operated at 280°C to test further the shaft seal, mechanically mounted alumina bearings, and various impeller vane shapes.

Repair of the 200Z pump stator was completed, and pump performance and thrust tests were conducted. The 200Z impeller has vanes on the lower shroud to reduce the pressure drop across the lower wear ring, thereby reducing erosion and the probability of burning in oxygen. Effective vaned and vaneless diffusers were also performance-tested.

Two diaphragm slurry pumps were disassembled after 6200 hr of circulating ThO_2 slurries at pressures up to 1500 psi. One diaphragm was in excellent condition; the other diaphragm was covered with many "worm tracks," but appeared to be still usable. The aluminum oxide pump check valves had deteriorated virtually to failure in 3200 hr. (AEC Activity 4151.1)

Development of Remote Maintenance. - Development of tools for modifying the HRT core and for plugging the hole was concluded successfully, and the tools have been placed in standby. An underwater Heliarc cutting torch and manipulator was perfected for cutting screens into strips which could be removed through the 2-in. access to the core. An alternative electrolytic dissolution method was demonstrated in a full-scale model test. A flange reamer was developed for removing an interfering weld bead in the 2-in. access opening. A method of replicating a Dip Seal impression of the hole in the core, with rubber and epoxy castings, was developed but requires additional refinement to control surface contamination. (AEC Activity 4151.1)

Fuel Systems Development. - The HRT Mockup completed run CS-25, a 1050-hr run at 280 to 300°C which was intended to explore the consistency of inventory measurements. The run included two "excursions" in which heavy-liquid-phase UO_2SO_4 was produced in the pressurizer boiler. Loop inventories of U, Cu, SO_4, and excess acid were steady through initial operation and were completely recovered after the first excursion. Complete analytical reports of the last part of the run have not yet been received. (AEC Activity 4151.1)

Molten-Salt Reactor Program

Chemistry. - Further study of quenched samples from the $NaF-ThF_4-UF_4$ system showed that the compounds $3NaF \cdot 2ThF_4$ and $5NaF \cdot 3UF_4$ form a solid solution with an apparently continuous primary-phase field and that the liquidus along the solid solution line exhibits a point of inflection in descending from the melting point of $3NaF \cdot 2ThF_4$ at 712°C to 648°C for $5NaF \cdot 3UF_4$. The compounds have lower limits of stability, and the temperature below which the solid solution is unstable decreases almost linearly from 683°C for $3NaF \cdot 2ThF_4$ to 630°C for $5NaF \cdot 3UF_4$.

A check on the adsorption of xenon by bulk reactor-grade graphite was made in connection with the retention of fission products. Graphite (AGOT) having a surface area of 0.8 m^2/g adsorbed 0.036, 0.0017, and 0.0008 cc of xenon per square meter at -79, 0, and 30°C and pressures ranging from 14 to 62 mm Hg. At 80°C the amount adsorbed was below the sensitivity of the method.

Samples of a relatively impervious graphite were impregnated to an extent of 2 vol % by a ThF_4 (15 mole %) fuel under a pressure of 75 psia. Under approximately the same conditions AGOT graphite is permeated to 10 vol %, and because of the relatively high solubility of UO_2 in the fuel, no difference in permeability is noted whether or not a pretreatment of the graphite with hydrogen is employed to remove oxide.

AGOT graphite was about 2-1/2 times more permeable to fuels than TSF graphite (2% permeation) when compared at pressures of the order of 1 atm.

Samples of an impervious graphite which had been exposed to fuel in a corrosion test loop for one year have been analyzed. Fuel constituents, presumably transported by the vapor phase and fixed by reaction with surface oxides, were found in the bulk graphite, typically to an extent of 100 ppm for beryllium and 15 ppm for uranium; the uranium concentration in the center portions, more than 3/8 in. deep, was about sixfold higher.

Continuing analyses of periodic samples from an Inconel corrosion test loop showed further evidence of malfunctioning and contamination. The chromium concentration in the melt has risen to 0.54 wt % and the uranium concentration has decreased from 2.8 to 2.0 wt %, presumably because of precipitation as oxide. Other sampled loops, of INOR-8, have maintained low steady-state chromium concentrations of 300 to 500 ppm in the circulating fuel. (AEC Activity 4152.1)

Metallurgy. — The effect of grain size on tensile properties of INOR-8 was determined at room temperature, 1000, 1200, and 1500°F for specimens of two different grain sizes, coarse (ASTM 2) and fine (ASTM 5). The following conclusions are based on the tensile data obtained: (1) Fine-grained material has a higher tensile strength than coarse-grained material up to 1200°F. (2) Fine-grained material has a higher yield strength than coarse-grained material up to 1500°F. (3) Fine-grained material has poorer ductility, especially at 1200 and 1500°F. (4) Aging at 1650°F for 40 hr improves the ductility at 1500°F but does not seem to affect the tensile properties at lower temperatures.

Through metallographic examination it was found that an aging treatment at 1650°F for 40 hr produces carbide precipitation and coarsening, especially along the grain boundaries. At 1200°F and above, intergranular cracks begin to appear, and at 1500°F the fracture is entirely intergranular. Metallography reveals that the material tears in the vicinity of the carbides before fracture.

Permeability was determined for two grades of graphite (TSF and a low-permeability grade with 1.86 g/cc bulk density) by a salt having the composition 62 LiF–37 BeF_2–1 UF_4 (mole %) at 1300°F and a pressure of 150 psig. The TSF grade was found to have a permeability similar to that of AGOT graphite and showed a weight increase of 15%, which indicates the filling of approximately 50% of the accessible void space. The low-permeability grade showed a weight gain of only 4.8% and the filling of 26% of the accessible void space.

Metallographic examination has been completed on rod specimens of grade GT-123 graphite which were exposed for one year to a salt mixture (62 LiF–37 BeF_2–1 UF_4, mole %). No signs of deterioration or corrosion of the graphite were observed; however, radial cracks were found in both as-received and as-tested specimens. No fuel penetration occurred into either the cracks or existing voids. (AEC Activity 4152.1)

Corrosion Tests. — Six INOR-8 and four Inconel pumped loops have operated for more than one year. Five of the INOR-8 loops are still operating; the oldest one has operated for 14,300 hr.

Chemical and metallographic examinations have been completed on an Inconel pumped loop which operated for one year while circulating a salt mixture of 53 NaF–46 BeF_2–1 UF_4 (mole %) at a maximum hot-leg temperature of 1300°F and a cold-leg temperature of 1100°F. Attack was observed in the heated section of the loop in the form of void formation ranging in depth from 18 to 38 mils. The unheated portion of the hot leg and the cold area showed only slight pitting and no metallic deposits. Analysis of the salt revealed a large increase in the chromium concentration, to 4670 ppm, and a slight reduction in the uranium concentration. Uranium dioxide was found to an extent that indicated extensive extraneous oxidation of the salt. (AEC Activity 4152.1)

Pump Development. — A helically grooved bearing containing smaller grooves than those used in previous tests was operated in molten salt at 1200°F, 1200 rpm, and loads up to 300 lb. Failure occurred after a total operating time of 136 hr, the last 14 hr of which was at a 300-lb load. Another similar bearing is undergoing test at 1200°F; this bearing contains flow restrictions at the discharge (upper) end of the helical grooves. It has been tested at loads up to 500 lb at 1200 rpm. (AEC Activity 4152.1)

Remote Maintenance Demonstration. — The stress relieving of the piping in the remote maintenance demonstration system was accomplished by electrically heating all the loop piping and holding at 1500°F for 2 hr, then slowly cooling to room temperature.

Installation of remotely operated stereotelevision equipment is in progress. (AEC Activity 4152.1)

Heat Transfer and Physical Properties. — Measurements have been completed on the enthalpy, heat capacity, and heat of fusion of the BeLT-15 mixture (LiF-BeF_2-ThF_4, 67-18-15 mole %) over the temperature range 100 to 800°C. For the solid (100 to 400°C), the enthalpy (in calories per gram) can be expressed as

$$H_t - H_{30} = -6.5 + 0.190t + (6.5 \times 10^{-5})t^2$$

and for the liquid (550 to 800°C) as

$$H_t - H_{30} = -38.5 + 0.418t - (7.0 \times 10^{-5})t^2.$$

The heat of fusion at 500°C was 48 cal/g. (AEC Activity 4152.1)

Fuel Processing. — Solutions of nitrogen dioxide in liquid hydrogen fluoride are being investigated for molten-salt reactor fuel processing by the HF dissolution method. Nitrogen dioxide affects solubilities in somewhat the same manner as water when added to anhydrous hydrogen fluoride, in that it increases the solubility of beryllium fluoride. In a solution containing approximately 50% NO_2, the BeF_2 solubility was about 50 mg/ml, but the LiF solubility was only 4 to 5 mg/ml. In HF containing about 20% NO_2, the LiF and BeF_2 solubilities were 26 and 44 mg/ml, respectively. Uranium tetrafluoride had sufficient solubility in these solutions to permit complete dissolution of salt containing 3% uranium. Fission product analyses indicated that rare earths were relatively insoluble, 0.002 mg per milliliter of solvent, as they are in 90% HF. (AEC Activity 4152.1)

Thermal-Breeder Evaluation

Thermal-Breeder Evaluation Studies. — The GNU program for the IBM-704 has been selected for performing the primary nuclear calculations. It was established that the cost of the calculation with this code is approximately one-third of the cost of using the Oracle program Cornpone, and it was verified, by a series of calculations in which identical 34-group cross sections were used, that the results obtained from the two codes are substantially in agreement. Reactors moderated with graphite, D_2O, and H_2O and having 50% leakage into the blanket were included in the comparison.

For those reactors in which parametric studies are needed, the program Wanda (IBM-704) will be used with few-group constants prepared from the multigroup results.

In order to treat heterogeneous reactors, a multigroup method of calculating the equivalent "homogenized" reactor with the use of group disadvantage factors for fuel and moderator was developed. In the epithermal region the group disadvantage factors are evaluated from a multigroup diffusion-theory calculation (GNU) of a single heterogeneous unit cell (having zero leakage at the boundary) and are defined in such a way that, when used to multiply the cross sections of fuel and moderator, respectively, the "homogenized" reactor calculation substantially reproduces the neutron spectrum and balance obtained from the heterogeneous-cell calculation. For the thermal group the disadvantage factor is evaluated by means of a transport calculation in which the slowing-down density from the last epithermal group is used as the source. (AEC Activity 4160)

NUCLEAR POWER PLANTS

Small-Size Nuclear Power Plants. - It was determined by the AEC that the first power plant to be constructed in the small-size nuclear power plant program would utilize a pressurized-water reactor and have a gross electrical output of 16,500 kw. The participating cooperative, when selected, will have the option of adding a superheater, fired with fossil fuel, at its expense to improve steam conditions and to increase the electrical output to about 22,000 kw. The selection of the firm of Gibbs and Hill, Inc., of New York to act as architect-engineer for this plant was announced on August 27. A cost-type contract with the Oak Ridge Operations Office (AEC) is being negotiated.

The ORNL Advisory Group continued to furnish technical assistance to OROO on this program. Additional reactor design criteria were prepared and submitted to OROO for inclusion in an information package to be furnished to the architect-engineer. (AEC Activity 4160)

MARITIME SHIP REACTOR PROGRAM

Maritime Reactors. - An environmental study of the effects of operation of the NS "Savannah" reactor and the maximum credible accident (MCA) at the New York Shipbuilding Corporation dockyard at Camden, New Jersey, was completed. The conclusions of this study indicate the following:

1. The MCA with adverse weather conditions could result in a maximum total body dose of 200 millirems to individuals at the location of maximum concentration on the ground. The population exposure due to atmospheric dispersion for the MCA could be as high as 6000 man-rems.

2. If the MCA results in the introduction of the activity directly into the Delaware River, the maximum concentration in the river water will be less than two times the occupational maximum permissible concentration. For this same case the maximum submersion exposure will be less than 1 mr/hr.

3. The low exposure rates resulting from the MCA (which assume long-term operation of the reactor) indicate that it will not be necessary to place limitations on the normal power level and operation time at the initial startup. (AEC Activity 4510)

GENERAL REACTOR RESEARCH

Radiation Damage: Advanced Engineering and Development. - Basic beryllium acetate complex, $Be_4O(OCOCH_3)_6$, completely substituted with deuterium, has been suggested as a moderator material for nuclear reactors. The stability of this compound in the mixed radiation field of a reactor has been measured at room temperature. The irradiations were carried out at a flux of 6×10^{11} thermal neutrons$\cdot$cm$^{-2}\cdot$sec^{-1} in a water-cooled hole of the Graphite Reactor. Total exposures ranged up to 1.6×10^{18} neutrons/cm^2, corresponding to a total energy deposition of 2.9×10^{22} ev/g (4.6×10^8 rads) from fast neutrons and gamma radiation.

Measurements of the evolved gas provided an index of the radiation-induced decomposition. The yield, in molecules per 100 ev, was 0.41 for gas not condensed at liquid-nitrogen temperature, and the total gas yield was 0.52. The x-ray diffraction pattern did not change upon irradiation. The melting point decreased from the initial value of 287°C to 251°C at the longest exposure, 2.9×10^{22} ev/g. The decrease in melting point was linear with the exposure over the range covered. The gas evolution rate is less than half that of such organic polymers as polyethylene and polymethyl methacrylate but is an order of magnitude greater than the evolution rate of polystyrene. (AEC Activity 4202)

Power Reactor Fuel Reprocessing: Darex Process. — Twenty-five unirradiated simulated Consolidated Edison fuel pins were dissolved in succession in a titanium dissolver according to the Darex-Thorex flowsheet. Uranium losses to the 5 M HNO_3–2 M HCl decladding solution varied in a random manner from 0.1 to 0.5% over the 25 cycles. Dissolution of the 95.5% ThO_2–U_3O_8 core pellets (~80% of theoretical density) in 13 M HNO_3–0.04 M NaF–0.1 M $Al(NO_3)_3$ was complete at the end of each cycle. Dissolution of hydrogen-fired Consolidated Edison pellets in 13 M HNO_3–0.04 M NaF was complete in 2.5 hr. Dissolution times for the same dissolvent containing 1.0, 0.7, 0.4, or 0.1 M Fe^{+++} or 0.1 M Al^{+++} were 3.0, 3.5, 4.3, 5.7, and 7.9 hr, respectively.

Three methods were investigated for coagulating the silica contained in APPR-Darex dissolver solution; more than 96% of the silica in the solution was precipitated in all cases as a rapidly settling material containing over 50% SiO_2. The methods included (1) digestion of the raw dissolver solution with 12 g of Celite 545 per liter for 2 to 3 hr at boiling; (2) use of the same coagulant during the Darex standard chloride removal step; and (3) digestion at boiling with 0.004 M HF for 15 to 30 min, followed by cooling to 60°C and addition of 0.15 g of gelatin per liter or 2 g of albumin per liter. In all cases filtration of the silica was slow; however, centrifugation was satisfactory.

In order to determine whether NaK-bonded SRE-type fuel can safely be dissolved by the Darex process, five Darex dissolutions in 2 M HCl–5 M HNO_3 were made with simulated SRE fuel, a single stainless steel tube containing 5 cc of NaK (78% K, 22% Na) being used in each dissolution. The reaction of NaK with the acid during the first four dissolutions was accompanied by flashes of light and pressure surges. An explosion of sufficient violence to blow apart the Pyrex pipe dissolver occurred during the fifth dissolution, probably from a rapid gas-phase reaction between hydrogen and oxidizing gases such as NO_2.

Titanium corrosion tests for the Darex-Thorex system were continued for Consolidated Edison process conditions. The average corrosion rate over eight cycles in which the core oxides were dissolved in chloride-containing Darex decladding solution was 0.33 mil/month.

Welded titanium in starting, middle, and final Thorex dissolver solutions containing 100 ppm Cl^- and no Al^{+++} gave over-all maximum corrosion rates of 6.52, 0.49, and 0.18 mils/month, respectively, over 336 hr. Some pitting was observed in the initial and final solutions. In the absence of chloride, maximum over-all rates were 0.21 mil/month in the middle solution (1005 hr) and 0.21 mil/month in the final solution (840 hr). (AEC Activity 4340)

Power Reactor Fuel Reprocessing: Zircex Process. — Separation of uranium chlorides from $ZrCl_4$ by fractional desublimation at 300°C was demonstrated. A mixture of CCl_4 and HCl was used to remove all uranium from the hydrochlorination residue in the reactor to the 300°C desublimer. (AEC Activity 4340)

Power Reactor Fuel Reprocessing: Sulfex Process. — Twenty-five unirradiated simulated Consolidated Edison fuel pins were dissolved in sequence according to the Sulfex-Thorex flowsheet in a Ni-o-nel dissolver. Uranium losses to the 6 M H_2SO_4 decladding solution varied randomly from 0.1 to 0.9%. The 95.5% ThO_2–U_3O_8 (80% theoretical density) core pellets were not completely dissolved in any cycle; this effect is presumably due to contamination of the core dissolvent with sulfate. The average amount of undissolved oxide after two digestions was about 2%.

About 30% of high-density (~94% of theoretical) 95.8% ThO_2–$UO_{2.3}$ pellets was dissolved in 1 hr in a 200% excess of boiling 13 M HNO_3–0.04 M NaF–0.1 M $Al(NO_3)_3$. The amount dissolved in 1 hr decreased to 14 and 8% when the sulfate concentration in the dissolvent was 0.05 and 0.1 M, respectively.

A small-engineering-scale trial Sulfex run was successfully made in which nearly all of a dummy 11-plate stainless steel fuel assembly weighing 7.75 kg was dissolved in boiling 6 M H_2SO_4 at an approximate rate of 83 g of stainless steel per minute. Hydrogen was evolved into the nitrogen gas blanketing the system at about 1.28 cfm (STP). Foam in the 10-in. dissolver reached a height of 15 ft. (AEC Activity 4340)

Power Reactor Fuel Reprocessing: Zirflex Process. — The modified Zirflex process is being studied as a method for dissolution of U-Zr alloy fuels in stainless steel equipment. By adding ammonium nitrate or hydrogen peroxide gradually to oxidize uranium and tin as rapidly as the zirconium was dissolved in refluxing ammonium fluoride solutions, up to 7% U–Zr and 7% U–Zr–Sn alloys were dissolved completely. Uranium concentrations four times higher than those produced in hydrofluoric acid dissolution were obtained in 1 to 2 hr. No precipitation occurred at any time.

The solubility of zirconium in the core dissolvent nitric acid was 3 M at 2 M HNO_3 and 0.9 M at 6 M HNO_3 and above.

In Zirflex cyclic corrosion tests, welded specimens of Ni-o-nel and of type 309 SCb stainless steel corroded at maximum rates of less than 2 mils/month over eight and ten cycles. After three successive exposures of 18 to 24 hr each to boiling Perflex dissolvent (1 M HF–0.06 M H_2O_2), Hastelloy C showed relatively constant rates between 3.9 and 4.7 mils/month. The corrosion rate of Ni-o-nel in refluxing 6 M NH_4F–1 M NH_4NO_3 was 4 to 6 mils/month. (AEC Activity 4340)

Power Reactor Fuel Reprocessing: Mechanical Processing of Fuels. — Materials-handling flowsheets were completed for decladding, washing, recanning, and storing spent SRE uranium fuel slugs and for shearing and leaching stainless-steel-clad UO_2 and UO_2-ThO_2 fuels.

Shearing tests on stainless-steel-clad, porcelain-filled prototype fuel with eight different blades in combination with three different anvils were completed both locally and under two subcontracts. The conclusions from these tests were that (1) optimum initial clearance between blades is 1 to 3 mils, (2) rounding of the cutting edge of the shear blade to a 0.03- to 0.05-in. radius does not impair cutting efficiency and minimizes chipping of the cutting edge, (3) the combination of a stepped cutting blade and a concave anvil produces the best cuts and break-out of ferrules for minimum tonnage, and (4) a blade hardness between Rockwell C54 and C56 appears optimum; hardness above C56 results in cracking of the blades; hardness below C54 results in excessive wear, galling between blades, and breakage of the cutting edges when shearing through Nicrobraz 50, which has a Rockwell hardness of C51. (AEC Activity 4340)

Power Reactor Fuel Reprocessing: Processing of Graphite Fuels. — Leaching of high-density 4% uranium—graphite fuel with two portions of boiling 15.8 M HNO_3 resulted in 99.6% recovery of the uranium from -16 +30 mesh samples. Metal recoveries of about 92% were obtained when -200-mesh samples of General Atomic 1.54% uranium—7.18% thorium—graphite fuel were leached with either 15.8 M HNO_3 or 15.8 M HNO_3—0.04 M NaF—0.04 M $Al(NO_3)_3$. Recoveries were slightly lower with standard Thorex dissolver solution containing 13 M HNO_3—0.04 M NaF—0.04 M $Al(NO_3)_3$.

Fuel specimens containing 9% uranium in graphite disintegrated when immersed in liquid bromine at 25°C for 3 to 5 hr, but specimens containing 0.7% uranium did not disintegrate after 23 hr. A specimen of the General Atomic fuel also did not disintegrate. Leaching the powder produced by soaking the 9% uranium—graphite fuel in bromine with two portions of boiling 15.8 M HNO_3 resulted in about 99.8% recovery of the uranium; the same recovery was obtained by mechanically grinding to -4 +8 mesh. (AEC Activity 4340)

Power Reactor Fuel Reprocessing: Solvent Extraction Studies. — Scrubbing irradiated solutions of tributyl phosphate (TBP) in Amsco with aqueous sodium chloride shows promise as an analytical method for determining the amounts of nitric acid, dibutylphosphoric acid (DBPA), and monobutylphosphoric acid (MBPA) in irradiated solvent. By the contacting of TBP-Amsco solutions containing nitric acid, DBPA, and MBPA with three successive equal volumes of 25% aqueous sodium chloride solution, 99+% of the nitric acid was recovered in the aqueous phase and 95% of the DBPA in the organic phase; about 20% of the MBPA was transferred into the aqueous phase. (AEC Activity 4340)

Fuel Element Development. — The experimental aluminum fuel element for the ORR, containing 1000 g of U_3O_8 (20% enriched in the U^{235} isotope), was examined visually in the ORR hot cell. This element, which was subjected to 52% burnup of the U^{235} atoms, exhibited no observable defects. Manufacturing of this type of fuel element for experimental loadings in the Puerto Rico Nuclear Center Research Reactor has been initiated. (AEC Activity 4420)

Waste Disposal Research and Engineering: Geochemical Studies. — The sorption of cesium from 6 M $NaNO_3$ solutions by Wyoming bentonite was increased by heating the bentonite to approximately 600°C. Ion exchange determinations showed that the exchange capacity remained constant up to 500°C for calcium-saturated samples and up to 600°C for sodium- and potassium-saturated samples; cesium sorption up to these temperatures also remained relatively constant. At 600°C for the calcium bentonite and 700°C for sodium and potassium bentonites, the exchange capacity of the clays decreased to about 50% of the original values; it was at these temperatures that cesium sorption increased. Further heating decreased the exchange capacity even more, and cesium sorption was found to be reduced from the maxima. These data emphasize the importance of the structural factor and the relative unimportance of the total capacity of the clay for the selective sorption of cesium from waste solutions. On the concept that cesium sorption is favored by the 10-Å (001) spacing, potassium bentonite appears to be the most stable of the three forms of bentonite and calcium bentonite the least stable. (AEC Activity 4452)

Waste Disposal Research and Engineering: Disposal in Natural Salt Formations. — The synthetic wastes and test equipment for the 7.5-ft and 18-in. cavities have been delivered to the mine in Hutchinson, Kansas, for installation. A chemical laboratory has been installed in the mine, and personnel have been assigned to Hutchinson for the duration of the experiments.

The 7.5-ft cavities have been found to be hydraulically interconnected. Tests to determine the rate of flow through the salt are being made. The flow is believed to be due to the uplifting of the floor. The uplift is caused by the increased stress on the pillars resulting from the room excavation and will, therefore, tend to occur in most mines.

Final tests of components were carried out in a 10-in. cavity in a 24-in. salt block. Components tested and found to be effective are Cerrolow 117, a low-melting alloy for power connections, and a solid gas seal consisting of sand, silica gel, and charcoal.

An equation was derived which gives the vertical power distribution in the liquid in the salt cavities. The electrode resistance should not affect the temperature distribution in the 7.5-ft experiment with neutralized wastes but may produce 10 to 20° top-to-bottom differential in the acid-waste experiment. (AEC Activity 4452)

Waste Disposal Research and Engineering: Disposal of ORNL Radioactive Wastes. — To define the character of the overburden, the depth to ground water, the configuration of the water table, and the movement of radionuclides in the existing burial ground, 20 auger wells ranging in depth from 5 to 20 ft were completed. Future requirements of the new burial ground are based in part upon the volume of solid waste anticipated. From analyses of records of burial ground operations at ORNL since 1955, an additional 2.2×10^6 ft^3 of solid waste is expected through 1964. The estimated volume of waste consists of 1.2×10^6 ft^3 of alpha-contaminated material and 1.0×10^6 ft^3 of beta- and gamma-contaminated material.

In Bear Creek valley, four core holes 200 ft deep and ten churn-drilled wells ranging in depth from 53 to 157 ft were completed.

The projected operations of the Fission Product Pilot Plant (FPPP) will require the storage of wastes derived from the processing of young fuel elements. These wastes are expected to be high in Zr^{95} and Nb^{95} and may produce elevated temperatures in the tanks. Therefore emission of radioactive aerosols from the tank liquids may result.

In a preliminary study, the wastes were characterized by ionic and radiochemical analysis of a tank sample, a study of FPPP unit processes, and a scanning of FPPP operational logs for pertinent analytical data. An exploratory series of laboratory batch runs of neutralized simulated Purex waste at varying temperatures and air sweep rates was completed. By utilizing iron and nitrates as macrochemical indicators, it was found that entrainment did occur with a threshold at temperatures of about 90°C and that sweeping with air at less than 1 liter/min per square foot of liquid surface increased the carry-over several fold. Use of a radioactive tracer (a mixture of Fe^{55} and Fe^{59}) during three runs provided useful clues on the stratification of supernatant and solid zones. (AEC Activity 4452)

Waste Disposal Research and Engineering: Process Waste Water Treatment Plant Studies. — Weekly composite samples from the process waste water treatment plant were analyzed for calcium hardness and total hardness as well as radiochemically for strontium and total rare earths. The percentage removal of strontium and of total rare earths was graphed as a function of the residual calcium in the plant effluent. As the residual calcium increases, the percentage removal of strontium decreases, and the apparent relationship is expressed as a coefficient of correlation of -0.74; the 95% confidence limits are -0.57 and -0.86. Similarly, the coefficient of correlation between residual calcium and total rare earths is -0.66, with 95% confidence limits of -0.45 and -0.79. The real meaning of these relationships can be determined only after the mechanism of strontium and total-rare-earth removal has been established. Since a calcium determination can be made rapidly, it may continue to serve as a simple test to control the treatment process. (AEC Activity 4452)

Waste Disposal Research and Engineering: Clinch River Studies. — A series of river observations are being made in order to obtain further data on the dispersion of contaminants in the Clinch River and to define the factors that influence the distribution and concentration of radionuclides downstream. With the services of the AEC Radiological Physics Fellows and the cooperation of the U.S. Geological Survey, background radiation, temperature, and velocity measurements were made above and below the confluence of White Oak Creek and the Clinch River in preparation for the expanded river study program. Further analysis of the data from the earlier tracer tests has shown that under the river conditions at the time of the tests complete dilution was obtained at Clinch River Mile (CRM) 15.5, 5.3 miles below the mouth of White Oak Creek (CRM 20.8). A water level recorder was installed at CRM 20.7 to correlate the flow from Norris Dam and the water elevation at Watts Bar Dam. (AEC Activity 4452)

Waste Disposal Research and Engineering: Long-Range Evaluation of Over-All Waste Complex. — Contamination of the Ohio River following an assumed failure of a hypothetical high-level waste storage tank was investigated under several assumptions regarding the activity distribution within the ground-water reservoir. With the leakage rate from the ground-water reservoir assumed to be constant, the concentration and persistence of the contamination in the river were found to depend on the mixing characteristics of the waste within the ground water. Concentrations of the order of 10^4 times the maximum permissible concentration were found to result from leakage rates of 1% per day, a value which is high but not unreasonable. Diffusion within the ground water would tend to reduce the river concentrations but increase the persistence.

One of the more severe hazards thought to be associated with a major tank failure within the first few hundred years following the filling period would result from local heating. The sorptive characteristics of the soil would tend to retain a major portion of the waste within the area adjacent to the tank. The lack of artificial cooling would permit large rises in the temperature of the geologic structures containing the waste. Preliminary calculations indicate that local melting and possibly fission product vaporization would occur. (AEC Activity 4452)

PROGRAM 5000 — PHYSICAL RESEARCH

ISOTOPE PRODUCTION

Radioisotope and Stable Isotope Production. — The collections of the isotopes of neodymium, nickel, and copper were completed. Radiogenic lead enriched in Pb^{206} is being processed in the XAX channels. The isotopes of germanium are being collected in the XBX calutrons.

In addition to the calutron feeds for the isotope collections of neodymium, nickel, and copper, 8.5 kg of radiogenic lead enriched in Pb^{206} was separated from 20 kg of ore concentrate. This refined lead was converted to the chloride, which is the charge material being used for the collection of high-purity Pb^{206}.

New inventory lots of Gd^{152}, Lu^{176}, Mg^{25}, Rb^{87}, Sr^{84}, Sr^{86}, and Sr^{88} and returned lots of Ba^{135}, Cu^{65}, Nd^{145}, Pb^{208}, Sb^{121}, Si^{29}, Sm^{144}, and Sm^{148} were prepared for inventory following chemical refinement and spectrochemical and mass analyses.

Special service projects conducted include the conversion of $H_3B^{10}O_3$ to B_2O_3, preparation of $Li_2^6CO_3$ from lithium metal, and preparation of solutions of $D_3B^{10}O_3$ by dissolving known amounts of B_2O_3 in D_2O. Target work included 14 conversions and the preparation of 11 targets.

Radioisotope Research and Development. — The processing scheme in the di(2-ethylhexyl)phosphoric acid (D2EHPA) system in the Fission Product Pilot Plant was revised to include an extraction and stripping operation on the trivalent rare earths, which eliminates the major quantities of silica and manganese prior to the oxalate precipitation and thereby increases the filtration rate of this precipitate.

The first-order rate constant for cerium oxidation-extraction was determined to be 2.125 min^{-1} in 1 M D2EHPA and 0.536 min^{-1} in 0.5 M D2EHPA. The rate of oxidation is thus directly proportional to the square of the D2EHPA concentration. A clue to the mechanism of the reaction is indicated by the formation of a yellowish-white solid in the aqueous phase during the extraction of Ce(IV), which is extracted slowly into the D2EHPA. Preliminary investigation of the structure of this intermediate compound with Ce^{144} tracer indicated a ratio of one cerium atom to one D2EHPA molecule.

Temporary equipment was installed in Room 10 of Building 3026 to test the pyridine extraction procedure for purification of a technetium fraction from the Fission Product Pilot Plant.

The production of P^{32} was transferred to the ORR after three test production runs produced material that met product specifications.

The new thermal diffusion columns for neon isotope separation produced 425 standard cc of 59.0% Ne^{22}. (AEC Activity 5100)

Stable Isotope Development. — The sputtering ratio and sputtering angle of both target and incoming ion species are being measured at various ion energies as part of an isotope retention program. Past separation data are being analyzed in an attempt to correlate further the properties of the elements with the isotope retention and purity realized.

Receiving plates have been designed and fabricated for use in the calutron to deposit C^{13} directly upon a copper backing. This procedure, if successful, will eliminate the need for refinement of the isotope and its subsequent deposition upon the copper target.

As a preparatory step toward the future separation of osmium, a program has been established to study the properties of osmium tetroxide — the formation, transport, stability, solubility, etc. — in conditions simulating as nearly as possible those present in calutron operations. Results so far indicate that this compound may be useful as a charge feed if concentrations in work areas can be held below toxic levels. (AEC Activity 5100)

PHYSICS

Heavy-Particle Physics. — The elastic scattering cross section of nitrogen from aluminum is being measured by a coincidence method. Measurements at 2° intervals have been made from 50 to 136° in the center-of-mass system. The differential cross section exhibits a smooth drop-off with angle.

New measurements were made on the cross section for the radiative capture of nitrogen by phosphorus. In the course of these experiments an activity with an 8-min half life was observed and has been tentatively assigned to a new isotope of titanium, Ti^{42}, produced by the reaction $P^{31}(N^{14},3n)Ti^{42}$.

Germanium barrier counters for heavy-ion detection are being developed. Energy resolution better than 3% for 5.3-Mev alpha particles has been obtained. A significant portion of this 3% is due to preamplifier noise, indicating that better resolution is attainable. (AEC Activity 5220)

The 86-Inch Cyclotron: Nuclear Physics. — A time-of-flight system was installed for measuring neutron spectra from (p,n) reactions. The natural r-f bunching of the cyclotron beam is used to obtain short proton bursts. To increase the time available for measurements, seven out of eight bursts are eliminated by r-f sweeping of the beam. Neutrons are detected with a plastic scintillator, and zero-time pulses are obtained from protons striking another plastic scintillator. The peak due to target gamma rays triggering the neutron counter has a full width at half maximum of 3.5×10^{-9} sec. Much of this width is believed to be due to the width of the proton bursts. With this system and a 7-m flight path, neutron peaks due to the ground state and known excited states at 4.58 and 7.18 Mev were observed from the reaction $Li^7(p,n)Be^7$. It appears that the resolution of the present system is adequate for finding levels in some light nuclei. (AEC Activity 5220)

The 86-Inch Cyclotron: Nuclear Chemistry. — The fission of Ra^{226} by 11- and 20-Mev protons is being studied by catching the fission fragments on foil stacks and chemically separating and counting certain fission products. Preliminary results show the following: (1) The yield ratio Pd/Sr at 20 Mev is about twice that at 11 Mev, indicating that symmetric fission predominates at the higher energy, as expected. (2) The energy of the strontium fragment seems to be slightly lower at 20 Mev than at 11 Mev, while the palladium fragment energy is slightly higher at 20 Mev than at 11 Mev; this may indicate an increase in the relative number of highly excited fissioning nuclei leading to palladium (symmetric fission) as the proton energy is increased. (3) The total fission cross section is approximately three times as large at 20 Mev as at 11 Mev. (4) Neutron-induced fission is negligible with the present experimental setup. (AEC Activity 5220)

The 86-Inch Cyclotron: Applied Physics. — The isotopes Mn^{54}, Ge^{68}, As^{74}, I^{124}, and Bi^{206} were produced in service irradiations for outside customers, and Tc^{95m} was produced for the Isotopes Division. The Mn^{54} was produced on a target consisting of 1.3 g of Cr^{54}. It was run for 20 hr at an average current of 1735 μa. (AEC Activity 5220)

Spectroscopy Research Laboratory. — The spectra of rare-earth compounds burned in an oxyhydrogen flame have been photographed to furnish wavelengths for calibration of the ORNL flame photometer. Rough preliminary measurements have been made on these spectrograms, and tentative identification has been made of band systems in LuO, PrO, and GdO. In LuO, the system published by Watson and Meggers [J. Res. Natl. Bur. Standards 20, 125 (1938)] has been verified and slightly extended by the identification of the bands 1-0, 10-9, and 0-1. In addition, tentative identification of two new systems having 0-0 bands at $\sigma = 24406\ cm^{-1}$ and at $19162\ cm^{-1}$ has been made. In PrO, the system published by Watson [Phys. Rev. 53, 140 (1938)] has been verified and slightly extended by identification of the band 3-2, and a new system with 0-0 band at $\sigma = 11794\ cm^{-1}$ has been tentatively identified. In GdO, the systems I and II of Piccardi [Gazz. chim. ital. 63, 887 (1933)] have been verified and slightly extended. In the dysprosium spectrum, there appears to be a band system with 0-0 band at $\sigma = 18996\ cm^{-1}$ and of unknown origin. (AEC Activity 5230)

Electronuclear Machines. — The magnetic field design for the eight-sector spiral-pole electron cyclotron (Cyclotron Analogue II) is nearly completed. Conversion of the shape specification of the sector coils from a series of straight-line segments to circular arcs is being accomplished with only minor deviations from the desired flutter and spiral angle. Preliminary examination indicates that the average field can be fitted to within 2% by a simple array of circular coils. The close-spaced set of circular coils near the median plane will reduce the error to about 5 parts in 10^5. Mechanical design of the vacuum tank and pumping system is about 15% complete. (AEC Activity 5240)

Relativistic Isochronous Cyclotron (ORIC). — The preparation of title II drawings and specifications for the ORIC building and facilities is proceeding on schedule; Catalytic Construction Company is the architect-engineer. The work should be ready for bidding before November 1. Specifications for shielding doors were completed and invitations were issued for bids.

Design of the components for the cyclotron itself is also on schedule. The vacuum and water systems are essentially complete. Specifications for the power supplies for the sector coils, harmonic coils, and trimming coils were completed and are out for bid. Detailed preliminary instrument and control systems and the console design have been worked out. The ion source probe and vacuum lock are being detailed. The basic features of the resonator have been established, and the features of the movable shorting bar are being detailed. Design of the magnetic field will be implemented with the quarter-scale and full-scale magnet measurements.

The ORIC magnet is installed in a temporary structure adjoining Building 9204-3. The two main coils (9 tons of aluminum) were wound in the Y-12 shops and have been attached to the 200-ton yoke. The design for the pole-base disks was completed and a contract placed for their fabrication. A positioning system with an 80-in. disk is being fabricated for full-scale precision measurements of the magnetic field.

Installation of the quarter-scale magnet is complete, the 30-in. gear for precision measurements was received, and preliminary field measurements were begun.

Special Separations. — The separation of 500 mg of high-purity U^{234} has been completed in response to a recent AEC request. Ion currents equivalent to 0.667 g were collected. Two of the samples assayed 97.53% and 97.81% U^{234}, respectively; a third sample is awaiting assay.

The separation of approximately 2 kg of U^{233} containing only 5 to 20 ppm U^{235} has been completed. The product is being refined and charge feed is being recovered.

The third plutonium run of a series initiated to provide 300 mg of Pu^{241} produced ion currents equivalent to 0.0649 g of Pu^{241} and 0.479 g of Pu^{240}. Feed for this separation was 50 g of $PuCl_3$ (30% Pu^{240}) prepared from the recovered material of a previous plutonium run.

Eight shipments of uranium and three of neptunium were made. Plans and equipment are ready for the separation of feed for high-purity samples of Pb^{204} and Pb^{208}. (AEC Activity 5250)

Mass Spectrometry and Related Techniques. — Ion-molecule reactions involving low-energy (0—0.15 ev) CHN^+ and $C_2H_3N^+$ were shown conclusively to follow a hydrogen atom abstraction mechanism of the type

$$C_2H_3N^+ + H_2 \rightarrow C_2H_4N^+ + H$$

These reactions are independent of temperature and have bimolecular rate constants 10^3 to 10^5 times larger than are observed in the hydrogen atom abstraction reactions of free radicals. On the basis of this study, a definite correlation was made between the chemical reactivity of ions having unpaired electrons (ion radicals) and the behavior of free radicals. (AEC Activity 5250)

CHEMISTRY

Chemical Engineering Research: Aqueous-Organic Interface Resistance. — To study the kinetics and mechanisms of solvent extraction processes, the resistance of an aqueous-organic interface to molecular diffusion is being studied. The concentration gradients of uranyl ion are determined by photographically recording the transmittance of monochromatic (436 mμ) light through the test solution at an angle normal to the assumed direction of the concentration gradient in a glass diffusion cell. Owing to the complementary nature of Beer's law and photographic sensitometer relations, the concentration of uranium at any point in the diffusion cell is a linear function of the logarithm of the density of silver on the exposed photographic plate. (AEC Activity 5310)

Chemical Engineering Research: High-Speed Liquid-Liquid Contactor. — A high-speed liquid-liquid contactor was assembled which consists of a series of hydroclones stacked one above the other, with a pump for each hydroclone stage to provide recirculation. An induced flow of light organic phase has been made to move in a direction opposite to the flow of heavy aqueous phase down the walls, utilizing the induced underflow effect observed during the development of hydroclones for the aqueous homogeneous reactor. A vertically stacked three-hydroclone system was operated satisfactorily with kerosene and water, with the water phase continuous. With kerosene and benzoic acid solutions the systems operated with 100% stage efficiency. (AEC Activity 5310)

Chemical Engineering Research: Transpiration Corrosion Protection. — A porous wall may be protected from the attack of a corrosive medium by a film of noncorrosive material transpired through the porous wall. In this controlled film the concentration of the corrodent will be less than in the bulk. To measure the effects of transpiration flow velocity (at various bulk solution velocities) on the concentration of a corroding ion in the film at the porous surface, a small silver—silver chloride electrode was placed exactly at the surface of a fritted glass disk through which pure water could be forced into a surrounding solution containing chloride ion (0.1 M KCl solution); the bulk KCl solution could be agitated by a variable speed agitator. By measuring the Ag-AgCl cell potential at various conditions of transpirational flow and bulk solution velocity, the variation of the concentration of chloride in the film adjacent to the porous surface was determined. Good correlation of variables has been obtained which indicates that (1) at any given stirrer speed (i.e., bulk solution turbulence) the concentration of chloride at the wall falls rapidly to an almost constant value as the transpirational velocity increases to 5 to 10 cm/min; and (2) the chloride concentration can be reduced at the wall by a factor of 60 to 100 by a transpirational velocity of 5 to 10 cm/min. (AEC Activity 5310)

Thorium Oxide. — Removal of the existing thorium oxide production equipment from Building 3019 and reinstallation of this equipment in Building 2528 were completed. The plant, which has a capacity for producing large experimental batches of ThO_2 (1000 lb of ThO_2 per month), went on stream July 26, and seven production runs have been made. Orders for mixed oxide (1.2% uranium) with a particle size of 2.5 μ, fired at 1050°C, will keep the plant operating around the clock for five days a week. Four of the initial seven runs were to fill requests for pure thorium oxide. (AEC Activity 5310)

Feed Materials. — Studies of the adsorptive capacity of UO_2 powders for methylene blue from 0.2-g/liter chloride solutions in relation to the surface area and sinterability of the powders were continued. The presence of anions of weak acids in the solution inhibited the adsorption of the dye by UO_2. When 17 UO_2 powder samples were treated with 0.001 M $Na_4P_2O_7$—0.001 M NaOH at 100°C for 1 min and then washed free of phosphate with distilled water, their capacities for adsorption of methylene blue varied linearly with surface area, as measured by the Brunauer-Emmett-Teller nitrogen adsorption method, with a deviation of less than 10% from the mean. The correlation of methylene blue adsorption with sintered density of the powders was linear, but not so good as for surface area. (AEC Activity 5310)

Druhm Process. — Experimental data from seven runs showed that graphite liners 1/8 to 1 in. thick are usable for reactor temperatures above the boiling point of sodium. A copper nozzle, cooled with chilled helium, to introduce the UF_6 was not seriously damaged when the exit helium temperatures were kept below 100°C by adequate flow. More than 50% excess sodium is necessary to produce significant amounts of uranium metal rather than UF_4. In one run the recovery of uranium as consolidated uranium metal was 56.8%. In a study of liner materials and consolidation parameters it was found that some product penetration of both 1/8-in. high-fired MgO and 1/8-in. commercial-grade graphite occurs at 1200°C. Uranium metal consolidation appeared to be slow below 1180°C. (AEC Activity 5310)

Inorganic and Physical Chemistry: Inorganic Solution Chemistry. — The standard entropy ΔS^o and enthalpy ΔH^o of solution and the standard partial molal entropy $\overline{S}^o$ of Ag_2SO_4 have been determined as a function of temperature to 200°C:

t	ΔH^o(kcal/mole)	ΔS^o(e.u.)	$\overline{S}^o$ (e.u.)
25	4.5	-7.1	40.8
100	-0.8	-22.8	32.4
200	-9.6	-43.7	19.7

The negative value of ΔS^o and the decrease of both ΔS^o and $\overline{S}^o$ with temperature indicate that the presence of Ag_2SO_4 causes an increase in the amount of "structure" in the solvent water and that the increase (over the amount in pure water) is enhanced at higher temperatures.

Both activity coefficients and osmotic coefficients may be expressed as semiempirical two-, three-, or four-parameter equations containing a Debye-Hückel term plus one or more simple functions of the ionic strength. By making use of the Gibbs-Duhem relation, it was found that osmotic coefficient data in the range 1.0 to 3.5 solute molality can be used to evaluate activity coefficients in the range 0.1 to 5.0 $\underline{m}$ by the use of high-speed computing machines. (AEC Activity 5330)

Radiation Chemistry. — The hydrogen yields in the Co^{60} gamma irradiation of aqueous sodium nitrate solutions had been shown to be a linear function of the cube root of the sodium nitrate activity for solutions less than 1.30 $\underline{m}$. A rigorous test of this observation was obtained by altering the activity of the scavenging solute by the addition of the electrolytes NaCl (1.0–5.0 $\underline{m}$), $CaCl_2$ (0.30–2.02 $\underline{m}$), and $AlCl_3$ (0.066–1.02 $\underline{m}$) to 0.01 to 1.07 $\underline{m}$ $NaNO_3$ solutions.

At concentrations of $NaNO_3$ from 1.30 to 9.14 $\underline{m}$ the efficiency of the nitrate for scavenging hydrogen atoms, the assumed precursors of H_2, is reduced. This change in efficiency may be evidence that H_2 is formed by two mechanisms. The molecular hydrogen yields observed in solutions more than 1.30 $\underline{m}$ at 298°K are the same as those observed in identical solutions irradiated at 77°K, and extrapolate to $G(H_2) = 0.10$ in pure H_2O. The $G(H_2)$ observed in 0 to 1.30 $\underline{m}$ $NaNO_3$ solutions irradiated at 77°K are concordant with this extrapolation. The yield of H_2 derived from the more easily scavenged hydrogen atoms is given by $G(H_2) = 0.36 - 0.40\ [\gamma(NaNO_3)]^{1/3}$, and the yield from the less easily scavenged hydrogen atoms by $G(H_2) = 0.10 - 0.047\ [\gamma(NaNO_3)]^{1/3}$. (AEC Activity 5330)

High-Temperature and Structural Chemistry. — Measurement of the heats of fusion of the alkali halides was continued. The four cesium halides were found to form two groups, the entropy of fusion of CsF and CsCl being 5.3 e.u. and that of CsBr and CsI being 6.2 e.u. These entropies of fusion are characteristic of the melting of the two crystal structure groups, CsF and CsCl having the NaCl structure, whereas CsBr and CsI have the CsI structure. The new measurements replace earlier data contained in the literature and found to be erroneous.

Determination of the radial distribution function by means of x-ray diffraction in molten bismuth(I) chloroaluminate produced strong evidence in favor of the presence of the complex cation Bi_2^{++} postulated, on the basis of thermodynamic measurements, to be present in the Bi-$BiCl_3$ system. (AEC Activity 5330)

Analytical Chemistry Research. - A special high-temperature cell assembly has been designed and constructed for use with the Cary spectrophotometer model 14 M to record absorption spectra of molten fluoride salts. The spectra of such compounds as NiF_2, CoF_3, PrF_3, CrF_3, UF_4, and UO_2F_2, each dissolved in the eutectic mixture 46.5 LiF, 11.5 NaF, 42.0 KF (mole %), have been recorded at temperatures in the range 500 to 700°C. As a measure of the suitability of this cell assembly, the spectrum of UF_4 in this mixture has been observed as being essentially identical with that reported for UCl_4 in a molten eutectic mixture of LiCl and KCl. Similarly, the molar absorbance index of UF_4 at the absorbance peak of 1100 mμ is estimated to be about 15, a value which is in good agreement with a reported index of 13 for UCl_4 at 1090 mμ.

A study has been completed of a gamma-ray spectrometric method for the nondestructive analysis of various materials by means of the induced gamma radiation from radionuclides which are produced by the neutron irradiation of a sample. By a complement-subtraction technique, complex mixtures of radionuclides, emitting gamma radiation of various energies, have been analyzed by this rapid and easily applied method. As a result of this investigation, the gamma-ray scintillation spectrometer is being used to determine both short- and long-lived radionuclides in a variety of activated materials such as cement, limestone, sand, ores, water, metals and alloys, and tobacco.

METALLURGY AND SOLID STATE PHYSICS

Microstrains in Crystals. - A new x-ray diffraction technique for measuring the thickness of very thin oxide films on metals has been developed and has been tested for Cu_2O films grown on copper single crystals. The method consists in a measurement in absolute units of the integrated intensities of the Bragg maxima for the film. If the film is substantially a single crystal, such a measurement determines $(F^2_{hk\ell}\ V)$, the product of the square of the crystal structure factor and the volume of the film irradiated. If the area of the film irradiated, A, is measured and $F_{hk\ell}$ is either computed from theory or measured by a separate experiment, then the thickness of the film is determined:

$$T = \frac{(F^2_{hk\ell}\ V)}{F^2_{hk\ell}\ A} .$$

Such measurements for Cu_2O films grown on (110) faces of copper single crystals indicate that the method is promisingly accurate and consistent. (AEC Activity 5420)

Fundamental Physicometallurgical Research. — The hypothesis that metallic zirconium is approximately divalent, deduced from the effect of alloying on the allotropic transition temperature, has been examined by measurements of the effects of indium and tin on the electronic and lattice specific heat of zirconium. A valency of approximately 1.5 to 2.0 was indicated, the effects of tin on the magnitude of these quantities always being larger than the effects of indium.

Deformation twins were found in single crystals and coarse-grained specimens of high-purity niobium deformed by impact or slow compression at -196°C. The twinning elements were established to be K_1 (twin plane), $\{112\}$ and η_1 (shear direction), $\langle 11\bar{1} \rangle$. Cleavage fracture at -196°C was always found to occur on $\{100\}$ planes. (AEC Activity 5420)

Fundamental Investigations of Radiation Damage in Solids. — Polycrystalline samples of natural zirconium dioxide containing known concentrations of uranium and thorium impurities were exposed to thermal neutron dosages at approximately 85°C. It was found that the normal monoclinic oxide transforms under the action of fission fragments to form the high-temperature face-centered-cubic phase. By monitoring the number of fission events which took place in partially transformed samples, it was possible through x-ray diffraction techniques to estimate the number of atoms affected by a pair of fission fragments. It was found that each pair of fission fragments affects $\sim 2 \times 10^6$ atoms in what is believed to be a "fission spike" mechanism in which a region about 140 Å in radius is heated above 2000°C for a time as short as 10^{-11} sec. In a complementary study with synthetic pure single crystals of monoclinic zirconium dioxide it was observed that exposure to fast-neutron dosages in excess of 10^{20} neutrons/cm^2 produced no measurable change in the single-crystal perfection of the samples. This is believed to be further evidence supporting the "fission spike" phenomenon in zirconium dioxide. (AEC Activity 5430)

THERMONUCLEAR PROJECT

Carbon Arc and Vacuum. — An attempt has been made to achieve pressures in the 10^{-8} mm region in the presence of the carbon arc in the PIG Plate Facility. After several hours of (noncontinuous) operation, a pressure of 7.5×10^{-8} mm was observed in the presence of a 300-amp arc and 5.5×10^{-8} mm in the presence of a 250-amp arc. Evidence of wall cleanup was observed, and it was tentatively inferred that the observed base pressure was established primarily by repeated large pressure pulses which resulted from arc instabilities. (AEC Activity 5540)

DCX-1 Facility. — The main activities of DCX-1 during the past month were directed toward preventing the carbon arc from striking the anode baffle assembly and causing a water leak. It is not certain, as this is written, that adequate control measures have been taken.

The reduced time for good operation still permitted more than 600 oscillograms to be taken from detectors which measure the particles emerging from the central storage region. The initial inspection of these oscillograms indicates that the ion accountability may be somewhat higher than reported last month.

A traveling detector has been installed which should permit better axial scanning of the emerging particles. (AEC Activity 5540)

DCX-2 and Magnetic Studies. — A study was undertaken to investigate the degree of flexibility in magnetic field shapes attainable from simple current ratio manipulation in the field coils.

Various criteria for field shape were used, varying from fields with maximum homogeneity in the mid-plane region to fields which were thought to more nearly preserve the concept of over-all particle adiabaticity.

Evidently it is possible to obtain a field varying from one that is very homogeneous over a relatively large region to one that has a fairly constant shape over a relatively large region. (AEC Activity 5540)

PROGRAM 6000 — BIOLOGY AND MEDICINE

BIOPHYSICS

Internal Dosimetry. — When a radionuclide which emits gamma rays is taken into the body, the concentration of the nuclide in the bladder, in the gastrointestinal tract, or in other organs may contribute significantly to the dose that the gonads receive. The dose to the peripheral cortex of the ovary from an assumed concentration of the emitter in the bladder or GI tract has been computed by use of a single-hit theory as well as a straight-ahead approximation for gamma rays of energy between 0.05 and 10 Mev. It was found that in most cases the gastrointestinal contents deliver more dose to the gonads than the contents of the bladder do. Available data concerning the concentrations of most of these nuclides in human gonads are scanty and perhaps not accurate, but they indicate that for many radionuclides a large, even a preponderant, fraction of the genetic dose is due to the presence of the nuclide in the bladder or in the GI tract. (AEC Activity 6440)

Waste Disposal Research and Engineering: Evaluation of Soil Disposal. — Fabrication of a new radiologging unit, to be used primarily for well monitoring, was completed. The cable reel and feed mechanism incorporates such features as an overload clutch, automatic log, retract and shutoff, resettable level-wind index, optional two-speed hand operation, and a wide range of automatic logging speeds. Power is supplied from a 12-v storage battery which should provide at least 16 hr of operation without recharging. The electronic unit utilizes the average current from four G-M tubes and employs only one vacuum tube and one corona-type voltage regulating tube. A 12-v storage battery powers the electronic unit and should provide at least 35 hr of operation without recharging. The entire assembly, including the electronic unit, the reel assembly, the recorder, and the batteries, will be mounted on a frame for operation from a Jeep station wagon. (AEC Activity 6440)

OPEN LITERATURE PUBLICATIONS

Auerbach, S. I., and D. A. Crossley, "Strontium-90 and Cesium-137 Uptake by Vegetation Under Natural Conditions," Proc. U. N. Intern. Conf. Peaceful Uses Atomic Energy, 2nd, Geneva, 1958 18, 494 (1959).

Blewitt, T. H., R. R. Coltman, and C. E. Klabunde, "Stored Energy of Irradiated Copper," Phys. Rev. Letters 3, 132 (1959).

Boyle, J. W., S. Weiner, and C. J. Hochanadel, "Kinetics of the Radiation-Induced Reaction of Iron(III) with Tin(II)," J. Phys. Chem. 63, 892 (1959).

Bredig, M. A., "Dimeric Bismuth(I) Ion, $(Bi_2)^{2+}$, in Molten Bismuth Trihalide–Bismuth Systems," J. Phys. Chem. 63, 978 (1959).

Collins, C. J., and N. S. Bowman, "Molecular Rearrangements. XVI. The Pinacol Rearrangement of the Diphenyl-m-tolylethylene Glycols," J. Am. Chem. Soc. 81, 3614 (1959).

Cowser, K. E., R. J. Morton, and E. J. Witkowski, "The Treatment of Large Volume, Low-Level Waste by the Lime-Soda Softening Process," Proc. U. N. Intern. Conf. Peaceful Uses Atomic Energy, 2nd, Geneva, 1958 18, 161 (1959).

Crawford, J. H., Jr., and M. C. Wittels, "Radiation Stability of Nonmetals and Ceramics," Proc. U. N. Intern. Conf. Peaceful Uses Atomic Energy, 2nd, Geneva, 1958 5, 300 (1959).

Datz, S., and E. H. Taylor, "Some Applications of Molecular Beam Techniques to Chemistry," p 157 in Recent Research in Molecular Beams, ed. by I. Estermann, Academic Press, New York, 1959.

de Laguna, W., K. E. Cowser, and F. L. Parker, "Disposal of High-Level Radioactive Liquid Wastes in Terrestrial Pits; a Sequel," Proc. U. N. Intern. Conf. Peaceful Uses Atomic Energy, 2nd, Geneva, 1958 18, 101 (1959).

Hurst, G. S., and T. E. Bortner, "Attachment of Low-Energy Electrons in Mixtures Containing Oxygen," Phys. Rev. 114, 116 (1959).

Kerr, J. M., "Preliminary Tests on Clay Sinters to Retain Reactor Wastes," Am. Ceram. Soc. Bull. 38, 374 (1959).

Kraus, K. A., D. C. Michelson, and F. Nelson, "Adsorption of Negatively Charged Complexes by Cation Exchangers," J. Am. Chem. Soc. 81, 3204 (1959).

Lietzke, M. H., and R. W. Stoughton, "The Solubility of Silver Sulfate in Electrolyte Solutions. I. Solubility in Potassium Nitrate Solutions," J. Phys. Chem. 63, 1183 (1959).

Lietzke, M. H., and R. W. Stoughton, "The Solubility of Silver Sulfate in Electrolyte Solutions. II. Solubility in Potassium Sulfate Solutions," J. Phys. Chem. 63, 1186 (1959).

Lietzke, M. H., and R. W. Stoughton, "The Solubility of Silver Sulfate in Electrolyte Solutions. III. Solubility in Sulfuric Acid Solutions," J. Phys. Chem. 63, 1188 (1959).

Lietzke, M. H., and R. W. Stoughton, "The Solubility of Silver Sulfate in Electrolyte Solutions. IV. Solubility in Nitric Acid Solutions," J. Phys. Chem. 63, 1190 (1959).

Livingston, Ralph, "Physical Observation of Radiation-Induced Intermediates," J. Chem. Educ. 36, 349 (1959).

Mahlman, H. A., "Ceric Reduction and the Radiolytic Hydrogen Yield," J. Am. Chem. Soc. 81, 3203 (1959).

Ritchie, R. H., "Interaction of Charged Particles with a Degenerate Fermi-Dirac Electron Gas," Phys. Rev. 114, 644 (1959).

Rudolph, P. S., and C. E. Melton, "Mass Spectrometric Studies of Ionic Intermediates in the α-Particle Radiolysis of the C_2 Hydrocarbons. I. Acetylene," J. Phys. Chem. 63, 916 (1959).

Sherwood, J. E., and S. J. Ovenshine, "Nuclear Spins of I^{128} and I^{130}," Phys. Rev. 114, 858 (1959).

Struxness, E. G., and J. O. Blomeke, "Multipurpose Processing and Ultimate Disposal of Radioactive Wastes," Proc. U. N. Intern. Conf. Peaceful Uses Atomic Energy, 2nd, Geneva, 1958 18, 43 (1959).

Taylor, E. H., "Radiation Effects on Solids, Including Catalysts," J. Chem. Educ. 36, 396 (1959).

Taylor, E. H., "Samuel Colville Lind," J. Phys. Chem. 63, 772 (1959).

Wilson, J. C., "Effects of Irradiation on the Structural Materials in Nuclear Power Reactors," Proc. U. N. Intern. Conf. Peaceful Uses Atomic Energy, 2nd, Geneva, 1958 5, 431 (1959).

Zittel, H. E., F. J. Miller, and P. F. Thomason, "Amperometric Titration of Barium," Anal. Chem. 31, 1351 (1959).

FOREIGN VISITORS

Name	Affiliation	Citizen of
Baysal, B. M.	Ankara University	Turkey
Bromley, D. A.	Atomic Energy of Canada Limited	Canada
Christian, J. W.	Oxford University	Great Britain
Erichsen, L. V.	University of Bonn	Germany
Forland, T.	Norwegian Institute of Technology	Norway
Gronlund, F.	University of Indiana	Denmark
Halvorsen, K.	Joint Establishment for Nuclear Energy Research and Argonne Natl. Laboratory	Norway
Huber, K.	Physikalisches Institut der Univ., Basel	Switzerland
Hudson, R. P.	National Bureau of Standards	Great Britain
Inoue, K.	Hokushin Electric Works	Japan
Kleijn, H. R.	Delft Technological University	The Netherlands
McPherson, N. F.	Canadian General Electric Co.	Canada
Matsuda, T.	Kurita Kogyo Co., Ltd.	Japan
Miescher, E.	Physikalisches Institut der Univ., Basel	Switzerland
Sato, H.	National Institute of Radiological Sciences	Japan
Sawyer, W. A.	Canadian General Electric Co.	Canada
Shiga, T.	National Delegate from Japan	Japan
Takeuchi, T.	Brown University	Japan
Tiberghien, R.	Indatom, Paris	France
Tsukamoto, K.	National Institute of Radiological Sciences	Japan
Ukai, T.	College of Shizuoka	Japan
Vanhecke, L. H.	University of Louvain	Belgium

* * * * *

Reports issued in this series during the past year:

August 1958	ORNL-2595
September 1958	ORNL-2623
October 1958	ORNL-2636
November 1958	ORNL-2645
December 1958	ORNL-2664
January 1959	ORNL-2691
February 1959	ORNL-2706
March 1959	ORNL-2721
April 1959	ORNL-2736
May 1959	ORNL-2756
June 1959	ORNL-2781
July 1959	ORNL-2795

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

August 1959

Distribution

U.S. Atomic Energy Commission

Oak Ridge Operations Office

1-32. Research and Development Division

External

33.	Aeroprojects, Inc.	W. B. Tarpley
34-36.	Ames Laboratory: Iowa State College	F. H. Spedding
37-39.	Argonne National Laboratory	N. S. Hilberry
40.	Atomics International (N. Am. Aviation, Inc.)	E. E. Motta
41-42.	Battelle Memorial Institute	H. W. Russell
43-45.	Brookhaven National Laboratory	L. J. Haworth
46.	Bureau of Mines, Salt Lake City	J. B. Clemmer
47-48.	Chicago Operations Office	
49-54.	Hanford Atomic Products Operation	O. H. Greagor
55-56.	Idaho Chemical Processing Plant	R. L. Doan
57.	Kaiser Engineers	P. D. Bush
58-59.	Knolls Atomic Power Laboratory	K. R. Van Tassell
60-61.	Lawrence Radiation Laboratory	D. Cooksey
62-63.	Los Alamos Scientific Laboratory	N. E. Bradbury
64-65.	National Bureau of Standards	F. L. Mohler
66.	National Lead Company	C. K. McArthur
67.	Nuclear Metals, Inc.	A. R. Kaufmann
68.	Union Carbide Corporation	R. W. McNamee
69.	Union Carbide Nuclear Company	C. E. Center
70.	Union Carbide Nuclear Company	J. L. Gabbard
71.	Union Carbide Nuclear Company	D. M. Lang
72.	Union Carbide Nuclear Company	J. P. Murray

Internal

73. K. A. Allen
74. E. E. Anderson
75. E. A. Bagley
76. P. S. Baker
77. C. R. Baldock
78. L. H. Barker
79. S. E. Beall
80. E. E. Beauchamp
81. R. H. Beidel
82. P. R. Bell
83. D. S. Billington
84. C. A. Blake
85. R. E. Blanco
86. F. F. Blankenship
87. E. P. Blizard
88. A. L. Boch
89. E. G. Bohlmann
90. C. J. Borkowski

Distribution (continued)

91. G. E. Boyd
92. M. A. Bredig
93. J. C. Bresee
94. R. B. Briggs
95. R. E. Brooksbank
96. K. B. Brown
97. F. R. Bruce
98. W. D. Burch
99. T. H. J. Burnett
100. A. D. Callihan
101. A. E. Cameron
102. R. A. Charpie
103. C. E. Clifford
104. T. E. Cole
105. C. F. Coleman
106. E. L. Compere
107. D. D. Cowen
108. J. A. Cox
109. F. L. Culler
110. J. E. Cunningham
111. J. S. Drury
112. D. E. Ferguson
113. J. L. Fowler
114. J. H. Frye
115. W. F. Gauster
116. M. W. Gerrard
117. J. H. Gillette
118. W. Y. Gissel
119. H. E. Goeller
120. C. D. Goodman
121. A. T. Gresky
122. J. C. Griess
123. W. R. Grimes
124. P. A. Haas
125. C. S. Harrill
126. J. C. Hart
127. M. R. Hill
128. C. J. Hochanadel
129. A. Hollaender
130. R. W. Horton
131. F. T. Howard
132. A. S. Householder
133. H. H. Hubbell
134. G. S. Hurst
135. G. H. Jenks
136. R. W. Johnson
137. R. J. Jones
138. W. H. Jordan
139. P. R. Kasten
140. G. W. Keilholtz
141. C. P. Keim
142. M. T. Kelley
143. E. M. King
144. K. A. Kraus
145. Laboratory Shift Supervisor (D. Phillips)
146. E. Lamb
147. J. A. Lane
148. R. E. Leuze
149. W. H. Lewis
150. T. A. Lincoln
151. S. C. Lind
152. R. B. Lindauer
153. R. S. Livingston
154. J. T. Long
155. L. O. Love
156. R. N. Lyon
157. R. J. Mackin
158. H. G. MacPherson
159. F. C. Maienschein
160. W. D. Manly
161. M. B. Marshall
162. J. A. Martin
163. H. F. McDuffie
164. J. R. McNally
165. A. J. Miller
166. E. C. Miller
167. K. Z. Morgan
168. W. L. Morgan
169. E. J. Murphy
170. M. L. Nelson
171. C. E. Normand
172. J. L. Need
173. G. W. Parker
174. M. E. Ramsey
175. P. M. Reyling
176. A. W. Riikola
177. L. P. Riordan
178. G. A. Ropp
179. A. F. Rupp
180. A. D. Ryon
181. H. C. Savage
182. H. W. Savage
183. A. W. Savolainen
184. H. E. Seagren
185. E. D. Shipley
186. A. Simon
187. O. Sisman
188. J. R. Sites
189. M. J. Skinner

190. A. H. Snell
191. R. W. Stoughton
192. H. Stringfield
193. E. G. Struxness
194. W. H. Sullivan
195. C. D. Susano
196. J. A. Swartout
197. E. H. Taylor
198. C. D. Watson
199. B. S. Weaver
200. A. M. Weinberg
201. M. E. Whatley
202. J. C. White
203. C. E. Winters
204. R. E. Worsham
205. W. Zobel
206. A. Zucker
207-208. Central Research Library
209. Health Physics Library
210. ORNL - Y-12 Technical Library
Document Reference Section
211. Reactor Experimental
Engineering Library
212-403. Laboratory Records Department
404. Laboratory Records, ORNL RC