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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

MAY 1959

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

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

May 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. — An additional 8 metric tons of BNL irradiated uranium fuel, containing 3.6 kg of plutonium, was processed. Based on losses from the plant, uranium and plutonium recoveries were 99 and 99.25%, respectively. The plutonium product met gross gamma specifications. The uranium product gross gamma contamination was higher than the limits given in the specifications by a factor of 3 but was acceptable for shipment to Y-12. Uranium product decontamination was improving as the run progressed. (AEC Activity 2351)

In-Line Instrumentation. — The fluorothene-tube in-line gamma monitor has functioned successfully for five months. The background contamination builds up to 35% of the count rate of the process stream being monitored. Measurements, in gross gamma counts per minute per milliliter, agree to within ±50% of laboratory analysis, which is sufficiently close for process control. The presence of radioactive solids in the process stream is the chief cause for the disagreement with laboratory analyses.

A single-beam gamma absorptiometer is being evaluated for measuring heavy-metal ion concentration in monitoring, for example, uranium, plutonium, or thorium concentrations in aqueous and organic streams. In tests with a stream of lead nitrate solution, the plot of gamma absorption vs lead concentration was linear on semilogarithmic paper over the range 0 to 300 g/liter. No corrections were needed for decay of the source. (AEC Activity 2724).

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PROGRAM 4000 — REACTOR DEVELOPMENT

HOMOGENEOUS REACTOR PROJECT

Homogeneous Reactor Test. — During hydrostatic testing of the reactor high-pressure system piping and components in preparation for run 20, the fuel circulating pump stopped rotating. On the basis of an observed 200-lb surge in system pressure, it appears that the titanium impeller of this pump may have ignited as did that of the blanket pump, which was described last month. The fuel pump has been replaced, and it will be disassembled remotely and inspected at an early date. The replacement pump was fitted with a stainless steel impeller. (AEC Activity 4103.3)

Reactor Analysis. — The high-energy flux in the HRT was calculated as a function of position for blanket-to-core fuel-concentration ratios of 0 and 0.33. Only uncollided and once-collided neutrons were considered. The neutron flux above 1 Mev energy at the core-tank wall was about 8×10^{11} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1} at 1 Mw, with no fuel in the blanket. With fuel in the blanket region the corresponding flux was about 5.6×10^{11} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1}. At the pressure-vessel wall, the neutron flux above 1 Mev energy was about 7×10^{8} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1} per megawatt, with no fuel in the blanket region, and about 4×10^{10} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1} per megawatt with fuel in the blanket (blanket-to-core fuel concentration ratio about 0.3). The integrated fast flux (that above 1 Mev) at the inner surface of the pressure vessel was computed to be $<1.2 \times 10^{15}$ neutrons/cm^2 for HRT run 14, and 3.6×10^{17} neutrons/cm^2 for HRT run 17. (AEC Activity 4103.1)

Homogeneous Reactor Instrumentation. — A differential transformer for sensing the primary-element motions of sealed instruments and capable of operating for long periods at 300°C was developed. The transformer exhibited no change in characteristics when cycled from 25 to 350°C and has shifted less than 2% in zero and span in a two-month test at 300°C. The transformer is wound with ceramic-insulated wire on an Inconel form and has a sensitivity of 0.1 mv/v per 0.001 in. when used with appropriate exciting and impedance-matching circuitry.

A flexible bellows assembly fabricated from grade Ti-55 titanium was developed by Fulton Sylphon Division on an ORNL contract. One of the bellows, which are made up from three 0.100-in. plies, was cycled 11,600

times (1/8-in. strokes) in uranyl sulfate at 280°C and with a 2300-psi differential before failure occurred at a convolution. The bellows, which have an inside diameter of 13/16 in. and a spring rate of 300 lb/in., are intended for use as stem seals for control valves. (AEC Activity 4103.1)

HRP Design. — Equipment was designed for use in flushing the HRT core and blanket systems during maintenance operations. A removable, basket-type screen strainer is utilized in each system to trap a portion of the circulating solids.

Several special tools are being designed for use in viewing, cutting loose, cutting up, and removing the HRT core screens and for plugging the core hole.

The design of a small titanium loop for operation at 3100 psi and 375°C was completed. (AEC Activity 4103.1)

HRP Metallurgy. — By means of tungsten-arc melting or cutting torches, which may be useful in remote maintenance work, zirconium and titanium were cut successfully under water. The cutting torch, small enough to pass through a 2-in.-dia hole, cut 1/8-in. Zircaloy-2 material at rates as high as 100 in./min; it produced "shot-like" drops with no tendency to stick together. Melting torches adapted with either single or multiple electrodes cut at a rate of 15 in./min. The melting torches produced very few drops, since most of the melted material remained attached to the sides of the cut; however, when drops did form they were large and often stuck together. Careful planning and close control, either manual or by instruments, were required in making satisfactory cuts with the multiple-electrode torches. (AEC Activity 4103.1)

Laboratory Corrosion Studies. — A corrosion study was conducted to determine whether sensitized type 347 stainless steel that corroded badly in boiling 65% HNO_3 would be susceptible to intergranular attack in uranyl sulfate solutions. Different heats of the sensitized steel that corroded in the nitric acid at rates ranging from 1.3 to 39.2 mils/month were exposed to uranyl sulfate solutions at temperatures of 100, 200, and 300°C. (Rates in excess of 2 to 3 mils/month in the nitric acid test are usually associated with extensive intergranular penetration.) Results of the study showed that the heats of sensitized type 347 stainless steel that were highly

susceptible to intergranular attack in boiling 65% HNO_3 did not undergo intergranular penetration in the uranyl sulfate solutions, regardless of the test conditions or the particular heat of steel. On the other hand, sensitized specimens of the unstabilized type 310 stainless steel that corroded extensively in the nitric acid test also exhibited substantial intergranular attack in the uranyl sulfate solutions. Presumably, other sensitized, unstabilized stainless steels (exclusive of the extra-low-carbon grades) would show a behavior similar to that of type 310 stainless steel. (AEC Activity 4103.1)

Dynamic Solution Loop Corrosion Tests. — Several reagents were tested for possible usefulness in removing deposits of corrosion-product scale in the HRT. The two most promising reagents were Deoxidine-170 (inhibited 8 M phosphoric acid) and Versenol-120 (trisodium salt of N-hydroxyethyl-ethylenediaminetriacetic acid) buffered with ammonium acetate. The Deoxidine-170 solution completely dissolved a heavy deposit of corrosion products from a stainless steel loop in 36 hr of circulation at 125°C. Stainless steel and Zircaloy-2 specimens suffered negligible attack, but titanium corroded at a rate of 660 mils/year (mpy), and Stellite 98M2 at 250 mpy. Additional tests showed that reducing the concentration of Deoxidine-170 by a factor of 2 reduced the corrosion rates of titanium and Stellite 98M2 by a factor of 10; the rate of scale removal, as measured by the rate of increase of iron in solution, was reduced by only a factor of about 2.

A solution containing 20% Versenol-120 plus 25% ammonium acetate was circulated for 64 hr at 125°C in a previously used stainless steel loop. Type 347 stainless steel specimens prefilmed in 0.025 m UO_2SO_4 at 225°C lost most of their film, an average of 1.0 mg/cm^2, and were not corroded significantly. The appearance of the loop surfaces indicated a similar behavior. The solution did not dissolve the scale completely but peptized some of it into very finely divided solids. Some decomposition of the solution occurred as shown by an increase in loop pressure during the test and the strong odor of ammonia when the loop was vented at the end of the test. (AEC Activity 4103.1)

Small-Scale Dynamic Slurry Corrosion Tests. — To compare the effects of atmosphere and temperature on the attack of materials and on the stability of uranium in slurries prepared from thoria–8.8% urania, a series of 300-hr toroid tests was conducted at 280°C with slurries containing 500 g of Th-U per kg of D_2O. The relative susceptibility of four selected alloys to attack at 26 fps under an atmosphere of 25-psi oxygen, expressed as a ratio of corrosion rate of the given metal to the rate of type 347 stainless steel pins at the same conditions, was: type 347 stainless steel pins, 1.0; type 347 stainless steel piping, 0.2; titanium-75A, 0.3; Zircaloy-2, 0.1; and Inconel, 5.3. The normalized rate of type 347 stainless steel at 280°C under the oxygen atmosphere was 10 mils/year. Decreasing the test temperature from 280 to 200°C decreased the attack of type 347 stainless steel, titanium-75A, and Inconel to about 50% of the 280°C value. Under a deuterium atmosphere the attack of Inconel was reduced at both test temperatures, whereas attack of the other materials was increased by an average factor of 3.

No significant separation of uranium was reported under any condition.

A 30-mil fissured layer of zirconium deuteride was observed on the upstream abraded surface of a Zircaloy-2 pin in one of the above tests at 280°C and 26 fps, in which a 65-psi deuterium atmosphere was used and in which 0.0017 m CrO_3 had been added to the slurry. (AEC Activity 4103.1)

Dynamic Slurry Loop Corrosion Tests. — A slurry of thoria–8.8% urania at a concentration of 400 g of Th per kg of D_2O and with 490 std cc of D_2 per liter of slurry was circulated at 200°C for 724 hr in a 100A loop. This was an initial test of a potential fuel slurry. The over-all attack rate for the austenitic stainless steel loop piping (at 6.5 fps) was 2 mils/year (mpy). Attack on the titanium scroll of the circulating pump was 15 mpy and on the Zircaloy-2 pump impeller was 1 mpy. Attack rates of pin specimens exposed at velocities of 20 and 50 fps, respectively, were (in mpy): type 347 stainless steel, 0.3 and 81; Inconel, 0.1 and 4.8; titanium RC-55, <0.1 and 1.9; Zircaloy-2, 0.1 and 0.3; and Incoloy, 0.1 and 23. The U^{6+} content of the mixed thoria-urania was reduced to U^{4+} during the test.

When the slurry was recirculated at 200°C with an oxygen atmosphere (546 cc O_2 per liter of slurry), the U^{4+} in the slurry was reoxidized to

U^{6+}. At a slurry concentration of 200 g of Th per kg of D_2O the loop piping, pump scroll, and impeller were not detectably attacked. Attack rates of specimens exposed during the 407-hr test, at 20 and 40 fps, respectively, were (in mpy): type 347 stainless steel, Zircaloy-2, and titanium RC-55, <0.1 and <0.1; Inconel, 2.5 and 3.2; and Incoloy, <0.1 and 1.1. (AEC Activity 4103.1)

In-Pile Autoclave Solution Corrosion Studies. — In an experiment undertaken to obtain information concerning uranium deposition on hot surfaces in autoclaves, a hollow cylindrical Zircaloy-2 specimen filled with UO_2 enriched in U^{235} was exposed along with two coupon-type control specimens in a solution of 0.026 m UO_2SO_4, 0.02 m $CuSO_4$, and 0.016 m D_2SO_4 in D_2O, contained in a titanium autoclave in the HB-5 facility of the LITR. Fissioning inside the pin specimen produced a quantity of heat sufficient to create a range of heat fluxes along the length of the pin of 4 to 6.5 w/cm^2. It is estimated that these heat fluxes gave rise to temperature gradients between the metal surfaces and the solution of from 20 to 30°C. The solution fission power density was about 0.7 w/ml at the location of one control specimen and 1.5 w/ml at the other. The fission-heated pin was exposed to solution power densities of 0.8 to 1.3 w/ml. The radiation exposure period was 724 hr at temperatures, in chronological order, of 250°C, 37 hr; 280°C, 311 hr; 300°C, 140 hr; and 320°C, 236 hr.

Microscopic examination after exposure showed that the heated pin specimen was intact and that the corrosive attack was fairly uniform. The average amounts of corrosion as shown by weight losses after defilming were 0.22 and 0.43 mil, respectively, for the control coupons and 0.31 mil for the heated pin. The scales on the heated pin were different from those on the control specimen as shown by the manner in which scales were removed during the descaling operations. The descaling behavior of the control coupons was normal; essentially all the scale was removed during drying and very little was removed during subsequent cathodic descaling. Scale was also removed from the heated pin during drying, but an additional 3 mg/cm^2 was removed in cathodic defilming. The uranium content of all bulk scales examined was very low: 0.04% uranium for scale from the heated pin and 0.1% for scales from the coupons.

The corrosion shown by weight loss of the heated pin is in near agreement with that of the control specimens when adjusted for solution power density. Thus no effect of the heating on corrosion is apparent in the results. (AEC Activity 4103,1)

HRP Service Corrosion. — Examination of the corrosion-specimen array removed from the core of the HRT after run 17 revealed considerable amounts of scale, averaging 15 mg/cm^2, on the portion of the titanium holder and on the corrosion specimens which had been located in very low flux regions of the core outlet pipe. Zirconium was the major constituent (~50%) in these scales; the uranium present averaged about 1.5%; the titanium, about 1.4%; and the iron, about 4%. Activation measurements with Zircaloy-2 specimens indicate fluxes of 3.4×10^{10} to 4.4×10^{11} neutrons$\cdot cm^{-2} \cdot sec^{-1}$ at a reactor power of 3 Mw.

No accurate determinations were made of the amounts of scale on those holder and specimen surfaces which were located in the high-flux region of the core. However, the scale appeared very much thinner than that in the low-flux portion, and samples scraped from the titanium holder showed about 10% iron, 6% zirconium, and 9.2% titanium.

Corrosion rates were obtained on specimens exposed at fluxes near 1×10^{13} neutrons$\cdot cm^{-2} \cdot sec^{-1}$ at 3 Mw in the blanket region of the HRT, also during run 17. Based on weight-loss results and an exposure period of 890 hr, the titanium alloys exhibited corrosion rates of 0.3 to 0.5 mil/year (mpy); the stainless steel specimens, 2.4 to 4.5 mpy; and the zirconium alloys, 2.9 to 3.9 mpy. The corrosion rates for the zirconium and titanium alloys are in fair agreement with rates predictable from results obtained on specimens in other in-pile studies, but the stainless steel rates are about an order of magnitude greater than predicted for this low power density.

Corrosion specimens were removed from the fuel and blanket circulating lines after run 18. Almost all these specimens showed weight gains after repeated defilmings. Corrosion rates obtained on types 347 and 309 SCb stainless steel were less than 0.1 mpy. An average scale of approximately 2.5 mg/cm^2 was found on the blanket-system specimens, compared with 6 mg/cm^2 on the fuel-system specimens. (AEC Activity 4103.1)

Supporting Radiation Corrosion Studies. — A series of preliminary experiments was performed to study the effects of fission-fragment recoils on the corrosion of zirconium in an oxygen atmosphere at 250°C. The experiments were designed to provide for oxidation of zirconium specimens under the following conditions: (1) reactor radiations including fission-fragment recoils, (2) reactor radiations in the absence of fission-fragment recoils, and (3) no radiations. The irradiation experiments were conducted in Hole 16 of the ORNL Graphite Reactor, where the maximum thermal-neutron flux is 8.5×10^{11} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1} at a reactor power of 3.5 Mw. The specimen surfaces exposed to fission-fragment recoils were subjected to a calculated average recoil intensity of 5×10^{8} particles$\cdot$cm$^{-2}\cdot$sec^{-1}, with an average particle energy of about 10% of the initial energy. The experiments were run either one or two weeks.

Examination of the specimens revealed a significant difference between the oxide on the specimens exposed to fission fragments and that on specimens exposed to reactor radiations only. Under polarized light, the recoil-irradiated specimens were optically isotropic; the other specimens were optically anisotropic. Low-angle electron diffraction patterns of the surface of a specimen exposed to fission-fragment recoils showed very diffuse rings, while the patterns from the surface that was in reactor radiations only showed a relatively large number of spotty rings. A surface-topography study of the specimens by use of the electron microscope showed that the majority of the surfaces exposed to recoils had a rough, mottled surface, while the surfaces exposed only to other radiations were, in general, much smoother. Microscopic examination under normal illumination revealed that the interference colors displayed by the oxide film of the recoil-irradiated surfaces were consistently different from those of the other films.

It can be concluded from these observations that the normal oxide film on the surface that was exposed to recoils had developed a very fine particle size with random orientation. It is also possible to interpret the observed change in optical anisotropy in terms of a phase transformation of the oxide. (AEC Activity 4103.1)

MOLTEN-SALT REACTOR (MSR) PROGRAM

Chemistry. – Studies of the freezing behavior of fuel mixtures containing ThF_4 and UF_4 have shown that the substitution of UF_4 for ThF_4 in $LiF-BeF_2-ThF_4$ (67.5-17.5-15 mole %) depresses the liquidus temperature linearly from 500°C to 480°C for complete substitution. Similarly, with $LiF-BeF_2-ThF_4$ (70-10-20 mole %), the liquidus decreases from 534°C to 459°C on complete substitution of UF_4 for ThF_4. Crystallization paths for fuels of current interest have been charted with the aid of refined data.

The solubility of rare-earth fluorides in $LiF-BeF_2$ mixtures containing 15 to 20 mole % UF_4 or ThF_4 has been studied. Increased solubility of CeF_3 of about 0.8 mole % at 500°C and 3.3 mole % at 700°C was found, in comparison with the minimum solubility in the $LiF-BeF_2$ binary solvent containing 37 mole % BeF_2, which was 0.16 mole % at 500°C and 1.23 mole % at 700°C.

The use of BeO to separate uranium from rare-earth fluorides in $LiF-BeF_2$ (63-37 mole %) appears to be promising on the basis of parallel experiments in which UF_4 (1 mole %) was precipitated as UO_2 and CeF_3 (0.25 mole %) remained in solution.

Graphite (AGOT grade) rods soaked for 1000 hr at 1200°F in $NaF-BeF_2-UF_4$ (53-46-1 mole %) gained 0.5% in weight and showed relatively high concentrations of fuel in the center section, as compared with higher concentrations at the periphery and low concentrations at intermediate distances along the radius. (AEC Activity 4103.1B)

Physical Properties. – The experimental enthalpy-temperature data obtained on fused salt mixtures during the past few years are being critically reviewed with the intent of extending or improving the correlations of liquid heat capacities. Given the composition of a mixture, these correlations enable estimation of heat capacities for engineering design purposes. Thus, based on the data obtained with six BeF_2-containing mixtures, the following equation can be used to predict the heat capacity of other mixtures containing BeF_2:

$$(c_p)_{liq} = 3.821(\bar{A})^{-0.785} \quad ,$$

where $\bar{A}$ is the ratio of the average molecular weight of the mixture to the average number of atoms ($\bar{M}/\bar{N}$). (AEC Activity 4103.1B)

Graphite-Moderated, Molten-Salt-Fueled, Breeder Reactor. — A parametric study was performed of heterogeneous, graphite-moderated, 5-ft-dia reactors fueled with U^{233} in a salt mixture containing 13 mole % ThF_4, 71 mole % LiF, and 16 mole % BeF_2. The core of the reactor is surrounded by a 2.5-ft-thick blanket region filled with a salt mixture containing 13 mole % ThF_4. The lattice has a spacing of 8 in., with a triangular pitch. The fuel channel diameter was varied to yield different volume fractions of fuel. A peak initial breeding ratio of 1.16 was obtained at a fuel volume fraction of 0.12.

From a nuclear standpoint, these reactors look promising; however, there is some question concerning the temperature of the graphite if there is fuel penetration. If fuel penetrates the graphite to 1 vol %, about 15 Mw of heat would be generated in the graphite at a reactor power level of 125 Mw. A temperature rise in the graphite of less than 400°F above the graphite surface temperature would require reducing the lattice spacing to 3 in. (AEC Activity 4103.1B)

D_2O-Moderated, Molten-Salt-Fueled, Breeder Reactors. — The possibility of using D_2O as a moderator in molten-salt-fueled reactors was studied. In a first conception of the system, D_2O is contained in a large cylindrical vessel penetrated in a direction parallel to the axis by zirconium tubes about 5 in. in diameter. These tubes contain two concentric graphite tubes having walls about 1/4 in. thick, surrounding a graphite tube through which the molten-salt fuel is pumped. The fuel is composed of 71 mole % LiF, 16 mole % BeF_2, and 13 mole % ThF_4, together with sufficient $U^{233}F_4$ to make the system critical. The fuel channels are 1.875 in. in radius and are arranged in a triangular lattice having a spacing of 12.5 in. A concentration of about 0.15 mole % UF_4 gave a multiplication in the infinite reactor (k_∞) of 1.007 and an initial regeneration ratio (with no fission products or nonfissionable isotopes of uranium present) of 1.17.

This reactor appears to have considerable promise as a one- or two-region breeder having a probable doubling time of less than 20 years. (AEC Activity 4103.1B)

Creep Properties of INOR-8. – Results of the first 4000 hr of creep studies conducted on 0.060-in. INOR-8 sheet specimens in air at 1250°F have been obtained. The measured creep rates vary from 1.9×10^{-2}% per hour at 30,000 psi to 1.6×10^{-4}% per hour at 10,000 psi. The specimens used in these studies generally exhibited an initial period of nil and even negative creep followed by accelerated creep. (AEC Activity 4103.1B)

Triplex Heat Exchanger Tubing Development. – Two assemblies of triplex tubing have been constructed with different grades of porous nickel sheet between Inconel tubes. To obtain an adequate bond of the three layers, it was found necessary to anneal the tubing prior to drawing. Plug drawing was used, followed by a 2-hr sinter in H_2 at 2200°F. Evaluation of the final assemblies indicated that both tubes were bonded to the core and that the porosity of the nickel core was retained. (AEC Activity 4103.1B)

Brazing of Graphite. – A screening survey has been conducted to determine the wetting and flow characteristics of several high-temperature brazing alloys on TSF-grade graphite. A small high-frequency induction furnace was set up to braze graphite T-joints in helium. The graphite was vacuum-outgassed for 1 hr at 1000°C prior to the brazing experiments. Most of the brazing alloys tested contained large percentages of zirconium or titanium and were therefore expected to possess low coefficients of thermal expansion. An alloy with a low coefficient of thermal expansion is desirable because of the low coefficient of thermal expansion of graphite. Promising results were obtained with a 48% Zr–48% Ti–4% Be alloy, and a sample button has been prepared for corrosion testing in fused salts. (AEC Activity 4103.1B)

Removing Gaseous Contaminants from Graphite. – An attempt to getter gaseous contaminants from graphite was made by placing a graphite (AGOT grade) crucible and fuel 130 ($LiF-BeF_2-UF_4$, 62-37-1 mole %) in separate areas of an enclosed Inconel container. The system was then held at 1300°F for 20 hr in a vacuum of less than 1 μ. Radiographic examination of the system following the treatment revealed that uranium oxide had precipitated from the fuel 130 and thus had gettered the oxygen outgassed from the graphite. A new charge of fuel 130 was then placed in the graphite crucible without exposing the system to air, and the container was again heated for 20 hr

at 1300°F. Examination of the salt in the graphite crucible showed that uranium oxide had precipitated, but in a lesser quantity than in similar tests in which there was no pretreatment of the system. (AEC Activity 4103.1B)

GENERAL REACTOR RESEARCH

Reactor Evaluation Studies. – Effective resonance integrals as a function of concentration were obtained for U^{235} by multigroup, Monte Carlo, and statistical-model calculations. The same cross-section data were considered in the various calculations; the fuel concentration was varied from 1 to 200 g/liter. More self-shielding was obtained with the Monte Carlo method than with the multigroup method. The results from the statistical-model calculation agreed within 5% with the Monte Carlo results for both U^{235} and Pu^{239}, but agreement was relatively poor for Pu^{241}. Greater self-shielding was always associated with the statistical-model calculation.

An IBM-704 code was written to calculate the effective neutron temperature in heterogeneous reactors as a function of the fuel and moderator cross sections and the cell disadvantage factors. (AEC Activity 4119)

Basic Reactor Research: Neutron Decay Constant in Beryllium. – Measurements of the decay constant of the neutron density in beryllium blocks of dimensions 20 × 20 1/8 × 20 1/8 in., 16 × 14 3/8 × 14 3/8 in., and 14 × 14 3/8 × 14 3/8 in. have been made at temperatures from 0 to 500°C. The value of the transport mean free path of thermal neutrons in beryllium as a function of beryllium temperature has been determined. Measurements are in progress on an 8 × 8 5/8 × 8 5/8 in. beryllium block in an attempt to obtain the variation of the diffusion cooling coefficient as a function of temperature.

A refrigeration unit to be used to cool a 20 × 20 × 20 in. beryllium block to -100°C for further experiments has been ordered.

Work has been continued to improve the operation of the 300-kv accelerator. (AEC Activity 4202)

Basic Reactor Research: Measurement of Eta. – During the period March 26 to May 13, four measurements of the absolute value of eta of U^{233} by the manganese bath technique were made. Values of 2.311, 2.287, 2.282, and 2.283 were obtained, and a weighted average of 2.29 ± 0.02 was computed.

A first measurement of eta of U^{235} by the same technique was completed, but corrections to the experimental result have not yet been completely evaluated.

Additional measurements of eta of both U^{233} and U^{235} are in progress. (AEC Activity 4202) (See Activity 5220 for additional details.)

Basic Reactor Research: Reactor Calculations. – The number of fast fissions per thermal fission in the LTSF source plate has been calculated by a successive generations technique, and the value is about 0.02.

The diffusion cooling constant has been calculated exactly in the heavy-gas model for a bare, homogeneous pulsed assembly. It is found to be 4/3 times the value obtained with Nelkin's variational method with a Maxwellian trial function.

An analysis has been made of the factors determining the signal-to-noise ratio obtainable with a pile oscillator. (AEC Activity 4202)

Fuel Element Catastrophe Studies. – Initial measurements of the release of volatile fission products and radioactive particles during oxidation and melting of irradiated Hanford uranium in air and/or steam were completed.

The rate of oxidation of highly irradiated uranium was about 50% higher than that of unirradiated metal. Uranium irradiated in the ORNL Graphite Reactor to very low burnups showed the same oxidation rate as unirradiated metal. In steam-air mixtures the oxidation rate was noticeably suppressed.

At 1000°C, in 10 min, and at a trace concentration of fission products (low burnup), a typical sample was 25.6% oxidized and released 4.8% of the total gamma activity, 55.6% of the rare gases, 8.0% of the iodine, 6.5% of the tellurium, 0.6% of the cesium, 0.005% of the barium, and 0.22% of the ruthenium.

Under similar time-temperature conditions, fully irradiated Hanford metal in one case was 53.2% oxidized and released 0.1% of the total gamma activity, 100% of the rare gases, 67% of the iodine, 2.8% of the cesium, 0.13% of the ruthenium, 0.008% of the strontium, and 0.008% of the cerium. At 1200°C, 3.5% of the plutonium was released and was found on the furnace walls.

Of eight samples burned at temperatures of 1000 to 1200°C, only one gave a detectable amount of particulate material outside the furnace tube (other than volatile iodine and cesium). In this case black solid particles 0.3 to 1.0 μ in diameter, assumed to be UO_2, were collected on a Millipore filter. (AEC Activity 4252)

Reactor Safety Studies. – One set of standard fuel elements for the BSF stainless steel–UO_2 reactor (BSR-II) has been completed. Two of the required four control-rod elements have been completed, and one control-rod drive element has been assembled. A major portion of the special equipment required to conduct tests at SPERT I has been completed, checking the electrical control equipment will be undertaken in the next 30 days, and arrangements are being made to install the reactor in the BSF for testing as permitted by the Safeguards Committee.

The design of the reactivity insertion device to be used in the SPERT I tests has been changed because of fabrication difficulties. The three-plate blade now terminates in a boron-powder-filled can. Calculations have not yet been made to determine the effect of this change.

Meetings have been held with SPERT I personnel at NRTS concerning hardware, wiring, and installation. Questions were satisfactorily resolved. Present scheduling calls for the installation of the BSR-II in the SPERT I facility to begin in September 1959. (AEC Activity 4252)

Power Reactor Fuel Reprocessing: Zirflex Process. – In Zirflex process corrosion tests, Nionel and type 309Nb stainless steel were corroded in boiling 6 M NH_4F–1.0 M NH_4NO_3–0.05 M $Na_2B_4O_7$–0.05 M Na_2SiO_3 at maximum rates of 6.1 and 4.1 mils/month, respectively, in a 500-hr exposure. Monel was corroded at rates of 17 to 23 mils/month during four successive 24-hr exposures to boiling 1 M HF–0.06 M H_2O_2. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Mechanical Processing. — Dummy Yankee fuel elements, UO_2 pellets clad in 0.4-in.-OD, 0.1-in.-thick-wall stainless steel tubing, were sheared to 0.5-, 1-, 2-, 3-, 4-, and 5-in. lengths and were subsequently leached free of uranium with 173% excess of boiling 10 M HNO_3. The leaching times, 40 to 240 min, were greater by a factor of 2 than with 150% excess of 10 M HNO_3. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Darex Process for Yankee Fuel. — In a bench-scale test with unirradiated Yankee fuel samples (stainless-steel-clad UO_2) the complete Darex standard flowsheet was successfully demonstrated; chloride was removed to 440 ppm before dilution to flowsheet concentrations. With a fuel pin irradiated to 1000 Mwd/ton and decayed four months, the waste and recycle-acid streams contained approximately 0.01% of the activity in the dissolver solution. A small quantity of solid residue from the dissolution and feed-adjustment steps, principally iron and silica as shown by analysis, contained no uranium. Overall heat transfer coefficients were 200 to 300 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot (°F)^{-1}$ in a prototype steam-heated titanium feed-adjustment tank during the mixed-acid-removal and recycle-acid-recovery steps. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Miscellaneous. — In corrosion tests to determine the possibility of using Darex equipment for U-Al dissolution, four 72-hr exposures of titanium to boiling 8 M HNO_3—0.05 M $Hg(NO_3)_2$ resulted in an average corrosion rate of 0.73 mil/month. In tests to determine the possibility of using type 304L stainless steel piping for transferring cold (125°F) Darex dissolver solutions, this alloy was occasionally activated, resulting in serious corrosion. Probably the piping will have to be titanium.

From a simulated Darex—Foreign Research Reactor (U-Al alloy) fuel dissolution product [2 M $Al(NO_3)_3$—2 M HCl], chloride was removed to 10 ppm by the standard Darex procedure.

In a 0.15 M acid-deficient Redox solvent extraction run, uranium was extracted, with 0.04% loss, from a typical low-enrichment Darex fuel solution (0.4 M U, 0.8 M stainless steel, 0.004 M Si) with hexone in a 2-in.-dia, 25-ft-high (21-ft extraction, 4-ft scrub) column packed with 1/4- by 3/8-in. stainless steel Raschig rings. This is 20 times greater

than the loss in the same columns with Redox feed prepared from fuel containing no stainless steel. The feed/scrub/solvent flow ratio was 1/0.24/2.5. Flooding occurred at a flow rate of 485 $gal \cdot hr^{-1} \cdot ft^{-2}$. The extraction HETS was approximately 2 ft. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Long-Range Studies. – Previous studies have shown that nitric oxide may be used to induce the volatilization of chloride from nitrate solutions at room temperatures. Nitric acid was also found to be transferred by a similar reaction. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Sulfex Process. – In a 4-hr decladding test in boiling 6 M H_2SO_4 of a Yankee fuel pin (stainless-steel-clad UO_2 pellets) irradiated to 800 Mwd and decayed four months, uranium and plutonium losses were 0.03 and 0.07%, respectively, confirming previous loss data. Spectrographic analysis of the washed solid residue from the decladding and pellet dissolution showed only a trace of uranium. In all cases where the stainless steel was passivated to 6 M H_2SO_4, wrapping the pin with iron wire initiated the reaction.

The stainless steel tube that contains the thorium oxide–uranium oxide pellets of the Consolidated Edison fuel may be dissolved in 6 M H_2SO_4. The uranium loss from pellets of 95.5% ThO_2-U_3O_8 refluxed in 6 M H_2SO_4 in the absence of stainless steel for 6 hr (1 g of pellets per 5 ml of acid) was 0.5%. In identical tests with 95.5% ThO_2-UO_2 the uranium loss was only 0.02%. The presence of 0.5 M HNO_3 in the 6 M H_2SO_4 did not increase the uranium loss from ThO_2-U_3O_8 pellets. Losses from ThO_2-U_3O_8 pellets were <0.1% when stainless steel was simultaneously dissolved at a rate >10 $mg \cdot cm^{-2} \cdot min^{-1}$.

Tests on corrosion during decladding in 6 M H_2SO_4 and core dissolution in 13 M HNO_3–0.04 M HF (Thorex dissolvent) were continued. After 1000 hr of exposure to boiling 13 M HNO_3–0.04 M HF, welded Nionel, heat-treated at 1950°C and air-quenched, showed rates of 1.8 and 4 mils/month from weight loss and metallographic examination data, respectively. In eight cyclic Sulfex-Thorex dissolutions, Nionel corrosion rates varied between 1.8 and 2.7 mils/month. In boiling 6 M H_2SO_4 titanium was corroded at very high rates. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Darex Process for APPR Fuel. – At room temperature, air-sparging of a solution of APPR fuel in 5 M HNO_3–2 M HCl for 12 hr decreased the chloride insufficiently for process use, that is, from 0.65 to 0.41 M. At boiling temperature in 8 M HNO_3, either air- or nitrogen-sparging decreased the chloride ion concentration from 0.5 to 0.04 M. Filtration of the chloride-free solution of APPR fuel through bare porous metal appears feasible in a production plant, no difficulty in recovering uranium from the filter cake being anticipated. In a typical experiment with 10 liters of solvent extraction feed from APPR-type Darex head-end treatment, 35.6 in.^{3} of wet filter cake was collected on a 0.66-ft^2 filter. The cake was washed three times with 5 liters of water per wash. The uranium loss after one wash was <0.1% and after three washes was 0.005%.

Several gelation and filter-aid media were tested for removing silica from dissolver product, prior to chloride removal, at reasonable filtration rates. The best were 0.004 M HF with Celite 545 and 0.004 M HF alone; Celite 545 alone, activated alumina, ammonium molybdate, and silica gel were all unsatisfactory. Freezing of the solution was also ineffectual. Centrifugation in a 5-in.-dia solid-bowl centrifuge at 3000 rpm completely removed siliceous material at centrifugate flow rates up to 400 cc/min. The results of experiments with a 2.5% TBP solvent extraction flowsheet indicated that 5.5 extraction and 3.5 stripping stages are required for satisfactory uranium recovery. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Darex Process for Consolidated Edison Fuel. – Consolidated Edison fuel samples (stainless-steel-clad ThO_2-UO_2) were declad with 4 M HNO_3–2 M HCl with 0.2 and 0.02% loss of uranium and thorium, respectively. With 5 M HNO_3–2 M HCl, uranium losses were 0.3 to 0.4%. Losses were not increased by exposure of the core oxide to depleted Darex solution (1.3 M HNO_3–1.5 M HCl–50 g of stainless steel per liter) for times up to 3 hr at boiling.

In scouting tests, Consolidated Edison fuel dissolved completely in dilute aqua regia containing fluoride catalyst, producing a solution from which uranium and thorium may be recovered by solvent extraction. In preliminary Thorex solvent extraction experiments, excess aluminum ion (beyond

the amount that must be added to complex the fluoride ion in order to decrease corrosion in the dissolver) was not required for salting. This differs from the processing conditions for thorium metal, where $Al(NO_3)_3$ must be added. With the stainless-steel-containing fuel, salting strength is provided by the dissolved stainless steel.

Decladding in aqua regia and core dissolution in boiling Thorex dissolvent (13 M HNO_3–0.04 M HF) are being considered. Titanium, a possible material of construction, was corroded in the vapor phase at a rate of 8.1 mils/month in 756 hr of exposure. When 0.04 M Al^{+++} was present, the maximum rate was 1.7 mils/month. With <0.04 M Al^{+++}, pitting occurred. After 13 consecutive Darex decladding–Thorex dissolvent dissolutions, the maximum corrosion rate of titanium coupons was <0.2 mil/month. In a 253-hr test, titanium A-55 in the vapor and liquid phases of a modified boiling Thorex dissolvent (13 M HNO_3, 0.04 M F^-, 0.1 M H_3BO_3) containing 0.0, 0.5, and 1.0 M thorium from dissolved Consolidated Edison pellets was corroded at a maximum rate of 0.6 mil/month. The boric acid might be a useful inhibitor for vapor-phase corrosion.

Thorium and uranium were >99.9% extracted with 42.5% TBP in Decalin from a solution containing 1.5 M Th, 0.085 M U, 4.0 M H, 0.1 M Al, and 0.04 M F. Extraction with 42.5% TBP in Amsco was unsatisfactory because of third-phase formation. The scrub in both cases was 5 M HNO_3. Uranium only was extracted under similar conditions with 5% TBP in Amsco. (AEC Activity 4301)

Fused Salt–Fluoride Volatility Processing. – Fuel elements with high zirconium contents may be reprocessed by dissolution in molten fluoride in the presence of HF prior to volatilizing uranium with fluorine. In three dissolution experiments at 700°C in an INOR-8 dissolver using the 43-57 mole % NaF-LiF salt system, the effect of HF feed rate and initial ZrF_4 content on the dissolution rate of Zircaloy-3 was determined. In runs at 2.5 and 6 lb of HF per hour, average dissolution rates were 1.9 and 2.7 $mg\cdot(cm^2$ of original surface$)^{-1}\cdot min^{-1}$ with no initial ZrF_4. When the experiment with a feed rate of 6 lb/hr was repeated with 13 mole % ZrF_4 initially present, the average dissolution rate decreased to 2.0 $mg\cdot cm^{-2}\cdot min^{-1}$. The melting point of the salt containing 13 mole % ZrF_4 is approximately

the same as that of zirconium-free salt (650–660°C), but the dissolution temperature could be decreased to 650°C in the range 16 to 56 mole % ZrF_4 with a corresponding decrease in corrosion rates.

At higher temperatures, zirconium or Zircaloy reacts with HF in the absence of fused salt. A nine-plate Zircaloy-3 fuel assembly was reacted with HF at 1025°C at an average reaction rate of 6 $mg \cdot cm^{-2} \cdot min^{-1}$ based on the initial area (0.72 ft^2). The average HF utilization was 47%. The run was terminated after 85% reaction (3 hr) in order to measure the remaining reacting area (0.16 ft^2). (AEC Activity 4302)

Waste Disposal Research and Engineering: Field-Scale Experiment in Natural Salt Formations. – Excavation of the gas-seal trenches around the two cavities in the Hutchinson mine was completed. One hundred two 1 5/8-in.-dia wells around the cavities for thermocouples and moisture probes have been drilled. Additional core holes in the vicinity of the experimental area have been drilled to define the stratigraphy. The salt deposit 10 ft below the floor, as exposed by the excavated pits, is much less pure than the mined zone above it. Chemical and mineralogical analyses of sections (1 in. wide × 1/4 in. thick) from a side wall of each cavity and the core driller's logs will be used to prepare a geologic cross section of the floor in the area.

Cooperative arrangements with the Applied Physics Laboratory of the U.S. Bureau of Mines, the Department of Civil Engineering of the University of Texas, the Department of Geology of the University of Kansas, and the Geotechnical Corporation (Dallas) for the measurement of "flowage" were established. Each group will employ particular techniques and instruments to measure the strain (creep) created in the salt by the heated wastes. (AEC Activity 4352)

Waste Disposal Research and Engineering: Laboratory Studies of the Chemical Interaction of Waste and Salt. – Thin sections of rock taken from one side wall of each of the cavities in the Hutchinson mine, as well as representative samples from the cores recovered at each corner, were chemically analyzed for NaCl and insoluble material (shale and anhydrite). The south cavity is excavated in rock composed of 78% NaCl and 22% insoluble material, whereas the rock surrounding the north cavity is approximately

72% NaCl. The samples were also examined by x-ray diffraction technique for mineral identification. Halite (NaCl) crystals were identified in all samples representative of different depths; anhydrite ($CaSO_4$) and clay minerals (aluminosilicates) make up a large fraction of many samples. Of the clay minerals identified, illite was present in greatest amount; kaolinite and vermiculite were present in lesser amounts. These findings point to the unreliability of logs prepared by visual inspection of cores in the mine. The cores will have to be brought out of the mine and washed, and selected samples (or the whole core) analyzed chemically and mineralogically. (AEC Activity 4352)

Waste Disposal Research and Engineering: Disposal of ORNL Radioactive Wastes. – It has been observed that potassium treatment of vermiculite results in a collapse of the crystal lattice accompanied by an extreme increase in the affinity of the vermiculite for cesium. Potassium treatment of Arizona bentonite and Wyoming bentonite did not yield strong diffraction peaks at 10 Å, and the increased affinity for cesium was less marked than in the case of vermiculite. Also, the cesium sorption curves obtained were not comparable to the illite cesium sorption curve, as was the curve obtained for potassium-treated vermiculite.

Sodium vermiculite has been treated with potassium to yield vermiculite with less than 100% potassium saturation. X-ray diffraction data indicate that collapse occurs between some plates at potassium concentrations lower than those at which it occurs between others. This is evidenced by a continuous increase in the magnitude of the 10-Å peak with increasing potassium concentration rather than an abrupt change in spacing from 14 to 10 Å at some critical concentration. Exchange-capacity measurements indicate that the exchange capacity of the vermiculite is reduced by an amount equivalent to the amount of potassium sorbed. Cesium sorption curves will be obtained for these vermiculites and will be correlated with x-ray diffraction data for further evaluation of the role of structure in cesium ion exchange. (AEC Activity 4352)

Waste Disposal Research and Engineering: Corrosion Studies. – Tests of Haynes alloy No. 25, welded at ORNL, at 80°C in acid Purex waste saturated with NaCl revealed intergranular corrosion in areas adjacent to the welds. Localized attack did not occur away from the welds.

Sample	Welding Procedure	Corrosion Rates (in./year)		Remarks
		36 days	Last 24 days	
3G 1A	Heliarc with backup gas	0.0107	0.0103	Very severe intergranular corrosion next to weld; about 1/3 penetration
3G 2A	Heliarc without backup gas	0.0082	0.0061	General intergranular attack without appreciable penetration
3G 3A	Metallic arc	0.0085	0.0057	General intergranular attack without appreciable penetration

Specimens of butt-welded Haynes alloy No. 25 pipe have been exposed for 129 days at 80°C in acid Purex waste saturated with NaCl without visual sign of intergranular corrosion. A sample of the pipe containing a weld corroded at a rate of 0.003 in./year, and one without a weld corroded at a rate of 0.001 in./year. The cause of this attack is being investigated by ORNL and Haynes Stellite Company metallurgists. (AEC Activity 4352)

Waste Disposal Research and Engineering: Treatment of ORNL Wastes. – During 1958, approximately 330,000 ft^3 of solid waste was disposed of in the existing burial ground. ORNL contributed about 155,000 ft^3 or 47.0% of this total, Y-12 contributed 15,000 ft^3 or 4.5%, and off-site agencies contributed 160,000 ft^3 or 48.5%. Knolls Atomic Power Laboratory (52,000 ft^3), Argonne National Laboratory (34,000 ft^3), and the General Electric Company of Evendale, Ohio (19,000 ft^3), were the principal off-site contributors. (AEC Activity 4352)

Long-Range Planning Studies. – The analytical expression has been obtained which relates Cf^{252} production to exposure time and flux, utilizing Pu^{242} as the starting material. This expression was programed on the IBM-704. Starting with 100 g of Pu^{242}, the Cf^{252} production after two years of

exposure in an average flux of 3×10^{15} neutrons·cm^{-2}·sec^{-1} would be about 100 mg.

Additional nuclear calculations were performed for the HFIR (High Flux Isotope Reactor). The temperature coefficient (Δk_e per degree C) for the core was -7×10^{-5} at operating conditions. The void coefficient (ratio of Δk_e to the fraction of voids) for the fuel region ranged from -0.25 to -0.41 for initial voids from 0 to 35%. The void coefficient for the island region decreased from 0.02 at zero voids to zero at 25% voids; the total increase in reactivity associated with increasing the island voids from 0 to 25% was about $0.003\Delta k_e$. (AEC Activity 4510)

THERMONUCLEAR PROJECT

DCX-EPA. – This is the experiment in which the large (7-in.-dia) hollow arc is run. The D_2^+ ion source and injector system are now in operation, and a 15-kev D^+ beam has actually been trapped within the hollow carbon arc, (AEC Activity 4851)

DCX-1. – The regular DCX machine is being returned to operation after a long shutdown. The destination of the undissociated H_2^+ beam has been determined, so that a means of disposing of it can be designed. Some noise measurements on the carbon arc in DCX-1 have revealed a noisy mode of operation and a quieter one, The former can be transformed to the latter by increasing the magnetic field, increasing the arc current, or adding gas. The efficiencies of dissociation of H_2^+ in the two modes have not yet been compared. (AEC Activity 4851)

Theory. – A generalized theory of burnout has been established which presumes gas breakup (as in the Russian OGRA) and includes the effects of plasma pumping and of the gas contributed by the input beam. (AEC Activity 4851)

GAS-COOLED REACTOR PROJECT

A task force consisting of personnel from Kaiser-Engineers, the Nuclear Power Department of Allis-Chalmers Manufacturing Company (formerly a division of ACF Industries), AEC-ORO, and ORNL prepared a preliminary hazards analysis of the Experimental Gas-Cooled Reactor (EGCR), which is

to be built at an Oak Ridge site. This hazards summary report was submitted to the Advisory Committee on Reactor Safeguards (ACRS). An oral presentation is to be made to the ACRS in July. (AEC Activity 4110)

LITR Irradiation Experiments. – The fission products were recovered from experimental assemblies L-2b, L-14, and L-15, which were irradiated in the LITR to obtain data on the evolution of gas from UO_2 at high temperatures. Samples of the fission product gases evolved are being analyzed. The leak in capsule L-2 that resulted in loss of the evolved gases was found at the terminal of one of the internal thermocouple leads. Data from the five experimental assemblies removed from the LITR thus far are being assembled and correlated. (AEC Activity 4110)

Instantaneous Fission Gas Release Analysis. – Data were obtained for the release of fission gas from UO_2 as a function of temperature. For the first sets of data over the range 1200 to 1700°F at a constant fission rate, the gas release followed an exponential function. Preliminary analyses of data taken thus far (based on the activity of the entrained gamma-emitting fission products) indicate that 0.05% of the total amount of fission products generated escapes and approximately 1% of the products that might be expected to be volatile escapes at 1700°F. These calculations are based on theoretical predictions of gross fission product activities. The half lives of the observed products agree with predicted values. (AEC Activity 4110)

Graphite and Coating Studies. – The graphite specimen that had been coated to reduce permeability to gaseous material and then irradiated for approximately 5×10^{20} nvt is undergoing a high-temperature (1100°C) oxidation test in air. As was described previously, the specimen was undamaged by the irradiation, and no effects of exposure to air at 1100°C have yet been observed. Two similar samples were discharged from the reactor after approximately 7×10^{20} nvt. Two weeks will be allowed for decay before these samples are examined. (AEC Activity 4110)

Closed-Cycle Experiment. – The first fuel capsule was installed in the ORR and located at the center line of the reactor. With the reactor operating at 7 Mw, all components of the experimental assembly were functioning properly, except that the thermocouple on the sample read too low.

A better method will be sought for this thermocouple installation. For this run the sample temperature will be estimated from the shell temperature.

After 1 hr of operating time, the reactor was shut down because of high building air activity. Because of the possibility that this may have been caused by the B-9 experiment, the capsule was withdrawn to 2 ft 6 in. above the center line of the reactor. The loop gas was discharged to the off-gas system, and the loop was left under a negative off-gas pressure. The reactor resumed normal operation.

The experimental system will be checked before the capsule is reinserted to the center line of the reactor. (AEC Activity 4110)

General Electric Test Reactor (GETR) Irradiations. – Stringers 2 and 3, each of which consists of three stainless steel capsules filled with UO_2 pellets, are being irradiated in the GETR, which is operating at 30 Mw. The center temperatures of stringers 2 and 3 are 860 and 730°F, respectively. The uncorrected cladding temperatures vary from 300 to 700°F. The bulk gas temperature is 390°F. (AEC Activity 4110)

Radiation Effects in Structural Steel. – Operation of experimental assemblies ORR-3 and ORR-6 is continuing to be satisfactory. The second set of ORR-3 flux monitors was disassembled and transferred to the counting laboratory.

Testing of Charpy specimens in the hot cell is in progress. The Charpy machine operates very well in the hot cell, and the only modifications planned are to improve temperature control and to decrease the difficulty of moving the machine into and out of the hot cell. (AEC Activity 4110)

Graphite-UO_2 Reactions. – Pellets of UO_2 and graphite containing 50 wt % UO_2, -60 and -80 mesh, were fired for 100 hr at temperatures of 1350 and 1425°C. It was found that, under the test conditions, samples fired at 1350°C did not form amounts of carbide sufficient to be detected by x-ray diffraction or weight loss. However, as much as 50% of the UO_2 was reduced to the carbide in the samples fired at 1425°C. The product from the reaction was identified as UC.

These pellets were fired in a graphite container packed with lampblack, which should generate a partial pressure of CO during the test. The reaction will be studied at 1350°C and at lower temperatures in a system containing no carbon, other than that in the pellet, to determine whether the CO present in the graphite system had a partial pressure greater than the equilibrium pressure of the reaction. (AEC Activity 4110)

PROGRAM 5000 – PHYSICAL RESEARCH

Reactor Operations. – Operation of the ORR was routine during the last cycle. The safeguard report for annealing the stored energy in the Graphite Reactor apparently will not be approved until it has been reviewed by the Advisory Committee on Reactor Safeguards. This will probably cause the annealing to be delayed until next summer.

PHYSICS

High-Voltage Experimental Program: Manganese-Bath Eta Measurement. – Recent interest in the thermal Th^{232}-U^{233} breeding cycle has emphasized the need for a highly accurate determination of eta, the yield of fission neutrons per thermal neutron absorbed. An independent measurement accurate within 1% appears possible, the old manganese-bath technique being used for energy-independent detection of the neutrons. A beam of thermal neutrons is passed through a large (1-m) manganese bath to its center, either activating the bath directly, or being stopped in a "black" sample of U^{233} whose fission neutrons then activate the bath. The activity ratio measures eta. Most background and epithermal effects can be eliminated by taking cadmium differences. Highly precise monitoring of the beam intensity is achieved by using in the beam a thin sample of manganese, whose activity can then be compared with a sample of the bath solution under favorable, low-background counting conditions. Preliminary results give values of 2.27 to 2.31 for U^{233} and 2.02 to 2.08 for U^{235} at 2200 m/sec. (AEC Activity 5220)

High-Voltage Experimental Program: Total-Cross-Section Measurements in the Kilovolt Range. – Two instrumental improvements have been made in the measurement of the total cross sections of separated isotopes in the kilovolt range of neutron energies, which add to both the quality and the quantity of the data. The short-term resolution now attainable is 8 mμsec/m at 3 kev and less at higher energy. The long-term resolution is correspondingly 10 mμsec/m because of instrumental drift. In addition, the data are now punched on tape to permit processing on the Oracle in the same manner as the data from the fast chopper.

The most recent set of measurements included the following isotopes in the energy range 3 to 8 kev (the figures in parentheses are the energies at which resonances were observed): Co^{59} (4.22, 4.81, 4.97); Cu^{63} (4.33, 4.76, 5.28, 5.72, 7.40, 7.79); Ge^{70} (4.33, 4.95, 5.52, 6.65); Ge^{72} (4.52, 4.92); Ge^{74} (4.93); Ge^{76} (4.95?); Rb^{85} (3.18, 3.97?, 4.16, 4.87, 5.10, 5.58); Rb^{87} (3.76, 5.04); Sr^{86} (3.18) (observed only to 4.5 kev); and Sr^{88} (no resonances between 3 and 6.5 kev). (AEC Activity 5220)

Neutron Diffraction. – Neutron diffraction investigations on HoN have shown that this compound becomes ferromagnetic at low temperatures, and measurements of the magnetic intensities as a function of temperature predict that the transition occurs at about 18°K. Calculations based on the coherent magnetic intensities and on the difference in diffuse scattering above and below the Curie point give a value of 8.0 ± 1.0 Bohr magnetons for the atomic magnetic moments in the ordered ferromagnetic lattice. This value is considerably lower than the moment calculated for the free Ho^{+++} ion (10.0 Bohr magnetons), but the diffraction patterns at temperatures down to 1.30°K indicate an appreciable amount of inelastic scattering which has not been analyzed. Experiments performed in an external magnetic field show that the ferromagnetic moments are aligned parallel to the cube edges and that they cannot be rotated out of this direction with fields of 12 kilo-oersteds. Since relatively few nuclear reflections are observed from this compound, the paramagnetic scattering at high temperatures can be well determined, and this determination gives the magnetic form factor associated with the spin and orbital contributions to the atomic moments of the 4f electrons in the Ho^{+++} ion. (AEC Activity 5220)

Low-Temperature, Nuclear and Solid State Physics. – Preliminary results have been obtained on the angular distributions of alpha particles emitted from aligned Np^{237} nuclei in $NpO_2Rb(NO_3)_3$ at temperatures down to ~0.22°K. A complete experiment (germanium surface-barrier counter, rotatable single-crystal sample, angle-measuring devices, and a thermometer) was enclosed in a container filled with He^3 gas and thermally attached to a large (1/2 mole) body of paramagnetic salt, which could be magnetically cooled to low temperatures. The measurements indicate (1) that the alpha

particles are preferentially emitted from the tips of the prolate Np^{237} nuclei and (2) that the bonding calculations of Eisenstein and Pryce [Proc. Roy. Soc. (London) A229, 20 (1955)] do not adequately reflect the detailed structure of the electron cloud in the Np-O covalent bond.

A new apparatus has been used to study the angular distribution of fission fragments produced in thermal neutron bombardment of $U^{233}O_2Rb(NO_3)_3$ down to a temperature of ~1.3°K. The measurements show that the fission fragments are emitted isotropically within ~1%, even while alpha particles from the same group of nuclei are emitted anisotropically, in accord with earlier results (ORNL-2302, p 31). (AEC Activity 5220)

CHEMISTRY

Neptunium Recovery. – The neptunium (IV) loading of the solvent in extraction by 0.025 M trilaurylamine in xylene from 1 N HNO_3–5 N NH_4NO_3 was ~0.005 M; thus the molar ratio of amine:Np approached 4:1. Loading was similar with 0.025 M Alamine 336 (an n-octyl-n-decyl tertiary amine), but a second organic phase separated from 0.05 M triisooctylamine on loading. (AEC Activity 5310)

Extraction Reagent Performance. – Three aromatic hydrocarbons – isopropylbiphenyl, amylnaphthalene, and methylbiphenyl – as 5 or 10 vol % solutions in Amsco 125-82 did not appreciably change the rate of decomposition of tributyl phosphate (TBP) that was irradiated to 200–400 whr/liter. The yields of gas and of monobutylphosphoric acid (MBPA) were essentially unaffected by the presence of the aromatic hydrocarbons. In each test, 300 g of TBP was diluted to 1 liter with one of the above solutions and was contacted with 2 M HNO_3 prior to the irradiation. In the absence of nitric acid, MBPA yields are 0.1 to 0.15 molecule per 100 ev of energy absorbed. The MBPA yields in these experiments, 0.42 to 0.8 molecule per 100 ev of energy absorbed, support previous conclusions that prior contact with nitric acid increases the rate of hydrolysis of the primary acid product of radiolysis, dibutylphosphoric acid, to MBPA. (AEC Activity 5310)

Ion Exchange Technology. – Equilibrium uranium loadings were determined for Dowex 21K anion exchange resin from 0.3 to 0.5 M sulfate solutions at pH 2. The results at a given total sulfate concentration could be fitted by a Langmuir sorption isotherm. For 0.3 M sulfate the Langmuir constants were A = 1.33 liters per gram of dry chloride-form resin and B = 1410 liters per mole of uranium. For 0.5 M sulfate the A and B values were 0.86 and 920, respectively.

The rates of uranium loading from 0.025 to 0.20 M sulfate were measured, and from the results apparent uranium diffusion coefficients in the range 1.2×10^{-7} to 1.6×10^{-7} cm^2/sec were calculated. (AEC Activity 5310)

Solvent Extraction Research. – Although Pu(III) is much less extractable than Pu(IV) or (VI), its extraction is sometimes desired in order to maintain another metal at a favorable valence, for example, Np(IV) or U(IV), or to avoid the need of oxidizing a major solution component, for example, Fe(II). Primene JM-T (primary amine), 0.1 M, extracted Pu(III) with coefficients greater than 10 from 3 M H_2SO_4 containing 0.7 M Fe^{++} and 0.01 M Fe^{+++}, and 0.1 M triisooctylamine extracted Pu(III) with coefficients greater than 100 from 6 M nitrate containing 0.05 to 0.1 M acid and 0.1 M ferrous sulfamate, in which neptunium exists as the highly extractable Np(IV).

A scheme is proposed for recovering uranium and plutonium from sulfuric acid solutions used for removing stainless steel cladding. The U(IV) and Pu(III) are extracted in succession with 0.1 to 0.3 M primary amine and stripped from the combined extract with dilute nitric acid. The addition of 0.01 M Fe^{+++} after the uranium extraction increases the plutonium extraction. For amounts of uranium up to 3 g/liter in the decladding solution a concentration factor of at least 30 appears possible.

In a kinetic study of uranium extraction by tri-n-octylamine sulfate in benzene at constant activities of amine, uranium, and acid but with varying sulfate concentraction, the uranyl ion was three times as important as the neutral monosulfate complex in transferring uranium to the organic phase, and the neutral complex was twice as important as the dinegative disulfate complex.

The extraction of rare earths from concentrated nitrate solutions by tributyl phosphate varied greatly with the cation. The distribution coefficients for europium between 1 M tributyl phosphate in kerosene and 5.6 N nitrates ranged from 450 for $LiNO_3$ to 2 for NH_4NO_3. The order of the distribution coefficients for the nitrates tested was $LiNO_3 > Al(NO_3)_3 > Mg(NO_3)_2 > Zn(NO_3)_2 > Cu(NO_3)_2 > Fe(NO_3)_3 \cong NaNO_3 \cong Ca(NO_3)_2 > NH_4NO_3$. (AEC Activity 5310)

Chemistry of Heavy Elements. – The thermal neutron absorption cross section of Th^{233} and the resonance absorption integrals of Th^{232}, Th^{233}, and Co^{59} were found to be 1450 ± 100, 85 ± 8, 400 ± 100, and 75 ± 5 barns, respectively, by neutron activation measurements with reactor-spectrum neutrons and cadmium filters. These values are based on 2200-m/sec cross sections of 37.0 and 99.0 barns for cobalt and gold, respectively, and a resonance integral of 1560 barns for gold. (AEC Activity 5330)

Chemistry of Corrosion. – Cathodic polarization measurements were made on iron electrodes that were first passivated in oxygenated solutions of certain inhibitors of corrosion. The results show that the cathodic process consists chiefly in the reduction of oxygen, even when such potentially good oxidizing agents as chromate and pertechnetate ions are present; it was demonstrated also that such nonoxidizing inhibitors as benzoate and phthalate ions enable oxygen to produce passive potentials on iron. These results invalidate the argument, often made, that the only function of oxidizing inhibitors is to augment the cathodic oxygen current by their own reduction.

Precise experimental values of the dissociation quotient of the bisulfate ion were obtained by using potentiostatic and galvanostatic techniques. An accurate knowledge of this quotient is necessary in interpreting measurements in progress on the effects of hydrolysis of uranyl ion and of complexing of uranyl and sulfate ions on the kinetics of reduction of uranyl species on passive stainless steel in aqueous sulfate medium. For an ionic strength of 0.3 and over a temperature range from 22 to 85°C, the dissociation quotient K of the bisulfate ion is given by $\log_{10} K = -5.93 + 1.39 \times 10^3 (1/T)$, where T is the absolute temperature. The standard error of the least-squares estimate is 0.059. (AEC Activity 5330)

Physical Chemistry of Ion Exchangers. – Measurements of the standard free energies, heats, and entropies for the silver-sodium ion exchange were completed. Five differently cross-linked Dowex 50 cation exchangers were used. The integral standard heat values were positive, despite the fact that the differential heats of exchange became negative below equivalent fractions of silver of 0.1 to 0.2. The standard entropies of exchange were also positive. The unusual behavior of the enthalpy and entropy changes in this exchange appears to be consistent with the extensive ion association that occurs with silver salts in aqueous solutions. Negative contributions to the heat of exchange conceivably may arise through the formation of charge-transfer compounds of benzene and silver ions.

Studies of the bromide-chloride exchange reaction with liquid anion exchangers at high concentrations of HCl showed that the uptake of micro amounts of bromide ion could be accounted for quantitatively when corrections were made for thermodynamic ionic activity coefficients in the concentrated aqueous phase. The nonideal behavior of the ions in the ion exchanger evidently was not an important factor in determining the equilibrium partition of bromide ion. (AEC Activity 5330)

Chemistry of Technetium. – Pertechnetate ion (TcO_4^-) in HCl solutions is reduced to Tc(V) by Cl^-. The spectrum of the Tc(V) species exhibits two strong absorption maxima at 2300 and 2925 Å with molar absorbancy indices of 10,400 and 4700, respectively. A spectrophotometric study showed that the Tc(V) formed from TcO_4^- in HCl solutions is proportional to the sixth power of the HCl concentration, indicating a species $TcOCl_4^-$, or less probably $TcO_2Cl_4^{---}$. From the equilibrium data the free energy of formation of $TcOCl_4^-$ is calculated to be -127 kcal/mole. This gives -0.5 v for the potential of the $TcOCl_4^-$ –TcO_4^- couple. The $TcOCl_4^-$ ion is stable with respect to disproportionation into TcO_4^- and $TcCl_6^{--}$ but is slowly reduced to $TcCl_6^{--}$ by Cl^- in strong HCl solutions. This information, together with the above potential, permits an estimation of -0.3 to -0.4 v for the potential of the $TcCl_6^{--}$–$TcOCl_4^-$ couple. (AEC Activity 5330)

High-Temperature Aqueous Solution Chemistry. – Experiments were completed in which osmotic coefficients were determined at 99.6°C, extending the previously studied concentration range in most cases to saturated solutions of NaCl, KCl, LiCl, CsCl, K_2SO_4, UO_2SO_4, $BaCl_2$, and $MgSO_4$. At concentrations near the solubility limit the experimental precision was about 1%. In no case was a change in concentration associated with a change in the sign of the difference between the osmotic coefficient at 99.6 and that at 25°C.

The reaction rate constants for the homogeneously catalyzed activation of molecular hydrogen and deuterium by cupric ion were determined at 100, 110, 120, and 130°C for the following systems:

20 atm H_2, H_2O, 0.1 M $Cu(ClO_4)_2$, 0.01 and 1.0 M $HClO_4$

20 atm D_2, H_2O, 0.1 M $Cu(ClO_4)_2$, 0.01 and 1.0 M $HClO_4$

20 atm H_2, D_2O, 0.1 M $Cu(ClO_4)_2$, 0.01 and 1.0 M $DClO_4$

20 atm D_2, D_2O, 0.1 M $Cu(ClO_4)_2$, 0.01 and 1.0 M $DClO_4$

The rate constants were lower at the higher acidity in every case.

Investigation of the four-component system UO_3-CuO-SO_3-H_2O at 300°C (see "HRP Chemistry," ORNL-2736) was extended to include the solubility equilibria along a binary curve in which $UO_3 \cdot H_2O$ and $CuO \cdot 3UO_3$ are believed to be the solid phases. Investigation of the UO_3-HNO_3-H_2O system from 150 to 300°C was extended to 5 m HNO_3. The solid phase in equilibrium with solutions having a high concentration of HNO_3 was always $UO_3 \cdot H_2O$. (AEC Activity 5330)

Chemical Separation of Isotopes. – Single-stage fractionation factors, α, were measured for both oxygen and carbon isotopes between the liquid and vapor phases of amine carbamates ($RNHCO_2NH_3R$ or $R_2NCO_2NH_2R_2$) at or near their boiling points. The CO_2 for isotopic analysis was liberated by decomposing the carbamates with an excess of concentrated H_3PO_4. The separation factors for sec-butylamine carbamate and isobutylamine carbamate were measured directly by a single equilibration; the others were calculated from data obtained from multistage distillations. Oxygen-18 and carbon-12 were found to concentrate in the gas phase.

Amine Carbamate	Temperature (°C)	Single-Stage Factor (α)	
		$\left(\frac{O^{18}}{O^{16}}\right)_g \Big/ \left(\frac{O^{18}}{O^{16}}\right)_l$	$\left(\frac{C^{13}}{C^{12}}\right)_g \Big/ \left(\frac{C^{13}}{C^{12}}\right)_l$
Methylethyl	60	1.0032	
Dimethyl	62	1.0039	0.9984
sec-Butyl	76	1.0060	0.9976
n-Propyl	82	1.0014	0.9998
Isobutyl	86	1.0063	0.9972
n-Butyl	91	1.0044	0.9995

(AEC Activity 5340)

PROGRAM 6000 – BIOLOGY AND MEDICINE

BIOLOGY

Cytology and Genetics. – The enzymes involved in starch synthesis and degradation in maize endosperm are being studied. The experiments are designed to compare the enzymes produced by a series of mutants that show abnormal starch synthesis.

When enzymes from many of the starch mutant stocks which show different rates of migration in starch gel electrophoresis are mixed, they form only a single band in the mixtures, rather than two bands as expected. This does not necessarily mean that the enzymes are identical. The rate of migration of an enzyme band was found to be greatly accelerated when it moves through a region of the gel which had previously been traversed by the same enzyme. Preliminary studies indicate that the altered enzymes which do not complex in mixtures to form a single band in starch gel electrophoretic separation also do not cross-react immunologically, as determined by the agar diffusion technique. (AEC Activity 6130)

Pathology and Physiology. – Spleen cells of C57BL mice injected intraperitoneally (150 million cells per recipient) into sublethally (500 to 600 r) x-irradiated (C57BL × 101)F_1 mice killed all the recipients (29 in group) within 60 days. In contrast, the same number of C57Bl spleen cells injected after overnight incubation at 7°C with homogenized recipient-type liver killed only 17% of the irradiated F_1 recipients (29 in group) within 90 days. However, the C57BL spleen cells similarly incubated with homogenized donor-type liver killed all irradiated F_1 hybrid recipients (17 in group) at the same rate as did the nonincubated spleen cells. Since the majority of the surviving recipients had erythrocytes of C57BL (donor) type, as judged on the basis of hemoglobin solubility and electrophoretic mobility, it is believed that the grafted spleen cells remained functional despite incubation with recipient liver before implantation. These results are interpreted to signify that "immunological paralysis" of the graft may interfere with the foreign spleen reaction. They indicate, furthermore, the existence of a simple procedure for inhibiting the foreign spleen reaction, and possibly also the foreign bone marrow reaction, in vitro. (AEC Activity 6130)

Biophysics. — Intensive quantitative decay studies with the electron spin resonance spectrometer have been made of the magnetic centers produced by x rays in methionine and DL-leucine. The methionine data seemingly exhibit a constant shape, but the decay is characterized by a double exponential with half lives of about 53 and 2.4 hr. Changes in the absorption curve shapes of irradiated leucine clearly demonstrate the presence of at least two and probably more superimposed patterns. The strongest single absorption peak has the greatest half life of perhaps 35 hr, whereas the rest of the pattern taken as a whole has a half life of less than 6 hr. The presence of six resonance absorption peaks has been clearly shown for leucine. All these decay studies tend toward an understanding of the energy storage states and their kinetics for these irradiated materials.

The induction of free radicals by ultraviolet radiation in deoxyribonucleic acid and a number of proteins has been confirmed. Measurements indicate that proteins are slightly less sensitive to the action of ultraviolet than is deoxyribonucleic acid. In one of the simpler protamines, salmine, which shows some structure in the spin resonance pattern induced by ionizing radiation, an identical pattern is produced by ultraviolet irradiation.

The decay or "aging" of resonances induced by ionizing radiation is drastically increased by subsequent exposure to ultraviolet. This effect has been observed in some proteins as well as in the crystalline amino acids. (AEC Activity 6130)

Mammalian Recovery. — S,2-Aminoethylisothiourea (AET) given orally (16.0 mg) protected the intestine of irradiated mice better than AET given intraperitoneally (8.8 mg). In irradiated mice (900 or 1500 r) given AET orally 15 min before exposure, good histological recovery of the intestine was observed on the third and fourth days. This recovery was reduced about 50% when AET was given 30 min before 1500 r. Autoradiographic studies of tissues of normal mice given AET labeled with tritium showed rather diffuse labeling in most organs studied.

Within the first half hour after exposure of mice to 600 r, incorporation of H^3-thymidine by intestinal crypt cells is essentially unimpaired. Four days after doses up to 10,000 r, gut autoradiographs still show labeling of some small epithelial-like cells.

Mouse antisera against homologous-strain tissues appeared to be cytotoxic for hematopoietic cells of this strain, but not for the isologous hematopoietic cells. This phenomenon was used to identify the nucleated cells in lymph nodes, spleen, bone marrow, and blood from lethally irradiated mice treated with homologous fetal liver cells. These cells were shown to be of donor type.

Various additives have been tested for optimum low-temperature preservation of mouse bone marrow cells. The marrow cells are tested in lethally irradiated mice. Tyrode's solution, mouse serum, 7% polyvinylpyrrolidone, 0.5 and 0.1 M glucose, and 0.3 M methanol were not effective additives; 7% dextran, 15 and 25% bovine serum albumin, and 0.3 M ethylene glycol were partially effective; 15% glycerol, 0.5 M lactose, 0.3 M i-erythritol, and 0.3 M mannitol were very effective. The eosin dye method for determining viability of marrow cells after freezing did not correlate well with the ability of cells to enhance recovery in irradiated mice. (AEC Activity 6230)

Radiation Immunology. — The exposure of mice to various doses of x ray to the total body during the period of maximum depression of the secondary antibody response revealed that 800 to 850 r is required to reduce the immune status 50% as compared with normal, nonirradiated secondary responders. In contrast, 600 to 650 r is required to reduce the immune status 50% as compared with normal, nonirradiated primary responders. This difference can be reconciled provided the following information is available: (1) the number of antibody-forming cells at various time intervals after primary antigenic stimulation, (2) the radiosensitivity of antibody-forming cells as a function of maturation vector, and (3) the transmission of information as a function of cell division.

It has been shown previously that the immune status of young mice is low. It has now been found that only 30 to 40% of the four-week-old mice exposed to an LD_{99} dose of x ray can be protected with normal adult isologous bone marrow cells. Furthermore, the protective capacity of 1×10^6 nucleated bone marrow cells was comparable with that of 12×10^6 nucleated bone marrow cells. With increase in age of the recipient, a proportional increase in protective capacity of normal adult bone marrow cells was seen. (AEC Activity 6230)

PROGRAM 8000 — WORK FOR OTHERS

International Reactor Courses at ORNL. — The 26 students currently enrolled in the foreign programs have completed 15 weeks of school since the beginning of classes in February.

Plans have been made for starting a second session of courses in Nuclear Reactor Operations Supervision and Nuclear Reactor Hazards Evaluation this fall. Brochures and applications were distributed in April to 63 foreign countries and international organizations by the Atomic Energy Commission's Division of International Affairs. This distribution was made through foreign embassies and legations in Washington, D.C. Descriptive material for this program is being distributed to industrial organizations and governmental agencies in this country. This session will begin in November 1959, and the deadline for receipt of applications is July 10, 1959.

The Operations Supervision course is limited to ten students, and the Hazards Evaluation course is limited to sixteen. (AEC Activity 8717)

OPEN LITERATURE PUBLICATIONS

Anderson, N. G., R. E. Canning, M. L. Anderson, and R. H. Shellhamer, "Studies on Isolated Cell Components. XI. Rat Blood Proteins," Exptl. Cell Research 16, 292 (1959).

Arnold, W., and H. K. Maclay, "Chloroplasts and Chloroplast Pigments as Semiconductors," p 1 in Brookhaven Symposia in Biology, No. 11: The Photochemical Apparatus, Its Structure and Function, Brookhaven National Laboratory, 1959.

Atwood, K. C., and S. L. Scheinberg, "Isotope Dilution Method for Assay of Inagglutinable Erythrocytes," Science 129, 963 (1959).

Borie, B., "X-Ray Diffraction Effects of Atomic Size in Alloys. II," Acta Cryst. 12, 280 (1959).

Bridges, W. H., and E. L. Long, Jr., "A Technique for the Easy Removal of 'Direct' Replicas for Electron Microscopy," Am. Soc. Testing Materials, Spec. Tech. Publ. No. 245 (1958).

Brosseau, G. E., Jr., "Crossing Over Between Y Chromosomes in Male Drosophila," Drosophila Info. Ser. 32, 115 (1958).

Callihan, A. D., and J. T. Thomas, "Accidental Radiation Excursion at the Oak Ridge Y-12 Plant," Health Phys. 1, 363 (1959).

Dent, J. N., "A Technique for Removing Thyroid Glands from Anuran Larvae," Endocrinology 64, 314 (1959).

Dismuke, S. E.,"Proposed High-Radiation-Level Examination Laboratory," p 305-12 in Proceedings of the 7th Hot Laboratories and Equipment Conference (1959 Nuclear Congress) April 7-9, 1959, Cleveland, Ohio.

Galonsky, A., H. B. Willard, and T. A. Welton, "S-Wave Detector of Deuteron Polarization and 14-Mev Polarized-Neutron Source," Phys. Rev. Letters 2, 349 (1959).

Hollaender, A., C. C. Congdon, D. G. Doherty, T. Makinodan, and A. C. Upton, "New Developments in Radiation Protection and Recovery," p 139-50 in Progress in Nuclear Energy, Series 7, Medical Science, vol 2 (ed. by J. C. Bugher, J. Coursaget, and J. F. Loutit), Pergamon Press, London, 1959; also in Proc. Second Intern. Conf. Peaceful Uses Atomic Energy, Geneva, 1958 23, 3 (1958).

Kelley, M. T., H. C. Jones, and D. J. Fisher, "Electronic Controlled-Potential Coulometric Titrator," Anal. Chem. 31, 956 (1959).

Kimball, A. W., "Disease Incidence Estimation in Populations Subject to Multiple Causes of Death," Intern. Statist. Bull. 36(3), 193 (1958).

Kimball, R. F., "The International Congress of Radiation Research at Burlington," Intern. J. Radiation Biol. 1, 91 (1959).

Kimball, R. F., N. Gaither, and S. M. Wilson, "Reduction of Mutation by Postirradiation Treatment After Ultraviolet and Various Kinds of Ionizing Radiations," Radiation Research 10, 490 (1959).

Lindsley, D. L., and E. Novitski, "Localization of the Genetic Factors Responsible for the Kinetic Activity of X Chromosomes of Drosophila melanogaster," Genetics 43, 790 (1958).

McDowell, B. L., A. S. Meyer, Jr., R. E. Feathers, Jr., and J. C. White, "Spectrophotometric Determination of Traces of Nickel with 4-Isopropyl-1,2-cyclohexanedionedioxime," Anal. Chem. 31, 931 (1959).

McGowan, F. K., "Excitation Coulombienne" (Coulomb Excitation), p 225, Comptes rendus du congrès international de physique nucléaire, Dunod, Paris, 1959.

McGrath, R. A., "A Chambered Microscope Slide," Microscope 12, 57 (1958).

McGrath, R. A., and C. C. Congdon, "X-Ray-Induced Abnormal Differentiation of the Epithelium of the Small Intestine in the Mouse," Intern. J. Radiation Biol. 1, 80 (1959).

Miller, F. J., and P. F. Thomason, "Direct Thermometric Titration of Boric Acid," Talanta 2, 109 (1959).

Morgan, H. W., and J. H. Goldstein, "Microwave Spectrum and Molecular Structure of Vinyl Fluoride," J. Chem. Phys. 30, 1025 (1959).

Nystrom, C. W., N. E. Tolbert, and S. H. Wender, "Formation of β-Phenylglucoside in Plant Leaves," Plant Physiol. 34, 142 (1959).

Owen, R. D., "Genetic Aspects of Tissue Transplantation and Tolerance," J. Med. Educ. 34, 366 (1959).

Russell, W. L., "Radiation-Induced Genetic Damage in Mammals," p 41 in Genetics, Radiobiology, and Radiology Proceedings, Midwestern Conference, Thomas, Springfield, Ill., 1959.

Russell, W. L., L. B. Russell, and J. S. Gower, "Exceptional Inheritance of a Sex-Linked Gene in the Mouse Explained on the Basis That the X/O Sex-Chromosome Constitution Is Female," Proc. Natl. Acad. Sci. U.S. 45, 554 (1959).

Schwartz, D., "Further Studies on Breakage-Fusion-Bridge Cycles in Maize Endosperm," Maize Genet. Coop. News Letter 33, 88 (1959).

Soldano, B. A., R. W. Stoughton, R. J. Fox, and G. Scatchard, "A High-Temperature Isopiestic Unit," chap. 14 (p 224-35) of The Structure of Electrolytic Solutions, ed. by W. J. Hamer, Wiley, New York, 1959.

Susano, C. D., H. P. House, and M. A. Marler (eds.), Analytical Chemistry in Nuclear Reactor Technology. Part 1, Specific Applications of Diverse Methods of Chemical Analysis; Part 2, Instrumentation, Remote Control Techniques, and Nucleonics; Part 3, Particle-Size Analysis. Second Conference, Gatlinburg, Tennessee, September 29-October 1, 1958, TID-7568 (April 1959).

Thompson, D. O., and D. K. Holmes, "Dislocation Contribution to the Temperature Dependence of the Internal Friction and Young's Modulus of Copper," J. Appl. Phys. 30, 525 (1959).

Welshons, W. J., and L. B. Russell, "The Y Chromosome as the Bearer of Male Determining Factors in the Mouse," Proc. Natl. Acad. Sci. U.S. 45, 560 (1959).

White, J. C., and R. F. Apple, "The Spectrophotometric Determination of Cerium in Fluoride Salts," Talanta 2, 176 (1959).

Wilkinson, M. K., "Neutron Diffraction Investigations of Magnetic Phenomena in Crystalline Compounds," J. Appl. Phys. 30, 278S (1959).

FOREIGN VISITORS

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

May 1959

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