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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

APRIL 1960

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

APRIL 1960

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Oak Ridge, Tennessee
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U.S. ATOMIC ENERGY COMMISSION

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

April 1960

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 4000 – REACTOR DEVELOPMENT

GAS-COOLED REACTOR PROGRAM

Irradiation Experiments. – New values of burnup and fission-gas release were calculated for ORR-irradiated capsules 0-1, -2, -3, and -5, which consisted of hollow UO_2 pellets in type 304 stainless steel capsules. The new values, based on fill-gas pressure measurements that gave an absolute pressure of no more than 690 mm Hg prior to irradiation, are listed below:

Capsule	Burnup (Mwd/metric ton)	Fission-Gas Release (%)
0-1	1760	9.3
0-2	1910	11.6
0-3	1640	10.3
0-5	3260	8.1

In the instantaneous fission-gas-release experiment, a 200-channel analyzer is being used to obtain analytical data directly from the flowing gas. Daughter products Cs^{138} and Rb^{89} have been identified as well as Kr^{88}, Xe^{131}, Xe^{133}, Xe^{133m}, Xe^{135}, and Xe^{135m}. With the temperature low and the neutron flux at zero, Xe^{133}, Xe^{133m}, Xe^{135}, and Xe^{135m} (formed from decay of iodine that had left the fuel and plated on the sweep circuit) were present in the sweep gas. The fuel was allowed to decay for nearly two weeks and then reinserted into the reactor. A burst of activity resulted when the temperature was increased. Spectra of this burst were obtained at several-minute intervals. Since the release of the isotopes Xe^{131}, Xe^{133}, and Xe^{135} decreased with time, it is evident that they were stored in the fuel. The release of these isotopes reached equilibrium in about 70 min. The rate of evolution of xenon and krypton from the fuel was recalculated on the basis of a revised sweep-gas flow rate obtained with a better calibration of the flowmeter. At a temperature of 1050°C the calculated average

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diffusion coefficient for krypton was 5×10^{-15} cm^2/sec, and for xenon it was 3.5×10^{-15} cm^2/sec. These diffusion coefficients were based on a slab model rather than equivalent spheres.

Plans are being made for relocation of the ten full-diameter prototype EGCR fuel capsules currently being irradiated in the ETR and the insertion of four new units (E-11, E-12, E-1R, E-2R). The ETR experiment review committee is reviewing the plans prior to approval of the resumption of irradiation after the cycle 29 shutdown this month. The relocations are to be made to increase the heat rate from 30,000 to 35,000 Btu$\cdot$hr$^{-1}\cdot$lin ft^{-1}. (AEC Activity 4141.1)

Tests of Graphite and Structural Materials in Static Helium. — Isothermal pot tests have demonstrated the effect of metal-gas reactions on the impurity buildup in static helium as a result of the outgassing of graphite. Both the area and temperature structure of the metal surfaces become important factors in projecting the results of compatibility experiments to conditions of the EGCR. Accordingly, a series of polythermal pot tests was carried out in which the temperature and the surface area of the metal relative to the temperature and surface area of the graphite were made to agree with EGCR conditions. In these tests, tubular specimens of type 304 stainless steel, corresponding to fuel element cladding, were held at temperatures ranging from 1500 to 1800°F, and the temperature of the graphite and metal surroundings was maintained at 1100°F.

Two general test arrangements were utilized for the experiments. In one arrangement the ratio of metal surface area at 1100°F to that of the heated element was 250:1, and in the other arrangement it was 53:1. The ratio of the volume of gas to the volume of graphite was 4.9:1 in the first arrangement and 3.6:1 in the second.

The impurity concentrations measured in these tests and the reaction films produced on the heated elements differed markedly between the two arrangements. The initial concentrations of CO were of the same order of magnitude in all tests, but the CO concentrations declined much sooner and faster in the higher surface-area-ratio tests than in the lower. A similar effect was noted for the CO_2 concentrations in the two types of tests. The CO_2 level reached a maximum in the first few hours at temperature in both types of tests but tended to remain constant or decline very slowly in the lower surface-area-ratio tests, whereas it fell off sharply in the higher surface-area-ratio tests. In contrast, the concentrations of non-oxygen-bearing impurities were roughly comparable in both types of tests. The amounts of H_2 and CH_4 reached maximum levels in 100 hr and usually decreased steadily thereafter.

The differences in impurity concentrations between the high and low surface-area-ratio tests were also manifested in characteristic differences in the microstructures and surfaces of the type 304 stainless steel heated elements. The surface films of the tubes exposed in the higher surface-area-ratio tests were silver-colored and were shown by electron diffraction analysis to be composed of Fe_3O_4 and Cr_2O_3. In contrast, the surface films of the tubes from the lower surface-area-ratio tests were dark green to black in color and were found to be primarily $2Cr_2O_3\cdot Fe_2O_3$, which was covered in some instances by a layer of graphite.

The heated elements used in both types of tests showed an increase in carbon content compared with the "as-received" level of 0.02 wt %. The amount of carbon pickup increased with increased element temperature; the greater carbon increases occurred for specimens in the higher surface-area-ratio tests. Metallographic examination of these specimens showed measurable increases in carbide precipitation in all cases where the carbon amounts were above approximately 0.1 wt %.

The interplay between graphite-gas and metal-gas reactions in the higher surface-area-ratio tests produced an environment that was initially carburizing and only weakly oxidizing. Thus the relatively large carbon increase in specimens from these tests is easily understood, since a relatively clean metal surface was exposed to the carburizing atmosphere at temperatures at which the diffusion rate was rapid. The increase in carbon content with element temperature is due to the increase in the diffusion rate with temperature. The oxide films which formed were composed chiefly of Fe_3O_4, which has characteristically been found in tests with a high percentage of CO relative to CO_2.

The atmospheres in the lower surface-area-ratio tests, in contrast with those in the higher surface-area-ratio tests, were initially oxidizing and later carburizing and reducing. The formation of a tightly adherent oxide film prior to the occurrence of a carburizing potential in these tests would be expected to hamper the diffusion of carbon into the metal. The fact that a carburizing potential existed later in the test is evidenced by the deposition of carbon which, as noted above, was found on the oxide surface. It may be noted that no Fe_3O_4 was detected in the films formed on the type 304 stainless steel heated elements in these tests, in contrast with the higher surface-area-ratio tests; only the spinel $2Cr_2O_3 \cdot Fe_2O_3$, MnO_2, and Cr_2O_3 were detected.

A disturbing metallographic finding in these tests has been the occurrence of sigma-phase material. Special etching techniques have given a positive indication of sigma formation in most of the type 304 stainless steel heated elements. This effect was especially evident in the elements tested at 1700 and 1800°F; a nearly continuous network of sigma-phase material formed along the grain boundaries. No apparent correlation was found between the occurrence of this phase and the impurities in the gaseous environment or the subsequent differences in carbon content. The sigma phase was preferentially oxidized by the gaseous environments and, at high temperatures, there was intergranular attack to a maximum depth of approximately 3 mils. (AEC Activity 4141.1)

Evolution of Gas from Graphite. – Graphite specimens prepared from needle-type coke by two different manufacturers were found to behave differently upon heating from room temperature to 1000°C. Approximately 18 cm^3 of gas per 100 cm^3 of graphite was evolved by the specimens of one fabricator, whereas the specimens from the other fabricator evolved only about half as much gas under the same conditions. (AEC Activity 4141.1)

THERMAL-BREEDER REACTOR PROGRAM

Homogeneous Reactor Program

Homogeneous Reactor Test. — The reactor maintenance and alteration program continues. Arc-cutting of the five core diffuser screens, performed last month, produced about 1/2 liter of metal droplets and over 100 larger pieces of screen which were removed with simple tools. The core circulating pump was raised, and temporary crossover pipes were installed to permit back-flushing.

Plastic impressions of both core-vessel holes were made. The impression of the lower (first) hole will be used to replicate a contour patch; the upper impression will ensure the positioning of a tool which will remove a specimen of the vessel wall for metallographic examination.

Accessible portions of the fuel and blanket systems were surveyed, using an underwater radiation probe. Gamma activity on the surface of the high-pressure system piping was in the range of 1000 to 5000 r/hr. (AEC Activity 4151.3)

Homogeneous Reactor Blanket Studies. — Thoria spheres of 0.2-in. diameter were fabricated by pressing active thoria powders granulated with 5% aluminum stearate in a Stokes model E automatic press, followed by firing to 1750°C. The spheres had a density of 9.6 g/cc and lost only 0.4% of their original weight per hour in a spouting-bed attrition test. Porous, weak spheres (furnished by Davison Chemical Company) whose attrition losses were 21% per hour in the spouting bed were strengthened by addition of 1.5 wt % Al_2O_3 as dibasic aluminum nitrate, followed by drying and calcining to 1750°C. The attrition loss for the treated pellets was 0.8% per hour. (AEC Activity 4151.1)

Heterogeneous Equilibria in Aqueous Systems (See AEC Activity 5330, ORNL-2906). — Solubilities at 325°C in the system UO_3-CuO-SO_3-H_2O were determined along the binary curve for which the two compounds $CuO \cdot 3UO_3$ and $3CuO \cdot SO_3 \cdot 2H_2O$ were the saturating solid phases. The saturation concentrations at 325°C in a 0.065 m sulfate solution were found to be 0.024 m CuO and 0.031 m UO_3. The data indicated no marked decrease in solubility as the temperature was raised from 300 to 325°C at least over a concentration range from 0.003 to 0.065 m H_2SO_4.

The new compound $CuO \cdot 3UO_3$, reported previously (ORNL-2736, p 5), was found to be thermally stable to approximately 850°C over a one- to three-day period. The compound decomposed within a few hours at temperatures above 850°C. The x-ray diffraction patterns for the decomposed compound showed decomposition to U_3O_8 and to an unidentified compound or compounds. Some diffraction lines seemed to indicate CuO, but a complete set of lines for CuO was absent. (AEC Activity 4151.1)

Slurry Circulation Experiments. — In the 200B loop, additional turbulent-flow heat transfer measurements were made with a pure thoria slurry in the concentration range 260 to 830 g of Th per liter and the temperature range 180 to 275°C. At the lower concentrations it was inferred that fully turbulent flow was developed over the entire experimental range of 5 to 14 fps, whereas at the high concentration 14 fps was required.

The degree of nonuniformity of slurry flowing in a horizontal pipe at 280°C in the 200A loop was correlated with velocity and concentration as parameters. It was found that a uniform condition was approached asymptotically at high velocities. The approach to uniformity was more rapid at high concentration.

The 50A slurry loop was operated with a thoria slurry of 3.65 μ average particle diameter. The material exhibited mild dilatancy after pumping and after the addition of simulated corrosion products to a level of 27 mg of Fe per g of Th. (AEC Activity 4151.1)

Slurry System Development. — The 300-SM system, with a new full-flow heat exchanger and with modified valves, pressurizer, and feed system, was operated with thoria slurry at concentrations up to 950 g of Th per liter. The revised components performed properly, and operation of the system was considerably improved. (AEC Activity 4151.1)

Core and Blanket Vessel Development. — Calculations were made for estimating the temperature distribution within the core of the HRT with the flow reversed and the screens removed. At a power level of 5 Mw the maximum fuel temperature was calculated to be 305°C, assuming the core wall to be clean. In support of these computations, measurement of fluid velocities near the wall in the core-vessel model indicated a minimum of 1.5 fps.

Operation of a 2-ft-dia, 6-ft-long cylindrical model was begun. The model, which simulates a 40-Mw reactor or a scaled-down 400-Mw reactor, is of the annular re-entrant type with swirl vanes at the inlet. The first set of vanes tested induced excessive bypassing directly from inlet to outlet; they are being replaced with a modified design. (AEC Activity 4151.1)

Component Development. — An analysis was made of stresses induced in the HRT core vessel by operating the core and blanket regions at different temperatures. It was concluded that blanket temperatures down to 100°C would be permissible with the core at 280°C. (AEC Activity 4151.1)

Molten-Salt Reactor Program

Chemistry. — In order to determine whether diffusion coefficients for chromium, as determined by radiotracer and electropolishing techniques, are valid for predicting actual corrosion behavior, duplicate convection loops of INOR-8 were charged with a noncorrosive fluoride fuel solvent containing Cr^{51} as CrF_2. One loop was dismantled after 500 hr of operation to obtain diffusion coefficients, as manifested by exchange with the wall, under noncorrosive conditions. After the other loop had been operated for 792 hr, a selective oxidant for chromium, FeF_2, was added without otherwise disturbing the operation. The concentration profiles of tracer in the second loop, determined after an additional 264 hr of operation, agreed well with profiles predicted with data from the first loop. These results confirm both the diffusion coefficients and the mathematical analyses employed for treating metal diffusion in corrosion tests involving thermal-convection loops.

A fuel containing 14 mole % ThF_4, which permeates most graphites in a uniform manner, penetrates extruded CEY graphite tubing to an almost negligible extent and in amounts which decrease with distance from the exposed ends. Chemical analyses of the imbibed fuel show a selective depletion in the relative uranium concentration with increasing distance from the end.

The removal of trace quantities of Pa^{233} from solution in $LiF-BeF_2-ThF_4$ (67-18-15 mole %), which is of interest in connection with blanket processing, has been demonstrated by further experiments with BeO and CaO as precipitants. The addition of approximately 6000 ppm of the oxide precipitant reduced the Pa^{233} gamma activity from 10^5 counts$\cdot min^{-1} \cdot g^{-1}$ to background. The precipitated Pa^{233} activity was quantitatively restored to the solution by hydrofluorination of the melt. (AEC Activity 4152.1)

Metallurgy. – Thermal-cycling tests from 700°C to room temperature were made on molybdenum-to-Inconel braze joints for the purpose of selecting an alloy suitable for brazing these metals. Joints of both Coast Metals alloy 50 and Coast Metals alloy 53 withstood the thermal cycling, while an 82 Au–18 Ni (wt %) alloy joint failed by cracking. Because molybdenum promotes wetting on graphite, 25 alloys in the Au-Ni-Mo system were tested as possible brazing alloys for graphite-to-metal attachments. Five of these alloys were found to exhibit particularly good wetting properties and were used successfully to braze type RLM-42 low-porosity graphite tubes to molybdenum plates.

Uranium dioxide was precipitated from $LiF-BeF_2-UF_4$ salts when the salts were exposed to graphite. This was due to a reaction with contaminants in the salt. Several methods were demonstrated that will eliminate these contaminants. One method, that of decomposing ammonium bifluoride crystals in the system, removed oxide contaminants sufficiently from graphite so that subsequent exposure to molten salts at 1300°F resulted in no precipitation. A graphite-to-salt ratio of 27:1 was used, and final tests were made for 100 hr, whereas the UO_2 precipitation reaction is usually completed within 5 hr. (AEC Activity 4152.1)

Corrosion Tests. – Chemical and metallographic results were obtained for two additional INOR-8 forced-convection loops operated for one year with beryllium-base salts. Operating conditions for these loops were as follows:

	MSRP-8	MSRP-9
Salt composition*	124	134
Maximum salt-metal interface temperature, °F	1300	1300
Minimum salt temperature, °F	1100	1100
ΔT, °F	200	200
Reynolds number	4000	2300
Flow rate, gpm	2	1.8

*Salt 124: 58 NaF–35 BeF_2–7 ThF_4, mole %
Salt 134: 62 LiF–36.5 BeF_2–0.5 UF_4–1 ThF_4, mole %

The attack undergone by both loops, based on visual and metallographic evaluations, was extremely slight. Specimens examined metallographically revealed only light surface roughening in all the specimens removed from both loops. The start of a thin surface film was observed on several sections of both loops.

Before- and after-test chemical analyses of the salt circulated in loop MSRP-8 indicated very little change during the test in the concentrations of nickel and iron, but a significant increase in chromium concentration did occur. (AEC Activity 4152.1)

In-Pile Tests. – The second MSR graphite-fuel capsule irradiation experiment, ORNL-MTR-47, has satisfactorily completed 400 of the scheduled 1600 hr of irradiation. Operating conditions are normal.

The capsule end of the first experiment was cut off at the MTR hot cells and is being shipped to ORNL for disassembly and examination. (AEC Activity 4152.1)

Pump Development. – The molten-salt pump containing a salt-lubricated journal bearing was placed in operation at 1250 to 1275°F with molten salt 130. It contains a gimbals-mounted helical-groove bearing lubricated with the molten salt. Performance of the salt bearing is being investigated at various system resistances to flow and at shaft speeds of 800, 1000, 1200, and 1400 rpm. Performance will also be observed at temperatures ranging from 1100 to 1350°F.

The journal-bearing tester was placed in operation with molten salt 130 at 1200°F; seizure occurred after 497 hr of operation. The test bearing will be removed from the tester and inspected as soon as practicable.

The INOR-8 specimens from the self-welding experiment at 1500°F were examined; no evidence of self-welding was found.

The MF-F pump endurance test has operated continuously for 24,504 hr, circulating salt 30 at 1200°F. The PKP pump rotary assembly is being assembled prior to cold shakedown and subsequent hot operation with NaK. (AEC Activity 4152.1)

Remote Maintenance Demonstration. – An inventory of approximately 120 gal of molten salt at 1200°F has been successfully circulated through the loop of the remote-maintenance demonstration facility. The run was terminated as scheduled after 3 hr of salt circulation. The pump, pump motor, reactor vessel, heat exchanger, and sump tank are being remotely disassembled to study the effects of the frozen salt retained in the freeze flanges and pump.

A color motion picture is being made of the disassembly operations. (AEC Activity 4152.1)

Heat Transfer and Physical Properties. – The thermal conductivity of an INOR-8 alloy specimen has been determined in the temperature range 300 to 1000°F. The results can be represented to within ±5% by the equation $k = 7.2 + 0.0045\ t$, where k is the conductivity in $Btu \cdot hr^{-1} \cdot ft^{-1} \cdot (°F)^{-1}$ and t is the temperature in degrees F. The conductivities are somewhat lower than corresponding values for Inconel.

Approximately 3750 hr of operation has been accumulated on the system for determining the heat transfer coefficients of the fluoride mixture BULT-14 flowing through heated Inconel and INOR-8 tubes. A scheduled shutdown

was made in order to replace the thermocouples and the oxidized copper power leads of the INOR-8 section. On startup, it was observed that the turbine-type flowmeter was no longer functioning; flow rates are now being calculated from the pump speed by use of a calibration curve obtained during the period of initial operation of the system. Although the data exhibit considerable scatter, there does not appear to be a systematic variation in the heat transfer coefficient. Analysis and interpretation of the data are continuing. (AEC Activity 4152.1)

Fuel Processing. – A study of potential fuel processing costs and poison fractions for various fuel processing rates for a 1000-Mw (electrical) molten-salt reactor was completed. Conceptual plant design and cost estimates were made for the fluoride volatility process for recovery of uranium from both core and blanket salt and for the HF dissolution process for recovery of fuel carrier salt.

The estimated total fuel-cycle cost, which includes processing, inventories, and losses, is about 1.2 mills/kwhr for a breeder reactor with a conversion ratio near 1.06. The cost increases rapidly with increasing conversion ratio. The cost drops to 1.0 mill/kwhr at a conversion ratio of about 1.03, and to a broad minimum around 0.7 mill/kwhr for a reactor with a conversion ratio in the neighborhood of 0.95. Inventories account for about 40% of the cost, processing for about 50%, and salt replacement for the rest. The chief advantage of the molten-salt concept over other fluid-fuel reactors arises from the fact that the blanket is a solution which may be processed rapidly and cheaply for U^{233} recovery. (AEC Activity 4152.1)

MARITIME SHIP REACTOR PROGRAM

Two fuel-rod clusters were installed in the pressurized-water loop in the ORR and have been irradiated for one reactor cycle in loop water at 1750 psi and 500°F. The clusters consisted of three 0.5-in.-dia, 18-in.-long fuel rods fabricated by swaging fused and ground UO_2. The average heat flux, as determined from loop water flow and temperatures, was approximately 130,000 $Btu \cdot hr^{-1} \cdot ft^{-2}$. Irradiation of these two fuel-rod clusters will continue for several cycles. (AEC Activity 4510)

PACKAGE REACTOR DEVELOPMENT

Army Package Power Reactor (APPR-1). – Postirradiation inspection of a stationary fuel element and a control-rod fuel element after operation in core I of SM-1 (APPR) for 10 Mwyr revealed no significant effects attributable to irradiation. Aside from some "crud" deposition on fuel-plate surfaces, the over-all appearance of the elements was excellent.

Radiochemical burnup analyses of samples removed from both fuel elements, which contained 25.98 wt % UO_2, 0.14 wt % B_4C, and 73.88 wt % stainless steel, are summarized below:

	U^{235} Fissions (%)	Total Depletion (%)
Stationary fuel element No. 57		
Average burnup	13.00	16.0
Maximum burnup	31.00	38.13
Control-rod fuel element No. 13		
Average burnup	17.89	22.0
Maximum burnup	32.0	39.36

Plates have been fabricated for four Eu_2O_3-stainless steel absorbers to be used in core I of the SM-1A reactor at Fort Greely, Alaska. Two absorber assemblies have been completed. (AEC Activity 4630)

GENERAL REACTOR RESEARCH

Reactor Safety Studies. — A program has been initiated to study the release of fission products resulting from in-pile melting of fuel specimens of interest to civilian power reactors. Preliminary design of an experimental assembly to simulate "loss of coolant" melting of UO_2 specimens clad in stainless steel, to be installed in an existing facility at the ORR, has been completed. The assembly, as designed, consists of a stainless steel containment system which is penetrated by thermocouple wells and by inlet and exit lines for purging gases. The fuel specimen is supported on a pedestal of ThO_2 within an inner assembly of ThO_2 which is thermally insulated from the stainless steel container by ceramic material. The external system includes filters of graduated porosity to retain particulate matter and carbon traps for iodine and for krypton and xenon. Parts are being fabricated to permit assembly and out-of-pile testing of a completed mockup of this apparatus. Minor modifications to the facility in the ORR which are necessary to accommodate these experiments are in progress. (AEC Activity 4171.1)

Basic Shielding Research: Spectrum of Prompt Gamma Rays Associated with Thermal-Neutron Fission of U^{235}. — A calculation of the self-absorption effects in the encapsulated radioactive sources used for absolute calibration of Compton and pair spectrometers is nearly completed. The results will be applied to the continuing analysis of experimental data. An auxiliary experiment based on the use of the 4.43- and 11.7-Mev gamma rays resulting from the $B^{11}(p,\gamma)C^{12}$ reaction as spectrometer calibration sources was begun. Adequate target currents (0.4 ma) and target stability were obtained, but a serious breakdown of the BSF particle accelerator, being used as the source of the 200-kev protons for the reaction, has delayed completion of the experiment. (AEC Activity 4410)

Basic Shielding Research: Spectrum of Fission Product Gamma Rays Emitted from 1 to 1800 sec After Thermal Fission of U^{235}. — The effort to perfect the analysis of experimental data on fission product gamma rays has been renewed. Previous work has employed matrix techniques to correct for the nonuniqueness effect inherent in the crystal spectrometer, and the present work is along similar lines. An effort is being made to represent the experimental response functions by nonlinear analytic functions, with the IBM-704 as a computational aid. (AEC Activity 4410)

Fuel Element Development. — The fabrication of the experimental aluminum-clad, 20%-enriched U_3O_8-Al fuel plates for core II of the Puerto Rican reactor has been essentially completed. Approximately 500 fuel plates were manufactured with a rejection rate of less than 5%. Completion of the fuel assemblies will culminate the development of the 20%-enriched U_3O_8-Al fuel system.

Fabrication procedures are being established for producing burnable-poison-bearing, fully enriched fuel plates containing UO_2 and U_3O_8 dispersions in aluminum for irradiation testing at 200 and 500°F. Based on a limited number of experiments, U_3O_8-B_4C-Al powder-metallurgy compacts can be hot-roll-clad with type 1100 aluminum without difficulty. Homogeneity measurements on the fuel and poison by wet chemical methods indicate no detectable inhomogeneity within the accuracy of the analyses, which in the concentration range of interest was ±10% for the boron and ±2% for the uranium. The hot-roll-cladding of these dispersions with type X-8001 aluminum has been initiated. (AEC Activity 4420)

Power Reactor Fuel Processing: Mechanical Processing. — Initial operability tests of the equipment for dejacketing the SRE NaK-bonded stainless-steel-clad uranium fuel rods were successful. A prototype fuel rod was hydraulically expanded ~ 45 mils at 1300 psig pressure.

In fuel leaching tests the effective area of cylindrical UO_2 pellets dissolving in nitric acid was estimated from experimental rate measurements as a function of the fraction dissolved. The results can be approximated by $A_p/A_0 = 1 + 13x - 14x^2$, where A_0 is the area of the initial cylinder (diameter and length approximately equal), A_p is the effective area of the porous, partially dissolved pellet, and x is the fraction dissolved, by weight. (AEC Activity 4441)

Power Reactor Fuel Processing: Alternative Aqueous Processes. — In dissolution tests with uranium-molybdenum alloy fuel, 3 to 4 moles of nitric acid was consumed for each mole of U–8.4% Mo alloy dissolved in HNO_3 or HNO_3-$Fe(NO_3)_3$ solution. The solutions produced were 0.25 to 1 M in iron and 0.3 to 2 M in uranium. These results are valid only for the experimental conditions used, since the amount of acid apparently consumed is partially governed by the efficiency of the off-gas condensing system.

The dissolution rate of niobium in refluxing sulfuric acid increased from 0.75 to 6 $mg \cdot cm^{-2} \cdot min^{-1}$ as the sulfuric acid concentration was increased from 14 to 18.3 M.

Samples of graphite–uranium carbide fuel were disintegrated and leached twice at 25°C with 90% nitric acid (0.2% maximum dissolved oxide). Recovery

of uranium from fuel containing 2% uranium or more was 99.5%, as compared with 99.8% recovery for a 90% nitric acid disintegration and leach followed by a secondary leach with boiling 15.8 M HNO_3. When boiling 15.8 M nitric acid was used in the second leach, the uranium recovery from all fuels studied was independent of the temperature of the first leach with 90% acid between 25°C and the boiling point (~ 103°C). Since 25 to 30% of the total uranium recovered was found in the wash solutions, high recoveries can be achieved only after thorough washing of the graphite matrix. These tests indicate that adequate uranium recovery can be achieved at 25°C with 90% HNO_3 and water washing alone, thereby minimizing both corrosion of stainless steel and formation of colored, nitric acid—soluble graphite degradation products. (AEC Activity 4441)

Power Reactor Fuel Processing: Zircex Process. — Nitric acid—insoluble residues of ZrO_2-UO_2 obtained from the hydrochlorination of high-zirconium (STR) fuel samples were chlorinated with CCl_4 in an inert carrier gas. The reaction rate at 350°C was only one-third that at 600°C. No reaction was evident with high-uranium (EBWR) residues below about 500°C.

In Zircex cyclic corrosion tests, Chlorimet-2 and type 316 stainless steel showed attack rates of 5 to 18 and 19 to 21 mils/month respectively. The addition of CO to the CCl_4 part of the cycle did not improve the corrosion resistance of most alloys appreciably. In boiling 10 M LiCl—0.1 M HCl solution, Hastelloy A showed rates of less than 1 mil/month. (AEC Activity 4441)

Power Reactor Fuel Processing: Darex Process. — Dissolution rates for UO_2-ThO_2 pellets were 8 to 45% higher in 13 to 15 M HNO_3—0.04 M F^-—1 M stainless steel solution than in 13 to 15 M HNO_3—0.04 M F^-—0.1 M Al when the dissolvent was continuously recirculated between a refluxing dissolver and a well-mixed external reservoir. The highest rate for dissolvent containing stainless steel solution was obtained with 13 M HNO_3; with aluminum solution the best rate was achieved in 15.5 M HNO_3.

In corrosion tests under total reflux conditions, titanium showed local weld-metal attack in the condenser above boiling Thorex feed-adjustment solutions after 840 hr; for the first 100 hr of successive boildown, no localized attack was noted. In 0.05 M fluoride solutions the corrosion rate of titanium decreased almost logarithmically with increase in HNO_3 between 6 and 15 M. In 13 M HNO_3 the rate increased almost logarithmically between 0.005 and 0.05 M F^-.

Type 347 stainless steel exposed at 80°C for 672 hr showed beginning intergranular attack in Darex-Purex waste storage corrosion tests with stainless steel solution containing 5 M HNO_3 and 100 ppm Cl^-. In similar solutions only 2 M in HNO_3, neither type 347 nor type 304L showed signs of attack after 336 hr. (AEC Activity 4441)

Power Reactor Fuel Processing: Solvent Extraction Studies. — In a test of the NS "Savannah" first-cycle solvent extraction flowsheet in the Immi mixer-settlers with unirradiated fuel, uranium losses to the raffinate and stripped solvent were less than 0.01%. However, stage samples indicated excessive uranium in the extraction-scrub bank.

The volume of 0.008 M $Al(NO_3)_3$ required to strip thorium and uranium from a 30% solution of tributyl phosphate in Amsco in six stages varied

with the acidity of the organic phase. One and 1.2 volumes of aqueous phase were required to strip the uranium and thorium from organic phases containing 0.1 and 0.2 M HNO_3 respectively. In five stages, 1.5 volumes of aqueous phase was required to strip an organic phase containing 0.2 M HNO_3.

A final U^{233} purification system using 2.5% tributyl phosphate in Amsco and acid salting showed decontamination factors from ruthenium, zirconium-niobium, protactinium, and thorium of 2000, 200, 10, and 40 and a uranium loss of less than 0.1%.

Equilibrium data in a system consisting of 30% tributyl phosphate in Amsco, water, thorium nitrate, uranyl nitrate, and nitric acid showed that for systems containing less than 1 mole of total nitrate, the uranium extraction is approximately proportional to the total nitrate concentration and is independent of the cations present. (AEC Activity 4441)

Power Reactor Fuel Processing: Zirflex Process. – In continuing corrosion tests on Ni-o-nel, Zirflex dissolver solution (6.0 M NH_4F–1.0 M NH_4NO_3) containing dissolved Zircaloy attacked the Ni-o-nel at rates of 28 to 32 mils/month until the zirconium concentration reached about 0.4 M. Between 0.4 and 0.6 M Zr^{4+} the corrosion rate decreased to about 1 mil/month. In flowing solutions of 6 M NH_4F–1.0 M NH_4NO_3 containing 0.003 M H_2O_2, Ni-o-nel was corroded at maximum rates of 0.6, 5.5, and 1.4 mils/month in the vapor, at the interface, and in the solution respectively. (AEC Activity 4441)

Power Reactor Fuel Processing: Sulfex Process. – Results of the last two of four successive engineering-scale Sulfex decladdings and Thorex dissolutions of unirradiated stainless-steel-clad ThO_2-UO_2 Consolidated Edison prototype fuel were obtained. Decladding-solution thorium losses in the presence of a ThO_2-UO_2 heel were 0.2%, as compared with 0.08% in the absence of a heel. Neither boron nor sodium was lost by volatilization or precipitation during core dissolution. A 23-hr batch dissolution of the 3.6% heel left from the four semicontinuous runs resulted in dissolution of 99.98% of the total ThO_2-UO_2 charged. Stainless steel passivation was avoided in one run by addition of 0.1 M HCOOH.

The insoluble residue obtained in the dissolution of ThO_2-UO_2 pellets in boiling 13 M HNO_3–0.04 M NaF–0.1 M $Al(NO_3)_3$ in 3-min tests was identified as $6Al_2O_3 \cdot CaO$. The pellets contained about 0.4% Al. In other experiments with ThO_2-UO_2 pellets having densities 93% of theoretical, the initial 10-min dissolution rates were 2.1, 3.0, and 2.4 $mg \cdot min^{-1} \cdot cm^{-2}$ for pellets containing 2.95, 5.98, and 9.12% UO_2 respectively. Fused ThO_2 dissolved much more slowly than 95.8% ThO_2–$UO_{2.36}$ having 93% of theoretical density. Only 30% of the fused ThO_2 (-4 +8 mesh) dissolved in 5 hr, as compared with about 88% of ThO_2-UO_2 of the same mesh size. (AEC Activity 4441)

Waste Disposal Research and Engineering: Geochemical Studies. – The mineral variscite, $Al(OH)_2H_2PO_4$, was found to have a high affinity for strontium even in neutral pH ranges. Since calcium phosphate precipitates have been extensively investigated for strontium removal, comparative studies were made with aluminum phosphate precipitates. At pH 6.0, aluminum phosphate flocs removed 50 times more strontium than calcium phosphate flocs; at pH 12.0, however, calcium phosphate flocs removed 20 times

more strontium than aluminum phosphate. Whereas pH 12 and a Ca/P equivalent ratio of 0.4 were optimum conditions for strontium removal (94%) by calcium phosphate flocs, pH 9 and an Al/P equivalent ratio of 0.75 gave optimum removal (97%) by aluminum phosphate.

Although only a small amount of strontium was removed by aluminum phosphate at pH 12 by the precipitation technique, the mineral variscite removed the largest amount at this pH. The probable explanation is that very little if any precipitate was formed at the high pH, whereas the mineral which was added (80-μ particles) had not dissolved during the time that the experiment was conducted. (AEC Activity 4452)

Waste Disposal Research and Engineering: Disposal in Natural Salt Formations. — The volume of gas produced when NaCl-saturated Purex waste is heated to 80°C was measured and found to agree well with the calculated volume of NOCl and Cl_2 expected from the depletion of hydrogen ions in the aqua regia reaction. No product gases were released from the NaCl-HNO_3 system when the acid concentration was 3.3 M or less, even when the system was kept at boiling temperature for several hours. The thermochemical balance of the aqua regia reaction shows that this reaction is endothermic to the extent of over 10 kcal per mole of hydrogen ion consumed.

Samples of salt and impurities from the side walls of the experimental pits at Hutchinson, Kansas, were treated with simulated Purex waste solutions at room temperature. The shale samples produced measurable quantities of NO_2. Traces of CO_2 were obtained from some shale and from mixed shale-anhydrite samples.

The field experiment with neutralized waste has operated for 110 days. The average temperature in the pit containing neutralized waste has reached a maximum at approximately 73°C and has not dropped significantly. Eight feet out from the walls of the cavity the temperature is 31.5°C and at 30 ft it is 22.5°. The average temperature in the pit containing acid waste has reached a maximum of 63.5°C and continues at approximately that level. Temperatures in the salt at 8 ft and at 30 ft from the cavity are the same as for the neutralized pit, 31.5 and 22.5°C. Temperatures in the salt structure continue to slowly but steadily increase. (AEC Activity 4452)

Waste Disposal Research and Engineering: Disposal of ORNL Radioactive Wastes. — Ruthenium is not retained completely by the waste-pit system, and, consequently, a part of this material finds its way into the Clinch River after passing through White Oak Creek. The estimated amounts of Ru^{106} released from the pit area to the bed of White Oak Lake during 1957 and 1958 were 200 and 160 curies respectively. The volume of liquid lost by seepage from the pits was 4.92 million gal in 1957 and 4.84 million gal in 1958. Therefore the average concentration of Ru^{106} in liquid seeping to the streams was 0.011 μc/ml in 1957 and 0.0087 μc/ml in 1958. The cumulative average concentration of Ru^{106} in waste released to the pits was 0.47 μc/ml in 1957 and 0.35 μc/ml in 1958. Considering the concentrations of Ru^{106} in the waste and in the liquid seeping to the streams, the calculated reduction in Ru^{106} due to dilution, decay, and sorption was 97.7% in 1957 and 97.5% in 1958. (AEC Activity 4452)

Waste Disposal Research and Engineering: Clinch River Studies. — Since the Clinch River sediments contain a mixture of clay minerals, initial tests

on radionuclide uptake were made with pure samples of illite, vermiculite, kaolinite, and montmorillonite. These standard samples of clay minerals have been evaluated for sorption of cesium, cobalt, strontium, and zirconium-niobium. Portions of distilled water containing tracer amounts of the respective radionuclides (no carrier added) were mixed with known weights of clay. The selectivity of the clay, K_d, is expressed as the ratio of the fraction of radionuclide per unit weight of clay divided by the fraction of radionuclide per unit volume of solution. The extremely high K_d of illite for cesium (>150,000) is further evidence of the favorable structure of illite for specific sorption of cesium ions. The readily accessible ion exchange sites of kaolinite and montmorillonite are not particularly favorable for cesium sorption (K_d is 2000 for kaolinite and 1000 for montmorillonite).

For strontium sorption, the readily accessible exchange sites of kaolinite and montmorillonite favor increased sorption (K_d is 4000 for kaolinite and 2000 for montmorillonite); illite shows very limited sorption for strontium (K_d, 350). Particularly noteworthy is the high K_d of kaolinite for strontium. On the basis of weight alone, kaolinite exhibits the lowest exchange capacity of the minerals tested and should sorb the least amount of strontium. These findings again emphasize the importance of the surface features of clays for tracer-level ion sorption.

The reactions of cobalt and zirconium-niobium with the clay minerals appear to be colloid interactions rather than ion exchange reactions. Whereas cesium and strontium solutions were not affected by centrifugation, cobalt and zirconium-niobium removals were sensitive to centrifugation and pH. Another interesting feature of the results is that illite and vermiculite (this vermiculite, like most commercially available samples, is an illite-vermiculite material) show increasing uptake with time, whereas montmorillonite and kaolinite show a decreasing uptake with time. (AEC Activity 4452)

Waste Treatment and Disposal. — In a 4 × 18 in. pot, synthetic Purex waste containing sodium and calcium to decrease the volatility of sulfate was evaporated and calcined to 900°C; 5 gal of solid product (1.56 g/cc, 48% porosity) represented 1 ton of uranium processed. The thermal conductivity of the solid increased linearly with temperature, from 0.210 $Btu \cdot hr^{-1} \cdot ft^{-1} \cdot (°F)^{-1}$ at 230°F to 0.311 at 1330°F. The condensate contained 98% of the feed nitrate and 0.71% of the sulfate; the water in the recirculating off-gas scrubber contained 2 and 0.02% respectively. In an 8-in.-dia, 84-in.-high vessel the average feed rate was 16 liters/hr. Tracers (Cr^{51} and Nb^{95}) showed the solids deposition pattern to be partly radial growth from the side and partly vertical growth from the bottom. In small-scale batch studies an NO atmosphere decreased the volatility of ruthenium from 66 to 79% to 0.9 to 8.0%.

Air bearing $Cr(NO_3)_3$ aerosols (labeled with Cr^{51}) and containing 0.30 to 0.95 mole fraction of steam was decontaminated to background (decontamination factor at least 10^4) in a 3-in.-dia packed distillation tower 10 ft high. With no steam the decontamination factor was less than 10.

Synthetic Sulfex decladding waste (stainless steel sulfate) solidified with 0.26 part each of $Ca(OH)_2$ and sand and 0.55 part of portland cement per part of waste had a volume 45% greater than the original and a compression test strength of 3000 psi. Without sand the solid volume was 23 to 28% larger and the strength was 1250 to 1350 psi, which may be strong enough for transportation and storage. Synthetic Zirflex decladding waste (zirconium fluoride) partially evaporated with $Ca(OH)_2$ and then solidified with sand and portland cement had 23% less volume than originally and a compression test strength of 4200 psi.

Two thousand volumes of effluent from the low-level-waste lime-soda treatment plant containing added Cs^{137}, Sr^{85}, and Ce^{144}-Pr^{144} was filtered (removing 30% of the gross gamma activity) and then passed through one volume of sulfonated phenolic cation exchanger for an over-all removal of ~ 95%. From influent that was only made 0.01 M in NaOH and then treated by filtration and ion exchange, more than 99% of the tracer activity was removed; the better results are thought to be due to the lower sodium content. In upflow regeneration of the resin with 5 M HCl, 98% of the activity was removed in five bed volumes and 99.98% in ten. The second five bed volumes could be recycled. (AEC Activity 4452)

PROGRAM 5000 — PHYSICAL RESEARCH

ISOTOPE PRODUCTION

Radioisotope and Stable Isotope Production. — The collection of the isotopes of indium was completed with ion currents equivalent to 212 g. The collections of the isotopes of barium, lead, chromium, and osmium are now in process.

Charge materials were produced for the separation of the isotopes of barium, calcium, lithium, chromium, magnesium, and osmium. The runs with lithium and magnesium were exploratory. Osmium tetroxide was supplied so that the isotopes of osmium could be separated for the first time.

Large quantities of inorganic salts have been synthesized in the charge preparation laboratory for potential use as charge materials in expanded facilities located in Building 9204-3.

Thirty-three new lots of isotopes and twenty-two returned samples were prepared for inventory following spectrochemical and mass analyses.

Special service projects, initiated at users' requests, included: (1) the conversion of Hg^{202} metal to aqueous $Hg(NO_3)_2$ (including encapsulation in a Pyrex ampoule), $Cd^{113}O$ to CdI_2, elemental B^{11} to H_3BO_3, $Cr_2^{53}O_3$ to $Cr(C_2H_3O_2)_3$, Li^6 metal to LiF; (2) crushing $Cr_2^{52}O_3$ to 350 mesh; and (3) encapsulating $Si^{30}O_2$ and $Hg^{198,199,202,204}$ as $Hg(NO_3)_2$ in suitable containers for shipment. Target preparations included 17 conversions and 16 targets.

Radioisotope Research and Development. — Several inorganic ion exchangers were investigated for their efficiency in selectively removing Cs^+ from 6 M Na^+ aqueous solution. With an exchanger of this type, Cs^{137} could be concentrated from such fission product solutions as the basic supernatant in Savannah River storage tanks. The exchanger could be contained in a shipping cask. An aluminosilicate material, manufactured by the Permutit Company under the trade name Decalso, exhibits the most desirable properties of those tested. The Cs^{137} can be concentrated by a factor of 10 so that 10,000 curies can be shipped in one shielded transfer tank. The Cs^{137} is eluted from the Decalso with $(NH_4)_2SO_4$ solution.

A shielded cask for transferring dried fission product rare-earth double sulfate from Hanford to ORNL is being designed at Hanford. A scale model of the filter to be contained in the cask was fabricated, and the dissolution of the cake from the filter was tested. It was found that the double sulfate cake could not be dissolved completely in 4 M HNO_3 if the salt had been heated to 500°C, which is the maximum anticipated salt temperature. Apparently the dehydrated salt cakes tightly so that wetting of the interior of the cake with the dissolvent acid is impossible.

Test thermal diffusion columns for determining the characteristics of a parallel column system for enriching C^{13} are being installed. (AEC Activity 5100)

Stable Isotope Development. — Sputtering data accumulated during lead isotope separation have led to construction of a new collector pocket for Pb^{207}.

A second-pass separation of Pb^{207} is in progress. Enrichment factors, retention, and total elemental lead recovery are to be determined. Output achieved averaged in excess of 2 ma. Only 4.5 g of lead was included in the charge.

Developmental results have proved the practicality of separating the isotopes of osmium. In the current osmium separation, use is being made of calutron equipment modified to accommodate this element and of procedures outlined to provide personnel protection. (AEC Activity 5100)

PHYSICS

Scintillation Spectrometry and Instrument Development. — The singles and coincident spectra of the gamma rays from the decay of Rh^{102}, which was produced by the (p,n) reaction on ruthenium containing 97.2% Ru^{102}, have been measured with scintillation spectrometers. The energies (in kev) and the relative intensities of the gamma rays are: 415 ± 4 (3.2), 475 ± 5 (100), 630 ± 6 (17.4), 695 ± 7 (9.5), 745 ± 8 (0.7), 765 ± 8 (6.9), 1050 ± 10 (9.7), 1105 ± 8 (4.4), 1110 ± 11 (4.1), 1365 ± 10 (0.67), 1565 ± 13 (0.27), 1795 ± 15 (0.06), 2040 ± 14 (0.06), and 511 (31.6). Spins of the levels in Ru^{102} which are consistent with the directional angular correlations of the gamma-ray cascades are: 475 (2^+), 1105 (2^+ and 4^+, doublet), 1525 (3), 1840 (0^+), 1870 (3^+), 2040 (2), 2220 (3), and 2270 (?). A value of E2/M1 ≧ 225 for the 630-kev transition ($2^+ \rightarrow 2^+$) was deduced from the composite correlation of the 630-475-kev cascades. The branching ratio of cascade to crossover from the second 2^+ state is 1.5 ± 0.3. Gamma-ray spectra have been measured as a function of time for 250 days, and a change in the relative population of the states is observed. From this, the existence of a long-lived isomeric state in Rh^{102} is inferred. (AEC Activity 5220)

High-Voltage Experimental Program: Neutron Total Cross Sections in the Kilovolt Region. — The ORNL millimicrosecond neutron time-of-flight specitrometer is used, in part, for the measurement of neutron total cross sections in the kilovolt range of energies. Germanium is the first element for which it has been possible as yet to study a number of isotopes. Over the energy range 3 to 30 kev the following resonances have been found: Ge^{70}: 4.35, 4.95, 5.55, 6.65, 8.5, 9.8, 10.2, 11.7, 13.2, 18.5, 23.8, 25.7, 28.1; Ge^{72}: 4.52, 4.92, 8.9, 11.1, 19.2; Ge^{74}: 4.95, 19.3; Ge^{76}: 4.95, 14.9, 22.3, 29.3. While the peak positions are well established, some of the measurements have to be repeated before a really reliable figure can be given for the strength functions. Cobalt-59 was thought to have a relatively sharp resonance at 4.2 kev and a broad resonance at about 5 kev. With the best resolution achievable at the time, Co^{59} was re-examined; there was definite evidence that the resonance at about 5 kev is, in fact, two resonances peaking respectively at 4.8 and 5.0 kev. Finally, the measurements on Cu^{63} have been repeated and extended up to 10 kev with the following resonances: 4.35, 4.77, 5.14, 5.71, 7.40, 7.80, 9.00, 9.70. (AEC Activity 5220)

High-Voltage Experimental Program: Measurement of Thresholds for (p,n) Reactions. — Protons accelerated by the ORNL 5.5-Mev Van de Graaff were used to measure thresholds for (p,n) reactions in intermediate-weight nuclei. Targets of separated isotopes were electroplated or evaporated (except Nb^{93}) onto platinum blanks; the niobium target was a solid foil. Neutrons were

detected by a 4π counter. The target nuclei and their corresponding thresholds (in kev) are as follows: Fe^{57}, 1649 ± 3; Ni^{61}, 3076 ± 4; Zn^{68}, 3766 ± 5; Ga^{69}, 3052 ± 3; Ge^{74}, 3395 ± 5; Se^{80}, 2684 ± 4; Pd^{106}, 3790 ± 10; Pd^{108}, 2700 ± 100; and Cd^{112}, 3435 ± 15. Each threshold except that for Fe^{57} is consistent with energy differences obtained [Nuclear Data Sheets, Natl. Acad. Sci., Natl. Research Council (1958–59)] from positron decay or other threshold measurements. The Fe^{57} threshold corresponds to a Co^{57}-Fe^{57} ground-state energy difference of 838 ± 3 kev rather than 570 ± 30 kev as given in the nuclear data sheets. Thresholds were also observed on the Y^{89} target at 4257 ± 7 kev and on the Nb^{93} target at 2750 ± 100 kev. Each of these is interpreted as a threshold to a known [Nuclear Data Sheets, Natl. Acad. Sci., Natl. Research Council (1958–59)] excited state in the product nucleus; in each case the ground-state transition is inhibited by a large spin change. (AEC Activity 5220)

Heavy-Particle Physics. — A new technique utilizing a vacuum carbon arc was perfected for the manufacture of very thin unsupported carbon targets (15–50 $\mu g/cm^2$). These targets were used to extend the investigation of inelastic scattering to smaller angles.

A silicon barrier counter in conjunction with a very thin gold scatterer is now used to measure the energy and energy spread of the N^{+++} beam in a routine manner. The energy is 28.0 Mev and the spread ±300 kev at half maximum. This is in good agreement with previous results obtained from measurements of the range of recoil protons in emulsion. The energy and angular distributions of protons from the nitrogen bombardment of F^{19} were obtained. The energy distributions show an exponential dependence on the excitation energy of the residual nucleus, from which a temperature of ~ 2 Mev was obtained. The angular distributions are isotropic from 23 to 150° and for excitation energies from 12 to 26 Mev. (AEC Activity 5220)

The 86-Inch Cyclotron: Nuclear Physics. — An IBM card punch output was added to the equipment for measuring energy spectra from (p,p), (p,d), (p,t), and (p,α) reactions. This enables the data to be processed on the IBM-704 immediately upon the completion of a run (see ORNL-2925 for description of processing).

The (p,d) spectra from iron isotopes 54, 56, 57, and 58 are being measured; they are consistent with the single-particle interpretation which has been given to the (p,d) spectra from zirconium isotopes. The locations of the single-particle levels in the iron region do not agree with the simple predictions of the shell model of Mayer and Jensen. The locations of the levels are in qualitative agreement, however, with the calculations of Nilsson, in which Mayer and Jensen's special case of a spherically symmetric harmonic oscillator potential is generalized to allow a deformation of the average potential. (AEC Activity 5220)

The 86-Inch Cyclotron: Applied Physics. — The isotopes Fe^{55}, Co^{56}, Co^{57}, As^{74}, Y^{88}, and I^{125} were produced as service irradiations for customers in the United States and India, representing 50 hr of bombardment time. High-purity aluminum and high-purity magnesium targets are being irradiated on a long-term basis for the production of Al^{26} to be used in atomic-diffusion coefficient studies. Capsule targets were modified to

increase the average energy of the impinging proton beam for the production of Ca^{47}. A sample, representing approximately 200 mg of ~ 95%-enriched Ca^{48}, was irradiated at 100 μa for 1.5 hr in a test run. Yield calculations will be made on this sample. Cardboard cassettes are now being used as film holders for the autoradiographic studies made on flat-plate targets. This has resulted in a reduction of radiation dosage to personnel from about 100 to 150 mr/week to a present level of about 10 mr/week. The associated hazard of transporting a highly radioactive source through the building to the darkroom has been eliminated. Routine x-raying of capsule targets was introduced for the detection of nonuniformities in packing density. A system of ten automatic recording beta counters utilizing Geiger tubes has been assembled and placed in operation. Variable scaling and time factors are incorporated in the apparatus. (AEC Activity 5220)

Relativistic Isochronous Cyclotron (ORIC). — Construction of the ORIC building (Building 6000) has proceeded more rapidly with improved weather but is still behind schedule. The building is now ~ 20% complete (materials in place). The 7-ft shielding walls are in place to ceiling height except for placement of windows and doors. The cyclotron room floor and the second-floor slab in the laboratory-office section are poured.

Improvements in magnetic-field measuring and in data processing now make it possible to complete a 2400-point field plot in 4 hr and to obtain the evaluation within 24 hr. Approximately 30 field configurations have been evaluated. Conflicting requirements are encountered in attempting to optimize conditions for both AVF focusing forces and constant orbit frequency at the same time; some compromise may be required. At present a suitable field configuration for proton energies up to ~ 68 Mev is obtained from the quarter-scale magnet work.

Tests with the full-scale r-f mockup indicate that the available frequency range will exceed the design requirements of 7.5 to 22.5 Mc/sec. The power requirement, 210 kw at 22.5 Mc/sec for 100 kv dee-to-ground, is within the anticipated design range. A 1/48 sector of the full-scale shorting plane was operated to failure at between 5500 and 6000 flexures; at two changes in frequency per day this indicates a life of about ten years.

Design and fabrication of components of the cyclotron continue on schedule. The design work is about 60% complete. The first forgings for the pole tips were rejected because of inclusions indicated by ultrasonic tests; new forgings were made and are now being machined. Major components now being designed are the dee structure, the AVF coils, and the console and control panels. Studies of beam deflection and the ion optics system continue; the orientation of the AVF sectors with respect to the dee has been determined.

Special Separations. — The engineering for a proposed new plutonium area in Building 9204-3 has been started.

The following shipments were made: uranium, 7; plutonium, 4; neptunium, 2; thorium, 1. (AEC Activity 5250)

Mass Spectrometry and Related Techniques. — Elementary processes in catalytic reactions were studied by means of mass spectrometry. A previously postulated intermediate HCOOH was detected in the reaction

$$H_2 + CO_2 \xrightarrow{Pt} CO + H_2O$$

In addition, an unsuspected product, H_2CO_3, which retards the reaction, was identified. It was also found that the decomposition of 1-butene on platinum gave the following free radicals: C_4H_7, C_3H_7, C_3H_6, C_3H_5, C_3H_3, and CH_3.

Novel charge transfer reactions producing chemical changes in the neutral molecule, for example,

$$Ar^+ + CH_4 \rightarrow Ar + CH_2^+ + H_2$$

were proved by means of mass spectrometry. Charge transfer from Ar^+ to CH_4 produced methyl, methylene, and hydrogen radicals. The observations were used to explain previous data from alpha ionization of the Ar-CH_4 mixture in the mass spectrometer and from radiolysis. (AEC Activity 5250)

CHEMISTRY

Chemical Engineering Research. — Distribution coefficients were measured for the distribution of uranium between acid-deficient solutions of uranyl nitrate in water and 30% solutions of tributyl phosphate in Amsco. Measurements of the pH of the aqueous phase before and after extraction and stripping showed that only neutral (stoichiometric) uranyl nitrate was extracted. Feed solutions containing from 550 to 30 g of uranium per liter and having a deficiency of from 0 to 0.3 mole of acid per mole of uranium were used.

Flooding studies on the mark I stacked clone contactor with the system Amsco 125-82—0.08 M HNO_3 showed throughputs to be lower than with the Supersol—0.08 M HNO_3 system. This effect was more pronounced at low aqueous throughputs; for example, at an aqueous flow rate of 300 cc/min the Supersol throughput was 700 cc/min, whereas the Amsco throughput was 400 cc/min. The flooding curves for the two solvents approached each other at higher aqueous rates, the limiting organic throughputs at an aqueous flow rate of 1000 cc/min being 400 and 300 cc/min for Supersol and Amsco respectively. Flooding studies with Amsco were carried to an aqueous flow rate of 2000 cc/min, at which flooding occurred when the Amsco throughput was 50 cc/min. (AEC Activity 5310)

Thorium Oxide. — In out-of-pile hydrogen-oxygen combination experiments with a slurry of 1000°C-fired thorium-uranium oxide (75.6% Th, 11.7% enriched uranium) at a concentration of 100 g per kg of H_2O and containing a sol-prepared palladium catalyst (100 parts of palladium per million parts of solids), catalyst performance indices were 1.5 to 2.0 w/ml at 280°C under an oxygen overpressure.

A sample of slurry (1000°C-fired Th—0.5% U oxide, 1800 parts of Pd per million of Th) from the slurry loop mockup gas recombination tests showed a catalyst performance index of ~ 1 w/ml at 280°C. A synthetic mixture of the original charge gave an index greater than 30 w/ml, confirming qualitatively the dropoff in catalytic activity observed in the loop run. The dropoff in activity during pumping was coincident with a degradation of slurry solids.

A second series of thorium oxide powders fired at 650, 800, 900, 1100, and 1500°C and irradiated for 21-3/4 months in an LITR lattice position was recovered for examination. The samples fired at 650, 800, and 900°C showed red coloration and clinkering similar to those observed with oxides fired at these temperatures and irradiated 16 months. The samples fired at 1100 and 1500°C were powders and had the off-white and blue colors observed in the previous series. Heat transfer calculations indicated that the powder temperature during irradiation was below 300°C. (AEC Activity 5310)

Equipment Decontamination. — Steel bolts contaminated with irradiated particles of UO_2 were decontaminated by a factor as high as 5000 by a few minutes of spraying with a citrate-carbonate-bicarbonate-peroxide solution in steam, followed by a steam spray of Turco 4518. This method has been recommended for decontaminating the EGCR service machine. Darex-contaminated titanium was decontaminated by a factor of 1400 by boiling for 30 min with 10% Turco 4518. Oxalic acid (5%) was very effective for removing Zr^{95}-Nb^{95} from stainless steel, with a decontamination factor of about 3×10^4 in 15 min of boiling. Oxalic acid was also efficient on ruthenium-contaminated type 347 stainless steel which had been pretreated with alkaline permanganate, with a total decontamination factor of about 10^3. The addition of 1 M H_2O_2 to 10% oxalic acid decreased the corrosion rate for type 347 stainless steel at 100°C from 0.011 ± 0.002 to 0.000 ± 0.002 mil/hr and increased its decontamination effectiveness for ruthenium by a factor of 7 to 10. Peroxide addition did not alter the effect of oxalic acid on Zr^{95}-Nb^{95}. A 5% oxalic acid-peroxide mixture dissolved mild steel at a rate of about 12 mils/hr. (AEC Activity 5310)

Inorganic and Physical Chemistry: Inorganic Solution Chemistry. — A new anion exchange separation technique was investigated in which strongly adsorbed anions are removed from exchangers through complexing with metals. The technique appears particularly useful for thiocyanate analysis, where excellent removal of adsorbed SCN^- occurs through complexing with Fe(III).

During an extension of this study, it was demonstrated that certain complex cations may be strongly adsorbed by anion exchangers. In this "abnormal" adsorption, adsorbability increases with ionic strength, as expected from theoretical considerations; in "normal" exchange reactions adsorbability decreases with ionic strength. This demonstration of adsorption of cations by anion exchangers complements earlier work in this Laboratory where strong adsorption of anions by cation exchangers was proved. (AEC Activity 5330)

Organic Chemistry. — The utility of the combined radiochemical-stereochemical technique in detecting which side of the migration terminus is attacked by each of two migrating groups during reactions involving 1,2 shifts was previously demonstrated. These same techniques were applied to the mechanisms of the deaminations of (1) optically active threo- and erythro-1,2-diphenyl-1-amino-2-propanol, (2) optically active threo- and erythro-1-phenyl-2-p-tolyl-1-amino-2-propanol, and (3) optically active 1,1-diphenyl-2-aminopropane. Both stereospecific and nonstereospecific C^{14} labels were employed. In each case the optical isomers produced were isolated and the fate of each label was determined. Appropriate control experiments were performed with reactants which were labeled both with C^{14} and with deuterium.

The absolute configurations of reactants and products for each reaction were studied through synthetic sequences which do not affect the optical integrity of the asymmetric centers. From these data it was possible to arrive at certain definite conclusions concerning the mechanisms of the foregoing reactions. For example: (1) the fractions of each reaction proceeding through cis and trans transition states were evaluated, and (2) new information was gained concerning the importance or unimportance of "neighboring group participation." Both classical techniques and a novel radiochemical method were used in these experiments. (AEC Activity 5330)

Separation of Oxygen Isotopes. — Construction of the O^{17} facility is proceeding according to schedule. The following components have been installed: all of the water distillation columns, the water heaters and heat exchanger, the cooling tower, and the diesel-electric emergency power supply. Piping and electrical circuits associated with these units are about 50% complete. Fabrication of the total and partial condensers is nearing completion; construction of the reboilers for the distillation columns is 50% complete. Fabrication of the thermal diffusion columns was begun. It is expected that all phases of construction will be completed by the end of the fiscal year. (AEC Activity 5340)

Analytical Chemistry Research. — The absorption spectra of praseodymium, neodymium, and samarium fluorides dissolved in molten lithium fluoride (mp 848°C) were recorded at approximately 900°C, a temperature almost one-third greater than any used previously for spectral studies in molten fluoride salt systems. A Cary model 14 M recording spectrophotometer equipped with a specially designed high-temperature cell assembly was used to record the spectra. The molten salt was confined as a pendent drop in a platinum tube. The wavelengths of maximum absorbance of these rare-earth fluorides generally correspond quite closely with those observed for the trivalent rare-earth ions in perchloric acid solutions, which indicates that the absorbing species in molten fluoride and aqueous media are identical. A new absorption peak was observed for praseodymium fluoride in the fluoride salt melt at a wavelength of 522 mμ. The maximum molar absorbance indices are: Pr, 1.2 at 444 mμ; Nd, 1.7 at 582 mμ; Sm, 0.63 at 399 mμ.

A sensitive spectrophotometric method for osmium (ORNL CF-59-7-16) was applied to the determination of microgram quantities of this element in molybdenite ore for the purpose of establishing the age of the ore. From the osmium content, which is derived from the beta decay of Re^{187} present in the ore, and also the rhenium concentration and half life, which has been established, the age of the molybdenite can be calculated. To determine the osmium, the ore is dissolved in HNO_3, which oxidizes the osmium to the octavalent state. The osmium tetroxide is distilled and absorbed in 1 M H_2SO_4 together with lower oxides of nitrogen, which are oxidized by the addition of potassium permanganate. Chloroform is then used to extract the osmium tetroxide, after which the osmium in the organic extract is determined spectrophotometrically by the diphenylcarbazide method. From tests made with Os^{191} as a tracer, it was indicated that the over-all recovery is of the order of 85%. Five molybdenite concentrates analyzed by this method were found to contain less than 5 ppm of osmium. The method was also applied successfully to the determination of 6 to 10 ppm of osmium in samples of flue dust which contained rhenium in relatively high concentrations.

METALLURGY AND SOLID STATE PHYSICS

Microstrains in Crystals. — A refinement of the diffraction theory for temperature diffuse scattering (TDS) for cubic powders has been accomplished. New general relations among the various orders of TDS have been derived, which allow the elimination of second-order TDS from measured intensity distributions without assumptions. (AEC Activity 5420)

Fundamental Physicometallurgical Research. — A study has been made of deformation stacking faults in high-purity lead and alloys of lead with indium and silver. Deformation at 4.2°K gave a stacking fault frequency of one fault for every 100 planes, as compared with one in 70 for α-brass. This study has resulted in a new theory regarding the formation of stacking faults in the face-centered-cubic metals. (AEC Activity 5420)

THERMONUCLEAR PROJECT

Arc Research. — Spectroscopic studies were made of the long high-current (≧200 amp) carbon arc in the Long Solenoid Machine. Intensity measurements on spectral lines involving energy states of widely different excitation potentials indicate electron temperatures of about 28,000°K from lines of C^{++} and 58,000°K from lines of C^{+++}. The temperature is determined from the relation $\log I\lambda^3/fg = \log Cn_j - 5040E_i/T$, where I is the line intensity, λ is the wavelength, f is the line oscillator strength, E_i is the excitation energy, n_j is the density of the ion species, and C is a constant. The difference in temperature for the different ionic species may be real, since the radial variation of temperature through the arc is not sampled by the present integrated intensity measurements. Thus the core of the arc may have a point temperature in excess of 60,000°K. The calculated density ratio C^{++}/C^{+++}, based on the Saha relation for temperature equilibrium at 40,000°K and $n_e = 10^{14}$ electrons/cm^3, is in general agreement with mass spectroscopic measurements in the peripheral region of the arc. The excitation temperature is also in general agreement with microwave radiation measurements, which gave 40,000 to 90,000°K. It is probable that inelastic collisions with partially stripped ions depopulate the high-energy tail of the electron energy distribution and keep down the electron "temperature." (AEC Activity 5540)

DCX-1 Facility. — The installation of the DCX-1 in Building 9201-2 is almost complete. Measurements are being made of the magnetic field, the pumping speeds, and the ion source and accelerator characteristics to assure that the facility is again operational.

The measured energy spectra of neutral particles escaping from the trapped plasma, described in the previous report, have been corrected to give energy spectra of the particles inside the machine. For the usual 600-kev molecular ion injection energy, the mean particle energies of these spectra are approximately 300 kev. (AEC Activity 5540)

Larger Magnetic Facility. — The detailed design of this experimental equipment (DCX-2) is now in progress. (AEC Activity 5540)

Vacuum. — The 32-in. Consolidated Electrodynamics Corporation diffusion pumps, as presently installed on the DCX, have pumping speeds for hydrogen of 20,000 to 25,000 liters/sec per pump. The oil back-streaming was reduced to a tolerable 0.6 cc/day per pump by positively centering the jet assembly inside the pump barrel.

Experiments on the pumping of hydrogen by titanium gettering on the inside of the DCX liner (before the move from Building 9204-3), and also in a separate facility of comparable size, have yielded pumping speeds of 50,000 to 69,000 liters/sec in the pressure range from 2×10^{-7} to 2×10^{-6} mm Hg. Sufficient titanium can be evaporated at one time so that the surface is not saturated for a period of several hours at 2×10^{-7} mm Hg. Sticking probabilities of 1/20 to 1/27 have been found for hydrogen. On saturation it has been found that titanium will remove hydrogen from the system on a molecule-for-molecule basis.

The large Vacion pump by Varian Associates has been tested under conditions simulating its proposed use as a "sink" for the unused portion of the high-energy H_2^+ beam in the DCX. High pumping speeds (up to 11,000 liters/sec) have been observed briefly at the beginning of a test. After pumping for 1 to 2 hr at a moderate leak rate (less than 80 liter-μ/sec) the speed is about 6000 liters/sec. (AEC Activity 5540)

Sherwood Project Chemistry. — Several forms of aluminum oxide and several aluminum silicates which are commercially used as catalysts have been studied for pumping hydrogen, carbon monoxide, methane, butane, and air. Although the sorption of butane exhibited some anomalies which may indicate multilayer adsorption, for none of these gases was the pumping speed large enough to indicate general applicability.

Apparatus has been obtained and is being installed for the purpose of preparing phosphor films which may be suitable for the analysis of energetic neutral hydrogen atoms. (AEC Activity 5540)

PROGRAM 6000 — BIOLOGY AND MEDICINE

BIOPHYSICS

Radiation Dosimetry Programs: Theoretical Physics of Dosimetry. — A program for the IBM-704 computer has been developed for calculating by stochastic methods the distribution in energy, angle, and space of the gamma-ray flux and the distribution in angle and space of the physical dose from a source distribution that may be extended in both energy and space. The method consists in following the histories of individual photons. When a photon crosses a predetermined boundary, the flux and dose at that point in phase space are modified so as to include the contribution of that photon. Instead of allowing absorption to take place when collisions are experienced, a statistical weight is assigned to the photon emerging from a collision. An optional biasing scheme is available whereby attention may be focused on deep penetration. The program has been arranged so as to be adaptable to various complex recording geometries in heterogeneous media; however, present investigations have been restricted to a recording geometry of concentric spherical surfaces about the origin in an infinite homogeneous medium. (AEC Activity 6140)

Radiation Dosimetry Programs: Experimental Physics of Dosimetry. — The specific fluorescence dS/dx of anthracene has been shown by Birks to depend on the specific energy loss (LET), dE/dx, for protons, alpha particles, and high-energy electrons as follows:

$$\frac{dS}{dx} = \frac{A(dE/dx)}{[1 + B(dE/dx)]} ,$$

where the constants A and B are the same for all three radiations. Evaluation of data taken at this Laboratory on the pulse-height—energy characteristic for monoenergetic low-energy electrons (10 to 125 kev) shows that these data also are in accord with the above equation. The generality of this relationship has suggested the development of an LET meter based on the nonlinear response of anthracene. Such a device may be constructed from a sandwich of a thin layer of anthracene and a thin inorganic phosphor or proportional counter. A particle traversing these layers at any angle will cause a pulse whose height is proportional to dS/dx in the anthracene and a pulse proportional to dE/dx in the other layer. The dE/dx of the particle may then be computed from the ratio of the pulse heights in the two layers (which is independent of the angle of traversal) and the above equation. Experiments have been conducted with an anthracene layer 0.2 mg/cm^2 thick evaporated in a vacuum onto the face of a 6199 photomultiplier. Pulses from alpha particles are obtained which are readily distinguishable from the background. These experiments indicate that such a device may be feasible. (AEC Activity 6140)

Waste Disposal Research and Engineering: Evaluation of Soil Disposal. — In previous studies naturally occurring phosphate minerals have shown promise in the selective removal of strontium from basic or neutral waste solutions. This prompted a comparative study of commercially available mineral materials that might be considered in the development of waste disposal facilities.

Preliminary slurry studies were conducted with vermiculite, Tennessee rock phosphate, Florida pebble phosphate, and variscite for the removal of strontium from a solution 0.3 M in $NaNO_3$ and 0.2 M in NaOH and containing the mass equivalent of ~ 0.1 μc of strontium per milliliter.

The weight of the sorbent material was varied to obtain 50% removal of strontium, and the distribution coefficients, K_d, were calculated. The results showed that variscite, with a K_d of ~ 5000 for 50% strontium removal, was far superior to the other materials investigated. The corresponding K_d values for Tennessee rock phosphate and Florida pebble phosphate were about 1000, whereas vermiculite showed a K_d of only 25.

The use of variscite, an aluminum phosphate having the formula $Al(OH)_2H_2PO_4$, shows definite promise for selective sorption of strontium. Column studies are in progress for investigation of continuous exchange properties and for evaluation of physical flow parameters. (AEC Activity 6440)

ENVIRONMENTAL SCIENCES

Ecological Studies. – Insects resident upon White Oak Lake bed, an area contaminated with various radionuclides during waste-disposal operations, showed differential accumulation of Sr^{90} and Cs^{137}. Per gram concentrations of Cs^{137} in insects were about 50% of the concentrations in plants, but for Sr^{90} the concentrations in insects were only about 7% of the plant values. Laboratory experiments with grasshoppers showed that the discrepancy may be explained by differences in elimination rates. In *Melanoplus differentialis*, the biological half life (T_b) for cesium was about two days, whereas the T_b for strontium was about 14 hr. In *Melanoplus femurrubrum*, the T_b for cesium was about 30 hr and that for strontium was about 12 hr. If similar relationships hold for most of the insect species on White Oak Lake bed, the equilibrium concentrations for Sr^{90} in insects could be expected to be lower than those for Cs^{137}. (AEC Activity 6480)

PROGRAM 8000 – WORK FOR OTHERS

Fission Product Recoveries. – About 99% of the strontium and 95% of the rare earths were extracted from Purex process waste that had been adjusted to 1 M tartrate and to pH 6 with NaOH. The extractant was 0.3 M di(2-ethylhexyl)phosphoric acid (0.2 M D2EHPA + 0.1 M NaD2EHPA) plus 0.15 M tributyl phosphate in Amsco 125-82. Three stages were used. Strontium and rare earths were stripped from the organic phase with 2 M HNO_3, giving the following decontamination factors: ruthenium, 24; zirconium-niobium, 9; calcium, 26. The rare earths were separated from the strontium by a factor of 10^4 by adjusting the strip solution to 0.005 M tartrate and pH 1 and extracting the rare earths with the same solvent. The strontium was extracted from the same solution, with the same solvent, after adjustment of the aqueous phase to pH 6. The product concentrations of the rare earths and strontium were 20 and 3.3 g/liter, respectively, with concentration factors of 25 and 30. (AEC Activity 8632)

Eurochemic Assistance Program. – The Eurochemic charter has been ratified by all countries except Italy and Sweden. The board of directors has been elected and a general manager appointed. Project IV, a plant with a daily capacity for 350 kg of natural uranium or 250 kg of uranium enriched up to 5%, has been developed within the available monetary framework ($21.77 million). Seven people associated with Eurochemic visited ORNL in October 1959. Process chemistry and engineering were discussed, along with associated fields of health physics, criticality, analytical work, and building design. (AEC Activity 8743)

Professional Health Physics Training Program. – Enrolled in the 4-hr graduate course in health physics being given during the second semester at Vanderbilt University by members of the Health Physics Division are 18 AEC Fellows, two Air Force officers, five Vanderbilt students, and three foreign students.

A ten-week course in health physics was presented by ORNL and ORINS for personnel from the AEC and from state health departments. The first two weeks were spent at ORINS on review of basic radiation physics; the remaining eight weeks were spent at ORNL on fundamentals of health physics, including training in applied health physics.

A series of 12 lectures in health physics was presented at the request of the Chemical Technology Division to a group of 20 employees from that division.

One AEC employee from the Washington office, one Army officer from Fort McClellan, Alabama, and one Japanese citizen are presently with the Health Physics Division for training in applied health physics. (AEC Activity 8780)

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

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82.	Union Carbide Corporation	R. W. McNamee
83.	Union Carbide Nuclear Company	C. E. Center
84.	Union Carbide Nuclear Company	J. L. Gabbard
85.	Union Carbide Nuclear Company	D. M. Lang
86.	Union Carbide Nuclear Company	J. P. Murray
87.	USAEC, Lockland Aircraft Reactors Operation	H. H. Gorman
88-89.	USAEC, New York Operations	J. Smith
90-91.	USAEC, San Francisco Operations	R. Hughey
92-93.	USAEC, Savannah River Operation	H. Rahner
94.	Westinghouse Electric Corp.	A. Squire

Internal

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

192. W. L. Morgan
193. E. J. Murphy
194. M. L. Nelson
195. C. E. Normand
196. J. L. Need
197. G. W. Parker
198. M. E. Ramsey
199. P. M. Reyling
200. A. W. Riikola
201. L. P. Riordan
202. G. A. Ropp
203. A. F. Rupp
204. A. D. Ryon
205. H. C. Savage
206. H. W. Savage
207. A. W. Savolainen
208. H. E. Seagren
209. E. D. Shipley
210. A. Simon
211. O. Sisman
212. J. R. Sites
213. M. J. Skinner
214. A. H. Snell
215. R. W. Stoughton
216. H. Stringfield
217. E. G. Struxness
218. J. C. Suddath
219. W. H. Sullivan
220. C. D. Susano
221. J. A. Swartout
222. E. H. Taylor
223. C. D. Watson
224. B. S. Weaver
225. A. M. Weinberg
226. M. E. Whatley
227. J. C. White
228. C. E. Winters
229. R. E. Worsham
230. W. Zobel
231. A. Zucker
232-233. Central Research Library
234. Health Physics Library
235. ORNL — Y-12 Technical Library, Document Reference Section
236. Reactor Experimental Engineering Library
237. Biology Library
238-429. Laboratory Records Department
430. Laboratory Records, ORNL R.C.