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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

OCTOBER 1958

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UNION CARBIDE CORPORATION

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OAK RIDGE NATIONAL LABORATORY
STATUS AND PROGRESS REPORT

October 1958

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

TABLE OF CONTENTS

	Page
PROGRAM 2000 — SPECIAL NUCLEAR MATERIALS	1
Metal Recovery	1
In-Line Instrumentation	1
Dissolver Solution	2
PROGRAM 4000 — REACTOR DEVELOPMENT	2
HOMOGENEOUS REACTOR PROJECT	2
Homogeneous Reactor Test	2
HRT Chemical Pilot Plant	3
Fuel Processing	3
Gaseous Fission Product Disposal	3
Thorium Oxide Slurry Development: Slurry Irradiations	3
Thorium Oxide Slurry Development: Oxide Preparation	4
HRP Chemistry: Fuel Studies	4
HRP Analytical Chemistry	5
Slurry Blanket System Development	6
Engineering Research	6
Core and Blanket Vessel Development	7
Centrifugal Circulating Equipment	7
Miscellaneous Equipment	7
Heat Removal Equipment	8
Gas Handling Equipment	8
Fuel Systems Development	8
Feed Pump Development	9
MOLTEN-SALT REACTOR (MSR) PROGRAM	9
Design	9
Experimental Engineering	10
Metallurgy	11
Reactor Chemistry	12
Radiation Damage	13
Physical Properties of Molten Salts	13
CIVILIAN-POWER ADVANCED GAS-COOLED REACTOR	13
Site and Power Plant Analyses	13
Fuel Element Fabrication and Inspection	14
Radiation Effects Studies	14
Reactor Physics	15
Applied Mechanics	17
Instrument and Control Investigations	17
Hazards Evaluations	17
Design Study of a Contaminated-Coolant GCR	18
Materials Compatibility Tests	18

TABLE OF CONTENTS (Continued)

	Page
Mechanical Property Tests of Capsule Tubing	20
Compressor Development	20
MARITIME SHIP REACTOR PROGRAM	21
Maritime Ship Reactor	21
GENERAL REACTOR RESEARCH	21
Basic Reactor Research	21
Fuel Element Development	22
Power Reactor Fuel Reprocessing: Alternate Aqueous Processes	23
Power Reactor Reprocessing: Darex Process	23
Power Reactor Fuel Reprocessing: Hermex Process	23
Power Reactor Fuel Reprocessing: Sulfex Process	24
Power Reactor Fuel Reprocessing: Zirflex Process	24
Power Reactor Fuel Reprocessing: Solvent Extraction Studies	24
Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes by Sintering	25
Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes Into Deep Wells	25
Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes Into Salt Formations	26
Waste Disposal Research and Engineering: Process Waste Water Treatment Plant	26
Waste Treatment	26
PROGRAM 5000 — PHYSICAL RESEARCH	27
ISOTOPE PRODUCTION	27
Radioisotope and Stable Isotope Production	27
Radioisotope Research and Development	28
Stable Isotope Development	28
PHYSICS	29
High-Voltage Experimental Program: Decay of I^{130}	29
High-Voltage Experimental Program: Neutron Capture Cross Sections	29
High-Voltage Experimental Program: The $Al^{27}(n,\alpha)Na^{24}$ Reaction Cross Section	30
Atomic Beam Spectroscopy	30
The 86-Inch Cyclotron: Nuclear Physics	30
The 86-Inch Cyclotron: Applied Physics	31
The 86-Inch Cyclotron: Operation	31
Heavy-Particle Physics	31
Spectroscopy Research Laboratory	32
Special Separations	33
Mass Spectrometry and Related Techniques	33
Mass-233 Studies	33
Raw Materials Research and Development	34
Neptunium Recovery	34
Extraction Reagent Performance: Radiation Damage	34

TABLE OF CONTENTS (Continued)

	Page
Feed Materials Processing	35
Ion Exchange Technology	35
High-Temperature and Structural Chemistry	35
Organic Chemistry	35
Radiation Chemistry	36
Molten-Salt Thermodynamics	37
Analytical Chemistry Research	37
METALLURGY	38
Fundamental Physicometallurgical Research	38
Microstresses in Metal Crystals	39
Ceramics Research	39
Fundamental Investigations of Radiation Damage in Solids	40
PROGRAM 6000 — BIOLOGY AND MEDICINE	40
BIOPHYSICS	40
Ecological Studies	40
PROGRAM 8000 — WORK FOR OTHERS	40
Professional Health Physics Training Program	40
MISCELLANEOUS REPORTS	41
Open Literature Publications	41
Visitors	43

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

October 1958

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

Metal Recovery. — A total of 6.4 kg of enriched uranium was recovered from HRP reactor fuel. Only one cycle of solvent extraction was required for decontamination of the uranium from fission products. Solvent extraction losses were less than 0.1%, and a gross-gamma decontamination factor of 4×10^3 was obtained; 2.5% tributyl phosphate was used as solvent.

The plant has been decontaminated, and the equipment is being extensively modified to prepare the plant for a power reactor fuel reprocessing program.

In the Thorex Pilot Plant, processing of all the available irradiated thorium was completed. The Interim-23 single-cycle flowsheet was used; the processing rate was limited to 450 kg/day by the dissolution rate. The U^{233} purification cycle was operated to process accumulated analytical waste.

The head end of the plant was effectively decontaminated with Turco reagents, and equipment modifications necessary to accommodate the power reactor fuel reprocessing program was started. (AEC Activity 2351)

In-Line Instrumentation. — Surface monitoring, liquid scintillation monitoring, and tape monitoring were tested as possible alpha scintillation counting methods at 100 and 500 $counts \cdot min^{-1} \cdot ml^{-1}$ for the continuous monitoring of Pu^{239} loss in radiochemical process raffinate streams of low activity level. Surface monitoring had the advantage of simplicity and the disadvantages of poor counting efficiency and background build-up effects. Liquid scintillation monitoring, with a higher counting efficiency than any type of surface monitoring tested, appears to be a feasible method for further investigation. Tape monitoring was efficient, and all three methods discriminated against gamma and beta rays. Stainless steel alpha scintillation cells are being fabricated for continuously monitoring the surface and liquid of flowing streams, and a tape monitor has been designed for fabrication. The tape monitor

to be fabricated will consist of an absorbent fabric impregnated at intervals with polyethylene; it is continuously moved from a supply reel to a falling sample stream, to a flash dryer, to a scintillator, and to a storage reel. (AEC Activity 2724)

Dissolver Solution. — Precision studies were completed on the determination of uranium by the following procedure: (1) salting the uranium from its major contaminants into 5% triisooctylamine in xylene with dibasic aluminum nitrate, (2) removing the uranium from this organic solvent by two contacts with 0.05 M $HClO_4$–1 N H_2SO_4 solution, and (3) titrating the uranium in the strippings coulometrically at controlled potential. The entire procedure requires approximately 45 min and has a relative standard deviation of 0.2% for solutions having a uranium concentration of about 10 mg/ml. The procedure has been applied successfully to synthetic and nonradioactive dissolver solutions of the stainless steel type. With this type of sample, addition of a reducing agent such as hydroxylamine to the sample before extraction was found to be beneficial. The extraction behavior of uranium in this solvent with chloride salting is also being studied. (AEC Activity 2724)

PROGRAM 4000 — REACTOR DEVELOPMENT

HOMOGENEOUS REACTOR PROJECT

Homogeneous Reactor Test. — Detailed observations of the core vessel, which began in September, were continued. The hole which has been in the core vessel since April 4 was examined closely and was photographed. The hole seems to be very nearly circular in shape and between 1 and 2 in. in diameter.

Detailed observations of the north half of the core exterior and the pressure-vessel interior revealed no change from their appearance before run 17. There was no indication of localized attack on any of the sharp-edged hardware such as the pressure vessel inlet nozzles. There was a very shallow accumulation of solids in the bottom of the pressure vessel.

A series of 2-in. holes was drilled in the upper six diffuser screens. A tilting-mirror periscope with a light source under the mirror enabled observations to be made of the inner core wall between the various screens. Observation and probing with a lifting hook revealed that the major portions of the first, second, and fifth screens from the top are detached from the core vessel. Most of the 22 corrosion specimens left in the core were found to be below the top screen. Six of the specimens were recovered. Retrieving operations will continue.

The shield was flooded, and underwater installation of a new entrance line to the chemical processing plant is in progress. (AEC Activity 4103.3)

HRT Chemical Pilot Plant. — The results of the final 102-hr operating period of HRT run 17 were similar to those of the preceding 438-hr period (reported in September) except for a somewhat reduced collection rate for uranium. The corrosion-product oxides that were removed totaled 51 g and analyzed 49% Zr, 19% Fe, 4% Cr, and 1% Ti. Only 12.6 g of uranium was removed, apparently as a second phase. The removal rates of 0.5 g of corrosion products per hour and 0.12 g of uranium per hour compare with corresponding rates of 0.34 and 0.17 g/hr in the preceding run.

The initial fuel charge was transferred from the reactor site to Building 3503, where it was decontaminated by solvent extraction. Gadolinium was used as a soluble poison for criticality control to permit transfer of the 6-kg charge in one batch in an existing 125-liter shielded carrier. (AEC Activity 4103.1)

Fuel Processing. — At 250 to 300°C the valence-state distribution of iodine dissolved in simulated HRT fuel solution (0.04 m UO_2SO_4, 0.02 m H_2SO_4, 0.005 or 0.03 m $CuSO_4$, 10^{-5} m HIO_3) was not affected by varying the radiation intensity from 0 to 23 w per kg of H_2O or by increasing the initial oxygen pressure from 100 to 400 psi at room temperature. Experiments with sampling equipment similar to that used for HRT samples showed that iodine loss to the container walls and rubber closures under irradiation ranged from 30 to 85% after 24 hr at 23 w/kg.

Electrolysis of simulated HRT fuel solutions in D_2O concentrated to 60 g of U per kg of D_2O by use of a mercury cathode resulted in 91 to 99% removal of nickel, with uranium losses of less than 0.1%. At the normal fuel concentrations (~10 g of U per kg) nickel was removed less efficiently, and uranium losses were 25 to 30% under the same experimental conditions as those used for the more concentrated solution. (AEC Activity 4103.1)

Gaseous Fission Product Disposal. — Additional study of the gaseous products resulting from the combustion of charcoal in oxygen showed that the ratio of the concentration of carbon dioxide to that of carbon monoxide increases as the linear velocity of the oxygen sweep gas decreases, or, in effect, as the reaction temperature decreases. Under conditions simulating those present in the HRT charcoal beds, the ratio of carbon dioxide to carbon monoxide was high. This is in accordance with the observed abnormal behavior of the HRT charcoal bed during HRT run 17. (AEC Activity 4103.1)

Thorium Oxide Slurry Development: Slurry Irradiations. — The radioactivity from a gas sample obtained from the slurry of 1000°C-fired thorium-uranium oxides (2.8% U^{235}) which had been irradiated in a gas-production run in the ORNL Graphite Reactor is decaying with a 2.74-hr half life, suggesting either Kr^{88} or I^{132}. A slurry of 1000°C-fired thorium-uranium oxides (U/Th = 0.05) with a D_2 atmosphere under irradiation

in the LITR is operating normally and shows essentially no radiolytic-gas pressure.

Samples of the slurry (500 g of Th per kg of D_2O) of 1600°C-fired oxide prepared for the in-pile slurry loop and under test in a high-temperature loop run were obtained 67 and 168 hr after the addition of 0.008 m MoO_3 to the loop system. In gas-recombination studies both slurries showed an induction period for catalysis, but after three experiments with each slurry the recombination rates of stoichiometric mixtures of gas at 280°C were between 3 and 4 moles of D_2 per hour per liter at a D_2 partial pressure of 100 psi. At a concentration of 250 g of Th per kg of D_2O a slurry of 1000°C-fired thorium-uranium oxides (U/Th = 0.05) prepared from the coprecipitated oxalates and containing 1000 parts of palladium (added as the hydrous oxide) per million of thorium showed some deactivation on heating with oxygen at elevated temperatures. However, the recombination rate was still 3.5 moles of D_2 per hour per liter at 240°C and at a D_2 partial pressure of 100 psi. (AEC Activity 4103.1)

Thorium Oxide Slurry Development: Oxide Preparation. — A total of 300 lb of 1225°C-fired thorium-uranium oxide (U/Th = 0.005) was prepared for slurry loop tests. About 100 lb of this material was prepared from enriched uranium for use in the in-pile slurry loop run and associated out-of-pile tests. Thirty-five pounds of thorium-uranium oxide (U/Th = 0.08) was prepared for engineering tests by precipitating the uranium onto thoria and firing at 1050°C. Kilogram quantities of thorium-uranium oxides having U/Th ratios of 0.05, 0.08, 0.1, and 0.25 were also prepared by this technique for toroid tests.

A torch-feed assembly was developed which operates smoothly for the flame denitration of thorium-uranium nitrate solutions in an oxyacetylene flame. Mixed oxide of 1.5-μ mean diameter, 85 wt % below 5μ, having a surface area of 5.8 m^2/g was classified for removal of >5-μ material and submitted for toroid testing. Reflectors of MgO appear excellent for temperatures up to 1600°C but fail before 2000°C is reached. Fluidized-bed equipment installed for direct denitration of $Th(NO_3)_4$ solutions operated well with an initial ThO_2 charge and with water as the feed stream. Feed-system modifications were necessary to permit successful feeding of $Th(NO_3)_4$ solutions.

Cylindrical hydroclones successfully discharged circulating ThO_2 slurries into an induced underflow receiver. While conical hydroclones are more efficient, the cylindrical design reduced plugging of the underflow port. (AEC Activity 4103.1)

HRP Chemistry: Fuel Studies. — Equilibrium compositions of the two liquid phases in the system UO_3-SO_3-H_2O were determined at 300 and 325°C. From initial stoichiometric UO_2SO_4 solutions having concentrations of 0.8 to 2 m, the resulting light phases at 300°C show compositions from 0.3 to 1.5 m uranium and contain 5 to 10 mole % excess acid; the heavy-phase compositions are from 4 to 5 m in uranium with 1 to 8 mole % deficiencies

in acid. At 325°C the light-phase compositions are from 0.1 to 0.3 m in uranium and contain 15 to 20 mole % free acid; the heavy phases are from 6 to 8 m in uranium with 1 to 3 mole % deficiencies in acid. The exact final compositions depend on the specific molalities of the initial solutions.

The study of the five-component system UO_3-NiO-CuO-SO_3-D_2O was continued. Phase-equilibrium data were obtained in which three solid phases — $UO_3 \cdot UO_2SO_4$, $3CuO \cdot SO_3 \cdot 2H_2O$, and $NiSO_4 \cdot H_2O$ — are in equilibrium with solutions that contain varying amounts of total SO_3. Investigations of this type have substantiated prediction methods for solubility behavior in aqueous homogeneous reactor fuels.

The solubility of antlerite ($3CuO \cdot SO_3 \cdot 2H_2O$) at 300°C was determined in H_2SO_4-H_2O solutions from 10^{-3} to 1 m in total SO_3. For a saturated solution, at 10^{-3} m total SO_3, the mole ratio, m_{CuO}/m_{SO_3}, is 0.31; this ratio increases to a value of 0.73 at 1 m SO_3.

Laboratory experiments were performed to test whether the presence of peroxide in radioactive fuel samples would interfere with the free-acid determination. It was found that acid was liberated from peroxide deliberately added to the system, but the amounts of peroxide which could be present in a fuel sample without causing precipitation even before the analysis would probably not cause a major inaccuracy in the free-acid determination. The procedure was found to be very sensitive to the rate of addition of base near the neutral point of the titration.

Studies of the homogeneous catalysis of the hydrogen-oxygen reaction by dissolved copper salts in aqueous solutions at elevated temperature were extended to lower temperatures with improved temperature control. The effects of acidity, copper concentration, and temperature were determined at 175 and 200°C. Increasing the acidity definitely lowers the reaction rate at a given hydrogen pressure. (AEC Activity 4103.1)

HRP Analytical Chemistry. — The colorimetric method for determination of titanium, in which disodium 4,5-dihydroxy-m-benzenedisulfonate (Tiron) is used as the chromogenic reagent, was adapted to the analysis of slurries of thorium oxide which contain uranium, molybdenum, and chromium(VI). These three interfering elements are removed prior to formation of the colored titanium-Tiron complex in a 2 M acetate—0.05 M ethylenediaminetetraacetic acid (EDTA) solution. Molybdenum and chromate are extracted with hexone, or, if uranium is also present, the major portions of the molybdenum and chromate are removed by a single hexone extraction, after which the residual molybdenum and all the uranium are adsorbed on an anion exchange resin column. For titanium within the range 2 to 10 ppm, the coefficient of variation is about 10%.

A method was devised for the determination of uranium in tin. Over 99% of the tin is removed by volatilizing it as $SnCl_4$, after which the uranium in a 1 N HCl solution of the residue is determined by derivative

polarography by utilizing the uranium reduction wave at -0.22 v vs the S.C.E. For 0.5 to 2% of uranium in tin, the coefficient of variation is 3% or less.

Microgram quantities of iron in uranyl sulfate solutions containing copper, 0.5 M $CuSO_4$, 2 M H_3PO_4, 1 M H_2SO_4, and even 1 M HF have been successfully determined by a flame-photometric method. The iron is selectively extracted from 6 N HCl with 4-methyl-2-pentanone (hexone), and the emissivity is then measured in the hexone medium. For 2 to 10 μg of iron per milliliter, the coefficient of variation is 2%. (AEC Activity 4103.1)

Slurry Blanket System Development. — A fluidized bed of thoria or thoria-urania particles has many attractive features as a reactor blanket or fuel material. The greater part of the solids can be retained in the reactor proper, and the problems associated with the circulation of solids through pumps and heat exchange equipment are reduced. Preliminary tests were started to determine the behavior of fluidized beds of thoria. The objective was to determine whether excessive particle degradation would occur under out-of-pile conditions at temperatures of interest for homogeneous reactor operation. Spheres of thoria about 40 μ in diameter made by the Houdry process were fluidized in water at 285°C for 264 hr at a void fraction of about 0.71 (equivalent to a thoria bed density of 2900 g/liter). The measured attrition rate was 0.26 wt % per day, a satisfactorily low figure. (AEC Activity 4103.1)

Engineering Research. — Pressure-drop data obtained for the flow of non-Newtonian ThO_2-H_2O slurries through tubes 0.124, 0.318, and 1.030 in. in inside diameter have been analyzed. The friction factor associated with the nonlaminar flow data was correlated in terms of the Reynolds number, by replacing the viscosity with the coefficient of rigidity. The friction factor for the laminar flow data agreed with that predicted on the basis of the associated Hedstrom and Reynolds numbers.

Several 30-gpm loop runs were completed in which thoria caking factors were investigated. Autoclaving the thoria from batch LO-20 for 36 hr at 300°C in the presence of 800 ppm H_2SO_4 did not prevent cake formation in a subsequent loop run. Pretreatment of batch LO-20 thoria with a Waring Blendor resulted in a slurry which did not cake in subsequent loop runs. This change in caking behavior appears to be associated with the change in particle-size distribution.

Pressure and thermal stresses in bare spherical reactors were calculated for reactors operating at 2000 psi and 280°C and having diameters from 3 to 4 1/2 ft, thorium concentrations from 0 to 400 g of Th per liter, and either H_2O or D_2O as the moderator. It was assumed that the power density was the same in all systems and equal to that in a 4 1/2-ft-dia reactor operating at 5 Mw (3.7 kw/liter). For these conditions, reducing the core diameter reduced the total stress. Increasing the thorium

slurry concentration decreased the wall heat-generation rate and the thermal stresses. For stainless steel vessels, the stresses were below the permissible level for slurry concentrations greater than 200 g of Th per liter; slurry of any thoria concentration was permissible in stainless-steel-clad carbon-steel vessels. (AEC Activity 4103.1)

Core and Blanket Vessel Development. — A generalized correlation was developed for predicting wall shear stress or heat transfer coefficients in re-entrant cores. The key variable is a modified Reynolds number equal to $(R_s\theta\ V_j\rho/\mu)\ (R_j/R_s)$, where R_s and R_j are sphere and jet radii, respectively, $R_s\theta$ is the arc length between the jet impingement point and a particular point on the sphere, V_j is the jet velocity, ρ is the density, and μ is the viscosity.

The vaned annular inlet geometry being considered for the HRT replacement core was tested further. Vane angles of 0 and 10° produced weak unstable vortices, although solids were found to be removable from a bottom bypass.

Additional runs were made with the full-scale model of the present HRT core, in order to find an optimum location for a dip tube to be used in the reactor to collect solids. A heavy liquid (perchloroethylene), a nonflocculating solid (sand), and a flocculating solid (lead oxide) were all removed readily from a take-off near the junction of the top screen. However, since the mobility of these materials in the system is low, only a fraction of the amount injected was recovered.

Water runs made on a 30-in. slurry core model indicated that wall shear stresses of the magnitude required to prevent slurry settling were obtainable at a throughput of 300 gpm. (AEC Activity 4103.1)

Centrifugal Circulating Equipment. — The 300A loop pressurizer, which included a jet to contact the pressurizer vapor and circulating liquid, was changed to a more conventional design because of excessive gas entrainment into the loop. In early runs, leaks interrupted loop operations.

The hydraulic design of the Reliance 6000-gpm pump was completed; detailing of the motor portion is about 50% complete. Promising results were obtained in preliminary tests of a thermal-barrier seal which contains a soft-wire sealing insert. (AEC Activity 4103.1)

Miscellaneous Equipment. — Numerous tools and viewing devices were developed for the performance of specific hot operations in the HRT core and blanket. It was demonstrated that the hole in the HRT core could be located by permitting gas bubbles to issue from it and travel upward through water in the core until they were visible. This was performed successfully in the reactor, and the hole was located visually from the blanket side by means of an improved remote-mirror optical periscope.

Hole saws for cutting screen samples from the side of the top screen and from the center of the screens were developed. The latter tool was used to cut a 2-in. sample from each of the top six screens. The resulting channel was used to permit viewing of the hole in the core from the inside with a tilt-mirror device which had small lights fastened to the mirror.

Several devices for recovering samples, corrosion specimens, and lost objects on the top screen and in the screen system below were devised and used. (AEC Activity 4103.1)

Heat Removal Equipment. — The procurement of materials for test-model heat exchangers by the Griscom-Russell Company is proceeding satisfactorily, except for a delay in procurement of composite stainless steel—Inconel tubing. An order for such tubing on a best-effort basis is being placed.

Thoria slurries containing 200 to 1000 g of ThO_2 per liter were operated in a seven-tube horizontal heat exchanger. Although slurry dropped out at velocities below 1.8 fps, normal flow was resumed in all tubes when the head across the tube bundle was increased. Preliminary runs were made in an inverted-L-shaped tube bundle.

The titanium-lined heat exchanger made by the Pfaudler Company was tested hydrostatically and found to leak. Several cracks were found in the titanium lining of both heads of the exchanger. This experience is similar to previous experience with titanium-lined pipe. (AEC Activity 4103.1)

Gas Handling Equipment. — The Inconel natural-circulation recombiner began operation at 1500 psi pressure, with a gas input of 3/4 scfm of H_2 and O_2. The recombiner appeared to operate well, although some minor difficulties were experienced with the caustic electrolytic cells which supply gas to the system. (AEC Activity 4103.1)

Fuel Systems Development. — The HRT mockup, containing a titanium letdown heat exchanger and a titanium pressurizer in which the uranyl sulfate is boiled, was operated for 1726 hr at 1500 psi and 280°C without incident. The stainless-steel-to-titanium transition joints have shown no visible corrosion, and the generalized corrosion rate for the system has not exceeded 1.3 mils/year. The highest temperature in the system, however, was about 15°C below the heavy-phase formation temperature, according to fuel analysis.

The two fuel-feed-pump heads have operated 6936 hr and 11,128 hr, respectively, since the last diaphragm failure. The titanium letdown heat exchanger had operated for 4350 hr without trouble, and the titanium-to-stainless-steel joints have remained tight during this time. (AEC Activity 4103.1)

Feed Pump Development. — Prototype HRT fuel feed pumps and purge pumps continued to operate without difficulty. The normal life expectancy of these pumps in continuous service is now at least 1 1/2 years. The single-stage Roth turbine pump was dismantled for replacement of its shaft seal after 3370 hr of operation; the internal parts were found to be in excellent condition.

The duplex-diaphragm slurry feed pump operated satisfactorily for 1400 hr. Following this period, however, significant wear was observed in the tungsten carbide check-valve trim, which is being replaced with aluminum oxide.

Fixed capillary letdown devices were used successfully in the feed-pump system. A variable capillary valve also gave promising service. (AEC Activity 4103.1)

MOLTEN-SALT REACTOR (MSR) PROGRAM

Design. — A study of a natural-convection, molten-salt reactor of 576-Mw thermal output was made. The calculations indicate a salt volume of 1600 to 2400 ft^3, 20,000 to 50,000 heat exchanger tubes 6 to 17 ft long, and a riser 3.4 to 5.2 ft in diameter. The initial cost of a natural-convection reactor would be higher than that of a forced-circulation system, and the large number of heat exchanger tubes casts doubt on the reliability of the natural-convection system.

A rough survey of the nuclear characteristics of graphite-moderated molten-salt reactors that would utilize an initial complement of low-enrichment uranium fuel was made. It was found that reactors could be constructed that would operate with fuel enriched initially with as little as 1.25% U^{235}; initial conversion ratios as high as 0.8 could be obtained with a fuel enrichment of less than 2%. Highly enriched uranium would be added as makeup fuel, and such reactors could probably be operated to burnups as high as 60,000 Mwd/ton. The total fuel cycle cost would be approximately 1.3 mills/kwhr, of which 1.0 mill would be the cost of the enriched U^{235} that would be burned up.

An investigation of the initial nuclear characteristics of two-region, homogeneous molten-salt reactors fueled with U^{233} was completed. It was found that 8-ft-dia cores gave optimum regeneration ratios for a given fuel investment. A core having 1.0 mole % ThF_4 in the fuel salt required a fuel inventory of about 180 kg of U^{233} and gave a regeneration ratio of 0.905. Increasing the ThF_4 content increased both the inventory and the regeneration ratio; at 7 mole % ThF_4 the inventory is about 1300 kg and the regeneration ratio is 1.1.

A fuel drain tank design was evolved in which, for simplicity and ease of removal for maintenance, the coolant flows through a removable bayonet tube assembly that rests on the drain tank. No bolting or welding is required for replacement of the coolant assembly. (AEC Activity 4103.1B)

Experimental Engineering. — The assembly of the first molten-salt forced-circulation in-pile loop is nearly completed, and the loop is scheduled for insertion in the MTR during the December 1, 1958, MTR shutdown. The assembly of the second loop has proceeded to the point of installation of the pump, which is being held pending the outcome of pump tests that are under way.

The copper sealing rings in the 3-in.-dia pipe freeze-flange joints were replaced with aluminum rings of a modified design. A resin-base lubricant containing molybdenum disulfide and graphite was coated on the mating surfaces of the clamps and flanges and on the nuts, bolts, and washers. Room-temperature leak tests of the flanges with a helium leak detector yielded the lowest leakage rates obtained for any of the mechanical joints tested to date. The leakage rate was less than 6×10^{-11} cm^3/sec of helium. The flanges are being installed in loops for cycling tests.

The third in a series of tests of molten-salt-lubricated hydrodynamic journal bearings was conducted with the INOR-8 journal and bearing used in the second test and with the same salt mixture (62 mole % LiF—37 mole % BeF_2—1 mole % UF_4; salt 130) at a temperature of 1200°F. A total of 284 hr of operation at a journal speed of 1200 rpm was completed; during 168 hr of the test the radial bearing load was 200 lb, and during the remainder of the test the load was 300 lb. Also during the first 168 hr the shaft was started and stopped 30 times. Indications of very light rubbing were noted on both the journal and the bearing surfaces. Therefore a fourth test was started with new INOR-8 journal and bearing parts in order to determine whether the wear occurred because of the start-stop operations or was due to steady-state operation.

Water tests of the hydrostatic bearing were concluded. The experimental data which relate bearing flow, bearing load, and journal eccentricity to pump operating conditions are being analyzed. Bearings with 0.0075- and 0.005-in. radial clearance were tested. The scoring experienced in these tests was greatly reduced in comparison with that of the first test, in which a radial clearance of 0.003 in. was used, and can be attributed to a structural modification which provides uniform supply pressure to each of the four hydrostatic pockets.

Motorettes with heat and nuclear radiation-resistant electrical insulation were received from the Louis Allis Company. Apparatus for measuring the electrical properties of the test insulation at temperatures up to 1200°F was assembled. Tentative plans also call for testing the motorettes at elevated temperatures in a gamma-ray flux.

Five additional forced-circulation corrosion-testing loops were placed in operation, making a total of 15 operating loops. The loops consist of a centrifugal pump, a cooled section, and two resistance-heated sections, and they are circulating fused-salt mixtures at maximum temperatures of 1200 or 1300°F with temperature differences across the

loops of 200°F. Four of the loops are of Inconel and the remaining eleven are of INOR-8. The loops are intended for long-term operation; one loop has accumulated over 8500 hr of operation.

The molten-salt thermal-convection loop facility is operating at its full capacity of 31 loops. Three loops were shut down and four were started during this period. The operating loops include 16 fabricated of INOR-8 and 15 fabricated of Inconel.

A small frozen-lead-sealed pump that consists of a centrifugal pump mounted vertically, a frozen lead seal, and a fractional-horsepower motor drive has operated continuously for more than 3000 hr. The pump is circulating a fused-salt mixture isothermally at 1200°F.

Inconel and INOR-8 corrosion specimens exposed to aluminum chloride gas at 1200°F for 1000 hr in a small vapor-pressure testing apparatus were examined. There was intergranular attack of the Inconel specimens to a depth of 2 mils and grain boundary penetration of the INOR-8 specimens to a depth of 0.5 mil. This preliminary test suggests that if aluminum chloride gas proved to be a desirable reactor coolant, INOR-8 would probably be a satisfactory container material. (AEC Activity 4103.1B)

Metallurgy. — INOR-8 specimens subjected to 5000-hr aging heat treatments at temperatures from 1000 to 1500°F were found to have tensile strengths approximately equal to the tensile strength of annealed INOR-8. Thus it appears that aged INOR-8 has high-temperature stability.

Activation energy values of 82,700, 83,600, and 84,400 $cal \cdot mole^{-1} \cdot (^{o}K)^{-1}$ were established for recrystallization, relaxation, and creep, respectively, of INOR-8 in the temperature range from 1100 to 1700°F. INOR-8 creep tests conducted to establish reproducibility showed good correlation up to 5% creep. At 20,000 psi and 1250°F, rod specimens exhibit a creep rate lower than that of sheet specimens.

Tensile properties of INOR-8 and Inconel held in a sodium-graphite environment for 400 hr at 1200°F showed no change from the tensile properties of unexposed material. Creep tests of similar specimens held at 1600°F for 40 hr showed the carburized specimens to have improved in time to rupture, and they had lower creep rates. INOR-8 specimens exposed to a salt 130—graphite system at 1300°F for 2000 hr were slightly carburized and exhibited a 10% increase in tensile strength with an accompanying 10% decrease in ductility.

Improvement in INOR-8 weld metal ductility was attained by heat-treating at 1800°F for 4 hr to spheroidize the eutectic carbides. Low-carbon-content welding wire has been manufactured in an additional attempt to improve ductility through minimizing carbides.

The attachment of molybdenum to nickel-base materials was attempted with the use of copper, gold-nickel, and Coast Metals 52 braze materials.

Only copper-brazed joints were free of cracks. When the attachments were thermally cycled, all the joints cracked, but the Coast Metals 52 brazed joints cracked the least. (AEC Activity 4103.1B)

Reactor Chemistry. — Recent thermal-gradient quenching experiments have shown a eutectic with the approximate composition 50 mole % NaF—47 mole % BeF_2—3 mole % ThF_4 (mp 377°C). Solid phases present at this invariant point are β-$NaF \cdot ThF_4$, $NaF \cdot 2ThF_4$, and $NaF \cdot BeF_2$. The appearance of β-$NaF \cdot ThF_4$ at this invariant point indicates that it is an equilibrium phase in the system NaF-ThF_4.

Solubility data obtained with CeF_3-PuF_3 mixtures in the solvent LiF-BeF_2 (63-37 mole %) can be correlated with the use of the equation

$$N_{PuF_3(d)} = S^{o}_{PuF_3} N_{PuF_3(ss)} \quad ,$$

where $N_{PuF_3(d)}$ is the solubility of PuF_3 in the mixture at a specified temperature, $S^{o}_{PuF_3}$ is the solubility of PuF_3 in the pure solvent at the same temperature, and $N_{PuF_3(ss)}$ is the mole fraction of PuF_3 in the solute mixture. It was found that 1.0 mole % BaF_2 depressed the solubility of PuF_3 in the LiF-BeF_2 (63-37 mole %) solvent but to such a small degree as to indicate that divalent fission products in a plutonium-fueled fused-salt power reactor would be unlikely to have a significant effect on the solubility of plutonium in this solvent.

Preliminary thermal analysis data obtained with four LiF-PuF_3 mixtures indicate that there is a eutectic in this binary system that contains approximately 18 mole % PuF_3 and melts at 745°C. The apparatus used in these studies was tested with LiF-CeF_3 mixtures, and the thermal effects obtained with 0.8 g of LiF-CeF_3 (90-10 mole %) compared favorably with those obtained with 50- to 100-g quantities in conventional thermal analysis apparatus.

Extraction of SmF_3 from an LiF-BeF_2-UF_4 mixture with solid CeF_3 has been shown to be effective at low concentrations (800 ppm) of SmF_3. The concentration of SmF_3 remaining in solution can be calculated with fair accuracy from the solubilities of the two fluorides, the volume of liquid, and the amount of CeF_3 added.

Chromous fluoride for use in many experiments has been prepared by a process which includes, as its final step, the reduction of CrF_3 with H_2 at elevated temperatures. Such material is quite free of extraneous cationic impurities, but it is contaminated by small amounts of metallic chromium. It has been shown that complete conversion of metallic chromium to CrF_2 can be obtained by reaction with SnF_2 or PbF_2. An improved procedure for preparation of this material has been developed. (AEC Activity 4103.1B)

Radiation Damage. — Two graphite capsules enclosed in Inconel and filled with a molten-salt fuel were examined after irradiation in the MTR for 1610 and 1492 hr, respectively, at 1250°F to integrated power densities of 1520 and 1375 kwhr/cm^3. The fuel mixture that surrounded the graphite was $LiF-BeF_2-UF_4$ (62-37-1 mole %). Preliminary examinations have indicated that the graphite was unaffected by the irradiation. Thus it appears that the life of a fuel-cooled graphite moderator would not be limited by damage to the graphite by the fuel. (AEC Activity 4103.1B)

Physical Properties of Molten Salts. — The enthalpies, heat capacities, and heats of fusion have been determined for six additional salt mixtures. Of these, two were alkali metal—beryllium fuels, and the other four were chlorides of possible interest as secondary coolants. The heat capacities for these mixtures within the indicated temperature range of the liquid state are expressed by the following equations, in which c_p is in cal/g, and T is in °C:

(1) $NaF-BeF_2-UF_4$ (53-46-1 mole %), $c_p = 0.312 + (21.74 \times 10^{-5})T$ from 400—800°C;

(2) $LiF-BeF_2-UF_4$ (53-46-1 mole %), $c_p = 0.539 - (3.80 \times 10^{-5})T$ from 400—800°C;

(3) $NaCl-CaCl_2$ (49-51 mole %), $c_p = 0.358 - (15.40 \times 10^{-5})T$ from 550—850°C;

(4) LiCl-KCl (70-30 mole %), $c_p = 0.361 - (6.24 \times 10^{-5})T$ from 450—800°C;

(5) LiCl-KCl (60-40 mole %), $c_p = 0.315 - (0.36 \times 10^{-5})T$ from 400—800°C;

(6) LiCl-KCl (50-50 mole %), $c_p = 0.239 + (9.24 \times 10^{-5})T$ from 500—800°C.

Preliminary values for the surface tension of the mixture $LiF-BeF_2$ (63-37 mole %) can be described by the equation

$$\sigma \text{ (dynes/cm)} = 235.5 - 0.09T \text{ (°C)}$$

to within ±5% over the temperature range of 460 to 750°C. (AEC Activity 4103.1B)

CIVILIAN-POWER ADVANCED GAS-COOLED REACTOR

Site and Power Plant Analyses. — Detailed information was compiled on two local sites being considered by the AEC Site Selection Committee for construction of the Gas-Cooled Prototype Reactor (GCPR) currently being designed by Kaiser-ACF. Several advantages would accrue from the selection of a local site; in particular, the Clinch River provides adequate cooling water, and the AEC facilities at Oak Ridge need the electric power that would be supplied.

Studies of the problems of steam plant control and the integration of the steam system controls into the over-all control system for the GCPR were initiated. (AEC Activity 4110)

Fuel Element Fabrication and Inspection. — An apparatus for automatic inert-gas shielded-arc fusion welding of end caps to fuel capsules was completed, and samples for evaluation of welding conditions for three types of end caps were prepared. Capsules with end closures of the three types (plug caps, butt-welded caps, and dished-head seam-welded caps) were subjected to stress-rupture tests at 1500°F under a helium pressure of 1500 psi in order to detemine the short-time rupture strength of the end closures. All three specimens failed in the tube area in less than 5 min, and thus it appears that satisfactory welding conditions have been established for all three designs. Further capsule stress-rupture tests will be required in order to determine the optimum welding conditions.

Shrinkage factors for the manufacture of cored pellets of 1.0% enriched UO_2 were determined, and dies for molding and sintering cored pellets of the desired size were ordered. Fabrication of solid prototype pellets of normal-enrichment material was started. The pellets that do not require central holes are being machined from solid slugs. The fabrication of miniature pellets with a density of 95% (10% enriched) was nearly completed. A fuel element assembly area was set up, and equipment for small production runs is being assembled.

A survey of helium leak detection techniques applicable to the inspection of completed fuel capsules was made, and a preliminary procedure specification was prepared. Ultrasonic inspection methods are being used in the inspection of the type 304 stainless steel fuel capsule tubing used for experimental work, and ultrasonic techniques for inspecting end cap welds are being developed. (AEC Activity 4110)

Radiation Effects Studies. — The second miniature prototype fuel capsule was removed from the LITR on September 30. It had been exposed to a neutron flux of 2×10^{13} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1} for 1962 hr. The internal temperature at the top of the capsule during the irradiation was maintained constant at 1770°F. At termination of the irradiation the outer temperature at the top of the shell was 1140°F and at the bottom it was 1385°F. After a decay time of 12 hr the observed activity was 1000 r/hr at a distance of 1 ft from the capsule, and the activity through 6 in. of lead was 450 mr/hr. The total gas activity measured at contact was 800 mr/hr.

Gas samples were taken from both ends of the capsule. Spectrometer analyses, as yet incomplete, have shown a clear Xe^{135} peak; Kr^{85} was also detectable. Photographs of the capsule indicate that no swelling occurred. The capsule will be shipped to the General Electric Company Vallecitos Laboratory for disassembly and examination.

Two specimens of coated graphite were discharged following irradiation in the ORR for one reactor cycle. These samples received a dose of

about 6 x 10^{19} neutrons/cm^2, No changes in appearance, weight, or density were evident. A high-temperature oxidation test is under way for determining the postirradiation integrity of the coating. Two similar specimens were left in the ORR for more extended irradiation.

Thermocouple irradiations were concluded in the LITR, but similar tests are continuing in the ORR. Instrumentation is being prepared for testing the irradiated thermocouples in a hot cell.

Negotiations were completed and final plans were made for the hot-cell work to be done by G-E personnel at the Vallecitos Laboratory. Miniature and stressed UO_2 dummy capsule assemblies which duplicate the three capsules already irradiated were supplied to G.E. for testing hot-cell techniques of disassembly and examination.

A miniature prototype fuel capsule was inserted in the LITR on September 30. The pellets are in an argon atmosphere. The central pellet temperature is 2300°F, and the outer can temperature at the top of the can is 1100°F; at the bottom of the can the temperature is 1300°F.

Design work continued on equipment for irradiations in the ORR, the ETR, and the GETR. The unperturbed flux in one typical position proposed for these experiments in the ORR has been measured to be 2 x 10^{13} neutrons·cm^{-2}·sec^{-1} by activation of cobalt wire. (AEC Activity 4110)

Reactor Physics. — Calculations have been made of the neutron source strength required in the GCR-2 and in the GCPR to give a thermal-neutron flux of 14 neutrons·cm^{-2}·sec^{-1} in the thermal column, with a k_{eff} = 0.97. These calculations were made by using multigroup diffusion theory. The geometrical model in each case included the core, the reflector, the pressure vessel, and a graphite shell about 6 in. thicker than the thermal column. A portion of the reflector was spread out, with diminished density, to fill the space inside the pressure vessel. For GCR-2, the calculations indicate a thermal column flux of 1 neutron·cm^{-2}·sec^{-1} per watt of reactor power and for the prototype 100 neutrons·cm^{-2}·sec^{-1} per watt. For k_{eff} = 0.97, this means that a source of some 3.2 x 10^{10} neutrons/sec would be required for GCR-2, and for the prototype about 3.4 x 10^8 neutrons/sec would be required. The corresponding strengths of a Po-Be source are 10^4 and 10^2 curies, respectively. A calculation at ACF in which the source and the neutron flux were expanded in cylindrical harmonics and a calculation at ORNL in which the source and flux for a slab reactor were expanded in a cosine series both indicated that the fast-neutron leakage from the reactor is given quite accurately ($\sim$10%) if only the normal-mode flux distribution is taken into account, for k_{eff} = 0.97. Thus a multigroup calculation is adequate for the subcritical case.

In another calculation, the effect of boron in the outer 6 in. of the reflector was studied. It was found that (1) the reactivity of the reactor is diminished very slightly, because the flux rises somewhat less in the reflector, and hence the inward current of thermal neutrons into

the core is less; (2) the thermal flux in the thermal column is decreased by about a factor of 3; and (3) the thermal neutron flux in the pressure vessel is diminished a thousandfold and captures of all neutrons several hundredfold.

Results of the same multigroup calculations were used to predict the integrated neutron dose to the pressure vessel. The epicadmium flux obtained from this calculation is an order of magnitude lower than that estimated by ACF. Results of the calculations are summarized in the following table.

Neutron Energy*	20-Year Integrated Neutron Dose (neutrons/cm^2)		Attenuation of 6-in. Graphite**
	Inner Surface of Pressure Vessel	Outer Surface of Pressure Vessel	
>0.5 Mev	3.9×10^{17}	1.1×10^{17}	7.4
>10 kev	2.1×10^{18}	6.3×10^{17}	6.7
>100 ev	6.7×10^{18}	1.6×10^{18}	6.3
>0.4 ev	2.1×10^{19}	3.5×10^{18}	5.5

*Doses in each row are for all neutron energies above the energy given.
**Factor by which additional 6-in. reflector would reduce dose.

Calculations of the graphite-moderated Th-U^{233} breeder reactor were continued. The following configuration and parameters of the reactor were chosen as typical of a general case: core diameter, 40 in.; over-all length, 40 ft; void fraction, 0.2; initial carbon-to-U^{233} atom ratio, 4000:1 with no thorium in the core; blanket thickness, 3 ft (1.5 to 2 ft would be adequate); carbon-to-thorium atom ratio, 20:1; graphite buffer, 6-in. thickness between core and blanket. In the buffer the density would be adjustable by mechanical means from 20 to 100% of the graphite density in order to control core leakage. The results of the calculations indicate that the neutrons are not well moderated for low carbon-to-U^{233} ratios; since η for U^{233} is considerably lower in the resonance energy range than for thermal neutrons, poor moderation produces a low average value of η. For a well-moderated spectrum, on the other hand, parasitic absorption in the graphite becomes important. The product ηf has a maximum for a carbon-to-U^{233} ratio of about 5000. The value of η for an infinite core of this composition is about 2.18, compared with the value 2.28 taken for 2200-m/sec neutrons. This lower value of η for a poorly moderated spectrum is decisive. Neutron absorptions in graphite and in xenon and samarium, together with spectral shifts associated with the detailed core geometry, lower the achievable conversion ratio to about 1.06. During a core burnup cycle in which

approximately 30% of the U^{233} in the core is consumed, other fission poisons further reduce the cumulative conversion ratio to about 1.03 to 1.04. In addition, a loss by neutron absorption of about 1% of the Pa^{233} formed, together with a loss of 1% of reprocessed core material and 0.8% of the U^{233} in blanket recovery, further reduce the conversion ratio to about unity. Thus, even though the specific power in the core is as high as 10,000 kw per kg of U^{233}, the time required to double the whole inventory of U^{233} would be many decades, if breeding were in fact achieved. This result is very sensitive to the average value of η in the core. New information on $\eta(E)$ for U^{233} could significantly affect the breeding gain. Further, the use of a D_2O moderator, even for a heterogeneous reactor, might make possible a much more thermal spectrum without excessive neutron absorption in moderator or structure. (AEC Activity 4110)

Applied Mechanics. — Stress analyses were completed of the designs for the ETR and ORR gas-cooled in-pile loops. The stresses that would occur in the two jackets surrounding the fuel elements under normal operating conditions were calculated, and the results indicated that hemispherical heads should be used rather than the proposed dished heads. With this change, the maximum combined stress in the outside cylinder was found to be less than that allowed by the ASME code.

Design analyses of the GCPR were initiated. Stresses in the reactor pressure vessel and in the primary coolant piping are being studied. Since the designs have not been completely established, a range of parameters is being used in the calculations. (AEC Activity 4110)

Instrument and Control Investigations. — An information block diagram of the proposed GCPR safety system was prepared on the basis of preliminary reports and drawings received from Kaiser-ACF.

Information was compiled on the temperature profile across the GCR-2 fuel element, gas channel, and graphite for use in analog computer analyses of the reactor system. (AEC Activity 4110)

Hazards Evaluations. — A method was derived for calculating the diffusion of fission products from graphite-UO_2 fuel elements. The IBM 704 was coded for the diffusion calculation, including radioactive decay, and values of escaped activity were obtained as a function of temperature based on diffusion rates extrapolated from data reported in the literature for long-lived isotopes at higher temperatures. The calculations were made both for 91% dense UO_2 and for graphite containing UO_2 particles 200μ in diameter. More activity was found to escape from UO_2 than from C-UO_2 at high temperatures. This result was expected because of the high activation energy (45 kcal/mole) for the diffusion of xenon through graphite and the assumption (in lieu of experimental data) that the diffusion of the more volatile fission products was the same as that for xenon. (AEC Activity 4110)

Design Study of a Contaminated-Coolant GCR. — Preliminary layouts of a contaminated-coolant GCR nuclear power plant were made. The containment vessel is ~270 ft in diameter so as to provide access to components which are to be remotely serviced or maintained.

The power recovery equipment for the 300-psi system consists of eight scaled-up GCR-2—type steam generators equilaterally placed around the reactor, with each feeding steam to a turbine. Each of these heat exchangers removes 393.5 Mw of heat, including the 9000-hp input of each blower. Gas is introduced at 1500°F and is cooled to 506°F at the exit from the economizer. It then runs through the blowers, and the full flow is ducted upward through an annular region to cool the pressure shell, after which it leaves the steam generator through an annular pipe at 525°F.

The tube matrices and dimensions are identical with those in the GCR-2 steam generators, with the surface proportions in the superheater, the boiler, and the economizer regions altered to fit the new conditions. The total tube matrix height is only 3 ft more than that of the GCR-2, and each steam generator removes 2.25 times as much heat. Because of the radiation hazard, the steam headers and economizer headers will be exterior to the shell. The need to remove the superheater drum from the inlet hot gas, coupled with the desire to allow manual tube blanking in case of failure, justifies this design feature.

With 1450-psi, 1050°F outlet steam conditions, the allowable feed-water inlet temperature with a boiler peak temperature difference of 50°F is 480°F. A simple regenerative cycle with four feed-water heaters will yield a thermal efficiency of 40.47%; with six heaters the efficiency is increased to 40.85%. Without feed-water heating, the efficiency would be 35.9%. Cycle analyses show that at these steam conditions at least four feed-water heaters are justifiable. (AEC Activity 4110)

Materials Compatibility Tests. — Out-of-pile exposures of fuel capsule and reactor structural materials plus various grades and types of treated and untreated graphite in controlled gaseous environments were continued. Static pot test 6 was completed, and tests 7 and 8 have been started. Test 5 of a series of tests in a low-pressure natural-convection loop was initiated. Procurement and assembly work continued on both a high-pressure thermal-convection loop and a high-pressure forced-convection loop.

Static pot test 6, which contained helium, TSF graphite, and structural materials, had completed 100 hr at 500°F and 900 hr at 1250°F. Analyses of helium gas samples taken during the test indicated that the impurities given off by the structural materials and the graphite samples were less by a factor of 2 to 3 than those given off in a previous similar test at 1400°F. The major impurities found were H_2, CH_4, N_2, and CO. The surfaces of all the stainless steel and low-alloy steel samples showed only slight oxidation. The DB-2 alloy (7% Al—5% Cr—bal Fe) samples, however, had a tightly adherent black scale. The scale was

present both on the samples in contact with the graphite and on the samples suspended above the graphite.

An investigation of the reaction products which result from the exposure of type 304 stainless steel to low-pressure CO_2 at 1500°F was continued. The film which forms at the gas interface in an atmosphere of argon + 0.6% CO_2 was identified as rhombohedral $3Cr_2O_3 \cdot Fe_2O_3$ by transmission electron-diffraction methods. A very thin film, identified as $MnO \cdot Fe_2O_3$, was found between the bulk oxide and the metal interface. This film has a cubic spinel structure and exhibits preferential orientation. Spectrographic analyses of the stripped oxide indicate that Fe, Mn, and Cr are the principal constituents.

Type 304 stainless steel clippings 0.002 in. thick were exposed to CO_2 at a reduced pressure ($\sim$20 mm Hg) at 1500°F for 336 hr in a closed quartz capsule to determine what gaseous reaction products result from the oxidation reaction. Spectrographic analyses of the gases before and after test showed the following results:

Gas	Amount Present (ppm)	
	Initially	After Test
H_2O	33	9
Hydrocarbons	13	None
N_2 + CO	None	16
A	3	1
CO_2	25,300	1.3

A similar test was conducted in which the gas was CO. The test in this case was conducted to determine the ambient gas pressure change by manometric methods as the reaction proceeded. Because of an inadequate reaction chamber, the reaction rate could be followed for only 400 min, but during this time the CO pressure decreased either logarithmically or parabolically. The sample was oxidized, as predicted, and formed a tenacious, green, nonmagnetic film. A weight increase of 160 $\mu g/cm^2$ was noted after 125 hr at 1500°F.

These data show that both CO and CO_2 oxidize type 304 stainless steel at 1500°F. Evidence of either carburization or decarburization is not apparent from the limited number of analyses conducted. Tests are in progress to determine the reaction products between CO and the metal.

Under static conditions it appears that the principal gaseous reaction product in a CO_2 atmosphere is CO. The CO, in turn, reacts

with the metal to form more oxide. This conclusion is based on the fact that with a starting pressure of 20,750 μ CO_2, the final pressure was only 20 μ, virtually a 100% reaction. (AEC Activity 4110)

Mechanical Property Tests of Capsule Tubing. — A tube-buckling test at 1300°F with an external pressure of 300 psi was discontinued at the end of 2200 hr. The type 304 stainless steel tubes had been machined to 0.020-in.-thick walls. Diametral clearances between the tube walls and inserts were 0.002, 0.005, 0.010, and 0.020 in. All the specimens had collapsed to some degree. The specimen with the diametral clearance of 0.002 in. was the only one with even collapse. No wrinkles were formed on the other specimens, but they were considerably out of round. These specimens will be tested further to see whether wrinkles will form with additional collapse.

Another tube-buckling test of 0.010-in.-wall commercial type 304 stainless steel tubing was discontinued at the end of 500 hr at 1500°F, with a 300-psi external pressure. Diametral clearances between tubes and inserts were 0.002, 0.005, and 0.010 in. The 0.002-in.-gap specimen was evenly collapsed. The 0.005-in.-gap specimen buckled in a one-lobe mode. Two wrinkles were formed on the specimen with the 0.010-in. gap.

One creep-buckling specimen with a 0.010-in. gap between the tube and the inserts was tested at 1900°F under an external pressure of 300 psi. Examination of the specimen at the end of 8 hr showed that even collapse had occurred. A similar specimen examined after 25 hr at 1800°F with a 300-psi external pressure had also collapsed uniformly.

A tube-burst specimen of type 304 stainless steel subjected to 300-psi internal pressure at 1500°F ruptured in 454 hr. The fracture occurred around the end plug, as well as in the axial direction.

A hot-streak test conducted at a 300-psi external pressure with a uniform temperature of 1300°F and a hot streak of 1500°F was discontinued at the end of 100 hr. The original diametral clearance was 0.010 in. The tube was collapsed in an uneven manner around the insert, but no wrinkles were formed. This specimen will be tested an additional 500 hr and re-examined. (AEC Activity 4110)

Compressor Development. — Development studies were continued on two types of compressors for the gas-cooled loop for testing full-size GCPR fuel elements in the ORR. The A-Tool and Gage Company of Detroit, Michigan, was selected as the fabricator of the proposed grease-lubricated compressor being designed at ORNL. Consideration is being given to a proposal received from Armstrong-Siddeley, Conventry, England, for supplying a compressor with gas-lubricated bearings. The AiResearch Mfg. Co. is fabricating the compressor required for the out-of-pile loop in which argon will be circulated. Work continued on facilities for performance testing of bearings and compressors. (AEC Activity 4110)

MARITIME SHIP REACTOR PROGRAM

Maritime Ship Reactor. — The Laboratory continued to furnish technical assistance to the Maritime Reactors Branch (AEC) in the program to develop nuclear-powered merchant ships. Additional drawings and specifications for the NS "Savannah" reactor system were reviewed, and comments were transmitted to MRB. Laboratory field inspectors continued to witness inspections and tests performed at vendor's plants during the fabrication of reactor components. The area of inspection assistance was expanded to include fuel elements and other core components.

Deboronization experiments were conducted on type 304 stainless steel tubing which initially contained 240 ppm boron. The specimens were subjected to four different brazing atmospheres at four temperatures from 1850 to 2150°F. Results indicate that essentially no deboronization occurred in either argon or vacuum. However, significant losses were observed in hydrogen atmospheres, with the losses being greater when the dew point was increased from -95 to -40°F.

Experiments indicate that Au, Ni, and V wires may be useful in wire pyrometer techniques for measuring UO_2 pellet temperatures during irradiation tests. Photographs of the melting process were prepared for use as standards for postirradiation comparison. Platinum wire was found to be unsatisfactory; it apparently reacted with its surroundings instead of melting. Incorporation of boron into UO_2 by coprecipitation with ammonium diuranate from a uranyl nitrate solution did not improve boron retention during pellet sintering.

Design and procurement for the pressurized water loop to be installed in the ORR continued as scheduled. Loop design work is practically complete, and procurement action has been initiated for all major components except the in-pile tube. Work was started on the necessary alterations to the ORR building. (AEC Activity 4151)

GENERAL REACTOR RESEARCH

Basic Reactor Research. — The new 300-kv BSF accelerator was used successfully to produce neutrons for the time-dependent neutron diffusion experiment. The first measurements in which the accelerator was used were with beryllium at room temperature. In another experiment 4- and 12-Mev gamma rays were produced by the $B^{11}(p,\gamma)C^{12}$ reaction. During these operations it became apparent that when the accelerator is being used to produce neutrons, several precautionary measures must be taken to reduce the radiation level in the near vicinity. The necessary steps have all been initiated.

A new and stronger BSR-I support structure was installed. The new structure is similar to the Pool Critical Assembly structure, where all the support is attached to one side of the grid. This greatly facilitates

fuel element loading and will also simplify the installation of the calorimeter tank which will be used for power determinations.

Recent work in the experiment to investigate the spectrum of prompt gamma rays from U^{235} fission is being centered on the measurement of the spectrometer efficiency as a function of energy. This work is performed with radioactive sources whose disintegration rates are normally determined with the aid of a high-pressure ionization chamber. An Na^{24} source was calibrated by absolute coincidence-counting techniques with an estimated uncertainty of 1%. The results from separate ion chamber measurements were within 1% of the absolute coincidence-counting results for this single case, which substantiates the ion chamber calibration technique. A steady-state source of 7.4-sec N^{16} was calibrated absolutely within about 5% in a similar manner to allow an efficiency determination with the 6.1-Mev gamma rays from this decay. Considerably more work of this type will be required for a proper evaluation of the spectrometer efficiency.

A modification was made in the control system for the Pool Critical Assembly. When the servo calls for insertion of the regulating rod, the insertion will proceed only if a negative period of -50 sec or shorter has not been established (as seen by log N). The log N level reverse is also delayed about 1.5 sec. If during this time a -50-sec period has been established, the reverse will not occur. However, if the negative period goes away, the reverse will occur after the 1.5-sec delay if the instrument contact requesting it has not returned to normal.

The perturbation of the flux by indium foils in a graphite medium at various distances from a source is being determined in a series of measurements at the Standard Pile. Both bare and cadmium-covered foils of various thicknesses and areas are being used. (AEC Activity 4202)

Fuel Element Development. — An aluminum test element, containing a 37 vol % dispersion of U_3O_8 in aluminum, in which the uranium contained 20% of the U^{235} isotope, was irradiated for four cycles in the active lattice in the MTR. The estimated burnup of U^{235} atoms was 25%. No difficulty with this fuel element was observed during the irradiation period. A similar element was also inserted in the active lattice of the ORR and has now operated successfully for three weeks.

In an investigation of the factors contributing to uranium contamination on the surfaces of aluminum fuel plates, recent analyses of uranium in the aluminum cladding stock reveal that the concentration of this element is in the range 0.05 to 0.2 ppm.

Development of the technology of boron and boron compounds in stainless steel showed that stainless steel powder alloyed with boron is susceptible to boron loss when heat-treated under hydrogen at elevated temperatures. Specimens containing 0.06 wt % boron lost between 25 and 40% of the boron while being sintered for 1 1/4 hr at 1150°C. Similar results on samples sintered under the same conditions in a vacuum revealed no significant losses.

In the development of dispersions of UO_2 in ferritic stainless steels, a literature survey revealed that both type 446 and type 430 steels have sufficient corrosion resistance in water at 500 to 600°F to be considered corrosion resistant. Preliminary fabrication experience indicates that embrittlement of type 446 steel occurs during hot fabrication and that water-quenching is required in order to gain some ductility. Type 430 stainless steel showed good ductility after conventional fabrication at 1900°F and appears to be a better selection for this dispersion-type system. (AEC Activity 4203)

Power Reactor Fuel Reprocessing: Alternate Aqueous Processes. — The M-388 alloy (99% Al–1% Ni) used as cladding for BORAX-4 fuel dissolves at the same rate as pure aluminum in boiling 3 and 4 M NaOH.

Tests of boron and cadmium as criticality control additives to Consolidated Edison dissolver solution were continued. Dissolver solution boiled to 181°C lost only 6% of the added boron; only 3% was lost at 135°C, the normal boildown temperature. Insufficient cadmium was lost in similar boildown tests to permit its determination in the distillate. Preliminary tests indicate that the separation of boron and cadmium from thorium and uranium in solvent extraction is satisfactory. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Darex Process. — Prototype APPR fuel was processed in an 11 1/2-hr engineering-scale Darex run with the use of all the equipment and controls in the continuous loop system. Operation was satisfactory until the stripper was plugged by silica at the end of the run. Analysis of the siliceous material showed appreciable amounts of Si, Fe, Cr, Ni, U, and acid-insoluble materials. It was concluded that a bubble-cap tower is unsatisfactory in the chloride stripping operation for high-silica fuels.

In tests with a 1-in.-dia glass dissolver designed to test semicontinuous APPR dissolution and maintenance of constant liquid level by means of an air lift, three 24-in.-long miniature elements weighing 575 g each were dissolved in 21 liters of aqua regia in 65 min. Other results of these tests demonstrated that (1) bottom acid feed and side-arm product takeoff is the best flow arrangement, (2) liquid-level control by air-lifting of product is impractical in the dissolver because of the violence of the reaction but is feasible in the side arm, (3) dissolution is most pronounced at the base of the assembly, and (4) no advantage accrues from recycling either the dissolver product or the condensate. This study is being continued in 6-in.-dia equipment with full-size APPR prototype fuels.

In batch and semicontinuous tests on the removal of chloride from raw dissolver solution, the residual chloride concentrations were 400 ppm in a 2-liter batch test and 140 and 20 ppm in two semicontinuous experiments. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Hermex Process. — The solubility of thorium in mercury was found to increase from 0.0024 to 0.0264 wt %

as the temperature was increased from 40 to 356°C. The solubilities of thorium and uranium in a solution saturated with both increased from 0.00244 to 0.0118 wt % and from 0.0124 to 0.770 wt %, respectively, as the temperature was increased from 100 to 300°C. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Sulfex Process. — In further study of the Sulfex dissolution procedure for Consolidated Edison fuel, the 1-hr dissolution rate of type 304L stainless steel in 4 or 6 M H_2SO_4 containing 5 g of stainless steel per liter was less than 0.004 $mg \cdot min^{-1} \cdot cm^{-2}$. The initial rate of dissolution of type 304L stainless steel in 4 or 6 M H_2SO_4 containing 1 g of Cu^{++} per liter, as $CuSO_4$, was less than 0.02 $mg \cdot min^{-1} \cdot cm^{-2}$. The copper was present as a corrosion inhibitor. These data verified the low rates previously reported.

In continuing Sulfex corrosion studies, solution-annealed Carpenter-20Cb stainless steel corroded at a maximum rate of 4.6 mils/month over a 164-hr exposure to boiling 6 M H_2SO_4. The maximum rates for stress-relieved and solution-annealed—stress-relieved specimens under the same conditions were 7.5 and 8.8 mils/month, respectively. Contact with type 304 stainless steel dissolving in boiling 6 M H_2SO_4 lowered the corrosion rate of Carpenter-20Cb steel from 2.9 to approximately 0.85 mils/month. The vapor-phase corrosion averaged about 3.6 mils/month in each case. Welded, stabilized Ni-o-nel subjected to five Sulfex-Thorex dissolver cycles showed rates of 1.6 to 2.1 mils/month, with little or no intergranular corrosion. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Zirflex Process. — The high-activity-level, laboratory-scale Zirflex facility was started up on unirradiated ZrO_2-coated PWR sample fuel pins. The uranium loss to the decladding solution was 0.1%.

In a modified Zirflex low-corrosion flowsheet for U-Zr-Nb alloy fuels (EBWR core 1), the zirconium cladding and the underlying alloy are dissolved completely in 6 M NH_4F. The dissolution is followed by metathesis with NH_4OH, by which all the uranium and zirconium and part of the niobium are precipitated. The solids are filtered and the filtrate is recycled. The solids are then redissolved in HNO_3. Complete dissolution of thin EBWR fuel plate was accomplished in 20 hr. The waste solution from the metathesis, containing 0.4% of the uranium, 0.2% of the zirconium, and 6% of the niobium, was evaporated to 6 M in NH_4F and recycled to the next dissolution. Upon dissolution of the metathesis precipitate in HNO_3, 0.1% of the uranium was held up in the 88% of the niobium that was insoluble in HNO_3. Metathesis and filtration reduced the F/metal ratio from 8 to 2, resulting in a final F/Zr ratio of 5.9. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Solvent Extraction Studies. — Laboratory experiments have established a 1.0 M aluminum nitrate—salted 2.5% tributyl phosphate flowsheet for the solvent extraction recovery of enriched uranium from special stainless steel feed solution prepared by the Darex process.

Modifications in the Interim-23 flowsheet at varying acidities have been demonstrated in laboratory countercurrent batch extractions of Consolidated Edison fuel solutions. Use of water as scrub instead of 0.5 M $Al(NO_3)_3$ resulted in the refluxing of uranium to a concentration of 40 g/liter.

A combination of 0.1 M KOH and lime satisfactorily recovered solvent more highly degraded than is anticipated in the present power reactor fuel reprocessing program. (AEC Activity 4301)

Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes by Sintering. — Aluminosilicate sinters were prepared from illite (a natural clay) and from the chemical equivalent of illite in the form of oxides. The purpose of this test was to study the effect of the initial form of the sinter material on cesium retention. The waste used was neutralized Purex (6 M). The sinter formed from the clay was approximately ten times more effective in retaining cesium than the sinter produced from the oxides. The chemical analyses of the soluble constituents showed that alumina was not completely incorporated in the sinter produced from the oxides. Sinters produced from oxide ingredients which were reduced in alumina content were superior to those with higher alumina in retaining cesium.

Sinters were prepared from different clay minerals — illite, montmorillonite, and kaolinite. The results to date show that kaolinite is superior in fixing cesium on the basis of equal ignited weight of the sinters. Tests conducted on methods to determine the leachability of cesium from sinters show that the method recommended by the American Ceramic Society leaches approximately four times more cesium in 4 hr than the commonly used bottle test does in seven days. When tested by the former method, the kaolinite sinter retained 99.98 $\pm$ 0.02% after one 4-hr leach. This degree of retention is for a sinter prepared from 50 ml of 6 M $NaNO_3$ containing 0.1 g of cesium per liter and 87 g of kaolinite. (AEC Activity 4352)

Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes Into Deep Wells. — Safe disposal of high-level radioactive waste solutions into deep wells requires consideration of the resulting temperature rise due to fission product decay. Calculations suggest that conservative temperature limits will not be exceeded if the disposal formation holds no more than 2 to 4 curies/ft^3 of such long-lived nuclides as Cs^{137} and Sr^{90}. The concentration of these nuclides in the disposal formation will depend not only on their concentration in the liquid waste but also on the amounts sorbed by the clay minerals in the disposal formation, particularly near the injection well. Laboratory experiments with sodium-neutralized 6 M Purex-type waste showed that the order of relative sorptive capacity of the major clay minerals for radiocesium is illite > bentonite > montmorillonite > vermiculite > kaolinite > halloysite. Tests of illite slurries, in beakers, showed that at equilibrium about 50 to 60% of the cesium is sorbed. The proportion of cesium sorbed can be reduced by increasing the sodium to

cesium ratio, by lowering the pH, or by the addition of calcium carbonate. The use of competing ions such as sodium is one promising method of controlling and reducing the concentration of cesium by adsorption in the disposal formation. This may be illustrated by one hypothetical case calculated from the laboratory data. If 6 M sodium-neutralized Purex-type waste were diluted 30 to 1 with concentrated brine and injected into a formation having 25% porosity and containing 5% calcium carbonate and 5% illite, then the highest cesium concentration near the well would be 3.7 curies/ft^3, a value within the limits presently regarded as acceptable. Of this amount 1 curie would be in the liquid in the pores and 2.7 curies would be sorbed on the clay. Both these values could be further reduced if desired. (AEC Activity 4352)

Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes Into Salt Formations. — A preliminary design has been completed for an experimental field facility to be constructed in an existing salt mine. This facility will consist of a gas-tight cover over the cavity, an off-gas system, electrical heaters, power controls, and associated instrumentation. With this equipment a number of experiments will be performed to study the interactions of heated nonradioactive waste solutions and rock salt beds in place.

Corrosion studies of metals and materials necessary for field-scale experiments have shown that Hastelloy alloys, graphite, and Teflon are resistant to boiling solutions of acid and neutralized Purex waste saturated with sodium chloride.

Preliminary laboratory experiments, aimed at determining the practiability of simulating the nuclear heat of decay by means of graphite electrodes, are now in progress. These experiments will include quantitative and qualitative analysis of the off-gases released during the heating cycle. (AEC Activity 4352)

Waste Disposal Research and Engineering: Process Waste Water Treatment Plant. — A total of 119 million gallons of process waste was treated during the first 11 months of plant operation. This volume amounted to 50% of the total process waste. Radiostrontium removals ranged from 66 to 87% for monthly periods and were as high as 94% for individual periods of one week. The total rare earths were removed as efficiently as the strontium. A total of 48 curies of Sr^{89} and Sr^{90} and 44 curies of total rare earths were removed. Since September 1958 the addition of 100 ppm of clay (Grundite containing 65 to 75% illite) to the waste has resulted in the removal of 80% of the Cs^{137}. The radionuclides removed were associated with 387,000 gal of sludge containing an average of 52% solids by volume. The expenditure for chemicals, operators' time, and utilities was $21,500. The cost per 1000 gal of waste treated was 18¢. (AEC Activity 4352)

Waste Treatment. — In studies of the nitric acid—formaldehyde reaction for its application to the reduction of liquid waste to solid form, 2.12 moles of HNO_3 were destroyed per mole of HCHO fed.

When neutralized TBP-25 waste was reduced to dryness by heating, 76% of the nitrate ion was recovered in the condenser and off-gas equipment. When formaldehyde was added to the neutralized waste, only 23% of the nitrate ion was recovered. When the vapor was passed through a hot copper bed (400°C) only 27% of the nitrate ion was recovered for recycle.

The distribution coefficient study of waste treatment by ion exchange was extended to SIR, STR, and ORNL wastes. The values of the cesium distribution coefficient for Duolite C-3 and Duolite S-30, respectively, were 131 and 110 in neutralized SIR waste, 30.6 and 11.4 in neutralized STR waste, and 83.1 and 77.7 in neutralized ORNL waste.

A preliminary cost estimate was made for construction of a plant to decontaminate the ORNL intermediate-level waste now being discharged to the seepage pits. This waste varies widely in volume and composition; but this estimate was based on a volume of 7000 gal of waste per day, 0.4 M in Na^+ and 0.3 M in NO_3^-, containing 40 curies of cesium, 10 curies of total rare earths, 2 curies of ruthenium, and 40 mc of strontium. The process for decontaminating this stream is that proposed by I. R. Higgins (ORNL CF-57-10-129) and consists in an ion exchange step for removal of cesium and other cationic activities, followed by an electrolysis step for cathodic plating of ruthenium and reduction of nitrate. Laboratory studies indicate that such treatment would decontaminate the waste by factors of 10^4 for cesium, 10^3 for total rare earths and strontium, and 200 to 500 for ruthenium. Essentially complete reduction of nitrate to ammonia can be obtained.

The preliminary estimate ranged from $500,000 to $750,000 for a complete facility, including the building. (AEC Activity 4352)

PROGRAM 5000 — PHYSICAL RESEARCH

ISOTOPE PRODUCTION

Radioisotope and Stable Isotope Production. — The first production, 3000 curies of Cs^{137}, was processed through the Multicurie Fission Product Pilot Plant.

Calutron separations of the isotopes of calcium and potassium were completed. Ion currents equivalent to 407 g of total calcium and 39 g of total potassium were collected.

Numerous requests for the isotopes of lutetium have depleted the present inventory; therefore a second collection is being made with the remaining $LuCl_3$ charge material. Ion currents equivalent to 0.219 g of total lutetium have been collected thus far in this series, and it is estimated that approximately 1 g of Lu^{176} will be available from this second collection for inventory.

An exploratory attempt to deposit the isotopes of silicon on copper target backing for use in cyclotron studies is in process in a single beta channel. The second beta channel has been equipped with 190° shims and associated equipment for the development of an inhomogeneous field separator.

The charge materials supplied for isotope separation were $K^{40}Cl$ (fused), Ca metal, $SiBr_4$ (liquid), and $LuCl_3$ (fused).

Seven hundred twenty-five grams of Lu_2O_3 was recovered and is presently being converted to $LuCl_3$ to provide charge material for the second collection.

Potassium charge material, enriched in K^{40} to 0.19% by reactor irradiation, is being recovered and purified from graphite parts and calutron wash solutions following termination of an isotope collection.

New lots of Hf^{174}, Hf^{176}, Hf^{177}, Hf^{178}, Hf^{179}, Hf^{180}, V^{51}, Yb^{168}, and Lu^{176} and returned lots of Pb^{204} and Fe^{58} were chemically refined and submitted for mass analysis. New lots of Se^{74}, V^{51}, Yb^{168}, Lu^{176}, Hf^{174}, Hf^{176}, Hf^{177}, Hf^{178}, and Hf^{180} and a returned lot of Fe^{58} were prepared for inventory following refinement and spectrochemical and mass analysis.

Ten preliminary samples of K^{40} were recovered and partially refined to obtain combination and/or "in-run" assay data.

Radioisotope Research and Development. — Improvements were made in the chemistry of the Fission Product Pilot Plant process for separating fission products from Redox waste. Mercury, one of the macro constituents of the waste, is precipitated by reduction to the metallic state in 0.1 M NaH_2PO_2 and 0.3 N HNO_3.

The method of recovering cerium by precipitation as the arsenate was found to be impractical under actual operating conditions because of the coprecipitation of trivalent rare earths as double sulfate salts. A substitute method was developed in which cerium is oxidized to Ce(IV) and precipitated as a basic salt at pH 4.0. Laboratory experiments indicate that a 90% yield can be expected by this method.

After the installation of a column having an increased capacity, all development effort on the purification of Kr^{85} was completed and the process was assigned a routine production status. (AEC Activity 5100)

Stable Isotope Development. — A voltage regulator circuit capable of regulating to within 0.1% any potential desired within the range 25 to 460 v was designed and installed in each of the 0-to-2500-w XBX and XAX charge-oven heater circuits. Since the resistance of graphite heaters changes but slightly over a limited temperature increment, this device is expected to improve oven temperature stability within the calutron system.

A receiver was fabricated, and runs are in progress to test the feasibility of silicon target formation within the calutron. If successful, ions would be deposited directly on the target backing, and vaporization of purified isotopes onto a target would not be necessary.

Conventional shims were removed from XBX tank I to permit the installation of specially shaped 190° shims. The initial exploratory run of a series designed to test such a radially inhomogeneous field separator system is in progress. (AEC Activity 5100)

PHYSICS

High-Voltage Experimental Program: Decay of I^{130}. — The decay of I^{130} has been previously studied by two groups of workers (Roberts, Elliott, Downing, Peacock, and Deutsch, Phys. Rev. 64, 268, 1943, and Caird and Mitchell, Phys. Rev. 94, 412, 1954). As a result, the following level scheme for Xe^{130} has been proposed: 528 (2^+), 1190 (4^+), 1930 (6^+), and 2340 (5^-) kev. Only the states at 1930 and 2340 kev were found to be directly populated by beta decay. The measurement of spin 5 for I^{130} (see section entitled "Atomic Beam Spectroscopy") suggests that the 1190-kev state, if 4^+, should be populated by beta decay. In the present work, such a group was looked for but was not found. A lower limit of 9.8 was set for log ft. Gamma-gamma correlations were also made. The experimental correlation for the 1190-528-0 level sequence can be fitted by 4(Q)2(Q)0 or by 2(96% D + 4% Q)2(Q)0 sequences. The 2^+ assignment to the 1190-kev level requires a predominantly M1 transition for the 2^+ to 2^+ transition, in contrast with the predominantly E2 character of other known transitions of this type. Therefore, either 4^+ or 2^+ assignment to the 1190-kev state leads to an anomalous situation. (AEC Activity 5220)

High-Voltage Experimental Program: Neutron Capture Cross Sections. — The neutron capture tank used with the pulsed Van de Graaff has been operated with a timing resolution of 15 to 20 mμsec. The resolution on individual gamma rays is 25 to 30%. The 29-kev Li(p,n) threshold measurements previously reported have been repeated at a lower bias, and data have been taken at 63 kev. Nineteen samples had large enough cross sections to make a study of the capture cascade sum-spectra in the large liquid scintillator feasible. Conversion of the full binding energy to a light pulse, even with the over-40-cm gamma absorption path in the scintillator, seems to be relatively rare. The spectra are sufficiently characteristic, however, to identify neutron binding energies. Values of 6.5 ± 0.4 Mev for Tb^{159} and 6.2 ± 0.4 Mev for Ho^{165} were found in this way.

Scattered neutrons from a sample are mostly thermalized and captured at times that are long compared with the 1.4-μsec beam-off time. About 97% are captured by hydrogen, giving a 2.2-Mev gamma ray; the rest, absorbed chiefly in the aluminum inner liner of the scintillator, give capture gamma-ray pulses up to 8 Mev. A small proportion (<1%) of the scattered neutrons are found to be captured in hydrogen within 50 mμsec,

thus contributing 2.2-Mev pulses in the time delay window corresponding to neutron capture in the sample at slightly lower energies.

A collimator has been installed, allowing the use of neutrons down to 4 kev, with considerable sacrifice in intensity, and up to several hundred kilovolts. First results, with a flight path of 119 cm, include the observation of capture in the 27- and 50-kev resonances of fluorine and in the 83-kev resonance in magnesium. (AEC Activity 5220)

High-Voltage Experimental Program: The $Al^{27}(n,\alpha)Na^{24}$ Reaction Cross Section. — The cross section for the $Al^{27}(n,\alpha)Na^{24}$ reaction has been measured as a function of neutron energy in the range $6 < E_n \leqq 8.3$ Mev and at 14.8 Mev. Source neutrons were obtained from the $H^2(d,n)He^3$ reaction by using the 5.5-Mv Van de Graaff generator, and from the $H^3(d,n)He^4$ reaction by using the 250-kev Cockcroft-Walton accelerator. Neutrons in a small cone about 0° with respect to the incident charged-particle beam were incident on an aluminum metal sample and a thin U^{238} deposit placed adjacent to the aluminum sample. Fission fragments from the deposit were detected and counted in a gridded ionization chamber in order to determine the number of neutrons incident on the aluminum sample. The number of $Al^{27}(n,\alpha)Na^{24}$ events was determined by counting the residual Na^{24} activity in a calibrated sodium iodide gamma-ray scintillation detector. Although a number of peaks appear in the curve for cross section vs energy, there is a general increase in the cross section with increasing neutron energy from approximately 3.4 millibarns at 6.3 Mev to about 46 millibarns at 8.3 Mev. At a neutron energy of 14.8 Mev the $Al^{27}(n,\alpha)Na^{24}$ cross section was measured to be 117 ± 8 millibarns, in agreement with several previous measurements in this energy range. (AEC Activity 5220)

Atomic Beam Spectroscopy. — The atomic beam magnetic resonance technique has been used to determine the nuclear spin of I^{130}. The isotope was prepared by an (n,γ) reaction on elemental I^{129}, obtained as a fission product. Approximately 100% dissociation of the iodine vapor was obtained by superheating to above 650^oC in a gold oven. There are two observable $\Delta f = 0, \Delta m = \pm 1$ transitions for atomic iodine. Both have been detected repeatedly and at two different magnetic field intensities. In all cases the lines appear at frequencies corresponding to spin 5. This is in disagreement with the spin assignment (6) of Caird and Mitchell (Phys. Rev. 94, 412, 1954). (AEC Activity 5220)

The 86-Inch Cyclotron: Nuclear Physics. — The electron capture of Tm^{168} (87-day) to levels in Er^{168} was investigated with permanent-magnet spectrographs and coincidence scintillation spectrometers. The following levels in Er^{168} were established: 79.9 (2^+), 264.4 (4^+), 549.1 (6^+), 822.0 (2^+), 896.7 (3^+), 996.2 (4^+), 1095.1 (3^- or 4^- and $T_{1/2} = 1.2 \times 10^{-7}$ sec), and 1543.1 (4^-) kev. The internal-conversion data for Tm^{166} (7.7-hr) suggest levels in Er^{166} at 80.7 (2^+), 265.4 (4^+), 545.8 (6^+), 787.4 (2^+), 861.0 (3^+), and 957.9 (4^+) kev, with many more high-lying levels. Energy level schemes have been proposed for both Er^{168} and Er^{166}. The levels at 822, 897, and 996 kev in Er^{168} and at 787, 861, and 958

kev in Er^{166} may possibly be associated with electric quadrupole (gamma) vibrations (Kgl. Danske Videnskab. Selskab Mat. fys. Medd. 27, No. 16, 1953). (AEC Activity 5220)

The 86-Inch Cyclotron: Applied Physics. — The isotopes Co^{57}, Zn^{65}, As^{74}, Sr^{85}, Nb^{94}, and Au^{195} were produced in service irradiations for outside customers. Fluorine-18 was produced for the Chemistry Division. In a 4-hr run ~80 mc of Zn^{65} was produced by 22-Mev protons on a copper target at an average current of 2415 μa; this is a new level for routine operation.

For some target applications the high temperatures required and the chemical contaminants introduced by the usual bonding procedures introduce special problems. Tests show that targets soft-soldered to the target base can be operated at currents of 500 to 800 μa; hard-soldered targets are operated at 1000 to 1500 μa. A platinum sheet soft-soldered to a standard copper base was run for 3 hr at an average current of 780 μa.

A fabrication technique was developed for producing zirconium targets. A sheet of 0.020-in. zirconium is first plated on one side with nickel to a thickness of 0.001 in. This sheet is then heat-treated in an argon atmosphere to produce a diffusion bond between the nickel and the zirconium; finally it is soft-soldered to a standard copper flat plate target. One such target failed at 1100 μa; another, in which the heat treatment step was omitted, failed at 400 μa. In both cases the failure occurred at the nickel-zirconium interface. The successful target was operated at 750 μa for 2 hr. (AEC Activity 5220)

The 86-Inch Cyclotron: Operation. — The central maximum in the axial distribution of the cyclotron beam on an internal target now has a width of 0.9 in.; previous measurements gave a width of 1.7 in. This narrowing, which is probably due to some slight changes in positioning of the parts of the resonant system, is advantageous for deflecting the beam but produces more uneven distribution of the beam on internal targets.

In a check on the calibration of the coil which moves the effective center of the beam away from or toward the target, it was found that the extremes of operation changed the beam energy only 0.5 Mev. On the other hand, actually moving the target from the minimum position of 30 in. to 31.75 in. increases the energy 2.3 Mev. Further calibration tests are to be made. (AEC Activity 5220)

Heavy-Particle Physics. — The beam of 28-Mev nitrogen ions was used to measure the density of the carbon arc which dissociates molecular hydrogen ions in the DCX experiment. The measurement was performed by scattering nitrogen ions from carbon nuclei in the arc and detecting both particles in coincidence with thin CsI scintillation counters. Due to the paucity of counts, only an approximate value of the arc density was obtained. It is estimated that the density is greater than 5×10^{13} but less than 2×10^{14} atoms/cm^3.

Elastic scattering of nitrogen from beryllium was extended to 132° in the center-of-mass system. At backward angles the distribution levels off and has a small peak around 115°.

A search is being made for the double transfer reaction $Al^{27} + N^{14} \longrightarrow Mg^{27} + O^{14}$. So far, indications are that this reaction proceeds with a very small cross section, if at all.

A selected DuMont 6365 six-stage photomultiplier tube was found to give a resolution of 2.6% full width at half-maximum with 20-Mev protons on a NaI(Tl) crystal. This tube will be used in a particle-selecting proportional counter-scintillation counter telescope whose total length is 6 in. (AEC Activity 5220)

Spectroscopy Research Laboratory. — Recently, Lindner and Davis (J. Opt. Soc. Am. 48, 542, 1958) published Zeeman effect data for the spectra of erbium. Their attempts at classification of the spectra using the MIT digital computer proved unsuccessful. With their Zeeman data and the wave-length data of Gatterer and Junkes (Spektran der Seltenen Erden, Citta del Vaticano, 1945), eight low Er II levels and more than forty high levels have been established. The eight low levels are tabulated below:

Term	Level	J	g	g_{LS}
^4H	0.0	13/2	1.225	1.231
^2H	440.4	11/2	1.092	1.091
^4F	5132.7	9/2	1.234	1.333
^2F	5403.9	7/2	1.038	1.143
^4H	7149.7	11/2	1.110	1.133
^2H	7195.4	9/2	0.942	0.909
^4H	10894.1	7/2	0.837	0.667
^4H	11042.8	9/2	1.045	0.970

The low electron configuration is $4f^{12}6s(^{4,2}H)$, for which all six levels are believed to be identified. Two levels, probably from the configuration $4f^{12}6s(^{4,2}F)$, have been located about 0.6 ev above the lowest level. Splitting by the 6s electron in the $4f^{12}6s$ configuration is small and comparable to that for Tm (238 and 188 cm^{-1} for $4f^{13}6s$). This small s doubling of the $f^n s$ configurations is a characteristic of the singly

ionized spectra of the rare earths. The level splitting in the $4f^{12}(^3H_6 - ^3H_4)$ parent is of the order of 11,000 cm^{-1}, which may be compared to the $4f^{13}6s^2(^2F)$ splitting of 10,150 cm^{-1} in Yb I. Except for the inversion of terms it is expected that Er II will be quite analogous to levels built on the f^2 parent term in Ce II. (AEC Activity 5230)

Special Separations. — The U^{238} single-arc separation is virtually complete, with an estimated collection of 37.0 g from the 16 runs made during this period. Total estimated collection thus far in this series is 76.0 g.

The first U^{233} charge is practically exhausted, with five runs completed and an estimated collection of 7.0 g of U^{233}. Of this collection, 3.6 g has been purified and assayed, with an impurity of less than 10 ppm of other isotopes.

Calutron tank 611 is being converted into a facility suitable for the separation of U^{233} and Pu. The fabrication and assembly of the double-arc source units for U^{238} separations are approximately 80% complete. Corrosion maintenance of Pu recycle facilities is becoming more frequent due to the age of the equipment.

Preparation of U^{238} for civilian application sales was completed, and the uranium was delivered to Y-12 for metal fabrication.

Twenty-eight shipments consisting of uranium, thorium, and the transuranics were made from the research pool during this period. (AEC Activity 5250)

Mass Spectrometry and Related Techniques. — The negative-ion mass spectra of the following alcohols were investigated: methyl, ethyl, n-propyl, isopropyl, n-butyl, isobutyl, sec-butyl, and tert-butyl. In each of the alcohols, negative ions were formed by loss of a hydrogen from the hydroxyl group. Relative abundances of the dissociation product ions such as CH_3^- and CH_2^- could be qualitatively correlated with the structure of the parent molecule. In all four of the butyl alcohols, ions C_2^-, C_3^-, and C_4^- were observed.

Initial tests of the full two-stage mass spectrometer have shown that the instrument is easily capable of separating and measuring two adjacent nuclides whose abundance ratio is greater than one million. (AEC Activity 5250)

Mass-233 Studies. — Approximately 19.5 g of U^{233} of high isotopic purity was isolated from a Thorex Pilot Plant product. This material was recovered in the Pilot Plant by processing raffinate from short-decayed runs after allowing the Pa^{233} to decay about one year. Mass analysis of the product was 99.62% U^{233}, 0.04% U^{234}, and 0.34% U^{238}. The purified

material contained 0.3 g of impurities, primarily Al, Fe, and Sn. (AEC Activity 5310)

Raw Materials Research and Development. — Essentially complete recovery of thorium from a monazite sulfate liquor was obtained in a continuous run using Primene JM-T as extractant. The uranium had previously been removed from the liquor with a tertiary amine. The thorium product recovered from the nitrate strip solution contained <20 ppm uranium and < 1% total rare earths. Thorium was stripped effectively from di(2-ethylhexyl) phosphoric acid extracts with 4.5 M sulfuric acid. Most of the thorium precipitated in the stripping system, the precipitate containing 1.5 moles of sulfate per mole of thorium. The precipitate had excellent settling and filtration characteristics and did not cause any emulsion formation.

The extractability of Pu(IV) from nitric acid solution by amines in hydrocarbon solution was at a maximum at 8 M HNO_3 with primary, secondary, and quaternary amines, but was at a maximum near 2 M HNO_3 with tertiary amine. The curves for 0.1 M triisooctylamine and the dimethyldidodecenylammonium salt crossed at about 4 M HNO_3, with the extraction coefficient $E_a^o(Pu) \cong 50$. The extraction coefficients with primary and secondary amines were an order of magnitude lower (ORNL-2578).

Coefficients for the extraction of Fe(III) from acid perchlorate solutions by means of di(2-ethylhexyl)phosphoric acid in octane vary inversely with the third power of acidity, as expected for extraction by cation exchange, and directly with about the 2.5 power of dialkylphosphoric acid (HX) concentration. This may be due to the simultaneous formation of $Fe(OH)X_4H_2$ and FeX_6H_3 as the extracted species at low iron loadings. However, at higher loadings it is clear that additional species must be formed, since the saturation ratio of Fe(III):X is $\geqq 0.63$. (AEC Activity 5310)

Neptunium Recovery. — The remainder of the Np^{237} from Metal Recovery processing of fluorination ash has been processed by an oxidized fluoride precipitation to remove Th, Pu, and Cr, followed by a reduced fluoride precipitation to precipitate NpF_4 from Al, Fe, Ni, and U. Contaminants in the purified product averaged 0.1% Pu, 0.1% U, 0.1% Th, and <100 ppm of other metal cations. Approximately 25 g of Np^{237} in recycle solutions remains to be isolated. (AEC Activity 5310)

Extraction Reagent Performance: Radiation Damage. — Diethyl carbonate (DEC) has previously been reported to yield, primarily, gaseous radiolysis products which would not affect solvent extraction processes. Further tests have shown G_{-M} (DEC destruction) = 5.5 molecules per 100 ev of energy absorbed by the solution and G_{-P} (DEC converted to polymer) = 2.8 at a dose density of 392 whr/liter. The nature of the polymer has not been determined. Neither uranium retention nor emulsification tendencies are appreciably altered as a result of radiation damage of DEC. The yield of acid, G_{acid}, from the irradiation of dibutyl phenylphosphonate (DBPP) at

400 whr/liter is 0.78, contrasted with a value of 0.36 previously reported for irradiation at 1500 whr/liter.

When DBPP-Amsco solutions are contacted with an aqueous uranyl nitrate solution, a third phase may form, depending on the various concentrations. Both Decalin and Tetralin, in concentrations of 5 to 15% in the organic phase, eliminate third-phase formation. Tetralin is the better addend, since solutions containing it recover more quickly from the emulsified state. DBPP is of interest because it is several times more resistant to radiolytic decomposition than TBP. (AEC Activity 5310)

Feed Materials Processing. — In the reduction of UF_6 to uranium by lithium amalgam, the product of the reduction step, uranium quasi amalgam, was concentrated with respect to uranium by filtration, and the by-products LiF and UF_4 were washed from the concentrate with water. The resulting aqueous slurry was adjusted to pH 8, and the solids were separated by settling and centrifugation with losses of <0.25% of the U and <0.001% of the Hg. More than 99.9% of the mercury was removed from the residual solids by volatilization at 360°C. Fluorination separated >99.9% of the uranium from the lithium fluoride as UF_6.

Three metal buttons, weighing approximately 10 g each, have been produced by the lithium amalgam reduction process.

One hundred ninety-two pounds of sized UF_4 feed material was prepared for an extended operation of the Fluorox fluidized bed reactor. (AEC Activity 5310)

Ion Exchange Technology. — The average chloride capacity of Dowex 21K determined from five resin samples was 4.56 ± 0.19 meq per gram of dry chloride resin. Each sample was converted to the chloride form and then eluted with a sodium nitrate solution which was subsequently analyzed for its total chloride content. (AEC Activity 5310)

High-Temperature and Structural Chemistry. — Recent measurements, by others, of the decrease in the vapor pressure and melting point of bismuth trichloride on addition of bismuth metal were analyzed here with the assumption of the presence of dimers, both of bismuth monochloride, Bi_2Cl_2, and of bismuth metal, Bi_2. The equilibrium constant $K = N^2_{BiCl_3} \times N^2_{Bi_2}/N^3_{Bi_2Cl_2}$ was found to be 2.0 ± 0.2 at 300°C and 5.0 ± 0.5 at 356°C. In these results the expected tendency of the oxidation state Bi(I), in the form of Bi_2Cl_2, toward increased disproportionation with increasing temperature into the two oxidation states Bi(III) and Bi^o, the latter in the form of Bi_2 molecules, is reflected. This mechanism satisfactorily explains the $BiCl_3$-Bi phase diagram. (AEC Activity 5330)

Organic Chemistry. — For the photochemical oxidation of formic acid by chlorine,

$$HCOOH + Cl_2 \longrightarrow 2HCl + CO_2$$

the rate-controlling step is abstraction of hydrogen from carbon by chlorine atoms,

$$Cl\cdot + HCOOH \longrightarrow HCl + \cdot COOH$$

and the C^{13} rate isotope effect has the value $k_{12}/k_{13} = 1.006$ at 20°C. From the C^{13} effect the deuterium isotope effect with formic-d acid, DCOOH, is calculated to have a value of k_H/k_D = ca.2.

This deuterium isotope effect has now been measured by two independent methods:

Method A. A value of $k_H/k_D = 2.5$ was found by using a 50% DCOOH–50% $HC^{14}OOH$ mixture. The $C^{14}O_2$ specific activity at 5 to 10% reaction, S_5, was compared with the $C^{14}O_2$ specific activity at 100% reaction, S_{100}. If R is the exact molar ratio of HCOOH to DCOOH in the acid mixture used, it is easily shown that

$$\frac{k_D}{k_H} = (R + 1)\frac{S_{100}}{S_5} - R \quad .$$

Method B. The individual rates for HCOOH chlorination and DCOOH chlorination were measured separately in a mass spectrometer. The computed ratio at approximately 70°C was k_H/k_D = ca. 1.7. For measurement of the individual rates, reactants were mixed in the expansion bulb of a spectrometer, and light from a projector was focused on a point just before the leak leading into the ionization chamber. The per cent reaction after a steady flow rate was reached was estimated from the relative heights of the carbon dioxide (product) and formic acid (reactant) peaks. Identical light intensity, pressures, and other conditions were used for the two reactions compared. Since the per cent reaction was kept small in each run, the rate constant was proportional to the per cent reaction achieved at the steady flow rate.

Method B was also applied to the determination of the deuterium isotope effect with formic acid-d, HCOOD, in the photochemical oxidation of formic acid by chlorine. Initial results, which have not yet been verified, indicate a reverse effect: k_D/k_H = ca. 1.3. (AEC Activity 5330).

Radiation Chemistry. — Solutions containing both an oxidizing and a reducing agent provide useful scavenger systems for studying the intermediatesin the radiation decomposition of water. Two such systems were studied, and mechanisms consistent with observed results and with known radical yields in the radiolysis of water were derived. In a solution containing Fe(III) and Sn(II), iron is reduced with an initial yield $G = 6.62$, which is equivalent to $G_H + G_{OH}$, tin is oxidized with a yield $3.75 = G_{H_2O_2} + G_{OH}$, and hydrogen is produced with a yield $0.45 = G_{H_2}$. Initially, Sn(II) is oxidized by OH radical to Sn(III), Fe(III) is reduced to Fe(II)by Sn(III) and by H radical, and Fe(II), or Sn(II), is

oxidized by H_2O_2. The dependence of yields on relative concentrations indicates competition of Fe(II) and Sn(II) for OH radical, and of Fe(II) and Fe(III) for H radical.

Hydrogen enhances the yield for radiation-induced reduction of Ce(IV). The yield, however, increases with increasing concentration ratios $[H_2]/[Ce(III)]$ and $[H_2]/[H_2SO_4]$. This indicates competition of these three solutes for reaction with OH radical; the species produced by reaction of OH with H_2SO_4 (presumably HSO_4) reacts readily with Ce(III) but not with H_2. The relative reactivities of Ce(III), H_2, and H_2SO_4 with OH radicals are 650:135:1. (AEC Activity 5330)

Molten-Salt Thermodynamics. — A statistical mechanical treatment of a lattice model for the reciprocal molten-salt system A^+, B^+, C^-, D^- was carried out to determine the effect of nearest-neighbor interactions. For dilute solutions of A^+ in B^+D^-, the activity coefficient γ_{AD} was found to be related to N_{C^-}, the ion fraction of C^-, by

$$\gamma_{AD} = \frac{(1 - X)^Z \left[(1 + \frac{X}{\beta(1 - X)}\right]^{Z - 1}}{1 - N_{C^-}} ,$$

where Z is the lattice coordination number. The term X is the fraction of anion lattice positions adjacent to A^+ ions that are occupied by C^- ions; this is obtained from a distribution function based on an anion lattice consisting only of adjacent and nonadjacent positions which differ in energy by ΔE due to the formation of A^+C^- ion pairs ($\beta = e^{-\Delta E/RT}$).

The term $(1 - X)^Z$ is the fraction of A^+ ions which are "free" or "uncomplexed" by C^- ions and should, according to complex ion terminology, be equal to γ_{AD}. As seen from the equation, the value of γ based on the complex-ion concept is too small if β is small (weak complex) or if N_{C^-}, the concentration of complexing anions, is appreciable. (AEC Activity 5330)

Analytical Chemistry Research. — Further investigations of the applicability of tri-n-octylphosphine oxide (TOPO) as an analytical extractant for Ce(IV) have been carried out in nitrate systems. During the extraction, Ce(IV) is partially reduced to Ce(III); however, as much as 68 mg of Ce(IV) can be extracted from 1 to 7 M HNO_3 solutions in 1 min with 1 mmole of TOPO in cyclohexane. More than 95% of the Ce(IV) is removed. The percentage extracted is not affected by the ratio of aqueous to organic phase over the range 1 to 20, by increasing the concentration of HNO_3 up to 8 M, or by increasing the period of extraction to 20 min. Sulfate impairs the extraction from nitrate solutions. The species extracted, which exhibits a TOPO-to-Ce mole ratio of 2, absorbs light in the region of 280 to 290 mμ.

A spectrophotometric method has been developed for the determination of iron which is relatively insensitive to interference by the common anions which form masking complexes with iron. Iron in the trivalent state is separated and concentrated by extracting it from 6 M HCl with tri-n-octylphosphine oxide in cyclohexane, after which the iron is determined spectrophotometrically by formation of an Fe(II)—o-phenanthroline complex in the organic phase and measurement of the absorbancy. As a result of the concentration effected by the use of a high aqueous-to-organic phase ratio, the sensitivity can be increased at least twentyfold. This method can be applied to fluoride or phosphate solutions as high as 0.5 M. The coefficient of variation is less than 5%.

METALLURGY

Fundamental Physicometallurgical Research. — The heat capacities of high-purity titanium and hafnium were measured over the temperature range 1.2 to 4.5°K. The experimental specific heat for each of the metals can be expressed in the form $C_p = \gamma T + 19.437 \times 10^5 (T/\theta_D)^3$, where C_p is in millijoules per mole per degree and the constants γ and θ_D for titanium are 3.254 ± 0.015 and 385.8 ± 4.4 and for hafnium are 2.098 ± 0.010 and 246.0 ± 0.5, respectively. In each case the Debye temperature was found to be constant over this temperature range.

The Debye temperatures for titanium and zirconium are inversely proportional to the square root of the atomic mass, as would be expected from the central force model if the atomic forces are unchanged for the two metals. The Debye temperature of hafnium is higher than would be predicted on this basis, in agreement with the stronger atomic forces in hafnium as compared with the other two group IVA elements. The electronic specific heats decrease with increasing atomic number, indicating either a progressively larger degree of electronic interaction in the sequence Ti, Zr, Hf, or a variation of the Brillouin zone structure due to the different potentials of these elements.

As part of the investigations to determine the factors which affect the relative solubilities of elements in the two allotropic forms of group IVA elements, the vapor pressure of cadmium was measured for four beta-phase cadmium-zirconium alloys in the composition range 2.2 to 10.5% Cd and the temperature range 1175 to 1325°K. Within these limits the integral free energy of mixing is given within experimental accuracy by the formula $R(-2780X) + RT[X \ln X + (1 - X) \ln (1 - X) + 2.22X]$, where X is the mole fraction of cadmium in the alloy. Within the above range, the heat and entropy of formation are not temperature-dependent, so that a specific heat of mixing is not observed in the beta phase, and the Koop-Neumann rule holds. The entropy, on the other hand, is less than in an ideal solution by the amount 2.22XR, which indicates either deviations from perfectly random mixing or alloying contributions to the thermal entropy at lower temperatures.

Observations at 4.2°K have shown that both the face-centered cubic phase and the hexagonal close-packed phase of cerium transform to the "collapsed" face-centered cubic structure. The temperature of the beginning of the hcp → fcc' transformation upon cooling is between 77 and 40°K. Although the x-ray diffraction pattern at 4.2°K shows only the "collapsed" cubic phase to be present, upon heating both the normal face-centered cubic and the hexagonal phases reappear in approximately the same proportion in which they were present before the cooling to 4.2°K. This "memory" may be due either to a characteristic distribution of transformation stresses or to stacking faults. (AEC Activity 5420)

Microstresses in Metal Crystals. — A model for the short-range atomic arrangement in the alloy copper—16 at. % aluminum was devised. It is assumed that each aluminum atom has only copper nearest neighbors and that the second and third coordination shells about an aluminum origin are aluminum-rich. Essentially, the assumed atomic configuration is that of a distorted tetrahedron of aluminum atoms formed in the second and third coordination shells, with a copper nearest neighbor in the center of the tetrahedron. The crystal is assumed to be composed of such tetrahedra, or small groups of them, randomly oriented relative to each other and embedded in a pure copper matrix. Short-range-order parameters computed from this model are in good agreement with those obtained from a Fourier inversion of the diffuse scattering distribution after correction for "size effects." An attempt to fit the size-effect modulations of the diffuse scattering by assuming that aluminum atoms are displaced from their ideal sites in such a way that the tetrahedra become more regular appears promising.

Specimens of Cu_3Au that were prepared for a radiation damage study became polycrystalline after long-time homogenization anneals. This is evidently a result of a small amount of residual distortion in the specimens after they were cut from the ingot. To avoid this, the ingot will be recast and homogenized before it is cut. (AEC Activity 5420)

Ceramics Research. — Fuel elements of the Si-SiC type containing up to 10 wt % UO_2 were prepared for irradiation in the ORR. Attempts to add up to 20 wt % of UO_2 to the Si-SiC fuel element have not as yet been entirely successful.

A method of spheroidizing GC-grade (Brush Beryllium Company) beryllium oxide powder was developed. Selected powder is pressed in metal dies, crushed on a 16-mesh screen, and then rolled in a ball mill jar. Whether or not the material will spheroidize is largely dependent on the process history of the material and the speed and fall of the particles within the jar. The immediate application of such spheres (0.6 to 1.6 mm) is as die fill for fabrication by cold-pressing. Sintered spheres will be used as control samples for some physical tests.

Thoria powder has been treated in the same manner with some degree of success. Spheres up to 3.7 mm in diameter have been rolled and heated to 1500°C. Their approximate bulk density was 5.7 to 6.2 g/cc, the average value for 12 specimens being 5.9 (theoretical, 10.03). (AEC Activity 5420)

Fundamental Investigations of Radiation Damage in Solids. — The availability of a Cockcroft-Walton accelerator in the Biology Division has permitted irradiation-effect studies on germanium, utilizing the 14-Mev neutrons produced in a platinum-zirconium tritide target by deuterons. The actual total dosage at about 3 in. from the wall of the target tube is approximately 3×10^{11} neutrons/cm^2 for an irradiation time of 6 hr, which is also the expected lifetime of one target. High-purity single-crystal specimens of n- and p-type germanium have been irradiated with 14-Mev neutrons for the measurement of minority-carrier lifetime and for Hall coefficient and conductivity studies. The removal rate of conduction electrons in n-type germanium has been determined as about 8 per incident 14-Mev neutron. Samples of n-type germanium have been converted to p-type by successive irradiations. The addition rate of holes in p-type material has been determined as about 1 per incident 14-Mev neutron. Additional irradiations on material with a range of initial carrier concentrations and completion of the lifetime measurements and annealing studies should permit an understanding of the defect structure introduced by high-energy neutrons. (AEC Activity 5420)

PROGRAM 6000 — BIOLOGY AND MEDICINE

BIOPHYSICS

Ecological Studies. — In areas contaminated by waste disposal operations, herbivorous insects may accumulate fission products from their plant food. Studies have been conducted on the levels of fission product concentrations reached by insects in relation to the concentrations in their food. In the laboratory, grasshoppers (*Romalea microptera*) were fed bean plants inoculated with Cs^{137}. One cage of 16 grasshoppers received 0.125 μc per grasshopper on every other day, a second cage received this dose on every fourth day, and a third cage on every eighth day. Under these conditions equilibration with food occurred in less than three weeks. The biological half life was approximately four days. The per gram concentrations of Cs^{137} in grasshoppers in the first, second, and third cages were approximately 1/15, 1/30, and 1/50 of the per gram concentration in the food plants. (AEC Activity 6480)

PROGRAM 8000 — WORK FOR OTHERS

Professional Health Physics Training Program. — In September, 35 AEC Fellows, two Air Force officers, and two men from the Newport News Shipbuilding and Dry Dock Company were enrolled in the 1958—59 Radiological Physics Fellowship Program at Vanderbilt University. (AEC Activity 8780)

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August 1958	ORNL-2595
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Approved by Alvin M. Weinberg, Director

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56.	National Lead Company	C. K. McArthur
57.	Nuclear Metals, Inc.	A. R. Kaufmann
58-60.	Nuclear Products - Erco	E. A. Wick
61.	Union Carbide Corporation	R. W. McNamee
62.	Union Carbide Nuclear Company	C. E. Center
63.	Union Carbide Nuclear Company	J. L. Gabbard
64.	Union Carbide Nuclear Company	D. M. Lang
65.	Union Carbide Nuclear Company	J. P. Murray
66-67.	University of California Radiation Laboratory	D. Cooksey

Internal

68. E. E. Anderson
69. E. A. Bagley
70. P. S. Baker
71. C. R. Baldock
72. L. H. Barker
73. S. E. Beall
74. P. R. Bell
75. D. S. Billington
76. R. E. Blanco
77. F. F. Blankenship
78. E. P. Blizard
79. A. L. Boch
80. E. G. Bohlmann
81. C. J. Borkowski
82. G. E. Boyd
83. M. A. Bredig
84. J. C. Bresee
85. R. B. Briggs
86. K. B. Brown
87. F. R. Bruce
88. T. H. J. Burnett
89. A. D. Callihan

Distribution (Continued)

90. A. E. Cameron
91. R. A. Charpie
92. C. E. Clifford
93. T. E. Cole
94. E. L. Compere
95. J. A. Cox
96. F. L. Culler
97. J. E. Cunningham
98. J. S. Drury
99. D. E. Ferguson
100. J. L. Fowler
101. J. H. Frye
102. W. F. Gauster
103. H. E. Goeller
104. J. C. Griess
105. W. R. Grimes
106. C. S. Harrill
107. J. C. Hart
108. M. R. Hill
109. C. J. Hochanadel
110. A. Hollaender
111. A. S. Householder
112. H. H. Hubbell
113. G. S. Hurst
114. G. H. Jenks
115. R. W. Johnson
116. W. H. Jordan
117. P. R. Kasten
118. G. W. Keilholtz
119. C. P. Keim
120. M. T. Kelley
121. E. M. King
122. K. A. Kraus
123. Laboratory Shift Supervisor (D. Phillips)
124. J. A. Lane
125. T. A. Lincoln
126. S. C. Lind
127. R. B. Lindauer
128. R. S. Livingston
129. R. N. Lyon
130. H. G. MacPherson
131. F. C. Maienschein
132. W. D. Manly
133. L. A. Mann
134. H. F. McDuffie
135. J. R. McNally
136. E. C. Miller
137. K. Z. Morgan
138. E. J. Murphy
139. C. E. Normand
140. G. W. Parker
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143. L. P. Riordan
144. A. F. Rupp
145. H. C. Savage
146. H. W. Savage
147. A. W. Savolainen
148. H. E. Seagren
149. E. D. Shipley
150. A. Simon
151. O. Sisman
152. J. R. Sites
153. M. J. Skinner
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155. R. W. Stoughton
156. H. Stringfield
157. E. G. Struxness
158. W. H Sullivan
159. C. D. Susano
160. J. A. Swartout
161. E. H. Taylor
162. B. S. Weaver
163. A. M. Weinberg
164. J. C. White
165. C. E. Winters
166. H. P. Yockey
167. W. Zobel
168-169. Central Research Library
170. Health Physics Library
171. ORNL - Y-12 Technical Library, Document Reference Section
172. R.E.E.D. Library
173-360. Laboratory Records Department
361. Laboratory Records, ORNL R.C.