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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

AUGUST 1958

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Date Issued: SEP 8 1958

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

TABLE OF CONTENTS

	Page
PROGRAM 2000 — SPECIAL NUCLEAR MATERIALS	4
Metal Recovery	4
In-Line Instrumentation	4
Dissolver Solution	5
PROGRAM 4000 — REACTOR DEVELOPMENT	5
HOMOGENEOUS REACTOR PROJECT	5
Homogeneous Reactor Test	5
HRT Chemical Pilot Plant	6
Thorium Oxide Slurry Development: Oxide Preparation	6
Thorium Oxide Slurry Development: Irradiated Oxide Studies	6
HRP Chemistry: Fuel Studies	7
HRP Analytical Chemistry	7
Slurry Testing and Evaluation	8
Slurry Blanket System Development	8
Engineering Research	9
Core and Blanket Vessel Development	9
Centrifugal Circulating Equipment	10
Miscellaneous Equipment	10
Heat Removal Equipment	10
Gas Handling Equipment	10
Fuel Systems Development	11
Feed Pump Development	11
MOLTEN-SALT REACTOR (MSR) PROGRAM	11
Molten-Salt Reactor: Design	11
Molten-Salt Reactor: Experimental Engineering	12
Molten-Salt Reactor: Metallurgy	12
Molten-Salt Reactor: Chemistry	12
Molten-Salt Reactor: Heat Transfer and Physical Properties	13
Molten-Salt Reactor: Radiation Damage	13
Molten-Salt Reactor: Fuel Processing	14
CIVILIAN-POWER ADVANCED GAS-COOLED REACTOR	14
Gas-Cooled Reactor: Design	14
Gas-Cooled Reactor: Experimental Engineering	14
Gas-Cooled Reactor: Instrumentation and Controls	14
Gas-Cooled Reactor: Metallurgy	15
MARITIME SHIP REACTOR PROGRAM	16
Maritime Ship Reactor	16
PACKAGE REACTOR DEVELOPMENT	16
Army Package Power Reactor (APPR-1)	16
GENERAL REACTOR RESEARCH	17
Basic Reactor Research	17
Fuel Element Development	18
Power Reactor Fuel Reprocessing: Alternate Aqueous Processes	18
Power Reactor Fuel Reprocessing: Hermex Process	19

TABLE OF CONTENTS (Continued)

PROGRAM 4000 (Continued) — Page

Power Reactor Fuel Reprocessing: Darex Process 19
Power Reactor Fuel Reprocessing: Equipment Development 20
Power Reactor Fuel Reprocessing: Solvent Extraction Studies . 20
Power Reactor Fuel Reprocessing: Zircex Process 20
Fused Salt—Fluoride Volatility Processing 21
Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes by Sintering 21
Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes into Deep Wells 22
Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes into Salt Formations 22
Waste Treatment: Sulfex Waste 23
Waste Treatment: Ion Exchange Methods 23
Waste Treatment: Darex Waste 23

PROGRAM 5000 — PHYSICAL RESEARCH 24
ISOTOPE PRODUCTION 24
Radioisotope and Stable Isotope Production 24
Radioisotope Research and Development 24
Stable Isotope Development 25
PHYSICS 25
High-Voltage Experimental Program 25
The 86-Inch Cyclotron: Nuclear Physics 26
The 86-Inch Cyclotron: Applied Physics 26
Spectroscopy Research Laboratory 26
Electronuclear Machines 27
The 48-Inch Cyclotron 28
Special Separations 28
CHEMISTRY 28
Raw Materials Research and Development: Dapex Process 28
Raw Materials Research and Development: Amine Extraction . . . 29
Raw Materials Research and Development: Rare Earths 29
Chemical Engineering Research 30
High-Temperature and Structural Chemistry 30
Molten-Salt Thermodynamics 30
Analytical Chemistry Research 30
METALLURGY 31
Fundamental Physicometallurgical Research 31
Microstresses in Metal Crystals 32
Ceramics Research 32

PROGRAM 6000 — BIOLOGY AND MEDICINE 32
BIOPHYSICS 32
Radiation Dosimetry Programs 32
Waste Disposal Research: Soil Disposal of Intermediate-Level Radioactive Wastes 33
Ecological Studies 33

PROGRAM 8000 — WORK FOR OTHERS 34
Nuclear Technology Education for University Faculty Members . . 34
Professional Health Physics Training Program 34

MISCELLANEOUS REPORTS 35

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

August 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. — Two neptunium product recycle runs were completed to remove the major contaminant, thorium, from product solutions. A total of 640 g was processed with 8.0 M nitric acid as the salting agent in the feed and 5% tributyl phosphate as the extractant. Neptunium losses for the runs averaged 0.15%. The resulting product solution was transferred to the Chemical Development Section for purification. (AEC Activity 2351)

In-Line Instrumentation. — Fabrication is 60% completed for the first of two in-line gamma-activity monitors which will be used in the ORNL power reactor fuel reprocessing program, and a second monitor is 30% completed. The first monitor is similar to the in-line gamma-activity monitors which are in use in various process streams at the Hanford separations plants. This monitor will be used in the pilot plant's first-cycle organic product stream (the HAP stream) and will measure the gross gamma activity, the zirconium-niobium gamma activity, and the ruthenium gamma activity of this stream in counts per minute per milliliter. The monitor consists of (1) a 12-in. section of fluorothene tube with a 1/2-in. bore which is very highly polished and which is filled with the HAP stream, (2) a gamma scintillation detector head, (3) a DD-2 linear amplifier, (4) a logarithmic count-rate meter, (5) a Brown Electronik recorder, (6) a high-voltage supply, (7) a Co^{60} standard, and (8) a manually operated rinsing system for the fluorothene tube. This monitor differs from the Hanford models in that a gamma source, instead of an oscilloscope, is used for standardizing the monitor, and a manual control, instead of an automatic programmer, is used for rinsing the fluorothene tube and for measuring the tube background.

In the second monitor a freely falling stream is employed in order to avoid background radiation effects in the fluorothene tube. This monitor will be used in the metal recovery plant's second-cycle organic product stream (the IAP stream). Aside from a different hydraulic arrangement, this monitor has the same components as the previously listed monitor, and it makes the same measurements. An air-operated valve interrupts the falling stream while the background and standardization measurements are made. (AEC Activity 2724)

Dissolver Solution. — Studies have continued on the use of triisooctylamine in xylene as an extractant for separating uranium from various dissolver solutions prior to its determination by controlled-potential coulometry. Experiments have established that the optimum concentration of amine in xylene is 5% and that the most satisfactory stripping solution is 1 N H_2SO_4, 0.05 M $HClO_4$. At least 12 mg of uranium can be salted into 5 ml of this extractant from 5 ml of dibasic aluminum nitrate. The uranium can then be removed by two 5-ml portions of stripping agent, these being drained directly into the coulometric cell for titration. The separation procedure requires 20 to 30 min and introduces no difficulty from organic material. Studies of the precision of the entire procedure are under way. Experiments to establish the decontamination efficiency of the separation procedure are planned. (AEC Activity 2724)

PROGRAM 4000 — REACTOR DEVELOPMENT

HOMOGENEOUS REACTOR PROJECT

Homogeneous Reactor Test. — The HRT operated continuously during August in run 17 at power levels as high as 5 Mw. Nearly 2000 Mwhr was accumulated, mostly at 3.5 Mw. Although the reactor performance was completely free of mechanical difficulties, the run was distinguished by occasional short bursts of power and by an apparent discrepancy in the two methods of determining uranium inventory.

The power bursts, sometimes negative and sometimes positive, were studied throughout the run, but they have not been fully explained. The bursts vary in magnitude to 2 Mw and in duration to several seconds. They are accompanied by periods as short as 5 sec, and sometimes result in a temperature rise of 1 or 2°C, indicating a sudden reactivity addition of ~0.5% of k. Presumably they are a consequence of fuel nonhomogeneities, but this remains to be proved.

Estimates of the uranium inventory of the reactor are based on the critical temperature and on the chemical analyses. The chemical inventory has

consistently indicated a decrease in inventory with increasing reactor power. At the 5-Mw level the "loss" of U^{235} is 1-1/2 to 2 kg; recovery is complete within a few hours after power removal is stopped. However, the inventory based on the critical temperature indicates that the reactivity of the core is essentially undiminished. This inconsistency can be explained by the collection of uranium in the core as the power increases, but at present there is incomplete evidence to support this argument. (AEC Activity 4103.3)

HRT Chemical Pilot Plant. — Analysis has been completed of the data obtained during two runs made in conjunction with HRT run 16. The rates of removal of solid corrosion products were essentially the same for both operating periods. In the first run, which lasted 290 hr and in which the reactor produced 249 Mwhr of heat, 100 g of solids was removed. During the last 340 hr, in which the reactor produced 1230 Mwhr, 107 g of solids was removed. The fission products that concentrated in the underflow receiver included ruthenium, zirconium, tellurium, and molybdenum. The rare-earth group has not yet reached the limit of solubility. The hydroclone loop is being operated during HRT run 17, but analytical data are not yet available.

High temperatures (200 and 140°C) occurred in the charcoal adsorber beds as a result of a recombination of D_2 and O_2 that had passed through the low-pressure system of the reactor. There is evidence that the charcoal in one of the beds ignited and that 6 to 8 lb of the 520 lb of charcoal burned. A breakthrough of activity from that bed occurred five days after the incident; the bed was isolated, and a spare bed was used while the incident was investigated and the short-lived activity decayed. All the adsorption beds are operating normally at the present time. (AEC Activity 4103.1)

Thorium Oxide Slurry Development: Oxide Preparation. — A total of 586 lb of regular-production (1600°C-fired and classified) thorium oxide product was prepared for slurry engineering tests. An additional 175 lb of 1600°C-fired oxide, classified to less than 5 μ and then reclassified to less than 3 μ, was prepared for in-pile slurry loop tests.

Thorium-uranium oxides containing 4, 8, and 12% uranium oxide were prepared by precipitating uranium on a 650°C-fired thoria from an ammonium uranyl carbonate solution and firing the mixed solids. A mixed oxide containing 9% uranium oxide retained the average particle size of the original thoria when fired at temperatures up to 1000°C, but showed an increase in particle size after being fired at or above 1125°C. Successive partial dissolutions of the 1000°C-fired thorium-uranium oxide with HNO_3-H_2O_2 showed the uranium to be nearly homogeneously distributed in the mixed oxide. (AEC Activity 4103.1)

Thorium Oxide Slurry Development: Irradiated Oxide Studies. — The recovered solids from a slurry of 1600°C-fired oxides which had been irradiated for 3235 hr at 300°C in the LITR(nvt = 3 x 10^{20}) in a settled condition were black in color and sandlike. A slurry of 1000°C-fired thorium-uranium oxides

(U/Th = 0.05) irradiated in the LITR showed no radiolytic gas. Out-of-pile measurements in the dash-pot-stirred irradiation bomb with a similar slurry showed that the initial rate of recombination of gas was 0.03 mole of D_2 per hour per liter at 300°C and a D_2 partial pressure of 100 psi. The recombination rate increased by a factor of 3 after one week of operation and showed no further increase after two additional weeks of operation.

In other out-of-pile studies with stoichiometric mixtures of D_2 and O_2, reaction rates in slurries of 1000°C-fired thorium-uranium oxide (U/Th = 0.05) containing 1000 g of Th per kg of D_2O and 0.04 m in MoO_3 were between 1 and 2 moles of D_2 reacted per hour per liter of slurry at 280 to 290°C and a D_2 partial pressure of 100 psi. With palladium catalyst in the ratio 1 Pd/1000 Th (added as $PdO \cdot H_2O$), the rate at 148°C was 4.6 moles of D_2 per hour per liter at a D_2 partial pressure of 100 psi. (AEC Activity 4103.1)

HRP Chemistry: Fuel Studies. — In the temperature range 60 to 90°C the rate of the decomposition of peroxide in uranyl sulfate solutions was increased by the addition of cupric sulfate catalyst, as had previously been reported for 100°C by Silverman, Watson, and McDuffie (Ind. Eng. Chem. 48, 1238, 1956), but the addition of catalyst did not lower the activation energy for the process.

A mechanism is being sought for the precipitation of uranium and copper from the HRT fuel solution in run 17. Two experiments were done on a solution that simulated the reactor fuel solution. Hydrogen peroxide was added quickly at 90°C to precipitate a portion of the uranium. The mixture was immediately filtered, and no copper was found in the precipitate. A partial reduction of a simulated HRT fuel solution with hydrogen gas at 250°C caused substantial loss of hexavalent uranium from solution but led to no detectable production of tetravalent uranium and no detectable loss of copper from solution. The lost uranium, not removed by water washes and presumably deposited upon the walls of the autoclave in a reduced form, was easily dissolved by nitric acid and recovered. It appears unlikely that the simultaneous disappearance of copper and uranium from the HRT fuel solution in run 17 was caused by precipitation of peroxide or partial reduction of the solution.

Hydrogen-oxygen recombination studies were performed at 200°C, and the results were compared with those of similar studies in which the oxygen gas was replaced by CrO_3 dissolved in the solution. The observed rate constants for the decrease in hydrogen pressure were indistinguishable from those for the decrease in hydrogen-oxygen pressure. Thus, the presence of oxygen gas does not invalidate the interpretation of the results of laboratory experiments that has previously been used in the study of homogeneous catalysis. (AEC Activity 4103.1)

HRP Analytical Chemistry. — A simple, yet fairly sensitive procedure for the separation and subsequent flame-photometric determination of aluminum in thorium oxide was developed. Aluminum is separated from thorium and other

interfering elements by extracting it with 8-quinolinol in hexone from a solution saturated with ammonium carbonate. The emissivity of the aluminum in the organic medium is then measured at 484 mμ.

A study was made of the extractability of thorium from 0.1 M $HClO_4$ solutions with 2-thenoyltrifluoroacetone (TTA) in carbon tetrachloride as a function of the acetate concentration of the aqueous phase. It was established that acetate enhances the extractability, owing to the formation of Th-TTA-HAc adducts, the principal species of which were identified and related to the acetate concentration.

A procedure was devised in which the interference of copper could be avoided in the colorimetric determination of iron in uranyl sulfate solutions without resort to time-consuming electrolytic deposition techniques for the removal of copper. The iron is extracted with a mixture of 4-methyl-2-pentanone (hexone) and amyl acetate from an 8 N HCl solution, re-extracted from the organic phase, and then finally determined by the α,α'-bipyridyl method.

A new coulometric titration method for copper and uranium was developed, based on the fact that copper reduction at a mercury cathode is reversible, whereas the uranium reduction is irreversible in the 1 M sulfuric acid electrolyte. (AEC Activity 4103.1)

Slurry Testing and Evaluation. — The effect of uranium and molybdenum oxides on the long-term behavior of high-fired thoria slurries was explored in run 200A-15. This run, in which the slurry contained 650 g of Th per kg of H_2O (1600°C-fired ThO_2) with 1/2 wt % uranium added as UO_3 and with 1.16 mole % molybdenum oxide, was terminated on schedule after circulation for 3822 hr at 300°C. The slurry was easily drained and washed from the loop. Examination of the pump impeller and casing revealed that titanium is far superior to stainless steel as a structural material in regions of high velocity and flow separation. The pump was in generally good condition except for the stainless steel impeller and thermal barrier. The titanium scroll liner showed little attack, which was in the region near the eye of the impeller. The presence of uranium and the molybdenum had no adverse influence on the behavior of the slurry.

A 100-gpm run at 280°C in which the thoria was coated with SiO_2 led to the conclusion that this composition was not stable at elevated temperatures. The silicate slowly left the thoria. (AEC Activity 4103.1)

Slurry Blanket System Development. — The 300-SM blanket test facility was shut down on June 24, 1958, after having been operated for 645 hr with a slurry containing 323 g of Th per kg of H_2O for final measurements of circulation in a blanket of HRT geometry. About 85.2% of the thoria charged was recovered from the storage tanks after draining of the circulating system,

and 9.1% additional was recovered after rinsing. The remainder of the charge has not yet been fully accounted for.

The pump inspection showed little damage, and no repairs are considered necessary. The Zircaloy-2 impeller lost only 0.4% of its weight during the run and was in excellent condition. Like titanium, Zircaloy-2 seems to be a good material of construction in regions of high slurry velocity.

The system is being altered to permit operation at 2000 psi. (AEC Activity 4103.1)

Engineering Research. — A vertical capillary-tube viscometer for routine determination of the shear diagrams of thoria slurries has been designed. The viscometer has a length-to-diameter ratio of no less than 1000 and will require about 200 ml of slurry.

Dye-retention studies of the flow patterns in two HRE-3 blanket models were completed. The last visible trace of dye was always associated with thin fluid filaments which clung to the outside of the shroud.

A fluid-flow visualization loop utilizing Milling-yellow solution as a working fluid was constructed. Flow patterns have been obtained in a number of two-dimensional models. Results indicate that the stability of flow through a divergent channel is dependent upon the included angle of the channel; at an angle of 60 deg the flow was stable, while at an angle of 30 deg it was unstable.

An analysis of the boundary-layer equations for Bingham plastics showed that these equations essentially reduce to those for Newtonian fluids if the Hedstrom number is less than the Reynolds number raised to the 1.5 power.

Fluidized-bed studies with thoria spheres indicated a slight decrease in slurry concentration with increasing bed height. The net fluid velocity required for slurry fluidization was nearly the same as the slurry settling velocity. (AEC Activity 4103.1)

Core and Blanket Vessel Development. — Tests conducted with an HRT-size core model, containing a north polar inlet and concentric annular outlet, showed that suspended solids were removed readily. Heat-meter measurements at points on the core tank surface indicated heat transfer coefficients of 800 to 2000 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot (^{o}F)^{-1}$ with reactor flow conditions. The normal flow conditions were not disturbed by insertion of corrosion specimens.

An alternate geometry being considered for the HRT replacement core, containing a vaned annular inlet and concentric polar outlet, was also tested. This geometry produced a strong vortex which trapped solids. The vibrational level was high.

Velocity measurements made in a 4-ft spherical re-entrant type of core vessel revealed the existence of a strong boundary layer at the tank wall. The observed velocities were in good theoretical agreement with the high heat transfer coefficients which had been measured previously. (AEC Activity 4103.1)

Centrifugal Circulating Equipment. — The initial slurry run in the 300A loop was interrupted, after 41 hr of satisfactory operation, by a leak in a fitting. The run was resumed following decontamination of the loop shielding and repair of the leak.

Conversion of the 400A-2 solution pump to the 200Z slurry pump was completed. Special features of the new pump are aluminum oxide radial bearings, a cast Zircaloy impeller with radial vanes on the lower shroud to reduce wearring pressure drop, and a cast Zircaloy diffuser to reduce radial thrust. Loop modifications are about 50% complete. (AEC Activity 4103.1)

Miscellaneous Equipment. — Development of devices for viewing the break in the HRT core tank was continued. The TV camera, together with the manipulator, was tested in the HRT mockup vessel and found to give an excellent picture of the outside of the core tank. Search patterns for use of the camera in the reactor are being programed first in the mockup. Improvements to increase mechanical reliability and illumincation were made in the folding-arm periscope device used previously. (AEC Activity 4103.1)

Heat Removal Equipment. — Detailed design of test-model heat exchangers, incorporating advanced features, was completed by Griscom-Russell Company and approved by ORNL. Purchase of material has begun.

No stress-corrosion cracking was observed in the metallographic examination of five small tube bundles which had been thermal-cycled in chloride-containing boiler water for up to 5000 hr. This unexpected good behavior might be attributed to cathodic protection of the stainless steel tubes by the carbon steel tube sheet.

A small plastic horizontal-tube heat exchanger was operated in slurry. No difficulty was experienced with plugging or with resuspension of the slurry. (AEC Activity 4103.1)

Gas Handling Equipment. — The solubility of a stoichiometric mixture of hydrogen and oxygen in light water was found to be about the same as the individual gas solubilities at 275°C. This was in contrast with previous runs at 250°C, which gave high solubilities for the mixture. Further runs will be required to check these results.

Construction of an Inconel natural-circulating high-pressure recombiner was completed, and the assembly was attached to the high-pressure recombiner loop. (AEC Activity 4103.1)

Fuel Systems Development. — The HRT fuel mockup system has been operating continuously since July 23, 1958, with a titanium pressurizer and heat exchanger in which the uranyl sulfate fuel is boiled. As of August 22 the system had run 692 hr at 280°C and 1500 psi. The pressure control has been very good with the Dowtherm boiler operating at 90% of its capacity. The fuel has appeared to be chemically stable so far, in contrast with the loss of uranium and copper that takes place under similar conditions in a stainless steel pressurizer. The heavy-phase formation temperature for this fuel composition (simulating the HRT fuel) was found to be 329°C, whereas the highest system temperature, occurring in the titanium heat exchanger, does not exceed 318°C. The fuel temperature in the pressurizer is 313°C.

The system will be shut down after 1000 hr of operation for inspection of the titanium pressurizer and piping. (AEC Activity 4103.1)

Feed Pump Development. — Prototype HRT fuel feed pumps and purge pumps continued to operate without difficulty during the report period. The same is true of the single-stage turbine pump.

A duplex diaphragm slurry feed pump was operated, and it circulated slurry from an atmospheric feed tank to a reservoir at 1800 psi. The pump's Stellite check valve trim was badly worn in several days; however, tungsten carbide trim has remained operable after 180 hr. A small canned-rotor pump is used in the suction line to ensure high volumetric efficiencies. (AEC Activity 4103.1)

MOLTEN-SALT REACTOR (MSR) PROGRAM

Molten-Salt Reactor: Design. — Reactor calculations were completed that provide a comparison of U^{235} and U^{233} as fuel for a two-region, homogeneous, molten-fluoride-salt reactor. The reference design reactor having an 8-ft-dia core and an external fuel volume of 339 ft^3 was assumed for these calculations, and for the long-term calculations it was assumed that reprocessing of the fuel began immediately after startup of the reactor. For a U^{235}-fueled reactor with 1 mole % thorium in the fuel salt, the initial inventory of 616 kg increased in 20 years to 1015 kg, the integrated net burnup was 2010 kg, and the initial regeneration ratio of 0.64 decreased to 0.592. For the U^{233}-fueled reactor, 7 mole % thorium was added to the fuel in accordance with the higher fission cross section of U^{233} in the epithermal region. The initial U^{233} inventory of 1078 kg increased to only 1451 kg in 20 years, the regeneration ratio decreased from 1.072 to 1.027, and there was a net gain of about 100 kg of fuel.

Design studies of a large power reactor system indicate that the minimum-cost heat transfer system that provides safety comparable to that of a boiling water reactor will be provided by direct transfer of heat from fuel to steam. A double-walled heat exchanger and a positive method of leak detection will be required. (AEC Activity 4103.1B)

Molten-Salt Reactor: Experimental Engineering. — A loop is being constructed for forced circulation of molten-salt fuels in a radiation field in the MTR. Tests with a prototype pump and loop demonstrated the feasibility of filling the loop at ORNL, allowing the salt to freeze, and then remelting the salt after installation of the loop in the irradiation facility.

The freeze flange and indented-seal flange joints developed for molten-salt systems were shown to be satisfactory for use in sodium systems. Also, two freeze flange assemblies on 4-in.-dia salt-filled pipes were thermally cycled 50 times between 1100 and 1300°F, and there was no salt leakage.

A second test of an INOR-8 vs INOR-8 molten-salt-lubricated hydrodynamic bearing was completed. The bearing operated at 1200°F for 500 hr through 57 stop-starts. Lubrication was satisfactory for extended periods of operation at 1200 rpm with a radial load of 200 lb (45 psi on the bearing projected area).

The small frozen-lead-sealed pump that was designed for operating submerged in the molten salt being pumped has continued to operate satisfactorily. (AEC Activity 4103.1B)

Molten-Salt Reactor: Metallurgy. — All planned thermal-convection loop corrosion tests of INOR-8 in contact with molten salts were completed. Metallurgical examinations revealed no measurable attack in any of these loops. Similarly, a section of the hot leg of an INOR-8 forced-circulation loop that circulated an $LiF-BeF_2-UF_4$ salt mixture for 3300 hr at a maximum temperature of 1300°F did not show evidence of attack.

Qualitative tests of the mechanical properties of INOR-8 weldments verified the bend-test evidence that the ductility of INOR weld metal decreases above 1200°F. The ductility can, however, be improved by aging.

INOR-8 specimens exposed to graphite and an $LiF-BeF_2-UF_4$ salt for 2000 hr at 1300°F showed no signs of carburization. The carbon content of the salt was unchanged. After 500-hr tests designed to determine whether an $LiF-BeF_2-UF_4$ salt would penetrate CCN graphite, there was no evidence of penetration.

Brazing alloys with silver contents ranging from 5 to 100 wt % were found to have poor corrosion resistance to $LiF-BeF_2-UF_4$ salts. (AEC Activity 4103.1B)

Molten-Salt Reactor: Chemistry. — A phase diagram showing the liquidus surfaces of the system $LiF-BeF_2-ThF_4$ was completed. Compositions and temperatures of each of the invariant points occurring at the liquidus were determined with a precision of ± 1 mole % and $\pm 3°C$. An unusual solid solution involving dissolution of both BeF_2 and ThF_4 in the solid phase $3LiF \cdot ThF_4$ causes the single-phase region $3LiF \cdot ThF_4$ solid solution to occur. A limit of the single-phase region occurs along the composition line between $3LiF \cdot ThF_4$ and

the composition $LiF-ThF_4-BeF_2$ (58-29-13 mole %) in the solid state. This single-phase solid solution prevents the phase $7LiF \cdot 6ThF_4$ from contacting the primary phase of an $LiF-BeF_2$ compound and thereby excludes the occurrence of a ternary invariant point involving $7LiF \cdot 6ThF_6$ and $LiF-BeF_2$ solid phases.

It was determined that CeF_3 dissolved in a mixture of $LiF-BeF_2-UF_4$ (62.8-36.4-0.8 mole %) can be extracted and that LaF_3 can be substituted for it simply by contacting the liquid with solid LaF_3. This process is brought about by the formation of CeF_3-LaF_3 solid solutions. A similar behavior was encountered previously with a solvent consisting of $NaF-ZrF_4-UF_4$ (50-46-4 mole %). The tendency of the rare-earth fluorides to form solid solutions under these conditions may be useful in the replacement of rare-earth nuclear poisons in molten fluoride solvents by simple substitution of less poisonous rare earths. The calculations are amenable to treatment analogous to that used in simple solvent extraction processes. Experimentally determined partition coefficients for the rare-earth fluoride distribution between the solid and liquid phases agree well with the coefficients calculated a priori from numerical values of the solubilities, in the liquid, of the particular rare-earth fluorides involved.

The 250-lb fluoride production facility was activated to process eight batches of beryllium-containing fluoride compositions for use in corrosion and component testing. The eight batches of varying composition totaled 640 kg, which is expected to be sufficient to meet all major requirements for the remainder of this fiscal year. (AEC Activity 4103.1B)

Molten-Salt Reactor: Heat Transfer and Physical Properties. — Experimental measurements of the heat capacity of the salt mixture $LiF-BeF_2-UF_4$ (53-46-1 mole %) have shown that the enthalpies are a function of the heating-cooling-cycle history of the sample. For samples heated to 450°C and suddenly quenched in the calorimeter, an average enthalpy of 225 cal/g was obtained. Similar data taken after heating to 500°C, cooling in the furnace to 450°C, and then quenching gave an average value of 209 cal/g. Data on four samples heated to 500°C and suddenly quenched resulted in an average enthalpy of 220 cal/g, a value less than that for the original 450°C measurements. Since BeF_2 is known to form glasses, it is suspected that the anomalous behavior is associated with this phenomenon. Thus it is possible that the prior temperature history of the sample determines whether the molten salt supercools to a glass, crystallizes to a stable solid, or forms a mixture of these two states. Data are now being obtained for samples cooled slowly in the calorimeter rather than quenched. (AEC Activity 4103.1B)

Molten-Salt Reactor: Radiation Damage. — The resistance-heated mockup of the thermal-convection loop designed for circulating molten salts in a radiation field in the LITR was placed in operation. Modifications of the air-cooling system are being made in order to improve the temperature distribution around the loop. (AEC Activity 4103.1B)

Molten-Salt Reactor: Fuel Processing. — A molten-salt reactor blanket salt ($LiF-BeF_2-ThF_4$, 71-16-13 mole %) containing sufficient irradiated thorium to give 5.6×10^7 Pa counts/min per gram of salt ($\sim 10^{-9}$ mole % Pa) was fluorinated for 2 hr at 600°C. None of the protactinium was removed. However, this does not prove that protactinium would not be removed by fluorination at the higher concentration (0.002 mole %) expected in the MSR.

Preliminary solubility measurements for UF_4 and ThF_4 in HF solutions indicate solubilities generally of the order of 0.01 mg/ml. With iron and nickel metals present to prevent uranium oxidation, solubilities in 100, 90, and 80 wt % HF were 0.02, 0.01, and 0.06 mg/ml, respectively. In 100, 95, and 90 wt % HF saturated with LiF the solubilities were 0.02, 0.006, and 0.005 mg/ml, respectively. Uranium solubilities in the absence of iron and nickel were scattered but generally higher, probably due to oxidation. The solubilities of ThF_4 were approximately the same as those of UF_4 under the same conditions. (AEC Activity 4103.1B)

CIVILIAN-POWER ADVANCED GAS-COOLED REACTOR

Gas-Cooled Reactor: Design. — The procedural aspects of ORNL participation in the AEC review of gas-cooled reactor design and fabrication were studied. Recommendations were submitted to the AEC relative to such review responsibilities, and the areas in which ORNL will be prepared to provide technical assistance were outlined. (AEC Activity 4110)

Gas-Cooled Reactor: Experimental Engineering. — A survey of facilities for in-pile testing of fuel-element materials was completed, and plans were formulated for capsule tests in various facilities. A loop is to be constructed and to be operated in the General Electric Test Reactor (GETR) that will provide forced circulation of the cooling gas. Equipment is being designed and constructed for capsule irradiation tests in the ETR, the ORR, and the LITR. Design and construction of equipment for out-of-pile heat transfer and materials compatibility tests are under way. Design work is continuing and components are being fabricated for the ORR gas-cooled loop, previously described. (AEC Activity 4110)

Gas-Cooled Reactor: Instrumentation and Controls. — Situations which would produce large temperature excursions in fuel elements are being examined on the analog facility of the ORNL Reactor Controls Department. Three of the situations are (1) sudden loss of a large fraction of the heat removal from the reactor with the reactor initially at full load, (2) insertion of the control rods at emergency speed with the reactor initially at full power, and (3) continuous withdrawal of the control rods with the reactor shut down or at a rather low power. Although the reactors that have been examined thus far were not complete GCR-2 systems, the results indicate the magnitude of the transients that may be encountered by the GCR-2. (AEC Activity 4110)

Gas-Cooled Reactor: Metallurgy. — Nondestructive evaluation of several lots of type 304 stainless steel capsule tubing was completed. Ultrasonic and dye-penetrant inspections showed surface roughness and many scratches and pinholes. Two lots of tubing were severely warped, and of twelve lots, only six were completely within commercial tolerances for wall thickness and eccentricity. The need for rigid specifications for this thin-walled tubing is evident.

The joining of beryllium sheets by resistance welding was attempted. Each spot weld showed radial cracks through the weld and base metal. Small root cracks were found that appeared to result from lack of breakdown of the oxide film. A pickling procedure was therefore made available for removal of the film prior to welding. Additional fusion welds will be prepared after design and assembly of preheating apparatus are completed.

The development of a continuous process for the production of UO_2 from UF_6 advanced rapidly as a result of changes in the calcining conditions. A recent batch of powder from this process was found to have the forming and sintering characteristics necessary for the production of 95%-dense pellets by the minimized fabrication process developed by the ORNL Ceramic Laboratory.

Metallographic examinations were completed of metal specimens exposed to helium and TSF graphite in static pot tests at 1400°F for 1000 hr. The appearance of the stainless steel samples was unchanged, but specimens of low-alloy steels showed slight attack. Hardness measurements on as-received types 304, 310, 316, and 410 stainless steel were compared with data on samples from the static pot tests. The observed decreases of 20 to 30 in diamond-point hardness numbers are attributed to the annealing effects at the high operating temperatures.

Metal samples were also exposed to TSF graphite and helium or to TSF graphite and CO_2 in a natural-convection loop in which the cold leg was maintained at 500°F. The samples exposed to helium were held for 2100 hr at 1400°F, and the samples exposed to CO_2 were held for 1000 hr at 1000°F. Prior to operation with CO_2, helium was circulated in the loop to remove any contaminants which might interfere with the evaluation of the CO_2-graphite reaction. The types 304 and 310 stainless steel specimens showed similar weight gains in both tests. Type 316 stainless steel, on the other hand, gained weight in the helium environment but lost weight in CO_2, while type 410 stainless steel specimens gained about ten times as much weight in CO_2 as in helium. All the stainless steel specimens tested in CO_2 had iridescent surface films, and the low-alloy steels had dark gray surfaces. X-ray diffraction examinations of the films are under way. The cold-leg side arms of the apparatus were examined. The cold leg used for the CO_2 test had a black deposit which was heaviest at the elbow. This deposit is being studied by x-ray and electron-diffraction techniques. (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. A new area of assistance was initiated in which Laboratory inspectors are witnessing, on behalf of MRB, inspections and tests performed during the fabrication of the reactor system for the NS "Savannah."

A survey of ozone explosions which have occurred in traps cooled with liquid nitrogen in the presence of radiation indicated that the proposed nitrogen-cooled charcoal trap in the gaseous waste system for the NS "Savannah" is not an explosion hazard.

Samples of boronated stainless steel tubing (330 ppm boron), similar to tubing proposed for NMSR fuel rods, were received from the reactor fabricator. Nondestructive tests indicated the tubing to be of high quality. Several welds of the boronated tubing to enclosures of the reference-cup type were made, and all the welds were found to be free from cracks, pores, and other defects.

A wire-pyrometry technique for measuring UO_2 pellet temperatures during irradiation tests is being studied. Small-diameter gold and nickel wires were inserted into sample pellets of depleted UO_2, and the melting points of the inserted wires were determined.

Designs were completed for several special components for the pressurized water in-pile loop, including the pressurizer, heaters, purification system, strainers, and filter. Design drawings of structural work were released for inclusion in a contract with other ORR building alterations to be done by an outside contractor. The location of the in-pile section was changed to lattice positions A_1 and A_2 in the ORR in order to simplify the mechanical design.

A detailed outline of the health physics monitoring program for the NS "Savannah" was developed and is now being reviewed for completeness and for final choice of presentation sequence. In waste disposal experiments, Cr^{51} was eluted from a mixed resin bed to the extent of 80% by 15 column volumes of sea water, and Ru^{106} was eluted to the extent of 30% by 16 volumes. The remaining fractions of Cr^{51} and Ru^{106} were eluted very slowly, and it appears that hydrolysis occurred during the elution. (AEC Activity 4151)

PACKAGE REACTOR DEVELOPMENT

Army Package Power Reactor (APPR-1). — Specifications were issued for fabrication of the fixed fuel elements for core II. An information meeting was then held at the Laboratory on July 31 for prospective bidders on a contract to furnish the fixed fuel elements for core II.

A program was initiated to determine more accurately the boron content of core I. Analytical standards were prepared for calibration of analytical procedures by mixing weighed quantities of metal powders and UO_2 to simulate the composition of a fuel plate. When analytical procedures are confirmed, several fuel plates that were fabricated for core I but which were rejected for minor flaws will be analyzed for boron.

At the request of the Army Reactors Branch, proposed design requirements for a reactor similar to APPR-1 (designated as APPR-2) were reviewed, and comments were submitted to the AEC. (AEC Activity 4601)

GENERAL REACTOR RESEARCH

Basic Reactor Research. — In the problem of resonance absorption of neutrons in nuclear reactors, formulas were developed for the effective resonance integral of single resonances in the "narrow resonance" approximation in homogeneous media, and in the "narrow resonance" and "infinite mass absorber" approximations in heterogeneous media. The Doppler effect in homogeneous media can be expressed through a certain function of two variables, $J(\xi,\beta)$, the properties of which were studied in detail. The effect of interference between resonance and potential scattering on the effective resonance integral was studied. The Doppler effect in heterogeneous media in the "narrow resonance" case can also be expressed through the function $J(\xi,\beta)$ if a rational approximation for the average escape probabilities, due to Wigner, is introduced. Furthermore, a formal identity exists between the homogeneous and heterogeneous cases for "narrow" resonances. The error caused by the rational approximation was studied in detail, and an improved formula was suggested for the case of no Doppler broadening and no interference scattering. In heterogeneous media in the "infinite mass absorber" case, an approximate expression for the albedo of an absorber lump was suggested on the basis of heuristic arguments. Precise calculations of the albedo based on a variational method of solving the monoenergetic transport equation were compared with this approximation and showed it to be quite accurate. Use of the approximate albedo again permits the expression of the Doppler effect through the function $J(\xi,\beta)$. The effect of interference between potential and resonance scattering in the "infinite mass absorber" approximation was also studied. These results are applicable to calculating the absorption in low-energy resolved resonances for which the widths and energy are known. In the region beyond the experimental limit of resolution, statistical considerations were employed. The formulas for single resonances were averaged over the probability distributions of the partial widths. The theory was applied to the calculation of the effective resonance integrals of uranium and thorium rods. Good agreement was obtained for uranium and fair agreement for thorium.

In the investigation of the spectrum of prompt gamma rays from U^{235} fission, observations of gamma rays in coincidence with fission have been concluded for gamma-ray energies greater than 0.4 Mev. Experimental work is

now confined to the determination of small corrections and the calibration of test sources of gamma rays. Current efforts are aimed at standardizing a source of O^{16} gamma rays to allow a measurement of spectrometer efficiency for 6-Mev photons.

The equipment required for measuring the neutron diffusion parameters in beryllium has been tested. A box lined with boron powder (a 20-in. cube) was filled with the beryllium, and the system was tested to determine the time required for cooling the beryllium to the temperature of liquid nitrogen. The time was 4 hr. (AEC Activity 4202)

Fuel Element Development. — One fuel element containing a dispersion of U_3O_8 in aluminum, in which the uranium contained 20% U^{235}, was successfully made and shipped to the MTR for irradiation testing. The second fuel element was rejected after processing because of improper handling. The fuel plates are now rolled to attain proper flatness.

Recent experimental results on the diffusion of uranium into the type 1100 aluminum cladding of composite fuel plates containing an 18.5% U—Al alloy have been extremely erratic, and are being attributed to contamination during handling subsequent to fabrication. The composites are now being encapsulated after fabrication in order to prevent contamination of the exterior. (AEC Activity 4203)

Power Reactor Fuel Reprocessing: Alternate Aqueous Processes. — The complete decladding of an EBWR fuel sample with 6 M NH_4F and dissolution of the core alloy by addition of 7 M HNO_3—0.6 M $Al(NO_3)_3$ required 10 hr. The final uranium concentration was about 0.3 M. The conversion of NH_4UF_5 to UO_2 by metathesis with NH_4OH is more than 95% complete in 1 hr at temperatures from 35 to 87°C.

Mercuric nitrate—catalyzed nitric acid is being used as a dissolvent for foreign reactor fuels. Boric acid, added to the dissolvent as a neutron poison, was carried through the fuel-dissolution and gelatin head-end treatments without loss. The removal of colloidal silica by formation of a gelatin-silica polymer prevented the formation of emulsions during solvent extraction. The gelatin-silica polymer was not degraded under Co^{60} gamma irradiation at levels up to five times those expected in processing. A flowsheet based on solvent extraction by 6% tributyl phosphate was developed to fit existing equipment in the 3019 pilot plant and was tested in 0.75-in.-dia pulsed columns; the uranium and plutonium losses were less than 0.01% when extrapolated to full process levels.

The tin in Zircaloy-2 is efficiently dissolved upon the addition of 0.6 to 1.0 M NH_4NO_3 to the 6 M NH_4F decladding solution used in the Zirflex process for the decladding of PWR blanket fuel. Cold 0.05 M $Al(NO_3)_3$ appears to be a satisfactory reagent for the removal of fluoride sorbed on UO_2 pellets during decladding.

Testing, with unirradiated fuel, of the equipment for the decladding of PWR blanket elements with NH_4F-NH_4NO_3 and with HF was continued. The uranium loss to the decladding solution was 0.1% with 6 M NH_4F—1.0 M NH_4NO_3 and 2.9% with 6 M HF. The cladding of oxide fuel was not completely dissolved upon boiling for 17 hr in 6 M NH_4F—1.0 M NH_4NO_3. The fluoride concentration in the uranyl nitrate product solution was between 0.08 and 0.14 M in all runs.

The dissolution of ThO_2-UO_2 pellets (simulating Consolidated Edison fuel) in 13 M HNO_3—0.04 M NaF—0.04 M $Al(NO_3)_3$—0.3 M H_3BO_3, in which the boron is added for criticality control, resulted in an unstable product solution. Precipitation of $Th(NO_3)_4$ occurred when the final 1 M $Th(NO_3)_4$ solution was cooled to room temperature. The use of 0.07 M $Na_2B_4O_7$ in the dissolver solution did not cause precipitation. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Hermex Process. — The solubility of thorium in mercury increases from about 0.002 to 0.03 wt % as the temperature is increased from 40 to 356°C. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Darex Process. — Modifications to the continuous Darex loop equipment to provide for further automatic control were completed; the loop, except for the feed adjustment tank, was then operated satisfactorily in a 5-hr trial run. In separate operation of the feed adjustment tank, continuous adjustment of the tank overhead for use as feed to the stripper reboiler was demonstrated with 22.5 M HNO_3 as makeup and with density control of the surge tank.

A batch Darex flowsheet was developed for the preparation of APPR feed containing 60 g of metals per liter. Feed containing <50 ppm Cl^- was obtained and all the recoverable HNO_3 and HCl was found to be re-usable if 95% HNO_3 was added after the dissolution in order to remove the chloride. In this operation, solvent extraction feed, 3 N in HNO_3, was satisfactorily obtained by adding 0.15 volume of 95% HNO_3 to the original unit volume of dissolver product, boiling to 0.3 volume, adding a second 0.15 volume of 95% HNO_3, refluxing, and finally diluting to the original dissolver volume.

Laboratory studies of the dissolution of APPR (free of brazing alloy) and of type 304L stainless steel showed the following:

Insolubles, %	0.19
Uranium loss to insolubles, %	0.006
Off-gas volume, liters/kg	460
Consumption of H^+, moles/g	0.068
Consumption of NO_3^-, moles/g	0.016 (APPR) and 0.017 (SS)
Consumption of Cl^-, moles/g	0.0049 (APPR) and 0.0021 (SS)

(AEC Activity 4301)

Power Reactor Fuel Reprocessing: Equipment Development. — It was found that a 10-in.-thick sand bed could be removed from both 6-in. and 12-in. experimental filters by using a steam jet and backwashing the bed with a water bleed to the suction side of the jet.

For the hydraulic transfer of pellets, a vessel with a conical bottom was equipped with a 1-in. transfer line. A nozzle had to be installed opposite the opening of the transfer line in order to accomplish transfer. With a 1/2-in. nozzle, transfer was accomplished, but flooding of the container occurred. With a 1/4-in. nozzle, flooding was eliminated until the slug of the pellets jammed in the elbows of the transfer line. Transfer was accomplished satisfactorily after sweep elbows were installed. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Solvent Extraction Studies. — Studies were made of the flooding of 2-in. glass pulsed columns in which the aqueous phase was water and the organic phase was a solution of 30% tributyl phosphate (TBP) in Amsco. When the ratio A/O of the flow rate of the aqueous phase to the flow rate of the organic phase was 0.4, the flooding rate ranged from 770 $gal \cdot ft^{-2} \cdot hr^{-1}$ at 50 cycles/min to 465 $gal \cdot ft^{-2} \cdot hr^{-1}$ at 80 cycles/min; at A/O = 0.8, the ratio for the Yankee Atomic Darex flowsheet, the range was 775 to 520 $gal \cdot ft^{-2} \cdot hr^{-1}$; the logarithm of the flooding rate decreased linearly with increasing frequency of pulsing. Flooding for the Yankee Atomic flowsheet occurred at 615 $gal \cdot ft^{-2} \cdot hr^{-1}$ at 60 cycles/min with relative flow rates of 1.5 for the solvent (30% TBP in Amsco), 0.2 for the scrub (3 $\underline{M}$ HNO_3), and 1.0 for the feed (126 g of U per liter, 30 g of stainless steel per liter, 2.16 $\underline{M}$ HNO_3). A successful run of 10 hr was made for determining the height equivalent to a theoretical stage; the material balance was 99%, and the extraction loss was 0.13% for an 8-ft extraction height. Iron, chromium, and nickel did not extract.

In laboratory-scale solvent recovery tests it was found that solvent degraded either thermally or by irradiation may be recovered by treatment with lime if the concentration of TBP in Amsco is less than 10%. Any concentration of thermally degraded solvent may be treated with lime following treatment with 0.1 $\underline{M}$ KOH, but high concentrations of irradiation-degraded solvent formed emulsions with lime and water even when treated with KOH. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Zircex Process. — In a 10-hr Zircex run, 52.5% of a 7.72-kg section of an EBWR assembly was hydrochlorinated at an average reaction rate of 3.3 $mg \cdot cm^{-2} \cdot min^{-1}$. The buildup of a crust of uranium chlorides on the surface of the plates retarded the reaction. The loss of uranium to the sublimate was 0.004%. The amount of reacted uranium insoluble in 10 $\underline{M}$ HNO_3 was 0.021%. (AEC Activity 4301)

Fused Salt—Fluoride Volatility Processing. — Hydrofluorination tests of 3-, 6-, and 12-in.-high fuel elements, in which the acid feed rate and the size of the initial charge of NaF-LiF salt were the same for all the tests, indicated that the specific dissolution rate is inversely proportional to the height of the fuel element. The specific rates for the three tests were 5.5, 1.7, and 0.88 $mg \cdot cm^{-2} \cdot min^{-1}$, respectively.

A process flowsheet for STR fuel alloy has been devised which minimizes corrosion and enhances decontamination of the product UF_6 from activity and from other volatile fluorides. A salt mixture containing 42 mole % NaF and 58 mole % LiF is used in the hydrofluorination step, with the final composition being 31 mole % LiF—24 mole % NaF—45 mole % ZrF_4, a composition close to a ternary eutectic having a melting point of 440°C. The fluorination of uranium from this base salt is thus feasible in the range 450 to 500°C. This flowsheet supersedes the previously reported LiF-NaF method, where the final salt composition contained 55 to 60 mole % ZrF_4 and fluorination would have required a temperature of 600 to 700°C.

Tests have demonstrated that uranium precipitates from 2% aqueous solutions of NaF, possibly as $NaUF_5$, which is less soluble than UF_4. Either iron reduction followed by $NaUF_5$ precipitation or anion exchange with Dowex 21K resin appears feasible as a method for the recovery of small amounts of uranium from the NaF waste resulting from pilot-plant testing of the volatility process. (AEC Activity 4302)

Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes by Sintering. — In the investigation of the treatment of wastes with clays, attempts to correlate the sorption of cesium with the base exchange capacity and surface area data for the clays have been unsuccessful. Tests conducted on clays whose structural features were modified by selected treatments have proved helpful in elucidating the mechanism of cesium sorption.

Of the many clays investigated, illite proved to have the highest affinity for cesium. In order to secure a clay whose structural parameter was more nearly like that of illite, vermiculite was treated with KCl. The modified vermiculite sorbed more cesium than did a comparable weight of illite, showing the importance of steric properties on sorption. For an exchange reaction to occur, ions in solution must migrate through the particles as well as compete with each other for the exchange. The width of the "openings" leading to the charged sites is 7.8 Å in untreated vermiculite and 3.4 Å in treated vermiculite and illite. In a sodium-neutralized Purex waste, the strongest competitor of cesium is sodium. In this system the highly hydrated sodium ion must lose some of its water of hydration, thus decreasing its effective diameter, in order to enter the "opening." When the entrance dimension was reduced to 3.4 Å, the amount of water lost by sodium was insufficient to enable it to enter the particle. With less competition from sodium, the sorption of cesium was increased.

After the KCl treatment the local shale, containing 25% vermiculite and 40% illite, showed a threefold increase in its ability to sorb cesium. The increased sorption after treatment was due to the changes induced in the vermiculite and not to a change in the illite. As illite contains potassium in its natural state, it was not affected by the treatment; hence no change in its cesium sorption was anticipated. (AEC Activity 4352)

Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes into Deep Wells. — Complexing agents, such as citric acid and tartaric acid, may be added to fuel reprocessing wastes to keep in solution iron and other ions normally precipitated by neutralization. Unconcentrated Purex waste (1 M HNO_3, 0.05 M Fe) was complexed by the addition of 112 g of citric acid per liter of waste. Two liters of solution treated in this way was neutralized to a pH of 7 and diluted to 18 liters without precipitation. The addition of 12 g of tartaric acid per liter of Purex waste achieved the same result at a cost of approximately $0.05 per gallon, as compared with $0.28 per gallon for citric acid. Tests with smaller volumes of waste indicate that the amount of citric acid may be reduced.

Neutralized and diluted Purex waste was run through a 10-ft-long glass column packed with a quartz and limestone sand mixture without plugging when the waste contained either citric or tartaric acid as a complexing agent. There was no apparent reaction between the sand mixture and the waste, and the permeability of the sand remained unchanged. The treated waste solution after passing through the column remained clear, forming no precipitate. Dilute, partially neutralized (pH 5) waste solution containing citric acid passed through the column without precipitation of iron, although some carbon dioxide was formed. Dilute, acid Purex waste containing no complexing agents tended to plug the sand mixture in the column both by copious evolution of CO_2 and by precipitation of ferric hydroxide in the pores. (AEC Activity 4352)

Waste Disposal Research and Engineering: Disposal of High-Level Radioactive Wastes into Salt Formations. — Preliminary laboratory investigations of the production of gas from mixtures of synthetic waste solution and rock salt were completed. The rate of gas production for synthetic acid aluminum nitrate solutions and Purex waste solutions, of varying acidity and saturated with salt, was found to be less than 10 cc (STP) per liter of solution per day. However, when the same solutions were exposed to 1.0×10^8 r of Co^{60} gamma radiation, gas generation was increased by a factor of 100. At the dose rate employed, 3.5×10^6 r/hr, neutralized waste solutions saturated with salt produced the greatest amount of gas. Subsequent experimentation indicated that the $Fe(OH)_3$ precipitate in the neutralized waste solution serves to increase gas production.

Quantitative analysis of the gases produced in the solutions during gamma radiation indicated that O_2 and H_2 were the two major gaseous products, with trace amounts of Cl_2, NO_2Cl, and NO_2. (AEC Activity 4352)

Waste Treatment: Sulfex Waste. — The composition of the condensate derived from the evaporating or drying of Sulfex waste varied with the rate of heating and the maximum temperature attained. For the drying step the amount of activity per unit volume of condensate increased with the amount of condensate collected. Two synthetic condensates (1.5×10^4 beta counts/min/ml and 4×10^3 gamma counts/min/ml) were passed through a 16-in. bed of Dowex 50-X8 at 100 gal/hr/ft^2; neither approached breakthrough after the passage of 75 and 100 column volumes. The activity was stripped from the bed by 9 to 11 column volumes of 6 M H_2SO_4. The most attractive method for reducing Sulfex waste to solid form involved evaporating the acid waste to 70 to 80% of its original volume, neutralizing with caustic, then evaporating to dryness at 600°C. Most of the fission products evolved during the drying step were retained in the condensate; the rest were caught in a packed-bed scrubber. No activity was found in caustic bubblers nor in a charcoal trap cooled with liquid nitrogen, which were downstream from the scrubber. The use of either 2% Carbopol 934 (a water-soluble carboxy vinyl polymer emulsifying agent) or 8% Orzan A (an ammonium lignin sulfonate sequestering and binding agent) yielded hard, homogeneous solids which formed without bumping or spattering. The results of all the work to date on Sulfex waste were used to draw up a flowsheet representing a reasonable method of processing for the activity level studied. (AEC Activity 4352)

Waste Treatment: Ion Exchange Methods. — In scouting studies of waste treatment by ion exchange, Decalso, Duolite C-3, and Duolite S-30 showed volumetric distribution coefficients of 102, 25, and 26, respectively, for cesium in neutralized Purex waste. In acidic Darex waste, Duolite S-10, Duolite A-40, and Dowex 21K showed distribution coefficients for plutonium of 15, 103, and 81, respectively. (AEC Activity 4352)

Waste Treatment: Darex Waste. — In studies of the destruction of nitrate ion in acidic Darex waste, it was found that the methanol in formalin (37% formaldehyde, 15% methanol, in water) was responsible for 25 to 50% of the total destruction of nitrate. With formaldehyde only (prepared by refluxing paraformaldehyde in water), approximately 1.5 moles of HNO_3 were destroyed per mole of HCHO. At acid concentrations >1 M, the off-gases are primarily NO_2 and N_2O_4. At lower acid concentrations they are primarily NO and N_2O.

A sample from one of the resin beds of the APPR-1 demineralizer was eluted with sea water, nitric acid, and aqua regia. The sea water eluted most of the activity in the resin, but the activity associated with solid corrosion products filtered out by the resin could be eluted only by aqua regia. (AEC Activity 4352)

ISOTOPE PRODUCTION

Radioisotope and Stable Isotope Production. — The largest single shipment of Co^{60} (25,000 curies) was made. Additional C^{14} with isotopic purity greater than 70% was separated. The first irradiations for radioisotope production were made in the ORR; the short-half-life products had specific activities exceeding those previously obtained in other ORNL reactors by a factor of 4.

Calutron separations of the isotopes of copper, hafnium, and lutetium were completed. Ion currents equivalent to 100 g of copper, 8 g of hafnium, and 48 g of lutetium were monitored in the collectors. This was the first calutron separation of lutetium, and emphasis was placed upon achieving high isotopic enrichment. Approximately 1.25 g of 70% Lu^{176} and 41 g of 99.95% Lu^{175} were produced; isotopic enrichment factors were 90 and 51, respectively.

Separation of K^{40} from KCl enriched in K^{40} to 0.19% by reactor irradiation is in progress. Alcohol cooled to -70°C is being used to reduce product contamination by KCl vapors. Ion currents equivalent to 19.5 g of total potassium have been monitored.

The separation of calcium isotopes to replenish the depleted supply of Ca^{46} and Ca^{48} is in progress with two-arc alpha equipment. Estimated total collection to date is 95 g.

Charge materials supplied for isotope separation were $LuCl_3$ (fused), $K^{40}Cl$ (fused), $HfCl_4$, and Ca metal.

Twenty kilograms of radiogenic lead (88.3% Pb^{206}) was converted to anhydrous $PbCl_2$ for a future collection of high-purity Pb^{206}.

Numerous chemical conversions were performed in the special services program. Examples of some conversions performed are $Sm_2^{147}O_3$ to anhydrous $SmBr_3$, $K^{39}Cl$ and $K^{41}Cl$ to KI, $Hg^{202}(NO_3)_2$ to HgO, Li^6 metal to Li_2CO_3, $Li_2^7CO_3$ to LiF, and B^{10} element to 270-mesh H_3BO_3.

New lots of Ag, V, Lu, Yb, and Cu isotopes were refined during the period. New lots of V (two samples), Yb, and Lu and returned lots of Ba, Gd, Eu, and Sr isotopes were prepared for inventory following chemical recovery and refinement.

Preliminary samples of Lu^{176}, K^{40}, and K^{41} were recovered and partially refined to obtain "in-run" data.

Radioisotope Research and Development. — Cerium-144 was separated from a Redox fission product waste solution and converted to 38.6 g of CeO_2 containing 770 curies of Ce^{144} activity. The CeO_2 was pressed into pellet form and sintered at 1500°C. The sintered pellets had an average density of 5.5 g/cc and appeared to be suitable for the fabrication of large radiation sources.

The thermal energy evolved in the radioactive decay of Ce^{144} in equilibrium with its daughter product, Pr^{144}, was measured calorimetrically. The measured value of 0.0075 w per curie of Ce^{144} is 7% greater than the value calculated from data taken from ORNL-2127 (part I, vol I, p 37).

Fission product rare gases concentrated from uranium dissolver off-gas contain significant quantities of nitrogen oxide gases. The nitrogen oxides absorb along with the rare gases on carbon chromatographic columns used to separate and purify the rare gases. A procedure has been perfected for the conversion of nitrogen oxides to N_2 and CO by passing the mixed gases through an incandescent carbon column operating at 1000°C. Nitrogen and CO do not adsorb on the chromatographic column; therefore the capacity of the system for the rare gases is substantially increased.

Fission Product Pilot Plant operation consists in the separation of fission products from Redox waste solutions. The first purified fission product prepared in the plant was 3,000 curies of Cs^{137}. (AEC Activity 5100)

Stable Isotope Development. — The extension of high-purity techniques into copper separations yielded an estimated 100 g of material. Mass analysis of a 5-g sample of Cu^{65} indicated that the purity achieved was 99.70% Cu^{65}, which corresponds to an enrichment factor of 740. The best Cu^{65} produced previously was 99.2% pure, which corresponded to an enrichment factor of 275.

Fabrication of the grid-type ion source to be used in evaluating an inhomogeneous field separator is estimated to be 75% complete. (AEC Activity 5100)

PHYSICS

High-Voltage Experimental Program. — A new ion source, called the Duo Plasmatron (M. von Ardenne, Tabellen der Elektronenphysik, Ionenphysik und Übermikroskopie, Berlin, 1956), shows some promise as a source of H_1^- and He^{++} ions in addition to its remarkable properties as a source of H_1^+ and He^+. A miniature version of the original 1-amp proton source has been developed with a proton current of 10 ma for use in pulsed accelerator work. With the extremely high proton source-brilliance now available, the current densities are limited entirely by space charge forces even at gradients above the vacuum break-down strength for carefully polished electrodes. For protons and He^+ ions, the gas efficiency is very nearly 100%.

A current of 11 μa of H_1^- ions was observed by simply reversing the extractor polarity. The size of the focal spot obtained was 1 mm. At the same time, electron focal-spot size was $\ll$1 mm, and electron current was 5 ma.

With helium operation, singly charged ion performance was similar to proton performance. Doubly charged ion yields of 5 μa have been observed with

a gas consumption less than 0.5 atmospheric cubic centimeter per hour. (AEC Activity 5220)

The 86-Inch Cyclotron: Nuclear Physics. — Measurements were made of the energy distribution of emitted protons from nuclear reactions induced by protons of various energies between 11 and 23 Mev in elements of atomic number 22 to 30. At intermediate energies the spectrum shows two maxima; there is strong evidence that the low-energy maximum is due either to "second protons" or to simultaneous emission of two particles. Measurements of nuclear temperatures at 18 Mev are strongly distorted by these effects. At lower bombarding energies, the low-energy contribution may be subtracted, and distortion of the high-energy part of the spectra by direct ejection is not large, so that the statistical theory of nuclear reactions may be studied. Values obtained for the level-density parameter a are independent of the energy of the emitted particle, independent of bombarding energy, slowly varying with atomic weight, and consistent with values obtained from neutron-induced reactions. But when these values are used to calculate cross sections for (n,p) and (p,p') reactions, it is found that protons are emitted with excessive probability in the latter. Some possible explanations for this are under consideration.

The earlier survey of (p,α) reactions is being extended by observing energy spectra of the alpha particles emitted from targets of a large range of atomic number and for incident proton energies ranging from 23 Mev down to 13 Mev. The peaks of the alpha energy distributions are found to occur at the same energy over the entire range of bombarding proton energy. Excitation functions for (p,α) reactions are also determined; the sensitivity to Q value is observed and is smaller than the usual theories predict.

Angular distributions of 23-Mev protons elastically scattered from U^{235} and U^{238} were measured; a small difference in scattering cross sections was observed. If this is due to an isotopic effect, it will be more pronounced for targets of lower atomic number. This is being checked by measurements of the proton elastic scattering cross sections for separated nickel isotopes over an angular range of 10 to 170 deg. (AEC Activity 5220)

The 86-Inch Cyclotron: Applied Physics. — The isotopes F^{18}, As^{74}, Sr^{85}, Pm^{148}, and Bi^{206} were produced in service irradiations for outside customers; F^{18}, Tc^{95m}, and Rh^{102} were produced for other ORNL divisions. A total of 330 mc of As^{74} was produced in three irradiations. (AEC Activity 5220)

Spectroscopy Research Laboratory. — The first breakthrough in assigning atomic energy levels to the rare-earth element erbium, Z = 68, was made during this period. Zeeman data of Lindner and Davis (J. Opt. Soc. Am. 48, 542, 1958) and wavelength data of Gatterer and Junkes (Spektren der Seltenen Erden, Citta del Vaticano, 1945) have been used in obtaining a preliminary set of energy levels. Two levels from $4f^{12}6s\ ^{4,2}H$ and two levels from $4f^{12}5d\ ^{4}G$ plus numerous higher odd-parity levels have been identified, the $f^{12}s$ pair being

approximately 0.6 ev more tightly bound than the $f^{12}d$ pair. The low levels are as follows:

Level (cm^{-1})	Probable Identification	J	Observed g
0.0	$4f^{12}6s\ ^4H$	13/2	1.225
440.4	$4f^{12}6s\ ^2H$	11/2	1.092
5132.7	$4f^{12}5d\ ^4G$	9/2	1.234
5403.9	$4f^{12}5d\ ^4G$	7/2	1.038

Measurements of the $J = 0 \longrightarrow 1$ and $J = 1 \longrightarrow 2$ transitions in the microwave spectra of the isotopic vinyl fluoride molecules $C^{12}H_2C^{12}HF$, $C^{13}H_2C^{12}HF$, and $C^{12}H_2C^{13}HF$ have been used to evaluate the structure of the molecule. Parameters obtained from the spectra are b = 10636.83, 10295.26, and 10635.02 Mc, and c = 9118.18, 8859.05, and 9082.78 Mc, respectively. Moments were computed for 1134 assumed structures and were used to obtain derivatives of the moments with five structural parameters. Least-squares determinations gave the following structure: r(CC) = 1.337 ± 0.002 Å, r(CF) = 1.343 ± 0.002 Å, ∠CCF = 121.0 ± 0.2 deg, $r(C_1H)$ = 1.075 ± 0.005 Å, and $r(C_2H)$ = 1.080 ± 0.005 Å. Angle CCH was assumed to be 120 deg. Measurements of the Stark effect of $C^{12}H_2C^{12}HF$ gave 1.40 ± 0.05 Debye units for the resultant dipole moment. (AEC Activity 5230)

Electronuclear Machines. — A fixed-frequency electron cyclotron, Analogue II, is being designed as an operating model of the proposed 850-Mev AVF proton cyclotron. Design of the eight-sector spiral magnetic field was corrected to eliminate variations in the spiral angle which caused undesirable fluctuation in the axial focusing properties of the field. The latest calculations show that smoother focusing properties are obtained and that the number of turns per coil can be reduced from 4 to 1. All coils in the electron model are air-cored.

Coil fabrication and cooling methods were tested. The presently favored technique consists in bonding a rectangular copper conductor directly to a heavily anodized water-cooled aluminum plate. The anodized layer, approximately 0.0002 in. thick, provides electrical insulation. Efficient heat transfer and close mechanical tolerances are achieved with this system. Tests indicate that current densities as high as 30,000 amp/in.^{2} should be feasible. The strict mechanical tolerances (± 0.001 in.) are to be maintained by locating the conductor on the aluminum plate with a master jig during bonding. Placing the conductor along circular arcs rather than along a series of straight-line segments is expected to largely eliminate field errors caused by deviations from the calculated geometry at corners. A program is being written for the Oracle to compute the magnetic field from circular arc conductors of finite cross section with arbitrary radius, orientation, length, and location.

The beam deflection system requires accurate regulation of the r-f accelerating voltage (116 Mc/sec). Precisely regulated d-c power supplies for this system have been completed and are being tested. (AEC Activity 5240)

The 48-Inch Cyclotron. — A site for the Oak Ridge General Research Accelerator has been tentatively selected in the X-10 area. This site is 1350 ft "east" of Building 4500; the front of the GRA building will be aligned with the front of Building 4500. Core drillings were made, and the suitability of the site is being established.

The basic magnet design, as obtained by extensive model tests, is now firm. The threefold azimuthally varying magnetic field is designed to provide suitable focusing forces to permit the fixed-frequency acceleration of protons to 75 Mev. With a wide range of control of average field strength and of field configuration, and with a variable r-f system, it will be possible to accelerate protons to all energies up to 75 Mev and to accelerate heavier particles (helium, carbon, nitrogen, oxygen, neon, and argon) to variable energies up to approximately 100 Mev. The azimuthal configuration of the field is enhanced by means of three pairs of sector coils in the pole pieces; additional focusing is obtained by a small amount of spiral in these coils.

The building design for the preliminary proposal for the GRA is complete. The building will have outside dimensions of 120 x 210 ft. A shielded cyclotron room and two shielded experimental areas occupy the high bay area; in the laboratory area there are two floor levels and a basement. (AEC Activity 5240)

Special Separations. — Calutron separation of uranium included four runs for U^{238} and two runs for U^{236}; equipment was assembled and vacuum-checked for use in a U^{233} separation. A 26-hr separation run to separate Pu^{241} yielded 20.5 mg of 90.6% product. Processing of separated U^{235} was completed, and final salvage from the recycle is 90% complete. An 800-g batch of U^{233} was received from the Chemical Technology Division, and a 250-g portion of this material was converted to UCl_4 calutron feed. Two batches of green salt were prepared as part of a coordinated program to provide metal foils required by Commission users. A 7-g amount of americium was received into the research pool. Heavy-element isotope shipments included uranium (13), plutonium (4), and americium (1). (AEC Activity 5250)

CHEMISTRY

Raw Materials Research and Development: Dapex Process. — Dibutylphosphoric acid, a radiation and hydrolysis degradation product of tributyl phosphate, precipitates thorium, and the precipitate severely impairs radiochemical solvent extraction operations. Other dialkylphosphoric acids, di(2-methylbutyl)-,

di(2-ethylbutyl)-, di(2-methylpentyl)-, and di(2-ethylhexyl)-, show no thorium precipitation under conditions giving copious precipitation with dibutylphosphoric acid (0.1 M reagent in Amsco 125-82 contacting 0.002 M thorium in 2 M HNO_3). This suggests use of the corresponding trialkyl phosphates as alternates for tributyl phosphate in solvent extraction processes involving thorium.

The stripping of Dapex extracts with ammonium carbonate may show appreciable advantage over the sodium carbonate stripping method from the standpoint of reagent costs and product composition (a sodium-free product is obtained in the former case). The strip solution is recycled in order to precipitate the uranium as ammonium uranyl tricarbonate, which has good settling and filtration characteristics and has shown no tendency to cause emulsions in the stripping system. This stripping method has also shown promise for recovering uranium from amine extracts.

The scaleup methods for mixer-settlers that were developed from a unit-operations study of mixing and phase separation were confirmed in the Dapex extraction circuits at Kerr-McGee Oil Industries, Shiprock, New Mexico, and Climax Uranium Company, Grand Junction, Colo. The plant data support the validity of the scaleup of geometrically similar mixers by the use of constant power input per unit mixer volume for mixers 6 to 36 in. in diameter. The phase separation in plant settlers 7 and 16 ft in diameter agrees with that predicted from small units by the use of constant aqueous flow rate per unit settler area. (AEC Activity 5310)

Raw Materials Research and Development: Amine Extraction. — The extraction coefficients $E_a^o(U)$ for the extraction of uranyl sulfate by benzene solutions of di-n-decylamine sulfate (DDAS) increased markedly on slow re-equilibration through a quiescent interface after equilibration by the usual violent shaking. The coefficient increased by a factor of ~5 at 0.006 M and by a factor of ~1.5 at 0.05 M uncomplexed DDAS. The slope of the plot of log $E_a^o(U)$ vs log M (DDAS) was nearly zero for the data taken after the slow equilibrations, in contrast with the usual slope of ~1 after violent shaking. Zero slope is consistent with the observed existence of DDAS as aggregates and of the DDAS—uranyl sulfate complex as a monomer in benzene (ORNL-2506). These results suggest that the lower coefficients and unit slope ordinarily found are caused by the formation of metastable species in the extractant phase during violent mixing. (AEC Activity 5310)

Raw Materials Research and Development: Rare Earths. — The rare earths were precipitated from monazite sulfate liquors by dissolving sodium chloride (60 to 80 g/liter) or sodium sulfate in the liquor after the removal of thorium and uranium by amine extraction. The product was principally the sodium rare-earth sulfate double salt. (AEC Activity 5310)

Chemical Engineering Research. — The interfacial area produced in a small turbine mixer was measured for a variety of conditions by means of the (α,n) technique. Specific interfacial areas as high as 500 cm^2 per cc of mixture were measured between a liquid fluorocarbon and a dilute aqueous polonium solution. For given mixing conditions, the specific interfacial area depended almost linearly on the volume fraction of the dispersed phase. (AEC Activity 5310)

High-Temperature and Structural Chemistry. — Observations on the transport, in an electrical field, of the color carriers existing in solutions of metals in their molten salts afford one of various ways of attacking the problem of the molecular and electronic structure of such solutions. Apparatus was designed and assembled in which the motion of a boundary formed within a porous alumina plug between a colored solution of a metal in a molten salt and the pure molten salt could be observed visually. In a test of the equipment, the color of a solution of cadmium metal in molten cadmium chloride was seen to be transported toward the negative electrode, in agreement with observations by others. The color existing in a solution of sodium in sodium bormide was found to move in the opposite direction (toward the positive electrode), indicating, in agreement with conductivity measurements, the presence of negatively charged, colored carriers ("trapped" but readily mobilized electrons). (AEC Activity 5330)

Molten-Salt Thermodynamics. — The vapor pressure of solid NiF_2, as measured between 1075 and 1285°C by a method which gives total pressures, can be represented by $\log P = 11.385 - 14{,}860/T$, where P is the vapor pressure in millimeters of mercury, and T is the absolute temperature. Extrapolation to permit comparison with sparse data in the literature, obtained from effusion and transpiration studies at lower temperatures, indicates that NiF_2 is not associated in the vapor phase. The same vapor pressure was obtained from an intimate mixture of NiF_2 and nickel as was obtained from NiF_2 contained in a platinum crucible; hence there was no evidence of a proposed reaction $NiF_2 + Ni \rightleftharpoons 2\ NiF$. Also, NiF_2 had not melted at 1285°C, which shows that the melting point is higher than previously estimated. (AEC Activity 5330)

Analytical Chemistry Research. — In a continuation of the investigations on the analytical applications of tri-n-octylphosphine oxide (TOPO), this reagent was found to be useful for extracting cerium(IV) from sulfuric acid solutions. By the use of this material it appears that cerium(IV) can be separated from other rare-earth elements and thorium. In solutions of 0.5 to 7 M H_2SO_4, the extraction coefficient, E_a^o, exceeds 1; it reaches a maximum value of 3.6 for solutions of 3 to 4 M H_2SO_4. Cerium(III) cannot be extracted by 0.1 M TOPO in cyclohexane from strongly acidic sulfate solutions.

In the development of a spectrophotometer suitable for use with molten LiF-NaF-KF and other fluoride salt systems, both reflectance and transmittance-

cell instruments as well as a pendant-drop method are being evaluated. No metal completely inpervious to attack by fluoride salt mixtures has, as yet, been found suitable for use as a mirror in the reflectance instrument; however, platinum is the best of the metals tested. Crystalline MgO, which does not appear to react with pure molten LiF-NaF-KF, offers some promise for use in the construction of a transmittance cell for spectrophotometric measurements above 300 mμ. In the pendant-drop method, the molten mixture can be suspended on a 1-cm platinum ring. It appears feasible to devise suitable optics for the spectrophotometric measurement of the pendant drop; however, measurement of the thickness of the drop poses a major problem.

METALLURGY

Fundamental Physicometallurgical Research. — New investigations of the Zr-Ga phase diagram show that a eutectic occurs in this system between 15 and 20 at. % Ga and that the first intermediate phase is Zr_3Ga, in good agreement with recent investigations of Anderko. The Zr_3Ga phase is very hard (Vickers hardness number of 700 $\pm$ 50). Gallium and aluminum are similar in valence and atomic size, and yet gallium is found to be relatively insoluble in beta zirconium, as compared with aluminum: 5% Ga vs 16% Al at 1100°C. This suggests tighter binding in Zr_3Ga than in Zr_3Al.

Measurements of the vapor pressure of cadmium over Zr-Cd alloys (2 to 10% Cd) continue to show negative heats and excess entropies of alloy formation in the cubic form of zirconium and over a portion of the temperature range for the hexagonal form. Positive values for the heat and the excess entropy of formation appear in the alpha phase just below the allotropic transition. A more complete study of equilibrium times was started to confirm the latter result.

The specific heat of one of the Zr-In alloys was remeasured at liquid-helium temperatures as a test of reproducibility. The coefficients of the electronic and lattice terms of the specific heat as evaluated from the two experiments were found to agree within 0.8%.

A study of the change in preferred orientation in uranium rod during creep testing at 500°C is being made. Preliminary analysis of the data indicates that the tension texture in uranium is not significantly influenced by the radial compressive stresses imposed during rolling, drawing, or swaging. The observed changes in orientation during creep can be explained on the basis of slip and twinning, and indicate that such deformation mechanisms as grain-boundary flow and cell formation are of minor importance. Deformation twinning probably occurs to a much greater extent than was believed previously. (AEC Activity 5420)

Microstresses in Metal Crystals. — A diffraction theory for very thin imperfect surface oxides on copper single crystals has been developed. A preliminary comparison with experiment appears encouraging. If more precise measurements of the contours of the oxide diffraction maxima confirm the correctness of the theory, new information on the structure of such films may result.

Single-crystal specimen preparation of Cu_3Au for a radiation damage study continues. It was found that to avoid gross segregation very-long-time homogenization anneals just below the melting temperature are necessary.

Diffuse scattering measurements on a single crystal of approximate composition Cu_6Al have been completed. A Fourier transform of the diffuse scattering will provide short-range-order parameters. The scattering distribution indicates the presence of an unusual short-range-order arrangement. (AEC Activity 5420)

Ceramics Research. — A quantity of highly sinterable LOH grade BeO was received from Brush Beryllium Company for experimentation. This material is derived from the hydroxide and calcined at low temperatues ($<1000^{\circ}C$) to maintain a very fine particle size (approximately 0.1 μ). Densities of 94 to 98% of theoretical were obtained by hot-pressing at temperatures of 1050, 1150, and $1250^{\circ}C$. This high densification was attributed partly to some impurity which was hygroscopic at $1050^{\circ}C$. Material purified with HCl was hot-pressed to densities of 68, 86, and 97%, respectively, at these temperatures. (AEC Activity 5420)

PROGRAM 6000 — BIOLOGY AND MEDICINE

BIOPHYSICS

Radiation Dosimetry Programs. — Gershman (zhur. Eksptl. i Teoret. Fiz., 1953) has discussed certain disperson relations in a plasma at low temperatures. This report is concerned with a study of a modification of the general dispersion relation (Eq. 24',p 460 of Gershman's paper). A solution to this problem would involve expressing k, the wave number, as a function of w, the frequency.

General results so far obtained concern the nature of a solution. It has been shown that if k(w) is a function satisfying the dispersion relation, then the negative conjugate of this function also satisfies it. Since it can be shown that there are no real solutions, all waves are either damped or amplified. Any continuous solution of the dispersion relation must be zero when

w is zero, and perturbed forms of the general equation suggest that all solutions have an infinite derivative at this point.

Numerical results include an evaluation of k when w is equal to the plasma frequency, which is the case of ionic resonance. In this case the effect of the electrons is neglected. Values of k at resonance have been expressed as a function of the temperature, the number of electrons, and the strength of the magnetic field. The derivative of k with respect to w at resonance has also been investigated. A table of values for k which are purely imaginary has been computed for w between zero and the plasma frequency. (AEC Activity 6140)

Waste Disposal Research: Soil Disposal of Intermediate-Level Radioactive Wastes. — Conasauga shale has demonstrated its ability to remove Cs^{137} from waste released to the pit system. Sodium and nitrate are the primary chemical constituents in the waste, averaging 0.4 mole/liter.

The affinity of the shale for cesium is due to the illitic type of clay mineral that comprises a large fraction of the shale. Successive clay mineral lattices of illite are stacked with open hexagonal networks of superimposed oxygen atoms. The void created can accomodate ions with a radius of $(\sqrt{5} - 1)$ times the radius of the surrounding oxygen atoms, or 1.64 Å, without disrupting the lattice. Cations of this approximate size can move into these exchange sites and become tightly bonded (Soil Sci. Soc. Amer. Proc. 4, 150, 1939). Cesium, with an ionic radius of 1.65 Å, is tightly held by these exchange sites. However, K^+ (r = 1.33 Å), NH_4^+ (r = 1.42 Å), and Rb^+ (r = 1.48 Å) effectively compete with cesium for these sites. Smaller ions, such as Na^+ or Ca^{++}, are less tightly held since they form less efficient contact with the surrounding oxygen atoms and consequently are not as competitive as K^+, NH_4^+, and Rb^+ for the available exchange sites.

The capacity of the shale for retaining Cs from an 0.18 M and a 0.50 M $NaNO_3$ solution containing the equivalent of 0.66 μc of Cs per milliliter was found to be 670 and 282 μc/g, respectively. No cesium was retained by the shale when a 0.50 M NH_4NO_3 carrier solution was employed. (AEC Activity 6440)

Ecological Studies. — The rate of release of assimilated radionuclides from fallen leaves is an important consideration in the biological cycling of radionuclides in waste disposal areas. Preliminary attempts to evaluate this rate of release were made with pine-needle litter collected in the vicinity of the ORNL waste pits. Concentrations of nuclides (principally Ru^{106}) were too low (2×10^{-3} to 2×10^{-4} μc/g, dry weight) to be traced through soil and biological materials with ordinary counting facilities. Therefore 5.5 mc of Ru^{106} was injected into a pine (*Pinus virginiana* Mill); one month later, needles were collected for use as experimental materials. The needles averaged 4×10^{-2} μc per gram of dry weight. Removal of the ruthenium by water

was found to be negligible; leaching of the needles with deionized water removed less than 3% of the activity. Ground needles lost about 5% of their activity when leached. By contrast, leaching of ground needles with a 1 N NaCl solution removed about 25% of the activity. Leachings of naturally contaminated needles from trees in the vicinity of the waste pits produced similar results. (AEC Activity 6480)

PROGRAM 8000 — WORK FOR OTHERS

Nuclear Technology Education for University Faculty Members. — Three visiting faculty members finished their six-month assignments at ORSORT and returned to their respective universities: Kenneth Bell, Case Institute of Technology; William Simpson, Georgia Institute of Technology; John Wethington, University of Florida. (AEC Activity 8736.4)

Professional Health Physics Training Program. — Twenty-four AEC fellowship students, one civilian from Watertown Arsenal, and one Air Force trainee completed their training in Applied Health Physics at the Laboratory in August. One Air Force officer, who received his academic training at the University of Rochester, completed the course with the group at ORNL.

Of the twelve students who received extensions to their fellowships, ten are remaining in the Health Physics Division to do research work for their M.S. degrees. Of the two remaining students, who were offered extensions and who took their field training at Idaho Falls, one declined because of a desirable military assignment, and the other will do his research work at Vanderbilt University or Idaho Falls, Idaho. (AEC Activity 8780)

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VISITORS

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

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121. R. B. Lindauer
122. R. S. Livingston
123. R. N. Lyon
124. H. G. MacPherson
125. F. C. Maienschein
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129. J. R. McNally
130. E. C. Miller
131. K. Z. Morgan
132. E. J. Murphy
133. C. E. Normand
134. G. W. Parker
135. M. E. Ramsey
136. P. M. Reyling
137. L. P. Riordan
138. A. F. Rupp
139. H. C. Savage
140. H. W. Savage
141. A. W. Savolainen
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143. E. D. Shipley
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156. B. S. Weaver
157. A. M. Weinberg
158. J. C. White
159. C. E. Winters
160. H. P. Yockey
161. W. Zobel
162-163. Central Research Library
164. Health Physics Library
165. ORNL—Y-12 Technical Library Document Reference Section
166. R.E.E.D. Library
167-432. Laboratory Records Department
433. Laboratory Records ORNL, R.C.