CENTRAL RESEARCH LIBRARY
DOCUMENT COLLECTION

[redacted]

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES
3 4456 0350601 6

ORNL-2487
Progress Cy. 164

Classification Cancelled
Or Changed To
By Authority Of RDO 9.13.71
By E. Gellberg Date 11.12.71

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

FEBRUARY 1958

CENTRAL RESEARCH LIBRARY
DOCUMENT COLLECTION
LIBRARY LOAN COPY
DO NOT TRANSFER TO ANOTHER PERSON
If you wish someone else to see this document, send in name with document and the library will arrange a loan.

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

[redacted]

[redacted]

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY
UCC
POST OFFICE BOX X
OAK RIDGE, TENNESSEE

March 12, 1958

C. R. Lib

Report No.: ORNL-2487 Copy : 164-165 Classification: Confidential Undoc.

Author(s): W. H. Sullivan

Subject: Oak Ridge National Laboratory Status and Progress Report February 1958

Request compliance with indicated action:

Please make the following change in your copy of the subject report.

CHANGE DATE ISSUED-APRIL 10, 1958 to MARCH 10, 1958

N. T. Bray

N. T. Bray, Supervisor
Laboratory Records Department
Technical Information Division

NTB:WCB:dbt

ORNL:RC

[redacted]

ORNL-2487

Copy No. 164

Contract No. W-7405-eng-26

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

February 1958

Date Issued: ~~APR~~ March 10 1958

OAK RIDGE NATIONAL LABORATORY
Operated by
UNION CARBIDE NUCLEAR COMPANY
Division of Union Carbide Corporation
Post Office Box X
Oak Ridge, Tennessee

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES

3 4456 0350601 6

[redacted]

[redacted]

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

February 1958

Distribution

U. S. Atomic Energy Commission

Oak Ridge Operations Office

1-21. Research and Development Division

22. Technical Information Service Extension

External

23.	Aeroprojects, Inc.	W. B. Tarpley
24-26.	Ames Laboratory: Iowa State College	F. H. Spedding
27-29.	Argonne National Laboratory	N. S. Hilberry
30.	Atomics International (N. Am. Aviation, Inc.)	E. E. Motta
31-32.	Battelle Memorial Institute	H. W. Russell
33-35.	Brookhaven National Laboratory	L. J. Haworth
36-41.	Hanford Atomic Production Operation	A. B. Greninger
42.	Idaho Chemical Processing Plant	C. E. Stephenson
43-44.	Knolls Atomic Power Laboratory	K. R. Van Tassell
45-46.	Los Alamos Scientific Laboratory	N. E. Bradbury
47-48.	National Bureau of Standards	F. L. Mohler
49.	National Lead Company	C. K. McArthur
50.	Nuclear Metals, Inc.	A. R. Kaufmann
51.	Union Carbide Corporation	R. W. McNamee
52.	Union Carbide Nuclear Company	C. E. Center
53.	Union Carbide Nuclear Company	J. L. Gabbard
54.	Union Carbide Nuclear Company	D. M. Lang
55.	Union Carbide Nuclear Company	J. P. Murray
56-57.	University of California Radiation Laboratory	E. O. Lawrence
58.	University of Rochester	H. A. Blair

Internal

59.	E. E. Anderson	65.	P. R. Bell
60.	E. A. Bagley	66.	E. S. Bettis
61.	P. S. Baker	67.	D. S. Billington
62.	C. R. Baldock	68.	R. E. Blanco
63.	L. H. Barker	69.	F. F. Blankenship
64.	S. E. Beall	70.	E. P. Blizard

[redacted]

[redacted]

Distribution (Continued)

71. E. G. Bohlmann
72. C. J. Borkowski
73. G. E. Boyd
74. M. A. Bredig
75. J. C. Bresee
76. R. B. Briggs
77. K. B. Brown
78. F. R. Bruce
79. T. H. J. Burnett
80. A. D. Callihan
81. A. E. Cameron
82. G. H. Cartledge
83. R. A. Charpie
84. C. E. Clifford
85. T. E. Cole
86. E. L. Compere
87. D. D. Cowen
88. J. A. Cox
89. J. H. Crawford
90. F. L. Culler
91. J. E. Cunningham
92. J. S. Drury
93. E. P. Epler
94. D. E. Ferguson
95. J. L. Fowler
96. J. H. Frye
97. W. F. Gauster
98. H. E. Goeller
99. J. C. Griess
100. W. R. Grimes
101. C. S. Harrill
102. J. C. Hart
103. M. R. Hill
104. C. J. Hochanadel
105. A. Hollaender
106. A. S. Householder
107. H. H. Hubbell
108. G. S. Hurst
109. G. H. Jenks
110. R. W. Johnson
111. W. H. Jordan
112. P. R. Kasten
113. G. W. Keilholtz
114. C. P. Keim
115. M. T. Kelley
116. E. M. King
117. K. A. Kraus
118. Laboratory Shift Supervisor (D. Phillips)
119. J. A. Lane
120. T. A. Lincoln
121. S. C. Lind
122. R. B. Lindauer
123. R. S. Livingston
124. R. N. Lyon
125. H. G. MacPherson
126. F. C. Maienschein
127. W. D. Manly
128. L. A. Mann
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
142. H. E. Seagren
143. E. D. Shipley
144. A. Simon
145. O. Sisman
146. J. R. Sites
147. M. J. Skinner
148. A. H. Snell
149. R. W. Stoughton
150. H. Stringfield
151. E. G. Struxness
152. W. H Sullivan
153. C. D. Susano
154. J. A. Swartout
155. E. H. Taylor
156. B. S. Weaver
157. A. M. Weinberg
158. J. C. White
159. [illegible]. E. Winters
160. [illegible] P. Yockey
161. [illegible] Zobel
162-163. Biology Library
164-165. Central Research Library
166. Health Physics Library
167. ORNL-Y-12 Technical Library Document Reference Section
168. R.E.E.[illegible]. Library
169-174. Laboratory Records Department
175. Laboratory Records ORNL, R.C.

[redacted]

[redacted]

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

TABLE OF CONTENTS

	Page
PROGRAM 2000 — SPECIAL NUCLEAR MATERIALS	6
Dissolver Solution Analysis	6
PROGRAM 3000 — WEAPONS	6
Special Separations	6
PROGRAM 4000 — REACTOR DEVELOPMENT	7
HOMOGENEOUS REACTOR PROJECT	7
Homogeneous Reactor Test	7
HRT Chemical Pilot Plant	8
Fuel Processing	8
Thorium Oxide Slurry Development	8
HRP Analytical Chemistry	9
Slurry Component Development and Slurry Testing	9
Slurry Blanket System Development	10
Blanket Engineering Research	10
Core and Blanket Vessel Development	10
Centrifugal Circulating Equipment	11
Miscellaneous Equipment	11
Heat Removal Equipment	12
Gas Handling Equipment	12
Fuel Systems Development	12
Feed Pump Development	12
Gaseous-Fission-Product Disposal	13
CIVILIAN-POWER ADVANCED-TYPE GAS-COOLED REACTOR (GCR)	13
Gas-Cooled Reactor: Design	13
Gas-Cooled Reactor: Metallurgy	14
Gas-Cooled Reactor: Experimental Engineering	15
Gas-Cooled Reactor: Radiation Effects	15
Gas-Cooled Reactor: Instrumentation and Controls	15
MOLTEN-SALT REACTOR (MSR) PROGRAM	15
Molten-Salt Reactor: Design Studies	15
Molten-Salt Reactor: Experimental Engineering	15
Molten-Salt Reactor: Chemistry	16
Molten-Salt Reactor: Metallurgy	17
PACKAGE REACTOR DEVELOPMENT	18
Army Package Power Reactor (APPR-1)	18
Gas-Cooled Reactor Development	18
Maritime Ship Reactor	19
GENERAL REACTOR RESEARCH	20
Basic Reactor Research	20

[redacted]

[redacted]

TABLE OF CONTENTS (Continued)

(Program 4000, Continued) Page

Fuel Element Development 20
Thorex Process 21
Power Reactor Fuel Reprocessing: Mechanical Processing 21
Power Reactor Fuel Reprocessing: Sulfex Process 21
Power Reactor Fuel Reprocessing: Zircex Process 21
Power Reactor Fuel Reprocessing: Phosphex Process 22
Power Reactor Fuel Reprocessing: Ion Exchange 22
Power Reactor Fuel Reprocessing: Darex Process 22
Power Reactor Fuel Reprocessing: Electrolytic Dissolution . . 22
Fused Salt—Fluoride Volatility Process 22
Waste Treatment and Disposal 23
Waste Disposal Research and Engineering:
Disposal of Radioactive Wastes into Geologic Structures . . 23

PROGRAM 5000 — PHYSICAL RESEARCH 24

ISOTOPE PRODUCTION 24
Radioisotope Production 24
Radioisotope Development 25
Stable Isotope Separation 25
Stable Isotope Development 26

PHYSICS 26
High Voltage Program 26
The 86-Inch Cyclotron: Nuclear Physics 26
The 86-Inch Cyclotron: Applied Physics 27
Heavy-Particle Physics 27
Spectroscopy Research Laboratory 27
Electronuclear Machines 28

CHEMISTRY 28
Raw Materials Chemistry 28
Americium Recovery 28
Feed Materials Processing 29
High-Temperature and Structural Chemistry 29
Organic Chemistry 30
Radiation Chemistry 30
Molten Salt Thermodynamics 31
Analytical Chemistry Research 31

METALLURGY 31
Fundamental Physicometallurgical Research 31
Microstresses in Metal Crystals 32
Fundamental Investigations of Radiation Damage in Solids . . . 32

PROGRAM 6000 — BIOLOGY AND MEDICINE 33

BIOPHYSICS 33
Radiation Dosimetry Programs 33
Waste Disposal Research: Soil Disposal of Solid Radioactive
Wastes 34

PROGRAM 8000 — WORK FOR OTHERS 34
Nuclear Technology Education for University Faculty Personnel . 34

MISCELLANEOUS REPORTS . . . [redacted] 35

[redacted]

[redacted]

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

February 1958

This Status and Progress Report summarizes somewhat more than one-half the activities of the Laboratory. Some of the topics are included every month, but many of them are reported on a bimonthly schedule. Program 4700 is covered in a separate report.

PROGRAM 2000 — SPECIAL NUCLEAR MATERIALS

Dissolver Solution Analysis. — The coulometric titration of uranium at controlled potential is being studied with the hope of applying this method to the determination of uranium in various dissolver solutions. This titration has been found to be capable of the high degree of accuracy and precision necessary for such application. The application of citric acid as the supporting electrolyte is being studied because interferences from other dissolver-solution components are minimized with the use of this acid.

A transistorized controlled-potential coulometer has been fabricated by the Instrumentation Group and is being evaluated. Studies of nitric acid interference are nearing completion. Results at this time indicate that milligram quantities of uranium in solutions whose nitric acid concentration is as high as 5 M can be titrated with excellent precision and accuracy. Higher nitric acid concentrations may be tolerated with some increase in titration time. (AEC Activity 2724)

PROGRAM 3000 — WEAPONS

Special Separations. — The U^{234} collection schedule for 1957 was completed in a series of runs ending December 26, 1957; 119 mg averaging 97.5% U^{234} was collected.

Runs for the collection of U^{236} by use of high-resolution equipment yielded approximately 0.400 g of U_3O_8 ranging from 95 to 98.5% in U^{236} content. Three more collection runs at higher outputs yielded 3.3 g of U_3O_8 containing U^{236} in the 93% range.

[redacted]

Approximately 1 kg of UCl_4, assaying 93% U^{235}, has been prepared as charge for the collection of high-purity U^{235}. An additional 2 kg is being prepared for the same series of collections. The equipment for the first of these runs is ready for startup.

One $PuCl_3$ charge was prepared from 60 to 80% range Pu^{240} product collected over the past four years. This material yielded 3 g of $PuCl_3$ for use in making a second-stage Pu^{240} separation, with UCl_3 to support the arc. The equipment for this collection has been installed in the tank and is ready to run.

The 5% Pu cycle is in the salvage phase of recovery for final balance. The 13% Pu cycle is 80% purified to oxalate for the final balance. (AEC Activity 3610)

PROGRAM 4000 — REACTOR DEVELOPMENT

HOMOGENEOUS REACTOR PROJECT

Homogeneous Reactor Test. — Final checking of the shield leak rate at 7-1/2 and 30 psia was completed on February 2. The shield temperature was held constant over a 12-hr period for each test, and rates of 2-1/2 liters/min at 7-1/2 psia and 4 to 4.3 liters/min at 30 psia were established.

Prior to the initiation of power operation of the HRT, the cell was entered to make repairs to the power wiring for the blanket space coolers and the terminal connections of the fuel circulating pump. At the same time, a short circuit in the stator test wiring was found and removed. The shield was again sealed on February 7, and the reactor was brought to criticality that evening. Low-power experiments were conducted from February 8 through February 12, to a maximum power level of approximately 1 Mw. On the evening of February 12, chemical analyses showed that the nickel content of the fuel solution had been rising rapidly during operation and was currently in the region of 1000 ppm. Although this concentration of nickel ion would result from a uniform corrosion of only 0.5 mil, the reactor was diluted and made subcritical for further analysis of the data. During the power run, 33 Mwhr of thermal energy was generated.

After reviewing the data, it was decided to establish a corrosion rate in the absence of fission power and to proceed with the power experiments. On February 17, the core concentration was raised to 6.9 g of U^{235} per kg of D_2O, and the reactor was operated subcritically at 250°C and 1700 psi for further study of the behavior of the system.

On February 18, electrical breakdowns occurred in the power wiring to the blanket space coolers and circulating pump and made it necessary to shut down. Efforts to dry the glass-braid-insulated cables to the disabled equipment were unsuccessful, and it was decided to open the reactor container so that the faulty wiring could be replaced. This work began on February 24 and will require several weeks for completion. (AEC Activity 4103.3)

HRT Chemical Pilot Plant. — Final closure of the processing cell, leak-testing of the cell, and repairs of the leaks were completed. Leak rates were reduced from an initial value of 25 liters/min at 30 psia to 6 liters/min. Most of the leakage was around the two sample housings in the corners of the cell and from conduit seals. No leakage at the welds of the top seal pans could be found. All detectable leaks have been repaired.

Design of facilities to handle waste uranium solutions transferred from the cell was completed. (AEC Activity 4103.1)

Fuel Processing. — Only insignificant amounts of simulated reactor corrosion-product solids traced with Fe^{59} or Zr^{95} and suspended in simulated reactor fuel solution were deposited on stainless steel test pins after several hundred hours at 275 to 300°C.

Prior to testing the UO_4 precipitation technique for decontamination of HRT fuel solution in the hot cells, experiments are being conducted in the laboratory to develop suitable equipment and procedures for such testing. Uranium losses in waste solutions as low as 0.01% have been routinely observed, as have nickel decontamination factors greater than 100. (AEC Activity 4103.1)

Thorium Oxide Slurry Development. — In studies of the effect of chemical and physical variables of the thorium oxalate precipitation step on the properties of oxide produced by thermal decomposition of the oxalate at 1600°C, the average particle sizes of the oxides decreased with decreasing precipitation temperature and increasing reagent concentration. More uniform materials were produced at lower precipitation temperatures (10°C), and the fraction of oversize particles decreased with increasing stirring rate and reagent addition rate. Well-digested, more uniform precipitates showed less localized sintering during firing at 1600°C.

About 1000 lb of 1600°C fired oxide classified by sedimentation to contain 95 wt % particles <5 μ and 180 lb of 1600°C fired oxide classified to contain 95 wt % particles <1 μ were prepared for engineering development studies. Special oxide prepared for classification to other size ranges of interest totaled 600 lb.

In further slurry irradiations in the LITR, a D_2O slurry of 1300°C fired thorium-uranium oxide (0.5 wt % U^{235}) containing 0.05 m MoO_3 gas recombination catalyst was irradiated without apparent damage and with no radiolytic-gas pressure in excess of steam.

Out-of-reactor measurements of the combination rate of stoichiometric D_2-O_2 mixtures in D_2O slurries of 1300°C fired thorium-uranium oxide (0.5% U^{235} prepared from coprecipitated oxalate) containing 1000 g of Th per kg of D_2O showed that the addition of 0.02 to 0.228 m MoO_3 catalyst should maintain the D_2 partial pressure at less than 100 psi at 280°C in LITR fluxes. Heating the slurries without agitation at 280 to 285°C with a small overpressure of O_2 for about 200 hr did not affect the combination rates. (AEC Activity 4103.1)

HRP Analytical Chemistry. — A flame photometric method was devised for the direct determination of silver in uranyl sulfate and other solutions by measurement of the emissivity of the silver at 350 mμ. Silver in concentrations as low as 1 μg/ml can be determined with a coefficient of variation of approximately 10%.

A spectrophotometric method was developed for the determination of microgram amounts of zirconium in slurries of thorium oxide in order to provide a means for evaluating the corrosion rate of Zircaloy components of test loops. One part per million of zirconium can be determined; over the range of 1 to 10 ppm the coefficient of variation is about 3%.

A polarographic method was used to determine uranium in low concentrations which had been added to slurries of thorium oxide during circulation tests. By using the ORNL derivative polarograph, from 200 to 500 ppm of uranium was determined with an average coefficient of variation of about 3%.

A gamma-counting procedure was applied to the determination of thorium in fractions of thorium oxide slurries derived in the measurement of particle-size distribution by sedimentation methods. It is only necessary to determine the ratio of thorium in the settled portion to total thorium in the sample. This ratio can be established by the gamma-count method with an accuracy of 2% in about one-third the time required by the conventional spectrophotometric method which was previously used. (AEC Activity 4103.1)

Slurry Component Development and Slurry Testing. — Run 200-A-14, to determine the degradation characteristics of 1600°C fired oxide when circulated at a concentration of 500 g of Th per kg of H_2O at 300°C and 2000 psig, was terminated after 3787 hr. The mean particle size decreased during the run from 2.5 μ to 1.8 μ. However, nitrogen absorption surface area remained constant at 1.6 m^2/g, and the generalized corrosion rate stayed at 1.2 mils/year. The slurry was easily removed, and no cakes or films were found. The pump scroll liner was heavily eroded near the seal rings, but the other parts showed surprisingly little wear.

Several runs were made in the 100-gpm loops to determine the effect of additives on the caking and sphere-forming tendencies of 800°C fired oxide. Apparently the presence of 1/2 wt % of uranium as UO_3, 1.16 mole % MoO_3, or 2500 ppm sulfate as $Th(SO_4)_2$, singly or combined, is sufficient to prevent formation of spheres or cakes in a slurry which has been observed to form cakes and spheres without additives.

The high-pressure capillary-tube viscometer has been refined to the point where useful rheological data are being obtained at temperature on slurries withdrawn directly from the circulating loops. (AEC Activity 4103.1)

Slurry Blanket System Development. — Run SM-4 was terminated on schedule after 2400 hr of operation. The maximum charged concentration was 720 g of Th per kg of H_2O at 200°C. The circulating concentration generally was lower by 30 to 80 g of Th per kg of H_2O than the charged concentration. Two uranium additions were made in order to trace the interchange of oxide between the circulating stream and the blanket contents. These additions confirmed previous indications that stagnant thoria existed in the bottom of the blanket, on top of the core vessel, and in the pressurizer.

At charged concentrations above 500 g of Th per kg of H_2O, pressure control became difficult owing to accumulation of dense thoria masses in the pressurizer.

The pump was found to be generally in good condition. The titanium impeller showed only slight attack at flow-separation regions. (AEC Activity 4103.1)

Blanket Engineering Research. — Digestion of freshly precipitated thorium oxalate before filtration and firing resulted in ThO_2 with particles which were resistant to degradation. The cake resuspension index of ThO_2 prepared from digested precipitates was lower than that of ThO_2 prepared similarly but without digestion. The addition of $NaAlO_2$ to a flocculated slurry reduced the yield stress significantly. The effect of the $NaAlO_2$ was not altered by autoclaving at 280°C.

It was demonstrated that a hydroclone can be used to effect a size classification of thoria particles. For the material studied, after the second pass, the overflow from the hydroclone contained material in which 98 wt % was less than 2 μ in diameter, while the corresponding value for the original material was 65 wt %. (AEC Activity 4103.1)

Core and Blanket Vessel Development. — Design of a 30-in. low-pressure blanket-vessel model was completed, and procurement and fabrication were initiated. A plastic core vessel with a shroud assembly was purchased. The plastic model will be used to determine the flow behavior and flowrate—pressure-drop relationship in the annulus between the core vessel and the shroud.

Drainage tests of sedimented slurries through a 60-deg-included-angle cone showed no gross retention of sediment, although a slime layer of 1/8- to 1/2-in. thickness remained on the cone wall.

Heat transfer measurements made at room temperature on a 4-ft-dia spherical re-entrant type of core vessel indicate a heat transfer coefficient between wall and fluid of approximately 2000 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot (^{o}F)^{-1}$ at a flow rate of 6000 gpm. (AEC Activity 4103.1)

Centrifugal Circulating Equipment. — The spare HRT fuel and blanket circulating pumps were assembled with new seal-welded thermal barriers and are ready for installation in the reactor when needed.

Detail design was initiated for the conversion of the 400A-2 pump to a 200-gpm slurry circulating pump.

The 300A-1 pump was modified for installation in the new 300A ThO_2 slurry loop. A Koppers floating-bushing shaft seal was installed in the pump for test and evaluation. The loop was completed except for instrumentation, insulation, and shielding erection. Pressurizer flow studies were conducted in a glass mockup of the loop.

Installation of the radiation test stator in the HRT is complete except for minor electrical and instrument work outside the cell. A leak in the stator housing will prevent measurement of hydrogen buildup pressure. The two other test stators were returned to Reliance Electric Company for weld repair.

The low-pressure system of the 4000-gpm Byron Jackson pump loop is being redesigned for operation with ThO_2 slurry. Pressure control will be by a feed and letdown system.

The preliminary layout for the 6000-gpm HRE-3 prototype fuel circulating pump was received from Reliance Electric Company. The major problem involved is that of providing a reliable seal around the thermal barrier. (AEC Activity 4103.1)

Miscellaneous Equipment. — The ability of the thermal-cycle facility to meet the thermal cycle specified for the HRT replacement heat exchanger was demonstrated in tests with an autoclave as the test piece.

Tests of HRT-type flanges made up without bolting ferrules gave leak rates comparable with those obtained when ferrules were used. The higher bolt stresses present at temperature caused no damage.

In cooperation with HRT personnel, a method has been developed for removing most of the light water from a flange which has been opened while submerged.

Equipment was fabricated to investigate various methods of producing purge water from slurry. Methods to be investigated include settling, filtering, and centrifuging.

Both of the titanium-lined pipes being fabricated by Crane Company cracked during a hydrostatic test. Repairs are being attempted. The Watervliet Arsenal will bore the extruded tianium pipe which will be used to construct an all-titanium pressurizer for the HRT mockup. (AEC Activity 4103.1)

Heat Removal Equipment. — A contract has been placed with Griscom-Russell Company for the fabrication of the heat exchanger test facility, which is scheduled for completion in about nine months.

A loop has been designed to test the effect of geometry on slurry heat exchanger design. Plastic models will be used to study settling, plugging, pressure drop, and other problems applicable to heat exchanger design. (AEC Activity 4103.1)

Gas Handling Equipment. — The long-term run of the high-pressure recombiner loop was terminated after 600 hr at 1000 psi and 250°C with a 3% volume fraction of radiolytic gas. Termination was due to caustic-induced stress corrosion of the type 347 stainless steel boiler weld metal. The boiler is being replaced with an Inconel vessel. The recombiner performed satisfactorily throughout the test.

The natural-circulation titanium loop in which uranyl sulfate is boiled at 1500 psi was operated for 1600 hr with no indication of corrosion. The loop will be operated with a solution of higher concentration with two liquid phases present. (AEC Activity 4103.1)

Fuel Systems Development. — The titanium letdown heat exchanger, the 80-kw loop heater, and the 400-gpm pump were installed in the HRT mockup system. Run 22, with simulated HRT fuel, was started to study the corrosion resistance of the titanium heat exchanger. After 340 hr of operation the average corrosion rate, based on increase in soluble nickel, was 0.7 mil/year for the system, compared with 1.3 mils/year for the corresponding period of the previous run with a stainless steel heat exchanger. Since titanium is almost insoluble in fuel solution, it is not possible to follow the attack rate on the titanium from solution analyses. The rate will be determined from examination of the exchanger after the run. (AEC Activity 4103.1)

Feed Pump Development. — One feed pump, containing a 0.019-in. diaphragm and 40-mesh screening, suffered a diaphragm failure after 9000 hr of operation. Heads now being tested have operated for 7899, 7262, 7262, 2491, and 832 hr. Purge-pump heads have operated for 3431, 2281, and 2281 hr. The double-diaphragm pump has operated for 1100 hr at 64 strokes/min, while pumping 2.0 gpm of fuel solution at 2000 psi discharge pressure. The head screened with 20-μ Poroloy exhibited pumping characteristics identical to those of coarse-screened heads. It has operated for 832 hr in the valve test loop, which is considered to impose severe test conditions on the pump screening.

A remote-leg slurry feed pump is being operated at atmospheric pressure to determine its practicality and characteristics. The pump has been operated at flow rates up to 1.2 gpm and with slurry specific gravity up to 2.3. It appears possible to maintain almost pure water in the head while pumping slurry through the check valves.

The slurry check-valve-material test loop was operated with 300- and 670-g/liter slurries. Aluminum oxide trim performed satisfactorily for 311

hr in slurry and for 50 hr in various rinses, while Kennametal K-501 and boron carbide trims were severely damaged in similar rinses in less than 24 hr. A 1/16-in.-ID by 10-ft-long stainless steel capillary letdown tube served for over 500 hr with no performance change and only one case of plugging.

The latest results from Ohio State University tests of diaphragm materials indicate an endurance limit of 62,000 psi in water or uranyl sulfate solution for titanium 110AT, and 68,000 psi in water and 38,000 psi in uranyl sulfate solution for AM-350 stainless steel. (AEC Activity 4103.1)

Gaseous-Fission-Product Disposal. — The charcoal used in the HRT beds was observed to ignite at 280°C in a flowing stream of oxygen. The presence of 5 to 7% water on Linde molecular sieves or HRT charcoal lowered the capacity of the adsorbents for krypton about equally. With 15% water on the molecular sieves their capacity for krypton was only 10% of the capacity of the dry material. (AEC Activity 4103.1A)

CIVILIAN-POWER ADVANCED-TYPE GAS-COOLED REACTOR (GCR)

Gas-Cooled Reactor: Design. — The first phase of the design study of an enriched-fuel gas-cooled graphite-moderated reactor has been completed, and the report requested by the AEC has been written. The principal specifications of the system that were evolved in this study are the following:

Core diameter	30 ft
Core height	20 ft
Thermal power	700 Mw
Helium inlet temperature	450°F
Helium outlet temperature	1000°F
Helium pressure	300 psia
Steam temperature	950°F
Steam pressure	950 psia
Gross electrical output	252 Mw
Net electrical output	225 Mw

The fuel element comprises a cluster of seven 3/4-in.-dia UO_2 rods, canned in type 304 stainless steel 0.020 in. thick. The optimum fuel enrichment is 2 to 2.5%; 2% was chosen for the reference design. An average fuel exposure of 7400 megawatt-days per metric ton was calculated, but some of the fuel may reach twice that exposure.

Operating costs of the plant, including maintenance and fuel costs, were calculated to be 2.6 mills/kwhr; the over-all power cost was estimated to be about 10.5 mills/kwhr at present cost levels. This is to be compared with an average cost of 7.5 to 8 mills/kwhr for modern coal-fired steam plants of similar capacity. These are 1958 construction costs, and they do not include an escalation factor that would have to be applied to the cost of construction completed in 1960. (AEC Activity 4119)

Gas-Cooled Reactor: Metallurgy. — Two static pot tests, as described previously (ORNL-2471, Feb. 7, 1958), are in operation at 1400°F. The first pot contains TSF-grade graphite and has been at temperature for more than 300 hr. The second contains AGOT-grade graphite and has been at temperature for slightly over 200 hr. Both tests are being run in order to evaluate the outgassing properaties of the graphite and the effects of impurities evolved by the graphite on structural metals. Gas analyses obtained thus far for both pots showed a significant increase in the percentages of H_2, CO_2, N_2, and H_2O as the temperature of the pots increased, but the levels of these impurities reached a maximum soon after the pot temperature was stabilized at 1400°F and have decreased significantly with time at temperature. Since the pots contain samples of relatively active metals, for example, zirconium and niobium, it is probable that the decrease in impurities is caused by the scavenging effects of these metals.

In general the following differences are apparent between the outgassing characteristics of the AGOT and TSF grades of graphite. The maximum CO_2 content in the pot with AGOT-grade graphite was a factor of 6 greater than that in the pot containing TSF-grade material. A difference was also apparent in the H_2 and H_2O content of each pot, with the levels of these impurities again being greater in the case of the AGOT-grade graphite. While these differences can possibly be associated with the storage of the graphite samples prior to test, it should be noted that the AGOT samples were stored approximately six months and that the TSF samples were stored close to one year.

Investigation of the creep buckling characteristics of thin-wall tubes has continued with Inconel tubing, which was available and which is similar to the required stainless steel tubing that will not be available for some time. A test of four 3/4-in.-OD, 20-mil-wall Inconel specimens was conducted at 1675°F with an external pressure of 300 psi. The diametral clearances between the tube wall and the inserts (to simulate fuel slugs) were 5, 10, 20, and 100 mils. At the end of 500 hr, the equivalent of a year at 1500°F, the test was discontinued and the specimens were examined. The specimens having diametral clearances of 5 and 10 mils were collapsed evenly around the inserts. However, the specimens with the 20- and 100-mil diametral clearances had buckled in a one-lobe mode. In a previously reported test of 35-mil-wall specimens (ORNL-2471, Feb. 7, 1958), even collapse around the inserts occurred for the specimen having a 20-mil diametral clearance. This indicates that, as the thickness of the tubing wall is decreased, it will be necessary to decrease the diametral clearance between the wall and the inserts if buckling is to be avoided.

Inconel specimens without inserts were subjected to an external pressure of 300 psi at 1500°F for various lengths of time to determine when buckling would occur. At the end of 50 hr no evidence of buckling could be observed; however, a specimen tested for 65 hr completely collapsed. These results seem to indicate that buckling does not occur gradually, but, rather, once buckling starts, that collapse occurs almost instantly. (AEC Activity 4119)

Gas-Cooled Reactor: Experimental Engineering. — One of two thermal-convection systems for testing the compatibility of various materials in flowing helium was completed and placed in operation at 1400°F, and the second was nearly completed. These loops contain 8-ft mockup sections of graphite and fuel-element-cluster assemblies. Design work is under way on a forced-convection test system. (AEC Activity 4119)

Gas-Cooled Reactor: Radiation Effects. — Fabrication of the equipment for irradiation of a stressed fuel element is nearing completion, and some preassembly testing of component parts was started. (AEC Activity 4119)

Gas-Cooled Reactor: Instrumentation and Controls. — The nuclear instrumentation of the GCR was selected. The location of the nuclear detectors permitted the use of standard chambers with standard electronic amplifiers; however, this location limited the nuclear flux available to the chambers. The source strength required to be able to use the standard instruments during startup was computed on the basis of the subcritical multiplication with all the control rods fully inserted and the minimum neutron flux necessary for adequate nuclear information. (AEC Activity 4119)

MOLTEN-SALT REACTOR (MSR) PROGRAM

Molten-Salt Reactor: Design Studies. — Oracle calculations were made with the use of the Sorghum code to show the effect of fuel processing on critical inventory and regeneration ratio. For these calculations it was assumed that the reactor had an 8-ft-dia core, that the U^{235}-enriched fuel contained 1 mole % ThF_4, and that processing was started after 300 days of operation and then continued at the rate of one fuel volume per year. The results of the calculations showed an increase from the original critical inventory of 668 kg to 898 kg in 300 days and a further increase to only 921 kg in 610 days; for the same periods the regeneration ratio changed from 0.62 to 0.53 to 0.52. Thus, in one to two years of operation under these conditions, the critical inventory and the regeneration ratio would reach nearly steady-state values.

A design study of the fuel fill-and-drain system was made. Filling and draining of the fuel circuit will be accomplished by gas pressurization, and there will be no valves in liquid lines. The fill-and-drain tank will consist of forty-eight 20-ft lengths of 12-in.-dia pipe arranged in six vertical banks connected on opposite ends with mitered joints. Preheating will be accomplished with bayonet-type heaters in tubes located axially in the pipes. Removal of afterheat will be accomplished by thermal radiation to banks of water-filled pipes that would normally be dry. (AEC Activity 4103.1B)

Molten-Salt Reactor: Experimental Engineering. — The finish on the gas-sealing surfaces of the freeze-flange joint described previously (ORNL-2458, Jan. 9, 1958) was improved so that cold leakage rates of less than 10^{-7} cm^3/sec of helium were obtained. A second type of joint was constructed that consists

of a tongue-and-groove arrangement in which a seal is formed by melting a silver alloy to form a "cast" ring between the tongue and groove. Both seals were tested through 75 thermal cycles between 1100 and 1300°F, and there was no salt leakage.

A heat transfer test of the mixture $LiF-BeF_2-UF_4$ (62-37-1 mole %, fuel 130) was completed in a small heat exchanger test stand. Data were obtained for heat transfer correlations and a determination of the heat transfer coefficient.

Initial tests of a pump journal bearing and seal lubricated with Dowtherm A at 1200°F gave satisfactory results, and further tests are under way. A modified seal having a split purge-gas flow, in which part of the purge gas is directed downward toward the process fluid and the remainder flows toward the seal, has operated satisfactorily at 1200°F for more than 3200 hr.

A design study was completed of a pump that does not require oil-lubricated bearings. The lower bearing is salt-lubricated; the upper (spherical) bearing is lubricated by the purge gas, and it serves as a seal as well as a bearing.

The design was completed on the in-pile portion of the forced-circulation loop which is to be fabricated of INOR-8 and which is to circulate fuels of interest in an irradiation facility of the MTR. A pump with the required sump volume and extended shaft was assembled and found to have a satisfactorily high critical speed. A test heat exchanger and an electrical heater were fabricated. (AEC Activity 4103.1B)

Molten-Salt Reactor: Chemistry. — Invariant equilibria in the system $LiF-ThF_4-UF_4$ were shown to occur at the composition 58 mole % LiF—23 mole % ThF_4—19 mole % UF_4 at 605°C, at the composition 71.5 mole % LiF—7 mole % ThF_4—21.5 mole % UF_4 at 497°C, and at the composition 72 mole % LiF—1.5 mole % ThF_4—26.5 mole % UF_4 at 488°C.

The solubility of PuF_3 increases from 0.25 to 0.45 mole % in $LiF-BeF_2$ mixtures at 565°C as the BeF_2 concentration is increased from 36 to 48.4 mole %. The saturating phase is PuF_3 in all cases. Melting points of binary mixtures containing appreciably less than 35 mole % BeF_2 are too high to permit solubility measurements. In the ternary mixture $NaF-LiF-BeF_2$ (56-18-26 mole %) the solubility is considerably higher than that in the binary mixtures containing 36 mole % BeF_2; the saturating phase in this ternary solvent is $NaPuF_4$.

The solubility of CeF_3 in $LiF-BeF_2$ mixtures was determined as a function of temperature and solvent composition. The solubility of CeF_3 goes through a minimum at about 63 mole % LiF, and at this solvent composition amounts to only 0.18 mole % CeF_3 at 500°C. This low value is encouraging in view of possibilities of removing rare-earth fission-product fluorides by solid extraction with CeF_3.

Several separations of solutes appear to be possible on the basis of selective precipitation as oxides from LiF-KF (50-50 mole %) with CaO as the precipitating agent. Titration curves with sharp end points were obtained upon addition of CaO to solutions containing UF_4 and CeF_3; the uranium appears to precipitate selectively before the removal of cerium begins. A similar separation of zirconium from cerium was observed.

The solubilities of neon and argon in NaF-KF-LiF (42-11.5-46.5 mole %) were determined. The numerical values, expressed in moles of gas per cubic centimeter of melt per atmosphere at 600°C, are 4.36×10^{-8} for neon and 0.90×10^{-8} for argon. The corresponding values at 800°C are 11.18×10^{-8} and 3.40×10^{-8}.

The solubility of HF in mixtures of NaF-ZrF_4 was determined as a function of temperature, pressure, and solvent composition. The results show a remarkable increase in solubility with increasing concentration of NaF. A change in composition from 45 to 80.5 mole % NaF brings about an increase in Henry's law constant from 0.5×10^{-5} to 4×10^{-5} mole of HF per cubic centimeter of melt per atmosphere at 800°C. The solubility constant increases to about 22×10^{-5} for HF in NaF-KF-LiF (42-11.5-46.5 mole %) at 800°C. (AEC Activity 4103.1B)

Molten-Salt Reactor: Metallurgy. — All the Inconel thermal-convection-loop tests programed in the first phase of the compatibility evaluation of 12 fused-salt mixtures being considered for MSR application were completed. Void formations of a minor nature were observed. Of 12 loops examined, seven had void formation of less than 1 mil. The remaining five had varying depths of attack that approximated 2.5 mils. Hot-leg sections of an Inconel forced-circulation loop which operated for 6300 hr at 1250°F with NaF-ZrF_4-UF_4 showed no measurable attack. Two INOR-8 thermal-convection loops that completed 1000 hr of operation at 1250°F with NaF-BeF_2-ThF_4 (58-35-7 mole %) and LiF-NaF-KF, respectively, showed no measurable attack.

Seesaw-furnace tests and thermal-convection-loop tests of Inconel-graphite-salt systems at 1300°F showed no detectable carburization of the Inconel in times up to 1000 hr. Both Inconel and INOR-8 were carburized in static pot tests at 1500°F, with the INOR-8 being carburized more extensively than the Inconel. The Inconel was carburized to a depth of 10 mils in 100 hr, and a 40-mil-thick specimen of INOR-8 was completely carburized. The gold-bearing brazing alloy being considered for use in fabricating INOR-8 components was found to have good corrosion resistance to LiF-BeF_2-UF_4 at 1200°F in 500-hr seesaw-furnace tests.

The Young's modulus for INOR-8 was determined by sonic methods from room temperature to 1900°F. The effect of aging of INOR-8 for 500 hr at 1000, 1100, 1300, and 1400°F on room- and elevated-temperature tensile properties was investigated. The material had a slight tendency to age; it attained a minimum ductility of 20%. Stress-strain curves were developed for INOR-8 at temperatures ranging from room temperature to 1400°F. Creep data are being

developed for INOR-8 at 1100, 1200, 1250, and 1300°F at a stress range of 10,000 to 30,000 psi. Creep of INOR-8 was shown in relaxation tests to occur at temperatures as low as 1100°F.

Procedure and qualification specifications were completed for the welding of INOR-8 tubing. Side-bend tests of INOR-8 weldments indicated that the as-welded material has good ductility to approximately 1200°F and gradually poorer ductility in the range 1200 to 1500°F. Side-bend tests at room temperature of specimens aged for 200 hr at temperatures up to 1300°F showed no loss in ductility. (AEC Activity 4103.1)

PACKAGE REACTOR DEVELOPMENT

Army Package Power Reactor (APPR-1). — The Laboratory continued to furnish technical support to the Army Reactors Branch (AEC-APB) on the 10-Mw pressurized-water reactor being tested at Fort Belvoir, Virginia, by Alco Products, Inc. The six-month test operation and training program which started July 25, 1957, was completed without major difficulties. Accumulated thermal power production is now approximately 4 megawatt-years or about one-third that anticipated during the core life.

Improved fuel elements and control materials are being developed for core II of the APPR-1. In order to minimize the transport of Co^{60} activity, type 304L stainless steel with reduced cobalt content (approximately 0.006% Co) was selected for core structural material. Specifications were prepared, and procurement of the steel is in process. A program was initiated for evaluating the stability of boron nitride in stainless steel and the possibility of using such boron-containing stainless steel as a means for attaining increased boron content in core II.

A preliminary proposal was submitted to AEC-APB that outlined the metallurgical work required for development of modified core components for a nuclear power plant of the APPR-1 type having a thermal capacity of 25 Mw. The possibility of using 400-series stainless steel as structural material will be considered in order to minimize Co^{58} activity.

The first APPR-1 demineralizer was removed from service and shipped to the Laboratory during the fall of 1957 for examination. Complete analyses of the resin samples were made, and the principal activation products found were Co^{58}, Co^{60}, Fe^{55}, Fe^{59}, Mn^{54}, and Cr^{51}.

Analyses of U^{235} and boron content were made on 94 samples of tubular fuel elements supplied by the Martin Company. (AEC Activity 4601)

Gas-Cooled Reactor Development. — An analog computer study for determining both the transient and steady-state operating characteristics of the gas-cooled loop scheduled to be operated in a test facility in the ORR is nearly complete. Preliminary information was obtained on the time rate of response to changes in coolant flow, heater temperatures, and fuel positions

under various assumed operating conditions. Construction was completed on a test loop that will be used to determine the rates of piston ring wear and gas leakage from two of the reciprocating compressors of the gas-cooled loop.

Construction work on the loop has conformed to the revised schedule. Invitations to bid were sent out to prospective vendors for the fabrication of the economizer unit, which is based on an ORNL design. The vendor's drawings for the high-temperature gas filter were reviewed. All the equipment for the reactivity and flux plotting was fabricated, including the enriched fuel samples to be used in the disadvantage-factor measurements. The equipment was checked dimensionally in the core of the ORR, and tests will be initiated after a criticality test of the ORR has been completed.

A literature search was made, and a summary was prepared of data on fuel element burnup as a function of irradiation temperature for UO_2—stainless steel dispersion-type samples. (AEC Activity 4603)

Maritime Ship Reactor. — The Laboratory continued to furnish technical assistance to the Maritime Reactors Branch (AEC) in the program for developing nuclear-powered merchant ships. Design specifications and criteria for several parts of the NS "Savannah" reactor system were reviewed, and proposed development programs were evaluated.

A limited supporting effort in materials development was initiated. Methods for incorporating burnable poison into UO_2 are being developed. Preliminary physics calculations indicated that boron is the most promising poison element, and the compounds ZrB_2 and TiB_2 are being investigated. An experimental welding and brazing program was formulated for a study of problems in fuel-rod closure and fuel-bundle assembly.

The Laboratory was requested to consider construction and operation of a pressurized-water test loop. This facility will be used to obtain supplementary radiation-effects data for the first merchant ship and will be available for future development programs. After an investigation of the test facilities of several possible reactors, it was decided to operate the proposed loop in the ORR. Design criteria were established. Engineering drawings and performance data for similar existing loops will be used as a basis for the proposed loop in order to minimize the design effort.

A study of the feasibility of sea disposal of radioactive wastes from nuclear-powered merchant ships is being considered as a joint effort of the Health Physics and the Chemical Technology Divisions. The National Research Council Committee on Ocean Disposal of Radioactive Wastes was requested to select an oceanographer for assignment to ORNL to participate in this study. (AEC Activity 4151)

GENERAL REACTOR RESEARCH

Basic Reactor Research. — Recent experiments showed that an aqueous solution of UO_2F_2, enriched to 93% in U^{235}, constituting a critical spherical volume having a diameter of 55.9 cm and surrounded by an effectively infinite water reflector, has an $H:U^{235}$ atomic ratio of 1258; removal of the reflector requires a decrease of the $H:U^{235}$ ratio to 1088. The ratio for an unreflected sphere of the solution with a diameter of 69.2 cm is 1363. The temperature coefficients of reactivity of the three volumes, measured over a narrow temperature range about room temperature, are -0.76¢/°F, -2.0¢/°F, and -1.5¢/°F, respectively. The uncertainty in the first of these coefficients is ± 5%, and in the last two it is ± 10%.

The measurement of fission gamma radiation delayed up to a few microseconds after fission was completed. Although the results have considerable qualitative interest, only weak quantitative conclusions are possible. Equipment is now being prepared for the final determination of the spectrum of prompt gamma rays with energies >1.5 Mev.

A safeguard report for the stainless-steel-clad core for the BSR was completed.

The fabrication of the Pool Critical Facility at the BSF was completed. A final checkout of the entire control system is currently in progress, and preparations are being made for critical experiments. (AEC Activity 4202)

Fuel Element Development. — In developing composite aluminum plates, containing a 48% U—3% Si—bal Al alloy, it has been demonstrated that thinning of the cladding at the core ends is negligible when such an alloy is framed in alclad 5154 aluminum and clad with type 1100 aluminum. Miniature composite specimens containing the silicon-modified alloy have been prepared and are being shipped to the MTR for irradiation testing to U^{235} burnup as high as 60%.

Preparation of dispersions of 42 vol % U_3O_8 in aluminum for this application has met with difficulty because of segregation of the oxide. The work has been directed toward a reduction of the oxide volume to 35%, and recent radiographic results have revealed acceptable homogeneity. Miniature composites containing 35 wt % depleted oxide have been successfully prepared, which demonstrates that miniature specimens containing the 20% enriched U_3O_8 can be made.

The project of developing dispersion of UC_2 in aluminum has been completed. It was demonstrated that fuel plates could be fabricated and that elements could be brazed; however, corrosion data revealed that the dispersion was susceptible to catastrophic corrosion in the event of an inadvertent cladding defect. The full-size irradiation test element has therefore been canceled, and irradiation testing will be confined to miniature samples. (AEC Activity 4203)

Thorex Process. — Pilot-plant operations have been discontinued, and alterations are being made for future plant use.

Further studies on decontamination of the Thorex product from ruthenium confirmed earlier indications of the beneficial effect of bisulfite. When synthetic feed solutions were adjusted to 0.2 M acid deficiency with dibasic aluminum nitrate and heated to 50°C for 2 hr with 0.06 M bisulfite and then extracted once, the decontamination factor from ruthenium was 200, compared with 2 in a control. The decomposition rate of sodium bisulfite in synthetic Thorex feed solution exposed to a Co^{60} source was 0.004 mole/whr but in water was only 0.001 mole/whr. The factor of 4 difference is presumably due to nitrite generation.

Laboratory experiments indicated that the radioactive daughters of the U^{232} decay chain are removed from the U^{233} product by the modified third uranium solvent extraction cycle or by cation or anion exchange. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Mechanical Processing. — A Yankee Atomic Power Reactor fuel element, UO_2 pellets enclosed in stainless steel, was sheared to separate the pellets from the jacket. The length of the cut portion had to be less than the length of a pellet for complete dislodging of the pellet material. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Sulfex Process. — In the Sulfex process, stainless steel jackets are separated from the cores by dissolution in sulfuric acid. Initial dissolution rates of type 347 stainless steel were 2.0, 5.5, 12, and 22 $mg \cdot cm^{-2} \cdot min^{-1}$ in boiling 2, 4, 6, and 8 M H_2SO_4, respectively. Losses to boiling 4 M H_2SO_4 of uranium and thorium from UO_2-ThO_2 pellets were 0.3 and 0.01%, respectively. The losses were independent of time and of ratio of volume of acid to weight of thorium. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Zircex Process. — In the Zircex process, zirconium-containing fuel elements are hydrochlorinated to form volatile $ZrCl_4$, the residue being processed for uranium recovery. In a 35-hr run, 97.4% of a 4.4-kg section of an STR subassembly was reacted with anhydrous HCl fed at a rate of 6 moles/hr. The HCl was 90% utilized. The uranium loss to the $ZrCl_4$ sublimate was 0.095%. Hydrogen was evolved at a practically constant rate, ~2.7 ft^3/hr, indicating that the reaction rate was independent of total area.

Zircaloy-2 dissolved in 9.6 M HCl in ethanol to a zirconium concentration of 0.75 M. The insoluble residue was about 0.02% when the water content of the solvent was <0.5 wt %. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Phosphex Process. — In continuing studies on dissolution of fuel in molten $3ZrCl_4 \cdot 2POCl_3$ (360°C), dissolution rates were ~10 $mg \cdot cm^{-2} \cdot min^{-1}$ for all fuel alloys tried. These included Zircaloy-2, 85% U—5% Zr—10% Nb, 94% U—5% Zr—1% Nb, and 90% U—10% Nb. Uranium was 97% recovered from the melt by nitric acid in the one run for which a material balance was made. A nitric acid—insoluble residue contained 2.5% of the uranium that had dissolved in the reagent. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Ion Exchange. — In the cation exchange separation of plutonium from a 4 M H_2SO_4 decladding solution for stainless steel, iron in the product was decreased from 22% to 6.8% by scrubbing the loaded resin with 0.5 M H_2SO_4 before eluting the plutonium with 2 M HNO_3. The additional loss of plutonium was only 0.1%. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Darex Process. — In the Darex process, stainless-steel-containing fuel elements are dissolved in aqua regia, and the chloride is stripped off before the uranium is recovered by solvent extraction. A 46-hr continuous dissolution and stripping run was made without difficulty. Because the dissolver discharged its product at a varying rate, the stripper flooded at cyclic intervals, resulting in varying chloride stripping (100 to 1500 ppm chloride in the product). A jackleg has now been installed to decrease such fluctuations. Estimated over-all material balances for HNO_3, H_2O, and HCl were 104, 98, and 97% for the stripper and 82, 96, and 95% for the dissolver.

In preliminary bench-scale studies on continuous concentration of dissolver product containing 900 ppm of chloride without removing the chloride, the concentrated product still contained the chloride. The concentrated product, containing 280 g of stainless steel per liter and 2.9 M HNO_3, was metastable, requiring as long as 16 hr at room temperature for solids to appear. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Electrolytic Dissolution. — When Zircaloy-2 was dissolved electrolytically in 3 M HCl at a current density of 0.37 amp/cm^2, 13.7% of the zirconium appeared as a metal sludge. The net current efficiency was 105%. In 3 M HNO_3—0.2 M HCl at current densities up to about 0.4 amp/cm^2, only small amounts of metal were present in the sludge, which consisted chiefly of hydrolyzed zirconium. Only 30 to 50% of the anode weight loss appeared as dissolved zirconium. (AEC Activity 4301)

Fused Salt—Fluoride Volatility Process. — In the fused salt—fluoride volatility process, uranium-zirconium fuel elements are dissolved in molten $NaF-ZrF_4$, and the melt is hydrofluorinated to form UF_4; this is fluorinated to volatile UF_6, which is collected in NaF traps.

Tests with 400°C beds of NaF gave inconclusive differences between 12 to 20 mesh and 1/8-in. pellets, although decontamination factors were highest with the material having the highest surface area. More than 50% of the fission products, plutonium, or chromium removed was generally found on the first quarter of each bed.

In corrosion studies on fused ZrF_4-NaF at 650°C with HF bubbled through at a rate of 10 g/hr, Hastelloy W gained weight slightly after 250 hr; INOR-8 and nickel A lost weight corresponding to a corrosion rate of about 0.089 and 8.8 mils/month. The HF used contained 100 to 1000 ppm of sulfur, presumably as fluorosulfonic acid, which was shown to cause severe embrittlement of nickel, 80-20 nickel-cobalt alloy, and monel. Inconel, Hastelloy B, and INOR-8 were not affected, with INOR-8 showing the least embrittlement.

Volatility processing of uranium-zirconium fuel elements following a Zircex head-end treatment is being considered. Loss of uranium in the $ZrCl_4$ sublimate from the Zircex hydrochlorination was only 0.002% when the off-gas was passed through a packed nickel filter. The uranium would be recovered from the filter in the fluorination step.

Barren salt flushes and rinses with ammonium oxalate solution and alkaline-peroxide-tartrate of contaminated laboratory equipment effected a decontamination factor of approximately 10. The corrosion attack of 0.4 M ammonium oxalate solutions at reflux on Inconel was established to be less than 0.001 mil/hr on the basis of weight-loss tests.

In preliminary studies on the feasibility of volatility processing of the molten fuel LiF-BeF_2-UF_4, the solubilities of LiF and BeF_2 in aqueous and anhydrous HF were determined. At 12°C, in 70% HF, the solubilities of LiF and BeF_2 were 23 and 11 g/liter, respectively; in anhydrous HF, the solubility was about 103 g/liter. This is significant for recycling of the Li^7. (AEC Activity 4302)

Waste Treatment and Disposal. — The nitrate in 5 M HNO_3 wastes was electrolytically destroyed, with 100% current efficiency and a power consumption of 3.3 kwhr per pound of nitrate. With synthetic ORNL waste, which is only 0.1 as concentrated, typical current efficiencies were 30%, and the power consumption was 18 kwhr per pound of nitrate. Nitrate destruction was independent of temperature and electrode potential but was inversely proportional to current density. Plating out of ruthenium was proportional to temperature but was independent of current density and electrode potential.

In laboratory studies, Purex-type waste was reduced in volume by a factor of 15 when evaporated to dryness. (AEC Activity 4305)

Waste Disposal Research and Engineering: Disposal of Radioactive Wastes into Geologic Structures. — A report containing instructions for preparing five waste types for deep-well disposal has been completed, and samples of all five types have been prepared. Experiments with glass columns filled with synthetic aquifer sands are in progress to determine if chemical reactions between waste solutions, aquifer solids, and aquifer brines may cause precipitation and plugging. So far there has been little evidence of plugging.

Preliminary correspondence with the Illinois, Kansas, and Oklahoma Geological Surveys suggests the possibility of arranging with them for a detailed review of the geology of areas in their states to determine the apparent feasibility of deep-well disposal. The Continental Oil Company of Ponca City, Oklahoma, believes it is possible to put large volumes of liquid waste down a deep well drilled in shale by pressure fracturing the shale. This is of considerable general interest as a novel approach to the problem, and is of particular interest to ORNL because of its possible practical application to the local waste disposal problems. This question will be pursued further with Continental Oil.

The studies of the solubility of NaCl, $MgCl_2$, $CaCl_2$, $CaSO_4$, and Kleer mine salt in neutralized Purex and acid aluminum nitrate wastes are being concluded. To date there has been no great change in the solubility of these salts in neutralized wastes, compared with their solubility in acid wastes.

There has been no evident production of gases when the neutralized wastes and the salts are mixed. To date the two main conclusions to be drawn from the chemical compatibility tests are that the wastes will have to be neutralized and saturated with sodium chloride prior to disposal in a salt mine.

The heat calculations of Birch of Harvard University, Schecter of the University of Texas, and Crowell and Parker of ORNL have been compared and were found to be substantially the same.

It appears that dissiptation of heat will not be a major problem but that the converse will be true, namely, how to obtain enough curies to run a meaningful field experiment. Birch has indicated that the solution of a more sophisticated mathematical model is not worthwhile, since the physical system will vary to such a great extent.

Steps are being taken to effect the purchase or lease of the field experiment site. Preliminary plans for the field experiment have been made. (AEC Activity 4305)

PROGRAM 5000 — PHYSICAL RESEARCH

ISOTOPE PRODUCTION

Radioisotope Production. — The extraction of rare earths from 450 gal of 4-year-old Redox waste is progressing satisfactorily. The primary emphasis is on the purification of Pm^{147} for the distribution program.

The production of I^{131} and P^{32} has been expanded to provide short-term requirements from Canada and England.

At the Fission Products Pilot Plant the contracted portion of the modification phase for Redox waste processing has been completed.

A total of 8879 curies of Kr^{85} was processed during January to bring the inventory to 17,692 curies.

Radioisotope Development. — Promethium can be separated with americium from other rare earths by cation exchange with 0.2 M citrate as elutriant at elevated temperatures; the americium can then be separated from the promethium as a thiocyanate complex by anion exchange with the use of a 5 to 8 M ammonium thiocyanate elutriant at room temperature. In studies of a single absorption-elution cycle, the glycolate complex was found unsuitable for the separation of americium and promethium on Dowex 1 over a pH range of 4 to 7 and a glycolate concentration of 0.5 to 5 M.

In further studies of source preparation from Fission Products Pilot Plant products, cerous oxalate was precipitated from a solution 1.5 N in HNO_3, filtered, and calcined to CeO_2 at 600°C.

Design was completed on modifications to the Kr^{85} processing equipment to permit purifying and storing of 50,000 curies per month. (AEC Activity 5100)

Stable Isotope Separation. — The separation of 880 mg of C^{13} enriched to 90% from 1.1% charge feed has been completed. Elemental carbon was removed from the collector pockets in nine separate batches from which mass analyses were determined. The material presently is being homogenized into final sample form.

Calutron equipment required for the initial separation of the seven isotopes of ytterbium has been assembled. Very pure ytterbium oxide is being converted to anhydrous ytterbium(III) chloride to be used as charge material.

The recovery of erbium charge material from the recent production collection has been completed. A total of 1338 g of erbium oxide, having a chemical purity of >99.9%, is now available for future use. This, together with the enriched samples, represents a recovery of approximately 70% of the original charge.

The current separations of vanadium and selenium isotopes continue.

Chemical refinement has been completed on new lots of Rb^{85} and Rb^{87} and on seven new lots of erbium, namely, Er^{162}, Er^{164} (2 lots), Er^{166}, Er^{167}, Er^{168}, and Er^{170}. In addition to these, returned samples of Sr^{84}, Ba^{130}, Ba^{132}, Cu^{63}, Gd^{155}, Gd^{157}, and Gd^{158} were chemically reprocessed. Twenty-six new lots of erbium, rubidium, rhenium, platinum, and tin were prepared for inventory. Of these, Er^{162}, Er^{164}, Er^{166}, and Er^{168} were inventoried for the first time. Returned lots of Eu^{151}, Re^{185}, Sn^{118}, Sn^{119}, and Sn^{120} were also prepared for inventory.

Stable Isotope Development. — The feasibility of dividing a Beta-type calutron track into separate segments and operating each segment at a different magnetic field strength has been verified by experiments conducted upon a 1/16-scale model magnet. Approximately 90 tons of iron is required in each yoke for satisfactory magnetic isolation of the sections.

The use of a Beta-type ion source having two filaments has been extended to vanadium separations. The ion reception time per run is increased by a factor of approximately 2 by the use of this source. (AEC Activity 5100)

PHYSICS

High Voltage Program. — In order to observe an asymmetry in Mott scattering of beta particles from gold, it is necessary to convert the longitudinal spin of the electron into a transverse spin. This has been done by bending the electrons through 136 deg with the electric field in a 1-cm gap between two spherical aluminum plates of mean radius 25 cm. The 624-kev Cs^{137} K electrons are transmitted when the plates are charged to ± 17.5 kv. With a ± 6-deg azimuthal acceptance angle at the source, the transmission is 5×10^{-5}. Because of relativistic effects, there is no radial focusing, but in the direction of the expected scattering asymmetry the image of a 1.5-mm source is <2.5 mm wide at half maximum. Electrons which are scattered from the plates (and possibly depolarized) are not focused and appear as tails on the intensity distribution of the image. By inserting Lucite apertures between the plates, the ratio of peak to tail intensity for a P^{32} source has been made >100. Polarization measurements of P^{32} electrons are in progress.

The neutron differential elastic-scattering cross section of beryllium has been measured at a number of energies between 0.7 and 3.0 Mev. Neutrons produced by the $H^3(p,n)$ reaction were scattered by cylinders of beryllium and detected as a function of neutron scattering angle with a shielded propane recoil counter. The center-of-mass differential cross sections in the nonresonant region between 1.0 and 2.0 Mev show an appreciable cos θ term in the angular dependence, thus indicating a relatively large p-wave contribution. The elastic scattering in the region of the broad peak in the total cross section around 2.8 Mev changes from being nearly symmetric about 90 deg at 2.44 Mev to being strongly peaked forward at 2.92 Mev. Qualitatively, this suggests two resonances in this region — a resonance involving no parity change in forming the compound state, followed by a resonance in which the parity is changed. This information, together with the data on the $Be^9(n,\alpha)$ cross section (Phys. Rev. 106, 1252, 1957), probably identifies the very broad state at about 2.9 Mev neutron energy as being formed by p-wave neutrons, with J = 2+ for the compound state of Be^{10}. (AEC Activity 5220)

The 86-Inch Cyclotron: Nuclear Physics. — Measurements were made of the energy distributions of protons, deuterons, and tritons from (p,p'), (p,d), and (p,t) reactions induced by 23-Mev protons in various elements through the periodic table and at several angles relative to the incident beam.

The proton energy distributions may be qualitatively explained as Maxwell distributions multiplied by Coulomb barrier penetration factors, except that there is an excess of low-energy particles. Both the deuteron and triton distributions are peaked near their maximum energies. Their fall-offs with decreasing energy are quite rapid and cannot be explained by Coulomb barrier effects; however, they can be simply explained by parentage considerations if (p,d) and (p,t) reactions are assumed to proceed very predominantly by neutron pickup. The angular distributions of all the particles are peaked in the forward direction. Typical cross sections for (p,p') and (p,d) reactions are 300 and 60 mb, respectively; they vary slowly through the periodic table. The cross sections for (p,t) reactions increase with increasing atomic number; for heavier elements, they are typically ~15 mb.

A precision measurement was made of the cross section for proton-induced fission in U^{238} in order to determine the total reaction cross section. The preliminary result is in reasonable agreement with predictions of the optical model. (AEC Activity 5220)

The 86-Inch Cyclotron: Applied Physics. — The isotopes V^{48}, Fe^{55}, As^{74}, Sr^{85}, Rh^{102}, Ce^{139}, Au^{194}, and $Bi^{205,206}$ were produced in service irradiations for outside customers. Gallium-67 was produced for the ORNL Chemistry Division. A target of vacuum-evaporated CaF_2 on an aluminum backing was successfully run at a current of 500 μa.

Test runs were made in an attempt to deflect a mass-3 singly charged ion. An internal H_3^+ beam was obtained, and its character was studied; a target for the detection of the deflected beam has not yet worked satisfactorily. (AEC Activity 5220)

Heavy-Particle Physics. — Angular distributions of light particles from the nitrogen bombardment of oxygen show an essentially isotropic distribution for protons and a strong forward peak for alpha particles. Light-particle spectra from the nitrogen bombardment of carbon, nitrogen, and oxygen are being investigated at 45 deg in the laboratory system. A reaction chamber for the study of the elastic scattering of nitrogen from beryllium was put into operation, and preliminary results were obtained. (AEC Activity 5220)

Spectroscopy Research Laboratory. — Two lines have been found in the spectrum of a high-current arc in mercury which fit the transitions $6s6p\ ^1P_1^o - 6p^2\ ^3P_0$ and $6s6p\ ^1P_1^o - 6p^2\ ^3P_1$ of Hg I, confirming the assignments of the levels $6p^2\ ^3P_{0,1}$ by Garton and Rajaratnam (Proc. Phys. Soc. A68, 1107, 1955). Combination of the frequencies of these lines with the measurements of Selwyn (Proc. Phys. Soc. 41, 392, 1929) gives for $6p^2\ ^3P_0$ the value 90097.5 cm^{-1} and for $6p^2\ ^3P_1$ the value 93976.6 cm^{-1}, relative to $6s^2\ ^1S_0$. Prediction of the position of $6p^2\ ^3P_2$ in Hg I has been made by extrapolation from Zn I and Cd I, with the use of the equations for the configuration p^2 in intermediate coupling (The Theory of Atomic Spectra, Cambridge University Press, 1935, p 274). The coupling parameter χ was determined from the ratio of the intervals $^3P_2 - {}^3P_1$ and $^3P_1 - {}^3P_0$ to be 0.030 for $4p^2$ of Zn I and

0.131 for $5p^2$ of Cd I. A linear extrapolation of $\chi^{1/3}$ vs Z then gives χ = 0.632 for $6p^2$ of Hg I. The interval ratio is found to be 0.760, placing $6p^2\ {}^3P_2$ at about 96,900 cm^{-1}. The corresponding transition $6s6p\ {}^1P_1^o$ — $6p^2\ {}^3P_2$ could not be definitely established. Because of the relative weakness of the line $6s6p\ {}^1P_1^o$ — $6p^2\ {}^3P_1$ and the absence or weakness of the transitions $6s6p\ {}^3P_{1,2}^o$ — $6p^2\ {}^3P_2$ in Hg I, it is concluded that the line $6s6p\ {}^1P_1^o$ — $6p^2\ {}^3P_2$ is probably absent. (AEC Activity 5230)

Electronuclear Machines. — Additional studies of the performance of focusing grids in the cyclotron central region were made in the ORNL 44-Inch Cyclotron. The focusing grids are graphite structures placed on the entrant side of each dee so as to improve the electric focusing at each accelerating gap. The most recently tested structure has a 2-in. axial aperture; 3 to 4 ma of protons were programed through five turns in this structure. The energy gain per turn in these tests was set at 85 kev so that turn separations would correspond to those expected in the proposed Oak Ridge General Research Accelerator. (AEC Activity 5240)

CHEMISTRY

Raw Materials Chemistry. — The basis for the design and scaleup of gravity settlers for the Amex uranium extraction process was determined in 6- and 48-in.-dia tanks. The nominal flow capacity for water-in-oil dispersions was 0.4 gpm of aqueous phase per square foot of settler area, and for oil-in-water dispersons the aqueous-phase capacity ranged from 0.8 gpm/ft^2 when the ratio of aqueous phase to organic phase was 1:1 to 2.9 gpm/ft^2 at a ratio of 4:1. Modification of the kerosene diluent with 3% tridecanol increased the flow capacity for water-in-oil dispersons from 0.4 to 0.7 gpm/ft^2 but had no effect on oil-in-water dispersions.

In the stripping of uranium from amine extracts (Amex process) by nitrate, the ammonium nitrate formed in the regeneration of the amine contains most of any molybdenum in the original leach liquor. This molybdenum was recovered in very pure form by adsorption on a column of activated carbon, elution with ammonium hydroxide, and evaporation of the eluate to dryness.

Phosphate was removed from Dapex vanadium strip solutions by a column of activated zirconium oxide. The column was regenerated readily with dilute caustic. The removal of the phosphate from the strip solution allows precipitation of the vanadium as a hydrated vanadyl oxide (by neutralization with ammonia) without exceeding the phosphorus specifications for commercial vanadium products. (AEC Activity 5310)

Americium Recovery. — A solution received from Los Alamos, containing approximately 40 g of americium contaminated with kilogram amounts of Al, Fe, Cu, Ce, and La and smaller amounts of Cr, Mg, Pu, and U, was processed through two solvent extraction cycles in the Metal Recovery Plant. In the first cycle, americium, lanthanum, and cerium were extracted by 100% tributyl phosphate from

a nitrate solution containing no free acid. In the second cycle 99.7% of the americium, containing about 2% of the lanthanum, was extracted into 100% tributyl phosphate from 15 to 17 M HNO_3. Corrosion during the evaporation of the second-cycle product resulted in considerable contamination with iron, chromium, and nickel. One batch of this product containing 18 g of americium was concentrated by sorption on Dowex 50 resin followed by elution with acidified ammonium citrate solution. About 90% (16 g of americium) was collected in a concentrated product cut; the other 10% will be recycled through ion exchange. A second batch of Metal Recovery second-cycle product, containing more corrosion products, could not be satisfactorily processed by ion exchange. The americium, lanthanum, and cerium in this product are being removed from the corrosion products by oxalate precipitation. (AEC Activity 5310)

Feed Materials Processing. — In studies on the reduction of UF_6 by sodium amalgam at 200 to 360°C, the UF_6 was 10 to 20% reduced to metal and 46 to 90% reduced to UF_4. The reactants were mixed by gas sparging. The UF_6 was fed to the reactor in a stream of argon which picked up the UF_6 upon passage over a bed of UF_6-NaF solid complex heated to 250 to 360°C. The contacting of UF_6-NaF solid complex with sodium amalgam at 300°C in a high-shear mechanical mixer reduced 2% of the UF_6 to uranium metal and 79% to UF_4.

In studies on the Druhm process, 94 g of UF_6 was 31% reduced to uranium metal by sodium vapor (81% material balance) in a 17-min run in which a graphite liner extended to the top of the reduction contactor. The contactor temperature, initially 800°C, rose to 1030°C. (AEC Activity 5310)

High-Temperature and Structural Chemistry. — The specific electrical conductance of solutions of potassium metal in molten potassium iodide was found to increase with metal concentration and temperature in a manner quite similar to that of the solutions in molten chloride and bromide. The three potassium systems differ radically from the corresponding sodium solutions by a monotonic increase, rather than an initial decrease, of the equivalent conductance of the metal with metal concentration. Significantly, the equivalent conductance of potassium metal in KI is, at low metal concentrations, higher than in either KBr or KCl, but at higher concentrations it is lower than in KBr, or intermediate between the conductances in KBr and KCl. Thus, again, the exceptional behavior of all the bromide systems studies thus far in this work seems to be reflected, namely, the consistently greater ideality, that is, lower degree of metal association, as compared with that in either the chloride or iodide solutions. The equivalent conductance of the metal at low concentrations, where every solution is more nearly ideal, being highest in the iodide system, is thought to reflect again, as it did in the sodium systems, the influence of the greater polarizability of the anion upon the mobility of the metal electrons.

In the study of the phase transformations of calcium carbonate by grinding, the metastable third form of $CaCO_3$, μ-$CaCO_3$, or vaterite, was prepared and was observed, on grinding, to undergo disintegration to a structure amorphous to x rays. (AEC Activity 5330)

Organic Chemistry. — During an attempt to study the kinetic O^{18} isotope effect in the dehydration of formic-O^{18} acid in a large excess of 96% sulfuric acid, an oxygen exchange between the reaction product, carbon monoxide-O^{18}, and the sulfuric acid was observed. The table indicates approximately the rate of loss of O^{18} from the carbon monoxide phase, as obtained from mass spectrometric data. The sample of COO^{18} represents the initial formic acid added to the sulfuric acid and hence also the initial isotopic composition of the carbon monoxide produced in the dehydration reaction, which was complete in a few minutes. The three analyses of CO^{18} represent successive samplings of the carbon monoxide gas. With additional studies it may be possible to suggest a mechanism for the exchange.

EXCHANGE OF CO^{18} WITH 96% H_2SO_4

Sample Analyzed	Contact Time		% O^{18}
	HOT	COLD	
COO^{18}			1.07*
CO^{18}	80°C, 15 min	4 days	0.54
CO^{18}	80°C, 15 min	4 days	0.49
CO^{18}	60°C, 2 hr	Shaken 60 hr	0.35
Normal composition			0.2

*After division by 2.

Continued consideration of the polyphenyls as reactor materials led to the study of the effect of radon alpha particles (heavy particles) on biphenyl. Gas (largely hydrogen) was generated; most of the diphenyl was recovered unchanged, and the polymer was separated into two fractions (benzene-soluble and benzene-insoluble).

Tributyl phosphate was shown to be extracted from hydrocarbon solutions with glycols and glycol derivatives. This fact can be used to develop (1) analytical methods for the estimation of tributyl phosphate in process solutions and (2) methods for the decontamination and purification of process solvent. (AEC Activity 5330)

Radiation Chemistry. — The experimental evidence from numerous different experiments on the polymerization of carbon monoxide by alpha particles indicates that the over-all mechanism is best described by the following series of reactions:

$$4\overline{CO} \overset{\alpha}{\rightleftharpoons} \overline{CO_2} + \overline{C_3O_2}$$

$$x\overline{C_3O_2} \longrightarrow \underline{(C_3O_2)_x}$$

$$\underline{(C_3O_2)_x} \xrightarrow{\alpha} x\overline{CO_2} + \underline{2xC}$$

$$\underline{(C_3O_2)_x} \xrightarrow{\alpha} 2x\overline{CO} + \underline{xC}$$

(The bar above the formula indicates a gas and below the formula a condensed state.) (AEC Activity 5330)

Molten Salt Thermodynamics. — As part of a continuing study of activities in molten salts, attention has been given to the interaction between Ag^+ and SO_4^{--} ions in solution in molten KNO_3. From measurements in dilute ranges in which Nernst's law was followed, changes in the activity coefficient of $AgNO_3$ (at mole fractions less than 0.006) due to the presence of SO_4^{--} (at mole fractions less than 0.010) were determined electrometrically, using Ag^o electrodes, at 360 to 450°C.

Relatively small negative deviations were found, corresponding from one point of view to a preferential occurrence of Ag^+ and SO_4^{--} ions as neighbors at about three times the random frequency. A previous study of Ag^+ and Cl^- ions under equivalent conditions in the same solvent had indicated that Ag^+ and Cl^- ion pairs occur about 20 times as often as expected for a random distribution. (AEC Activity 5330)

Analytical Chemistry Research. — Spectrophotometric methods have been developed for the determination of cerium and of titanium in mixtures of fluoride salts. The method for cerium is based on the absorbancy of HCl solution at 253 mμ. At this wavelength the molar absorbancy index is 755 and Beer's law is obeyed over the concentration range of 0.02 to 2 mg of cerium per milliliter. The method for titanium is based on the absorbancy, at a wavelength of 432 mμ, of titanium thiocyanate in a cyclohexane solution of trioctylphosphine oxide. The molar absorbancy index is 41,000 over the range of 0.02 to 1 μg of titanium per milliliter.

A nondestructive method of activation analysis has been developed for the determination of trace quantities of argon in a number of gases. The gamma radioactivity of A^{41}, from the reaction $A^{40}(n,\gamma)A^{41}$, is measured by means of a scintillation counter. Activation methods of analysis have also been adapted to the determination of the particle-size distribution of a wide variety of solid materials, such as nickel and iron powders, nickel oxide, aluminum oxide, titanium oxide, germanium oxide, and platinized aluminum oxide catalysts. Particle-size distributions ranging in diameter from 0.2 to greater than 120 μ have been evaluated.

METALLURGY

Fundamental Physicometallurgical Research. — Berg-Barrett x-ray reflection micrographs showed strain-free grains to be present in the extruded aluminum rods used for preferred orientation studies. This confirms the conclusion based on pinhole x-ray photographs and optical metallography that recrystallization occurs during deformation in 99.99% aluminum even at room temperature.

Vapor pressure data were obtained which suggest that the heat of formation and excess entropy of formation are positive in the alpha phase of Zr-Cd and negative in the beta phase. The result is based on two alloys and should be considered with caution until additional alloys have been measured.

In the zirconium-cadmium system, the temperatures at which the beta phase is in equilibrium with vapor at 1 atm of pressure are 1170, 1040, and 940°C for the compositions 2, 5, and 10.5 at. % Cd, respectively.

The low-temperature specific heat of zirconium appears to have small systematic deviations from the $\gamma T + \beta T^3$ law. Experiments are in progress to determine whether these effects are real or are caused by small experimental errors. (AEC Activity 5420)

Microstresses in Metal Crystals. — A new fundamental research group in the Metallurgy Division became active in September 1957. This group will be interested in the study of crystal imperfections by means of diffraction techniques. Order-disorder phenomena, cold work, thermal motion, size effects, and radiation damage are examples of the type of imperfection with which the group will be concerned.

Since its inception, the major activity of this group has been the design, construction, and acquisition of x-ray diffraction equipment appropriate for the study of defective crystals. A General Electric XRD-5 diffractometer has been purchased and modified so that it may be used to measure the distribution of diffuse x-ray scattering caused by the presence of imperfections in a crystal. The main features of this instrument are a lithium fluoride monochromator, doubly bent to achieve maximum intensity and maximum resolution, and a scintillation detector and pulse-height discriminator to eliminate the half-wavelength component of the beam obtained with the monochromator. Pending completion of the detector system, this instrument is being operated and tested with a Geiger counter detector. (AEC Activity 5420)

Fundamental Investigations of Radiation Damage in Solids. — Upon irradiation by fast neutrons, p-type polycrystalline specimens of the semiconducting intermetallic compound indium arsenide decrease in electrical conductivity to a minimum value and then increase. Subsequent Hall coefficient measurements indicate a conversion from p-type to n-type material. The primary result of lattice damage in InAs is, therefore, the introduction of additional donor-type centers. The addition rate of such centers is fairly equivalent to the total integrated fast-neutron flux received, for both n- and p-type material. The effect of annealing at a moderate temperature (100°C) indicates that the changes in the concentration of current carriers which are produced by irradiation are stable to at least this temperature.

The concentration of current carriers in high-purity n-type samples of germanium has been investigated as a function of temperature after such samples have been converted to p-type as a result of irradiation by Co^{60} gamma rays. The low-temperature slope of the Hall coefficient curve corresponds to an apparent activation energy of 0.2 ev or greater even after very

extensive irradiations. Annealing at a moderate temperature (100°C) increases the hole concentration in the low-temperature range but does not significantly alter the low-temperature activation energy. Continued irradiation subsequent to such a moderate-temperature annealing actually decreases the apparent hole concentration to approximately its value before the annealing. These results are not in agreement with the model of preferential interstitial migration previously suggested, since according to that model the apparent hole concentration should have increased.

The minority-carrier lifetime in fast-neutron- and gamma-irradiated samples of n- and p-type germanium has been measured as a function of radiation dosage and temperature. The lifetime of electrons in p-type germanium has a sensitivity to irradiation that is comparable with that of the lifetime of holes in n-type germanium. These measurements are being employed to determine the capture cross section and energy position of the recombination centers introduced by irradiation. (AEC Activity 5430)

PROGRAM 6000 — BIOLOGY AND MEDICINE

BIOPHYSICS

Radiation Dosimetry Programs. — Presently accepted estimates of radiation exposure are based on the number of ions or electrons liberated by the passage of radiation through matter. The necessity for the introduction of relative biological effectiveness (RBE) values in order to estimate biological damage from these considerations suggests that other factors may be involved. Among these is the distribution in energy of the liberated electrons, as opposed to considerations of only the number of such electrons. Parallel-plate ionization chambers have been constructed of graphite, aluminum, and copper, and current-voltage curves for these have been obtained at effectively zero gas pressure when the plates are bombarded with approximately 3 r/sec of lightly filtered x rays generated by a 250-kv constant-potential machine. A saturation current is obtained at a chamber potential of about 20 v, and this current corresponds to 2.3×10^{-4}, 11×10^{-4}, and 63×10^{-4} $esu \cdot cm^{-2} \cdot r^{-1}$ for the graphite, aluminum, and copper plates, respectively. In the case of a metal, the energy distribution of the electrons inside the metal may be found from the current-voltage curve. These distributions are observed for only those electrons which have at least enough kinetic energy to escape from the metal (15.9 ev for Al and 11.4 ev for Cu). The flux of such electrons inside the metal declines monotonically from the value at the energy given above, dropping to half value at an energy 0.5 ev higher and reaching zero value at about 20 ev higher. (AEC Activity 6140)

Waste Disposal Research: Soil Disposal of Solid Radioactive Wastes. — As a result of the increased use of the ORNL burial facilities by outside agencies and the increased generation of solid wastes at X-10 and Y-12 over the past years, the average annual burial rate has increased from about 1.5 acres per year to 5.0 acres per year. Consequently, at the present rate of burial, the existing burial ground will be filled in six to eight months. A field reconnaissance of potential sites for solid waste disposal was completed, and a 20- to 25-acre tract of land in Melton Valley was selected for further investigation. The site meets the preliminary requirements of ample area for a 4- to 5-year operating lifetime. It is located within the White Oak Lake drainage basin near the Laboratory, and is in the Conasauga formation, in which a trench is easily excavated and is not subject to collapse. The area is of gentle relief for ease of operation and yet is not subject to flooding by surface water. Ground-water and rock conditions are being investigated prior to use of the site. (AEC Activity 6440)

PROGRAM 8000 — WORK FOR OTHERS

Nuclear Technology Education for University Faculty Personnel. — W. A. Lloyd, of Pennsylvania State University, completed his six-month program at ORSORT on February 14. During the latter part of February, three university faculty members reported for six-month programs, as follows: Kenneth Bell, Case Institute of Technology; W. C. Simpson, Georgia Institute of Technology; John Wethington, University of Florida. (AEC Activity 8736.4)

MISCELLANEOUS REPORTS

OPEN LITERATURE PUBLICATIONS

Barnett, C. F., and H. K. Reynolds, "Charge Exchange Cross Sections of Hydrogen Particles in Gases at High Energies," Phys. Rev. 109, 355 (1958)

Barnett, C. F., and P. M. Stier, "Charge Exchange Cross Sections for Helium Ions in Gases," Phys. Rev. 109, 385 (1958)

Barton, C. J., H. A. Friedman, W. R. Grimes, H. Insley, R. E. Moore, and R. E. Thoma, "Phase Equilibria in the Alkali Fluoride—Uranium Tetrafluoride Fused Salt Systems: I. The Systems LiF—UF_4 and NaF—UF_4," J. Am. Ceramics Soc. 41, No. 2, 63 (1958)

Cantor, S., R. F. Newton, W. R. Grimes, and F. F. Blankenship, "Vapor Pressures and Derived Thermodynamic Information for the System RbF-ZrF_4," J. Phys. Chem. 62, 96 (1958)

Davis, F. J., and P. W. Reinhardt, "Instrumentation in Aircraft for Radiation Measurements," Nucl. Sci. and Eng. 2, No. 6, 713 (1957)

Fisher, D. J., "Falling-Drop Timing Circuit with Automatic Reset," Anal. Chem. 30, 308 (1958)

Fowler, J. L., and H. O. Cohn, "Oxygen Differential Neutron Scattering and Phenomenological Nuclear Potentials," Phys. Rev. 109, 89 (1958)

Fulmer, C. B., and B. L. Cohen, "Equilibrium Charges of Fission Fragments in Gases," Phys. Rev. 109, 94 (1958)

Harvey, J. A., and D. J. Hughes, "Spacings of Nuclear Energy Levels," Phys. Rev. 109, 471 (1958)

Householder, A. S., "A Survey of Some Closed Methods for Inverting Matrices," J. Soc. Ind. Appl. Math. 5, No. 3, 155 (1957)

Howden, H. F., and S. I. Auerbach, "Some Effects of Gamma Radiation on Trogoderma Sternale Jayne," Entomological Soc. Am. 51, No. 1, 48 (1958)

Johnson, C. H., J. P. Judish, and C. W. Snyder, "Focusing Procedures for Electrostatic Accelerators," Rev. Sci. Inst. 28, 942 (1957)

Jones, A. R., "Radiation-Induced Reactions of Potassium Bromide with Air," Science 127, 234 (1958)

Kasten, P. R., and R. E. Aven, "Fuel Costs in Batch- and Continuous-Processed Homogeneous Reactors," Ind. Eng. Chem. 50, 171 (1958)

Kraus, K. A., and F. Nelson, "Radiochemical Separations by Ion Exchange," Ann. Rev. Nucl. Sci. 7, 31 (1957)

Kraus, K. A., "Ion Exchange," Trace Analysis, J. Y. Yoe, Ed., John Wiley and Sons, Inc., N. Y., p. 34 (1957)

Macklin, R. L., "Graphite Sphere Neutron Detector," J. Nucl. Inst. 1, 335 (1957)

Macklin, R. L., and J. H. Gibbons, "A Study of the $T(p,n)He^3$ and $Li^7(p,n)Be^7$ Reactions," Phys. Rev. 109, 105 (1958)

Mahlman, H. A., and J. W. Boyle, "Free Radical and Molecular Yields in the Gamma-Ray Decomposition of Heavy Water," Rad. Res. 7, 436 (1957)

Mahlman, H. A., and G. K. Schweitzer, "Radiation Induced Nitrite Formation from Concentrated Nitrate Solution," J. Inorg. and Nucl. Chem. 5, 213 (1958)

Milford, R. P., "Engineering Design of Oak Ridge Fluoride Volatility Pilot Plant," Ind. Eng. Chem. 50, 187 (1958)

Morgan, K. Z., "Radioisotope Concentration Standards," Safety Aspects of Nuclear Reactors, C. R. McCullough, Ed., D. Van Nostrand Co., Inc., N. Y., p. 83 (1957)

Osborne, R. K., "Some Characteristics of the Thermal Neutron Scattering Probability," Nucl. Sci. and Eng. 3, 29 (1958)

Pigg, J. C., and C. C. Robinson, "Radiation Effects in Semiconductor Devices," Proc. of Transistor Reliability Sym. Sept. 17-18, 1956, New York University Press, p. 77 (1958)

Poppendiek, H. F., and L. D. Palmer, "Heat Transfer in Heterogeneous Circulating-Fuel Reactors," Nucl. Sci. and Eng. 3, 85 (1958)

Schwartz, E. S., "Hot Cell Operation at the ORNL Solid State Division," Proc. Fifth Hot Labs. and Equip. Conf., Pergamon Press, p. 63 (1957)

Stevens, D. K., W. J. Sturm, and R. H. Silsbee, "Magnetic Susceptibility of Neutron-Irradiated Quartz," J. Appl. Phys. 29, 66 (1958)

Strickler, T. D., H. E. Gilbert, and J. A. Auxier, "Fast Neutron Scattering from Thick Slabs," Nucl. Sci. and Eng. 3, No. 1, 11 (1958)

Williams, L. G., and H. D. Swanson, "Concentration of Cesium 137 by Algae," Science 127, 187 (1958)

Worsham, J. E., M. K. Wilkinson, and C. G. Shull, "Neutron Diffraction Observations on the Palladium-Hydrogen and Palladium-Deuterium Systems," J. Phys. and Chem. of Solids 3, No. 3/4, 303 (1957)

Yoder, R. E., and E. M. Empson, "Experimental Sand Filter for Air-Borne Radioactive Particulates," Public Works 88, No. 12 (1957)

VISITORS

Analytical Chemistry Advisory Committee:

Eyring, H.
Friedlander, G.
Furman, N. H.
Laitinen, H. A.
Larson, C. E.

Libby, Commissioner W. F., AEC, Washington
Tonks, L., University of California Radiation Laboratory
Wigner, E. P., Princeton University

FOREIGN VISITORS:

International School of Nuclear Science and Engineering:

Akpinar, Sait, Turkey
Akram, Kazi H., Pakistan
Alvarez-de-Toledo Morenes, Carlos, Spain
Bottoni, Giorgio, Italy
Chelet, Yves Jean-Pierre, France
Chi, Changyul, Korea
Choudhury, Md. Mujibur, Pakistan
Clouet D'Orval, Christian M., France
Cova, Carlo M., Italy
Eugenio, Manuel R., Philippines
Fornaciati, Paolo, Italy
Frederico, Jose Claudio, Brazil
Frisancho, Pineda Ignacio, Peru
Gain, Rolf K., Germany
Galvan Cabrerizo, Jose-Ramon, Spain
Goerlings, Maurits Willem, Netherlands
Gilli, Paul V., Austria
Gonzales-Moreno, Juan-Manuel, Spain
Heda, Mikio, Japan
Hirata, Mitsuho, Japan
Kaegi, Jakob, Switzerland
Koenig, Peter J., Switzerland
Kunzel, Karlheinz O., Germany
Kuramoto, Masaaki, Japan
Lee, In-King, China
Lim, Iong Kyu, Korea
Lwin, U Soe, Burma
Melches, Carlos, Spain
Moon, Suchyong, Korea
Pelled, Zeev, Israel
Prabhakar, Bangalore S., India
Prosdocimi, Armando, Italy
Quraishee, Malik M., Afghanistan
Rahman, Mohammad N., Pakistan

Rani, Antonio, Italy
Rietman, Jan, Netherlands
Ringels, Wilhelm K., Germany
Rogla Altet, Vicente, Spain
Rosatelli, Armando C., Italy
Sacay, Teiti, Brazil
Samson, Rogelio G., Phillipines
Sasaki, Shiro, Japan
Schludi, Harry, Germany
Shafique, Muhammad, Pakistan
Shukla, Agam P., India
Sukijbumrung, Krivuthi, Thailand
Sumawong, Watana, Thailand
Tseng, Der, China
Van de Linde, Jan G., Netherlands
Vasquez-Barete, Manuel, Mexico
Vianna, Antonio Didier, Brazil
Vidal, Rene, France
Waspodo, Soeroto, Indonesia
Wiriatmadja, Raden Djohari, Indonesia

Japanese Electric Power Team:
Fujita, Tomojiro, Kansai Electric Power Co., Osaka, Japan
Hatamori, Ichiro, Kansai Electric Power Workers' Union, Osaka, Japan
Imai, Masaaki, Japan (Interpreter)
Ishigame, Tsutomu, Federation of Electric Power Companies, Tokyo, Japan
Ishikawa, Takeshi, Tohoku Electric Power Workers' Union, Iwate-ken, Japan
Ito, Osamu, Chubu Electric Power Workers' Union, Nagoya-shi, Japan
Matsushita, Masaaki, Japan (Interpreter)
Misaka, Mitsukazu, Kyushu Electric Power Workers' Union, Fukuoka-shi, Japan
Shirakawa, Masanori, Tohoku Electric Power Co., Ltd., Japan (Team Leader)
Shirasawa, Tomiichiro, Tokyo Electric Power Co., Inc., Tokyo, Japan
Sunahara, Seiji, Hokkaido Electric Power Workers' Union, Hokkaido, Japan
Takamura, Yoshihiro, Ministry of Int'l. Trade and Industry, Tokyo, Japan
Takemura, Shigetake, Hokuriku Electric Power Co., Toyama City, Japan
Yasutake, Masatake, Tokyo Electric Power Co., Inc., Tokyo, Japan

Appelqvist, K., Uddholm Co., Stockholm, Sweden
Barricelli, N. A., University of Rochester (Citizen of Italy)
Bedoya, H., U. S. Information Agency (Citizen of Colombia)
Benedetti, M., Government of Uruguay
Bernot, J., French AEC
Bradley, R. A., Virginia Poly. Institute (Citizen of Canada)
Collery, L., Veterinary College, Dublin, Ireland
Duemmler, F., German Atomic Study Group, Germany
Eng., B. C. V., Degenfors Iron Works, Sweden
Fischer, E. H., Reactor Corporation, Hamburg, Germany
Gatheron, B., U. S. Information Agency (Citizen of France)
Jonnerby, E., Degenfors Iron Works, Sweden
Kistemaker, J., Netherlands

VISITORS (Continued)

Laska, L., German Atomic Study Group, Germany
Lindberg, G., A. B. Atomenergi, Stockholm, Sweden
Luthman, P., A. B. Atomenergi, Stockholm, Sweden
Meier, Rhys, Rhodesia Herald, Southern Rhodesia
Moore, J., United Kingdom Atomic Energy Authority, Great Britain
Roebert, G. A., Reactor Corporation, Hamburg, Germany
Schiller, A. K., German Atomic Study Group, Germany
Schulz, E. H., German Atomic Study Group, Germany
Vaccaro, H., U. S. Information Agency (Citizen of Argentina)
Von der Decken, C. B., Krupp, Brown, and Boveri Working Group, Germany
Wenzel, J., German Atomic Study Group, Germany
Wiesenack, L. G., German Atomic Study Group, Germany
Wolff, J., German Atomic Study Group, Germany
Wundram, O., German Atomic Study Group, Germany
Zeller, K., Wayne University (Citizen of Germany)

-oOo-

Reports in this series issued during the past year:

February 1957	ORNL-2281
March 1957	ORNL-2299
April 1957	ORNL-2316
May 1957	ORNL-2342
June 1957	ORNL-2355
July 1957	ORNL-2370
August 1957	ORNL-2393
September 1957	ORNL-2411
October 1957	ORNL-2426
November 1957	ORNL-2437
December 1957	ORNL-2458
January 1958	ORNL-2471

Approved by Alvin M. Weinberg, Director