MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES
3 4456 0365205 4

[redacted]

ORNL-3321
C-44a – Nuclear Technology – Materials
M-3679 (26th ed., Rev.)

147

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

JUNE 1962

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]

Downgrade to Unclassified After July 6, 1963

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

[redacted]

ORNL-3321

Copy No._____

Contract No. W-7405-eng-26

OAK RIDGE NATIONAL LABORATORY
STATUS AND PROGRESS REPORT
JUNE 1962

DATE ISSUED

[illegible]

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

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES

3 4456 0365205 4

[redacted]

SPECIAL NUCLEAR MATERIALS PROGRAM..... 1
Chemical Plant Criticality Studies..... 1
REACTOR DEVELOPMENT PROGRAM..... 1
GAS-COOLED REACTOR PROGRAM..... 1
Operation of the EGCR with GCR-3 Fuel Elements..... 1
Measurement of Conversion Ratio in EGCR..... 2
Heat Transfer of EGCR Fuel-Element Cluster..... 2
GCR-ORR Loop No. 2 Construction..... 3
Temperature Distribution Along a Roughened-Surface Fuel Element..... 3
Friction Factors on Roughened Surfaces..... 3
High-Temperature Measurements for Gas-Cooled Reactor... 3
Irradiation Effects on Beryllium Oxide..... 4
Compatibility of Steel with CO-CO_2 in Helium..... 4
Simulated Fuel-Element Studies with Chipped UO_2 Pellets..... 4
Stability of Carbon-Coated Uranium Carbide Particles... 5
THORIUM UTILIZATION PROGRAM..... 5
Thoria Pellet Development..... 5
Fuel Cycle Development: Chemistry of Solid Fuel Element Materials..... 6
Fuel Cycle Development: Fuel Preparation..... 6
Fuel Cycle Development: Fuel Fabrication Studies..... 7
Thorium-U^{233} Fuel Rod Fabrication Facility..... 8
MOLTEN-SALT REACTOR PROGRAM..... 8
MSRE Design..... 8
MSRE Procurement and Construction..... 8
Development of Components and Systems..... 8
MSRE Pump Program..... 9
Metallurgy..... 10
Instrumentation Development..... 10
Reactor Analysis..... 11
REACTOR SAFETY..... 12
Release of Fission Products on Out-of-Pile Melting of Reactor Fuels..... 12
Release of Fission Products on In-Pile Melting of Reactor Fuels..... 12
Deposition and Removal of Fission Products from Gases.. 13
Nuclear Safety Pilot Plant..... 14
FUNDAMENTAL HEAT TRANSFER AND FLUID DYNAMICS..... 14
Thermal Transient Studies..... 14
Boiling Studies..... 14
MARITIME REACTORS PROGRAM..... 15
Iodine Filter Testing..... 15
Shielding Survey..... 15
Water Chemistry..... 16
NUCLEAR TECHNOLOGY AND GENERAL SUPPORT..... 16
Power Reactor Fuel Processing: Corrosion..... 16
Power Reactor Fuel Processing: Solvent Extraction..... 17
Power Reactor Fuel Processing: Sulfex Process..... 17
Power Reactor Fuel Processing: Uranium Dicarbide Hydrolysis..... 17

[redacted]

[redacted]

Power Reactor Fuel Processing: Zirflex Process..... 17
Radioactive Waste Treatment and Disposal: High-Activity Waste Calcination..... 18
Radioactive Waste Treatment and Disposal: Low-Activity Wastes..... 18
Radioactive Waste Treatment and Disposal: Disposal in Deep Wells..... 19
Radioactive Waste Treatment and Disposal: Disposal in Natural Salt Formations..... 19
Radioactive Waste Treatment and Disposal: Engineering, Economic, and Hazards Evaluation..... 19
Radioactive Waste Treatment and Disposal: Clinch River Study..... 20
Radioactive Waste Treatment and Disposal: Fundamental Studies of Minerals..... 20
Dissolver Solution Analyses: High-Sensitivity Coulometric Titration of Uranium..... 20
Dissolver Solution Analyses: Elimination of Molybdenum Interference in the Coulometric Titration of Uranium..... 21
Dissolver Solution Analyses: Determination of U(IV) and U(VI) by Controlled-Potential Coulometric Titration..... 21

PHYSICAL RESEARCH PROGRAM..... 22
PHYSICS AND MATHEMATICS RESEARCH..... 22
High-Energy Physics..... 22
High-Energy Electronuclear Machines: Cyclotron Analogue II..... 22
Basic Studies of Cryogenic Electromagnets..... 22
Physics of Fission..... 22
Scintillation Spectrometry and Instrument Development..... 23
High-Voltage Experimental Program..... 23
Multiparticle Medium-Energy Nuclear Physics: The 86-Inch Cyclotron..... 23
Multiparticle Medium-Energy Nuclear Physics: Heavy-Particle Physics..... 24
Electronuclear Machines: The Isochronous Cyclotron.... 24
Stable Isotopes Development..... 24
Special Separations..... 24
CHEMISTRY RESEARCH..... 24
Volatility Studies: Pilot Plant..... 24
Equipment Decontamination..... 25
Protactinium Chemistry..... 25
Surface Studies: Chemical Development..... 26
Instrumentation for Analytical Chemistry Research..... 27
Radiation Chemistry..... 27
Transuranium Elements..... 28
Uranium-232 Preparation..... 29
Target Preparation..... 29
METALLURGY AND MATERIALS RESEARCH..... 29
Research and Development on Pure Materials..... 29

[redacted]

[redacted]

Theory of Alloying..... 30
CONTROLLED THERMONUCLEAR RESEARCH..... 30
Plasma Physics and Arc Research..... 30
DCX-1 Facility..... 31
DCX-2 Facility..... 31
Vacuum..... 31

BIOLOGY AND MEDICINE PROGRAM..... 32
ENVIRONMENT RADIATION STUDIES..... 32
Terrestrial Ecology: Evaluation of Fission Product Distribution and Movement in White Oak Drainage System..... 32
RADIOLOGICAL PHYSICS, HEALTH PHYSICS, AND INSTRUMENTATION... 32
Internal Dose Estimation..... 32

TRAINING, EDUCATION, AND INFORMATION PROGRAM..... 33
Professional Health Physics Training..... 33

ISOTOPE DEVELOPMENT PROGRAM..... 33
Recovery of Fission Products..... 33
Radioisotope Research and Development..... 34

OPEN LITERATURE PUBLICATIONS..... 35

FOREIGN VISITORS..... 36

[redacted]

[redacted]

OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

JUNE 1962

This Status and Progress Report summarizes the unclassified portion of the Laboratory's work. Some of the topics are included every month, but the majority are reported on a bimonthly schedule.

SPECIAL NUCLEAR MATERIALS PROGRAM

Chemical Plant Criticality Studies. — The material buckling of a mixture of an aqueous solution of 93% U^{235}-enriched $UO_2(NO_3)_2$ and borosilicate glass Raschig rings was found to be negative. The uranium concentration of the solution was 415 g/liter. The rings (1.56-in. OD, 1.25-in. ID, 1.75 in. long) had a natural-boron content of 5.9 wt %. A random array occupied approximately 24% of the volume of the 48-in.-diam test vessel in which the exponential experiment was performed. The result, which assures that even an infinitely large volume of the mixture is subcritical, was not unexpected, because of an earlier observation of low apparent source-neutron multiplication by 386 kg of U^{235} in these conditions. The information may serve as a basis for specifying operations with enriched-uranium solutions.

REACTOR DEVELOPMENT PROGRAM

GAS-COOLED REACTOR PROGRAM

Operation of the EGCR with GCR-3 Fuel Elements. — Problems associated with a transition of the EGCR fuel loading from the design elements to the advanced clad elements of the GCR-3 design are being examined. The transition would begin with the testing of a few elements in the open core and would lead to operation with a full loading of GCR-3 type elements. Because of higher power per channel, the core would generate its full rated power of 84.3 Mw with fuel elements in only 72 of the 232 fuel channels. Therefore the possibility of operation with a two-zone core is being considered to provide for testing advanced elements at high power ratings without the commitment of an entire core loading and to extend the exposure to the original EGCR elements.

[redacted]

A difficulty is that four to eight control rods must normally be fully inserted for shaping the power distribution, and thus the amount of excess multiplication actually available for burnup is limited. For a given pattern of fully inserted rods, the allowable range of continuous control provided by bank operation of the remaining rods is ~2.5%. This will permit the central core zone to accumulate an exposure of 1500 to 2000 Mwd per metric ton of uranium, but the lower-power EGCR elements will accumulate only about 500 to 700 Mwd per metric ton in this period. If the EGCR elements are to accumulate, for example, an exposure of 5000 Mwd per metric ton during the two-zone experiments, at least three patterns of fully inserted rods must be employed during each of two or three loading cycles for the central zone. The selection of rods that produce the correct reactivity decrement and power flattening is quite limited, and it appears unlikely, at present, that the reactor can be controlled in this way. Other means are available, however, such as burnable poison in the high-power zone, and these must be examined. Although a significant thermal test of the advanced fuel elements would be provided, it is not yet clear that the power distribution and reactivity could simultaneously be controlled for a long enough period to result in a significant extension of exposure for the original fuel elements.

Measurement of Conversion Ratio in EGCR. – Calculations were performed to obtain data for use in planning a proposed experiment to measure the conversion ratio for the EGCR during the physics phase of the startup program. The U^{238} absorption rate will be determined from counting the Np^{239} gamma rays, and the U^{235} absorption rate will be determined from fission product activity. Previous experiments of this type have required chemical extraction of the neptunium to reduce the fission product background in the Np^{239} counting. New coincidence counting techniques eliminate the necessity for any chemical separation and require rather simple, conventional counting instruments. In essence, the coincidence between a pair of Np^{239} gamma rays is used to discriminate against the fission product background. The calculations indicate that the necessary foils must be irradiated for ~1 hr at a flux of 10^8 neutrons cm^{-2} sec^{-1} in order to produce the desired counting rates.

Heat Transfer of EGCR Fuel-Element Cluster. – Analysis of data on the effects of Reynolds modulus variation on the thermal properties of the EGCR fuel-element cluster is continuing. The asymptotic value of the Nusselt modulus at the downstream end of the cluster agrees within a few percent with that predicted by the Dittus-Boelter equation ($N_{Nu} = 0.023\ N_{Re}^{0.8}\ N_{Pr}^{0.4}$) over the range N_{Re} = 40,000 to 63,000. In this same range the asymptotic values of the Nusselt modulus for the peripheral tubes just upstream of the midcluster spacer are consistently below the predicted curve but run parallel to it.

Experiments in which the two clusters are rotated with respect to each other are continuing. Data obtained for 15, 30, and 90° relative rotation are now being reduced.

GCR-ORR Loop No. 2 Construction. — The two main-loop throttle valves were installed in GCR-ORR loop No. 2. The installation of the constant air monitor in the cell was completed. The sampling system is approximately 80% complete. Modifications to the side-stream loop-dump system are approximately 50% complete. The evaporator level-element manifold was fabricated and is now undergoing operation tests and leak checks. Installation drawings were completed and work was started on installation of the cell refrigeration unit in its new location. The chromatograph gas-supply manifolds were fabricated.

Assembly of the 18-in. plug (in-pile section) was completed, and a hot shakedown test is under way. One aluminum cold finger was completed, and a spare unit is about 20% complete. The latch mechanism was fabricated.

Temperature Distribution Along a Roughened-Surface Fuel Element. — The temperature variations between turbulence-promoter ribs or wires on a fuel element were calculated from data reported in USAEC report IDO-28573. The data were obtained by mass-transfer techniques and showed a coefficient of variation between ribs of about 3 to 1. These data were applied to an element 0.85 in. in outside diameter with a linear heat rating of 80,000 Btu hr^{-1} ft^{-1}. The ribs were assumed to be 0.006 in. high with a pitch of 0.096 in. The results showed a temperature variation of less than 20°F, with the maximum temperature about 10°F above the average.

Friction Factors on Roughened Surfaces. — Friction factors were determined, with air, for an annulus consisting of a 0.750-in.-diam rod wrapped with a 0.0063-in. wire on a 0.091-in. pitch inside a 1.245-in.-ID sleeve. The Fanning friction factor was determined over the Reynolds number range 200 to 280,000. For the range of interest in GCR-ORR loop No. 1, 20,000 to 40,000, the friction factor was very nearly constant at 0.011.

High-Temperature Measurements for Gas-Cooled Reactor. — Studies of the stability of bare-wire refractory-metal thermocouples were continued. Tungsten vs tungsten–26% rhenium thermocouples were tested at 1200°C in flowing helium containing low oxygen concentrations. These thermocouples failed almost immediately when exposed to helium containing 100 ppm O_2. Electromotive force measurements were erratic within 2 hr, and a high emf of reversed polarity occurred in one case. Such results are thought to be due to spurious emf's developed between metals and oxides. Additional tests in helium containing 10 ppm O_2 gave emf drifts almost immediately. After about a 2-hr exposure, the emf error was +0.68 mv (34°C). Emf profile measurements showed that the nonhomogeneity that caused the errors occurred in the regions of the wires exposed at 950 to 1150°C. The thermocouples failed within 18 hr.

Exposure of the thermocouples to helium containing 100 ppm CO at 1750°C showed that the emf's of both wires were affected. When equal-size wires were used, the wires from two manufacturers gave about the same results. As might be expected, 0.010-in.-diam (30-gage) wires drifted much faster than 0.20-in.-diam (24-gage) wires. The tests were performed in the graphite-helix furnace at 1750°C with 30-cm^3/min gas

flow through the furnace. Rates of emf drift of individual couples varied considerably. Drifts of about +3.0 mv (180°C) in 90 hr were representative for 24-gage wires. The thermocouples with 30-gage tungsten–26% rhenium wire vs 24-gage tungsten wire drifted at rates of about 4.0 mv (235°C) in 24 hr. Greater drifts of 7.0 mv (400°C) in 24 hr occurred when both wires were 30 gage.

Recent tests in helium containing CO indicated that emf outputs of pure rhenium and iridium–60% rhodium wires were not appreciably affected by exposures at 1750°C. Therefore, pure rhenium wires are being used as reference wires. Subsequent tests showed that both tungsten and tungsten–26% rhenium wires drifted rapidly in the same direction during exposure. The drift of the alloy is more rapid than that of the pure tungsten; therefore the resultant drift for the thermocouple is slower than for the single wires. In one case it was shown that the 24-gage tungsten wire drifted over 8.0 mv in 160 hr.

Irradiation Effects on Beryllium Oxide. – Beryllium oxide specimens irradiated in experiment No. 6 are being analyzed. Fracturing, dimensional change, and gas generation appear to be similar to those found after earlier irradiations. These relatively large (1.180-in.-diam, 3-in.-long) low-density (2.52 g/cm^3 or 84% of theoretical) specimens did not show abnormalities that could be attributed to cladding or absence of cladding. No stress failures were observed at points of changes in geometry, such as thermowell cavities.

Experimental assembly No. 7 was removed from the ETR after an exposure of 15,554 Mwd to an estimated fast-neutron dose of 10^{21} neutrons/cm^2. Every capsule was found to be intact, although there is some evidence of internal pressure buildup and metal grain growth in some capsules. The capsules will be opened for examination of the irradiated BeO.

Experimental assemblies 8 and 9 were delivered to the ETR, and new increased-capacity control equipment was installed. Irradiation is scheduled to begin with the June 11 reactor shutdown.

Control-test capsules were fabricated and the required operating equipment was constructed for a series of thermal-history tests. The control capsules are to be tested with the same temperature, time, and cycling history as that of the irradiated BeO in order to determine which of the observed changes are caused by irradiation.

Compatibility of Steel with CO-CO_2 in Helium. – Two tests of approximately 850 hr duration have shown that the oxidation rates, as well as the extent of carburization, are greater for alloy GT-45 (18% Cr–14% Ni–2.5% Mo–4% Cu) than for type 304 stainless steel when exposed to CO-CO_2 mixtures in helium at temperatures up to 1000°C. In addition, the results show that as the ratio $(P_{CO})^2/P_{CO_2}$ is increased, the carburization of GT-45 increases, whereas the carburization of type 304 stainless steel decreases.

Simulated Fuel-Element Studies with Chipped UO_2 Pellets. – Chipped UO_2 pellets supplied by Westinghouse were clad with stainless steel and thermally cycled 200 times between 1450 and 500°F. The diameters of the pellets varied along the 10.5-in.-high stack from 0.7065 to 0.7080 in.

The 0.020-in.-wall cladding collapsed during cycling because of the 300-psi pressure differential imposed.

Ratchetting of the cladding was observed from the onset of cycling, in contrast to previous results. The ratchetting phenomenon was unexpected but may have resulted from the ovality of the tubing being sufficient to cause collapse of the cladding about the pellets at the high temperature of the cycle. Further, the chipped surfaces of the pellets provided the rough contact surface between the cladding and each fuel pellet that has been demonstrated to be a factor in fuel-element ratchetting. After the test, circumferential ridges were present in the cladding along the element axis at positions corresponding to pellet-to-pellet interfaces. In addition, protrusions were found along the cladding surface at intervals and positions that corresponded to the chipped places on the pellets. The protrusions may be due to accumulations of UO_2 fines in the chipped areas that inhibited complete local collapse of the cladding. This possibility will be investigated when the capsules are sectioned for examination.

Stability of Carbon-Coated Uranium Carbide Particles. — A major concern with respect to coated fuel particles is the possible migration of uranium through the coatings. One technique used to determine the extent of migration involves heat-treating fuel elements containing pyrolytic-carbon-coated uranium carbide particles before neutron activation. If uranium migrates through the coating to a distance within the recoil range of the outer surface of the particles, recoil Xe^{133} is expelled into the graphite matrix and released rapidly during postactivation annealing at 1000°C. Results of a number of tests on fuel elements containing fuel particles with both laminar and duplex coatings indicated that such preactivation heat treatment for several hours at a temperature of 1700°C or higher usually resulted in a higher rate of release of Xe^{133}, but some anomalous results were obtained. The increased release could also be caused by the cracking of a few coatings during heat treatment.

To resolve these questions, a low-voltage radiographic technique was used for determining the extent of uranium migration. Two samples of coated uranium carbide particles (3M 101-61 and HTM-2) were heat-treated for 95 hr at 1600°C. The major difference between the samples was that the 3M sample had a laminar coating and the HTM sample had a columnar coating. After heat treatment the particles were placed directly on a sheet of film and subjected to 10-kv x rays. Duplicate exposures were made on as-received particles for comparison. Significant uranium migration was found to have occurred in both laminar and columnar coatings. The penetration by uranium was nonuniform in both types of coatings. Further, it was very rapid in some areas of some particle coatings and undetectable in others. This observation suggests that the migration occurs along defects in the structure.

THORIUM UTILIZATION PROGRAM

Thoria Pellet Development. — Two hundred grams of sintered ThO_2 spheroids was prepared for the Battelle Memorial Institute pellet

development program. In this work an attempt will be made to coat the 1/8-in.-diam spheres with Al_2O_3 as a means of obtaining increased attrition resistance. The shapes were prepared by cold-forming cubes to a density of 4.1 g/cm^3, isostatically pressing at 35,000 psi, tumbling to 0.187 in., and sintering in H_2 for 40 min at 1750°C. The spheres were then polished by wet ball-milling with Al_2O_3 grog for approximately 30 hr.

Fuel Cycle Development: Chemistry of Solid Fuel Element Materials. — The determination of phase transitions in the system ThO_2-UO_2-O_2 by inference from dissociation pressure measurements is hampered by the time-consuming necessity of demonstrating establishment of gas-solid equilibrium. The use of low-density, high-surface-area solids should accelerate the exchange of oxygen between the solid and the gas phase. A method of preparation is being tested which should yield such materials, inferred by analogy with the behavior of ThO_2; it consists of coprecipitation of Th and U(IV) oxalates (isomorphous) in the desired Th/U ratio followed by thermal decomposition of the oxalate at comparatively low temperatures. An equimolar solution of thorium nitrate and uranyl sulfate was fumed with concentrated H_2SO_4 to remove the nitrate ion, the U(VI) ion was reduced to U(IV) on a mercury cathode, and the mixed oxalates were precipitated by adding hot oxalic acid solution. Portions of the washed and dried oxalate were decomposed under helium at 415, 500, and 610°C respectively. The dried oxalate contained 0.2% U(VI) and 0.4% sulfate. The Th/U mole ratio was 1.25. All oxide samples showed cubic structure by x-ray diffraction, with marked improvement in the sharpness of the pattern with increasing firing temperature. The 415°C-fired oxide had a surface area of 5.2 m^2/g, an order of magnitude lower than would have been anticipated from an analogous pure ThO_2 preparation. Additional studies will involve variations in chemical parameters, temperatures, and Th/U ratios.

In a continuing study of the determination of the chemical composition and crystal structure of ThO_2-UO_{2+x} equilibrium phases, eight ThO_2-UO_{2+x} solid solutions (10 to 80 mole % ThO_2) were equilibrated in air at 190°C for six months and quenched. As expected, the total oxygen content was still lower than for those equilibrated at 288°C (ORNL-3223, p 4). X-ray and metallographic methods demonstrated that in each annealed specimen two phases were present, though resolution of identifying lines in x-ray diffraction films was possible only for thoria concentrations of 60 and 70 mole %. With the same methods, it was established that on oxidation of ThO_2-UO_{2+x} solid solution with thoria content less than 40 mole % at 190°C, β-U_3O_7, reported by Westrum and Gronvøld (1962), is produced in small amounts as the second phase, while the oxidation product formed on annealing the solid solutions at 288°C is U_3O_8.

Fuel Cycle Development: Fuel Preparation. — In the preparation of sol batches containing up to 7 kg of ThO_2, conductometric titration of small samples of thoria suspended in nitric acid or uranyl nitrate was found to be a convenient and reliable method of determining the optimum nitrate/thorium mole ratio. At the end point, the pH of a 2 M ThO_2 sol is ~3.3 and does not vary significantly when the sol is diluted with H_2O or when concentrated by evaporation.

Procedures proposed for mixed thoria-urania sol preparation in the BNL Kilorod Facility are being tested and modified. It was found that wetted thoria denitrator product can be charged without difficulty into the circulating $UO_2(NO_3)_2$ solution. Good mixing was achieved in the critically safe slab tank. Tests for uranium segregation and analytical accuracy with samples of mixed sol and mixed oxide products showed that product compositions can be adequately controlled and verified.

Seven 39-in.-long rods loaded with sol-gel ThO_2-UO_2 to a bulk density of 8.8 g/cc and four 11-in.-long pins containing ThO_2-PuO_2 passed all inspection tests and were shipped to Chalk River. These will go into the NRX reactor for irradiation in a holder, replacing the one with eight 11-in.-long fuel pins loaded with sol-gel ThO_2-UO_2. The eight pins removed will have accumulated 10,000 to 17,000 Mwd per ton of thorium since April 1961. Based on center-line temperatures and the design heat generation rate of capsules in the ORR poolside facility, the thermal conductivity of the vibratorily compacted sol-gel ThO_2-UO_2 is 1.2 to 1.5 Btu hr^{-1} ft^{-1} $(°F)^{-1}$ at average fuel temperatures of 2400 to 2000°F.

A sol-gel procedure for preparing rounded thoria blanket pellets produced three 100-g batches of pellets having hourly attrition or breakage loss rates of < 0.1% in spouted-bed tests. Rounded 10 wt % uranium-thorium oxide pellets having average diameters of 100 μ and ~30% residual shrinkage were prepared by the sol-gel process.

Fuel Cycle Development: Fuel Fabrication Studies. – Twelve batches of ThO_2–3.4 wt % UO_2, produced by the sol-gel process in pilot plant runs, were evaluated for suitability as feed material for vibratory compaction. One of the most significant variables examined in this group of experiments was the point in the feed-material preparation process at which the oxide was comminuted to the required particle distribution. It was demonstrated that material sized after calcination at 1150°C could be compacted to densities 2 to 4% higher than similar material sized before firing. It was also shown, however, that particle mixtures in which only the coarse fraction was produced prior to calcination could be compacted to the same high densities (approximately 90% of theoretical density) as could similar particle mixtures containing material sized entirely after firing. These observations suggest a method by which improved yields of calcined particles with the desired size distribution can be attained. Compaction experiments with two particle-size distributions have indicated that densities of 89 to 90% of theoretical can be produced. This compares favorably with the densities obtainable with mixtures which contained three particle-size ranges.

Gamma scanning of a group of 45-in.-long, 0.50-in.-OD fuel rods has revealed density decreases of 2 to 6% near the top of most of the rods. Experiments are in progress to establish the source of this problem and to eliminate it.

Two 5/16-in.-diam, 11-in.-long vibratory-compacted fuel rods exposed in the MTR process water to an estimated burnup of 13,000 Mwd per metric ton are being examined. Information available to date indicates that no significant dimensional changes or gross changes in fuel structure occurred at linear heat ratings of approximately 40,000 Btu hr^{-1} ft^{-1}.

Thorium-U^{233} Fuel Rod Fabrication Facility. — Phase I construction of the fuel rod fabrication facility in cell 4, Building 3019, is approximately 70% complete; phase II is approximately 30% complete. Final design of all equipment except the fuel rod gamma-scanning device has been finished, and equipment fabrication or procurement is well advanced. The vibratory-compaction assembly has been tested and is being modified. A mockup of the powder-conditioning line is being assembled.

MOLTEN-SALT REACTOR PROGRAM

MSRE Design. — Drawings showing the drain-tank and coolant-cell-salt piping were approved. Design of the remote-handling jack for separating freeze flanges was completed. Drawings were issued for the reactor-cell assembly jig and the auxiliary-line layout for the fuel pump. Design of the fuel- and coolant-pipe support is nearing completion. All remaining drawings on the process system have been itemized; drafting is being concentrated on those items which will be needed first when installation begins.

Electrical design was completed on the 480-v one-line diagram, as well as design of control panels for heater circuits. Furnace design for the fuel pump is 40% complete. Other electrical layout is proceeding on schedule.

Design of the vapor-condensing system is 75% complete. Component cooling-system design is 95% complete. Out-of-cell air, gas, water, and steam piping layout is 50% complete.

Wiring diagrams of the main control board were completed and checked, and interconnection wiring diagrams were started. Specifications have been issued for most instruments, valves, pressure switches, etc.

MSRE Procurement and Construction. — Progress continues to be made in all areas of the procurement and construction of the MSRE. The project, overall, is approximately 38% complete as of June 30. The fabrication of components is approximately 30% complete, and the site preparation work is approximately 70% complete.

Machine work is complete on all heads for reactor components. The cylindrical sections for all components have been seam-welded and are being machined. Component assembly was started.

The second pour of concrete for the drain-tank cell was made. The reinforcing steel was set, in preparation for the final concrete pour. Progress continued to be made on the modification to the 24-ft-diam containment vessel. The cell penetrations are 75% complete, and the joint preparation was made for the 7-ft extension to the vessel.

Work continued according to schedule on the outside construction work. The concrete work on the filter pit was approximately 60% complete, and forms were set and the base was poured for the stack.

Development of Components and Systems. — Flow measurements in the full-scale MSRE reactor vessel mockup showed that the rotational flow in the cooling annulus along the reactor vessel wall is almost completely changed to radial flow in the bottom head by the antiswirl vanes. Tests

with granular solids in the lower head indicated that they are gradually entrained by the fluid moving radially inward above them. Heat transfer coefficients measured at one point indicated that the cooling of the head will be adequate. Results of earlier tests in the 1/5-scale model were somewhat different, in that some swirl persisted in the lower head. This difference was demonstrated to have been caused by an irregularity in the vanes of the flow model. The purely radial flow appears to provide better solids removal and improved heat transfer.

The engineering test loop continues to operate with flush salt at 1100°F. The graphite for use in the loop was received and was found to be structurally inferior; however, it is being processed for installation. Further tests on the 8-in.-diam graphite access port indicate that 30 scfm of air flow, with removal of 3 to 4 kw of heat, will establish a dependable frozen seal. It was necessary to thaw the seal in order to open the port after draining the loop.

Design of the reactor-grade sampler-enricher system was completed, and procurement of some components was initiated. Fabrication of a system to test most of the design is 50% complete.

The MSRE control-rod test has operated successfully at 1200°F for 13,500 cycles. Instances of rods sticking in the thimble during the first 1500 cycles were attributed to insufficient clearance and excessive internal restraint in the flexible element. After these conditions were corrected, the test operated without incident.

The MSRE freeze-flange joint with an octagonal ring was thermal-cycled from 150°F to above 1200°F 25 times without difficulty. For the last three cycles the leak rate from the helium buffer zone was constant at 5×10^{-5} cc/sec at 150°F ambient temperature and 2.5×10^{-6} cc/sec at 1250°F bore temperature. No permanent flange distortion was observed, and the maximum hot distortion was less than 15 mils. Attempts to freeze stagnant salt in the bore of the flange with the adjacent heaters off and the pipe heaters at 1200°F showed that this could not be accomplished with the flanges radiating heat only to ambient air.

A satisfactory brazed joint was made in a piece of pipe that had contained molten coolant-salt flow and still retained a few grams of solidified salt.

The detailed design of tooling and in-cell equipment for the remote assembly and disassembly of MSRE freeze flanges was completed. The operations involved are: (1) removing clamps from and replacing them on the flanges, (2) stowing the clamps, (3) forcing the flanges apart and back together, (4) removing and replacing the gasket assembly, (5) removing and replacing temporary flange covers, and (6) properly aligning the piping for flange assembly. This tooling is designed to operate with periscope-type viewing and dry-maintenance portable shielding.

A six-unit thermocouple junction box complete with 1/8-in.-OD stainless steel leads was tested for remote maintainability and was found to be satisfactory.

MSRE Pump Program. — Assembly of the hot-test stand for the prototype fuel pump was completed, and testing was started. Assembly of the rotary element for the fuel pump was completed, and room-temperature testing was started.

The design of the vessel for the drive motors to be used with the fuel and coolant pumps was approved. The leak-test procedures for the vessel were resubmitted by the vendor and are being reviewed; the welding procedures are being requalified by the vendor prior to resubmission. Drawings for the coolant pump tank were approved and released for fabrication.

Two volute and two impeller castings of INOR-8 for the coolant pump were weld-repaired satisfactorily. One impeller casting for the fuel pump was satisfactorily weld-repaired, and one volute casting is being weld-repaired. The founder will ship two volute and two impeller castings for the fuel pump by June 22.

The Inconel centrifugal molten-salt pump (PKP type) continues to operate satisfactorily after 3700 hr in an endurance test with molten $LiF-BeF_2-ThF_4-UF_4$ (65-30-4-1 mole %) at 1225°F, 1950 rpm, and 510 gpm.

Metallurgy. — Welding and brazing procedures were established for tube attachments on the MSRE heat exchanger. Several joint designs and fabrication conditions were investigated in four test assemblies. An automatic Heliarc welding procedure which uses no filler metal was selected from this work. An 82% Au–18% Ni braze metal was preplaced as rings in grooves surrounding the tubes. The optimum brazing temperature was found to be 1010 to 1035°C; however, rate of temperature rise within the limits of 75 to 425°C per hour had no significant effect on quality. A 12-in.-diam assembly with 90 tube joints was fabricated to investigate size effects.

Ultrasonic inspection of this heat exchanger joint is being developed; a Lamb-wave scanning probe and a solid couplant are being used. A theoretical angle for Lamb-wave generation was calculated, and a probe was fabricated. This equipment is being tested and calibrated.

Instrumentation Development. — The initial phase of long-term testing of the ball-float level transmitter (developed for use in measurement of molten-salt levels in the MSRE pump bowl) was completed with a four-month test at 900 to 1300°F. During the last three months, the temperature was maintained at 1200°F and the molten salt was held at a constant level. The maximum variation recorded during this period was 0.1 in. Span of the instrument is 5 in. Tests made at the end of the four-month period revealed no evidence of calibration shift or change in the performance characteristics of the instrument. After completion of the performance tests, the instrument was drained and the temperature was cycled three times between 500 and 1000°F. The instrument was not damaged by this repeated heating and cooling.

A model of the conductivity level probe (developed for use in single-point detection of level in MSRE storage tanks) was fabricated, and preparations are being made for testing this device under simulated service conditions.

Design of an alarm unit for use with the differential scan-alarm system (being developed for use in monitoring temperatures on the MSRE during heatup) was completed, and a prototype alarm unit was tested. Except for occasional random noise bursts, which apparently originate in the mercury commutating switch and which produce spurious alarms, the scan-alarm system operates satisfactorily. An effort is being made to

determine the cause of this defect. Also, the feasibility of replacing the mercury switch with a commercially available dry-reed commutating switch is being investigated.

Developmental testing of mechanical methods of attaching mineral-insulated sheathed thermocouples to thin-wall tubes on the MSRE radiator is continuing. Results indicate that errors can be limited to 50°F under MSRE operating conditions if sufficient contact area, clamping pressure, and insulation are provided. A braze-metal saddle, shaped to fit the contour of the tube, was required at the hot junction to obtain enough contact area. The effects of oxide film buildup do not appear to be objectionable, provided that sufficient contact surface and clamping pressure are maintained.

The stability of calibration of weld-attached metallic-sheathed thermocouples under long-term test at MSRE temperatures continues to be satisfactory. In furnace tests the maximum change in calibration for any of seven mineral-insulated thermocouples is less than 2°F after more than 3300 hr at 1250°F. Eight MSRE-type thermocouples were installed on the MSRE engineering test loop. After 45 days of operation at temperatures between 1100 and 1200°F, the test couples attached on the pipe wall agreed within 2°F with a nearby reference couple in a well.

Nine additional couples were installed on the MSRE pump test loop. These couples will be operated under conditions closely resembling thermocouple installations on MSRE piping.

Reactor Analysis. — ZORCH, an IBM 7090 code for computing the kinetics of the MSRE, was developed and is being checked out. ZORCH is an improvement over earlier kinetics codes in that it calculates the temperature distributions in the core fuel and graphite and uses weighted average temperatures for determining effects on reactivity. Input provides for arbitrary variation of reactivity and inlet temperature. Output includes power, core pressure, axial temperature profiles, and nuclear-averaged temperatures. Preliminary calculations show that during steady operation at 10 Mw, with 1175°F inlet and 1225°F outlet fuel temperatures, the fuel nuclear average temperature is 1210°F and the graphite nuclear average temperature is 1267°F.

Effective delayed-neutron fractions in the MSRE were calculated, taking into account the energy and spatial source distributions. Taking the actual delayed-neutron fraction to be 0.64%, the effective fraction in the static core is 0.67%. The increase is due to the relatively low energies of the delayed neutrons, which affect the nonleakage probabilities. Calculations assuming uniform fuel velocity in the core predict that when the fuel is circulating at 1200 gpm, the delayed neutrons emitted in the core will amount to 0.38% of all the neutrons. The effective delayed-neutron fraction for this case is only 0.31%, however, because the distortion of the spatial source distribution by circulation reduces the nonleakage probability for the delayed neutrons below that for prompt neutrons.

Calculations were made which indicate that if the 1 mole % thorium specified for the fuel salt is omitted, the critical uranium concentration (93% U^{235}) at 1200°F will be reduced from 0.26 to 0.15 mole %. Removal of the thorium would result in an increase in the temperature coefficient of reactivity from $9 \times 10^{-5}/°F$ to $12 \times 10^{-5}/°F$.

REACTOR SAFETY

Release of Fission Products on Out-of-Pile Melting of Reactor Fuels. – Recent development studies in out-of-pile melting of UO_2 have been directed toward attempts to work with much larger fuel samples. In this manner it is hoped that more realistic experiments will be performed in the simulation of reactor accidents and that the fission product release data may be extrapolated with more confidence for reactor hazards analysis.

A provisional scaleup factor of 10^3 for melting highly irradiated UO_2 has been obtained by the use of induction-heated thin-wall tungsten crucibles. Compatibility of tungsten with molten UO_2 has been the key factor in allowing freedom in melting. Early results also show that a W vs W–26% Re thermocouple may be used to ascertain the melting temperature of irradiated UO_2. Unirradiated UO_2 was observed to melt at 2830°C in one experiment. Highly irradiated UO_2 will be melted in a new hot-cell containment facility. With the larger experiments, it is expected that fractional release data for some elements will be lower than the corresponding data obtained in the microscale.

A second, perhaps more important, development for extrapolation confidence, consists in heating UO_2 rods with a central tungsten resistor. This design will be extended to a cluster of at least seven rods, six of these surrounding a single irradiated center rod. The cluster will be heated to partial melting of the outer rods and to complete melting of only the center one. Such a spatial array provides a maximum similarity to a partial reactor meltdown including the plateout surfaces of nearby hot fuel rods. Experiments carried out so far have shown an interesting effect in the extrusion of molten UO_2 by internal gas pressure from cracks along the lengths of the bare fuel following melting of the cladding. The origin and nature of this gas are being determined by spectrometric analysis.

Release of Fission Products on In-Pile Melting of Reactor Fuels. – Radiochemical analysis has been completed for experiments 2, 3, 4, and 5 on the in-pile melting of UO_2 fuel specimens. Satisfactory material balances were obtained for individual fission products and for uranium.

The percentages of various fission products released from the UO_2 in the puddle on the floor of the fission-heated furnace were:

Experiment No.	Time Molten (min)	Fission Products Released from UO_2 Puddle (%)					
		Sr^{89}	Zr^{95}	Ru^{106}	I^{131}	Ba^{140}	Ce^{144}
2	10	99.8	87.9	89		99.7	93.1
3	5	92.8	98.5	40		96.1	89.0
4	5	71.8	55.5	80	99.6	94.1	57.6
5	5	82.0	70.9	84.2		89.2	72.6

The noble gases and iodine were released essentially quantitatively. More than half of each of the other fission products was released. One-third to one-half of the UO_2 was transferred from the puddle to other regions of the furnace. Fission product release was generally greater in experiment 2, which was held molten longer, and had been irradiated at 1200°C for 24 hr after the cladding was inadvertently melted. This experiment had produced crystalline deposits of uranium oxide not found in the other experiments. The release of Sr, Ba, and Ce was substantially greater than had been observed in out-of-pile experiments. This difference may result from the longer molten period or the higher melting temperature in the absence of BeO.

The percentages of various fission products and of fuel which escaped from the furnace were:

Experiment No.	Fission Products and Fuel Escaped from Furnace (%)						
	Sr^{89}	Zr^{95}	Ru^{106}	I^{131}	Ba^{140}	Ce^{144}	UO_2
2	21.1	0.1	3		26	0.6	4.0
3	1.1	0.1	0.1		0.5	2.1	0.07
4	2.5	0.04	5	91	0.9	0.4	0.04
5	2.0	0.03	5.0		3.2	0.6	0.05

The Ba^{140} in experiment 2 was found in the charcoal trap, where it had been carried as Xe^{140} during irradiation of the UO_2 without cladding.

Although most of the fission products were released from the UO_2 puddle, they were almost entirely retained inside the high-temperature zone of the furnace, with the exception of iodine and the noble gases. This result suggests that effective removal of fission products from gases may be expected by high-temperature UO_2-covered surfaces near the melted zone in reactor accidents.

Experiments 6 and 7 are assembled and scheduled to be run in June and July. The irradiation facility for building up fission products in encapsulated UO_2 specimens is scheduled for installation in July.

Deposition and Removal of Fission Products from Gases. – Previously, the development of a method for radioactive vapor and aerosol characterization and its application to determining the efficiencies of various filter media with regard to very fine particles were discussed. The method, applicable to vapors and to particles having diameters up to approximately 1 μ, involves calculation of the corresponding diffusion coefficients from radioactivity measurements made after deposition of the aerosol in channels operating under laminar-flow conditions.

In the filter-media experiments, the particles have been either those emanating from a hot stainless steel ribbon or the now-well-defined 0.004-μ aluminum oxide particles. They were labeled with I^{131} and carried in an air stream at 5 fpm. Both high-efficiency particulate filters and a charcoal bed (3/4 in. deep, 6 × 16 mesh) have been tested. In the charcoal-bed test, the inlet stream included I^{131} vapor plus I^{131}-labeled Al_2O_3 particles. As expected, the iodine as vapor was almost completely

removed, but only about 75% of the particles were removed. The efficiency of high-efficiency particulate filters also appeared relatively low for particles in this size range, but additional experiments are needed for both types of media to further clarify the situation with respect to very fine particles.

Plans are to continue the filter-media efficiency studies, and also preparations are being made for extension of the use of this tool, for characterizing aerosols, to plant-size equipment designed for removal of radioactive particulate matter.

Nuclear Safety Pilot Plant. – Design and procurement of equipment is on schedule. The model containment vessel was designed, bids were requested, and process flowsheets were essentially completed. Instrumentation requirements were established, and a plasma-torch heat source was purchased for furnace development work. Design of the furnace is nearing completion. Small-scale tests with a borrowed plasma torch showed that melting of UO_2 could be performed by this means and that the character of the particulate matter in the off-gases was similar to that obtained with induction heating. Filters and traps were designed, and a detailed analysis of remote-handling procedures was started.

FUNDAMENTAL HEAT TRANSFER AND FLUID DYNAMICS

Thermal Transient Studies. – A four-channel magnetic-tape recorder was obtained to record, at low speed (7.5 rpm), the signals from fast-response thermocouples located at the wall-to-fluid interface of the test section of a horizontal pipe. The signal record can be replayed at a higher speed to alter the signal frequency sufficiently for use with the Panoramic frequency analyzer. Preliminary data obtained on surface-temperature fluctuations during forced-convection boiling of water are being analyzed.

Boiling Studies. – Additional data were obtained in the study of pool boiling inside single, horizontal, open-ended, electrically heated tubes immersed in water, and evaluation studies of the data were continued. The previously developed two-term correlation for predicting subcooled boiling burnout was further strengthened by the inclusion of additional literature data. In the boiling term of this correlation, considerable variation of the coefficient about the mean (±21%) was noted. While this is believed to result from the intrinsically random nature of the boiling process, no direct evidence to support this contention is available. A series of repetitive pool burnout tests with water under carefully controlled conditions was initiated. Comparison of the scatter in the results with the known precision of the experiment should give the desired information.

The study of heat transfer and critical heat fluxes with potassium in forced-convection saturation boiling in vertical tubes was resumed with the new high-flux boiler. This boiler consisted of a stainless steel tube jacketed with copper disks to provide (1) a measure of the heat flux and the inside tube surface temperature, (2) high thermal flux

at the boiling interface through the converging geometry, and (3) adequate protection against physical burnout. Heat flux levels of the order of 250,000 Btu hr^{-1} ft^{-2} were achieved. Operation at higher levels is restricted by pumping limitations. A single measurement of the critical heat flux (221,000 Btu hr^{-1} ft^{-2} at a flow of 7.7×10^4 lb hr^{-1} ft^{-2}) was obtained which, as for the low-flux measurements, agreed reasonably well with water data obtained under similar geometric and flow conditions. The heat transfer coefficients show some systematic deviation with flux level and day of operation.

MARITIME REACTORS PROGRAM

Iodine Filter Testing. — In-place tests were conducted on board the NS "Savannah" for determining the iodine-removal efficiency of the adsorber units installed in the reactor compartment main and auxiliary ventilation systems. Prior to the in-place tests, a series of laboratory experiments showed that the charcoal-adsorber medium in the auxiliary ventilation system was capable of an iodine-removal efficiency of 99.84 ± 0.1%. In the in-place tests, elemental iodine labeled with I^{131} tracer was injected into the duct leading from the reactor compartment to the filter-adsorber assembly. Samples of air entering and leaving the filter assembly were radioassayed. The iodine was injected over a period of 15 min, and the sampling was continued for 2 hr at the ambient temperature of the reactor compartment.

Two tests were conducted on the auxiliary ventilation system with different amounts of tracer. No radioiodine was found downstream from the adsorber within the limits of detection. The limits of sensitivity were such that 1/1400 and 1/15,000, respectively, of the amount of iodine collected upstream could have been detected in the downstream collector. The results of the tests indicated that the charcoal adsorber assembly, as installed, is capable of the iodine-removal efficiency observed in laboratory tests.

A similar in-place test was conducted on the reactor compartment main ventilation system. The silver-copper mesh iodine adsorber had been installed on December 14, 1961, and had been operated almost continuously up to the time of the in-place test, June 2, 1962. In this test two downstream iodine collectors were used, one immediately following the air-cleaning assembly and the other on the deck of the ship at the base of the stack. The amount of iodine was only negligibly lower in the air sample taken at the latter point. The observed iodine-removal efficiency was 95%. This result is consistent with laboratory tests on silver-plated copper mesh in air at room temperature.

Shielding Survey. — The survey of the NS "Savannah" reactor shield was conducted May 20 to May 25, 1962, during a cruise of about 2800 miles. All measurements were made with the reactor at essentially full power.

Prior to the start of the survey, the secondary shield was marked in a 3-ft grid. The general procedure consisted in making measurements at each of the grid points with both thermal-neutron and gamma-ray dosimeters. After each grid-point count, the detector was swept slowly over

the shield surface between grid points to detect any small area of unusual leakage. The entire shield was covered in this manner for both thermal neutrons and gamma rays. Fast-neutron dose rates were measured at carefully chosen locations over the shield in sufficient quantity to permit a good mapping of the ratio of fast to thermal neutrons. With these measured ratios, the thermal-neutron dose rates observed over the entire shield can be used to give a good estimate of the fast-neutron dose rate.

In addition, several special points were surveyed for which predicted dose rates were available from the Babcock and Wilcox Company shield design calculations. Finally, an experiment was performed for determining the effect of the accidental loss of water from the "deep tanks" beneath the reactor of the "Savannah." In this experiment, fast-neutron, gamma-ray, and thermal-neutron detectors were positioned in three locations at tank-top level: fore and aft of the reactor shield on the vessel's center line and approximately amidships at one side of the shield. Counts were made at closely spaced, regularly timed intervals as the water was being pumped from the tanks. After the tanks were emptied, the counters were interchanged, and the experiment was repeated while the tanks were being refilled.

Although reduction of the shielding survey data is not yet complete, it appears that the shield contains no significant voids or cracks and that the stringent dose-rate specifications have essentially been met. In addition, the data obtained by emptying the deep tanks indicate that no significant hazard will arise because of accidental emptying.

Water Chemistry. — A high-temperature filter unit for testing the filtration and ion exchange properties of magnetite and other inorganic materials was installed in the ORR loop sample station. When in operation, hot water flows continuously (~0.08 gpm) through the unit, causing the temperature to rise to 400 to 425°F. Samples are withdrawn upstream and downstream of the filter cartridge.

Two filtration tests were carried out by using natural magnetite previously cleaned with carbon tetrachloride and acetone. The percentages of activity removed were estimated from the short-lived Mn^{56} activity and the residual longer-lived Mn^{54} + Co^{58} (~0.82 Mev) and Co^{60} + Fe^{29} (~1.31 Mev) activities. In general, the initial test results were encouraging, especially since the major portion of the activity removed was probably in the form of dissolved material. Flow rate and quantity of filtrate taken did not have a pronounced effect on the fraction of activity removed by the magnetite.

NUCLEAR TECHNOLOGY AND GENERAL SUPPORT

Power Reactor Fuel Processing: Corrosion. — Aluminum, grade 6061-T6, gained weight when exposed for 4 to 48 hr in refluxing HNO_3–1 M HF when the molarity of the HNO_3 was 14 M or greater. When the molarity of the HNO_3 was decreased or the exposure temperature was lowered, the corrosion increased sharply and occurred at the interface or in the solution instead of in the vapor phase. In refluxing HNO_3–1 M HF,

maximum rates were 85 (vapor), 105 (vapor), 3000 (interface), and 11,500 (solution) mils/month when the molarity of the nitric acid was 13, 12, 10, and 4 M respectively. In 14 M HNO_3–1 M HF at room temperature, the maximum rate (solution) was 65 mils/month. In 14 M HNO_3–5 M HF, maximum rates (solution) were 4.4 and 294 mils/month at reflux and room temperature respectively.

Power Reactor Fuel Processing: Solvent Extraction. – The gross gamma decontamination factor was approximately 500 when Thorex pilot-plant U^{233} product after adjustment to solvent extraction feed conditions was extracted with either 2.5% tributyl phosphate in diethylbenzene or 2.5% di-sec-butyl phenyl phosphonate in diethylbenzene. The extract was scrubbed with 0.8 M $Al(NO_3)_3$, which was 0.1 N acid deficient. The decontamination during startup was nearly the same as at equilibrium.

Power Reactor Fuel Processing: Sulfex Process. – In 19 Sulfex decladding experiments with type 304 or 316 stainless-steel-clad UO_2 pellets, 93 to 96% of theoretical density, that had been irradiated to levels of 1545 to 28,200 Mwd/ton, all specimens were passive in 200% excess of 4 M H_2SO_4. The reaction had to be initiated with iron wire. Losses of uranium and plutonium to the decladding reagent were generally about 0.05%, although in isolated cases they were as high as 0.3%. The decladding solutions, containing up to 10^{10} dis min^{-1} ml^{-1} from activated stainless steel components, required filtration or centrifugation to remove undissolved scale, UO_2 fines, etc.

Power Reactor Fuel Processing: Uranium Dicarbide Hydrolysis. – Hydrolysis of uranium dicarbide ($UC_{1.84}$ with enough free carbon present to make a total C/U ratio of 1.97) in water at 80°C yielded 44 ml (STP) of gas per gram of carbide consisting of 45% hydrogen, 15% methane, 24% ethane, 9% saturated C_3-C_8 hydrocarbons, 6% olefins, and 0.2% acetylenes. Of the 7.54 mg-atoms of total carbon (0.52 mg-atom of free carbon) present in each gram of the original carbide, 33% was found in the gas phase, 26% in the wax, and 9% in a water- and ether-insoluble residue; the remaining 31% of the carbon presumably goes to water-soluble compounds which have not yet been isolated. The uranium product is a gray-green insoluble amorphous tetravalent compound which turns black when exposed to air.

Hydrolysis of uranium dicarbide in 6 M H_2SO_4 at 80°C was quite similar to that in water, except that the rate of gas evolution was slower and the uranium product was $U(SO_4)_2 \cdot 4H_2O$, which was insoluble in the sulfuric acid.

Reaction of the dicarbide with 6 N HCl was noticeably different; 56 ml (STP) of gas was produced consisting of 66% hydrogen, 11% methane, 12% ethane, 3% saturated C_2-C_8 hydrocarbons, 6% olefins, and 2% acetylenes. The carbon distribution in the HCl reaction was 25% in the gas phase, 58% in the wax, 8% in a water- and ether-insoluble residue, and 9% unaccounted for. The uranium product was soluble uranous chloride, and the reaction proceeded at about the same rate as the water reaction.

Power Reactor Fuel Processing: Zirflex Process. – Size distribution of PWR UO_2 (93 to 95% density) irradiated to between 9700 and 12,900 Mwd/ton averaged 98.9% larger than 10 mesh, 0.4% smaller than 100 mesh,

and the remainder between 10 and 100 mesh. Solvent extraction runs with the modified Purex flowsheet confirmed earlier results, giving uranium losses to the raffinate between 0.001 and 0.012% and plutonium losses between 0.07 and 0.12%; the addition of sulfuric acid to make the feed 0.1 M in sulfate had essentially no effect on losses to the raffinate. Uranium decontamination factors from beta- and gamma-emitting contaminants ranged between (2.1 to 5.4) $\times 10^4$ and 7.1×10^3 to 1.9×10^4 respectively.

Radioactive Waste Treatment and Disposal: High-Activity Waste Calcination. — Aluminum borophosphate glasses were prepared which incorporated up to 17% Darex waste oxides, represented volume reductions of 2.5 to 3.2, had bulk densities between 2.44 and 2.74 g/cc, and had initial softening points of 750 to 850°C. Attempts to incorporate 25% Darex waste oxides in similar glasses resulted in products which became devitrified to crystalline or segregated glassy materials.

In a semicontinuous fixation of 28 liters of TBP-25 waste in lead phosphate glass, ruthenium volatilization was about 16%, but physical entrainment was only about 0.1%. Operation with a rising liquid level and a minimum inventory of unmelted solid appears to be advantageous for ruthenium control. A container of a similar glass held at 900°C for four weeks slowly developed a vacuum which eventually reached 27 in. H_2O.

Type 304L stainless steel coupled to titanium was corroded at an apparently uniform rate of 77 mils/month during the final 2 hr (900°C) of a TBP-25 (aluminum nitrate) waste evaporation-glass fixation cycle, compared with an estimated 61 mils/month when exposed alone. The coupled titanium specimen was corroded at a rate of 31 mils/month.

In an engineering-scale test with simulated TBP-25 waste, 70% of the mercury in the waste volatilized from the pot calciner during calcination; 20% was removed in a 150°C trap located between the calciner and off-gas condenser, 15% plated out in the 450°C off-gas line, and 30% went into the condensate.

Chemical and material balance flowsheets were prepared for the pot-calciner pilot plant which is to be installed at Hanford. It was demonstrated that the currently designed pots can be satisfactorily sealed by welding. A test rack was designed for testing calciner feed pumps, and a design study indicated the practicability of installing radioactive-cell equipment for evaporation-calcination tests in an existing facility originally intended for self-sintering experiments with high-activity wastes.

Radioactive Waste Treatment and Disposal: Low-Activity Wastes. — In bench-scale equipment, circulation of 5 to 10% of the clarifier solids to the flash mixer lowered the total hardness (as $CaCO_3$) of the clarifier effluent to 8 to 14 ppm. Addition of copperas in the flocculator instead of in the flash mixer lowered the value to 5.2 ppm.

The presence of 1.7 to 2.4 ppm of Turco 4324 in ORNL tap water had no noticeable effect on the total hardness of the effluent, but 3.2 ppm increased hardnesses in the flash mixer and clarifier effluent to 44 and 37 ppm respectively. Operation of this process must be demonstrated on

ORNL waste water, which contains 4 to 10 times as much dissolved CO_2, Na^+, Cl^-, F^-, and total solids as tap water and even higher ratios of phosphate.

Vermiculite, used to remove cesium and strontium from the clarifier effluent, performed best at impracticably low flow rates. At 0.25 ml cm^{-2} min^{-1}, 50% cesium breakthrough was reached at 1000 bed volumes, although the decontamination factor from strontium was ~1000 up to 1500 bed volumes. CS-100 resin, on the other hand, gave decontamination factors of 1100 and 1800 for cesium and strontium, respectively, up to 1500 bed volumes at flow rates of 5 ml cm^{-2} min^{-1}.

Radioactive Waste Treatment and Disposal: Disposal in Deep Wells.— A tentative flowsheet, an operating procedure, and a layout for the hydrofracture experiment have been prepared.

The attainable pumping rates were estimated for well casings of various sizes. It was concluded that:

1. A 5-in. casing would probably be adequate.
2. Bottom-hole injection pressure is a quite significant variable and should be experimentally determined before pumping equipment is procured.
3. Extra pumping capacity can probably be supplied more cheaply than can a larger-diameter casing. This possibility should be verified by a cost comparison.
4. Limited available information indicates that for multiple injections the required injection pressure increases sharply with the number of injections. The extent of this pressure increase should be determined promptly, since an extreme pressure rise will seriously limit the future utility of this means of waste disposal.

The low-level wastes presently being considered for disposal by deep-well injection require containment for about 200 years. Intermediate-level wastes, requiring containment for 450 years, may be considered for disposal by this method if low-level wastes can be handled satisfactorily. It is unlikely that high-level wastes resulting from chemical processing of spent reactor fuel elements will be released to the ground without first immobilizing the fission products.

Strontium-90, the major contributor to the hazard of the wastes, will move only 1 to 10% as fast as the transporting water because of ion exchange. Precipitation reactions will provide additional restrictions to radiostrontium movement. Cesium-137 will be expected to move much more slowly than Sr^{90} because of highly selective exchange reactions. Ruthenium-106 movement is not likely to be hazard-controlling.

Radioactive Waste Treatment and Disposal: Disposal in Natural Salt Formations. — Salt samples from the Carey mine at Hutchinson, Kansas, have been observed to fracture at about 250°C and release trapped moisture which totals 0.106 to 0.293% of the weight of the salt. Salt from the Carey mine at Lyons, with a moisture content of 0.048 to 0.100%, breaks up at a lower temperature, about 215°C.

Radioactive Waste Treatment and Disposal: Engineering, Economic, and Hazards Evaluation. — A cost study of the shipment of calcined solids in 6-, 12-, and 24-in.-diam cylinders was completed. Costs were lowest for shipment in the largest casks considered manageable, which were 60

in. in inside diameter and contained four 24-in.-diam, nine 12-in.-diam, or thirty-six 6-in.-diam cylinders. Some wastes would have to be aged before shipping to allow decay of generated heat. The longest decay time required was 11 years for acidic Purex waste in 24-in.-diam cylinders.

Weights and costs of casks at minimum ages were about 100 tons and $50,000 for iron casks, 80 tons and $120,000 for lead casks, and 65 tons and $650,000 for depleted-uranium casks. Shipping costs were lowest in all cases for lead casks, and ranged from 0.70×10^{-3} mill/kwhr (electrical) for acidic Purex waste in 24-in.-diam cylinders for a 1000-mile round trip to 32×10^{-3} mill/kwhr for reacidified Thorex wastes in smaller cylinders for a 3000-mile round trip.

Radioactive Waste Treatment and Disposal: Clinch River Study. — The Subcommittee on Bottom Sediments, Clinch River Study, met on May 10, 1962, to compile and evaluate all available data pertaining to deposition of, and radioactivity associated with, bottom sediments of the Clinch and Tennessee Rivers. A draft of the evaluation is in preparation for review in early July 1962.

A contract has been awarded to Sprague and Henwood, Inc., Scranton, Pennsylvania, to collect bottom sediment cores in 14 sections of the Clinch River from miles 1.3 to 32.5, in two sections of the Emory River, and in one section of Poplar Creek. The distribution of radioactivity, particle size, and mineralogy of the bottom sediment are to be studied by use of these cores.

A piston core sampler has been developed and tested for use in coring of fine sediments. The sampler was tested in water 12-1/2 ft deep. Core recovery was 73 and 81% for bottom sediments 5.5 and 4.3 ft deep.

Radioactive Waste Treatment and Disposal: Fundamental Studies of Minerals. — A highly selective sorbent for strontium has been prepared by heating gibbsite above its decomposition temperature of 150°C. At pH 10 and with a solution 0.1 M in $NaNO_3$ and 1×10^{-5} M in $Sr(NO_3)_2$, the distribution coefficient ($\overline{K}_d$) ranged from 4000 to 40,000 when the solid/solution ratio was varied from 0.001 to 0.05 g per 50 ml. The higher K_d for the solution containing more solids suggests that sites highly selective for strontium are present in this material. The highest surface area of 303 m^2/g was obtained for the sample heated to 200°C for 173 hr; however, the sample heated for 17 hr at 400°C sorbed strontium more effectively, even though the area was reduced to 280 m^2/g. Higher heating temperatures resulted in decreased strontium sorption and reduced surface area. Arkansas bauxite, a raw ore material for aluminum and containing gibbsite, was also found to be effective for strontium when heated to decompose the gibbsite. Mineral allocation based on x-ray diffraction analysis and thermal measurements showed the Arkansas bauxite sample to contain 60% kaolin and 40% gibbsite.

Dissolver Solution Analyses: High-Sensitivity Coulometric Titration of Uranium. — Studies have been made of the coulometric titration of uranium, ranging from about 0.8 to 8000 μg in a volume of 10 ml, utilizing

a high-sensitivity, controlled-potential coulometer. The test results were as follows:

U Taken (μg)	U Found (μg)	Error (%)	σ (%)
7990	7970	−0.2	0.1
799	795	−0.5	0.2
79.9	79.5	−0.5	0.5
8.0	7.9	−1	2.3
0.8	0.7	−10	10

Satisfactory results are obtained at low levels only if: prereduction is made at +0.160 v and reduction is made at −0.200 v vs the S.C.E.; the current consumed is corrected graphically for continuous faradaic current by recording coulombs Q vs time t and measuring Q after extrapolation to t = 0; Q is corrected for contributions from charging and faradaic impurity currents by subtracting a reagent blank obtained by carrying out a titration in the absence of uranium.

Construction of a titration vessel of 1-ml capacity is planned in order to study further the applicability of this procedure for the determination of microgram amounts of uranium.

Dissolver Solution Analyses: Elimination of Molybdenum Interference in the Coulometric Titration of Uranium. — Further studies were made of a method, described in previous progress reports, for the determination of uranium in dissolver solutions in which the uranium is extracted into a cyclohexane solution of tri-n-octylphosphine oxide (TOPO), back-extracted, and then titrated coulometrically. Attempts were made to apply this method to dissolver solutions containing molybdenum in a U/Mo weight ratio less than 10. To eliminate the molybdenum interference at this level, the uranium and part of the molybdenum are extracted into TOPO-cyclohexane, after which the uranium is back-extracted with $Na_5P_3O_{10}$. Preliminary work indicates that uranium is quantitatively recovered by two back-extractions with 0.5 M H_2SO_4 containing 6 w/v % $Na_5P_3O_{10}$. After adjusting the pH to 7–8, the uranium is coulometrically reduced at −1.35 v vs the S.C.E. The relative standard deviation was found to be 0.45% at the 5-mg uranium level. Further work will include a study of the tolerance limit for molybdenum.

Dissolver Solution Analyses: Determination of U(IV) and U(VI) by Controlled-Potential Coulometric Titration. — A method was developed by which the U(IV) and U(VI) content of a sample can be determined by controlled-potential coulometric titration. Both the oxidation and reduction titrations take place at a mercury pool electrode in a supporting electrolyte solution of pH 7 to 9, containing 6 w/v % sodium tripolyphosphate. The same sample test portion is used for both titrations. In the range 2 to 10 mg of total uranium, the error of the method is approximately 1%.

PHYSICAL RESEARCH PROGRAM

PHYSICS AND MATHEMATICS RESEARCH

High-Energy Physics. – Film obtained with the liquid-helium bubble chamber exposed to separated K^- mesons was searched for events that produced Σ^- hyperons interacting with helium. Over 100 such events were found. Roughly one-half of them produced a lambda seen to decay in the chamber. Two-thirds of them had a visible stub, which could be a proton, a deuteron, or a triton. Preliminary results of kinematically fitting these events indicate that the stub is a triton in most cases. This means that the K^- + He interaction results most of the time in a three-body breakup to a lambda, a triton, and a neutron.

High-Energy Electronuclear Machines: Cyclotron Analogue II. – Experiments were performed in the 450-kev eight-sector electron cyclotron to verify the expected effects of imperfection harmonics in the magnetic field on radial stability. The radial distribution of the beam was monitored with a radially segmented probe with sections 1/16, 1/16, 1/8, and 3/8 in. wide, in order of increasing radius. The addition of the first and the fourth harmonics in the magnetic field has no effect on the radial distribution. As expected, the addition of a second harmonic introduces strong radial instability at the radius where $\nu_r = 2$. This is the mechanism planned for beam extraction in the Analogue and in the Mc^2 cyclotron. With no second harmonic present, all the current is received on the first probe segment; with a second harmonic equal to 0.15% of the mean field, the current distribution on the four segments is as follows: first, 30%; second, 30%; third, 30%; and fourth, 10%.

Basic Studies of Cryogenic Electromagnets. – Extended tests with Nb-Zr wires in the high-field (100-kilogauss) magnet clearly demonstrate that the best results are obtained when the superconducting specimens, including joints, are cast in epoxy resin to hold them rigidly in place. In longitudinal fields, dc densities up to 8×10^5 amp/cm^2 were reached; at this level (400 amp, 0.010-in. wire) joint resistance is less than 10^{-7} ohm. In the superconducting state at high current the resistance of a 50-ft sample of Nb-Zr is less than 10^{-9} ohm; the resistivity is less than 4×10^{-14} ohm-cm, limited only by circuit noise. A 1/2-in.-ID solenoid of 0.010-in. Nb-Zr wire interwound with copper wire, equal size and turns, but not cast in epoxy resin, carries 24 amp either with or without the secondary connected. These results are inconsistent with a previously described experiment in which the current in a similar magnet, but cast in epoxy, increased from 18 amp to 30 amp when the secondary circuit was closed.

Physics of Fission. – Energy correlation measurements of particles emitted in three-particle fission have been carried out in a three-parameter experiment involving $128 \times 128 \times 4$ channels. Particles from the thermal-neutron-induced fission of U^{235} were incident on silicon surface-barrier detectors. Silicon of 600 ohm-cm was used for the heavy-fragment detectors; silicon of 3400 ohm-cm was used for the two third-particle detectors. All detectors were about 4 cm^2 in area and were operated under

essentially optimum conditions [Bull. Am. Phys. Soc. 6, 240 (1961)]. The energy of each of the two heavy fragments was recorded in 128 channels; the third-particle energies were divided into four groups in the range 6 to 30 Mev. Differences in the fragment energy-correlation spectra are observed, depending on the energy of the third particle. In particular, the shape of the fragment mass distribution varies as a function of the third-particle energy. The strong preference for emission of the alpha particles at an angle somewhat less than 90° with respect to the light fragment, as reported by Titterton [Nature 168, 590 (1951)], was observed.

Scintillation Spectrometry and Instrument Development. — Measurements of the reflectivity $R(v,\theta)$ of a second enriched (91.2%) Fe^{57} mirror for the 14.4-kev gamma rays from a Co^{57} source (unsplit Mössbauer line) as a function of source velocity v and small glancing angle θ have been made by a method previously reported [Bull. Am. Phys. Soc. 6, 443 (1961)]. For $\theta = 4 \times 10^{-3}$, R(v) exhibits the six peaks and valleys associated with the six resonant frequencies of the nuclear Zeeman effect in ferromagnetic iron. The observed asymmetry of R(v) is attributed to interference between the coherent contributions of the individual resonances with each other and the interference of these with the nonresonant electronic contribution to the forward-scattering amplitude. For $\theta = 2 \times 10^{-3}$, R(v) shows the six resonant dips expected plus an additional wide valley at v = 0. The latter may be associated with the vanishing of the magnetic moment of very thin iron films [Z. Physik 137, 169 (1954)]. A small magnetic field at the mirror aligned perpendicular to the gamma-ray direction is found to deepen the valleys due to the two $\Delta m = 0$ transitions.

High-Voltage Experimental Program. — Low-lying levels of Br^{79} have been Coulomb-excited with alpha particles of variable energies ranging from 2.5 to 7.1 Mev. The gamma rays from these levels have been measured with a 3 × 3 in. NaI scintillation spectrometer. Their energies are 218 ± 2, 264 ± 4, 302 ± 5, 307 ± 3, 308 ± 4, 393 ± 6, 399 ± 6, 527 ± 5, 545 ± 7, 609 ± 9, and 767 ± 9 kev. The spectra in coincidence with the 218- and 307-kev gamma rays were also investigated. The results of these studies indicate that levels at 218, 264, 307, 399, 526, 610, and 765 kev are populated by the E2 Coulomb excitation process. From the gamma-ray yields the reduced E2 transition probabilities for excitation of these levels were found to be (in units of $e^2\ cm^4 \times 10^{-50}$) 3.91 ± 0.33, 0.069 ± 0.006, 0.209 ± 0.018, 0.37 ± 0.05, 9.7 ± 0.9, 1.49 ± 0.23, and 11.9 ± 1.1 respectively. Since the ground-state spin of Br^{79} is 3/2, the possible spins of the excited levels are 1/2, 3/2, 5/2, or 7/2.

Radiative capture cross sections of a number of nuclei have been measured for neutrons near 30 and 60 kev with the 1000-liter liquid scintillator. These measurements increase the knowledge of capture at these two energies for almost all elements with A < 210 except gases. Effects of sample thickness were examined for gold and indium. It was found that the "sum spectrum" shape changed with sample thickness, but the computed capture cross section was essentially independent of sample thickness if the corresponding spectrum shape was used.

Multiparticle Medium-Energy Nuclear Physics: The 86-Inch Cyclotron. — Radiations from the electron-capture decay of 11-hr Pt^{189} suggest that bands are populated in Ir^{189} based on configurations $3/2^+$ [402] at ground

and $1/2^+$ [400] at 94.2 kev. The Coriolis effect may alter the rotational energy spacing of these bands. The $11/2^-$ [502] intrinsic state, at 372.1 kev, is de-excited by gamma rays of 71.6 kev (M2: $11/2^- \rightarrow 7/2^+$) and 258.3 kev (E3: $11/2^- \rightarrow 5/2^+$) to levels of the ground-state band. The E3 transition competes to the extent of only 8%. An estimate of the half-life of this state is 1 μsec. The dominant branch of electron-capture decay leads to a 721.8-kev level, which suggests a $5/2^+$ [402] orbital assignment for that level.

Multiparticle Medium-Energy Nuclear Physics: Heavy-Particle Physics. — The Tandem Accelerator was successfully used for heavy-ion experiments. About 0.04 μa of N^{5+} and 0.5 μa of O^{5+} were obtained after energy analysis. The alpha-particle spectra and angular distributions from a 30-Mev oxygen bombardment on carbon and aluminum were measured. Results are now being analyzed. The angular distributions of the transfer reaction $O^{16}(Al^{27},N^{15})Si^{28}$, leaving the N^{15} in its ground state and the Si^{28} in its ground and first excited state, were measured. Preliminary results show that the distributions peak around 55° (lab), with the ground-state peak slightly more forward than the excited-state peak. The ground-state cross section is about 50% smaller than the excited-state transfer cross section.

Electronuclear Machines: The Isochronous Cyclotron. — Since initial operation in April, the dynamics of the beam of the Isochronous Cyclotron are being studied in detail so that the remaining portions of the deflector system can be designed for the most efficient transmission of an extracted beam of high quality. The temporary rf amplifier used in initial tests is sufficient for an 8-Mev proton beam at full radius. About one-third of these protons can now be separated from the circulating beam with an electrostatic deflector, as much as 120 μa being deflected. Measurements of the quality of this beam show that there are no serious defects, such as large radial or azimuthal divergence. The components of the full-power rf system are nearly completed; installation is scheduled to begin during July. Computations on the IMB 7090 computer show that a 75-Mev beam of good quality can be extracted by means of a harmonic bump produced with the valley coils, an electrostatic deflector-system channel, and a magnetic channel of 6 kilogauss and about 3 ft long.

Stable Isotopes Development. — It has been determined that ion beam instabilities which occur during certain periods of calcium separations are related to arc conditions and to residual pressure in the system. The oscilloscope monitoring method for detection of these instabilities, which contribute to isotopic contamination, is applicable in all high-purity separations.

Special Separations. — The interiors of 60 hollow cylinders are being plated with 0.3 mg of U^{235} per cm^2, and the interiors of 30 cylinders with 60 matching end caps are being plated with 0.15 mg of Np^{237} per cm^2.

CHEMISTRY RESEARCH

Volatility Studies: Pilot Plant. — Two successful runs were made with > 15%-burnup, 6.5-year-decayed, zirconium-uranium alloy fuel. Charges

of 48 and 52 kg dissolved at 650–500°C at an HF flow rate of 150 g/min. Dissolution rates in the $NaF-LiF-ZrF_4$ salt were 1.8 and 3.1 kg/hr, respectively, based on 90% completion. Corresponding HF utilization efficiencies were 17.1 and 29.8%. The dissolver off-gas scrubber system performed well with a recycle HF flow rate of ~350 g/min. Uranium was decontaminated from volatile fission products (Ru^{106}, Sb^{125}, Nb^{95}) during dissolution by factors of 2 to > 175. The primary fission products detected in the dissolver off-gas were Sb^{125} and Ru^{106}, with only 0.4 and 0.08% of the theoretical feed activity for these two isotopes reaching the off-gas filter in run R-1.

The bulk of the uranium decontamination was obtained in the uranium recovery system (fluorination to UF_6 and sorption-desorption on NaF), with factors of $\sim 10^8$ to 10^9 from Cs^{137}, Sr^{90}, and rare earths, and of $\sim 10^5$ to 10^7 for Ru^{106}, Sb^{125}, and Zr^{95}-Nb^{95}. Specific fission product activities in the UF_6 product were below the analytical limits of detection. Fluorine utilization was low in run R-1 (2.7%), but above average in run R-2 (7.3%). In run R-1, 2.2% of the uranium was transferred to the waste salt because of excessive discharge from the absorber.

The product total cation impurity level for run R-1, < 724 ppm, was slightly lower than in the 732 ppm for the TU test series (unirradiated fuel), but in R-2 the level was higher, < 874 ppm. However, contaminants in the product for the irradiated fuel runs included plutonium (< 10 ppm), neptunium (< 234 ppm), and technetium (< 69 ppm).

Equipment Decontamination. – Development and testing of low-corrosion decontaminating solutions for carbon steel, stainless steel, and aluminum were continued. Although acidic fluoride is an excellent decontaminant for several fission products, plutonium, and americium, its corrosivity to most metals has been a drawback to wide usage. Fluoride was safely added to oxalate-peroxide, however, at increased peroxide concentrations or when aluminum ion was also added. In 0.4 M ammonium oxalate–0.2 M ammonium acetate–0.16 M ammonium citrate–0.05 M fluoride–0.05 M aluminum–0.34 M hydrogen peroxide, with 100 ppm Fe^{3+} added to catalyze H_2O_2 decomposition, carbon steel at 95°C was corroded < 0.001 mil/hr in the first 11.5 hr when the initial pH was 4.0. The rate then rose to 0.15 mil/hr for about 1 hr, after which it dropped to 0.09 mil/hr. The corrosion rate of aluminum in 0.4 M oxalate–0.2 M acetate–0.2 M fluoride–1 M H_2O_2 was 6 to 9 mils/hr, but the addition of 0.2 M Al^{3+} decreased it to 0.27 mil/hr. In 0.4 M oxalate–0.2 M acetate at 95°C and pH 4.0, carbon steel became passive at about 0.15 M H_2O_2, but when 0.2 M fluoride was added, 1.1 M H_2O_2 was required for passivity. When 0.2 M Al^{3+} was also added, passivity occurred at 0.38 M H_2O_2.

Protactinium Chemistry. – In fundamental studies on solvent extraction chemistry of protactinium, the relative extractabilities of sulfate and bisulfate from sulfuric acid by N-benzylheptadecylamine were investigated. Potentiometric titration of 1 ml of amine-diethylbenzene, previously equilibrated with aqueous sulfuric acid, in acetone indicated that the amine exists almost entirely as the bisulfate when equilibrated with 10 N sulfuric acid. The bisulfate fraction decreases approximately linearly to 60% as the sulfuric acid concentration decreases to 2 N, and the remainder of the amine is in the sulfate form. The sulfate was determined by titration with KCl solution. In the presence of excess KCl to complex

the amine, the first end point reached with standard NaOH corresponded to the bisulfate ion and the second to the total amine.

Preliminary experiments on adsorption of Pa^{233} from nitric acid solutions on freshly prepared silica gel, commercial silica gel, and unfired Vycor glass showed adsorption coefficient maxima, which occurred in 6 to 8 M HNO_3, at 830, 190, and 325 respectively. These results were obtained from 1-hr batch contacts of 20 to 40 mesh adsorbent with aqueous nitric acid containing $\sim 5 \times 10^5$ counts min^{-1} ml^{-1} of Pa^{233}.

Surface Studies: Chemical Development. – Calorimetric determinations of the heat of wetting of ThO_2 at 25°C as a function of firing temperature and outgassing temperature have yielded the following data:

Firing Temperature[a] (°C)	650	800	1000	1200
Surface Area (m^2/g)	14.7	11.5	5.64	2.20
Outgassing Temperature[b] (°C)	Heat of Immersion ($ergs/cm^2$)			
100	488		528	
150	532		563	443
200		724		444
250	622		607	443
300		837		466
350	755		703	544
400	915		708	592
450	946		742	496
500	1106			540

[a]All samples were from the same lot of oxalate-precipitated ThO_2. Portions were heated at the indicated temperatures for 4 hr and stored in bottles in 1958.

[b]Samples were held at the indicated temperature for 24 hr at a pressure of less than 10^{-5} mm Hg and then sealed in vacuum in glass bulbs for insertion in the calorimeter.

An inspection of the data indicates that ThO_2 surfaces "remember" their previous history, at least the maximum temperature to which they have been exposed. The data suggest that the greatest heats are observed with a firing temperature in the vicinity of 800°C. These observations are probably related, at least in part, to the ability of ThO_2 to adsorb water irreversibly, as reported by A. L. Draper and W. O. Milligan of Rice University.

The increase of the heat of immersion with increasing outgassing temperature is greatest for the 650°C-fired material (and decreases with

increasing firing temperature), probably because the low-fired material possesses a "dirtier" surface in its initial state.

Instrumentation for Analytical Chemistry Research. — Chronopotentiometry was evaluated as a method for determining the concentration of an electroreducible species in an aqueous solution containing variable concentrations of one or more electroreducible species. The investigation was also restricted to chronopotentiometry with constant current and with mercury cathodes. Chronopotentiograms photographed on an oscilloscope were initially measured. Because this method is too tedious and imprecise for routine analyses and in order to direct attention to electrochemical limitations, a chronopotentiometer was designed which, through the use of operational amplifier circuits, regulates the cell current to a chosen constant value, drives a timer which measures the transition time, and can also drive a chart recorder or an oscilloscope to record the entire chronopotentiogram. The performance of this instrument proved to be very satisfactory.

It was concluded that the principal limitation of chronopotentiometry for measuring concentration arises from the nature of the electrochemical process: while total cell current is held constant, the cathode potential is observed as a function of time. The usual equations relating transition time to concentration assume that all the cell current is utilized for the primary faradaic process. In practice, a variable fraction of the current is stolen by charging and surface processes and by faradaic processes for trace impurities such as oxygen. It is feasible, however, to design procedures which minimize errors from convection or from variable temperatures, current, or cathode area.

Either the method of standard addition or calibration curves can be used to determine the concentration of an ion in the presence of more easily reduced ions if electrolysis conditions are carefully chosen for appropriate ranges of concentration and transition time, and for kinetically fast processes, since, under these conditions, the calibration curve is linear and passes through the origin. However, if concentrations and current density are such that the charging current is appreciable, the calibration curve does not pass through the origin. Best chronopotentiometric results are obtained at concentrations exceeding 1 m$\underline{M}$; at lower concentrations they become increasingly unsatisfactory. On the whole, this chronopotentiometric method was not found to have significant analytical advantages over polarographic methods for the determination of concentrations of ionic species.

Radiation Chemistry. — Plots of turbidity vs concentration of KNO_3 or KNO_2 dispersed in pressed KBr disks showed discontinuities which appeared to correspond to the solubilities of KNO_3 or KNO_2 in KBr (ORNL-3176, p 28). In order to determine if such discontinuities would appear when the dispersed ion became too large to be incorporated in the KBr lattice, and thus to confirm or deny the correspondence of the discontinuity with the solubility, similar experiments were made for ions of increasing volume. The concentration at the point of discontinuity (for two wavelengths) for the selected substances, which include the neutral molecules hydroquinone and anthracene, is listed below. This concentration decreases as the volume of the dispersed substance increases.

Concentration for the Appearance of Turbidity
(ions or molecules per g of KBr)

Substance	3000 A	5000 A
	($\times 10^{17}$)	($\times 10^{17}$)
BrO_3^-	420	480
CH_3COO^-	47	95
$C_6H_6O_2$ (hydroquinone)	21	42
$C_{14}H_{10}$ (anthracene)	17	15
$(C_6H_5)_3C^+$	9	14
$C_{20}H_{10}O_5^{2-}$ (fluorescein)	3	3

It seems unlikely that the bulky, and even nonionic, substances are incorporated into the lattice, replacing bromide ions, as was proposed for the NO_3^- and NO_2^- cases. Instead, it is now believed that the behavior may be compared to that of "colloidal electrolytes" in liquid solutions, which show turbidity at a concentration coincident with that at which the solutions become thermodynamically nonideal.

This is interpreted to mean that although there is undoubtedly some true solubility of small ions such as NO_3^- and NO_2^-, the remainder of the KNO_3 and KNO_2 and all the bulkier ions and neutral molecules exist in these mixtures as adsorbed or occluded aggregates ranging upward in size from individual ions or molecules.

Transuranium Elements. — The behavior of ruthenium and zirconium in the tertiary amine—lithium chloride system for separating actinides and lanthanides is being studied. Zirconium was quantitatively extracted into 0.6 M Alamine 336 in diethylbenzene diluent from 11 M LiCl. Ruthenium was quantitatively extracted only when H_2O_2 or some other oxidant was present. In stripping with 8 to 9 M HCl, zirconium distribution coefficients (organic/aqueous) were 50 to 200. In stripping with 5 to 9 M HCl containing H_2O_2, ruthenium distribution coefficients (organic/aqueous) were ~100. In the absence of an oxidizing agent, ruthenium was partially stripped. In re-extraction of this ruthenium into fresh solvent from 5 M HCl containing H_2O_2, ruthenium distribution coefficients were only 5 to 10. This anomalous behavior of ruthenium may be due to formation of different complexes.

Decontamination factors of an americium solution of $> 10^3$ were obtained by sorption of these contaminants on Dowex 1 or Permutit SK resin from 5 M LiCl—3 M HCl containing H_2O_2 at 60 to 80°C.

In a small-scale demonstration run on the proposed ion exchange process for recovering Am-Cm from plutonium-processing raffinate, Am-Cm recovery was 99.4%, and cerium, the principal rare-earth fission product, was recovered in this product cut. Decontamination factors from Cs, Ru, Zr, and Sb were 500, 20, 100, and 33 respectively. It was estimated that

65% of the Ru, 50% of the Zr, and 40% of the Sb remained on the resin column. The run was made in a shielded cell with 120 ml of plutonium raffinate reading > 100 r/hr. The feed was adjusted to neutral 2.34 M $Al(NO_3)_3$ by evaporation. Americium, curium, and rare earths were sorbed from 15 column displacement volumes onto Dowex 1-X10, 50 to 100 mesh, resin. The resin column was washed with five displacement volumes of 8 M $LiNO_3$, and the product was eluted with seven displacement volumes of 0.65 M HNO_3.

Uranium-232 Preparation. — Four samples of high-purity U^{232} were isolated from irradiated Pa^{231} by ion exchange. Isotopic analyses of the products, containing 32.9 mg of uranium, showed 98.49 to 99.66% U^{232}, 0.0127 to 0.0311% U^{233}, 0.0032 to 0.0109% U^{235}, and 0.31 to 1.47% U^{238}. The principal product contained 21.6 mg of uranium, which was 98.90% U^{232}, 0.0127% U^{233}, 0.0095% U^{235}, and 1.075% U^{238}. In mass spectrometer analyses, a mass-231 peak was detectable in only one product, indicating < 5% Pa^{231} contamination of the uranium.

Protactinium was prepared for reirradiation to produce about 1 g of U^{232} by neutralizing the HCl-HF solutions with ammonium hydroxide, which precipitated protactinium hydroxide. The precipitate was washed and dried and then calcined at 1000°C. The oxide contained ~44 g of Pa^{231} and 9 g of aluminum and was fabricated into six aluminum-jacketed Al-Pa_2O_5 cermet targets which are being irradiated in the ORR. About 13 days will be required to produce the 1 g of U^{232}.

Target Preparation. — Deuterated polyethylene films having a lower thickness limit of 0.1 mil have been prepared by blowing molten material (similar to glassblowing). Heavier films (> 1.5 mils) are formed by hot-pressing or rolling methods.

Self-supporting targets of isotopic boron from 30 to 500 $\mu g/cm^2$ thick have been prepared. When a potential of 4 kv (negative) was imposed between an electronic grid system and the evaporating boron, ions formed during evaporation of the bulk material by electron bombardment were collimated. Films were formed in less time and with greater thickness than when fabricated without the grid.

Self-supporting elemental metal foils of scandium, vanadium, and indium have been produced by rolling techniques. The lower limits of thickness are 1 mg/cm^2 for scandium and vanadium and ~4 mg/cm^2 for indium. Scandium was found to be pyrophoric at its lower limit of thickness.

A 6-in.-diam disk was formed from 236 g of U_3O_8 highly enriched in U^{235} and sealed in a 1-mil-thick iron foil container. This target will be used in making neutron cross-section measurements. A similar disk was prepared from 491 g of highly depleted U_3O_8 which had been specially purified from daughter products.

METALLURGY AND MATERIALS RESEARCH

Research and Development on Pure Materials. — A second LiF crystal (99.99% Li^7), weighing 260 g, was produced and loaned to the Materials Research Center, Cornell University. Portions of the first crystal (ORNL-3269, p 28) were analyzed by emission spectroscopy at ORNL and found to

contain, in ppm, Ca, 35; Al, < 100; Mg, 75; Na, 75; Si, < 100; Ti, < 100. A small amount of Mn (70 ppm) was found in the upper portion of this crystal but not elsewhere. Small amounts of the purest sections were provided to the Analytical Chemistry Division for the preparation of new spectroscopic standards for LiF, and to the Chemistry Division for neutron scattering experiments. Infrared absorption spectra were obtained at ORNL on the first crystal; absorption bands for OH^- were absent from these spectra, indicating that effective dehydration of the charge preparation had been obtained. Approximately 300 g of LiF (98.06% Li^7) was dehydrated, selected by hand, and loaded into the Stockbarger furnace for the production of a third LiF crystal.

Preliminary investigation of the stability of MgO in molten salt solutions, conducted for purposes of selecting high-temperature solutions from which crystals of pure MgO can be grown, has demonstrated that MgO forms simple eutectic systems with each of the alkali fluorides. Several MgO crystals of dimensions 5 × 2 × 2 mm were grown in four days at 700°C from solution in molten NaF-LiF eutectic mixture circulated over nutrient crystals at 800°C by convection in a nickel loop. Growth of a 0.2-mm thickness on an MgO seed was obtained by evaporation of a KF-MgO solution at 1100°C. Needles of KF 9 × 1/2 × 1/2 mm were grown from the gaseous phase in the same experiment. Evaporation of a KCl-MgO solution at 1100°C produced 0.1-mm crystals of MgO. Crystals of MgO 0.1 mm on an edge were also produced by bubbling helium saturated with H_2O through molten magnesium at 650°C.

Theory of Alloying. – Low-temperature measurements of the transverse magnetoresistance of pure tungsten crystals show that the magnetoresistance increases nearly as the square of the field strength for all field directions when the current is in the directions <100> and <110>. In terms of the Lifshitz theory relating the magnetoresistance to the presence of open and closed Fermi surfaces, tungsten resembles an even-valent metal with closed surfaces and equal numbers of holes and electrons. Other directions in tungsten crystals are being investigated.

The electronic and lattice specific heats of α-zirconium with additions of antimony and cadmium have been measured at 1 to 4°K. These properties are now known for the whole series of solutes Ag, Cd, In, Sn, and Sb. The results show linear increases in electronic specific heats which are proportional to the valency of the solute. The Debye temperatures have a more complicated variation, with a minimum decrease for indium and larger decreases for the solutes Ag, Cd, Sn, and Sb.

CONTROLLED THERMONUCLEAR RESEARCH

Plasma Physics and Arc Research. – Injection of ions random in energy and direction into a mirror magnetic geometry appears to form a plasma which, when the source potential is removed, decays with an e-folding time longer than that permitted by a dissipative instability. Helium and deuterium plasmas have been measured. It has been argued that the rate of energy loss from ions to thermal electrons will not permit a decay constant as long as 0.5 sec for the deuterium plasma (ORNL-3286, p 27). However, it has also been suggested that it is very difficult for electrons

below 8 ev to lose energy. An 8-ev electron sink will permit a decay constant of about 15 msec, which is also longer than the decay by an instability. A decay constant of this length for the deuterium plasma has been repeatedly observed and is dependent on mirror ratio and background pressure, as expected. A source, other than the plasma, of the 0.5-sec decay constant for deuterium (previously reported) has not yet been ascertained.

DCX-1 Facility. – A new set of bakeable shields was installed in the intermediate vacuum region. This set is equipped with heavier-duty tubing and is spaced farther from the walls of the vacuum tank. It is believed that these changes will permit longer superheated bakeouts of the intermediate region. The new set also isolates the 14-in. diffusion pumps from the intermediate region with liquid-nitrogen-cooled chevron baffles.

When operation is with the new set of shields, but with the liner removed, pressures as low as 6×10^{-9} torr (uncorrected ion-gage reading) were recorded in the plasma region. Under these conditions, the only fast component of signal to neutral-particle detectors in the south detector port was apparently due to neutrals from dissociation of the incoming H_2^+ beam.

Several pinhole leaks were detected in the tubing on the liner, which had given satisfactory service for five months. The old tubing was stripped off and new tubing was attached. The liner is being installed.

DCX-2 Facility. – The H_2^+ component of a 100-ma ion beam was passed into the DCX-2 on April 24 and 26. It was found that the extra longitudinal compensating coils did eliminate excessive field penetration through the injection duct. Inadequate instrumentation and high pressure prevented a useful study of the trapped plasma at this time. There followed a shutdown period of several weeks to improve the vacuum and add instruments. Operation on May 14, 16, and 17 with four foil neutral detectors spaced along the axis, magnetic and electrostatic rf probes, and a beam pulser showed evidence of trapping and of ion plasma pumping, but indicated also that the central field was not flat. (A mirror-coil short was found to be responsible.) Calorimetric power measuring equipment and a fast neutral spectrometer were available, but did not operate properly. The vacuum was still unsatisfactorily high. Operation was suspended to install equipment needed for bakeout of the vacuum system, for cleaning, and for a magnetic-field survey, including the effect of the injection duct.

Vacuum. – Continued tests of the experimental surface-boiling diffusion pump were made to ascertain if any increase in pumping efficiency for hydrogen relative to air occurred as a function of boiler temperature. It was found that at a temperature of approximately 284°C, the pumping speed for hydrogen was 1.5 ± 0.3 times that for air. At 296°C boiler temperature, this ratio increased to 2.25 ± 0.4. The high thermal stability of the experimental polyphenyl ether diffusion-pump oil holds considerable promise for improved diffusion pumping of hydrogen.

Ion-gage studies on nude gages showed that in the absence of a screen in a clean glass chamber, the pressure readings were substantially the same as for the nude gage with a shielding screen. However, after the chamber was exposed to an operating nude gage long enough for a dark deposit to appear on the glass (usually several days of operation), it was

found that the gage readings obtained without the added screening element were approximately one-half of those observed in the presence of a screen. The use of a shielded screen in order to obtain consistent readings with a nude gage is recommended.

BIOLOGY AND MEDICINE PROGRAM

ENVIRONMENT RADIATION STUDIES

Terrestrial Ecology: Evaluation of Fission Product Distribution and Movement in White Oak Drainage System. — A series of soil samples was taken within the bed of former White Oak Lake and radiochemically analyzed to determine the distribution and total amount of Ru^{106} in the soil. The activity flows onto the bed from two streams that drain the intermediate-level waste pit area. Results of the investigation show that (1) as of February 1962 there was approximately 1200 curies of Ru^{106} in the bed of former White Oak Lake; (2) the highest concentrations of Ru^{106} are found in the uppermost 2 in. of soil; (3) about 75% of the activity found in the lake bed is associated with the first 2 ft of soil; (4) although Ru^{106} is being transported by ground water through the lake bed soil, a relatively small amount of it is reaching the creek in this manner.

RADIOLOGICAL PHYSICS, HEALTH PHYSICS, AND INSTRUMENTATION

Internal Dose Estimation. — Maximum permissible concentration values are estimated by using a "standard man" and thus, in general, do not take account of variations correlated with age, sex, weight, individual habits, etc. Yet a population, even an adult population, exposed to environmental contamination at a constant level would include individuals whose total doses, or annual doses, differed markedly. In general, the variations in dose would be proportional to the body burdens or organ burdens of the radionuclide in individuals of this population. The variation of these organ burdens, or concentration of the radionuclide in tissue, can be estimated by studying the concentration of naturally occurring isotopes in the tissues of individuals, since these would result from chronic exposure to trace quantities of the isotopes in the diet. The data on concentration of trace elements in tissues of grossly normal humans have been obtained and are being analyzed. To delineate as objectively as possible the extent of this dose variation for the case of chronic low-level exposure, two statistical methods have been applied.

By use of nonparametric statistical tests, a band about the sample cumulative distribution can be determined. The probability that the population cumulative distribution lies in this band is 0.95 (Introduction to Statistical Analysis, McGraw-Hill, New York, 1957). A second band gives a 95% confidence interval for each percentile of the population cumulative distribution (An Introduction to Mathematical Statistics, Ginn, New York, 1960). These tests involve no assumptions concerning the form of the distribution function. They have been applied to data for Cd in

the kidneys, Ra^{226} in the skeleton, Fe in the spleen, Al in the lungs, and Sr/Ca ratio in stillbirths.

The Federal Radiation Council and the ICRP have estimated that the majority of the population will not exceed three times the mean value. From these curves the percent of the population which would be expected to exceed this value may be asserted with 95% confidence. From such statistical tests it is possible to estimate the variation of dose due to certain nuclides for the case of chronic low-level exposure. In all cases tested thus far the fraction of the population exceeding three times the sample mean does not exceed 5%, except for one case involving the lung, where the fraction is 8%.

TRAINING, EDUCATION, AND INFORMATION PROGRAM

Professional Health Physics Training. – A three-month training period in health physics was begun by 16 AEC Fellowship students, one student from India, and one U.S. Air Force officer. Twelve of the Fellows have received extensions in order to complete the research and thesis requirements for the master of science degree at Vanderbilt University.

The Education and Training Section has been responsible for the coordination of programs for both foreign and American visitors. Thirty foreign visitors and 44 American visitors have received health physics training. These include an 11-member Japanese study team, a 13-member health specialist group from the University of North Carolina, and a 27-member Public Health Service group from the Taft Memorial Center in Cincinnati.

Samuel Helf, Picatinny Arsenal, New Jersey, has completed a one-year training period in health physics. His program was coordinated by this section.

ISOTOPE DEVELOPMENT PROGRAM

Recovery of Fission Products. – Substituted phenols continued to give good results as extractants for recovery of cesium from Purex waste solutions. A new phenol, 4-*sec*-butyl-(2-α-methylbenzyl)phenol (BAMBP), extracted cesium much more strongly than any previously tested compound. In a batch countercurrent test with 1 *M* BAMBP in diisopropylbenzene, 99.9% of the cesium was recovered from simulated Purex waste (tartrate-complexed and adjusted to pH 12.7) in five stages at an organic/aqueous ratio of 1/1. The cesium was 99.8% stripped in a batch contact with 0.05 volume of 0.05 *M* HNO_3 to give a solution containing 2.4 g of Cs and 0.2 g of Na per liter (overall Cs decontamination from Na by a factor of ~6000). In a test with 1 *M* *o*-phenylphenol in diisopropylbenzene, > 97% of the cesium was recovered from adjusted simulated Hanford tank farm supernatant at an organic/aqueous ratio of 1/1 in six stages. A study of cesium extraction coefficient dependence on phenol concentration showed that it is at least third power. In tests with concentrated basic cesium solutions, the maximum organic cesium loading was 1 mole per 6 moles of phenol.

Radioisotope Research and Development. — Cesium silicate glasses are being investigated for the retention of Cs^{137} in an insoluble form. It is expected that these glasses will find wide application in gamma-ray sources and in heat sources. Currently the properties of cesium glasses containing CaO, PbO, Al_2O_3, and B_2O_3 are being studied. One method being investigated for fabrication of the glass into rectangular rods consists in casting of the molten glass (at ~1000°C) in metal and graphite molds.

OPEN LITERATURE PUBLICATIONS

Alexeff, I., and R. V. Neidigh, "Observation of Ionic Sound Waves in Gaseous Discharge Tubes," p 1523 in Proc. Fifth Intern. Conf. Ionization Phenomena in Gases, vol 2, Munich, Aug. 28–Sept. 1, 1961.

Auerbach, S. I., "Analytical Requirements for Environmental Surveys," Nuclear Safety 3(3), 51 (1962).

Boyle, J. W., "Radiation Chemistry of Aqueous Sulfuric Acid Solutions," Radiation Research 16, 565 (1962).

Boyle, J. W., and H. A. Mahlman, "Iron III Molar Extinction Coefficients in Light and Heavy Water Solutions," Radiation Research 16, 416 (1962).

Carroll, R. M., "The Continuous Release of Fission Gas from UO_2 During Irradiation," p 110 in Symposium on Radiation Effects in Refractory Fuel Compounds, ASTM-STP-306, American Society for Testing Materials, 1961.

Davis, F. J., and P. W. Reinhardt, "Radiation Measurements over Plane Sources," Health Phys. 8, 233 (1962).

Gauster, W. F., and C. E. Parker, "Some Concepts for the Design of Superconducting Solenoids," p 3 in Proc. Intern. Conf. High Magnetic Fields, MIT, Nov. 1–4, 1961.

Gibbons, R. A., and R. J. Mackin, Jr., "Development and Study of a Highly Ionized Steady-State Deuterium Plasma," p 1769 in Proc. Fifth Intern. Conf. Ionization Phenomena in Gases, vol 2, Munich, Aug. 28–Sept. 1, 1961.

Griffin, P. M., "Echelle Grating," p 602 in Encyclopaedic Dictionary of Physics, vol 2, Pergamon, London, 1961.

Hochanadel, C. J., "Photolysis of Dilute Hydrogen Peroxide in the Presence of Dissolved Hydrogen and Oxygen. Evidence Relating to the Nature of H and OH Produced in the Radiolysis of Water," Radiation Research 16, 555 (1962).

Hochanadel, C. J., and J. A. Ghormley, "Effect of Temperature on the Decomposition of Water by Gamma Rays," Radiation Research 16, 653 (1962).

Lietzke, M. H., and R. W. Stoughton, "The Thermodynamic Investigation of Aqueous Electrolytes to 275°C," J. Chem. Educ. 39, 230 (1962).

Lundy, T. S., and J. F. Murdock, "Diffusion of Al^{26} and Mn^{54} in Aluminum," J. Appl. Phys. 33, 1671 (1962).

May, F. T., and R. A. Dandl, "A Sub-Millimicroampere Current Amplifier Utilizing an Unusual Transistor Effect," IRE Trans. on Nuclear Sci. NS-8(4), 16 (1961).

McNally, J. R., Jr., "Hollow Cathode Discharge Tube," p 454 in Encyclopaedic Dictionary of Physics, vol 2, Pergamon, London, 1961.

Melton, C. E., "Observation of NH_4^+ in the Decomposition of Ammonia on Platinum," J. Am. Chem. Soc. 84, 1491 (1962).

Merzbacher, E., "The Single-Valuedness of Wave Functions," Am. J. Phys. 30, 237 (1962).

Noggle, T. S., and J. O. Stiegler, "Fission Fragment Tracks in Thin Films of UO_2," p 47 in Symposium on Radiation Effects in Refractory Fuel Compounds, ASTM-STP-306, American Society for Testing Materials, 1961.

Postma, Herman, "DCX-1 Instrumentation," IRE Trans. on Nuclear Sci. NS-8(4), 77 (1961).

Rush, R. M., J. S. Johnson, and K. A. Kraus, "Hydrolysis of U(VI). Ultracentrifugation and Acidity Measurements in Chloride Solution," Inorg. Chem. 1, 378 (1962).

Schroeder, H. A., J. J. Balassa, and I. H. Tipton, "Abnormal Trace Elements in Man — Nickel," J. Chronic Diseases 15, 51 (1962).

Sonder, E., and S. T. Sekula, "Effects of Gaseous Impurities on the Magnetic Properties of Copper," J. Phys. and Chem. Solids 36, 93 (1962).

Stoughton, R. W., J. Halperin, R. E. Druschel, F. J. Johnston, P. M. Lantz, J. H. Oliver, and G. W. Parker, "Measurements of Thermal Cross-Sections and Resonance Integrals Using Pile Neutrons and Activation Techniques," pp 239–46 in Proc. IAEA Symposium — Pile Neutron Research in Physics, Vienna, Austria, October 1960.

Toner, D. F., and J. L. Scott, "Study of the Factors Controlling the Release of Xe^{133} from Bulk UO_2," Am. Soc. Testing Materials, Spec. Tech. Publ. 306, 86 (1961).

Trammell, G. T., "Elastic Scattering at Resonance from Bound Nuclei," Phys. Rev. 126, 1045 (1962).

Trammell, G. T., "Sound Waves in Water Containing Vapor Bubbles," J. Appl. Phys. 33, 1662 (1962).

Witkamp, M., "Breakdown of Forest Litter in Relation to Environmental Factors," Plant and Soil 15, 295 (1961).

FOREIGN VISITORS

Name	Affiliation	Citizen of
Artaud, J. G.	French CEA	France
Baur, F.	William Ainsworth, Incorporated	Switzerland
Becker, M. O.	NEKEM	Germany
Billard, F.	French CEA	France
Boer, A.	Institute for Atomic Energy	Indonesia
Boulenger, R. R.	Centre d'Etude de l'Energie	Belgium
Bullen, M. A.	Bristol General Hospital, England	United Kingdom
Burger, A. P.	South Africa Atomic Energy Board	South Africa
Burtscher, A.	Studiengeselschaft fur Atomenergie	Austria
Campbell, R. H.	United Kingdom Atomic Energy Authority	United Kingdom
Chaput, M. C.	Centre d'Etudes Nucleaires	France

Name	Affiliation	Citizen of
Cigna, A.	Casaccia Nuclear Center	Italy
Cocconi, G.	Cornell University	Italy
Colley, J. R.	South Africa Atomic Energy Board	South Africa
Cowper, G.	Atomic Energy of Canada	Canada
Craig, D. K.	University of Rochester	South Africa
Dambrine, C. P.	French CEA	France
Dautrey, R. I.	French CEA	France
de Latil, P. P.	Le Figaro Science Etavenir	France
Denilou, G.	French CEA	France
Deshpande, K. B.	University of North Carolina	India
du Toit, S. J.	South Africa Atomic Energy Board	South Africa
England, C. J.	United Kingdom AEA	United Kingdom
Felix, F. W.	Hahn-Meithner Institute	Sweden
Galvagni, R.	CNEN, Italy	Italy
Grenon, M.	Communaute Europeenne de l'Energie Atomique	France
Guieysse, L.	French Government	France
Haung, P. Y.	University of Virginia	China
Haynes, R. H.	University of Chicago	Canada
Hermans, A. M. E.	KEMA Laboratories	Netherlands
Hirata, M.	University of Tokyo	Japan
Jobson, D. A.	United Kingdom Atomic Energy Authority	United Kingdom
Lalere, J.	French CEA	France
Latham, R.	University of Cambridge	Great Britain
Lell, E. D.	Self	Germany
LeTheule, J.	French National Assembly	France
Leupen, F. F.	Netherlands Dept. of Agriculture	Netherlands
Lichtenberger, F. A.	Region d'Equipment Thermique Nucleaire	France
Lind, R.	United Kingdom Atomic Energy Authority	United Kingdom
Mehl, J.	International Atomic Energy Authority	Austria
Mehlhorn, W.	Vanderbilt University	Germany
Meranda, D.	Israel AEC	Israel
Nagino, Y.	Japan Atomic Power Company	Japan
Neuilly, M.	French CEA	France
Noda, R.	Case Institute of Technology	Japan
Pannetier, R.	French CEA	France
Pate, B. D.	Washington University	United Kingdom
Paul, M.	Gutenberg University	Germany
Perez, R.	University of Florida	Spain
Pomarola, J.	French CEA	France
Pouteaux, M.	French CEA	France
Pradel, J.	French CEA	France
Pretsch, J.	German Atomic Ministry	Germany
Prince, K.	United Kingdom Atomic Energy Authority	United Kingdom
Prot, A. C.	French CEA	France
Ratumbuysang, A. A.	Institute for Atomic Energy	Indonesia

Name	Affiliation	Citizen of
Rautenbach, W. L.	Council for Scientific and Industrial Research	South Africa
Richards, B. N.	Department of Forestry	Australia
Rothwell, E.	United Kingdom Atomic Energy Authority	United Kingdom
Schmeling, P. E.	Hahn-Meithner Institute	Sweden
Schmidt, H. W.	BBC, Krupp	Germany
Schulten, R.	BBC, Krupp	Germany
Sheldon, E.	Laboratory fur Kernphysik, Zurich	Switzerland
Shinohara, M.	Erb and Gray Scientific, Inc.	Japan
Siwac, G.	French Embassy	France
Snepvangers, J. J. M.	Reactor Centrum Nederland	Netherlands
Soebagio, R.	Institute for Atomic Energy	Indonesia
Steinmann, W.	Universitat Munchen	Germany
Sugimoto, A.	Japan Atomic Energy Research Institute	Japan
Terasawa, S.	Hitachi Central Research Laboratory	Japan
Tokizawa, M.	Nippon Atomic Industry Group Company	Japan
Vernin, E.	SAMES Corporation, France	France
von der Decken, C. B.	BBC, Krupp	Germany
Wang, L.	Solid State Radiation, Inc.	China
Went, J. J.	KEMA Laboratories	Netherlands
Wiehle, H. R.	University of Illinois	Germany
Woldringh, S. H.	Reactor Centrum Nederland	Netherlands
Wolf, H.	KEMA Laboratories	Netherlands
Wood, E.	United Kingdom Atomic Energy Authority	United Kingdom
Wormser, G.	French CEA	France
Yacob, Y.	University of Cincinnati	Iraq
Zimen, K. E.	Hahn-Meithner Institute	Sweden

Reports issues in this series during the past year:

June 1961	ORNL-3156
July 1961	ORNL-3174
August 1961	ORNL-3190
September 1961	ORNL-3211
October 1961	ORNL-3223
November 1961	ORNL-3237
December 1961	ORNL-3242
January 1962	ORNL-3256
February 1962	ORNL-3269
March 1962	ORNL-3279
April 1962	ORNL-3286
May 1962	ORNL-3307

ORNL-3321
C-44a – Nuclear Technology – Materials
M-3679 (26th ed., Rev.)

INTERNAL DISTRIBUTION

1. I. Alexeff
2. C. F. Baes
3. E. A. Bagley
4. P. S. Baker
5. C. R. Baldock
6. L. H. Barker
7. S. E. Beall
8. E. E. Beauchamp
9. P. R. Bell
10. D. S. Billington
11. C. A. Blake
12. R. E. Blanco
13. F. F. Blankenship
14. E. P. Blizard
15. J. O. Blomeke
16. A. L. Boch
17. E. G. Bohlmann
18. C. J. Borkowski
19. G. E. Boyd
20. M. A. Bredig
21. J. C. Bresee
22. R. B. Briggs
23. R. E. Brooksbank
24. K. B. Brown
25. F. R. Bruce
26. W. D. Burch
27. T. H. J. Burnett
28. A. D. Callihan
29. A. E. Cameron
30. T. E. Cole
31. C. F. Coleman
32. E. L. Compere
33. D. D. Cowen
34. J. A. Cox
35. F. L. Culler
36. J. E. Cunningham
37. R. A. Dandl
38. J. S. Drury
39. J. L. Dunlap
40. A. C. England
41. M. F. Fair
42. D. E. Ferguson
43. J. L. Fowler
44. J. H. Frye, Jr.
45. J. L. Gabbard (K-25)
46. W. F. Gauster
47. M. W. Gerrard
48. J. H. Gillette
49. W. Y. Gissel
50. H. E. Goeller
51. C. D. Goodman
52. A. T. Gresky
53. J. C. Griess
54. W. R. Grimes
55. P. A. Haas
56. C. S. Harrill
57. J. C. Hart
58. M. R. Hill
59. C. J. Hochanadel
60. A. Hollaender
61. L. B. Holland
62. R. W. Horton
63. F. T. Howard
64. A. S. Householder
65. H. H. Hubbell
66. G. S. Hurst
67. G. H. Jenks
68. R. W. Johnson
69. R. J. Jones
70. W. H. Jordan
71. P. R. Kasten
72. G. W. Keilholtz
73. G. G. Kelley
74. M. T. Kelley
75. E. M. King
76. K. A. Kraus
77. Laboratory Shift Supervisor
78. J. A. Lane
79. D. M. Lang (K-25)
80. C. E. Larson
81. R. E. Leuze
82. W. H. Lewis
83. T. A. Lincoln
84. S. C. Lind
85. R. B. Lindauer
86. R. S. Livingston
87. J. T. Long
88. L. O. Love
89. R. N. Lyon
90. R. J. Mackin
91. H. G. MacPherson
92. F. C. Maienschein

93. W. D. Manly
94. J. A. Martin
95. H. F. McDuffie
96. J. R. McNally
97. A. J. Miller
98. E. C. Miller
99. K. Z. Morgan
100. W. L. Morgan
101. F. J. Muckenthaler
102. E. J. Murphy
103. J. P. Murray (K-25)
104. R. V. Neidigh
105. M. L. Nelson
106. C. E. Normand
107. G. W. Parker
108. R. B. Parker
109. J. J. Pinajian
110. M. E. Ramsey
111. A. W. Riikola
112. L. P. Riordan
113. A. F. Rupp
114. A. D. Ryon
115. H. C. Savage
116. H. W. Savage
117. A. W. Savolainen
118. H. E. Seagren
119. C. H. Secoy
120. E. D. Shipley
121. O. Sisman
122. J. R. Sites
123. M. J. Skinner
124. A. H. Snell
125. W. S. Snyder
126. W. L. Stirling
127. R. W. Stoughton
128. H. Stringfield
129. E. G. Struxness
130. J. C. Suddath
131. W. H Sullivan
132. C. D. Susano
133. J. A. Swartout
134. E. H. Taylor
135. R. E. Thoma
136. C. D. Watson
137. B. S. Weaver
138. A. M. Weinberg
139. M. E. Whatley
140. J. C. White
141. E. O. Wollan
142. R. W. Worsham
143. R. G. Wymer
144. W. Zobel
145. A. Zucker
146-147. Central Research Library
148. ORNL — Y-12 Technical Library Document Reference Section
149. Reactor Division Library
150. Biology Library
151-338. Laboratory Records Department
339. Laboratory Records, ORNL R.C.

EXTERNAL DISTRIBUTION

340. Division of Reactor Development, AEC, Washington (J. M. Simmons)
341-349. Division of Research and Development, AEC, ORO
350. Aeroprojects, Inc. (W. B. Tarpley)
351. Allis-Chalmers Manufacturing Co. (W. F. Banks)
352-354. Ames Laboratory, Iowa State College (F. H. Spedding)
355-357. Argonne National Laboratory (A. V. Crewe)
358. Atomics International (H. Pearlman)
359. Babcock and Wilcox Co. (Atomic Energy Division)
360-361. Battelle Memorial Institute (H. W. Russell)
362-364. Brookhaven National Laboratory (M. Goldhaber)
365. Bureau of Mines, Salt Lake City (J. B. Clemmer)
366-373. General Atomic (General Dynamics Corp.) (Manager)
374-375. General Nuclear Engineering Corp. (J. West)
376. Grand Junction Operations Office (W. C. Hutchinson, Jr.)
377-378. Idaho Chemical Processing Plant (R. L. Doan)
379. Kaiser Engineers (P. D. Bush)
380-381. Knolls Atomic Power Laboratory (K. R. Van Tassell)
382-383. Lawrence Radiation Laboratory (H. Fidler)

384-385. Los Alamos Scientific Laboratory (N. E. Bradbury)
386-387. National Bureau of Standards (L. S. Taylor)
388. National Lead Company (C. K. McArthur)
389. Nuclear Metals, Inc. (A. R. Kaufmann)
390-391. Philadelphia Electric Co. (L. R. Gaty)
392. Sanderson and Porter (Atomic Energy Division)
393. Union Carbide Corporation (R. W. McNamee)
394. USAEC, Assistant General Manager (S. G. English)
395. USAEC, Division of Biology and Medicine (C. L. Dunham)
396-398. USAEC, Division of Isotope Development (P. C. Aebersold)
399. USAEC, Division of Production (F. P. Baranowski)
400. USAEC, Division of Raw Materials (J. C. Johnson)
401-415. USAEC, Division of Reactor Development (F. K. Pittman)
416-421. USAEC, Division of Research (P. W. McDaniel)
422. USAEC, Division of Technical Information (Technical Reports Library)
423-424. USAEC, Chicago Operations Office
425. USAEC, Lockland Aircraft Reactors Operations Office (H. H. Gorman)
426-427. USAEC, San Francisco Operations Office (R. Hughey)
428-429. USAEC, Savannah River Operations Office (H. Rahner)
430. Westinghouse Electric Corp. (A. Squire)
431-669. Given distribution as shown in M-3679 (26th ed., Rev.) category C-44a and those addresses shown in C-44b and C-44c which do not receive copies under C-44a