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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

MAY 1960

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

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Oak Ridge, Tennessee
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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

May 1960

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

PROGRAM 2000 — SPECIAL NUCLEAR MATERIALS

Power Reactor Fuel Processing Pilot Plant. — The evaporator loop involved in the November 1959 explosion in Building 3019 was flushed with 500 liters of 6.0 M HNO_3 containing 0.1 M boron and 0.01 M fluoride prior to its removal from the cell. The wash removed about 100 g of plutonium. The U^{233} solvent extraction system was removed from cell 3 in preparation for changes required for the power reactor program. Painting of the penthouse and the pipe tunnel is continuing. (AEC Activity 2461)

PROGRAM 4000 — REACTOR DEVELOPMENT

GAS-COOLED REACTOR PROGRAM

EGCR Control-Rod Studies. — A series of two-group, two-dimensional control-rod calculations was performed for a homogenized EGCR core at operating temperature. It was assumed that the four small experimental loops were filled with homogeneous fuel-graphite specimens having a carbon-to-uranium atom ratio of 200 and that the four large loops were filled with graphite. The rod worths and the effects of insertion of various rods and combinations of rods on the radial core power distribution and experimental loop power were determined. The results are summarized below:

[redacted]

Control Rods Inserted			Radial Core Power Distribution, P_{max}/P_{av}	Total Control-Rod Worth (Δk, %)	Average Power per Loop (Mw)
Number	Coordinates x (in.)	Coordinates y (in.)			
None			1.758		1.338
1	0	0	1.352	2.46	1.073
5	0	±24	1.201	8.7	0.682
	±24	0			
	0	0			
4	±24	±24	1.188	8.0	0.866
2	±24	0	1.637	4.0	1.144
4	±48	±48	2.149	2.6	1.48
2	±48	0	1.878	1.45	1.29
1	+60	0	1.852	0.68	1.37
2	0	0	1.622	3.8	1.007
	+24	0			

The shutdown worth of the EGCR control-rod bank was also calculated for the case of empty experimental tubes and for a typical experimental assembly with a carbon-to-uranium ratio of 200 in each tube. These calculations showed that the presence of the experimental assemblies reduced the worth of the bank of rods by $\Delta k_{eff} = 0.03$. The results of the calculations are summarized below:

Experimental Loop Condition	Number of Rods Inserted	Δk_{eff}
Tubes empty	21	0.201
Tubes empty	25	0.269
Experimental assembly in each tube	25	0.240

(AEC Activity 4141.1)

Irradiation Experiments in the ORR. – Irradiation of the second group of hollow UO_2 pellets in type 304 stainless steel capsules continued as scheduled at a reactor power level of 16 Mw. A recent set of operating temperatures and calculated total burnups is presented below:

Capsule No.	Per Cent H_2 in Thermal Barrier Gas	Temperature (°F)				Calculated Burnup (Mwd/metric ton)
		Design	High	Low	Average	
01-2	40	1300 ± 50 (av)	1380	1045	1314	1590
02-2	80	1600^{+0}_{-50} (high)	1570	1262	1460	1630
03-2	42	1300 ± 50 (av)	1350	935	1300	1670
04-2	28	1300 ± 50 (av)	1420	980	1361	1670
05-2	33.5	1600^{+0}_{-50} (high)	1580	1370	1482	1700
06-2	59	1600^{+0}_{-50} (high)	1555	1412	1488	1620
07-2	55	1300 ± 50 (av)	1475	1240	1337	1100
08-2	100	1600^{+0}_{-50} (high)	1550	1370	1467	1180

(AEC Activity 4141.1)

Irradiation Experiments in the LITR. — Five experiments on the release of fission product gases from UO_2 are in progress in the LITR. The UO_2 pellets being irradiated are fabricated of 95%-dense material. The operating conditions of these experiments are described below:

Experimental Assembly No.	LITR Facility No.	UO_2 Enrichment (% U^{235})	Central Temperature (°F)	
			Capsule a	Capsule b
L-22	C-57	10	1400	1600
L-23	C-42	15	2700	2800
L-24	C-44	20	2600	3100
L-27	C-46	15	2100	2600
L-28	C-47	30	2200	3800

(AEC Activity 4141.1)

Closed-Cycle Experiment in the ORR. — Two UO_2 samples encased in siliconized silicon carbide-coated graphite were installed in the B-9 closed-cycle facility in the ORR. The capsule has been irradiated at a flux of 7×10^{13} neutrons·cm^{-2}·sec^{-1} for more than 100 hr at a coating temperature of 1500°F. No accurate fission product leakage rate has yet been determined, but it is interesting that the activity of the nitrogen coolant has not increased above normal background. (AEC Activity 4141.1)

Examination of Irradiated UC_2-Graphite Fuel. – The graphite capsule irradiated farthest from the core of the MTR was opened without cutting into the UC_2-graphite fuel element. The element slid from the bottle by its own weight as a solid right cylinder, which visually appeared to be unchanged from the preirradiation condition. This observation is significant because thermal stress calculations indicated that the matrix might fragment or crack. (AEC Activity 4141.1)

Examination of Irradiated Beryllium. – Preliminary density and dimensional data were obtained for 16 beryllium samples that were exposed to a dose of 10^{20} neutrons/cm^2 (> 0.6 Mev) at 500 to 600°C. Some swelling is indicated by the data. (AEC Activity 4141.1)

Mass-Transfer Experiments. – Twenty-six mass-transfer tests have been completed using a full-scale model of an EGCR cluster with the ORNL-designed 60°-pad midcluster spacer and modified top and bottom spiders. A full cluster (with the same type of spiders and spacers) preceding the test unit permitted studies of the effect of relative rotational orientation of the adjacent clusters. (Mass transfer from a naphthalene-coated rod is studied as a means of estimating heat transfer.) For each run, mass-removal traverses were made on one rod location at seven axial positions along the rod; thus, 182 mass-removal profiles have been obtained. The data are being analyzed. (AEC Activity 4141.1)

THERMAL-BREEDER REACTOR PROGRAM

Homogeneous Reactor Program

Homogeneous Reactor Test. – Progress continued on HRT core modifications. Molds were made from the plastic impressions of the two holes in the core tank, and metal patches are being fabricated from these castings. A model of the No. 2 hole patch has been tested for fit in the hole. The inner surface of the core was wire-brushed in an effort to remove any scale that might be remaining on the wall.

Outside the reactor core, two low-pressure-system valves were replaced and a new secondary recombiner was installed to prevent radiolytic gas from escaping to the carbon beds. Checks of the various reactor service lines continued in order to make certain that the containment has not deteriorated. (AEC Activity 4151.3)

Reactor Analysis. – The pressure rise associated with a cold-water accident in the two-region HRT was studied for upward core-fluid flow and for reversed flow. With the reactor initially at 260°C, the analysis indicated that introduction of fuel fluid at 100°C with upward flow of fluid would cause the pressure to rise rapidly by about 400 psi; with reversed fluid flow, the corresponding pressure rise would be 550 psi. For the above calculations it was assumed that one pressurizer was available for expansion of the core fluid; connecting the core and blanket pressurizers decreased the 400 value to 210 psi and the 550 value to 375 psi. (AEC Activity 4151.1)

Homogeneous Reactor Instrumentation. – The instrumentation and control system of the HRT has been reviewed to determine the control requirements for two-region operation subsequent to patching of the holes in the core and reversal of the direction of flow through the reactor. The major control problem associated with the return to two-region operation has been found to be the control of the differential pressure and temperatures between the core and the blanket so that any leakage at the patches will be in the direction from blanket to core and at a rate below that at which condensate can be transferred from the core low-pressure system to the blanket low-pressure system. Several methods of control are under consideration. The final selection of the control method will be based on the core-to-blanket leakage rate observed during the initial operation of the reactor subsequent to patching of the core. The instrumentation and control circuitry is being revised in accordance with the control method that has been selected for the case of low leakage rate. Additional instrumentation and circuitry are being installed which will permit changeover to one of the other methods of control if necessary. (AEC Activity 4151.1)

Design. – Additional tools designed for maintenance operations at the HRT included a tool for wire-brushing the core wall to remove accumulated uranium-bearing deposits and scale and a device for measuring the thickness of the 30 and 90° conical sections of the core tank. (AEC Activity 4151.1)

Metallurgy. – The variables associated with fabricating round-ended pellets from ThO_2 powder have been studied in an effort to establish a method for making pellets to be used in the reactor blanket-material program. The methods used for making flat-ended pellets were not satisfactory for the fabrication of round-ended pellets because laminations were formed at the transition of the round end and the cylindrical section. Partial disintegration of the pellets occurred at the laminations during abrasion testing.

Pellets with several compositions and fabricated by various methods were made and submitted for abrasion testing. Both flat-ended and spherical-ended pellets were made from material calcined at three different temperatures. One set of these contained 0.5% CaO as a densifier, and another contained only ThO_2. Half the specimens were pretumbled to remove easily abraded projections. As a result of these tests, a method was selected for fabricating 5 kg of pellets. They will be made of D-40 ThO_2 powder calcined at 1425°C and will be pretumbled. (AEC Activity 4151.1)

Laboratory Corrosion Studies. – At 100°C in oxygenated deionized water, alumina specimens of 93 to 95 and of 98% purity were found to corrode at rates of 0.3 and 0.1 mil/year, respectively, during a 1500-hr period. The rates for the same period in oxygenated water at 300°C were 1 mil/year for the 93-to-95%-purity specimen and 2.7 mils/year for the 98%-purity specimen. The materials are being considered for use as a nonconducting liner in a magnetic flowmeter.

One combination undergoing examination for use in valve seats and flowmeters is aluminum oxide brazed, preferably, to stainless steel. Recently, 1000-hr tests were completed with specimens of alumina brazed to titanium which were immersed in a simulated uranyl sulfate fuel solution at atmospheric boiling temperature and in oxygenated water at 300°C. Titanium was

used for the initial tests since its coefficient of expansion is closer to that of aluminum oxide than that of stainless steel. The brazing alloys contained various percentages of beryllium, iron, titanium, vanadium, and zirconium. Preliminary results indicate satisfactory corrosion behavior of the brazing alloys in the boiling fuel solution; some doubt exists as to the suitability of the alloys in water at 300°C. (AEC Activity 4151.1)

Dynamic Solution Stability Tests. — Tests of the chemical stability of very dilute uranyl sulfate solutions at high temperatures were carried out to determine what would happen if small amounts of fuel solution leaked around the patches to be installed in the Zircaloy-2 core tank of the HRT. A 0.04 m UO_2SO_4–0.025 m $CuSO_4$–0.035 m H_2SO_4 solution was diluted 200-fold with water to produce a solution containing nominally 50 ppm uranium, 8 ppm copper, and 50 ppm sulfate, with a pH of 3.5. The tests were carried out in a stainless steel 100A loop, and during the test fresh oxygen-saturated solution was fed to and let down from the loop at a rate of 2 to 3 liters/hr until 180 liters of the dilute solution had been passed through the loop. At the end of the run the deposited uranium and copper were dissolved from the loop with nitric acid.

Tests at 300 and at 325°C yielded essentially the same result. About 50% of the uranium and 50% of the copper were deposited in the loop, presumably as oxides, since sulfate was not lost from solution. The pH of the solution let down from the loop was 3.2, as compared with 3.5 for the starting solution. Tests at lower temperatures are in progress.

On the basis of the above tests, a low leakage rate of fuel from the core of the HRT to the blanket would probably result in the precipitation of uranium and copper, possibly on the blanket side of the core, if the surface temperature of the Zircaloy-2 is as high as 300°C. (AEC Activity 4151.1)

Dynamic Blanket Materials Tests. — In the thoria-pellet test program, 26 different preparations which had been fabricated from the same 800°C-calcined ThO_2 base powder were tested to determine the effects on pellet integrity of pellet shape, additive (0.5 wt % CaO), die lubricant, ThO_2 prefabrication firing temperature (800, 1200, or 1425°C), and wet-tumbling after calcination. All the finished compacts were calcined at 1650°C.

Portions of each preparation were subjected, at room temperature, to two 1-hr exposures in a spouted-bed test (ten pellets, 0.2 fps superficial velocity, 24 fps jet velocity), to ball-milling for two 1-hr periods, or to exposure to distilled water for 72 hr in static autoclaves at 260°C.

Attrition (weight loss) rates in the various tests, expressed as per cent per hour, were: (1) in the spouted-bed tests, 0.12 to 4.3 (46% of the preparations gave rates less than 0.5); (2) in the ball-mill tests, 0.36 to 7.14 (60% of the preparations gave rates less than 1.5); (3) in the static autoclave tests (only pellets containing 0.5 wt % CaO have been tested to date), 4 to 580×10^{-5} (83% gave rates less than 40×10^{-5}).

With 0.2-in. right cylinders having flat ends, the attrition rates increased as the ThO_2 precalcination temperature was increased from 800 to 1425°C; however, the attrition rates of 0.2-in. cylinders with rounded ends decreased as the ThO_2 precalcination temperature was increased. Pellets of either shape containing CaO additive gave slightly lower rates than pure ThO_2 compacts. During 72 hr exposure in static autoclaves, 0.1 to 3.5% of

the calcium present was leached from the pellets. Pellets prepared from 1200°C-precalcined ThO_2 gave the lowest rates of leaching.

Wet-tumbling after final calcination reduced the attrition rates in the first 1-hr tests by factors of 3 to 7 for both cylinder shapes.

No correlation with die lubricant (kerosene vs polyvinyl alcohol) was observed. (AEC Activity 4151.1)

In-Pile Slurry Corrosion Loop. — A 5-gpm in-pile slurry loop designed for operation in beam hole HB-2 of the LITR has been fabricated. Out-of-pile testing of the loop, now in progress, has proceeded for 450 hr at 280°C with a thorium oxide slurry containing 0.5 wt % enriched uranium, based on thorium, which is the material proposed for in-pile use. Loop operation has been entirely satisfactory in circulating this slurry at concentrations up to 1300 g of Th per kg of D_2O.

In contrast with previous thorium oxide preparations (which contained 0.5% depleted uranium) circulated in this loop, there has been very little degradation of the particle size of the thoria used in this test. During 400 hr of circulation the average mean particle diameter has remained in the range from 1.6 to 1.7 μ.

A system to remove samples of slurry from the loop during in-pile operation is incorporated in the in-pile facility. Out-of-pile tests of this system in conjunction with the in-pile loop have been trouble-free, and slurry concentrations obtained in a series of samples have shown good agreement with calculated values of the loop concentrations, which ranged between 1300 and 500 g of Th per kg of D_2O in the tests.

The thoria—0.5% enriched-urania preparation to be used in-pile has given a longer retention of catalytic activity than the slurry of thoria—0.5% depleted urania during circulation at temperature with added palladium catalyst; half the activity remained after four weeks of circulation during tests in the prototype loop. Similar activity and life have been manifested in current tests on the actual in-pile loop loading. An in-pile radiolytic-gas generation rate of 0.06 to 0.08 mole of D_2 per hour has been estimated, and a quantity of palladium catalyst sufficient to recombine this radiolytic gas at a low partial pressure has been added to the slurry now being circulated in the loop. (AEC Activity 4151.1)

In-Pile Autoclave Slurry Corrosion Studies. — The first in-pile Zircaloy-2 autoclave experiment with a thoria-urania slurry containing 13 wt % enriched uranium (based on thorium) (experiment L5Z-153S) was irradiated for 136 hr in the LITR HB-5 facility, after a preirradiation period of 249 hr at the experiment temperature of 280°C. The slurry concentration was 820 g of Th per kg of D_2O with 0.019 m palladium added as a recombination catalyst, and an oxygen atmosphere was maintained. On the basis of an indicated flux of 1×10^{13} neutrons·cm^{-2}·sec^{-1}, the fission power density achieved in the fully inserted position was 26 w/ml. However, since excessive radiolytic gas was generated in the more deeply inserted positions, 18% of the flux at the fully inserted position was the highest at which the autoclave was irradiated for extended periods. The average power density during the in-pile period of the experiment was 2.6 w/ml.

During the first reactor shutdown after insertion, a lag in pressure response indicated partial plugging of the water-filled 20-mil-ID capillary line leading to the pressure cell, and the experiment was terminated.

Subsequent observation of the gas pressure indicated that less than a 1-psi decrease had occurred during the 172-hr in-pile period, indicating negligible corrosion. (AEC Activity 4151.1)

Molten-Salt Reactor Program

Chemistry. – For purposes of design, the fuel composition for the MSRE has been set as Li^7F-BeF_2-ThF_4-$U^{235}F_4$ (65-30-4-1 mole %), corresponding to about 5 wt % U^{235}, although it is anticipated that less U^{235} will suffice for the actual reactor experiment. This composition represents a balance between low melting point (450°C) and other physical properties, determined mainly by the proportions of Li^7F and BeF_2. The proportion of ThF_4 was somewhat arbitrarily chosen, with a view to low uranium inventory and to improved chemical behavior of the fuel.

Petrographic and x-ray examinations of quenched fluoride melts which had been equilibrated for one week at 450 to 550°C were made in order to determine the effect of substitution of small amounts of oxide, added as BeO, for fluoride. A blanket mixture containing no UF_4 showed deposition of ThO_2 when 0.3 wt % oxide was present, but not at lower amounts of oxide. When 0.5 mole % UF_4 was present (LiF-BeF_2-ThF_4-UF_4, 67-18-14.5-0.5 mole %), UO_2 rather than ThO_2 was found when 0.3 wt % oxide was present. For oxide additions of 0.8 wt % and higher, both UO_2 and ThO_2 were deposited (as a solid solution at temperatures above 500°C).

The selective precipitation, by means of small amounts of added oxides, of protactinium (up to 50 ppm) and of uranium as oxides from molten-fluoride blanket mixtures continues to give promising results as far as analyses of the supernatant liquid are concerned; separation of the precipitated phases from the liquid also appears feasible, on the basis of preliminary experiments. (AEC Activity 4152.1)

Metallurgy. – Four additional grades of graphite were subjected to the standard screening permeation test, that is, exposure for 100 hr to LiF-BeF_2-ThF_4-UF_4 (67-18.5-14-0.5 mole %) (BULT-14-0.5U) at 1300°F at a pressure of 150 psig. Two of the new grades, S-4-LB and CS-112S, met the MSRP specification that no more than 0.5% of the bulk volume of the graphite shall be filled with fluoride salt. This brings the number of potentially acceptable grades of graphite to four.

An apparatus has been developed to leak-check brazed joints of tubular graphite to metal end caps. This is accomplished by pressurizing the graphite tube with isopropyl alcohol, a material with a viscosity approaching those of fused salts. To date, two gold-nickel-molybdenum braze alloys have been tested; both leaked under pressures of less than 5 psig. The reason for leakage is being investigated, and improved joint designs will be tested. (AEC Activity 4152.1)

Corrosion Tests. – Metallographic examination has been completed on nine additional INOR-8 and seven additional Inconel thermal-convection loops operated with fluoride mixtures for one year. These tests mark the completion of all phases of the molten-fluoride corrosion program dealing with thermal-convection loops. Of the INOR-8 loops, two exhibited occasional

subsurface attack in the hot leg to a depth of 1 to 1-1/2 mils; the remainder exhibited either surface roughening, pitting, or shallow subsurface voids to less than 1/2 mil. Attack in the form of intergranular and general subsurface void formation occurred in the hot-leg regions of all seven Inconel loops. In loops which were operated at 1250°F, the depth of attack ranged from 4 to 7 mils. In the loops operated at 1350°F, the depth of attack ranged from 7-1/2 to 15 mils. Attack in the form of surface roughening and pitting to a maximum depth of 1 mil was found in the cold-leg regions of the loops. No evidence of mass-transfer deposits was found in the cold-leg regions.

Pumped corrosion loops, ten of INOR-8 and two of Inconel, have been in operation with various fluoride salts during the month. The loops have operated at high wall temperatures of 1200 to 1500°F for periods of 3 to 23 months. Operation of an INOR-8 loop was terminated after 15,140 hr for the removal and examination of the third jacketed weight-loss specimen. (AEC Activity 4152.1)

In-Pile Tests. — The second MSR graphite-fuel capsule irradiation experiment at the National Reactor Testing Station, ORNL-MTR-47, has satisfactorily completed 1000 hr of a scheduled 1600 hr of irradiation. Operating conditions are normal.

The nose end of the first experiment is in storage at Oak Ridge awaiting the availability of hot-cell facilities for disassembly. (AEC Activity 4152.1)

Pump Development. — The molten-salt pump containing a salt-lubricated journal bearing continued in operation and has completed 1000 hr in the present run. Its performance has been observed at temperatures from 1100 to 1350°F, at flows from 75 to 150 gpm, and at shaft speeds from 800 to 1400 rpm.

The molten-salt journal bearing which failed in the tester, reported last month, was examined. The failure was a result of the dam over the end of the grooves rubbing against the journal. The dam was not concentric with the bearing bore as intended. Steps are being taken to ensure concentricity in the future.

The MF-F pump endurance test has operated continuously for over 25,100 hr, circulating molten salt 30 at 1200°F. The PKP pump rotary assembly was operated at room temperature to check the performance of the seals prior to hot operation. Seal performance was found to be acceptable from the standpoint of oil leakage, but modifications are planned to enhance the benefits of gas flow in the shaft annulus on minimizing the flow of NaK vapors to the seal and of oil vapors to the system. (AEC Activity 4152.1)

Remote Maintenance Demonstration. — The salt circulating pump, pump motor, reactor-vessel mockup, heat exchanger mockup, and pipe preheaters were successfully removed from the remote-maintenance demonstration facility after molten salt was circulated through the loop. A remotely operated pneumatic bolt runner and torque tool, specially designed, was used for the first time to disassemble the 3-1/2- and 6-in. freeze-flange clamps.

Filming of the motion picture "Remote Maintenance of Molten-Salt Reactors" was completed. (AEC Activity 4152.1)

HIGH FLUX ISOTOPE REACTOR

Reactor Physics Calculations. — Further studies have been made in connection with the HFIR temperature and void coefficients of reactivity. The most recent temperature coefficient calculations indicate that the water island and control region coefficients are positive and that the fuel region coefficient is negative over the entire temperature range of interest. Above 110°F the over-all coefficient, with all regions at the same temperature, is negative, but below 110°F the calculations indicated that the particular over-all coefficient might be positive with a maximum positive change in k_{eff} of about 0.002 from 70°F. The actual existence of an over-all positive coefficient has been verified to some extent by HFIR critical experiments and by ASTR calibration experiments (ANP Doc. No. NARF-57-48T). In both these experiments the temperatures of all regions were the same. When the changes in the temperatures of the various core regions are not the same, the effective over-all coefficient is different. For instance, in the case of the HFIR the island water temperature increases only one-tenth as much as the fuel region coolant temperature during a power increase, and therefore the effective over-all temperature coefficient is less positive than, and becomes zero at a lower fuel region temperature than, if all regions increased in temperature the same amount. To illustrate the importance of the power interpretation of the over-all coefficient a case was considered in which the control plates jammed upon reaching criticality at zero power, the initial core temperature was 40°F, and the inlet temperature to all regions remained at 40°F during the subsequent power transient. The new steady-state average fuel region coolant temperature reached was 84°F, which corresponded to a power level of 124 Mw. This compares with a 400-Mw increase with all regions at the same temperature. By increasing the relative flow rate through the island and/or increasing the initial core temperature the change in power level resulting from the postulated startup accident can be reduced significantly.

Void coefficients have been calculated for the case of uniform voids in all regions of the core. The results indicate that both the over-all and fuel region coefficients are negative, while the island coefficient is positive up to about 45% voids. The calculated maximum positive change in k due to the 45% island void is 2.5%. With the 200-g plutonium target in the island the corresponding maximum change in k is reduced to about 1%. (AEC Activity 040406090)

Heat Transfer. — Four HFIR heat transfer tests with slightly flattened axially oriented spacer strips have been completed. The average maximum temperature and burnout penalties incurred by the spacers at a heat flux of 1,500,000 $Btu \cdot hr^{-1} \cdot ft^{-2}$ were approximately 26°F and 17% respectively. Further tests with axial spacers of square cross section will be made as sections become available.

An approximate thermal and flow analysis of the WTR core at time of meltdown will appear as ORNL CF-60-5-33. This memo includes certain techniques of analysis which may also be of value for HFIR evaluation. (AEC Activity 040406090)

Critical Experiments. — The neutron flux distribution measured with small BF_3-filled proportional counters, both bare and cadmium-covered, in the critical assembly supporting the design of the HFIR is not significantly different from that reported previously from measurements with fission chambers. The dependence of the flux distribution upon fuel loading was investigated by increasing the fuel solution concentration from the original 92.5 g of U^{235} per liter to 119.7 and, subsequently, to 137.1 g/liter with, in both cases, re-establishment of criticality by addition of boron to the D_2O annular reflector. The natural boron concentrations were 37 and 56 mg/liter respectively. Flux distributions measured by small fission chambers at these increased concentrations were not greatly different from the earlier results.

The hydrogen density in the water-filled central column of the experiment was changed by decrements by the addition of hydrogen-displacing materials to, finally, a complete removal of the water. The reactivity reached a maximum of approximately +6% with the hydrogen content about 40% of that of water and remained slightly positive even with all water removed. These studies were designed to investigate the hazards associated with bubble formation in the central reactor column. (AEC Activity 040406090)

Corrosion. — Studies concerned with the corrosion of type 1100 aluminum under heat flux have been extended to include a determination of the effect of acidifying the cooling water on corrosion. Also, a preliminary test with type 6061 aluminum, the preferred alloy for cladding the fuel elements, has been started. All tests have been carried out under conditions approximating those that will exist in the HFIR: heat flux, 1.6×10^6 $Btu \cdot hr^{-1} \cdot ft^{-2}$; coolant velocity through channel in the aluminum test specimen, 33 to 36 fps; outlet water temperature, 190°F (220°F in one case); pressure, 900 psi (300 psi in one case); time, ten days.

With high-purity water as the coolant and type 1100 aluminum as the test material, the temperature measured on the outside of the test specimen increased at a rate of 18 to 22°F per day as a result of a tightly adherent oxide coating forming on the water-cooled surface. In addition, "film stripping" and localized attack near the outlet of the specimen were occasionally observed. Under identical conditions with the pH of the water adjusted to 5 with nitric acid, the rate of temperature increase was 4 to 6°F per day and "film stripping," significant general corrosion, and localized attack were absent. Increasing the outlet temperature of the water (pH 5) from 190 to 220°F (hot channel condition) changed neither the rate of temperature increase nor the apparent corrosion damage.

The first test with type 6061 aluminum is in progress (water at pH 5 with nitric acid), and based on temperature measurements only, its behavior is not significantly different from that of type 1100 aluminum. (AEC Activity 040406090)

Metallurgy. — The inner section of a mark II type fuel element has been reassembled to meet additional requirements for the hydraulic flow test. Twenty-eight hundred eighty measurements have been made of the water gaps between plates with the channel-spacing gage, and it was found that (1) the mean water gap was 0.050 in., (2) 89% of the measurements were within ±0.005 in. of this dimension, and (3) three measurements were not

within ±0.010 in. of this dimension. It was noted that one of the major reasons for the out-of-tolerance was due to the plates deviating from a true involute curve in the as-fabricated condition, and a comparison of measurements at radial positions showed that 98 to 99% of the measurements were within ±0.005 in. of the mean value at the respective radial position. To rectify this situation a Marforming punch is being reworked in order to produce fuel plates which more nearly conform to true involutes.

Examination of three fuel plates, containing fuel cores consisting of 30% U-2% Si-Al, revealed three potential problems — two associated with blistering and one related to "dog-boning" of the fuel core. These plates were rolled at 600°C to a reduction in thickness of 6 to 1, after a previous prerolling of 2-to-1 reduction in thickness at 600°C. After an anneal at 500°C, two of the plates developed blisters, which in all cases but one proved to be associated with the aluminum cladding. The other was attributed to a misfit between core and frame.

Metallographic examination of transverse sections indicated that it was possible, by a knowledge of the original fuel-core dimensions, to predict the presently specified trapezoidal cross section within ±5%. Examination of longitudinal sections of two plates revealed anomalous "dog-bone" cladding thickness. The other plate showed no dog-bone effects. Reasons for this anomaly are yet unsolved. (AEC Activity 040406090)

Design: Gamma and Beta Heat Generation Rates. — A calculation was made to determine fuel plate heat fluxes resulting from after-shutdown fission product heating. Maximum heat flux values are as follows:

Decay Time (sec)	Maximum Fuel Plate Heat Flux ($Btu \cdot hr^{-1} \cdot ft^{-2}$)
1.0	42,000
10	30,000
10^2	18,000
10^3	9,500
10^4	4,000
10^5	1,300

The core coolant gamma heating rate during operation was also calculated using the same techniques but adding in the fission and capture gamma sources. The average coolant gamma heat generation rate is about 33 w/cc at the start of the cycle when no fission products are present, and 57 w/cc after the fission products build up. (AEC Activity 040406090)

Design: HFIR Fuel Plate Stability. — An analytical study for predicting the elastic behavior of involute fuel plates was completed, and a code was written to be used with the IBM-704. A comparison of the experimental and analytical results for a 40-mil plate with pinned ends was made. The analytical curve seems to correlate the experimental results for the involute plates with pinned edges very well. The theory and experiment agree on positions of maximum, zero, and minimum deflection. In the case of an

HFIR fuel plate the maximum deflection will be less than 1-1/2 mils per psi pressure differential. (AEC Activity 040406090)

MARITIME SHIP REACTOR PROGRAM

The two fuel-rod clusters that were being irradiated in the pressurized-water loop in the ORR were removed after two reactor cycles. They will be reinserted for further irradiation when the reactor power level is increased to 30 Mw. A second set of fuel rods was inserted that is identical with the set removed. The fuel-rod clusters consist of three 0.5-in.-dia, 18-in.-long fuel rods fabricated by swaging fused and ground UO_2.

Calculations were made which indicate that the worth of the control rods for the NS "Savannah" reactor would be increased by ~ 0.6% if the boron concentration were increased from 1.35 to 1.71 wt % and that the removal of the flow blocks from the NS "Savannah" core would increase the reactivity by 1.20% Δk. The latter result compares favorably with the value of 1.26% Δk measured in a critical experiment. (AEC Activity 4510)

PACKAGE REACTOR DEVELOPMENT

Four Eu_2O_3–stainless steel absorber sections for use in the SM-1A reactor were manufactured and shipped to Alco Products, Inc. (AEC Activity 4630)

GENERAL REACTOR RESEARCH

Reactor Evaluation Studies. – The mathematical model used in fuel value determinations has been improved; it now permits consideration of materials in the chain from Th^{232} through Am^{243}. A revised formulation of the effects of temperature and isotopic concentration on effective resonance integrals has been completed. In this, the self-shielding effect is associated only with the value of the resonance integral in excess of the 1/v contribution. (AEC Activity 4160)

Reactor Safety Studies. – Tests of the ORNL-designed safety system for the Bulk Shielding Reactor II, recently completed, indicate performance exceeding specifications. With the level safety trip set at a normal 100 kw, a transient having a 0.0045-sec period was held to a peak power of less than 100 Mw. The period safety channel, set to come on scale at ~ 100 mw, subdued a 0.0030-sec-period excursion, holding peak power to less than 100 Mw. The peak powers reached during reactor shutdown by the safety system agreed substantially with calculations. It is concluded that the safety system is particularly useful in dealing with excursions having periods in the range from 0.015 to 0.005 sec, which would produce severe damage if they were limited solely by the self-shutdown characteristics of the reactor. (AEC Activity 4171.1)

Fuel Element Catastrophe Studies. – Since power reactor fuels are predominantly based upon sintered UO_2 clad in a metallic sheath, a necessary aspect of reactor safeguards is examination of the various conditions of reactor malfunctions which might lead to fission product release by loss of integrity of the fuel cladding and the resulting exposure of irradiated UO_2 to a corrosive atmosphere such as air or steam.

In some exploratory studies, the rate of reaction of sintered pure UO_2 with air to form U_3O_8 was shown to reach a maximum between 400 and 600°C, with complete reaction in about 1 hr. The resulting fission product release in air at these temperatures is highly dependent upon the oxygen available for reaction. The volatile fission elements, such as rare gases, iodine, and tellurium, are released in high percentages (30 to 50%) at these temperatures, while at temperatures above 1000°C, release occurs only by diffusion and is less by an order of magnitude. Melting experiments on a modified UO_2 containing other oxides have shown nearly quantitative releases of iodine, tellurium, ruthenium, and cesium, with only negligible releases of the other fission products. (AEC Activity 4171.1)

Radiation Detector Development. – The design of a subminiature fission detector, to be employed with the plate-type reactors at the Bulk Shielding Facility (BSF), has been completed. The device consists of a thin surface-barrier silicon diode, $\sim 1\ cm^2$ in area, with a foil of U^{235} on one surface, at the end of a 24-in. probe. The total over-all thickness of the detector, including its aluminum protective covering, has been held to < 0.1 in., so that it can be inserted between fuel plates in the reactor core to obtain a complete scan of the fission rate within the elements. The small size will also ensure minimum perturbation of the flux being observed. When placed in operation the detector will augment the previously reported calorimeter measurements leading toward the determination of the absolute power of the BSF reactors. (AEC Activity 4410)

Long-Range Planning Studies. – Some two-group, two-dimensional nuclear calculations were performed on a typical aqueous homogeneous breeder reactor to determine the effect of inlet pipes on nuclear performance. The results for a reactor with a 4-ft-dia, 8-ft-long core with a 2-ft-thick blanket showed a negligible change in breeding ratio when a 2-ft-dia inlet pipe at the top was added to the basic configuration.

Preliminary nuclear calculations for the MSRE indicate that for a 4-ft-dia, 5-1/2-ft-long core the critical fuel loading amounts to approximately 1.1 mole % U^{235} in the fuel salt containing 64 mole % LiF, 31 mole % BeF_2, and 4 mole % ThF_4.

Studies have been made concerning the adequacy of four-group, two-dimensional nuclear calculations in predicting the correct critical mass and flux distributions in the High Flux Isotope Reactor critical experiments. The results indicate that for these experiments the four-group, two-dimensional model is inadequate, but that substantially better agreement between calculations and experiment will be obtained with the use of a multigroup, two-dimensional calculation. (AEC Activity 4410)

Fuel Cycle Development. – Simulated reactor fuel tubes filled to 90% of theoretical density with thorium-uranium oxides were produced by vibratory compaction of the oxides in tubes. With an air vibrator, ThO_2–3.4 wt %

UO_2 was densified to 8.5 to 8.9 g/cc in 2-ft lengths of 3/8-in.-OD stainless steel tubes. The use of a large electronic vibrator of 5000 lb thrust gave an average density of 8.39 g/cc, while additional pneumatic vibration of the same tubes gave an average density of 8.63 g/cc. Temperatures measured by thermocouples attached to the tube surfaces during vibration were 20 to 50°C higher for oxide-filled than for empty tubes. (AEC Activity 4420)

Power Reactor Fuel Processing: Darex Process. — Samples of 95% ThO_2–$UO_{2.05}$ fuel pellets (92% of theoretical density) which had been irradiated in the ORR for 3 min at power densities ranging from 16 to 160 Mw per metric ton of thorium were dissolved in boiling 13 M HNO_3–0.04 M NaF–0.1 M $Al(NO_3)_3$. In a 5-hr digestion with 200% stoichiometric excess of dissolvent, an average of 88.5% of the pellets dissolved irrespective of the irradiation history.

Instantaneous dissolution rates of unirradiated ThO_2-UO_2 pellets were determined in boiling Thorex solutions for 27 combinations of concentrations of nitric acid and thorium (8 to 18 M HNO_3 and 0.02 to 1.0 M Th). The solutions were 0.1 M in $Al(NO_3)_3$ and 0.04 M in NaF, and the ratio of thorium to uranium was 20. A linear correlation of the data was developed which expressed the reaction rate as a function of the thorium concentration, with the nitric acid concentration as a parameter. The rates without thorium present maximized at ~ 14 M HNO_3, and the slopes decreased from -0.14 to -1.04 as the HNO_3 concentration increased from 8 to 18 M.

In feed clarification tests on Darex solutions containing siliceous materials, the caustic-insoluble residue contained niobium carbide.

In further corrosion tests, titanium continued to show localized weld attack in the condensing region above refluxing Thorex feed-adjustment solutions after 1008 hr exposure. After 125 hr exposure to the vapor from successive Thorex boildowns, titanium welds showed only a slight etch. In Thorex rectifier studies, the rate of corrosion of type 309SCb stainless steel increased with increasing concentrations of HNO_3 and HF, from 1.0 mil/month in 3 M HNO_3–0.005 M HF to 56 mils/month in 15 M HNO_3–0.05 M HF. In Darex-Purex extraction waste storage tests, welded types 347 and 304L stainless steel showed intergranular attack in 5 M acid solutions at 65 and 80°C and in 2 M acid solution at 80°C after exposures of 840 hr or more. (AEC Activity 4441)

Power Reactor Fuel Processing: Sulfex Process. — In feed clarification tests on Sulfex solutions of stainless steel cladding, carried out in Ni-o-nel and Carpenter-20 stainless steel equipment, the removed solids contained 10% copper.

In Thorex dissolver corrosion tests of welded Ni-o-nel, specimens were heat-treated at various temperatures prior to welding. The highest corrosion rates, for specimens heat-treated at 1725°C, were 19 to 20 mils/month. In 13 M HNO_3–0.04 M NaF with and without additions of 0.1 M Al^{+++} and 0.05 M $Na_2B_4O_7$, the lowest rate, 3.9 mils/month, was in the solution containing both additives. In boiling 6 M H_2SO_4, Haynes experimental alloys EB4358 and 5459 corroded at maximum rates of 2.7 and 2.4 mils/month, respectively, in 360 hr exposure. (AEC Activity 4441)

Power Reactor Fuel Processing: Zircex Process. — Preliminary evaluation of Zircex as a head-end process for U-Mo alloy fuel was begun. In 1-hr studies at 500°C, 91.6% U-Mo alloy reacted at an average rate of 3.6 $mg \cdot cm^{-2} \cdot min^{-1}$ when exposed to pure HCl gas; in exposure for 1 hr to a cycle of 10 min HCl and 1 min air, the rate was 5.0 $mg \cdot cm^{-2} \cdot min^{-1}$; 86% of the molybdenum was converted to highly volatile molybdenum oxychloride and separated from the uranium products.

In a cyclic Zircex corrosion test, Pyroceram 9608 showed a rate of 1.3 mils/month, as compared with 6.5 to 13 for Nichrome V and from 42 to 130 for various cermets. In boiling 10 M LiCl–0.1 M HCl, Hastelloy B showed a maximum rate of 3.5 mils/month in 24 hr exposure. In the presence of approximately 0.01 M H_2O_2, rates varied from 7.9 to 44 mils/month. (AEC Activity 4441)

Power Reactor Fuel Processing: Chemical Processes for Advanced Fuels. — Dissolution of U–3% Mo alloy in boiling 3 to 11 M HNO_3 and in 5 M HNO_3 containing up to 1 M $Fe(NO_3)_3$ resulted, respectively, in the consumption of 4 to 7 and about 4 moles of acid per mole of alloy dissolved. Pure molybdenum dissolved at a rate of 1.8 $mg \cdot cm^{-2} \cdot min^{-1}$ in refluxing 18.3 M H_2SO_4. The rate decreased to nearly zero as the acid concentration was decreased to 14 M or increased to 20 M.

Five successive 4-hr leaches of 15-g specimens of graphitized uranium-graphite fuel with 70% HNO_3 resulted in the recovery of 60% of the uranium in fuel containing 0.7% uranium, 64% for 2.5% fuel, and 75% for 5% fuel. No swelling or disintegration of the graphite occurred. The same fuels were disintegrated and leached with two portions of boiling 90% nitric acid; the uranium recoveries were 97, 99.5, and 99.85% respectively.

Iodine monochloride disintegrated samples of 10 and 14% uranium-graphite fuels to -20 mesh; but the halogen, primarily iodine, could not be readily removed from the graphite and inhibited the subsequent leaching with nitric acid. Two leaches with 15.8 M HNO_3 recovered 97% of the uranium from the 14% fuel, and only 85% from the 10% fuel. Disintegration and leaching with 90% HNO_3 recovered 99.95% of the uranium from similar specimens of these fuels.

In two scouting experiments methane and hydrogen were identified as the chief gaseous products from the hydrolysis of uranium monocarbide. (AEC Activity 4441)

Power Reactor Fuel Processing: Mechanical Processing of Fuels. — Mechanical-operability cold testing of the SRE mechanical dejacketing process equipment and its auxiliaries was completed. This equipment is installed in the segmenting cell, Building 3026, to demonstrate the production feasibility of mechanical dejacketing of NaK-bonded stainless-steel-clad SRE fuel. The hydraulic fuel-tube expander, operating at 800 to 1950 psig pressure, expanded 3/4-in.-dia prototype 10-mil-thick stainless steel SRE jackets by 0.5 to 365 mils, the rupture point of undamaged tubing. Prototype fuel slugs were easily discharged from the expanded jackets by a hydraulic pressure of less than 100 psig. (AEC Activity 4441)

Power Reactor Fuel Processing: Solvent Extraction Studies. — The substitution of 40% tributyl phosphate in Amsco for 30% TBP–Amsco in the Acid Thorex flowsheet resulted in equal decontamination, less waste volume, and

possibly increased plant capacity. The uranium and thorium from the Acid Thorex extraction column may be satisfactorily partitioned with five strip and five solvent scrub stages. Final U^{233} purification according to an acid-salted 5% TBP–Amsco flowsheet gave decontamination factors from ruthenium gamma, zirconium-niobium gamma, protactinium gamma, and thorium of 300, 2400, 400, and 10 respectively. A countercurrent batch extraction for final purification of U^{235} according to an acid-salted 15% TBP–Amsco flowsheet resulted in ruthenium gamma and zirconium-niobium gamma decontamination factors of 3200 and 14,000 respectively.

A graphical method based on the McCabe-Thiele diagram was developed to define operable flow ratios and to calculate the number of theoretical stages required for partitioning of uranium and thorium in a flowsheet for processing Consolidated Edison fuel. The calculations, confirmed experimentally in batch countercurrent tests, showed that partitioning is satisfactory in a two-section cascade consisting of five stages in each section. The graphical method also showed the ranges of solvent-to-aqueous flow ratios and uranium-thorium concentrations that can be used successfully. (AEC Activity 4441)

Fluoride Volatility Processing. – In two engineering-scale runs, Zircaloy-2 was dissolved in 43 mole % NaF–57 mole % LiF melts. A copper-lined dissolver was used to minimize contamination from corrosion products. The average rate of dissolution changed from 0.461 to 0.273 $lb \cdot hr^{-1} \cdot ft^{-2}$ as the ZrF_4 concentration in the melt increased from 0 to 45 mole %.

In laboratory studies carried out in a radioactive cell with "Nautilus" fuel, molybdenum, technetium, neptunium, and chromium were the chief contaminants of the UF_6 product. There was no plutonium in the product. The molybdenum varied from 1250 to 10,100 parts and the technetium from 56 to 1020 parts per million parts of uranium. The chromium contamination may have arisen in hydrolysis of the UF_6 in a stainless steel vessel. The presence of molybdenum, technetium, and neptunium indicates that they behave similarly to the UF_6 in the sorption-desorption process. The first NaF bed contained some plutonium but no neptunium; 5% of the neptunium was removed from the product stream by the second NaF bed. Small differences in fluorine flow rates or temperatures are believed to account for the variations in the molybdenum and technetium. Vapor pressure measurements on the MoF_6-NaF complex indicate that its decomposition temperature is 125°C (at 760 mm), and that absorption of UF_6 by NaF above this temperature would lead to a marked decrease in the contamination of the UF_6 by molybdenum. (AEC Activity 4442)

PROGRAM 5000 – PHYSICAL RESEARCH

Reactor Operations. – The Graphite Reactor was shut down on May 9 in order to install the cross ducts needed for the annealing. This work was to be completed about May 23. If instrumentation can be completed in time, it is planned to conduct the annealing operation about July 4.

The ORR is scheduled to be shut down for three weeks starting July 10 for installation of the new 30-Mw cooling system. Studies are in progress on the use of aluminum-cadmium shim rods in place of the cadmium-fuel shim rods now used. If the new rods are successful, they should last longer and provide more uniform control than the present rods.

Laboratory Facilities. – Interim monitoring is now installed on the 3039 stack so that the approximate amounts of gaseous and particulate radioactivity can be measured. Better monitoring systems are being designed and will be installed next year.

The waste disposal trench, designed to take the place of the leaking waste pit, has been completed and is now being tested for seepage rate. Some success has been achieved in the use of copper ores to fix ruthenium in the soil. The new seepage trench will be treated with copper to impregnate the surrounding soil before the active waste is added. A monitoring station is being installed in White Oak Creek to attempt to locate the source of the Sr^{90} which is measured at the White Oak Dam but is unaccounted for in any known release.

PHYSICS

Neutron Diffraction. – Since previous neutron diffraction investigations of chromium [Revs. Modern Phys. 25, 100 (1953); Phys. Rev. Letters 3, 211 (1959); Doklady Akad. Nauk S.S.S.R. 128, 1153 (1959)] have indicated different Néel temperatures for powders and single crystals, experiments have been performed with high resolution to determine whether magnetic short-range order could have influenced the observations from powders. In examinations of three powdered samples, one showed a Néel temperature of 308°K, in agreement with single-crystal determinations, but the others gave values close to 450°K. The two powders with high transition temperatures showed three distinct magnetic reflections near the (100) position, which could be explained qualitatively by the antiphase-domain model [Phys. Rev. Letters 3, 211 (1959)] if the length of the unit cell decreased with decreasing temperature. One of these powders also showed a transition at about 110°K, where a single reflection replaced the group of three, suggesting that the moment orientation flipped to a direction perpendicular to the antiphase-domain wall. The third sample, with the low Néel temperature, gave antiferromagnetic reflections consistent with either a simple body-centered-cubic antiferromagnetic structure or an antiphase-domain model with the moment direction never parallel to the domain wall. (AEC Activity 5220)

Theoretical Physics. – Depolarization processes affecting mu mesons are conveniently divided into two types of mechanisms: those associated

with initial formation of the mu-mesic atom and those associated with subsequent transitions to the ground state. For the former, the process considered is nonradiative capture with ejection of an atomic electron. The depolarization mechanism is the spin-orbit coupling, and the mu polarization in the initial orbital depends strongly on the amount of scattering preceding capture. It is found that the absorption is much smaller than the scattering cross section at relevant energies, so that the assumption of random initial motion is well justified for almost all atomic numbers. Then, independently of the physical model, the polarization in the initial bound state is essentially 1/3. Incomplete considerations of the depolarization due to subsequent radiative cascades indicate that the final polarization in the 1s state is 0.15 to 0.20, in agreement with observations. The smaller polarization observed in helium [Bull. Am. Phys. Soc. 5, 75 (1960)] is qualitatively explained by the larger ratio of absorption to scattering for lower atomic numbers. The distribution of mesons in the initial orbitals, as given by the Born approximation, is quite different from conjectured results appearing in the literature. (AEC Activity 5220)

Fast-Chopper Time-of-Flight Spectrometer. — Total-cross-section measurements have been made upon samples of fission product iodine (71% I^{129}, 29% I^{127}), fission product samarium (3.88% Sm^{151}, 1.63% Sm^{149}), and a reactor-irradiated fission product samarium (1.9% Sm^{151}, 0.012% Sm^{149}). A fission product samarium sample was irradiated in the ORR for ten days to burn out the Sm^{149}, which has a large cross section in the thermal neutron energy range and masks the effect of the Sm^{151} in the sample. The data on the fission product iodine sample gave an absorption cross section of I^{129} which varies as 1/v in the thermal energy range and has a value of 28 ± 2 barns for 2200-m/sec neutrons. The measurements upon the fission product samarium sample result in an absorption cross section of Sm^{151} whose value for 2200-m/sec neutrons is 15,400 ± 1500 barns and which departs strongly from a 1/v dependence in the thermal energy range (by a factor of ~ 5 from 0.02 to 0.2 ev). The cross section of Sm^{151} can be fitted in this thermal energy range by postulating that there is a bound level with the parameters $E_0 = -0.015$ ev, $\Gamma = 0.120$ ev, and $2g\Gamma_n^\circ = 2.6 \times 10^{-4}$ ev and that the background correction is 600 barn·$ev^{1/2}$. Transmission measurements have also been made upon all the samples up to 300 ev, and the data are being analyzed to obtain the parameters of the resonances. (AEC Activity 5220)

High-Voltage Experimental Program. — The ORNL 5.5-Mv Van de Graaff has been used to obtain Ne^{20} ions with energies up to 11 Mev. The Coulomb excitation produced by these ions has been studied in a number of nuclei ranging from Li^7 to Ir^{193}. Neon-20 ions in this energy range have the advantage that nuclear reactions other than Coulomb excitation have extremely small cross sections. However, with the presently available ion beam intensities, 0.2 μamp of Ne^{5+}, for example, the Coulomb excitation gamma-ray yields are so low that room background counting rates are often troublesome. To alleviate this problem, a lead shield for the scintillation detector has been built which has 4-in. walls and inside dimensions of 48 × 40 × 18 in. To obtain information on spins and decay modes of excited states, the 0-90° asymmetry ratios were measured for the 128-kev state in Mn^{55} and the 440-kev state of Ne^{23}. Since the Ne^{20} ions produce relatively few K x rays

during the slowing-down process in the target, they are useful for measuring K-shell conversion coefficients of Coulomb-excited states. The K-shell conversion coefficients for the 136-kev state in Ta^{181} and the 123-kev state in W^{186} have been measured.

In effect, measurements on a sample of lead containing 72% Pb^{208}, 26% Pb^{207}, and 2% Pb^{204} have been made by subtracting out the effect of Pb^{206} from normal lead by use of an 88% Pb^{206} sample. The total neutron cross section of this sample, as well as that of normal lead, has been obtained from 1 to 4.1 Mev with 25 kev resolution. A resonance, almost certainly due to Pb^{208}, of about 100 kev width shows up at 1.75 Mev. Its height corresponds to a $J = 3/2$ level. The differential cross sections were measured at an energy (1.2 Mev) at which the total cross section is flat, and at energies (2.2 and 3.2 Mev) where the total cross section is rising to a broad maximum centered around 4 Mev. Neutrons produced by the T(p,n) reaction were scattered by cylinders of lead and detected as a function of angle with a shielded propane counter. Since Pb^{208} is a nucleus with a doubly closed shell, it is expected that a phase-shift analysis of the data together with a knowledge of the bound states in Pb^{209} will allow a phenomenological potential to be defined which describes these bound and virtual states with large reduced widths [Bull. Am. Phys. Soc. 5, 34 (1960)]. (AEC Activity 5220)

CHEMISTRY

Solvent Extraction Research. – The extraction of technetium, neptunium, and uranium from uranium hexafluoride transfer cylinder wash by a commercial tertiary amine, which was satisfactory in laboratory-scale mixer-settler tests (ORNL-2906), was also satisfactory in continuous countercurrent tests. Neptunium was lost to the uranium product only when the extract was aged excessively. Stripping the neptunium with HNO_3 or HNO_3-H_2O_2 instead of H_2SO_4 eliminated sulfate from the neptunium product. Fractional stripping of nitrate permitted collecting the technetium product at ~ 5 g/liter in 4 M NH_4NO_3 or $NaNO_3$, with a concentration factor approaching 50.

In studies of the dependence of the uranium distribution coefficients on extractant concentration in two amine systems, di-n-decylamine sulfate in benzene and tri-n-octylamine sulfate in benzene vs dilute sulfuric acid, metastable conditions induced by vigorous agitation in the liquid-liquid equilibrations led to anomalous results. Gentle equilibrations through limited interfaces resulted in nearly theoretically ideal dependences for these systems. An explanation based on adsorption of the critical components at the much larger interface generated during vigorous agitation accounts reasonably well for the results.

The response of Ru^{106} extraction (as the nitrato nitrosyl ruthenium complexes) to variations in the treatment of the solvent (tributyl phosphate in Amsco 125-82) was tested in a manner similar to that used previously for Zr^{95}-Nb^{95}. In contrast with the Zr^{95}-Nb^{95} results, the Ru^{106} extraction properties of fresh TBP in fresh Amsco were not significantly altered by chemical degradation of the diluent, by simulated degradation of

the TBP (addition of the principal hydrolysis product), or by irradiation of the complete solvent. Only in tests where the Ru^{106} was extracted into degraded solvent and aged several days did simple scrubbing with acid or alkaline solutions not effect complete stripping. (AEC Activity 5310)

Ion Exchange Technology. – ORNL low-activity process waste, containing cesium and strontium tracers, was made 0.01 M in NaOH, filtered, and passed through Duolite C-3 phenolic-sulfonic resin or Duolite CS-100 phenolic-carboxylic resin at a rate of 600 resin bed volumes per day. The cesium broke through at about 1500 resin bed volumes. The breakthrough of strontium was significantly behind that of cesium. More than 99.9% of the activity was removed from the C-3 resin with ten volumes of 5 M HCl, and from the CS-100 resin by five volumes of 2.5 M HCl, both upflow. The effluent from the resin regeneration was evaporated to half volume before NaCl crystals formed.

The addition of oxalic acid (to a concentration of 0.12 M) to tenfold-diluted simulated Purex 1WW waste increased the 50% breakthrough capacity of Dowex 50W-X8 resin for strontium from 7 to 22 resin bed volumes and the 1% breakthrough capacity for cerium from 16 to 67 volumes. The increase results from complexing of iron in the feed, decreasing its competition for the resin. In a flowsheet demonstration run both strontium and cerium were loaded onto one Dowex 50 column. The strontium was eluted from this column, and loaded on a second, with 1 M HNO_3; the cerium was eluted from the first column with 0.5 M citrate at pH 3.5 and the strontium from the second with 0.07 M ethylenediaminetetraacetate (EDTA) at pH 6.1.

By a single-bead contacting technique the rate of loading of uranyl sulfate on nitrate-equilibrated 1200-μ Dowex 21K resin was shown to be independent of the sulfate concentration of the loading solution; it corresponded to an apparent uranium diffusion coefficient of 1.6×10^{-7} cm^2/sec, but the results were scattered because of the nonuniformity of the beads. A new contactor that brings several beads into contact with a flowing solution under defined and reproducible hydrodynamic conditions consists of a submerged pump with a removable resin holder located at the pump outlet. Elution from 1200-μ Dowex 21K resin with neutral 1 M $NaNO_3$ at a flow rate of 20 cm/sec gave results that fell near the curve predicted for an apparent uranium diffusion coefficient of 1.8×10^{-7} cm^2/sec. (AEC Activity 5310)

Inorganic and Physical Chemistry: Thermochemistry and Calorimetry. – The solubility of K_2ReCl_6 in 0.01 M HCl was found to be 0.1664 M at 25°C and 0.0832 M at 0°C. The standard heat of solution of K_2ReCl_6 in 0.01 M HCl was determined by measuring the integral heat of solution as a function of the final concentration of the salt and extrapolating the results to infinite dilution with the aid of the Debye-Hückel limiting law. The standard heat of solution is 10,397 ± 15 cal/mole. The standard heat of solution calculated from the temperature coefficient of the solubility, assuming $\Delta\bar{C}_p = 0$, is 9930 cal/mole, which is within 4-1/2% of the measured value. These data, together with the known entropies of solid K_2ReCl_6 and of $K^+(aq)$ and an estimate of the activity coefficient of K_2ReCl_6 in the saturated solution, give 59.5 + 0.4 $cal \cdot deg^{-1} \cdot mole^{-1}$ for the entropy of $ReCl_6^{--}(aq)$. The entropy of formation is calculated to be -141.6 $cal \cdot deg^{-1} \cdot mole^{-1}$. (AEC Activity 5330)

Inorganic and Physical Chemistry: Physical Chemistry of Ion Exchangers. — An accurate and convenient method was developed for the estimation of the weight swelling of strong-acid-type cation exchangers and strong-base anion exchangers when they are in equilibrium with aqueous electrolyte solutions. The procedure was based on measurements of the increase observed in the concentration of linear polyelectrolyte solutions when a known weight of dry exchanger was immersed in the solution, assuming that no linear polyelectrolyte was taken up by the resin. Estimates were made also of the invasion of the exchanger by electrolyte (e.g., NaCl) from measurements of the increase in electrolyte concentration, made on the same aqueous phase. This increase was not as great as for polyelectrolyte, of course, because of the loss of electrolyte into the exchanger. The molal Donnan distribution coefficient for electrolyte partition was found to decrease as the concentration of electrolyte decreased, and to approach zero as the latter became increasingly small. This finding agrees with theoretical expectations but not with published experimental measurements where different techniques were utilized. (AEC Activity 5330)

Inorganic and Physical Chemistry: Chemistry of Technetium. — The acid H_2ReCl_6 was prepared, and an attempt was made to determine the degree of dissociation of both hydrogen ions. Nonaqueous titrations of the compound dissolved in ethylene glycol against 0.1 N piperidine gave one-step neutralization curves resembling that for perchloric acid. Solutions of H_2SO_4 in the same solvent gave titration curves with two steps. It was concluded that H_2ReCl_6 is a strong acid and that dissociation to give a second hydrogen ion is greater than in the case of sulfuric acid.

The visible and near-infrared (5000 to 12,000 Å) spectra of K_2ReCl_6 (in 0.01 M HCl) and K_2ReBr_6 (in 3 M HBr) were measured. Both spectra exhibit an unusually distinct fine structure of vibrational bands superimposed upon the electronic absorption bands, especially when the compounds are in cyclohexanone solution. The energy separation of the vibrational bands in K_2ReCl_6 is approximately 150 cm^{-1}, a very low vibrational energy. The corresponding spectrum of K_2PtCl_6 (in 1 M HCl) shows no vibrational bands. At 25°C, the excess entropy of K_2ReCl_6 over that of the isomorphous diamagnetic compound K_2PtCl_6 is 7.25 $cal \cdot deg^{-1} \cdot mole^{-1}$. Of this difference, R ln 4 (= 2.75 $cal \cdot deg^{-1} \cdot mole^{-1}$) arises from the magnetic entropy of the K_2ReCl_6. A low-energy vibrational state at approximately 150 cm^{-1} would give rise to an entropy of 2.68 $cal \cdot deg^{-1} \cdot mole^{-1}$ at 25°C. This leaves an entropy of approximately 1.8 $cal \cdot deg^{-1} \cdot mole^{-1}$ still unaccounted for. The vibrational fine structure observed in the visible and near-infrared spectrum of K_2ReCl_6 shows two possibly different energy separations. The entropy considerations also indicate that perhaps there are two low-energy vibrational states. (AEC Activity 5330)

Chemistry of the Heavy Elements. — An aqueous solution composed of UO_2SO_4 (0.0123 m), $CuSO_4$ (0.0085 m), and D_2SO_4 (0.0072 m) in D_2O (98.5 at. % D) was examined spectrophotometrically from 0.34 to 1.2 μ over a period of 50 days and over a temperature range from 4 to 280°C.

The 0.42-μ band of U(VI) and the 0.82-μ band of Cu(II) do not overlap and could be studied independently. As the temperature was increased from 4 to 280°C, the molar absorptivities of the bands increased monotonically

by factors of 2.3 and 2.0 for U(VI) and Cu(II) respectively. At 280°C the half-intensity band width of U(VI) was 2330 cm^{-1} (6% narrower than at 25°C) and that of Cu(II) was 5480 cm^{-1} (3% broader than at 25°C).

In contrast with the earlier study of 0.017 m UO_2SO_4 in 0.01 m D_2SO_4, in which there was no indication of soluble lower-valence species, the data for this solution after a heating cycle indicated the presence of U(IV) at lower temperature, beginning with an estimated 0.05% (just detectable) at 50°C and increasing to 1.5% at 4°C. The appearance and disappearance of U(IV) were observed in four successive thermal cyclings. (AEC Activity 5330)

Reactor Neutron Cross Sections. – A program has been initiated to measure the neutron cross sections of several of the unstable fission product nuclides, among them 32-day Ce^{141} and 285-day Ce^{144}. Initial measurements of the cross section of Ce^{144} by the activation method are completed. Neutron capture in Ce^{144} produces a 3-min Ce^{145}, which decays by beta emission to 5.9-hr Pr^{145}. The Pr^{145} activity, which was identified through its 1.7-Mev beta ray, was used as a measure of the Ce^{144} absorption cross section. After a short irradiation and a short cooling period the praseodymium fraction was rigorously purified by 15 successive extractions into di(2-ethylhexyl)phosphoric acid. The decay of the 5.9-hr activity was followed through ten half lives. The only remaining activity was 13.8-day Pr^{143}. The results from these preliminary investigations indicate an effective neutron cross section for Ce^{144} of $(6 \pm 2) \times 10^{-24}$ cm^2/atom, which is of the same order as that for the 33-hr Ce^{143}.

A number of irradiations of U^{233} were carried out with and without cadmium covers in the LITR and in the ORR. The analyses for the fission products Sr^{89}, Zr^{95}, Mo^{99}, I^{131}, and Ba^{140} have been completed, and mass analyses for U^{233} and U^{234} are in progress. A resonance integral of 870 barns for the fission of U^{233} was obtained on the basis of the Sr^{89}, Mo^{99}, and Ba^{140} analyses. Both Zr^{95} and I^{131} yielded somewhat higher values for the resonance integral, but their epicadmium fission yields are thought to be higher than their yields at thermal energies. The resonance integral of 870 barns obtained experimentally may be compared with 820 barns calculated by Terasawa and with 900 barns calculated from the resonance data given in BNL-325. (AEC Activity 5330)

Nuclear Chemistry: Characterization of Short-Lived Fission Products. – Methods for the rapid separation of fission product molybdenum and technetium were developed, and a search was made for the short-lived activities of these elements. Three hitherto unreported periods were observed: 1.6-min Mo^{103}, 7-min Tc^{105}, and 18-min Tc^{104}. The chemical yield of technetium was estimated from the radioactivity of 60-day Tc^{95m}; the molybdenum yields were measured by colorimetric determination of the quercetin complex formed in chloroform after extraction of the element into the solvent with α-benzoinoxime. As yet, however, accurate fission yield determinations have not been made.

The decay scheme of 2.7-hr Pm^{150} is being studied by scintillation methods in collaboration with the Nuclear Data Project of the National Academy of Sciences. Analysis of the gamma-ray spectrum obtained with a single-crystal spectrometer showed clearly resolved peaks at 0.333, 0.41, 0.71,

0.86, 1.18, 1.33, 1.77, 1.96, 2.53, and 2.91 Mev, together with less intense peaks at 1.0, 1.6, 2.06, 2.29, 2.41, 2.75, and 3.08 Mev.

Fermi analysis of the beta-ray spectrum obtained with a single-crystal spectrometer disclosed groups at 3.16 ± 0.11 Mev, 2.3 Mev, and lower energies. The gamma-ray spectrum in coincidence with beta rays of energy greater than 1.8 Mev showed a strong gamma-ray peak at 0.33 Mev, and peaks of low intensity at 0.4, 0.86, and 1.18 Mev. Beta-ray spectra were determined in coincidence with five of the prominent gamma rays, with the following results: the 3.16-Mev beta group is in coincidence with the 0.333-Mev gamma ray only; the 2.3-Mev beta group is coincident with gamma rays at 0.333, 0.86, and 1.18 Mev; a 1.83-Mev beta group is in coincidence with 0.86- and 1.33-Mev gamma rays; and a 1.4-Mev beta group is coincident with gamma rays of 0.333, 1.33, and 1.75 Mev and possibly with others.

These results show that the most energetic beta group decays to the well-known first excited state of Sm^{150} at 0.333 Mev; hence the Pm^{150} beta disintegration energy is 3.49 ± 0.11 Mev, in good agreement with the energy systematics. From the available preliminary information a partial decay scheme can be formulated with excited states in Sm^{150} at 0.333, 1.18, 1.66, and 2.08 Mev. (AEC Activity 5330)

Nuclear Methods of Chemical Analysis. — Four 20-mg AmO_2 gamma-ray sources have been fabricated for use in investigations involving low-energy gamma absorption measurements. The sources have the following characteristics:

Radiochemical purity, %	>99.9
Chemical purity, %	>80
Ratio of 60-kev gamma rays to alpha particles for AmO_2	0.28
Per cent of gamma activity above 80 kev	<0.67
Total gamma emission from source, photons/min	$>10^{10}$

The AmO_2 is contained in a welded stainless steel cylinder 1 in. in diameter and 0.5 in. thick that is sealed with a 0.01-in. stainless steel window. The 60-kev gamma radiation emitted by the source makes it possible to use these sources in measurements of heavy-element concentrations by absorptiometry. In addition, sources of this kind can be readily used in gamma radiography applications. (AEC Activity 5330)

Physical Chemistry of Molten Salts. — Sodium fluoride, which does not form solid solutions with low concentrations of various solute fluorides, has been used as a cryoscopic medium to observe the effects of several divalent fluorides as solutes. The freezing point depressions due to the alkaline-earth fluorides in concentrations up to 20 mole % are inversely proportional to the cation-anion distance; smooth curves are obtained by plotting the reciprocal of the interionic distance in these fluorides against the difference, for each case, between the observed and the ideal freezing temperature for the solution. When the solutes are the isomorphous transition-metal difluorides the freezing point lowering follows the order $NiF_2 > CoF_2 > ZnF_2 > FeF_2 > MnF_2$. A plot of the excess partial molar

free energy of mixing of NaF vs the atomic number of the transition-metal cation gives a humped curve that is very similar to the curves of lattice energy and heat of hydration of the transition-metal halides. These results indicate that the coordination numbers of Mn^{++}, Fe^{++}, Co^{++}, Ni^{++}, and Zn^{++} in these fluoride solutions are approximately the same. (AEC Activity 5330)

Chemical Separation of Isotopes. — The single-stage separation factors for the exchange of boron between BF_3 (gas) and the triethylamine·BF_3 and phenol·BF_3 complexes (liquid) were studied as a function of temperature. The separation factor for the phenol system varied from 1.035 at -10°C to 1.022 at 37°C. The separation factor for the triethylamine system ranged from 1.021 at 30°C to 1.017 at 50°C.

The alkaline hypobromite oxidation of urea labeled with N^{15} indicated that the nitrogen gas produced came from individual urea molecules. Nitrogen exchange did not occur during the oxidation reaction. (AEC Activity 5340)

METALLURGY AND SOLID STATE PHYSICS

High-Temperature Reactions. — It has long been known that bismuth metal dissolves in molten bismuth trichloride, but despite intensive studies of thermodynamic properties the chemical species present in solution have not been identified. Accordingly, such solutions have been examined by absorption spectrophotometry over the concentration range from ca. 0.001 to 0.4 M and at temperatures of 260 to 433°C. These measurements show unequivocally that the metallic bismuth is partitioned into more than one species, only one of which can be measured accurately. The fraction of bismuth partitioned into this species has been determined as a function of total dissolved bismuth and temperature. For example, at a dissolved bismuth concentration of 0.01 M the partitioning fraction Q is virtually unity at all temperatures, whereas at 0.25 M, Q is 0.26 ± 0.02 at 260°C and is 0.67 ± 0.05 at 433°C. The partitioning equilibrium is between a monomer ion and a polymer ion with a possible change in the ionic charge per bismuth atom. It is the monomer which is observed. The determination of Q was made possible by the discovery of an intense band at 560 mμ with an oscillator strength which changes by no more than a few per cent for a 30% change in absolute temperature. (AEC Activity 5420)

THERMONUCLEAR PROJECT

DCX-1 Facility. — The installation of DCX-1 in Building 9201-2 was completed. Measurements were made on the magnetic fields, pumping speeds, and accelerator characteristics to determine that the facility is operational. Experience with the 32-in. diffusion pumps indicates that they are back-streaming excessive amounts of oil. Water-cooled baffles were installed on the two diffusion pumps on the anode side of the machine. This has resulted in a decrease in pumping speed of 50% for these two pumps. The

effect of the baffles on oil back-streaming is being observed.

Both argon and deuterium gas arcs have been operated. The minimum pressure achieved with D_2 arcs has been 3×10^{-5} mm Hg in the liner region, while the large pumps are capable of maintaining the pressure in the outer region at 9×10^{-5} mm Hg. For argon arcs the liner pressure is of the order of 10^{-6} mm Hg with a pressure differential factor of 50 to 100 between the inner and the outer vacuum system. (AEC Activity 5540)

DCX-EPA. – This machine is again operable, and the characteristics of several arc types are being studied. These are: (1) gas arcs operated with a large rotating water-cooled anode, (2) symmetrical reflux gas arcs using double anode and cathode systems, and (3) metallic and carbon reflux arcs using sputtered feed material. In these tests the operating voltage often exceeds 1000 v and the current varies from 50 to 200 amp. (AEC Activity 5540)

Ion Source Development. – Hydrogen ion beams from a source of the Von Ardenne type have been mass-analyzed at total currents up to 100 ma. Parameters affecting the mass ratios have been studied. The maximum relative yields of H_2^+ obtained to date have been 77% H_2^+ at 40 ma total output and 68% H_2^+ at 100 ma total output.

The 600-kev beam test facility is under vacuum. Final electrical connections are being made. (AEC Activity 5540)

PROGRAM 6000 – BIOLOGY AND MEDICINE

BIOLOGY

Genetic and Cytogenetic Effects of Radiation: Polyamines and Chromosome Coiling. – It has been found that the polyamines putrescine, cadaverine, and spermine upset the normal course of mitosis in root tips of *Vicia faba*. These compounds prevent chromosome uncoiling at telophase and interfere with the movement of chromatids to the poles at anaphase. The effects of these polyamines are also seen in chromosomes that reach metaphase after the beginning of treatment. The chromosomes condense, but they form a single amorphous mass.

The effects on stages from metaphase to telophase indicate that polyamines can interfere with the normal processes of chromosome coiling and uncoiling. These results are significant because no mechanisms controlling chromosome coiling and uncoiling are known; it appears that such mechanisms may involve polyamine compounds, which are widely distributed in plant and animal cells. (AEC Activity 6130)

Genetic and Cytogenetic Effects of Radiation: Effect of pH and Salt Conditions on Protection with AET·Br·HBr Against X Rays. – Experiments have been carried out to study protection with S,β-aminoethylisothiuronium bromide hydrobromide (AET) against the mutagenic and lethal effects of x rays in *Neurospora crassa*. In earlier studies this compound has occasionally given some positive protection, but since the results have proved difficult

to reproduce, it appeared that more investigation was needed with regard to such conditions as pH during the preparation of the AET solution, pH during the preparation of the conidia suspension, and salt effects of saline and buffer solutions.

The toxic effect of AET was found to be at a minimum in the pH range 5.5 to 7. A protection factor of 1.5 to 2 against mutation was found at pH 6.6 with doses below 40 kr. At higher doses the mutation rate is higher in the pretreated conidia; this is similar to the effect found with cysteamine and endogenous anoxia, which was assumed to be the result of protection against a selective killing of induced mutants. Protection against lethality at doses higher than 40 kr is best expressed by a protection factor of ca. 1.5. (AEC Activity 6130)

Effects of Radiation on Paramecium. — It has been known for some time that paramecia develop at least ten times as many recessive lethal mutations when they are irradiated within the first 2 hr after division as when they are irradiated later in the interdivision interval. There were indications that this abrupt change in susceptibility was associated with chromosome duplication in some way, but no direct information on the time of duplication was available. It has now been found that the volume of the nucleus increases quite abruptly to approximately double its initial value between 2 and 3 hr after division. Moreover, when one of the two events is delayed, the other is also and to the same extent, as shown by studies with a slowly growing clone and with caffeine-treated animals. It seems probable that the increase in volume is caused by chromosome duplication and, consequently, that the change in susceptibility is the result of this event. (AEC Activity 6130)

Insect Cytology and Genetics. — The labeling of ribonucleic acid with tritiated cytidine has been followed with time by autoradiography in the protozoan Tetrahymena. During the first 12 min in the radioactive medium, all incorporation of tritiated cytidine is restricted to the nucleus. From 12 to 40 min the nucleus becomes more heavily labeled but the amount of radioactivity in the cytoplasm rises steadily. From 40 min on, both nucleus and cytoplasm are about equally labeled with tritiated cytidine. This pattern of labeling suggests a movement of RNA from the nucleus into the cytoplasm. Incorporation of the label into the nucleus continues during nuclear division but falls briefly to a low level in cells that have just completed cytokinesis. (AEC Activity 6130)

Chemical Protection and Proteolytic Enzyme Studies. — Physical characterization of the major component of beef heart malic dehydrogenase established the sedimentation constant as 4.3 S (1.5% protein concentration). Moving boundary electrophoresis studies have shown the protein to have an isoelectric point close to pH 6.2 and a rather low net charge, the mobility at pH 7.5 in phosphate buffer being -0.65×10^{-5} $cm^2 \cdot volt^{-1} \cdot sec^{-1}$. Kinetic studies with this preparation reveal that the enzyme very slowly catalyzes the oxidation of mesotartrate and L-tartrate but not D-tartrate. When the activity of this mitochondrial malic dehydrogenase is compared with the malic dehydrogenase of the supernatant fraction, differences in kinetics are obtained using diphosphopyridine nucleotide (DPN) and its analogs. (AEC Activity 6130)

Pathology and Physiology. – A fatal disease developed in sublethally (600 r) x-irradiated adult (C57BL × 101)F_1 mice within two to three weeks after intraperitoneal or intravenous implantation of liver cells from mice of the parental strains. The mortality 30 to 45 days after exposure to 900 to 950 r of x radiation was increased to 100% in F_1 hybrid mice protected with isologous or parental bone marrow and injected at the same time with parental liver cell suspensions. The most important histologic changes occurred in the lymphatic tissues. These were similar to the changes observed during the foreign spleen reaction in irradiated mice and included replacement of the lymphoid follicles by proliferating reticuloendothelial cells. No mortality or related histologic changes have been observed in irradiated F_1 hybrids after implantation of isologous (F_1) liver cells or of liver cells from irradiated (950 r) parental donors. Irradiated C57BL mice (500 r) and 101 mice (550 r) do not show any disease after implantation of F_1 hybrid liver cells. The liver cell suspensions have been obtained from perfused and unperfused tissue. The fatal reaction caused by implantation of adult parental liver cells in irradiated (C57BL × 101)F_1 mice is ascribed to an immune reaction of immunologically competent grafted cells against the recipient. The identity of these cells remains to be established. (AEC Activity 6130)

Biophysics: Magnetic Resonance. – The production and decay of free radicals in dry lettuce seeds irradiated with x rays is being studied by electron spin resonance techniques. The initial results for whole seeds indicate that perhaps 600 ev is dissipated per long-lived radical formed, that a relatively rapid decay to about two-thirds of the initial concentration in a couple of days is followed by a long decay with a half life of many days, and that no clear fine structure is evident. Electron spin resonance studies are difficult but possible at dose levels feasible for germination studies. A promising attempt is in progress to correlate the effects on free radical concentration and on germination of seeds soaked in a solution of S,β-aminoethylisothiuronium bromide hydrobromide (AET) or in water and then vacuum dried.

A device has been built and put into use for high-intensity (~ 100 kr/min) irradiation of small (~ 0.2 cc) samples in the beam of an x-ray machine. Filtration with 1 mm of aluminum, a plastic layer for secondary-electron equilibrium, reproducible positioning and centering, and water-cooling are incorporated in the device. (AEC Activity 6130)

Biophysics: Ultraviolet Studies. – The synergism observed in the combined action of ultraviolet and x radiation in producing chromosome breakage in *Tradescantia* pollen has been repeatedly confirmed, and more quantitative studies of the phenomenon are in progress. The effect has also been found for pollen exposed to moderate doses of radiation from fluorescent lights and from sunlight (15 min). These investigations clearly indicate that the well-known day-to-day variation in the x-ray sensitivity of *Tradescantia* pollen arises from variations in the near-ultraviolet radiation received by the pollen from fluorescent lights or from sunlight either before or after x-ray treatment.

Experiments with the luminescent bacterium *Photobacterium fischeri* (order Pseudomonadales, family Pseudomonadaceae) have shown that a moderate

photoreactivation can be obtained. Photoprotection has not been found, although studies were conducted with various wavelengths of light throughout the near ultraviolet and at two different survival levels. These results may be accounted for by the unusual sensitivity of this organism to the lethal effects of near ultraviolet. Previous experiments on Pseudomonas aeruginosa, of the same order and family, showed large effects of both photoreactivation and photoprotection. The new results confirm the existence of photoreactivation in the order Pseudomonadales; they also show that photoprotection is a less universal phenomenon than photoreactivation, since it occurs in one, but not the other, member of the same family. (AEC Activity 6130)

Radiation Protection: Living Cells. — Most types of living cells have the potentiality to repair part of the damage produced by ionizing radiation. It is of real importance to understand the metabolic or biochemical processes involved in cellular repair. Experiments in this Laboratory have shown that the conditions of growth, which control the metabolic activity of bacterial cells, are determinants of the radiosensitivity of the resulting cell population. Radioresistant cells resulting from certain growth conditions may very well have greater potentiality to repair damage induced by ionizing and nonionizing radiation as well as by heat. An alternative explanation for the resistance of these cells is that they form, as a result of metabolic activity, a "built-in" protective chemical compound. Up to this time, the experiments conducted have failed to demonstrate the presence of such a compound in extracts of resistant cells. (AEC Activity 6230)

Mammalian Radiation Recovery. — In BDF_1 mice that have been lethally irradiated and treated with isologous bone marrow, recovery of the level of blood platelets is present eight days after irradiation and treatment and is apparently complete (normal counts in blood) by the third week. When homologous $(C3H \times 101)F_1$ marrow is used, recovery begins at the same time, but its course fluctuates considerably, and normal counts in the peripheral blood are not reached even after six weeks, when most of the mice have died from secondary disease.

Recent observations show that the livers of mice that have been treated with foreign bone marrow undergo a marked enlargement about 12 days after treatment. In some mice the liver is nearly twice normal size. This is thought to be a compensatory mechanism to produce extra materials for tissue regeneration.

Studies of the morphology, function, staining characteristics, and labeling of nucleoli in Ehrlich ascites tumor cells indicate that the nucleolar-like material is associated with the chromosomes during metaphase, anaphase, and telophase. (AEC Activity 6230)

Radiation Immunology. — Studies on the immune status of lethally irradiated mice treated with homologous adult bone marrow and fetal liver cells showed the latter group to be superior in response to RBC (red blood cell) antigens. The antibody response of the animals treated with homologous fetal liver was comparable with that of animals treated with isologous adult marrow; however, both revealed an immune response subnormal to that of the normal, nonirradiated control mice. These data suggest that either irreparable damage of the immune mechanism occurred after 950 r, or the "isologous" combination was in fact homologous as a result of gene drift.

Specific immune sera enhance phagocytosis of RBC of various species by mouse peritoneal exudate cells. The degree of enhancement appears to be dependent upon the genetic relationship between the antigen and the phagocytic cells. With mixtures of two types of RBC, in the absence of antisera, one type is preferentially engulfed. This can be reversed by addition of the specific antiserum.

Further studies with Millipore diffusion chambers revealed the following. Isologous spleen cells, when incubated with sheep RBC antigens, elicited both primary and secondary antibody responses in chambers with a pore size of 0.1 mμ when these chambers were inserted into the peritoneal cavity of mice exposed to 500 r. Spleen and bone marrow cells remained viable for 3 to 4 days in 0.1- and 0.45-mμ pore size chambers when these chambers were inserted into the peritoneal cavity of nonirradiated mice. Finally, antibody synthesis was detected by recipient mice when chambers (0.01 and 0.1 mμ) containing only sheep RBC or rat bone marrow antigens were inserted. Since these chambers are impenetrable to intact cells, these results suggest that cell fragments (haptene units) are able to diffuse out of the chambers and stimulate the host antibody-forming cells. Consequently, past interpretations of the host-donor reaction phenomenon based on diffusion chamber studies must be taken with reservation. (AEC Activity 6230)

PROGRAM 8000 – WORK FOR OTHERS

International Reactor Program at ORNL. – Currently enrolled in the two international programs, Reactor Operations Supervision and Reactor Hazards Evaluation, are 23 students representing 13 different countries. The students have completed 28 weeks of school since the beginning of classes in November.

Plans have been made for conducting a third session of international programs, starting in November 1960. Brochures and applications were distributed to 63 foreign countries and international organizations by the AEC Division of International Affairs. This distribution was made through foreign embassies and legations in Washington. Descriptive material for this program is also in the process of being distributed to industrial organizations and governmental agencies in this country. The deadline for receipt of all applications is July 31, 1960. (AEC Activity 8717)

ASEE-AEC Engineering Institute. – Plans are being made for presenting a series of ORSORT experiments in physics and radiochemistry to participants in the 1960 ASEE-AEC Summer Institute, conducted by North Carolina State College during the week of August 8, 1960. (AEC Activity 8736.3)

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