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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

MARCH 1959

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Oak Ridge, Tennessee
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OAK RIDGE NATIONAL LABORATORY
STATUS AND PROGRESS REPORT

March 1959

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 Reprocessing Pilot Plant. – During this report period 8.5 metric tons of uranium which had been irradiated at BNL and which contained 4.9 kg of plutonium was processed. Feed for solvent extraction was prepared by the continuous dissolution of metal in the Metal Recovery Plant dissolver. In order to dissolve 0.5 ton/day of uranium, it was necessary to maintain a 200% metal heel in the dissolver. In dissolution, 7.5 M nitric acid was used as the dissolvent, and 4.2 moles of nitric acid was consumed per mole of uranium. A standard Purex flowsheet was used for the recovery and decontamination of uranium and plutonium from fission products. After three complete cycles of solvent extraction the uranium and plutonium were decontaminated from gamma-emitting fission products by factors of 5×10^4 and 9×10^5, respectively. The major fission product contaminants in the uranium product were zirconium and niobium. Both plutonium and uranium products met purity specifications for subsequent processing. On the basis of losses from the plant, the product recoveries for uranium and plutonium were 99.0 and 99.7%, respectively.

The plant was shut down at the end of this report period for four days for replacement of a concatenated pulse column with a conventional pulse column. (AEC Activity 2351)

In-Line Instrumentation. – In-line scintillation spectrometers were operated in the Power Reactor Fuel Reprocessing pilot plants for continuously recording the level of the gamma activity (gross or any selected energy) of process streams. A monitor having a polished fluorothene tube as the sample cell has operated in the first-cycle product stream since January 11, 1959.

With a background of 2×10^4 counts/min, at which level the background from the fluorothene tube itself is insignificant, and with a stream reading 3×10^4 counts/min, the accuracy of the instrument was within 10%. A falling-stream-type monitor is now being tested in the second-cycle product stream.

A stainless steel test model of an in-line monitor for continuously recording the alpha activity of process waste streams has been installed in a test stream. A silver-activated zinc sulfide phosphor with multiplier phototube is encased in 1/4-mil Teflon and placed in contact with the stream. (AEC Activity 2724)

PROGRAM 4000 — REACTOR DEVELOPMENT

HOMOGENEOUS REACTOR PROJECT

Homogeneous Reactor Test. — During the run 19 startup, it became apparent that the flow rates in the main circulating loops were significantly less than the design flows. Flow determinations indicated rates between 180 and 200 gpm in the core loop and between 60 and 80 gpm in the blanket loop. The observed core differential pressures corresponded to normal flow (over 400 gpm) and normal pump discharge pressure. Similar measurements in the blanket system indicated that the blanket circulating pump was not delivering its required head.

Investigations of fuel stability have been postponed until the behavior of the main circulating systems is explained. There are several possible explanations for the low flow rates, and a systematic study is being made to determine whether they result from plugging by scale accumulation, mechanical failure of components, or other causes. (AEC Activity 4103.3)

Mathematics and Computation: Homogeneous Reactor Development. — Two programs were written for the IBM-704. One calculates the neutron flux present in a reactor due to uncollided neutrons and to those which have undergone one collision; the other calculates the Maxwell-Boltzmann-averaged cross sections of various materials, along with the associated f factor, as a function of moderator temperature. (AEC Activity 4103.1)

Reactor Analysis. — The power response of the HRT following a reactivity addition was calculated for various initial powers, on the basis that 40% of the reactor power is generated in the blanket. At an initial total power level of 1 Mw, the peak power following a reactivity addition of 0.5% Δk_e was about 4 Mw.

The amount of natural boron required to decrease the critical temperature from operating conditions to 20°C was calculated for three slurry reactors. In typical reactors about 50 ppm boron would decrease the critical temperature from 280 to 20°C in a thorium breeder having a 12-ft-dia core region; the corresponding boron concentration was 550 ppm in a 4 1/2-ft-dia reactor; about 700 ppm boron would decrease the operating temperature from 200 to 20°C in a 2 1/2-ft-dia reactor.

Samples of U^{233} irradiated in the LITR have yielded apparently low values (~0.03) for α^{23}. New U^{233} samples are in preparation for exposure in the ORR; boron and cobalt are associated with these new samples (to monitor the flux), along with U^{235} (to monitor the chemical procedures following irradiation). (AEC Activity 4103.1)

Homogeneous Reactor Instrumentation. – The miniature television camera being developed for viewing in radioactive areas was tested in a high gamma flux produced by Co^{60}. The camera head is 2 in. in diameter and uses a 1-in. Vidicon tube and a remote preamplifier that is connected to the camera by a 10-ft cable.

There was no observable instantaneous effect on the camera operation when it was lowered into a 1.5×10^6-r/hr field. However, the picture faded rapidly, required maximum target voltage after 20 min, and was not readable after 1 hr of additional exposure in a reduced field of approximately 5×10^5 r/hr. After further exposure for a total dosage of 5×10^6 r, the camera head was removed, and the camera lens was found to be practically opaque. The camera operated satisfactorily but with some loss of sensitivity after replacement of the lens.

A lens made from nonbrowning glass was ordered, and it is planned to repeat the test in a gamma field of about 10^5 r/hr. (AEC Activity 4103.1)

HRP Design. – Design drawings were made which show how the HRT primary circulating system piping can be altered to reverse the flow direction. The design also incorporates an assembly of hydroclones in a bypass stream for removal of solids. The underflow from these hydroclones would become the chemical-plant feed stream.

Several special tools are being designed and tested for possible use in removing the screens and repairing the hole in the HRT core tank. (AEC Activity 4103.1)

HRP Metallurgy. – Metallurgical examination of the specimens removed from the HRT was completed. It is still concluded that all damage to the Zircaloy-2 specimens resulted from their proximity to the titanium holder. A single titanium core specimen was recovered, and it showed only two small areas that had been at high temperatures. This specimen showed less "burning" than the Zircaloy specimens from the same station, indicating that the

geometry, rather than the composition, of the sample holder caused the "burning" difficulties.

Three Zircaloy-2 ingots were successfully fabricated commercially into a variety of sizes and shapes, by means of the fabrication schedule developed by ORNL. Rod and bar from 5/32 to 2 1/2 in. thick and both wide and narrow plates from 1/16 to 1 1/2 in. thick were produced. The procedures included a beta quench, followed by cold working and an alpha anneal, with no working permitted in the alpha-plus-beta region. (AEC Activity 4103.1)

Dynamic Slurry Loop Corrosion Tests. — The second phase of a 100A loop test at 280°C with a slurry of 0.9-μ, 650°C-calcined ThO_2 (furnished by the Westinghouse PAR Project) was completed. [See ORNL Status and Progress Report for January 1959 (ORNL-2691) for a description of the first phase of the test.] The second phase, under an oxygen atmosphere, was carried out at a circulating concentration of 378 g of Th per kg of D_2O, 0.5% U/Th (added as $UO_3 \cdot H_2O$ rods), and 0.001 m palladium-metal catalyst. The mean 528-hr attack rate of the austenitic stainless steel components (including pump impeller) of the loop was 2 mils/year (mpy). In the first phase, under an A-D_2 atmosphere, for 508 hr, the attack rate of the loop components was 1 mpy.

No detectable change in ThO_2 particle size, slurry settling rate (0.8 cm/sec at 280°C), or yield stress (0.1 psf at 25°C) occurred as a result of circulation under either atmosphere. At a pipeline flow velocity of 4.6 fps, 13% of the slurry solids was lost as a sludge from the circulating stream during the first phase of the test; in the second phase, 19% of the solids was lost, both by sludging and by some deposition as thin cakes on the loop piping.

With the O_2 atmosphere, attack rates, computed from undefilmed pin weight losses at 20 and 40 fps, respectively, were (in mpy): type 347 stainless steel, 1 and 7; Inconel, 13 and 22; titanium, 0.7 and 4; Zircaloy-2, 0.2 and 1.6; Croloy 5, 1 and 14; and Incoloy, 1 and 6. The attack rate of the circulating-pump scroll and thermal barrier, fabricated of titanium, was 8 mpy.

In another loop test, an oxygenated slurry of 1000°C-calcined, 10- to 30-μ thoria microspheres was circulated at 200°C for 553 hr at a concentration of 295 g of Th per kg of H_2O; the attack rate of the stainless steel

loop piping was 0.4 mpy at an average pipeline velocity of 8 fps. The spherical thoria particles did not degrade; however, nonspherical particles, which comprised approximately 20% of the charge, degraded, and thin deposits were found in the loop piping and circulating-pump impeller. Attack rates computed from weight losses of pin specimens exposed at 18 fps for 264 hr and at 26 fps for 289 hr were (in mpy): type 347 stainless steel, 0.3; Inconel, 5; titanium, 0; Zircaloy-2, <0.1; Croloy 5, 3.5; Incoloy, 0.5. (AEC Activity 4103.1)

In-Pile Autoclave Slurry Corrosion Studies. — A Zircaloy-2 autoclave experiment (L6Z-129S) containing thoria-urania slurry at a concentration of 1090 g of Th per kg of D_2O, 5 wt % U (coprecipitated) based on thorium, 0.0016 m Pd, and an oxygen atmosphere was terminated after nine weeks of in-pile operation. Following a week of operation at 280°C out-of-pile, it was inserted in LITR beam hole HB-6 and operated for three weeks at 280°C, one week at 320°C, two weeks at 240°C, and then three additional weeks at 280°C. The loop was operated for short periods at 200°C for gas generation and recombination studies.

During the first week of in-pile operation, the corrosion rate was 3 mils/year (mpy) at an average fission power density of 5 w/ml. Values for corrosion at 280°C were fitted by the equation

$$\text{mpy} = \frac{70}{\text{hr}} + \frac{130}{\text{hr}} \times \frac{\text{w}}{\text{ml}} \quad .$$

The radiation coefficient changed to 320/hr at 320°C and 3 w/ml. At 0.5 w/ml, no corrosion was detected at 240°C. During the first week after the return to 280°C, the incremental corrosion rate was 0.5 mpy at 3 w/ml. At 10 w/ml for the final two weeks of operation, the corrosion rate was 0.2 mpy.

Radiolytic-gas recombination by the palladium catalyst was studied by means of insertion and retraction experiments. A useful catalyst performance index, both applicable to reactors and observable in experiments, is defined as the power density (w/ml) resulting in an equilibrium pressure of 100 psi radiolytic D_2. It is related to catalytic activity and gas production as follows:

$$\text{w/ml (at 100 psi } D_2) = \frac{\text{catalyst activity (as moles } D_2 \text{ per liter per hr at 100 psi } D_2)}{0.38G_{D_2}} .$$

For the first few hours after the original insertion, this index was observed to be in excess of 38 w/ml for the 0.0016 m Pd catalyst at 280°C, but by the end of the first day it had dropped to between 5 and 6. After six more days of operations, the performance index was 4 w/ml. At the end of the third week at 280°C, the index was observed to be 2. Catalyst performance index values were between 1 and 2 at 320°C, 0.6 at 240°C, and 0.09 at 200°C. After the return to 280°C, index values of approximately 2 were observed during a week of operation. During the final two weeks most of the operation of the autoclave was at 12 w/ml and the index value was between 3 and 4 w/ml. (AEC Activity 4103.1)

Supporting Radiation Corrosion Studies. — By use of a special ion exchange column, the adsorption of uranium from UO_2SO_4 solutions on hydrous zirconium oxide was studied at high temperatures. Results from 3 1/2-hr runs at 250°C indicate uranium capacities of about 31 and 85 mg per g of oxide at uranium concentrations of 5 and 55 g/liter, respectively. All solutions were 0.02 m in excess H_2SO_4. These results are in agreement with those reported by Heston at the lower concentration, but are considerably higher than his results extrapolated to the higher concentrations.

Heston's data were obtained from batch-type experiments lasting about 1 hr, and there are indications from the present experimentation that the surface area of the oxides is decreased by extended exposures to high temperatures (250°C). The results of a 6-hr run indicate that the surface area reached an equilibrium value during this longer period. When plotted on a Langmuir-type plot of [(g of U per liter of solution)/(mg of U per g of oxide)] vs (g of U per liter of solution), the data from the 3 1/2-hr runs appear to give a straight line which may be used to predict the capacities at intermediate concentrations not studied experimentally. (AEC Activity 4103.1)

MOLTEN-SALT REACTOR (MSR) PROGRAM

Chemistry. — The compound $7NaF \cdot 6ThF_4$ is metastable with respect to $NaF \cdot ThF_4$ and $2NaF \cdot ThF_4$ in the NaF-ThF_4 binary system. However, this material is stabilized by a small concentration of dissolved UF_4. Accordingly, the

$NaF-ThF_4-UF_4$ ternary system shows a large primary phase field for the $7NaF \cdot 6ThF_4$–$7NaF \cdot 6UF_4$ solid solution; a similar phase field must be expected for the $NaF-BeF_2-ThF_4-UF_4$ quaternary system.

The solubility of CO_2 in molten $NaF-BeF_2$ (57-43 mole %) has been examined at 500, 600, 700, and 800°C at pressures from 0 to 2 atm. Experimental data demonstrate the linear dependence of solubility with saturating pressures. Henry's law constants, K, at these temperatures are $(7.930 \pm 0.355) \times 10^{-8}$, $(7.051 \pm 0.135) \times 10^{-8}$, $(8.324 \pm 0.081) \times 10^{-8}$, and $(10.760 \pm 0.453) \times 10^{-8}$ mole of CO_2 per cm^3 of melt per atm. Of all the gas solubility data for molten salt systems obtained to date, these are the first to show a nonlinear plot of log K vs 1/T. A minimum is observed on this plot in the vicinity of 600°C.

Molybdenum trifluoride has been prepared as a brown rhombohedral crystalline material by reduction of MoF_5 with metallic molybdenum. Identical material is obtained on melting of MoF_5 with excess iodine and subsequent removal of volatile products by heating under vacuum, and on treatment of MoF_5 with SbF_3.

In preparing complex fluorides of structural metals for corrosion research, it was found that molten NH_4HF_2 converts Fe_2O_3 and activated Al_2O_3 into $(NH_4)_3FeF_6$ and $(NH_4)_3AlF_6$, respectively, but does not attack commercial Cr_2O_3. Electrolytic iron powder reacts with NH_4HF_2 to form $(NH_4)_3FeF_6$. Granulated chromium metal, finer than 100 mesh, reacts to form $(NH_4)_3CrF_6$, but the reaction is slow.

The product obtained by fusing 1 mole of CrF_2 with 2 moles of SnF_2 appears to be stable at the boiling point of SnF_2 (~700°C). At 1000°C, SnF_2 had volatilized and left a residue of CrF_3. The probable intermediate, $Sn_3(CrF_6)_2$, has not been definitely identified, but the reaction is presumably

$$4SnF_2 + 2CrF_2 \xrightarrow{300°C} Sn_3(CrF_6)_2 + Sn$$

$$Sn_3(CrF_6)_2 \xrightarrow{1000°C} 2CrF_3 + 3SnF_2\uparrow$$

If less SnF_2 is used, a product which is thought to be $Cr_3(CrF_6)_2$ is obtained.

Cylindrical specimens (3/4 in. in diameter and 3 in. in length) of TSF graphite were outgassed at 1600°F by pumping for 24 hr and were subsequently

soaked for 48 hr in molten $LiF-BeF_2-UF_4$ (62-37-1 mole %) at 1250°F under a pressure of 15 psig of helium. Weight gains of the graphite that ranged from 0.2 to 1.9% were observed. While analyses of cuttings machined from the graphite rods are incomplete, available data show that the outer 1/16-in. layer contained most of the salt responsible for the weight gain.

Preliminary studies have shown that 40-g slugs of cast UF_4 can be dissolved in gently agitated $LiF-BeF_2$ (50-50 mole %) at 1250°F in less than 90 min. While additional experiments are necessary, it appears that fuel additions to compensate for uranium burnup in a molten-salt reactor can be accomplished by adding solid UF_4 (rather than low-melting mixtures of UF_4 with other fluoride salts) during reactor operation. (AEC Activity 4103.1B)

Physical Properties. — The experimental evaluation of the enthalpy and heat capacity of the salt mixture $LiF-BeF_2-ThF_4-UF_4$ (62-26.5-1-0.5 mole %) was completed for the solid state (100 to 450°C); measurements for the liquid state were reported previously. The enthalpy can be expressed as

$$H_T - H_{30°C}\ (\text{cal/g}) = -4.9 + 0.224t + (34.65 \times 10^{-5})t^2 \quad ,$$

where t is the temperature in °C. The heat capacity is then obtained as

$$c_p\ [\text{cal}\cdot g^{-1}\cdot(°C)^{-1}] = 0.224 + (69.3 \times 19^{-5})t \quad .$$

The heat of fusion at 450°C was found to be 64 cal/g. (AEC Activity 4103.1B)

Dynamic Corrosion of INOR-8 and Inconel in Molten Salts. — Three more INOR-8 thermal-convection loops have completed 1000 hr of circulation of BeF_2-base molten salts at a hot-leg temperature of 1250°F without any evidence of attack of the INOR-8. Two more Inconel thermal-convection loops completed one year of operation at a hot-leg temperature of 1250°F; heavy intergranular voids to a depth of 7 mils were found in the loop that circulated salt 130 ($LiF-BeF_2-UF_4$, 62-37-1 mole %) and to a depth of 9 mils in the loop that circulated salt 84 ($NaF-LiF-BeF_2$, 27-35-38 mole %). Inconel thermal-convection loops that operated for 1000 hr and for one year at a maximum temperature of 1350°F with $LiF-BeF_2-UF_4$ salts were found to have intergranular voids to depths of 4 and 15 mils, respectively. (AEC Activity 4103.1B)

INOR-8 Aging Studies. — Samples of INOR-8 have been aged at temperatures of 1000 to 1400°F for times up to 10,000 hr. Tensile tests showed no significant differences between the aged specimens and the annealed INOR-8. (AEC Activity 4103.1B)

Carburization of INOR-8. — Tests on two thermal-convection loops (one Inconel and one INOR-8) have been terminated, after operating for 4000 hr, for the purpose of observing possible carburization effects. The hot legs of these loops, which contained graphite inserts, were maintained at 1300°F. Preliminary metallographic examination of metal sections from these hot legs did not show carburization and thus confirmed other laboratory tests. (AEC Activity 4103.1B)

In-Pile Loop Tests. — The first MSR in-pile loop, which was operated in the MTR during the last quarter, was disassembled, and the plugging of the purge line was determined to have been the result of radiation damage to seal leakage oil. The internal purge system of the second loop is being modified to reduce the possibilities for purge-line plugging, and a charcoal trap will be installed on the inlet line to eliminate a repetition of the activity release. (AEC Activity 4103.1B)

Pump Development. -- An INOR-8 journal bearing having a 0.007-in. radial clearance was operated successfully under loads varying from 50 to 300 lb at 600 and 1200 rpm in molten salt 30 ($NaF-ZrF_4-UF_4$, 50-46-4 mole %). While the load was being increased from 300 to 400 lb at 600 rpm, the shaft power showed a marked increase at 382 lb, and the test was stopped. Examination revealed metal smeared over matching portions of the journal and bearing.

A centrifugal pump being operated in an endurance test has accumulated 15,144 hr of continuous operation with molten salt 30 at 1200°F, 645 gpm, and 2700 rpm. A similar pump equipped with a Fulton-Sylphon bellows-mounted seal has operated 12,478 hr with NaK at 1200°F, 1200 gpm, and 2800 rpm. (AEC Activity 4103.1B)

CIVILIAN-POWER ADVANCED GAS-COOLED REACTOR

LITR Irradiation Damage Experiments. — The experiment (L-1-ab) in which two stainless steel capsules (a and b) containing UO_2 pellets were irradiated in the LITR was terminated. Each capsule contained a 1-in.

length of pellets 0.156 in. in outside diameter and 0.078 in. in inside diameter. The density of the pellets was 95% of theoretical, and the UO_2 was 10% enriched. The duration of the experiment was 115 days. The central temperature for capsule "a" started at 1850°F and ended at 1500°F. A drop of 200°F occurred upon insertion of an adjacent experiment.

The "a" capsule gas sample was lost. The capsule holder stuck in the inner air tube, and while an attempt was being made to recover the capsule in the hot cell the capsule was inadvertently cut and the gases were released. Insufficient allowance had been made for the displacement of the capsule carrier. X-ray examinations are now being made to establish the initial position of the carrier in new experiments.

The "b" capsule gas may also have been lost. Gas samples were taken, but they showed no radioactivity on the gamma-ray spectrometer. The gas samples will be analyzed for argon and air on a gas spectrometer as soon as possible. Capsule "b" could have ruptured before the end of the test, but no activity was found by the exhaust monitors. It is also possible that the gas sample collected in the hot cell.

A cobalt flux monitor 1 in. from the capsules indicated that the flux during the irradiation was 2.8×10^{13} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1}, which is to be compared with a calculated flux of 2.1×10^{13} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1} and a value based on the temperature gradient of 2.5×10^{13} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1}.

Experiment L-2-ab was started February 26, 1959. The UO_2 pellets in the two capsules (a and b) are 0.156 in. in outside diameter, their density is 85% of theoretical, and they are 20% enriched. The initial central temperatures of the capsules were 2480°F, but the temperatures have dropped to 2060 ± 30°F. The change in temperature is attributed to the deterioration of the thermocouple, which could account for an initial drop in the temperature of 50°F per day, and to the insertion of an experiment in an adjacent facility that caused a 175°F drop in temperature. These capsules are to be irradiated for a period of 58 days. (AEC Activity 4110)

ORR Irradiation Experiments. — The equipment for the ORR irradiation experiments is being assembled and tested. Final checks are being made of

the instrumentation and controls for the experiments and the gas tubing system. The mechanical devices for use in inserting and removing the capsules are being tested. The various pinch valves in the system are being tried out to determine their integrity and the closing torque required.

A spot welder for seal-welding pinched tubes prior to removal was tested on a preliminary basis. Additional development work will be required to establish a totally satisfactory welding technique. (AEC Activity 4110)

GETR Irradiation Experiments. — The final components of the in-pile loop to be operated in the GETR were installed and leak checking was started. The compressors were operated on low-pressure nitrogen. Additional flux measurements are being made to determine the effect of the control rod position on the flux traverse at the capsule positions. (AEC Activity 4110)

Leak Testing of Fuel Capsules. — A simulated GCR fuel capsule containing steel pellets instead of UO_2 pellets was helium leak-tested to evaluate the leak test specification. The assembled capsule containing 1 atm of helium was tested with a leak detector which had been calibrated with a 10^{-6} std cm^3/sec leak. This is the smallest leak available at the present time for calibration purposes. The pumpdown time for the test chamber was 10 min, and the capsule showed no signs of a leak. A 0.013-in. hole was then drilled into the wall of the capsule to determine whether a leak would change the pumpdown time and whether such a gross defect would allow complete removal of helium from the capsule. The pumpdown time was increased to 12 min, and at the end of 15 min all traces of helium had been removed. A new check is being made with sintered UO_2 pellets to determine whether UO_2 will absorb enough helium to substantially increase the outgassing time. (AEC Activity 4110)

PACKAGE REACTOR DEVELOPMENT

Army Package Power Reactor (APPR-1). — The amount of boron loss during processing of APPR-1 composite fuel plates (26 wt % UO_2 and 0.13

wt % B_4C dispersed in stainless steel) was estimated by chemical analysis of fuel plates remaining from the manufacture of core 1 fuel elements. The analyses were performed by ORNL and by the Martin Company. The average of nine determinations by ORNL indicated that 23.9% of the original boron loading was lost during the manufacturing process, and the average of five determinations by the Martin Company indicated this value to be 12.9%. The over-all average boron loss of 20% is approximately double the losses anticipated while sintering fuel cores in hydrogen and indicates that appreciable losses occurred during preheating of billets prior to plate fabrication.

The reduced boron content as a result of the losses apparently has not affected the operation of APPR-1. The most recent data indicate that the reactor has performed satisfactorily for 10 Mwyr and will probably achieve the design life of 15 Mwyr. (AEC Activity 4601)

GENERAL REACTOR RESEARCH

Reactor Evaluation Studies. — Multigroup calculations have been used to calculate the effective resonance integral of U^{238} as a function of concentration in an aqueous, homogeneous system. The group structure and cross sections used did not produce the proper relation between effective resonance integral and uranium concentration; the results indicate that at strong resonances the group width and cross section have to be chosen on the basis of a trial-and-error procedure in order to obtain good agreement between multigroup calculation and experiment. (AEC Activity 4119)

Reactor Safety Studies. — The prototype control rod has operated successfully for over 2000 cycles. The determination of the curve of time vs displacement for a control rod during scram operation began on March 31. (AEC Activity 4252)

Power Reactor Fuel Reprocessing: Foreign Research Reactor Process. — The optimum dissolvent for the high-silicon Al-UO_2 Foreign Research Reactor fuel is 7.5 M HNO_3-0.005 M $Hg(NO_3)_2$. Use of this dissolvent results in a dissolver product 1.5 M in $Al(NO_3)_3$ and 1.3 M in HNO_3, the correct composi-

tion for the subsequent gelatin treatment that will be used to remove silica from the feed solution. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Darex Process. — The Darex process, involving dilute mixed nitric and hydrochloric acids, will be used for fuels containing mixtures of UO_2 and stainless steel, such as the APPR fuel. Confirming prior chloride removal tests, runs with the continuous acid addition, reflux, and double recycle flowsheets resulted in the removal of chloride from APPR dissolver product to concentrations of 200 to 240, 1240, and 30 to 55 ppm, respectively. The double recycle flowsheet has, therefore, been chosen as the reference flowsheet for pilot plant design. This flowsheet is a batch countercurrent system in which all the 13.3 M HNO_3 makeup is added in the final chloride removal step. The chloride concentration in the product from this final reflux step is inversely proportional to both reflux time and makeup HNO_3 concentration.

Hydrated silica in APPR feed from which the chloride was removed could not be dehydrated in boiling 10 to 16 M HNO_3 but was soluble in 8 M NaOH. Gravity setting of silica from Darex solvent extraction feed solutions for up to 48 hr produced supernatant liquors as clear as can be obtained with various settling and flocculating agents. Although APPR fuel element braze metal is soluble in fresh 5 M HNO_3-2 M HCl, it is not soluble in "mid-point" dissolver solution.

In corrosion tests, types 304 EEC and 347 stainless steel failed by local attack around weld areas in boiling 3 M HNO_3 containing 60 g/liter of dissolved stainless steel. The presence of chloride up to 800 ppm had no marked effect upon the severity of this attack. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Sulfex Process. — In further batch chemical decladding tests in which the stainless steel cladding of Consolidated Edison ThO_2-UO_2 fuel was removed in 5 hr with boiling 6 M H_2SO_4, uranium losses to the decladding solution increased from about 0.2% with 200% excess acid to 0.3 to 0.4% with 300% excess acid.

In tests on the continuous removal of the stainless steel cladding in boiling 6 M H_2SO_4, the results with respect to solution loading, reagent usage, uranium losses, and gas evolution were nearly identical with data from batch decladding tests. Continuous operation would be required in a geometrically safe Sulfex decladding-dissolving vessel.

In tests for determining the effect of undissolved stainless steel end plugs, etc., on dissolution of the ThO_2-UO_2, concentrations of ferric ion up to 1.0 M had no deleterious effect on the dissolution of the sintered pellets in boiling 13 M HNO_3–0.04 M NaF–0.04 M $Al(NO_3)_3$.

Preliminary data on the corrosion of titanium in boiling initial Thorex dissolver solutions show lower rates than expected, varying from 2.9 to 0.31 mils/month. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Zirflex Process. – In corrosion tests, welded specimens of type 309 Cb stainless steel showed maximum corrosion rates of 8.5 and 3.4 mils/month over a 72-hr exposure in boiling solutions representing initial conditions in the decladding and Zirflex core dissolution cycles, respectively.

In a new zirconium fuel process study on an alloy fuel containing zirconium, tin, and 7% uranium, 34.4 M HF–1.7 M H_2O_2, fed continuously to the dissolver, totally dissolved the alloy. The final F/Zr mole ratio was 6. Addition of $Al(NO_3)_3$ and HNO_3 to the dissolver product resulted in a stable solvent extraction feed containing more uranium than the normal Idaho Chemical Processing Plant feed and having no dissolved chromium. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Mechanical Processing. – In shearing tests on a mark I prototype fuel bundle (porcelain-filled stainless steel tubes), 22.2 g of metal fines ≤1680 μ in size was produced. Sufficient cold working was done by shearing to allow removal of all but 2.5 g with a small permanent magnet. Of the porcelain-metal fines mixture produced, 36.5% was less than 150 μ in size (vs 50% given for UO_2 last month).

A mocked-up Consolidated Edison fuel assembly consisting of 100 stainless steel tubes with 30-mil wall thickness joined together by 15-mil-wall ferrules was successfully disassembled, first into slabs, then into individual tubes with a simple portable impact wedge driven by a conventional air hammer. The Nicrobraz 50 bond between ferrules and tubes,

characteristic of this fuel assembly construction, has a tensile strength of 30,000 psi; the joints were easily broken with the impact wedge. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Solvent Extraction Studies. — A proposed Consolidated Edison second extraction cycle based on the TBP-25 system but with 0.1 M acid-deficient feed and with 2.5% tributyl phosphate as extractant gave a gross gamma decontamination factor of 10^5. Further improvements in this flowsheet cannot be studied in the present laboratory equipment because of the high radiation of the feed solution required. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Hermex Process. — In further studies to explore the use of mercury and amalgams as processing agents, additional determinations of the solubility of fission products in mercury have been made. The solubility of neodymium in mercury in the temperature range 100 to 300°C has tentatively been determined to be from 0.064 to 0.895 wt % at the respective temperature limits. Conflicting results of different analytical procedures require that the experiment be repeated. (AEC Activity 4301)

Power Reactor Fuel Reprocessing: Long-Range Studies. — Preliminary laboratory studies on the recovery of uranium and thorium from mixed unirradiated UC_2-ThC_2-graphite fuel (HTGR) indicate that grinding to -200 mesh and leaching with 15.8 M HNO_3 (and other acids) results in unacceptable (up to 10%) losses to the insoluble residue. Burning in oxygen and leaching of the resulting ThO_2-UO_2 ash with 13 M HNO_3—0.04 M NaF—0.04 M $Al(NO_3)_3$ proved satisfactory.

Since the first core of this fuel may be clad with stainless steel, the rate at which the fuel dissolved in two decladding reagents, dilute aqua regia and sulfuric acid, was determined. Sulfuric acid appeared to be the best choice for the decladding reagent, in that the uranium and thorium losses to boiling 6 M H_2SO_4 were less than 1.5% in 6 hr, whereas the uranium loss to boiling 5 M HNO_3—2 M HCl was approximately 33% in the same period of time. Mechanical removal of the stainless steel cladding is also being considered. (AEC Activity 4301)

Fused Salt-Fluoride Volatility Processing. – In a short-term corrosion test (27 hr), an Inor-8 dissolver vessel was exposed to a fused salt mixture of 55% NaF and 45% LiF in which the ZrF_4 concentration increased from 0 to 12 mole %. The corrosion rate at the interface between the salt and the hydrogen fluoride gas was 7.5 mils/day at 700°C. Corrosion rates are lower at lower temperatures; operation of a dissolver at 625°C with a salt mixture containing initially 10 mole % zirconium may be optimum. (AEC Activity 4302)

Waste Treatment. – Direct evaporation and calcination of acid wastes is being studied as a means of preparing high-level wastes for permanent disposal.

It has been found that the rate of heating affects the physical properties of the solids produced. For example, drying and decomposition of ferric nitrate at a furnace temperature of 300°C proceeded only about half as fast as at 400°C, but after calcination to 750°C the material dried at 300°C produced a cake of bulk density 2.2 (56% porosity), as compared with a cake density of 1.0 (80% porosity) obtained at the higher rate.

Batch evaporation of synthetic Darex (stainless steel) waste in glass to 136°C produced a free-flowing liquid suitable as feed to a calciner. The condensate from the evaporation step constituted about 85% of the volume and 75% of the weight of the original waste and contained about 65% of the nitrate.

When concentrated Darex waste was heated to dryness and calcined in a stainless steel pot, 8 in. in diameter and 18 in. high, it was found that the bulk density of the calcined solids decreased from 0.9 to 0.6 g/cm^3 when the average pot temperature during solids deposition was raised from 250 to 650°C.

Heat transfer calculations pertaining to the storage of cylindrical containers of waste showed that 20 to 50% more heat can be dissipated in air-cooled vaults than when the waste is buried in holes in the ground with a 1-in. air gap around the container. For example, the maximum temperature rise in an 8-in.-dia cylinder of solid waste evolving 1000 $Btu \cdot hr^{-1} \cdot ft^{-3}$ would be 800°F if it is buried in soil, 640°F if stored in a shaft in salt, and only 435°F if stored in a vault with air in natural convection. (AEC Activity 4352)

Long-Range Planning Studies. – In studies of high-flux isotope reactors, heat removal calculations indicate that average power densities up to 3.9 Mw/liter are achievable with water-cooled, plate-type fuel elements if the system is pressurized to 650 psi to prevent surface boiling. On this basis, 100 Mw of heat can be removed from a 14-cm-ID x 36-cm-OD x 30.5-cm-long annular fuel region, resulting in a thermal neutron flux of 3×10^{15} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1} in a 100-g sample of Cm^{244} placed in the inner island. (AEC Activity 4510)

THERMONUCLEAR PROJECT

DCX Facility. – The DCX machine is currently running with injected H_2^+ beams up to 3 ma in magnitude and with operating pressures less than 10^{-6} mm Hg. A typical operating condition is 1 ma of injected beam and a pressure of 3×10^{-7} mm Hg. The injected beam can, with effort, be raised above 3 ma, but the operating pressure is then likely to rise.

Emphasis has been placed upon study of the spreading of the volume occupied by the trapped H^+ ions. At pressures above 10^{-5} mm Hg, the trapped ions form a tight torus of major radius 5.3 in. and minor radius about 1/2 in. As pressures decrease, this region spreads until it occupies a volume about 14 in. long and perhaps 8 in. in radius. An obstacle introduced at one point causes a shrinkage all over, indicating that the particle orbits in the volume are thoroughly mixed. The extent of the spread can apparently be accounted for on a basis of the energies required of particles to reach given locations against the action of the mirror field, but the origin of the spreading remains obscure. Decay times of the accumulated trapped ions remain in the range of 6×10^{-3} sec when the injected beam is abruptly cut off. These observations have been made when the efficiency of dissociation of the H_2^+ beam was only 5%. This low value is believed to result from recent changes in the configuration of arc electrodes and baffles; it is believed that the former efficiencies of 20% or more are recoverable. (AEC Activity 4851)

Carbon Arc Studies. – In subsidiary experiments, it has been found that an H_2^+ beam need not pass directly through a carbon arc in order for appreciable dissociation to take place. If the H_2^+ beam passes at a distance of 1/8 in. or slightly more from the arc column as defined by its visible region and cathode dimensions, dissociation is almost as efficient as it is when the H_2^+ beam passes directly through the arc. (AEC Activity 4851)

PROGRAM 5000 — PHYSICAL RESEARCH

Reactor Operations. — The number of spurious shutdowns at the ORR has decreased to about one per cycle. Interaction between the beam holes and the neutron safety chambers makes it unsafe to drain and fill during operation. A special beam-hole plug and gamma safety chambers apparently will remedy this. A degasifier has been installed on the reactor water system and has markedly reduced the gaseous activity escaping into the reactor building.

Studies of the Graphite Reactor annealing have been completed. It is proposed to anneal by reversing the air flow and heating the graphite on the east side of the reactor. Most of the stored energy would be released at low temperatures so that no spontaneous release could occur.

PHYSICS

Neutron Velocity Selector. — Total neutron cross-section measurements were made on rolled and electropolished U^{233} and U^{235} metallic foils of very uniform thickness. These measurements were carried out with the new M2 rotor and associated equipment recently installed at the ORR. Both 11- and 45-m flight paths were used, and the new 2048-channel time-of-flight analyzer was employed in taking the data. Preliminary values of the total cross sections at 2200 m/sec are 588 ± 5 and 691 ± 6 barns for U^{233} and U^{235}, respectively. (AEC Activity 5220)

High-Voltage Experimental Program: Coulomb Excitation of the Second 2^+ State in Even-Even Medium-Weight Nuclei. — The location of the second 2^+ state has been established for seven even-even medium-weight nuclei by means of Coulomb excitation produced by 7- to 10-Mev alpha particles. The positions of the second 2^+ state are (in kev): Ge^{74}, 1200 ± 10; Ge^{76}, 1127 ± 10; Se^{76}, 1220 ± 8; Se^{78}, 1307 ± 9; Se^{80}, 1455 ± 11; Mo^{100}, 1050 ± 9; and Ru^{98}, 1402 ± 10. The values of B(E2) for excitation of these states are between 0.4 and 1.9 times the independent-particle estimate (with $R = 1.2 \times 10^{-13} A^{1/3}$ cm). With the branching ratio of cascades to crossovers obtained from radioactive-decay measurements, the values of the ratio of $B(E2;\ 2' \rightarrow 2)$ for the upper cascade to $B(E2;\ 2 \rightarrow 0)$ for the lower cascade are 0.7 ± 0.2 and 0.8 ± 0.2 for Se^{76} and Se^{78}, respectively. These values are appreciably smaller than the value of 2.0 predicted by the "near harmonic" model. Although corrections for double E2 excitation and angular

correlation effects from the triple cascade have not been included in these values, it appears rather unlikely that the ratio B(E2; 2' →2)/B(E2; 2 →0) will be larger than 1.0. With the known value of 49 for the E2/M1 ratio [Nuclear Phys. 4, 189 (1957)], the exceedingly small value of 3.9×10^{-4} is obtained for B(M1; 2' →2) for Se^{76}. (AEC Activity 5220)

High-Voltage Experimental Program: Neutron Radiative Capture Cross Sections from 10 to 150 kev. — The ORNL 3-Mv pulsed Van de Graaff accelerator has been used with a large liquid scintillator [Bull. Am. Phys. Soc. 4, 43 (1959)] to study neutron capture cross sections as a function of neutron flight time. The absolute cross section was determined earlier at 29 and 63 kev [Bull. Am. Phys. Soc. 4, 43 (1959)]. The relative values were normalized by means of gamma rays from the reaction $B^{10}(n,\alpha\gamma)Li^7$, whose cross section was assumed to vary as 1/v. The slope of the cross-section results ($\sigma \sim E^{-3/4}$ on the average) in the energy range 30 to 70 kev is in good agreement with the "threshold" measurements at 29 and 63 kev. Results have been obtained in the energy range 10 to 150 kev on the following elements: Ho, Tb, Ta, Pt, Pd, Ag, Au, Nb, In, and Sb. Departure of the cross section from a simple power dependence on energy is particularly noticeable for Pt, Ag, and Au. (AEC Activity 5220)

Theoretical Physics. — The V-A (vector—axial vector) theory of nuclear beta decay, as proposed by Feynmann and Gell-Mann and by Sudarshan and Marshak (1958), has met with considerable success in explaining most of the experimental results. But the question of the pseudoscalar interaction in nuclear beta decay has not been settled completely. Following the Rose and Osborn formulation [Phys. Rev. 93, 1315 (1954)] of pseudoscalar interaction, the longitudinal polarization of beta particles from 0 →0 (yes) transitions has been calculated with Coulomb effects included. The two-component theory of the neutrino and time-reversal invariance are assumed. A numerical analysis of the theoretical results is being carried out. (AEC Activity 5220)

Neutron Diffraction. — Neutron diffraction investigations on a fluoride of molybdenum have shown the existence of antiferromagnetic ordering in this material at low temperatures and should provide interesting information on the magnetic behavior of ions in the 4d transition series. This compound, which has been found by chemical analysis to be MoF_3, has a bimolecular rhombohedral unit cell very similar in dimensions to those which exist in

other trifluoride compounds [Acta Cryst. 10, 63 (1957)]. The molybdenum ions form hexagonal layers which are perpendicular to the unique axis of the crystal, and fluorine ions are located between these layers. Antiferromagnetic reflections are observed in the neutron diffraction patterns at low temperatures, and the disappearance of these reflections gives a Néel temperature of 185°K. The magnetic reflections indicate an antiferromagnetic structure in which the atomic magnetic moments are aligned parallel within each hexagonal layer and the moments of adjacent layers are antiparallel. In this type of magnetic structure, each metal atom is antiferromagnetically coupled to all of its six nearest neighbors. (AEC Activity 5220)

CHEMISTRY

Raw Materials Research and Development. — A two-step stripping method has been developed for the amine extraction (Amex) process. A concentrated $Ca(NO_3)_2$ solution is used to convert UO_2SO_4 in the organic phase to the nitrate prior to the water-stripping step. The product is a concentrated uranyl nitrate solution instead of the conventional solid sodium or ammonium uranate. Shipment of this solution to the refinery offers potential cost savings, since it would eliminate product calcining and packaging at the mill, simplify transfer of uranium between mill and refinery, and eliminate the nitric acid dissolution step at the refinery.

Emulsion difficulties in a number of solvent extraction systems are caused by silicic acid in the feed solutions. In the Amex process, emulsion formation has been reduced to acceptable levels by running with a continuous organic phase, since the water-in-oil dispersion is not stabilized by silica. Primary phase separation is rapid and loss of solvent by entrainment is reduced. Liquors containing 2 g of silica per liter have been handled successfully. (AEC Activity 5310)

Ion Exchange Technology. — The rate of sorption of uranium into 40-μ Dowex 21K resin particles from a 0.0042 M uranyl sulfate solution was studied by measuring the uranium loading on individual beads as a function of time. The beads became loaded to one-half their equilibrium value in 24 min, and the "apparent" over-all diffusion coefficient was 9×10^{-8} cm^2/sec. (AEC Activity 5310)

Nuclear Methods of Chemical Analysis. — In the investigation concerned with the use of short-half-life radionuclides in neutron activation analysis, it was found that Rh^{104m}, 4.4-min [produced by the neutron reaction Rh^{103}-$(n,\gamma)Rh^{104m}$], emits, in addition to reported gamma rays of 0.0511 and 0.0772 Mev, a 0.097-Mev gamma ray in its decay. This heretofore unreported gamma ray was characterized by decay studies and by special gamma scintillation spectrometry studies. It decayed with the same half life as that of the other gammas from Rh^{104m}, and the conclusion was reached that it is actually a gamma ray being emitted in the decay of Rh^{104m}.

As part of the short-lived-radionuclide investigation, experiments have also been performed on osmium. Short irradiations of 10 min each were made and the irradiated osmium samples analyzed by means of gamma scintillation spectrometry. In a decay study of the gamma-ray spectrum obtained, no full energy peak (photopeak) was found that could be associated with a short-lived osmium radionuclide. Therefore it was concluded that only the longer-lived osmium radionuclides, Os^{191m} (14-hr), Os^{191} (16-day), and Os^{193} (30.6-hr), could be used in the analysis of sample materials for trace osmium. (AEC Activity 5330)

PROGRAM 6000 — BIOLOGY AND MEDICINE

BIOLOGY

Cytology and Genetics: Insects. — Reversion of prophase chromatin to the interphase state is responsible for a large part of the mitotic inhibition that has been observed in cells of several types after irradiation. The present study shows that this phenomenon occurs in *Vicia faba* root tips, the cells of which are widely used to analyze radiation damage.

Since root-tip cells cannot be observed in the living condition, they must be fixed and stained for observation. Evidence of prophase reversion in such preparations is most convincing if there is complete disappearance of cells in mid-mitosis soon after irradiation, followed by a gradual decrease in the number of prophase cells and a concurrent increase in the number of interphase cells. A series of *Vicia* root tips given 1000 r of x rays satisfied these criteria — the number of cells in mid-mitosis dropped to zero 2 hr after irradiation and remained at that level for at least 6 hr. The number of cells in prophase began decreasing rapidly before this time and reached zero at about 5 hr after irradiation. The data obtained after exposures to 250 and 500 r suggest that prophase reversion also occurs at the lower doses. (AEC Activity 6130)

Pathology and Physiology. — Previous studies have shown that neutrons have a high relative biological effectiveness (RBE) for damaging the lens of the eye. This investigation was undertaken to determine the RBE of 14-Mev neutrons, as compared with Co^{60} gamma rays, in order to evaluate the influence of neutron energy in cataract formation. Young adult mice of the RF strain were exposed to gamma rays or neutrons (from the H^2-H^3 reaction) in a single or in two or three fractionated treatments (at a dose rate of 3 to 5 rads/min). Lens opacities, as viewed with a Zeiss-Opton slit lamp, developed in both gamma- and neutron-irradiated mice at a rate and to an extent that varied directly with the dose. Comparisons made at 20, 60, and 82 weeks after irradiation indicate that the RBE for 14-Mev neutrons ranged between 1.6 and 3.1. In general, the RBE tended to decrease slightly with increasing dose. A total of 750 rads of gamma rays or 400 rads of neutrons fractionated in two or three successive doses over periods up to 70 days was essentially as effective as the same total

dose administered in a single exposure. The RBE of 14-Mev neutrons for cataract induction is less than that of neutrons of lower energy, which was observed earlier to be between 3 and 4; however, it is greater than the RBE of the same radiations for 30-day lethality (1.5) or intestinal weight loss (2.0). (AEC Activity 6130)

Biophysics. — Electron spin resonance spectrometers are now being operated at both X and K bands. Indications have been found that the resonance patterns observed in crystalline glycine may be a function of the applied magnetic field.

Comparative studies of radiation-induced magnetic resonances and color centers in glycine, in glycine adsorbed on silica gel, and in glycine dissolved in urea continue to support the hypothesis that the magnetic centers are not isolated free radicals but more probably arise from some type of lattice defect.

Previous work on irradiated dry deoxyribonucleic acid has been confirmed. A weak resonance has been observed which is very stable if the material is kept dry; however, the radicals decay rapidly and irreversibly when the material is allowed to take up water. Similar behavior has been observed in irradiated *Tradescantia* pollen; there is an initial increase in resonance signal and a subsequent decay to an apparent steady state when the material is kept dry.

The decay of resonances in glycine, α-alanine, glutamic acid, and glycylglycine samples has been observed for 185 days following protracted (15 days), heavy irradiation (1.6×10^7 rads). Glycine showed a decrease to 40 or 60% of the initial signal at 40 days with very slow decay thereafter; the others varied less than ±20% over the entire period. DL-Leucine showed little decay in the first 12 hr and then a rapid, apparently exponential decay with a half life of about 30 hr; this suggests a short-lived precursor excited state, not detected by electron spin resonance techniques, which decays to give the detectable state, which in turn decays exponentially. (AEC Activity 6130)

Microbial Protection and Recovery. — Several years ago it was found that anaerobiosis and presence of glucose in a growth medium reduce the radiosensitivity of *Escherichia coli* B/r. This effect of oxygen during growth of the cells is not at all related to the well-known oxygen effect

during irradiation. The sensitivity change has the nature of producing a shoulder on the survival curve. The phenomenon has been demonstrated on a number of different E. coli strains, and some investigators have attributed the effect to clumping. Some recent experiments on the H_7 strain of E. coli showed the same effect. In the latter experiments, microscopic examination of the cell suspensions indicated no clumping; moreover, the amounts of DNA, RNA, and protein per viable cell were no different than for cell suspensions which showed essentially exponential inactivation curves. Thus it appears that clumping or production of multinucleate or polyploid cells cannot be the explanation for the differences in radiosensitivity. Recent experiments indicate that the resistance of the cells is not dependent on glucose or anaerobiosis per se, since highly buffered cultures or those in which the pH is prevented from falling during the culture cycle do not develop this resistance. The resistance appears to be related to some product of glucose metabolism at acid pH or to a physiological state of the cell induced by these conditions. (AEC Activity 6230)

Mammalian Recovery. — Chemical protection of the intestine of irradiated mice (x-ray dose, 900 to 2000 r) has been established. The chemical S,2-aminoethylisothiourea (AET) prevents (1) shortening of the small bowel, (2) intestinal weight loss, and (3) much of the histological damage to the intestine. Chromosome damage was reduced, and protein synthesis was considerably increased above normal in irradiated mice protected with AET. The decrease in intestinal potassium concentration was largely offset by AET, and there is cytological evidence that this chemical protects the cecum and colon of irradiated mice. A dependency of intestinal cell hypertrophy on x-ray dose has been observed. The underlying mechanisms of this observation are being examined by histochemical and autoradiographic methods.

Although AET has previously been shown to protect the bone marrow and intestine, this chemical does not protect the testis of irradiated mice as judged by organ weight and histological appearance.

Damage to bone marrow irradiated in vitro with AET is not reduced by making the suspensions anoxic. Increasing the concentration of AET in

irradiated bone marrow suspensions does not appear to enhance radiation resistance of the cells.

The possibility of slow freezing of mouse bone marrow in various additives is being investigated, with the hope of finding a more suitable additive than glycerine. Marrow cells are being tested for viability, after freezing, in lethally irradiated isologous mice and also by various dye uptake methods. A long-term program to test the protective ability of frozen mouse bone marrow after slow freezing in 15% glycerine and storage at -10, -30, -76, and -196°C for up to five years has been initiated. (AEC Activity 6230)

Radiation Immunology. — With the use of lethally x-irradiated mice as a tissue culture in vivo, it has been possible to study in detail the relation between antibody response and cell number by transplantation of isologous spleen cells from normal and preimmunized donors. A linear relation between the logarithm of the titer and the logarithm of the number of nucleated cells has been obtained, independent of the time of bleeding. The results indicate that the antibody-producing activity of 1.5×10^6 nucleated spleen cells or less can be readily detected, thus permitting measurement of the number of potential antibody-producing cells of mice after various treatments in vivo or in vitro.

In further studies on the antibody-producing mechanism of the mouse, lethally irradiated mice were injected with isologous nucleated spleen or bone marrow cells from normal or preimmunized donors. The data show that under appropriate experimental conditions as few as 12×10^6 spleen cells from preimmunized donors produced about as many circulating antibodies in irradiated hosts as in normal nonirradiated mice undergoing a secondary response. In agreement with past studies, the contribution of bone marrow cells was low or completely absent. (AEC Activity 6230)

PROGRAM 8000 — WORK FOR OTHERS

International Reactor Courses at ORNL. — Currently enrolled in the two specialized educational programs, Nuclear Reactor Operations Supervision and Nuclear Reactor Hazards Evaluation, are 26 students representing 14 countries. The students have completed seven weeks of school since the beginning of classes in February. Tentative plans have been made for starting a second session of the international program in the fall of 1959. (AEC Activity 8717)

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