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AEC RESEARCH AND DEVELOPMENT REPORT

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

AUGUST 1953

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

August, 1953

W. H Sullivan

and

F. T. Howard

Date Issued: SEP 8 1953

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August, 1953

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OAK RIDGE NATIONAL LABORATORY

STATUS AND PROGRESS REPORT

August, 1953

The subject material of this report represents approximately half of the Laboratory's program, which is covered with some exceptions on a bimonthly schedule.

PROGRAM 3000 - WEAPONS

Special Electromagnetic Separations - Since the ionization efficiency for cerium resembles that of plutonium, cerium trichloride feed was used in the development of a two-arc ion source for separating plutonium isotopes. By this technique a marked improvement in process efficiency was obtained, with no decline in product enrichment. A shipment of feed material from Chalk River, containing 13% Pu^{240} and 1.8% Pu^{241} will be used for the enrichment of the heavier isotopes of plutonium. A portion of the 5% Pu^{240} feed supply previously obtained from Chalk River was converted to the trichloride.

The residue from a separation run to enrich thorium-230 was recovered, purified, and converted to charge material in preparation for a second run. (AEC Activity 3601)

PROGRAM 4000 - REACTOR DEVELOPMENT

HOMOGENEOUS REACTOR PROJECT

Homogeneous Reactor Experiment - Installation and check-out of the Pulsafeeder pump were completed on August 10. Natural uranium fuel solution was charged into the reactor and a 72-hour corrosion run was made at 200°C with a fuel concentration of 35 gm. U/kg. of solution. The purpose of the natural uranium run was threefold: 1) to measure the overall corrosion rate of the system, 2) to obtain a material balance, and 3) to determine whether or not contaminating ions such as nitrate or chloride were present.

The corrosion rate, measured during this period at 200°C, was approximately 5 mils per year initially, with the rate approaching 1 mil or less per year after 72 hours. These corrosion rates were measured by the increase in concentration of nickel ions in the fuel solution.

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The uranium material balance for this run checked within <0.5% of the 3.5 kilograms of uranium charged into the system. There was no evidence of the presence of contaminating ions.

Following this run the natural uranium was washed from the system and preparations were made for the addition of enriched uranium fuel solution. During this time also repairs were made on the shim and safety rod mechanisms. This work is now complete and operation of the HRE at low power levels is scheduled to be resumed in September. (AEC Activity 4103.1)

HRP Chemistry - Experiments on the effect of reactor radiation on the static corrosion of type 347 stainless steel do not give a clear picture of the corrosion mechanism, but do indicate that in some cases the corrosion proceeds as a linear function of the square root of the time of exposure. In other cases the corrosion may start out in this manner but then change to give an even greater corrosion rate. This latter effect has been observed only with the highest fission density studied (equivalent to 6 kw/liter).

Present plans for the intermittent addition of acid to the "in-pile loop" call for the addition of a mixture, comprising a measured amount of sulfur dioxide and oxygen, to react in the presence of the ions of the solution to form sulfuric acid. A series of laboratory experiments have indicated that this procedure is feasible, with the desired oxidation occurring in less than one hour at 200°C without reduction of the uranyl ion.

Studies of the solubility of sulfates, phosphates, fluorides and selenates of thorium and of the complexing action of combinations of these anions have indicated there is little possibility of finding a satisfactory aqueous thorium breeder blanket system. Adequate solubility can be obtained only with the phosphate and nitrate. The phosphate solutions are extremely corrosive and very viscous. Thorium nitrate is sufficiently soluble but doubt exists concerning the stability of the nitrate ion under radiation. A fundamental study of the effect of various types of radiation on nitric acid, uranyl nitrate solutions, and thorium nitrate solutions is in progress. (AEC Activity 4109.31)

Homogeneous Reactor Fuel and Blanket Studies - It was reported earlier (ORNL-1545) that thorium oxide slurries showed promise for the homogeneous reactor blanket material, but that when the oxides were prepared by high temperature calcination (above 650°C), the slurries were abrasive to stainless steel. Thorium oxide, prepared by (1) autoclaving thorium hydroxide in water at 250°C and by (2) calcination of thorium oxalate at 500°C with subsequent hydrolysis of the solid in water at 250°C, was shown to form less abrasive slurries. When thorium tetrafluoride was added to the oxide (F/Th mole ratios of 0.5 and 0.9) prior to the autoclaving, the resulting slurries were more abrasive.

By autoclaving thorium oxide at 250°C in water with uranium, in the form of uranyl carbonate (ORNL-1545), uranyl sulfate, or uranyl fluoride and at low uranium/thorium ratios, a mixed oxide of uranium and thorium was produced. The addition of sodium fluoride before autoclaving prevented the formation of the mixed oxide.

The irradiation of thorium oxide pellets for three weeks in the LITR at room temperature at a thermal neutron flux of 5×10^{12} neutrons per square centimeter per second caused some pellet breakage.

Details of the homogeneous reactor fuel and blanket studies are reported in ORNL-1605. (AEC Activity 4109.31)

Engineering Investigation of Slurries - Uranyl phosphate slurry has been circulating for approximately 400 hours at about 100 gallons per minute in a stainless steel system at 250°C. Samples which have been withdrawn indicate little, if any, change in concentration; and a radiation survey around the loop indicates no large localized buildup of uranium.

Modifications to the pressurizer and pump of a simple high pressure, 250°C loop appear to have reduced considerably the tendency of UO_3 slurry to accumulate as a solid cake. The current run has now been in operation for about 150 hours. (AEC Activity 4109.31)

Dynamic Corrosion Studies - A 4400 hour continuous run at 250°C with a 5 g U/l uranyl sulfate solution containing 0.005M H_2SO_4 was completed recently. The corrosion rates of 300-series stainless steel pin specimens were uniformly less than 0.25 mil per year (mpy) for "as-machined" pins prepared from "as-received" rod, welded plate, and cast pipe. The bulk flow velocity of 11 to 15 feet per second over the specimens was perhaps a little low, but other flow data obtained in the same tests with different 347 stainless steel specimens indicated no increase in this corrosion rate up to a bulk flow velocity of 59 feet per second. Shorter runs have indicated that the metal weight loss, which is the basis of this 0.25 mpy corrosion rate, is accounted for in the first one or two hundred hours of exposure while the protective coating of oxides is forming. Thereafter no further weight loss is observed, within the limit of error of the measurements. Thus, the actual corrosion rate, once the oxide coating is formed, is substantially less than 0.25 mpy. A reasonable lower limit of about 0.05 mpy is indicated by long term static tests.

A series of preliminary tests at 275° and 300°C has also been completed. Indications are that there is less metal attack during the formation of the protective oxide coating and that this film is more resistant to higher bulk flow velocities, under a given set of solution conditions.

Preliminary tests of the sampler designed for use in the in-pile loop are in progress.

Two "torus rotator" assemblies were constructed and preliminary tests are in progress. The initial results suggest that correlation of these data with dynamic loop tests may not be as difficult as originally anticipated. If this is actually the case, the tori will be very useful in expanding information on the significant variables indicated by the loop studies.

A review of the corrosion test data on zirconium obtained thus far indicates there is no corrosion, within the limit of error of the weight change measurements on crystal bar or the 2.5% tin alloy, under exposure to any of the uranyl sulfate solution conditions tested. (AEC Activity 4109.4)

Boiling Reactor Research - In cooperation with Los Alamos personnel, a team from ORNL conducted a series of experiments on the Supo Water Boiler at Los Alamos. In these tests, reactivity, amounting in some cases to almost 0.7 of that required for "prompt critical," was introduced rapidly under both boiling and non-boiling conditions. In other tests, comparable amounts of reactivity were removed suddenly from the reactor. The neutron flux level of the reactor and the position of the "poison" rod were recorded simultaneously as a function of time by means of a two-pen Brush oscillograph. Also recorded were the fluctuation in neutron flux level during steady state boiling.

The following conclusions can be drawn so far from the results of the experiments:

1. There is a definite limit to the extent of power oscillation during steady state boiling in the Water Boiler at the power levels of operation (up to about 7 kilowatts).

2. There is also a definite limit to the reactor power period while boiling under these conditions.

3. At a given power level, the reactor compensates more rapidly for suddenly imposed increases in reactivity when boiling than when it is not boiling.

One minor incident occurred during the tests. As a result of boiling the reactor with too high a liquid level, some of the fuel solution was forced to the top of the narrow neck attached to the spherical reactor vessel, thereby setting off radiation alarms and scramming the reactor. There were no over-exposures of personnel. When the solution level was lowered, the reactor boiled with remarkable stability. (AEC Activity 4112.1)

AIRCRAFT NUCLEAR PROPULSION PROJECT

Aircraft Reactor Experiment - The modifications of the reactor and external fuel system, as required to reduce uranium investment in the ARE, have been specified. The volume of the external fuel system has been decreased from 7.5 cubic feet to 5.75 cubic feet by the elimination of one of the two parallel

heat exchangers, and the critical mass of the reactor itself has been reduced by the removal of some stationary structural poisons around the control rods. As a consequence of the above changes, the total uranium requirement for the ARE is now approximately 224 lb. An additional 101 lb of uranium has been requested from the AEC to supplement the original request for 152 lb. The extra 29 lbs will provide for such contingencies as may yet arise.

Although the removal of structural metal around the control rod will require higher control rod temperatures, which will be tolerable, the elimination of one heat exchange loop will reduce the ΔT across the reactor and hence the total reactor power by 50% (i.e., to 175°F and 1.5 Mw, respectively). It will be necessary to install a bypass with a throttling valve around the remaining heat exchanger in order to maintain the same total flow in the system without increasing the pressure drop across the reactor. The outlet fuel temperature is maintained unchanged at 1500°F.

As neither packed- nor frozen-seal pumps have demonstrated sufficient reliability for remote operation, the sump-type, gas-seal pump is now specified for use in both the reflector (sodium) circuit and the fuel circuit. Although the required head and flow can be attained in the fuel circuit by a single pump, a second pump, isolated by frangible-disk valves, will be available. A frangible-disk valve, which provides quick, certain operation (though a 'one-shot' proposition) will also be installed between the system and the hot-fuel dump tank. (AEC Activity 4401.1)

Experimental Engineering - Mechanical Contrivances - The 80 gallons per minute (gpm), sump-type, gas-seal centrifugal pump, which will be used in both the fuel and sodium circuits of the ARE, has satisfactorily pumped the mixture $NaF-ZrF_4-UF_4$ (50-46-4 mole %) at temperatures up to 1520°F and at ARE flow rates and pressures. The pump operated for 546 hours before being shut down to investigate bearing noises. The inner race on the lower bearing was found to be slipping on the shaft due to improper fit of the outer race in its housing; this was readily corrected. In other respects the pump performed reliably. It was easily primed and eliminated entrained gas in less than 10 minutes. Furthermore, the pump showed no tendency to cause gas entrainment throughout its entire operating range.

Development of alternative pumps, for both the fuel and sodium circuits, was continued at a reduced level. The ARE-size frozen-sodium-seal pump, which was altered to incorporate a short (1/2 in. long) seal with approximately 0.040 in. clearance, has operated for 150 hours with no leakage and a uniform power requirement of approximately 3/4 Hp (1/10 that of the 5-13/16 in. long seal). Three seals, packed with various BeF_2 mixtures and tested with an ARE-size fluoride pump, have given excessive leakage. A very short (1/4 in.) frozen fluoride seal operated satisfactorily on the same pump for one day before leaking seriously.

To determine the significance of the non-wettability of graphite by fluorides, a Graphitar-14 shaft sleeve rotating against a stationary Graphitar-14 sleeve was tested on a 1-3/16 in. dia. shaft. Provisions for both heating and cooling the seal housing made it possible to maintain near-zero clearance between the rotating shaft sleeve and the stationary member. The seal was operated for 195 hours sealing against the mixture $NaF-ZrF_4-UF_4$ (50-46-4 mole %) at pressures to 10 psi. Leakage rate was 4 grams/day and power dissipation in the seal was very low. Two successful stop-start tests were run. When examined, the Graphitar was found to be only slightly worn; apparently the non-wetting property of graphite is of considerable significance in seal design. (AEC Activity 4401.1)

Critical Experiments - Two modifications of the reflector-moderated circulating-fuel reactor, to evaluate the feasibility of using graphite instead of beryllium, were examined. One assembly, that had been critical with a beryllium island, fuel annulus, and beryllium-graphite reflector, was rearranged by removing the beryllium island and redistributing the fuel to form a two-region system with an 18" dia. core. This arrangement was far from critical. The assembly was then modified to give a 21" dia. core composed of 50-50 volume percent graphite and powdered fuel. Surrounded by a beryllium reflector 13-1/2" thick, the assembly had a critical mass of 15.2 kg U^{235}. Both the total and thermal neutron fission rates at the edge of the core were about four times those at the center. The addition of a cylindrical shell of stainless steel, 18" long and 0.06" thick around the mid-section of the core, increased the critical mass to 17.3 kg U^{235}. Not only is this critical mass considerably more than that of the comparable clean assembly with beryllium, but also the graphite-containing assembly had more reflector and a poorer power distribution.

Calculations, using the Goertzel-Selengut method with 32 lethargy groups, have been made for a few critical assemblies of aqueous solutions of a uranium salt, enriched to 93% in U^{235}. The values obtained for "k-effective" have been satisfactory over the range of hydrogen-to-U^{235} atomic density ratios from 50 and 750 (lower and upper limits, respectively, for such solutions). By comparison, the Fermi age model, using the same cross section values, gave a k-effective of 1.255 for a $H:U^{235}$ ratio of 50, the error being due primarily to an underestimate of fast neutron leakage. In contrast to other more approximate semi-empirical methods, the Goertzel-Selengut multi-group method depends only on basic cross section data and is applicable over a wide range of $H:U^{235}$ ratios, including intermediate and fast systems. Neutron leakage and absorption spectra are also obtained from the calculations. (AEC Activity 4401.2)

Radiation Damage - A study was made of the corrosion of Inconel capsules containing the fuel mixture, $NaF-ZrF_4-UF_4$, (50-46-4 mole percent respectively) as a function of time-of-exposure in the LITR (230 watts/cc irradiated fuel). The corrosion of several capsules irradiated so far at 815°C displayed a time dependence of approximately 1/2 mil per 100 hours, except for one sample which had only 1 mil total corrosion after 810 hrs.

In-pile creep tests were made in the LITR to determine the temperature-dependence of the creep rate of Inconel at 3,000 psi in the neighborhood of 1500°F. Tests with and without radiation in both air and an inert atmosphere showed no significant differences.

The ARE beryllium oxide moderator and reflector is cooled with sodium in an Inconel environment. To determine the extent of radiation-induced corrosion in such a system, Inconel capsules containing beryllium oxide immersed in sodium were irradiated in the LITR. After an exposure of about 2×10^{19} nvt (neutrons/cm^2) thermal at 1500°F, the beryllium oxide samples were removed and the sodium was carefully separated by vacuum distillation. The weights of the BeO samples after the test were within 0.1% of their original weights. No cracks or pits were found in the beryllium oxide and chemical analysis showed no beryllium in the sodium. (AEC Activity 4401.4)

Reactor Chemistry - Whereas the original calculations indicated that the ARE would go critical with some 4 mole percent U, the latest critical mass estimates indicate that some 6-1/2 mole percent U will be required. Measurements of the physical properties of new fluoride mixtures, as well as examination of the anticipated reactor loading procedure, showed that the new fuel concentration can be obtained with the $NaF-ZrF_4-UF_4$ system. The loading procedure involves the addition of an enriched uranium-bearing material to a barren mixture (in this case $NaZrF_5$) and is dependent upon 1) the temperature of the barren mixture, the enriched mixture and all intermediate mixtures, 2) the volume of the enriched mixture (only 1-1/2 cubic feet is available for the concentrate) and, 3) the uniformity of the mixtures. These requirements are best fulfilled if Na_2UF_6 is employed as the fuel concentrate. It is mixed with $NaZrF_5$ in such proportions that the resulting $NaF-ZrF_4-UF_4$ fuel mixture has a composition of 53.5-40.0-6.5 mole %, respectively.

Studies of purification treatments for Na_2UF_6 show that the hydrofluorination and hydrogenation which was effective for $NaZrF_5$ can be used without modification from a chemical standpoint although criticality considerations require modification of the apparatus' geometry. The use of zirconium metal as a reducing agent in the purification of $NaZrF_5$ apparently results in the retention of excess reducing power by the filtered melt; it is possible that soluble materials containing trivalent zirconium exist in these melts.

Quenching techniques were developed as part of a program for more detailed studies of important regions in the fluoride phase diagrams. The compound $NaZrF_5$, earlier regarded as melting congruently, was found to give a complex melting behavior, although the liquidus temperature was confirmed to 510°C. (AEC Activity 4401.5)

Corrosion Studies - Several static and rocking-capsule corrosion tests were made to determine the effect of the carbon, oxygen, and nitrogen contents of stainless steels on their resistance to attack by fused fluoride salts. The presence of these elements in concentrations up to 0.19% C, 0.2% N_2 and 0.031% O_2 had no measurable effect on the corrosion of the stainless steels. Inconel

alloys of varying purity were also prepared and tested for corrosion in the fluoride mixtures for 100 hours at 815°C. Except for the extremely high-carbon Inconel, which was more severely attached, the special Inconel alloys were not significantly different from commercial Inconel alloys with respect to attack by the fluorides.

Studies of the dynamic corrosion of fluorides in Inconel thermal convection loops have advanced to the point where the detailed corrosion mechanism may be observed. In particular, while the total corrosion at 1500°F and 1650°F is comparable after 500 hrs, the initial corrosion rate at 1650°F is much greater than that at 1500°F. Loop tests up to 3,000 hours show that the initial high corrosion rate (due to NiF_2, FeF_2 etc. impurities) is superseded after 250 hrs. by a much lower corrosion rate. Furthermore, in the loop operated for 3,000 hours, mass transfer of chromium was evident for the first time. In the operation of these loops the "hot-leg" deposit, formed when ZrH_2 was added to reduce impurities, was eliminated by filtering the mixture before filling the loop.

In the investigation of corrosion mechanisms, the limit of solubility of Fe^{+++} in ARE-type fuel at 600°C was about 0.43% by weight, whereas Cr^{+++} was soluble to 3% or more, as determined by a technique involving high temperature (800°C) mixing of the complex fluorides, lower temperature (600°C) filtration, and subsequent analyses of the filtrate.

From the operation of experimental fluoride systems it has been demonstrated that Type 316 stainless steel is preferentially attacked by fluorides in the presence of Inconel. Corrosion of Inconel was practically eliminated while an adjoining Type 316 stainless steel section was severely attacked.

Additional studies on the mass transfer of container materials in liquid lead using small thermal convection loops have given the following results: 410 stainless steel operated for 545 hours with only a small amount of mass transfer; nickel and Nichrome V plugged after two hours and twelve hours, respectively; chromium metal plugged in approximately 100 hours; and a 30% nickel-iron alloy plugged the loop after 275 hours of operation. (AEC Activity 4401.5)

Metallurgy - The creep and stress rupture tests for both coarse and fine grain Inconel, as measured in fluoride mixtures at 815°C, were completed for the stress range of 2,500 to 7,500 psi. Specimens have now been in test for 1,000 hours at 1,500 and 2,000 psi.

The Ni-P alloy does not have good resistance to air-oxidation or sodium corrosion. If, however, the brazed joint is subsequently electroplated with chromium, the corrosion resistance of the basic Ni-P eutectic is increased. This chromium plating is also advantageous since it increases the over-all load-carrying ability (although it does not impart more ductility) to the brazed joint. The satisfactory results obtained from the use of these alloys

is encouraging since the parts to be brazed can be preplated with the Ni-P braze alloy by the 'electroless' plating technique. Tests of tube-and-fin joints thus brazed indicate, however, that it will be necessary to plate more than 1 to 1-1/2 mils of this brazing alloy on the tube to produce a sound, well-filleted joint.

The cracking tendencies of the nickel-chromium-silicon alloy (69-20-11 wt %) when applied to a test radiator, were determined by thermally cycling the entire assembly from 1000°C to room temperature. Only one small leak occurred after a severe water-quench from 1000°C. (AEC Activity 4401.5)

Heat Transfer Research - Laminar and turbulent flow phenomena were studied in a full scale glass replica of the ARE fuel tube system. The apparatus was instrumented with pressure taps, an inclined manometer, a hypodermic dye tube, and a rotameter. Pressure drop measurements and dye filament diffusion observations were used to determine the flow characteristics at various Reynolds moduli. It was found that the flow changed from laminar to transitional flow at a Reynolds modulus of about 2,400 and that fully established turbulent flow was attained beyond values of 5,000.

Radial temperature measurements were made in a simulated circulating-fuel reactor system to determine maximum fuel wall temperatures. Volume heat generation was effected by passing current through sulfuric acid, as it in turn was pumped turbulently through the 9/32" dia. test section. The ranges of parameters investigated were: Reynolds number, 5,800 to 14,100; Prandtl number, 4.6 to 8.7; source strength, 0.1 to 0.4 kw/cc. Wall-fluid temperature differences ranged from 3.5° to 9°F for this system. These experimental radial temperature differences agreed within ±30% of the theoretical values (see ORNL-1395). When these results are extrapolated to the ARE fuel system, the radial temperature differences expected therein would be about 10° to 30°F. (AEC Activity 4401.5)

Physical Properties Research - The melting point of the latest ARE fuel mixture, $NaF-ZrF_4-UF_4$ (53.5-40.0-6.5 mole %), is about 570°C. Preliminary density and viscosity measurements were made on the new fuel. The density data are represented by the equation $\rho(gm/cc) = 4.06 - 0.00097\ T(^{\circ}C)$ over the temperature range, 600° to 800°C. The viscosity varied from about 16.2 centipoise at 580°C to 5.8 centipoise at 955°C.

Vapor pressure vs temperature curves were determined for six fluoride salt mixtures. The marked increase of vapor pressure, with increasing ZrF_4 content near 50 mole % ZrF_4, is currently being explored. (AEC Activity 4401.5)

Analytical Studies - Methods are being investigated for the determination of the reactants and products of the reaction

$$CrF_3 + UF_3 \longleftrightarrow Cr^{o} + UF_4.$$

This, or a comparable reaction involving CrF_2, may be a limiting corrosion mechanism in an Inconel-fluoride salt fuel system. The investigation is conducted in a medium of $NaZrF_5$ at temperatures of the order of 800°C. A solution of di-sodium di-hydrogen ethylenediaminetetraacetic acid (EDTA), buffered at pH 4.0 with sodium acetate and acetic acid, was used to leach CrF_3 selectively and separate it from chromium metal or the oxide, Cr_2O_3. The rate of this complexing action with EDTA was slow but nevertheless quantitative after several hours. Divalent chromium fluoride, CrF_2, forms a complex with EDTA rapidly at a pH of 4 and hence cannot be separated from CrF_3 by this method. Uranium tetrafluoride can be leached quantitatively with EDTA. Appreciable complex formation between UF_3 and EDTA has been observed. This reaction can be greatly retarded by increasing the basicity of the EDTA solution. The selective leaching of trivalent chromium and tetravalent uranium by complex formation with EDTA appears feasible.

The isotope dilution technique was applied to samples of the fluoride fuel which were then analyzed in the mass spectrometer. An accuracy of 1% was obtained routinely. This method was applied to the analysis of irradiated fuel specimens, where conventional chemical analysis had indicated wide discrepances. Spectrometer analysis of the uranium content of the fuel before and after irradiation permitted calculation of the burnup and the average value of the flux during irradiation, which was in excellent agreement with the known values of these parameters. (AEC Activity 4401.5)

Shielding Research - The neutron shielding qualities of natural uranium in an hydrogeneous shield has been determined as the result of parallel studies of the effective neutron removal cross section of uranium and neutron multiplication due to fission in uranium. In the latter part of this work the neutron intensity beyond a considerable thickness of water was measured as a function of the position of a uranium slab within the water. It was found, as expected, that, as the uranium was moved closer to the source and therefore into a large neutron flux, the neutron multiplication became more important. In other tests the uranium slab was covered on either or both sides with a sheet of boron-impregnated Plexiglass, which was quite effective in reducing the fission rate. Although nothing unexpected developed from these experiments, the measurements provide quantitative data for shield designs employing uranium.

The construction of the Tower Shielding Facility is proceeding very nearly on schedule. One leg is now 180 ft. high and another, 135 ft. The erection of the third and fourth legs has not yet been started. The reactor pool was completed and tested with water. The concrete hoist foundations were poured and the hoist building started. The control building construction was initiated, the outer walls being nearly complete. (AEC Activity 4401.7)

GENERAL REACTOR RESEARCH

Package Reactor - The design study and cost estimates for a low cost, 10-megawatt package reactor are nearly complete; a report (ORNL-1613) is being prepared. Solid fuel plates are specified and water (H_2O) is used as moderator and for heat transfer to the steam power circuit. The reactor will have a net electrical output of 1000 kw and will produce 3800 kw for steam heating purposes. (AEC Activity 4510)

General Reactor Radiation Damage - The dynamic mechanical properties, i.e. those governing response to sinusoidal stresses of elastomers, are being measured by the vibrating-reed method. The specimen is clamped at one end and placed in forced vibration. The amplitude-frequency relation is observed and the resonance frequency and band width are used to calculate the internal friction and dynamic Young's modulus. Measurements have been made on natural rubber and synthetic styrene-butadiene formulations. The dynamic Young's modulus is very sensitive to radiation exposure and shows a nearly linear increase over the range studied. It therefore provides a better measure of radiation hardening than other properties, such as breaking elongation and durometer hardness, which are restricted in sensitivity and range. The change in internal friction is not simply related to radiation exposure; however, this property is important for certain applications of elastometers such as tire design. The internal friction was little affected for natural rubbers but was increased for styrene-butadiene formulations by short-term irradiations.

An extensometer that has been under development for some time was tested for use on radioactive tensile specimens; it may be attached to the specimen by remote control. The output can be plotted against stress on an X-Y Recorder and the amplification (in terms of inches of chart movement per inch of specimen extension) can be continuously varied from 0 to 6,000. This is about six times the sensitivity of commercial recording extensometers, which generally are not adaptable for remote control operations. (AEC Activity 4540)

School of Reactor Technology - Ten design problems were undertaken by the students of the 1952-53 session of the Oak Ridge School of Reactor Technology, as "group theses" with group leaders selected by the ORSORT faculty. Final results were reported in meetings on August 17 and 18 to the Laboratory. The titles of the design problems were: "Heterogeneous Boiling Reactor Employing a Superheater," "10-Megawatt Aqueous Homogeneous Circulating Solution Reactor for Producing Electrical Power in Remote Locations," "10-Megawatt Homogeneous Boiler Reactor for Producing Electrical Power in Remote Locations," "Circulating Fluoride Reactor for a Submarine," "Gas-Cooled Package Power Reactor Employing Ceramic Fuel Elements," "Fluoride Thermal Breeder Reactor for Power," "Fused Salt Package Reactor," "Airplane Fireball Compressor Jet Reactor," and "Fluidized Solid Internal Breeder Reactor."

Selections for the 1953-54 session of ORSORT were completed, and classes for the new session will begin on September 14, 1953. (AEC Activity 4841)

PROGRAM 5000 - PHYSICAL RESEARCH

ISOTOPE PRODUCTION

Stable Isotope Production - During July and August gadolinium was processed for the first time. An estimated 61.4 grams of gadolium isotopes, along with milligram quantities of several of the isotopes of neighboring elements, were collected. The operating temperature range of the electron-bombardment-heated sources appears now to extend to at least 2800°C. Units of this type were used to continue the separation of ruthenium, and are now being used in the processing of iridium for the first time. An estimated 7.5 grams of ruthenium isotopes and 1.0 gram of iridium isotopes were collected.

Approximately 25 kilograms of 97-99.5% samarium oxide were separated from other rare earths by liquid-liquid extraction. Two kilograms of anhydrous gadolinium chloride were prepared for use as calutron charge material.

Forty-eight shipments and transfers of thirty-five isotopes of twenty-one elements were made. (AEC Activity 5121)

PHYSICS

High Voltage Program - Experimental runs were made to find the variation of ν for U^{235} fission as a function of neutron energy. Fission neutrons, detected by a recoil proportional counter, were counted in coincidence with fissions which took place in an annular counter arranged about the recoil counter. The recoil counter was shielded by a steel shadow cone from the direct D-D neutrons produced with the 2.5 Mev Van de Graaff machine. The counting system was calibrated with thermal neutrons. Preliminary data indicate the ratio of ν at 5 Mev to that at thermal is 1.75 ± 0.30. (AEC Activity 5211)

Short Lived Isomers - The directional angular correlations of three gamma-ray cascades in Ta^{181} were measured with a fast delayed-coincidence scintillation spectrometer, using NaI detectors. These cascades are 132 kev -- 480 kev, 132 kev -- 345 kev, and 345 kev -- 135 kev. An analysis of the results, in combination with the measured K-shell internal conversion coefficients, is not complete, but it appears that an unambiguous assignment of the angular momenta of these three excited states in Ta^{181} and a classification (multipole order and character) of these gamma-rays are possible. The results from the 132 kev -- 480 kev and the 132 kev -- 345 kev gamma-gamma angular correlations indicate that the angular distributions are attenuated appreciably by the interaction of the nuclear electric quadrupole moment of the 480-kev state with the gradient of the electric field from its surroundings. The angular correlation of the 345 kev -- 135 kev gamma-rays is the first correlation observed in which each transition is a mixture of electric and magnetic multipole radiation. (AEC Activity 5211)

Nuclear Physics, 86-Inch Cyclotron - Measurements of angular distributions of fission fragments from 22-Mev-proton-induced thorium fission were undertaken and data are now being accumulated on the angular distribution of the barium, silver, and strontium products of thorium fission.

The cross section for $U^{238}(p,\alpha)Pa^{235}$ was measured; the results are such that the accepted identification of Pa^{235} seems questionable.

Measurements of fission cross sections in medium weight elements were undertaken. The cross section for Na^{24} production from the bombardment of copper with 22-Mev protons was found to be less than 5×10^{-7} barns. The yield of N^{13} from (p,f) on Al is being investigated for correlation with data from the 63-inch cyclotron.

A search was made for the activity of Ca^{38} from (p,2n) on potassium. Its half life is either less than 2 seconds or greater than 3000 years. Searches for Mg^{22} and Ne^{18} from (p,2n) on sodium and on fluorine are underway. (AEC Activity 5211)

The 86-Inch Cyclotron - A target was fabricated which permits the bombardment of small quantities of chemical compounds at currents of 200 microamperes or greater. The present design consists of a thin-walled Inconel capsule centered within a larger thin-walled tube; both the tube and the capsule are cooled by water flowing at high velocity in the annulus between them. The use of thin-walled tubes (0.005") and a small annulus limits the degradation of proton energy to less than 5 Mev over most of the sample. This target was subjected to currents of 250 microamperes (5 kw) with no evidence of damage. The materials bombarded thus far in the target were strontium fluoride, magnesium chloride, magnesium oxide, and lanthanum oxide.

A second successful bombardment of lithium metal for Be^7 production was completed. The run lasted 5.2 hours with an average current of 600 microamperes (13.2 kw). From the 170-mc/mah yield obtained on the first run the estimated production of Be^7 was 535 mc. (AEC Activity 5261)

Fundamental Electromagnetic Research - A complex problem concerning the potential distribution encountered in a simple theoretical model of a controlled thermonuclear reactor was programmed, coded, and solved on the Laboratory's new high speed computing machine, the Oracle. For a given set of assumptions, solutions were obtained for many combinations of central ion density ($10^{11} \leq n_o \leq 10^{16}$) and temperature ($10^7 \leq T \leq 10^9$) in the reactor chamber. For a chamber of approximately 50 cm half-width it was found that the potential is small over most (i.e., >99%) of the region but rises sharply near the wall in a sheath effect to give high wall potentials. For this 50.1 cm chamber half-width, an ion density of $10^{15}/cm^3$, and a temperature of 10^8 °K, the wall potential is 160 kv. (AEC Activity 5261)

Isotope Analysis Methods Laboratory - The design work was completed for a new proton-free nuclear resonance "head" unit for studies of HF in UF_6. Microwave and infrared spectra were obtained on formyl fluoride, HCOF, a slightly asymmetric top molecule; also, infrared spectra were obtained for $LiAlD_4$ and $LiAlH_4$, $LiGaD_4$ and $LiGaH_4$, and solid $(ClCN)_3$. Zeeman spectra of uranium at 25,000 oersteds were obtained up to 9000Å. An echelle grating spectrograph for lithium isotope analysis was modified to permit sensitive spectrum scanning, and a curved entrance slit was constructed to offset image curvature at off-axis points. The isotopes, U^{233} and U^{235}, were excited in a high frequency source; the nuclear spins of both were found to be 5/2. The nuclear moments of U^{235} were revised to $\mu = -0.8$ nuclear magnetons and $Q = +8 \times 10^{-24}$ cm^2; for U^{233} μ and Q are both positive. X-ray diffraction data and calculations of interplanar spacings were completed on eighteen complex uranium compounds. (AEC Activity 5261)

Mass Spectrometer Laboratory - A number of equations were derived for application of the isotope dilution technique to the analysis of ARE fuels, and certain facts are apparent. (1) From the mathematical point of view, the isotopic ratio measurement of U^{236}/U^{235} on an irradiated, undiluted fuel sample would seem to give the best indication of fuel "burn-up." (2) The best method for the determination of total U^{235} loss appears to be obtained by diluting a sample of the irradiated fuel with natural U_3O_8 and measuring the U^{235}/U^{238} ratio. Any discrepancy between (1) and (2) above would give an estimate of U^{235} loss by processes other than fission and capture. (3) It is apparent also that isotopic ratios, measured with the mass spectrometer, can be used to check, or determine, the fission and capture cross sections for the various uranium isotopes. During the investigation 35 samples of ARE fuel were examined in the mass spectrometer.

The vacuum system of mass spectrometer MS-B2 (high temperature unit being added to the General Electric mass spectrometer) was assembled in place and checked to a base pressure of 1×10^{-6} mm of Hg. The electrical interconnection between MS-B2 and MS-B1 was essentially completed. A regulated AC supply for the ion source oven heater was completed and found to perform satisfactorily on dummy load tests. (AEC Activity 5261)

63-Inch Cyclotron Physics - Additional shimming of the magnetic field to define the median plane of the cyclotron resulted in a marked increase in circulating beam; 40 microamperes of N^{+3} ion were measured at the 24.3-inch radius. At the present stage of deflector development a deflected beam of 0.02 μa N^{+3} is obtained. (AEC Activity 5261)

Apotron Development - In an initial test of the operation of an apotron unit with thallium chloride feed material, the performance was much less satisfactory than for lithium feed. With this heavier feed material, instability of the ion beam was encountered at pressures for which stable operation was obtained with lithium. A 700-kw high voltage power supply for the apotron was assembled and found to perform satisfactorily. (AEC Activity 5261)

CHEMISTRY

Inorganic Solution Chemistry - It had been found earlier that uranium (VI) can be adsorbed from nitrate solutions by Dowex-1. In a reinvestigation of the adsorbability of U(VI) from aluminum nitrate solutions containing 0.5M HNO_3, it was found that excellent adsorption (distribution coefficients between 20 and 700) occurred in the aluminum nitrate concentration range of 1.0 to 2.5M. This implies that processing of uranium nitrate—aluminum nitrate solutions by anion exchange is feasible, particularly since these high distribution coefficients were observed when the resins were considerably loaded with uranium.

Following a suggestion of Professor George Scatchard, the ultracentrifuge equation for the distribution of polymers in the presence of supporting electrolytes was simplified sufficiently to permit estimation of the degree of polymerization of charged, low molecular weight, inorganic polymers. Combining results of ultracentrifugation of $ZrOCl_2$ solutions in 1M HCl, 1M LiCl—1M HCl and 1M CsCl—1M HCl solutions, it appears that zirconium exists as a trimer with charge not exceeding +1 per monomer unit. (AEC Activity 5311)

Chemical Physics - The apparent lack of reaction between a beam of molecular potassium at 500°K and a beam of hydrogen bromide at 150°K reported previously leads to a strong interest in observing the scattered products at angles nearer the center-of-mass vector, around 45° from the K beam. Since the present detector extends only about 15° from the K beam, a new detector mount is being built to allow observations up to 90°.

In the interim, measurements of total cross-section were started using a scattering volume instead of a spray of HBr. Measurements of beam intensity diminution were obtained at a number of pressures and temperatures and appear reasonable, but calculation of actual cross section values awaits recalibration of the HBr pressure gauge.

Cooperative microwave experiments with Duke University on the hyperfine structure of the $J = 2 \longrightarrow 3$ rotational transition of CH_3I^{131} were completed. The nuclear spin of the 8d. I^{131} was found to be 7/2 and its quadrupole coupling in CH_3I, -973 ± 9 Mc. This coupling gives -0.412×10^{-24} cm^2 for the quadrupole moment of I^{131}.

Neutron diffraction data on KH_2PO_4 at room temperature and at 140°K are consistent with a random arrangement of hydrogen atoms along hydrogen bonds at a distance of 1.08 ± 0.01 Å from oxygen atoms. The coherent scattering amplitude of phosphorus is 0.54×10^{-12} cm. (AEC Activity 5311)

Chemistry of Solid State - In continuing the study of equilibria between molten metals and their halides, the complete miscibility of potassium fluoride, a typical salt, with potassium, a typical metal, in the liquid state above 940°C was confirmed by further measurements. A solution of 40 mole percent of the

salt in the metal boiled, under an argon gas pressure of 1000 mm mercury, at approximately 35°C higher than the pure metal. The solubility of the solid salt in the liquid metal at 650°C, i.e. far below the monotectic temperature of 748°C where it amounts to 47 mole percent, is still as high as 7 mole percent.

The solubility of sodium metal in molten sodium chloride, bromide, and iodide, at the monotectic temperatures 793°, 740°, and 657°C was found to be 2.0 ± 0.2, 4.4 ± 0.3, and 3.0 ± 0.2 mole percent, respectively. The solubility was 27.0 ± 1, 15.0 ± 0.5, and 21.0 ± 0.5 mole percent, respectively, at 950°C, the highest temperature at which the present method gives reliable results.

The observed melting point depressions of 7° and 4°C for the bromide and iodide are much smaller than those calculated from the heat of fusion of the salts in ideal solutions of monotectic concentration and are, therefore, indicative of appreciable solubility of metal in the solid salt. Tests are underway to confirm this assumption. For the chloride, where experimental and calculated depressions agree (9°C), the solubility of sodium in the solid salt was found to be less than 0.2 mole percent immediately below the monotectic temperature.

On the other side (metal) of the equilibrium diagram the solubility of solid sodium chloride in liquid sodium was found to increase with temperature to 2.5 ± 0.2 mole percent near the monotectic temperature of 793°C. Above this temperature the solubility of molten chloride increases to at least 6 mole percent at 890°C and probably beyond this value at still higher temperatures. (AEC Activity 5311)

Radio-Organic Chemistry - A study of the isotope position isomerization of triphenylethylene-1-C^{14} in the presence of p-toluenesulfonic acid showed the reaction $RH + Ph_2\overset{*}{C}{=}CHPh \rightleftharpoons Ph\overset{*}{C}H{=}CPh_2 + RH$ to be first order with respect to the olefin and second order with respect to the p-toluenesulfonic acid. (AEC Activity 5311)

Radiation Chemistry - The yields, in molecules per 100 ev, of H_2, H_2O_2, H and OH produced in pure water and in 0.8N - H_2SO_4 solutions by irradiation with gamma-rays from a radio-cobalt source were evaluated to be:

	Pure Water	0.8N H_2SO_4
H_2	0.46	0.38
H_2O_2	0.67	0.67
H	2.74	3.7
O	2.32	3.1

These values were deduced from yields of products obtained by irradiating the following solutions: oxygen-free and oxygen-saturated potassium bromide also containing sulfuric acid, oxygen-saturated ferrous sulfate, oxygen-saturated ceric sulfate and pure water containing both dissolved hydrogen and oxygen.

By the technique of paper chromatography, the concentrations of non-volatile acidic products were measured qualitatively as a function of the gamma-ray dose received by various organic acids. Product concentrations usually showed an initial build-up followed by a decrease, with increased dose.

The oxidation of uranium(IV) by gamma-rays was studied as a function of dose for a wide range of concentrations of uranium(IV) and also with addition of either uranium(III), uranyl ion, hydrogen ion, hydrogen, hydrogen peroxide, or oxygen. The decrease in oxidation yield with increased dose cannot be explained by build-up of uranyl ion, hydrogen or hydrogen peroxide but is presumably due to the intermediate formation of uranium(V).

The radon collecting station is operating satisfactorily. A charge of 167 mc was withdrawn on August 18, which represents a minimum efficiency of 84 percent. Only one stroke of the Toepler Pump was employed for this extraction. The radon for this extraction "grew" for four weeks, during which time the pressure above the $RaCl_2$ solution increased only 1.3 cm Hg.

The study of the recombination of H_2 and O_2 in the presence of CO_2 is in progress. Preliminary evidence indicates that CO_2 does not enter into the reaction, nor does its presence affect the rate of recombination. The absence of a visible residue, $(HCHO)_x$, is interpreted as indicating no chemical reaction between the CO_2 and hydrogen. The rate constant derived from the pressure change of H_2 and O_2 only is about 10.7 ± 0.9 and the plot of log $P_{(H_2+O_2)}$ vs $e^{-\lambda t}$ results in a straight line. These criteria indicate that the CO_2 does not exhibit the inert gas effect. (AEC Activity 5311)

Chemistry of Fused Salts - The most significant results obtained by cryoscopic measurements in fused nitrates concern the stability of complex zirconium (IV) fluoride ions in the molten state. It has been demonstrated that $NaZrF_5$ dissociates essentially completely into Na^+ and ZrF_5^- in fused $NaNO_3$, and that the ZrF_5^- is stable at temperatures near 310°C. Likewise Na_2ZrF_6 dissociates completely into $2Na^+$ and $ZrF_6^=$, with $ZrF_6^=$ dissociating perhaps 5 or 10% further into ZrF_5^- and F^-.

Nearly completed studies of electrical conductivity and density of fused mixtures of lithium and potassium chlorides have demonstrated that these salts form essentially ideal liquid solutions, thereby confirming our hope that this system of solutions can constitute a useful reference standard. (AEC Activity 5311)

Uranium Chemistry of Raw Materials - Measurements of the solubility behavior or uranium(IV) orthophosphates in 1.36 to 15.24M total phosphate at 25°C are complete. Through this range, the solubility of uranium(IV) orthophosphate was found to increase with increasing total phosphate concentration from 1×10^{-4}M uranous ion at 1.36M total phosphate to 1.45M uranous ion at 15.24M total phosphate. The stable solid phase from 1.36 to 9.8M total phosphate is $U(HPO_4)_2 \cdot 6H_2O$ and above 9.8M total phosphate the stable solid phase is $U(HPO_4)_2 \cdot H_3PO_4 \cdot H_2O$.

In a continuous countercurrent mixer-settler system, only two extraction stages were necessary to obtain 99.8% extraction of uranium from typical sulfate leach liquors using 0.1M tri(n-octyl) amine in a hydrocarbon diluent. Tri(n-decyl) and dihexyl dodecyl amines have shown comparable high extractions.

The use of a hydroxy-phosphinic acid, 1-hydroxy-2-ethylhexylphenyl-phosphinic acid, for extracting uranium(VI) from solutions containing sulfate and phosphate yielded extraction coefficients higher than those obtained with any other pure organophosphorus compound yet tested. Coefficients of 10 have been obtained from 1.5M phosphate solutions at pH = 1 with 0.025M reagent in kerosene (a mole ratio of reagent to uranium of only 6). From concentrated phosphoric acid (21% P_2O_5) the coefficient is 25 with 0.1M reagent.

Continued burning tests with powdered lignite have shown that 75-80% of the uranium can be extracted from ash burned at 1000-1400°C if the burning time is brief. However, the acid consumption is invariably high due to the volatilization of most of the sulfate as SO_2. Ashes of lignite from the west side of the Slim Buttes region (S. Dakota) require 600 to 800 lbs. of H_2SO_4 per ton but ashes from the east side lignite require 1000 to 1100 lbs. of acid per ton of ash. (AEC Activity 5361)

Electrochemical Studies - As one phase of the work to improve the electrolytic method for preparing heavy water (ORNL-1448 and 1494), experiments were conducted on methods of increasing the ratio of deuterium to hydrogen in the gas stored by a palladium cathode in an electrolytic cell. As the cathode current density was increased, the percentage of deuterium in the electrode decreased, reached a minimum, and then increased to a value comparable to that at the lower density. Under the conditions studied, there would be no advantage in using the higher current densities. Deuterium enrichment factors of 3 to 6 were obtained with palladium cathodes in a bath containing 75% D_2O.

In studies of the properties of palladium electrodes it was shown that the resistance of a 10-mil-diameter palladium wire increased by a factor of 1.2 when hydrogen deposited in the wire, but when the hydrogen was discharged, the resistance of the wire returned to its original value. Palladium needles and coils that had been charged, as cathodes, with hydrogen-deuterium gas and then short-circuited through an ammeter were shown to act as hydrogen-deuterium anodes until 90 percent or more of the stored gas was used. The anodes then reversed polarity and took up hydrogen again.

Studies on nickel electrodes showed that the voltage required for electrolysis is 30 to 40 times less if a timer is used to change the polarity of the current periodically. A 60-cycle alternating current reversed the current too fast to accomplish this.

Work in the field of general electrochemistry included successful electrolytic decontamination of a sintered stainless steel disk that had been used to filter homogeneous reactor fuel. The filter was used as the anode of a cell in which the electrolyte was dilute sulfuric acid.

Small silver electrodes were placed in the sampling chamber of the homogeneous reactor in an attempt to remove trace amounts of chloride ion (about 3.5 grams) from the fuel as it circulated. About 1.5 percent of the chloride was removed in 3 hours (46 milliampere-hours) at approximately 100 percent current efficiency, but the low current density (about 1 milliampere per square inch), required to prevent dissolution of the silver anode, made removal of the rest of the chloride by this method not fast enough for the time allowed for this project.

Two electrochemical methods for removing mercuric nitrate from the "25" process waste, in order to reduce corrosion in the storage tanks, were investigated. In one method a mercury cathode, with the waste solution as the electrolyte, was found to remove 95% of the mercuric ion, but the problems involved in circulating the mercury and in making the apparatus leaktight made the method too complex for production use. In the other, 99% of the mercuric salt was removed by reducing the mercury with aluminum turnings or shot in a packed tower, but column plugging and violent evolution of gas accompanying dissolution of the aluminum made further study of the process unattractive. (AEC Activity 5361)

Volatility Studies - In studies on the chloride trifluoride method of processing Hanford-type irradiated slugs (preliminary studies reported in ORNL-1545), an exploratory experiment indicated that a continuous process for reacting chlorine trifluoride with uranium metal was feasible. The average dissolution rate of the uranium in this experiment was 25 grams/hr-cm^2, which compares favorably with batch dissolution data. (AEC Activity 5361)

Thorex Process Development - The evaporation-reflux procedure (report ORNL-1576) for clarification of the feed to the Thorex solvent extraction process was shown to convert some of the ruthenium in the feed to a less extractable form, thereby decreasing the ruthenium contamination of the product. The addition of reducing agents (sodium hydrosulfite, stannous chloride, and titanous chloride) to the feed solution was also effective in decreasing the amount of ruthenium extracted in the product. The details of laboratory development work are reported in ORNL-1610.

The design of the non-radioactive solution make-up area and the structural design of cells 5, 6, and 7 in the pilot plant were completed. Construction of the thorium nitrate tetrahydrate digester, for use in preparing feed for non-radioactive runs, was also completed. (AEC Activity 5361)

METALLURGY

Ceramics Research - In cooperation with the Fuel Survey, Fuel Technology Sub-Station at Hanford, end cap compacts for insulating uranium slugs are being prepared by hot pressing a 60-40 (wt percent) mixture of aluminum powder and bubble alumina in a "sandwich" between layers of aluminum powder. The finished specimen will consist of cylinders with layers of the 60-40 mixture between layers of aluminum.

Further efforts were made to improve silicon carbide-silicon cermets for reactor fuel use. Attempts to find more suitable volatile components for the basic graphite-volatile component mix continued. The wheat flour mixes did not give consistently good results. The microstructure of specimens made with napthalene were excellent, but large fissures often developed. New techniques for driving off the volatile materials are being investigated.

As a result of discussions held with the PWR Fuel Development Group at WAPD (Westinghouse Atomic Power Division), production of 18 specimen rods of UO_2 of maximum density was commenced. A pilot run using fused UO_2 gave a finished rod having a density of 96.5% of the theoretical value. This rod was prepared by milling the fused UO_2 in a steel-lined ball mill with steel balls and then leaching in HCl to remove the metal impurities picked up in milling. After drying, the powdered UO_2 was pressed into rod form, without a binder. The pressed rod was fired in a hydrogen atmosphere at 1750° for one and a quarter hours. (AEC Activity 5411)

Physical Metallurgy of Reactor Materials - Continuation of experiments to determine the factors affecting the impact strength of thorium has shown that 1/4 inch diameter specimens, having a notch with a root diameter of 0.150 inches, have ductile fractures at temperatures of 60°C and higher when tested in the range of -196°C to 180°C.

Investigation of possible factors influencing the suspected strain-aging phenomenon occurring during creep testing of thorium at 300°C was continued with the testing of high purity "iodide" thorium. It was expected that carbide precipitation might be responsible for the observed continued reduction in creep rate with time of testing. However, the results for one test did not confirm this, since the high purity thorium, tested at 8,000 psi, exhibited the same slowing-down of creep rate as Ames thorium. Completion of additional stress-to-rupture tests on Ames thorium at 300°C showed that pre-straining by loading to 4,000 or 18,000 psi had no appreciable effect on rupture stress, but may have reduced the elongation obtained at rupture.

As part of the continuing study of possible factors affecting the rupture of bonded slugs in the ORNL Graphite Reactor, length measurements have been made on discharged slugs. A maximum growth of 0.183 inches was noted for slugs that were not beta-transformed and which were irradiated for 464 days. Beta-transformed slugs irradiated under equivalent conditions showed a maximum growth of about 0.040 inches. (AEC Activity 5411)

Fundamental Investigation of Radiation Damage in Solids - In an effort to gain information about the "trapping centers" produced in germanium bombarded with fast neutrons, the magnetic susceptibility of n- and p-type material with various impurity concentrations is being studied as a function of temperature, down to 70°K. The Faraday force exerted on 0.5-cm cubes is measured with an electrodynamic microbalance built specifically for this work. Gaseous oxygen is used as a reference standard. Data on eight well-annealed samples prior to irradiation indicate that the diamagnetic susceptibility of germanium is temperature dependent, increasing with decreasing temperature, and has a value

of 1.05×10^{-7} cgs at room temperature. The contribution of the current carrier to total susceptibility was observed in 4 samples, the contribution being diamagnetic. The ratio of the rest mass to the effective mass for "holes" was found to be approximately 4 and, for electrons, 6.3. The values for the degeneracy temperature observed experimentally are in good agreement with the values calculated from degenerate statistics. These samples are now being irradiated in hole 51 of the Oak Ridge Graphite Reactor. (AEC Activity 5411)

Fundamental Physico-Metallurgical Research - Continued investigation of the alloying behavior of zirconium with the 5b subgroup elements has revealed the following information: 1) The crystal structure of γ phase in the Zr-Ag system (composition Zr_2Ag) is body-centered tetragonal with unit cell dimensions a = 3.25Å, c = 3.98Å. The X-ray diffraction data suggest that this phase may form a superlattice of twice these cell dimensions at low temperatures. 2) The solubility of silver in β zirconium increases from approximately 3 atomic percent at the eutectic temperature of 815°C to nearly 16 atomic percent at the eutectic temperature of 1160°C. 3) The peritectoid reaction, by which the δ phase in the Zr-In system is formed, occurs near 1090°C. 4) The solubility of indium in α zirconium at 600°, 700°, and 1090°C is 2.8, 4.8, and 11.8 atomic percent, respectively. 5) The solubility of indium in β zirconium at 1090° and 1250°C is 10.5 and 12-14 atomic percent, respectively. 6) The solubility of antimony in β zirconium at 965°, 1100°, and 1200°C is 0.7, 1.4, and 2.8 atomic percent, respectively.

A β heat treatment of 1 min at 730°C, followed by a water quench, altered the character and greatly diminished the degree of the preferred orientation developed in 0.210" uranium sheet straight-rolled 86% at 585°C and then stretch-straightened 1.4% at room temperature. The texture may be described as one in which the (001) plane is parallel to the rolling direction but tilted 45-70° from the plane of the sheet with the (110) pole in the major component and the (010) pole in the minor component being parallel to the rolling direction. The new orientation developed upon β heat treatment was very complicated, with no really outstanding component, but it was definitely not random. This effect of β heat treatment on sheet therefore is similar to that found for rolled rod. (AEC Activity 5411)

PROGRAM 6000 - BIOLOGY AND MEDICINE

BIOPHYSICS

Waste Disposal Research - In conjunction with the current study of the economics of power reactors, laboratory tests of the effectiveness of Conasauga shale for the removal of aluminum and mixed fission products from chemical processing wastes are in progress. When two liters of a 1.0M aluminum nitrate solution ($Al(NO_3)_3 \cdot 6H_2O$) at pH 2.2 was passed through a column containing 75 grams of the shale (60-mesh material), no removal of the Al^{+++} ions was

indicated by volumetric determinations of aluminum in the influent and effluent of the column. When two liters of a 2.0-molar solution of $Al(NO_3)_3 \cdot 6H_2O$, containing mixed fission products at a level of about 140,000 counts per minute per milliliter, was passed through a similar column the removal of radioactivity ranged from a maximum of 30% to a minimum of 4%. This may be compared with a 95% activity removal obtained when 15 liters of a solution of the same fission products in tap water was passed through such a column. (AEC Activity 6510)

Theoretical Physics - Using Bohr's scattering theory, it was found that about half the energy of a free atom moving in a monoatomic substance of like atoms is effective in producing displacements or vacant lattice sites. This remains true as long as the energy of the free atom is sufficiently low that electronic collisions can be neglected. For energies where electronic collisions are important, the calculations show that the total number of displacements does not substantially change as the energy of the free atom is increased; this is true for all elements of medium or large atomic number. Work on the problem of disordering of solids by heavy corpuscular radiation is continuing. The immediate questions of interest are computation of the displacements for elements of small atomic number, and the application of the results already obtained to compute the number of displacements caused by neutron or other heavy particle irradiation.

An age theory calculation was made of the neutron spectrum at 1500 yards from a fission neutron source in an infinite medium of air. The result is in good agreement with the threshold measurements made by G. S. Hurst at 1500 yards from Shot No. 1 of the spring test series, 1953. The calculation is being repeated, using a modification of the age theory. (AEC Activity 6510)

Physics of Nuclear Radiation - The variation of neutron dose with depth below the surface of a cylindrical phantom (2 meters in diameter, 30 cm in length, and composed of tissue equivalent material) was measured along the axis of the cylinder. When a Po-B source is placed at a distance D from the face of the phantom the dose at any point X inside the phantom is given by $N_0 B \frac{e^{-X/L}}{(D + X)^2}$ where N_0 is the dose incident to the surface (measured without the phantom present), B is the build-up factor due to the presence of the phantom and L is the relaxation length. For Po-B neutrons, L is 7.5 cm. Measurements to determine B have not been completed. Similar measurements in a smaller phantom (base 6" x 6", height 11") using both Po-Be and Po-B sources show the dose at any point X to be given by the same equation as that for the larger phantom but with different values for L and B.

For Po-B source, B = 1.20 L = 10.0 cm

For Po-Be source, B = 1.33 L = 11.3 cm

Extensive studies were made on the ionization produced in various gas mixtures by alpha particles from plutonium. Thus far, the results divide the gases into two groups. In the first group, the electron volts per ion pair (W_m) for the gas mixture is given by the empirical relation,

$$\frac{1}{W_m} = \frac{1}{P_1 + aP_2} \left[\frac{P_1}{W_1} + \frac{aP_2}{W_2} \right]$$

Where P_1 is the pressure of one gas in the mixture and W_1 is its W value while P_2 is the pressure of the other gas whose W value is given by W_2.

If it is assumed that there is no interchange of energy between the two components of the mixture, a would be the ratio of the stopping powers of the two gases, i.e., $\frac{S_2}{S_1}$. In some instances (for example, $N_2 - O_2$ mixtures) the ratio $\frac{S_2}{S_1}$ is very near equal to a; in other cases discrepancies as large as a factor of 2 are found.

In the second group, comprising mixtures of He and other gases, the value of W for mixtures with a low percent of argon, methane, and nitrogen, is approximately 30 electron volts/ion pair, while the W value for very pure He is 46 electron volts/ion pair. If 30 is used as the value of W for He in many mixtures, a good fit to the above formula is obtained by using the appropriate value of a for each mixture. (AEC Activity 6510)

Radioactive Particulate Problem - From the review and analysis of air monitoring data in the X-10 Area during the period from March 1949 to January 1953, the Special Isotope Recovery Process (Ra-La) was identified as a significant source of airborne radio-particulates.

In order to obtain more detailed information, this process was sampled extensively during the last run in the latter part of May 1953. Particulate samplers and fall-out trays were located in the process area and at four outlying stations within a radius of one mile from the process stack. Air samples of the off-gas and cell ventilation were collected with cascade impactors and Millipore filters, located in the system immediately before each duct discharges into the stack. A total of 300 air samples, taken during a seven day period, represented conditions before, during, and after the complete process cycle.

The samples were analyzed by radioautographic and radiochemical methods as well as by decay and absorption studies. Preliminary and incomplete analyses of the data reveal these findings: 1) The major constituent of the off-gas and cell ventilation systems is radioiodine. This activity is so predominant that it obscured the particulate activity in cascade impactor and Millipore filter samples, rendering them useless for further study, 2) the gross activity of samples collected at the outlying stations was too low to permit positive identification but the particles were numerous and sharply defined, 3) the cell ventilation Millipore samples and the particle samples collected at the outlying stations positively identify the Ra-La process to be a source of large numbers of sub-micron radioactive particles, and 4) unusual weather conditions, i.e., no precipitation and light winds, prevailed most of the time during the period of study, and prevented maximum ground concentrations from developing within the one mile zone around the stack. Hence, the outlying samplers could not be expected to collect representative samples for use in correlation with meteorological data.

The efficiency of the ORNL reactor filter house was investigated and found to be satisfactory. A flow-type ionization chamber was used to sample the effluent gases, and Chemical Corps No. 6 filter paper was used to collect the particulate matter. No alpha-particle contamination was found on the filter papers. The beta and gamma activity of the effluent, one minute or less after its release from the filter house, was of the order of 10^{-3} - 10^{-4} μc/cc. The activity had a short half-life and is due almost entirely to A^{41}, with relatively small amounts of Xe and Kr. (AEC Activity 6520)

Consultation and Special Problems - A detailed justification of the fencing for the Tower Shielding Reactor was prepared, based on direct radiation estimates. Also, calculations were made of the airborne hazards associated with two types of catastrophe postulated for this reactor. The first type involved a power excursion sufficient only to melt down the fuel plates with the resultant total release of halogens and rare gas fission products as a thermally "cold" cloud. In the second case, it was assumed that an Al-H_2O reaction resulted in the total release of all fission products in a "hot" cloud of considerable thermal content. From this study it was concluded that the "cold" cloud would exceed a 10 rep dose to the thyroid by a factor of 10 at a distance of 0.4 miles in the daytime and by a factor of 7 at 1.5 miles at night. The "hot" cloud was safer by a factor of 10 than a 10 rep internal dose to lungs and gastro intestinal tract at 10 miles at nighttime and by a factor of 100 at 10 miles during the day. The Reactor Safeguard Committee considered the airborne hazards less important than the "rain-out" contamination which could yield a dose rate of 500 r per hour at the X-10 laboratory area or 40 r per hour at a distance of 10 miles. (AEC Activity 6530)

Professional Health Physics Training - The twenty-one AEC Fellows in Radiological Physics completed their field training in Applied Health Physics August 28. Eight of the Fellows were granted extensions of the fellowship to complete the requirements for the Master's degree. (AEC Activity 6720)

RADIOISOTOPE SALES AND COSTS

Type of Transaction	July	FY to Date
*Domestic Sales	$103,763	$ 103,763
Foreign Sales	1,381	1,381
Project-Cash Sales	5,359	5,359
Project-Transfer Credits	---	---
Technical Cooperation Program Credits	299	299
Plant Credits	4,601	4,601
Civilian Defense Credits	---	---
Radioisotope Income	$115,403	$ 115,403
Other Income - No costs incurred	9,260	9,260
Total Income	$124,663	$ 124,663
Radioisotope Costs	$ 97,405	$ 97,405
*Cancer Program Costs	$ 15,968	$ 15,968
Radioisotope Shipments	949	949

*The Cancer Program costs subsidized by AEC included in "Domestic Sales".

GROSS OPERATING COSTS

	Cost for July	FY 1954 Cost to Date
Programmatic Operating Cost 1/	$2,427,308	$2,427,308
Plant and Equipment Cost	243,325	243,325
Construction Program "H"	1,952	1,952
Work for Other Parties - Transfers	46,704	46,704
Inventory Changes	20,904*	20,904*
Reimbursable Work for Other Parties	151,103	151,103
Deferred Charges	225	225
Total Laboratory Cost	$2,849,713	$2,849,713
Estimates Cost for Next Month	$2,850,000	$5,699,713

1/ Effective July 1, 1953 ORNL Program Costs are reflected on a Net Basis rather than Program Basis.

* Credit

[redacted]

PERSONNEL SUMMARY

	Number of Employees August, 1953	New Hires August	Terminations August
Administration	59	7	5
Operations*	118	1	1
Engineering, Shops and Mechanical	726	0	6
Laboratory and Research	1667	3	64
Protection	173	1	0
Service	402	2	8
Total	3145	14	84

*Includes Electrical Distribution and Steam Plant as well as the Operations Division.

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