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# HANFORD LABORATORIES
# MONTHLY ACTIVITIES REPORT
## OCTOBER 1964

NOVEMBER 16, 1964

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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
OCTOBER 1964

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November 16, 1964

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## Table I - Hanford Laboratories Force Report

Date: October 31, 1964

| | At Beginning Of Month | | At Close Of Month | | |
|---|---|---|---|---|---|
| | Exempt | Salaried | Exempt | Salaried | |
| Chemical Laboratory | 153 | 130 | 158 | 133 | 291 |
| Reactor & Fuels Laboratory | 212 | 196 | 208 | 200 | 408 |
| Physics & Instruments Laboratory | 128 | 81 | 130 | 80 | 210 |
| Biology Laboratory | 43 | 63 | 44 | 63 | 107 |
| Applied Mathematics Operation | 16 | 7 | 16 | 7 | 23 |
| Radiation Protection Operation | 49 | 85 | 49 | 86 | 135 |
| Finance & Administration Operation | 122 | 153 | 127 | 168 | 295 |
| Programming Operation | 5 | 2 | 5 | 2 | 7 |
| Test Reactor & Auxiliaries Operation | 60 | 299 | 62 | 303 | 365 |
| General | 2 | 5 | 2 | 5 | 7 |
| | 790 | 1021 | 801 | 1047 | 1848 |

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## BUDGET AND COST SUMMARY

October operating costs totaled $3,051,000, an increase of $293,000 from the previous month. Fiscal year-to-date costs aggregate $11,046,000 or 31% of the current control budget for FY 1965.

Hanford Laboratories' research and development costs for October compared with the previous month and the current control budget are shown below:

| (Dollars in thousands) | COST Current Month | Previous Month | To Date | Budget | % Spent |
|---|---|---|---|---|---|
| HL Programs | | | | | |
| 02 | $ 52 | $ 67 | $ 227 | $ 542 | 42 |
| 04 | 1 416 | 1 252 | 4 833 | 16 076 | 30 |
| 05 | 152 | 115 | 485 | 1 694 | 29 |
| 06 | 312 | 295 | 1 198 | 3 570 | 34 |
| 07 | -0- | 2 | 7 | -0- | -- |
| 08 | 47 | 37 | 165 | 500 | 33 |
| | 1 979 | 1 768 | 6 915 | 22 382 | 31 |
| Sponsored By | | | | | |
| NRD | 123 | 127 | 544 | 1 594 | 34 |
| IPD | 21 | 16 | 75 | 425 | 18 |
| CPD | 200 | 185 | 746 | 2 207 | 34 |
| | $2 323 | $2 096 | $8 280 | $26 608 | 31% |

## RESEARCH AND DEVELOPMENT

### 1. Reactor and Fuels

Iron-aluminum stabilized uranium irradiated to 2500 Mwd/ton swelled 1.2 vol%, or about half as much as a companion N-Reactor fuel composition.

Lithium aluminate and lithium silicate target elements are being irradiated in KE-Reactor to obtain data on tritium release and gas composition. In addition, more than 40 compounds of these salts with various



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addition agents have been made to evaluate their density, thermal stability, and ease of fabrication. Design of Al-3 wt% Li alloy target elements is proceeding. Work continued on preparation of Bi-9 wt% Li alloy.

No evidence of localized or accelerated corrosion attack was noted in preliminary examinations of N-Reactor graphite cooling system components.

Nickel-plated aluminum-clad fuel elements were discharged from C-1 loop after 30 days of exposure to 260 C water. Where the nickel plate could be distinguished through the thin deposit, it appeared to be intact.

Out-of-reactor ceramic fuel washout experiments with a section of a Zircaloy-2 clad, $UO_2$-1/2 wt% $PuO_2$ swaged PRTR 19-rod fuel element irradiated to 4085 Mwd/ton showed no detectable core material release (as determined by activity readings throughout the loop) through a slit defect 6 in. long by 1/16 in. wide after 14 days of exposure to flowing (13 fps) water at 300 C. Similar results were obtained last month during a 3-1/2-week test with a Vipac element.

The asbestos in a used PRTR nozzle-tube gasket showed surprisingly high chloride content, with the highest concentration (1430 ppm) occurring in the outer region where stress cracking had occurred in the SS spiral wound bands. The source of the chloride is presently unknown.

Extensive plutonium ceramic dissolution tests have shown that OPG solution (oxalic acid, peracetic acid, sodium gluconate) is the most effective of the standard decontaminating agents, but still has an undesirably slow dissolution rate-0.5 ppm after 1 hr at 40 C. More corrosive solutions (such as strong mixed acids) coupled with anodic protection are being evaluated.

A PRTR high power density core fuel element (58-in. length) produced less end bracket fretting damage in a 500-hr out-of-reactor test (at 145 ppm flow) than a standard PRTR fuel element tested under similar conditions (but at 120 gpm flow).

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The Technical Design Criteria for the PRTR High Power Density Core were issued.

The effect of reactor transient operation on fretting corrosion was clearly demonstrated in PRTR where the zirconium content of the primary coolant changed from <0.1 ppb to >5 ppb as the steam generator pressure was varied from 325 to 410 psi. The Zircaloy content returned to "normal" (< 0.1 ppb) shortly after completion of the test.

Calculations indicate that surface boiling should not occur on the PRTR shroud tubes in the 55-tube high power density core, unless the pressure tube is more than 75% eccentric.

Calculations were made to allow prediction of flow and enthalpy in the various flow channels in multirod fuel bundles during two-phase flow conditions. The study is not yet complete, but initial results allowed a prediction of two-phase pressure drop determined with electrically-heated test sections within a few percent.

Examination of PRTR process tubes shows that high flux significantly increases the corrosion rate.

A short core fuel element consisting of $UO_2$ fueled rods assembled into a 19-rod cluster was exposed in the EDEL-1 loop for 3 weeks at 540 F. Examination of the element showed minor wear (~ 0.1 mil) on the lower support pads and 5 to 10 mils in one rod-wrap.

Calculations indicate that 60% of the elements in the PRTR short core loading will operate with molten cores at some time during their irradiation.

A fuel element designed to demonstrate the feasibility of replacing individual rods in a PRTR cluster was charged to the MTR.

Capsules of Zircaloy-2 were exposed in an autoclave at 350 C, 150 psig in a continuing study of Zircaloy-2-$I_2$ interaction. After four days of



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autoclaving no cracking was observed; however, pits 5 to 10 mils deep were obtained in areas of high stress, in the presence of $I_2$.

$UO_2$-20% $PuO_2$ samples prepared from dry ballmilled and pneumatically impacted material continue to show excellent plutonium distribution.

Stainless steel-20 vol% $PuO_2$ cermets showed no changes in structure when heated to 900 C in a vacuum for 400 hr.

Two stainless clad, stainless-30 wt% $UO_2$ cermet pins (0.640 in. OD x 3-1/2 in. long) fabricated by pneumatic impaction were reduced by hot extrusion to 0.200 in. OD x approximately 36 in. long fuel elements.

The pneumatic impaction equipment was completely overhauled for the first time after 2-1/2 yr and approximately 1000 impaction.

The thermionic emission apparatus for measuring work functions of plutonium compounds has been successfully operated for the first time.

Beta plutonium sesquioxide exists at the composition O/Pu = 1.50. An oxygen excess or deficiency could not be produced.

A maximum exposure of $3.4 \times 10^{20}$ fissions/$cm^3$ (12,600 Mwd/ton $UO_2$) was achieved for the 12 EBWR prototypic fuel rod capsules still under irradiation.

Sol-gel thoria released considerably less gas than either low-fired gel thoria or Nupac denitrated thoria.

Full-size thorium-uranium tubular fuel elements operated under power reactor conditions in the ETR have now attained 9300 Mwd/ton exposure with a measured volume increase of 0.7%.

A sample of Haynes alloy 25 was exposed to high velocity helium, 500 fps, at 2050 F for 111 hr to determine if high gas velocities are important to corrosion in the ATR loop. Observed corrosion was in good agreement with data obtained in static oxygen at 18 μ or 42 μ pressure at 2050 F. These preliminary data indicate that gas velocity effects are not important.

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The first of a series of controlled pressure-temperature swelling capsules reached its goal exposure and was discharged. It had operated successfully at 450 C and 1000 psi. The construction of a second controlled pressure-temperature capsule was completed and the capsule was charged into a reactor.

As determined by a dead weight loading stress-to-rupture apparatus, the effect of irradiation [$1.8 \times 10^{20}$ nvt (E>1 Mev) at 50 C (122 F)] on the time to rupture of Inconel X-750 in the standard heat treated condition is as follows: at 48,000 psi stress and 732 C, the control ruptured in 175.2 hr, while the irradiated specimen broke in 9.5 hr.

Zirconium alloy samples exposed to the hydrogenated, low oxygen, coolant have now been weighed and corrosion rates calculated. Corrosion rates were reduced at least a factor of two by the hydrogen addition at all flux levels, but still were greater than out-of-reactor rates. The beneficial effects of hydrogen addition to the coolant were least at the highest flux levels.

Tests on a large bar of N-Reactor graphite show that 1000 ppm $H_2$ reduces the rate of oxidation by water vapor in concentrations of 100 to 1000 ppm.

Good agreement has been obtained on length changes in graphite tests at 600 C in the GETR and in K-Reactor. This indicates that the neutron dosimetry methods used have properly accounted for the differences in neutron spectra and flux intensity.

The thermal conductivity of graphite irradiated at 625 to 800 C to high neutron doses has been measured on N-Reactor and other graphites. A factor of 5 decrease in the room temperature thermal conductivity was found.

Radiation-induced expansion is now occurring in N-Reactor graphite samples irradiated in the ETR at about 1000 C to a dose of 4 to $6 \times 10^{21}$ neutrons/$cm^2$, E>0.18 Mev.

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Examination of electron micrographs made from molybdenum which was irradiated to $10^{20}$ nvt disclosed a nonuniform distribution of defect clusters. This suggests that clusters do coalesce to form dislocation rings.

A 2-hr anneal at 750 C of high purity molybdenum, irradiated to $1 \times 10^{20}$ nvt ($E > 1$ Mev), results in no appreciable change in lattice parameter but a marked decrease in line breadth. Foils containing carbon as a controlled impurity show a decrease in both lattice parameter and line breadth under the same conditions.

It has been inferred from experiments that diffusion of atoms is not the major mechanism in the beta to alpha transformation of pure plutonium.

2. Physics and Instruments

N-Reactor Phase III fuel cycle analysis using uranium metal fuel has been completed. Measurements of $k_\infty$ to support hazards evaluation of coproduct lattices are in progress.

Two more hybrid preamplifiers were completed for use with the regular fission detectors of the N-Reactor subcritical neutron flux monitoring system.

The addition of an $Am^{241}$ detection channel to the $Pu^{239}$ liquid sample counting system is being considered to compensate for $Am^{241}$ contribution to the readout, thereby improving the accuracy of the $Pu^{239}$ measurements.

Development is continuing on an ultrasonic test to detect clad thinning over the brazed regions of N-Reactor fuels (dog-bone test). A prototype probe was constructed using 2 Mc $BaTiO_3$ transducers, and preliminary results were encouraging.

A prototype ultrasonic tester for detecting the bond quality of lithium-aluminum target elements has been returned to the laboratory prior to its installation in the 306 Building for continued "in line" evaluation.

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X-ray fluorescence data indicate the instrument does not detect N-Reactor fuel closure contamination to the required sensitivity. Further laboratory work is planned.

An improved alpha particle detecting system to reveal uranium contamination on finished fuel surfaces has been designed.

Development work has been completed on a prototype dual bond tester to monitor the internal bond quality of AlSi fuels from the outer surface. This system eliminates the troublesome internal probe required by the present tester.

Exponential and neutron multiplication experiments were continued with enriched N-Reactor fuels in light water to provide data for nuclear safety guidance. Measurements have now been completed with two fuel assemblies in each of two lattices.

Subcritical neutron multiplication measurements were performed on arrays of SN and PR cans filled with plutonium nitrate solutions. Although some further experimental checks are needed, preliminary results show that SN and PR cans may be safely handled in much greater numbers than previously permitted.

A series of criticality experiments were completed with plutonium nitrate solutions in a 15.2 in. diam stainless steel sphere (30.2 liter volume). The data obtained from these experiments have provided critical mass values for plutonium solutions in the vessel bare and with reflectors of water and concrete.

Class I HL-designed shipping containers were subjected to drop tests (up to 30 ft) to simulate accident conditions. The containers survived these tests in fair order.

A Monte Carlo calculation has been used to provide an estimate of the critical mass for a bare sphere of $Pu^{238}$ metal. The results indicate the critical mass for $Pu^{238}$ may be as low as 5 kg, which is half the value for $Pu^{239}$ metal of 10 kg.

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Favorable comparisons have been obtained between measured and calculated criticality for $PuO_2$-polystyrene compacts at an H/Pu of 15, from both multigroup diffusion (HFN) and $S_N$ transport (DTF) calculations.

The modifications to the slow-neutron time-of-flight spectrometer were completed. The resolution elements of the spectrometer were measured using a new copper-crystal monochromator.

Another measurement series of 3- to 15-Mev total cross sections was completed. This series was a remeasurement of the separated isotopes which were measured last month. The results for neutron energies of about 9 Mev were lost from both runs due to a malfunction in the data storage analyzer.

Burnup data from the first of the 1% $PuO_2$-$UO_2$ fuel elements irradiated in the PRTR are now being received.

The initial experiment using 19-rod clusters 0.9% $PuO_2$-$UO_2$ fuel in a graphite lattice has been completed in the PCTR.

Extensive measurements of control element worth, reactor noise, and void coefficient measurements (Pu-Al rods in $H_2O$) were made in the PRCF.

A report describing measurements of the departure of the slowing down spectrum in the EBWR lattice from the zero absorption-(1/E) form is in preparation.

Comparisons of an iterated space-energy separable approximation to obtain unit cell parameters has been made with the integral transport simultaneous space-energy calculations for the EBWR lattice. The iterated thermal $\bar{\eta}f$ is shown to be from 0.35 to 1.35% higher than the simultaneous solution values.

Studies of the utilization of $Pu^{239}$ and $U^{233}$ in the 710 propulsion reactor are under way. Initial work covers both the statics and kinetics of the reactor. An immediate result is that the shutdown power density is ~1/3

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that of a uranium-fueled reactor (ratio of delayed neutrons is primary factor), and the shutdown cooling (propellant) requirements would be significantly lower in the plutonium-fueled case.

Reactor physics code development has continued at a relatively high level, including the completion of the ZODIAC code and descriptive documents.

The initial run with the HTLTR mockup has begun. The run is confined mostly to component testing and outgassing. Testing control element and fuel element components continues.

At the request of Washington AEC a study to determine plutonium values under private ownership conditions is under way to be completed early in November 1964. About 60 hr of 7090 computer time are involved.

Monte Carlo calculations in the RBU code have been used to calculate effective cross sections of $Pu^{239}$ as a function of concentration and geometry. Comparisons with other codes and experimental data will be attempted in order to determine the proper corrections of MELEAGER.

Cross section data have been derived from a recent version of the RBU Basic Library to make up three cross section data tapes for various temperature ranges for use in the MELEAGER Economics chain system.

The PROTEUS code (curve fits the output from the MELEAGER burnup code to the proper form for the QUICK economics code) has been revised to accurately treat private ownership of nuclear fuel.

A special data set for the final debugging of the VESTA code has been devised and is being run.

A catalog (designated FORV) of reactor fuel costs and fuel cost curves is being compiled for AEC-Washington.

Improved power supply regulation has eliminated instability of the PRTR underwater gamma scanner. System detector shielding was modified to accommodate various radionuclide calibration sources; fabrication was started on a device to permit closure of the aperture in the lower collimator.



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Two regenerating neutron flux detectors coated with about 90% $U^{234}$ and 10% $U^{235}$ were given preliminary tests (for gamma sensitivity and alpha particle induced current) and installed in KW-Reactor. A $B^{11}$ chamber and a number of metal-sheathed coaxial cables were also installed in a different KW-Reactor test facility. Prior exposure of the boron chambers to a $Co^{60}$ gamma flux of $7 \times 10^4$ R/hr revealed currents of one pA for the chamber and even less for the cables.

A thermostatically controlled bath was designed to provide molten Wood's metal for use in studying eddy current test coil design and the effect of flaw geometry on output test signals.

Analytical solution for thermal surface impedance as a function of materials, thermal wave frequencies, layer thickness and contact conductance has been evaluated for a wide range of possible test conditions.

The analytical and numerical attenuating wave propagation model studies are essentially complete. Shear wave attenuation measurements on stainless steel are complete except for data reduction.

Ultrasonic and eddy current precharacterization tests were conducted on fatigue test samples. Fatigue tests are being delayed while the four-point bending fixture is modified.

Two approaches are being conducted in an effort to produce a semiconductor junction in lithium sulfate monohydrate. The first is to grow the crystal from an aqueous solution and add the proper dopants during the growth. The second is to form deep traps by irradiation of a single crystal.

An analog computer program was prepared for the conversion of wind turbulence data into rectangular coordinate components.

An analog computer program was completed which defines the distribution and time transients of radioisotopes in the human body after ingestion, injection, inhalation, or absorption. Transients were observed after ingestion of single and multiple doses of $I^{131}$; results are graphed on a chart recorder.

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Successful operation was achieved with the new sliding valve mechanism for Biology canine inhalation studies. Thermistors control a linear actuator which, in turn, drives the sliding valve.

The Biology field analyzer and monitor was completed for field and laboratory use in the detection of radionuclides in experimental animals. Initial operation appeared satisfactory.

Experiments were conducted inside a reactor heat exchanger to determine if the animal physiological function telemetry system, being developed for Biology Laboratory, could be used to measure the temperature and/or pulse rate of humans working inside the exchangers. Results were favorable, although higher transmitting powers are required.

Equipment which will operate at 100 kc is being developed to penetrate isotope containment cells with ultrasound. This test will detect cracks within solid cores and reveal whether the core center is solid or molten.

The analytical solution for thermal surface impedance as a function of heat source parameters has been evaluated for a layer of Inconel on $SrTiO_3$. This method appears to reveal thermal properties of the core material while minimizing effects of contact conductance.

An ultrasonic test has been developed to determine the location of transition regions between as-extruded and annealed grain structures along the length of PRTR process tubes.

Analog computations were essentially completed on the routing problem of the Columbia River during a flood. The problem involved fitting of three parameters of a mathematical model to river flow data.

The isotope $Fe^{55}$ has been discovered in a blood sample from an Alaskan caribou.

Using the volume of a urine sample rather than the potassium content as an excretion index gives an average $Cs^{137}$ body burden estimate that agrees with whole body counting measurements made at the same time.



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The $He^3$ neutron spectrometer was successfully tested using the anti-coincidence circuits.

Recent unexplained difficulties in the small radiation calorimeter system have been traced to room temperature fluctuations. A pulsed read-out system for the small calorimeter was designed and built.

An analysis was made of two atmospheric physics experiments in which fluorescein and zinc sulfide tracers were released from a common source. By controlling the concentration of the slurry and the size of generated droplet, it was demonstrated that fluorescein particles can be produced which have aerodynamic properties similar to those of zinc sulfide, the tracer used at Hanford in the past.

3. Chemistry

Vapor deposition of nickel on copper and uranium has been accomplished by a promising new process using a nickel(II) compound of hexafluoroacetylacetone.

The volatility of trace quantities of polonium over bismuth oxide follows the equation

$$\log \pi \text{ (mm Hg)} = \frac{-1.63 \times 10^4}{T} = 15.62$$

where $\pi$ is the vapor pressure divided by the mole fraction of polonium in the oxide and T ranges from 858 to 1000 K.

Polonium distills out of nitric acid solutions containing 1 g/liter diphenylcarbazide (DPC). Acidities above 1 to 2M and/or the presence of bismuth suppress the distillation. Bismuthyl nitrate carries polonium almost quantitatively, offering promise as a process step for achieving concentration of polonium.

Irradiated thoria was dissolved with no difficulty and the resulting feed subjected to a manganese dioxide scavenging step for protactinium recovery. Radiation effects required the presence of a nitrite suppressant

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for proper precipitation of the manganese dioxide precipitate and prevention of its rapid redissolution. With addition of a nitrite suppressant satisfactory results were achieved with feeds substantially "hotter" than expected in the plant. Sucrose in 0.5M $H_2SO_4$ was demonstrated as a suitable reagent for conversion of manganese dioxide to ion exchange feed.

Precipitation as the hydroxide at pH 10 was tested in laboratory experiments and shown to be a promising method of separating magnesium from $Sr^{90}$, a separation need which may arise in future waste processing.

Sucrose denitration of technetium solutions was successfully demonstrated.

Reaction of BAMBP with nitric acid was shown to proceed through nitrous acid attack.

Continuing laboratory studies of thoria dissolution have indicated that aluminum does not catalyze reaction with nitric-hydrofluoric acids, contrary to SRL reports.

Previous work has identified the critical temperatures at which runaway chemical reactions can occur in ion exchange processes. Recent work has identified circumstances under which transient heat generation conditions can produce temperatures well in excess of ambient conditions.

Experiments to compare the effects of irradiation and chemical degradation on Ce(IV) extraction by D2EHPA-TBP-solvents indicated a marked (30-fold) decrease in extraction coefficients by exposure to 27 whr of radiation per liter.

A full-scale plastic-lined slag and crucible dissolver has successfully processed 190 cans of simulated feed material at the rate of two cans per hour.

A control system for use in plutonium electrorefining process studies has been designed, fabricated, tested and delivered to CPD.

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The GE-412 process control computer satisfactorily passed an extensive series of acceptance tests.

Essentially complete precipitation of plutonium dioxide in a salt cycle melt was demonstrated using a two-step process.

The separation of noble metals from uranium in salt cycle systems by electrochemical means has been shown to be unlikely. Preliminary experiments, however, indicate that precipitation is a possible separation step.

Relationships for the ratios of plutonium species in various chloride melts at 500 to 700 C have been determined spectrophotometrically.

A second hot cell phosphate glass waste fixation run was completed. A good quality vitreous product was obtained and little or no radioactivity was evolved to the off-gas or condensate streams.

The operating capacity for the prototype-scale spray calciner-melter (with no draft tube) was established at that equivalent to the waste from a plant processing about 2 tons of uranium per day.

The first operating tests of the Waste Solidification Engineering Prototype were successfully made on the pot calcination process.

An unexpected exothermic reaction occurred in an experiment in which an unirradiated N-Reactor coproduct element was heated in a steel tube; the experiment is to be repeated with a Vycor liner in the steel tube prior to tests with irradiated elements.

Design and procurement activities for the Containment Systems Experiment continued. Erection of the containment vessel in T Plant is scheduled to start in mid-November. An initial run was completed in the Aerosol Development Facility with carrier-free $I^{131}$. A system of equations relevant to reactor simulator blowdown was solved by an analog computer program.

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The "Steady-State Flow in Soils" computer program for analyzing liquid flow through porous media has been revised to form a new program to calculate gas flow through soils, related to the disposal of off-gas to ground.

A statistical analysis of numerical errors in permeability calculation methods reveals that remarkably accurate results can be expected in using the "Stream Computer Program" to analyze ground water flow systems.

Soil dispersion treatments which will aid in the separation and identification of the different mineral series in Hanford soils have been determined. Montmorillonite, mica, and a chlorite-like mineral have been identified in the clay fraction.

The high resolution characteristics of a lithium-drifted germanium detector were demonstrated on a mixture of gamma emitting fission products.

The mechanism of radiation-induced erioglaucine decomposition in aqueous systems was unequivocally established as attack by OH radicals. Irradiation of p-nitrobenzoate ion gave rise to free radicals, as studied by ESR, having half lives of several minutes.

Preliminary analyses of promethium from irradiated power reactor fuels showed $Pm^{146}$ contents of about 5 ppm after exposures of about 15,000 Mwd/Tonne, substantially less than published estimates.

The surface tension of molten neodymium (a promethium stand-in) was experimentally determined as 50 to 70 dynes/cm.

Strontium titanate of various stoichiometries was prepared for pneumatic impaction testing.

Two previously unreported gamma rays, 0.6 and 0.64 Mev, were tentatively identified as part of the americium decay scheme.

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4. Biology

Adult chinook salmon (20 fish) exposed by the contact method for 90 min to a virulent strain of Chondrococcus columnaris died 3 days postexposure. To our knowledge, this is the first laboratory demonstration of acute death of adult salmon from columnaris infection. Previous laboratory work was with juvenile salmon.

Acute myelogenous leukemia was tentatively diagnosed in a 2-1/2 yr-old female miniature pig in the 125 $\mu$Ci $Sr^{90}$/day group. This is the fifth hematopoietic tissue neoplasia observed in pigs ingesting $Sr^{90}$.

Lungs and bronchial lymph nodes from rats exposed twice weekly for 8 weeks to uranium ore dust show a selective retention of thorium in contrast to uranium. The concentration of thorium in these tissues is 2 to 6 times that of uranium although thorium and uranium were in equilibrium in the ore dust.

Initial tests were performed with DDT to determine experimental levels at which to stress flour beetle populations under conditions of both insecticide and radiation. Results indicated: (a) wild types had comparable sensitivities to DDT, (b) sooty mutants were about ten times more sensitive to DDT than were wild types, and (c) older larvae placed on 0.02% DDT rations had a higher survival rate than did younger larvae.

Alfalfa seeded into $Sr^{90}$ plots during late summer of 1963 was harvested six times during the 1964 growing season. Strontium concentrations increased slightly during the season, with no apparent effect of plant maturity upon the concentration. Current values are nearly identical to those found during 1958.

Preliminary calculations of biological half-life of $Cs^{137}$ in Anaktuvuk Pass adult Eskimos were made using whole-body burdens determined by the shadow-shield counter and $Cs^{137}$ intake calculated from dietary information.

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Substantial variation was found in the values derived, but generally indicated a half-time substantially shorter than the 100 to 150 days reported in the literature.

Whole-body counts of 12 Anaktuvuk Pass adult men were obtained by Radiological Physics Operation on October 13, utilizing the 3 in. crystal. $Cs^{137}$ body burdens were found to be about 20% lower than those measured on September 9-10.

TECHNICAL AND OTHER SERVICES

Work on the Gable Mountain Swamp to control contamination that entered the swamp on June 12 was essentially completed. Backfilling of the edges of the swamp buried the contamination in the near shore area and resulted in a rise in water level of about 1 ft. This action reduced radiation levels to a few mR per hour. The concentration of beta emitters in the swamp water had previously declined to about $10^{-5}$ $\mu c/cc$. Food plants for water fowl were either buried or burned. Ducks sampled during the latter part of the month had external contamination of less than 1000 c/m and $Cs^{137}$ concentrations in their flesh of about $10^{-4}$ $\mu c/g$. The bottom of the swamp is still highly contaminated, however, and will likely remain so for a year or two.

The concentration of fallout materials in the air of the Hanford environs was about 0.3 to 0.4 pc $\beta/m^3$ during the early part of the month, but rose to a peak of 4 pc $\beta/m^3$ on October 22-23, following the atomic test carried out by the Chinese on October 16. The level had declined to about 1 pc/$m^3$ by the end of the month. No significant increase in the $I^{131}$ concentration in local air or milk was evident by the end of the month.

During the installation of Project CGI-966 (Safety Circuit Trip Identification System) problems involving the interaction of row and column ground detectors were resolved through the study of a model assembled at the central maintenance shop in 100-H. Project acceptance tests were successfully



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completed at D-Reactor on October 30 following adjustments of the ground detector according to the instruction of "Invariance of Panellit Ground Detection Functions Under a Change of Supply Voltage," a letter from R. L. Richardson to the project engineer, dated July 20, 1964.

R. L. Richardson, Mathematics Operation, was coauthor of "Principal Variables in Vibrational Compaction," a paper presented at the U. S.-Japan International Exchange Meeting on Ceramic Nuclear Fuels.

HW-84388, "Finding Precision Lattice Constants of Hexagonal Crystals by Using a Computer," has been issued as partial documentation of work done on $Pu_2O_3$ crystals.

Installation of the GE 412 process control computer in the Hanford van next to the 321 Building was completed. The computer is now operational, and machine debugging of functional programs is in progress. Work continues on the coding of the general diagnostic (equipment failure and peripheral failure routines) program.

A procedure was developed for determining optimal numbers of process tubes to be measured in the continuing Alternate Tube-Corrosion Monitor Program.

An extensive set of power curves was developed to assist in the design of an acceptable run-to-rupture experiment to compare the HDS and AlSi fuel processes.

The initial mass spectrometer simulation study of the propagation of measurement errors was concluded.

The analysis of the data from the DNA study of mouse livers of radiation chimeras was completed.

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SUPPORTING FUNCTIONS

Plutonium Recycle Test Reactor output for October was 1312 Mwd for an experimental time efficiency of 76% and a plant efficiency of 60%. There were seven operating periods during the month, four of which were terminated manually, two were terminated by scrams, and one continued through month-end. A summary of the fuel irradiation program as of October 31, 1964, follows:

| | Al-Pu | | $UO_2$ | | $PuO_2$-$UO_2$ | | Other | | Program Totals | |
|---|---|---|---|---|---|---|---|---|---|---|
| | No. | Mwd | No. | Mwd | No. | Mwd | No. | Mwd | No. | Mwd |
| In-Core | 0 | | 7 | 1 850.9 | 78 | 14 128.0 | | | 85 | 15 978.9 |
| Maximum | | | | 349.7 | | 376.2 | | | | |
| Average | | | | 264.4 | | 181.1 | | | | |
| In Basin | 7 | 572.5 | 26 | 2 769.0 | 48 | 5 884.3 | | | 81 | 9 225.8 |
| Buried | | | | | | | 1 | 7.3 | 1 | 7.3 |
| Chemical Processing | 68 | 5 465.8 | 35 | 1 965.8 | | | | | 103 | 7 431.6 |
| Program Totals | 75 | 6 038.3 | 68 | 6 585.7 | 126 | 20 012.3 | 1 | 7.3 | 270 | 32 643.6 |

(Note: Mwd/Element x 20 ≃ Mwd/$ton_U$ for $UO_2$ and $PuO_2$-$UO_2$)

Estimated heavy water loss and indicated helium loss for the month were 975 lb and 131,596 scf, respectively.

Corrections to the Rupture Loop in the Fuel Element Rupture Testing Facility during the last outage have made a significant improvement in vibration level. Following loading of an intentionally defected element the loop was placed back into service. Some control problems continue at the higher temperature which are attributed to the low fuel power generation and the oversized trim in the heater bypass valve. New trim has been ordered for this valve.

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Total productive time in Technical Shops Operation for the period was 30,767 hr. Distribution of time was as follows:

| | Manhours | % of Total |
|---|---|---|
| N-Reactor Department | 2 933 | 9.5 |
| Irradiation Processing Department | 6 137 | 20.0 |
| Chemical Processing Department | 458 | 1.5 |
| Hanford Laboratories | 21 239 | 69.0 |

Nine craftsmen (carpenters, painters, and sheetmetal workers) were transferred from IPD to Technical Shops.

Total productive time in Laboratory Maintenance Operation was 18,100 hr of 20,200 potentially available. Of the total productive time, 95% was expended in support of Hanford Laboratories components, with the remaining 5% directed toward providing service for other HAPO organizations. Manpower utilization (in hours) for October was as follows:

| | | |
|---|---|---|
| A. Shop Work | | 2000 |
| B. Maintenance | | 6300 |
| 1. Preventive Maintenance | 1800 | |
| 2. Emergency or Unscheduled Maintenance | 1400 | |
| 3. Normal Scheduled Maintenance | 3100 | |
| C. R&D Assistance | | 9800 |

The heavy water inventory at the end of October 1964 showed a loss of 857 lb valued at $11,837 for the PRTR. Heavy water scrap generated during the month amounted to 1359 lb, resulting in a $2783 charge to operating costs. Total scrap on hand at October 31, 1964 amounted to 15,398 lb valued at $190,294.

Cumulative data of Hanford visitations:

| | Number of Visitors | |
|---|---|---|
| | In September | Since 6-13-62 |
| Visitors Center | 1569 | 81,107 |
| Plant Tours | 227 | -- |

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HAPO professional recruiting activity for October:

| | Plant Visits | Offers Extended | Offers Accepted | Offers Rejected | Offers Open |
|---|---|---|---|---|---|
| Ph.D | 5 | 5 | 3 | 3 | 3 |
| BS/MS (Direct Placement) | 0 | 1 | 2 | 0 | 0 |
| BS/MS (Program) | 0 | 0 | 0 | 1 | 0 |

Five Technical Graduates were placed on permanent assignment. One new member was added to the roll. Current program numbers 35. The Hanford Nuclear Program will be divided effective November 1. Of the 28 who have not yet selected off-program placement, 18 will remain with the Laboratories, and 10 will be assigned to the new HAPO program operated by N-Reactor.

Authorized funds for 13 active projects total $10,729,000. Total estimated cost of these projects is $11,880,000. Expenditures through September 30, 1964 were $5,066,000.

Manager, Hanford Laboratories

HM Parker:JEB:dh

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# REACTOR AND FUELS LABORATORY MONTHLY REPORT

## OCTOBER 1964

## TECHNICAL ACTIVITIES

## A. FISSIONABLE MATERIALS - 02 PROGRAM

### 1. Metallic Fuel Development

Comparative Swelling of Uranium Fuels. Two KSE-5 fuel elements irradiated in KER Loop 1 to 2500 Mwd/ton for the purpose of comparing swelling performance of two fuel compositions under identical irradiation conditions are being examined in the Radiometallurgy facility. As with a previous low exposure test that has been examined, one element contained uranium with a nominal "N-fuel" composition (140 ppm Fe and 120 ppm Si), and the other element contained uranium with 400 ppm Fe and 800 ppm Al. Swelling in the N-fuel uranium was 2.2 vol% compared to 1.2 vol% for the Fe-Al bearing fuel. Optical metallography reveals a slight amount of grain boundary tearing in the cooler zones (swirl microstructure) and clustered spherical porosity in the hotter zone (heavily twinned microstructure) of the N-fuel uranium. The fuel containing the Fe-Al additions reveals no obvious porosity of either of these two types. Some cracking in the center of the fuel was observed.

Post-Irradiation Examination of Co-product Driver Element. Radiometallurgical examination has been completed on a 1.25% enriched co-product driver element that was irradiated to 1850 Mwd/ton in KER-3. The element experienced a 3.3 vol% increase in the fuel as determined by element bulk density measurements as a consequence of the irradiation. The fuel was of standard N-composition, i.e., 150 ppm Fe and 100 ppm Si, and operated with a maximum fuel temperature of approximately 550 C.

Metallography of the fuel indicates two distinct types of porosity. Grain boundary tearing is evident in the fuel near the cladding, with more porosity on the OD than on the ID. The microstructure in these regions is swirled or hashed. Crystallographically oriented porosity is evident in the central region of the fuel. The microstructure in this higher temperature region is heavily twinned. Fuel swelling of 3.1 vol% has been determined from the comparison of pre- and post-irradiation uranium density as compared to 3.3 vol% determined from the comparison of pre- and post-irradiation fuel element bulk densities.

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Examination of both end closures, which were of the full-brazed, tapered or chevron design, revealed a good performance with no evidence of cladding deformation or damage in the vicinity of the closure. Some radial cracking in the fuel between the clad wall and the tapered side of the cap was observed, but the crack did not propagate beyond the uranium in any case.

Alternate Uranium Compositions. Studies are in progress to evaluate uranium base alloys in terms of composition, fuel fabrication history, corrosion behavior, and irradiation performance. Seventeen alloys planned for irradiation in NaK capsules have been extruded as 0.6 diameter Zr-2 clad rods. The alloys are now being heat treated as follows to provide specimens with second phase dispersions in various distributions and sizes:

Uranium with additions of up to 800 ppm Fe, Si, Al, P, or C,

1. beta quench
2. beta quench, alpha anneal
3. gamma quench.

Uranium with additions of approximately 2 wt% Zr, Mo, or Nb and up to 800 ppm Al or Si,

1. gamma quench
2. gamma quench, alpha anneal.

The machining of the irradiation specimens and capsule parts has been started and is scheduled for completion in November.

Target Element Development. As part of the continuing program of support for the N-Reactor multiproduct target element development, several studies are being conducted on the target materials and target element components.

To obtain data on the gas composition and tritium release during irradiation of ceramic core targets, two elements of $LiAlO_2$ and two of $Li_2SiO_3$ are now being irradiated in a KER Loop. During fabrication, careful control was exerted to assure that the target cores were free of moisture and extraneous gases. These elements will be discharged at the next reactor shutdown.

An irradiation test using Al-3 wt% Li alloy target elements inside tubular uranium driver fuel elements is planned. Fabrication of the metal target elements is scheduled to start early in November.

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Design of the target elements calls for a 0.35-inch gap between the end cap of the inner aluminum can and the end cap of the outer Zircaloy-2 can. There is concern that this free standing length of 0.736-inch OD x 0.030-inch thick Zircaloy-2 tubing may collapse or deform under the 1500 psi coolant pressure and lead to failure. To study the problems associated with this target design, a series of test samples simulating the target element configurations with free standing Zircaloy-2 tube lengths from 0 to 1 inch are being fabricated for thermal cycling in a steam autoclave at prototype pressure and temperature conditions.

Ceramic Target Cores. A program is under way to develop a ceramic core for the N-Reactor target elements capable of withstanding temperature excursions up to 1200 C for 2 hours. Two basic systems are under consideration, namely lithium aluminate and lithium metasilicate. More than 40 compounds of these salts have been made using additions of lithium carbonate, lithium hydroxide, sodium hydroxide, aluminum, lithium orthophosphate, lithium sulfate and graphite. The as-pressed density of the pellets averaged 68% of theoretical in almost every case. Higher pressures resulted in cracked pellets. In every case it was necessary to lubricate the punch and die with lithium stearate to allow ejecting an unbroken pellet. The pellets were sintered at 500, 750, 1000 C, and then heated to 1200 C for 5 hours as a proof of thermal stability. Evaluation of the sintered pellets is under way. Two methods will be used, namely schleroscope hardness and compression strength. Preliminary results indicate that 5% Al + $Li_2SiO_3$ and 5% $Li_2CO_3$ + $Li_2SiO_3$ are the easiest to fabricate and sinter to a hard pellet. The aluminate powders are extremely abrasive and require extensive die cleaning after each compaction.

Target Elements. Efforts to melt and solidify the Bi-9 Li alloy directly in low carbon steel tubes of final core size were twice unsuccessful because the steel wall melted through, allowing escape of the contents. From the sudden rapid heating of the charge, it became apparent that the formation of $Li_3Bi$ is a strongly exothermic reaction. In the third attempt the charge was segmented, pellets of Li being alternated with layers of Bi powder in the closed steel cartridge. This was cautiously heated in vacuo in a quartz tube to about 200 C by induction. Suddenly (over about 2 sec span) the temperature rose to greater than 1000 C, and the entire cartridge became yellow hot. Condensation of sublimed iron from the hot cartridge on the cooler quartz soon obscured the view of the pyrometer so that actual attained temperatures are not known. However, the cartridge remained intact and the specimen is to be used to study the effects of high pressure steam on the core material.



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Similar specimens are being prepared, from which the steel jackets are to be mechanically stripped and the sized core inserted into aluminum liners; the latter sealed and enclosed in Zircaloy jackets. These are to be used in tests for high temperature compatibility, irradiation behavior, and tritium extractability.

N-Fuel End Closure Development. Successful demonstration of clad sizing devices for truing the free standing cladding (after chemical milling) on N-outer fuel tubes resulted in a request to adapt this tooling to the driver tube geometry. Success of the "TIG Braze" closure may depend on this tooling. Joint fit up is extremely critical in this closure, and lack of cladding concentricity becomes magnified in the heavier wall fuel tubes. The necessary redesign is done, and fabrication of the driver tooling will be completed October 29.

N-Outer Fuel Supports. It will be necessary to form at least the first group of 1000 of our committed 6000 N-driver tube supports using the 2.5 KVA heat source. The pieces can be formed using a double pulsing of the heat source which at present introduces some variability in forming temperatures with resulting variability in parts. The 50 KVA unit excessed from UCLRL will provide adequate heat and control to minimize this problem.

Ceramic Target Outgassing. Samples of irradiated lithium aluminate and lithium silicate were outgassed. Relatively large moisture and $CO_2$ contents were observed in both compounds. The $CO_2$ apparently was generated by decomposition of unreacted $LiCO_3$ left from the preparation of the lithium aluminate and silicate. The aluminate was shown to be hygroscopic; up to 800 C, the $CO_2$ evolved was ~5000 ppm; the moisture evolved was ~1100 ppm. Helium-4 evolution was 5.7% of theoretical and tritium ($T_2$ + $\frac{HT}{2}$) evolution was 2.2% of theoretical. Moisture content of the silicate was 300 ppm; $CO_2$ evolution was ~1700 ppm; helium-4 evolution was ~2% of theoretical, and tritium evolution was ~0.25% of theoretical during outgassing to 370 C.

Samples of aluminate and silicate fired to 2000 F (1093 C) are currently being irradiated in the KER facility and will be discharged in November 1964.

2. Corrosion and Water Quality Studies

Hydriding of Zircaloy-2 at Low Temperature. Zircaloy-2 samples representing a variety of pretreatments were exposed to wet $H_2$ at 100 and 200 C in an attempt to reproduce reported low temperature

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hydride failures. It has been proposed that because of the slow rate of film formation some hydriding might occur before an inhibiting film could grow. Samples with thick films (applied at 450 C) did absorb a small amount of hydrogen, probably due to a memory effect, i.e., the films contained an oxygen vacancy concentration typical of 450 C which was permeable to $H_2$. Thin film samples, three-day autoclaved samples, and etched samples did not hydride. Hydriding in all samples appears to have stopped after two weeks of exposure.

Irradiation of Nickel-Plated Aluminum Fuel. Nickel-plated aluminum clad fuel elements were discharged from C-1 in-reactor loop after 30 days of exposure in 260 C deionized water. A thermocouple element operated at 282 C throughout the exposure despite deposition of a visible crud layer. Where the nickel plate could be distinguished through the thin deposit, it appeared to be intact.

Crevice Corrosion in Lithiated Water at pH 10. Two additional tests have been completed in the program to evaluate crevice corrosion at heated surfaces in lithiated and ammoniated water under conditions typical of N-Reactor. The test specimens have crevices formed on the surface of Zircaloy-clad, internally heated test assemblies by welding N-Reactor suitcase-handle fuel element supports to the cladding. Both tests were in TF-3 with lithiated water at pH 10, 293 C and 1350 psi; the first at 2000,000 Btu/ft$^2$-hr for 135 days, and the second at 550,000 Btu/ft$^2$-hr for 45 days. Preliminary examination of the crevice areas revealed relatively severe, localized corrosion in about one-third of the heated crevices. The thickest oxide film was estimated to be about 0.003 inch and was associated with the test assembly cladding. The corroded area near the crevice tip was almost equally divided between the cladding and the surface of the suitcase-handle supports. In similar tests using ammoniated water the thickest oxide film observed was 0.001 inch.

Water Quality Tests for N-Reactor. In-reactor tests are being conducted in KER-1 to evaluate possible effects of nonstandard water quality on corrosion and crudding in N-Reactor. Four separate conditions are being evaluated: (1) low pH, (2) high oxygen, (3) combined low pH and high oxygen, and (4) combined low pH, high oxygen and thermocycling. The low pH test and the high oxygen test have been completed.

A gradual buildup of crud on the fuel elements and an increase in the loop activity level occurred during low pH ($\sim$7) operation. The pre-test conditions of crud and activity were restored when the pH



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was returned to 10. No significant change in operating conditions occurred during high oxygen operation until the loop was shut down, at which time a severe crud burst occurred. The crud released at this time was brown and magnetic. The maximum crud concentration observed was 128 ppm (about 400 times the concentration under standard conditions). The water samples were opaque indicating a large amount of suspended crud. The activity as read through the sample bottle was 500 mr/hr as compared to a normal value of <10 mr/hr. Significantly, there was no indication of gross crud deposition on the fuel elements during this period.

Corrosion Evaluation of N-Reactor Graphite Cooling System. The tube sheet area of an N-Reactor graphite cooling system heat exchanger was inspected to determine if the high oxygen level induced by radiolytic decomposition of the cooling water is causing excessive corrosion. The stainless steel had a dull golden color with no evidence of accelerated attack. The carbon steel had a thin, tight black oxide with some mottled orange spotted areas. There was no evidence of accelerated attack. The stainless steel to carbon steel weld junction was in excellent condition. Additional tests are in progress to investigate possible localized attack in other areas of this system.

Anodic Inhibition of Corrosion by Acids (Decontaminants). Laboratory tests have shown that corrosion of carbon steel in oxalic acid, one of the candidate decontaminating agents, could be decreased by a factor of two or more by anodic protection. A pilot plant test was completed to demonstrate the feasibility of using this method to protect tanks used for storing this acid. A carbon steel tank (4.5 gallons) was filled with 0.3 M $H_2C_2O_4$ and protected by an impressed current of 0.08-0.10 ma/in$^2$. No difficulty was encountered in maintaining the passivity of the surface during the three-week test. The light uniform protective film was maintained, and no corrosion areas were apparent.

Corrosion Inhibition of Aluminum in the Single Pass Reactors. The second of two half-reactor tests to evaluate effects of pH on corrosion rates of aluminum was recently completed. In one test, one-half of the reactor was at 7.0 and the other half at 6.6; in a second test, one-half of the reactor was at 6.6 and the other half at 6.2. In both tests the corrosion coupons and the fuel elements showed the inhibitory effect of lower pH. No evidence of increased localized attack was observed. An over-all comparison indicated that reducing pH from 7.0 to 6.2 would decrease aluminum corrosion rates by a factor of 4 to 9. These large-scale reactor tests corroborate results from laboratory and pilot plant tests completed prior to initiation of the reactor tests.



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3. Gas-Atmosphere Studies

Graphite-Zirconium Compatibility Loop Experiment. Five additional tests have been completed on the new oxidation bar, described last month. The new bar has the same 20 interior cylindrical samples as in the first three tests and, in addition, a large number of surface samples. During the tests the temperature of the bar was 770 C one inch from the front, 820 C at the middle, and about 810 C one inch from the rear. The results show that 1000 ppm $H_2$ with or without CO is an effective oxidation inhibitor for water vapor in the range 100 to 1000 ppm. A comparison of interior samples with surface samples at the same longitudinal position, showed that in most cases the interior samples oxidized at about 1/10 the rate of the surface samples. In the last test, however, the ratio of surface sample rate to interior sample rate was only about two. Some factors that may be effecting this result are: (1) the higher water-vapor pressure (4000 ppm); (2) the bar has been partially oxidized which might enhance greater penetration of the oxidant; and (3) the ratio $P_{H_2O}/P_{H_2}$ was higher in the last test.

4. Thermal Hydraulic Studies

Analysis of Transient Heat Transfer Experiments for N-Reactor. Previously, laboratory experiments were conducted to determine the thermal and hydraulic conditions in a column of N-Reactor fuel elements following a gross rupture of an inlet riser supply line. These experiments were performed with a full-scale, electrically heated model of the downstream half of an N-Reactor fuel column with prototypic hydraulic connectors and fittings. The break leading to a water loss was simulated by quickly opening one end of the test section to atmosphere while isolating the recirculating loop by means of quick-acting valves. Water from a pressurized tank was available to the test section during the transient and the electrical heat generation was reduced in a manner to simulate N-Reactor scram curves.

During the past month, efforts were continued on developing an analytical model to describe such an event. It is felt that if the model can successfully account for all the conditions which occurred in the laboratory experiments, the model would be of great value in predicting conditions for the reactor which are beyond the scope of the experiments.

To date, it has been found that the temperature transient can be reproduced fairly accurately during the first few seconds by assuming a sudden shift from normal cooling to adiabatic heating



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of the fuel, with a redistribution of stored heat to provide uniform radial fuel temperatures. However, following this initial phase, the rate of temperature rise measured in the electrically heated fuel column model used in the experiments was less than that calculated by assuming no heat transfer out of the model.

Two sets of calculations were made in an effort to establish a valid prediction of heat loss from the test section after the first few seconds. The first was the calculation of heat transfer coefficients for steam or water cooling at various thermocouple locations. The calculated coefficients showed considerable variation, ranging from 15 to 40 Btu/(hr)(ft$^2$)($^o$F) along the upper part of the model and from 250 to 1500 Btu/(hr)(ft$^2$)($^o$F) along the lower part. Uncertainties in the specific heats of the model materials at high temperatures and in fluid temperatures inside the model probably account for some of the variations. It is suspected that some water remained in the bottom of the process tubes and that boiling of this water produced the relatively high coefficients at the lower part of the model.

The second group of calculations was used to determine rates of temperature rise assuming that heat was radiated from the fuel model to the process tube, and that the temperatures of the model and tube increased at the same rate. The rate of temperature rise calculated by this procedure was less than that measured by the thermocouples.

It must be concluded that none of the procedures examined thus far provide an accurate method for calculating the temperature transients after the first few seconds.

Hydraulic Tests for the Present Production Reactors. Vibration of inlet connectors for C-Reactor was studied briefly in laboratory experiments in the 189-D Hydraulics Laboratory. The connectors under study were 5/8-inch OD Inconel connected to a P-7537 C-Reactor nozzle on a uniform-wall ribbed aluminum tube with a CIIIE fuel loading. The operating flows for these tubes range from 53 to 55 gpm. The results of the investigation showed that:

(1) the vibration is definitely a characteristic of the nozzle and 5/8-inch OD connector combination over the flow rates considered (50-65 gpm);



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(2) the amplitude of the connector vibration on the laboratory mockup was about the same as that occurring on the reactor, that is, 1/16-inch peak-to-peak at 55 gpm;

(3) a major vibration at a frequency of about 50 cps was present with low-amplitude vibrations being superimposed at higher frequencies;

(4) at an operating flow rate of 55 ± 1.0 gpm, a slight noise in the nozzle assembly was evident, indicating that incipient cavitational conditions may be present in the front nozzle assembly during reactor operation.

5. Shielding Studies

N-Reactor Shield Evaluation. Final calculations are being run with MAC for rigorous comparison to experimental data. MAC-RAD is also being used on the N-Reactor problem to compare the codes.

Additional data reduction is in progress to improve experimental validity. The sulfur (n,p) cross section used in data reduction for fast neutron flux analysis by sulfur activation is spectrally weighted above 1 Mev. The MAC calculated neutron spectral, above 1 Mev, is now used to weight the $S^{32}$ cross section instead of the previously assumed fission neutron spectrum. A rough draft of the report on the N-Reactor top shield experiment is partially completed.

6. Graphite Studies

N-Reactor Graphite Irradiations. The series of long-term irradiations of N-Reactor graphite continues satisfactorily. The last of the fourth-generation capsules, H-6-4, has successfully completed the third cycle of irradiation, and the first of the fifth-generation capsules, H-4-5, has successfully completed one cycle.

Some difficulty has been experienced over the last two reactor cycles in maintaining sample temperatures in the H-6-4 capsule without exceeding former sample temperatures. This has been caused by an increase in the power level of the GETR from 30.0 Mw to 32.5 Mw, with a corresponding increase in gamma heating. Consequently, it has been necessary to rearrange fuel loadings, reposition adjacent fuel elements, and at times to adjust the power level to less than 32.5 Mw, to maintain the proper temperatures. These techniques have proved satisfactory to date, and sample temperatures duplicate those of previous capsules.



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A report, HW-81250 A, "The H-4, -5, -6, Irradiation Program Irradiation of N-Reactor Graphite--Interim Report No. 1," has been completed.

Hanford Irradiations of N-Reactor Graphites. The second of a series of length determinations was completed on N-Reactor reflector and core graphites irradiated at 550 to 600 C. The results given below summarize measurements made on four transverse and four longitudinal samples cut from each of four different bars of each type of graphite.

RADIATION-INDUCED LENGTH CHANGES OF N-REACTOR GRAPHITES

| Graphite Type | Orientation | Length Change, % After $7.8 \times 10^{20}$* | After $15.6 \times 10^{20}$* |
|---|---|---|---|
| GLC Reflector | ⊥ | -0.044 ± 0.009 | -0.180 ± 0.016 |
| | ∥ | -0.099 ± 0.007 | -0.334 ± 0.017 |
| AGOT-LS Reflector | ⊥ | -0.038 ± 0.007 | -0.146 ± 0.01- |
| | ∥ | -0.109 ± 0.009 | -0.304 ± 0.008 |
| | | After $6.0 \times 10^{20}$* | After $12.8 \times 10^{20}$* |
| TSX Core | ⊥ | -0.023 ± 0.010 | -0.122 ± 0.011 |
| | ∥ | -0.068 ± 0.008 | -0.231 ± 0.014 |

*Neutrons/cm$^2$, E > 0.18 Mev

Exposures in these irradiations are now high enough for direct comparison with the first GETR irradiations of TSX at 600 C. Exceptionally good agreement was found between the irradiations of small quarter-round samples in the GETR and the above results on 0.426-diameter cylinders irradiated at Hanford. This agreement indicates that the neutron dosimetry methods used properly account for differences in neutron spectra in water-moderated test reactors and graphite-moderated reactors.

B. WEAPONS - 03 PROGRAM

Research and development in the field of plutonium metallurgy continued in support of the Hanford 234-5 Building operations and weapons development programs of the University of California Lawrence Radiation Laboratory (Project Whitney). Details of these activities are reported separately via distribution lists appropriate to weapons development work.

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## C. REACTOR DEVELOPMENT - 04 PROGRAM

### 1. Plutonium Recycle Program

#### Fuels Development

PRTR Fuel Fabrication and Development. A cladding failure was discovered in the bottom end cap region of a swage-compacted $UO_2$-0.5 wt% $PuO_2$ PRTR fuel element (FE-5185) during routine underwater examination following discharge of the reactor. This element, with a burnup of 4870 Mwd/ton, was fabricated during the period when difficulties were being experienced with fluoride and moisture impurities in the fuel and several rods contained suspect material. The presence of this defective element in the PRTR core did not cause any operational problems. The defect will be examined to determine if it is similar to previous failures caused by these contaminants.

A short core fuel element consisting of $UO_2$ fueled rods assembled into a 19-rod cluster was exposed in the EDEL-1 loop for three weeks at 540 F. Operating conditions were maintained as close as possible to those sustained in the PRTR. Examination of the element showed minor wear (~0.1 mil) in the lower support pads and 5-10 mils in one rod-wrap. Apparently, this particular rod-wrap wire was loose and projected out far enough to contact the process tube. Examination of the process tube indicated no wear in the region of the support pads and 2-4 mils wear in the area adjacent to the fretted rod-wrap.

Maximum temperature conditions have been calculated for short (58-inch core length) $UO_2$-2 wt% $PuO_2$ elements operating in the proposed PRTR high power density core loading consisting of 55 fueled tubes. It is estimated that at a reactor power of 70 Mw 60% of the elements in the short core loading would operate with molten cores at some time (possibly very short) during their irradiation.

A fuel element designed to demonstrate the feasibility of replacing individual rods in a PRTR cluster was charged in the MTR. Each rod of the four-rod cluster has end caps designed to allow the removal and subsequent replacement of individual rods from the clusters. The test element will be returned to Hanford for examination.

Means for enrichment of uranium dioxide fuel elements with plutonium wires are being developed. The capability of the 280-ton



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extrusion press for extrusion of long lengths of plutonium wire is being investigated. Tooling was fabricated and tested with aluminum on the 125-ton unhooded extrusion press. Long lengths of aluminum wire were easily extruded. However, the small diameter extrusion ram buckled before the delta plutonium alloy extruded in the 280-ton press. Control of the maximum applied pressure to avoid overloading is difficult on the 280-ton press with the present system of valves and gages because the available pressure is enormous compared to the required pressure. Since short lengths of plutonium wire were extruded before buckling occurred, a higher strength tool steel was obtained for making new rams which, together with higher extrusion temperatures, may cause the plutonium to extrude within the present system of pressure control.

$UO_2$ fuel element 1037 was salvaged for further irradiation by banding it underwater in the PRTR basin. The old outer wire wrap had been broken, making banding necessary if irradiation of this long exposure (over 6000 Mwd/ton) $UO_2$ fuel element was to continue. During the banding process the element was completely banded once and then rebanded again to assure a tight bundle.

Cladding Evaluation. Capsules of Zr-2 were exposed in an autoclave at 350 C, 150 psig in a continuing study of Zr-2 - $I_2$ interaction. The capsules were charged with $I_2$ and stressed to 21,400 psi (55% YS) and 62,200 psi (>100% YS) using He pressurization or the thermal expansion difference of shrink-fitted stainless steel cores. Control capsules without $I_2$ were also exposed. After four days of autoclaving, no cracking was observed; however, in those capsules containing $I_2$, pits 5-10 mils deep were observed in areas of high stress. A light iridescent film too thin to identify was found in the $I_2$-charged capsules.

Underwater Equipment Development. Underwater spot welds made on Zircaloy-2 banding material, 3/16 inch wide by 0.010 inch thick, were shear tested to determine the comparative strength between samples which had undergone 64 hours of autoclaving and samples that were not autoclaved. Each sample had one spot weld holding two lapped pieces of material. There was a 2.3% reduction in strength after autoclaving. This strength loss should have little effect on the ability of the band to do its job over the required period of fuel element exposure.

Bands with two welds 1-1/8 inches apart were also tested. These specimens failed at slightly less than twice the force of the specimens with only one weld. The actual fuel element band would have at least four to six welds.



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## Corrosion and Water Quality Studies

Corrosion Evaluation of Irradiated PRTR Zircaloy-2 Process Tubing. Metallographic measurement of oxide thickness on PRTR Zircaloy-2 process tube (6079) substantiates previous results that the net in-reactor weight gains are higher than would normally be expected for Zircaloy exposed out-of-reactor. Process tube 6079 was exposed to a temperature of 500 F for approximately 370 days. Oxide thickness of two sections of the tube exposed to flux intensities of $3 \times 10^{12}$ nv (>1 Mev) and $3 \times 10^{13}$ nv (>1 Mev), respectively, were equivalent to in-reactor weight gains of approximately 40 $mg/dm^2$. Portions of the process tube exposed to the lesser flux intensities were not available for examination. But the oxide thickness of a lightly irradiated sample ($<1 \times 10^5$ nv) from process tube 6084 was less by a factor of five, even though process tube 6084 had been at a temperature of 500 F for a longer time than process tube 6079.

Thermal Cycling of Zircaloy-2. A Zircaloy-2 coupon has been corroded at 425 C (25 mm $H_2O$) for 80 days while suspended from a continuously recording balance. At least two and perhaps three multiple rate transitions were observed during isothermal operation. Thermal cycling to room temperature and back to 425 C induced a rate transition indicating that thermal shock can affect corrosion kinetics. These results confirm that mechanical breakdown at the oxide plays a role in the rate transitions of Zircaloy-2.

PRTR Pressure Tubes. Crack propagation in PRTR Zircaloy-2 pressure tubes is indicated to be independent of microstructure and neutron exposure. Seven specimens with annealed microstructure and two specimens with 45% cold worked microstructure have been tested. Three of the annealed pieces had received $2 \times 10^{20}$ nvt fast neutrons, and one of the cold worked pieces had received $5 \times 10^{20}$ nvt fast neutrons. The annealed specimens were prepared for testing by introducing cracks of 0.38 inch, 0.41 inch, 1.5 inches, and 2.0 inches length in the tube wall and oriented with the tube longitudinal axis. The method for doing this has been described previously.

The irradiated cold worked specimen was prepared with a 0.9 inch long crack and the unirradiated cold worked specimen was prepared with a 0.38 inch long crack. The product of the value of the hoop stress at which each of these cracks propagated and the crack length taken to the power 0.61 (not 1.477 as was reported last month) is indicated to be:

$$\text{Hoop Stress (crack length)}^{0.61} = 45{,}400.$$



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Varying the pressurization rate from 135 psig/min to 18,000 psig/min (1 to 133) produced no significant difference in maximum or bursting pressure of specimens of annealed-unirradiated PRTR Zircaloy-2 pressure tubes. The tests were performed at room temperature. The maximum pressure values ranged from 8000 psig to 8400 psig, and the bursting pressure values ranged from 7600 psig to 8200 psig. These data indicate that the maximum or breaking load capacity of this material is not sensitive to rate of application of pressure loads.

Rupture Behavior of PRTR Fuel Elements. The mixed oxide cores of irradiated PRTR fuel elements are proving to be very resistant to attack by high temperature water in ex-reactor tests. The second in a series of tests to determine washout behavior (release of core material) was conducted on a 12-inch length of a swaged PRTR rod, Zr-2 clad, with a core of $UO_2$-1/2% $PuO_2$, irradiated to 4084 Mwd/ton. A defect, 1/16" x 6", was milled in the jacket, and the element was exposed to 300 C flowing water. After 14 days of exposure, no detectable amount of core material was released into the water as determined by surveying the activity of loop components. In the first test a Vipac element, irradiated to 5000 Mwd/ton, was defected with a slit 1-1/2" x 1/16". No activity was released to the coolant after 3-1/2 weeks of testing, except for a minor amount thought to be caused by the defect machining operation.

PRTR Crevice Corrosion Tests. Out-of-reactor tests are being conducted in TF-3 to determine if accelerated corrosion will occur in crevice areas on the PRTR high power density core fuel elements. Crevices are found on PRTR fuel elements where the spiral wound wire wraps and cluster-bands contact the cladding. The potential for accelerated corrosion arises from possible concentration of LiOH (used for pH control) from boiling in the crevices.

Test assemblies, consisting of a cartridge-type heating element, spirally wound with two wraps and having a "cluster-band" at the mid-point of the heated length, are being used in the tests. The tests are being conducted in lithiated water at pH 10; temperature and pressure are such to promote surface boiling. Surface heat fluxes of the test assemblies are 550,000, 300,000 and 200,000 Btu/(hr)($ft^2$).

Stress Cracking of PRTR Gaskets. Several nozzle-tube gaskets (spiral wound SS-asbestos type) removed from PRTR pressure tubes showed stress cracking, especially along the outer ring (No. 1



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ring). The asbestos from one gasket was analyzed and found to be surprisingly high in chloride with the highest concentration (1430 ppm) occurring in the outer ring where the stress cracking had occurred. Concentration of $Na^+$ was high (710-1170 ppm), but concentrations of $Li^+$ were low (<0.05 ppm). Ring No. 2 had 680 ppm and Ring No. 3 had 280 ppm $Cl^-$. The inner ring was not analyzed. A leaking gasket from an ex-reactor loop was examined for $Cl^-$ and showed similar high concentration (1415 ppm) and concentration gradient (154-1415 ppm). The effect of the analytical procedure on $Cl^-$ release from the asbestos complex is being evaluated. It has been noted that the asbestos from used gaskets dissolves more readily than from new gaskets, suggesting that the PRTR service conditions tend to deteriorate the asbestos. Analyses on one new gasket gave the following values: Ring No. 1 (outer ring) 560 ppm $Cl^-$; Ring No. 2, 430 ppm $Cl^-$. The other rings are being analyzed. Both the high concentration (1430 ppm) and the concentration gradient (280-1430 ppm) are difficult to explain unless one assumes concentration by gasket leakage.

Decontamination Studies - Dissolution of Mixed Oxide Fuels. The sintered plutonium oxides have been very resistant to attack by the standard decontaminating agents. Of all tried so far, the best results have been with the OPG solution (oxalic acid, peracetic acid, sodium gluconate). Five hundred mg of the mixed oxides were treated in 50 ml of solution for one hour at 40 C. At the end of this time, only 0.025 mg Pu dissolved. This corresponds to a concentration of only 0.5 ppm and represents an undesirably slow dissolution rate.

Reagents which are capable of dissolving sintered plutonium oxides at a satisfactory rate are extremely corrosive to some reactor materials. It may be possible, however, to use these reagents for limited areas (such as deadlegs) or special cases if suitable inhibition can be provided. One reagent which had proved successful was 9 M HBr on refluxing for several hours. More dilute and inhibited solutions of HBr are being tried. In one test, 1 M HBr was ineffective.

PRTR Boron Shim Studies - Procedures for Analysis of Boric Acid. Work is continuing to develop procedures for measuring and controlling boric acid concentrations in the PRTR moderator system (Boron Shim). Several tests were made to evaluate analytical procedures (spectrophotometric methods) which may be used with the Technicon for analyses and control. The procedure used at other sites utilizes the reaction of carminic acid with boron to form a colored complex which may be measured quantitatively in low



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concentrations (~50 ppb of boron). However, the carminic acid must be dissolved in concentrated $H_2SO_4$, and this solution is so corrosive that a minor instrumental malfunction could cause extensive damage to the entire instrument. Some experiments were made to evaluate an aqueous solution of quinalizarin as an indicator, but no reliable procedure could be developed; at present, it appears that carminic acid is the best available reagent. Some modification of the ($H_2SO_4$) solvent may make the procedure more suitable.

Reactor Engineering Studies

Second Generation Mechanical Shim Rod. A test of the coolant flow versus temperature rise in the driving head was completed using the environmental rod test facility. With a differential pressure of 2 psi (minimum expected on-reactor), the flow was 8 gallons per hour and the coolant temperature rise was 35 F, which are satisfactory for PRTR application. Test conditions included an inside shell heater temperature of 450 F and an ambient air at the head of 350 F.

Design, fabrication, and testing of a special shim rod gasket has been completed. This gasket remains on the shim during in-reactor installation and removal. The second generation shim rod is being modified to incorporate this improvement. The control box, synchro readout equipment, and driving head have been delivered to PRTR for circuit continuity checking prior to in-reactor testing.

Fretting Corrosion Investigation - Ex-Reactor Tests. The fretting corrosion test in EDEL-1 with the high power density fuel element was terminated after 537 hours at 530 F, 1050 psi, and 145 gpm. The fuel element was removed, and the fuel element and pressure tube were then inspected. One relatively large burnished area was noted in the pressure tube which was caused by one lower support pad. The deepest part of this area was estimated to be 0.25 mil deep. The fuel element was inspected, and, on two of the four bundle bands, several of the spot welds on the band lap-over had failed. This condition was not noted following the relatively severe external excitation tests reported last month. This partial failure is attributed to the combined effects of the tests. From the tube response characteristics, the tube damage measurements and the comparison with the fretting corrosion tests with the regular 88-inch fuel element, the following qualitative statements are drawn:



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(1) The short (Mark-I) fuel element indicates no greater fretting tendency than does the regular fuel element under similar operating conditions and at lesser flow.

(2) The mode of vibration of the pressure tube has changed slightly with the short fuel element installed. This change may be significant when considering the interaction or coupling between the fuel element and pressure tube.

| | |
|---|---|
| Short Element | |
| inlet jumper | 1.3-2 mils double amplitude @25-35 cps |
| tube middle | 0.3 mils double amplitude @25-35 cps |
| Regular Element | |
| inlet jumper | 1.5-2 mils double amplitude @25 cps |
| tube middle | 0.1-0.13 mils double amplitude @45 cps. |

Fretting Corrosion Investigation - PRTR Tests. A PRTR test designed to yield correlations between steam generator pressure, tube vibration, pump (or piping) vibration, and zirconium content of the primary coolant has been conducted. Preliminary data are shown in Figure 1. From preliminary analyses of these data, considering only average maximum amplitudes, several tentative conclusions can be made:

(1) Tube vibration level tends to decrease with decreasing steam generator pressure. This trend confirms data from a June 1964 test where the same effect was noted. The source of excitation (steam generator, pumps, etc.) is not immediately evident. It is interesting to note that tube vibration levels with the reactor hot and pressurized, but with insignificant steam generation, are about one-tenth of the operating levels.

(2) Steam generator pressures of about 365-385 psi seem to cause increased inlet jumper-tube vibration; pressures of less than 350 psi (and 395 psi for the pump suction header) cause increased pump header vibration.

(3) Apparently all reactor channels do not exhibit similar fretting characteristics. Tube 1653 shows high zirconium content in the coolant at 365 psi, whereas the bulk coolant zirconium content was at a minimum. In general, where the primary system vibration level was high, the bulk coolant zirconium content was high.



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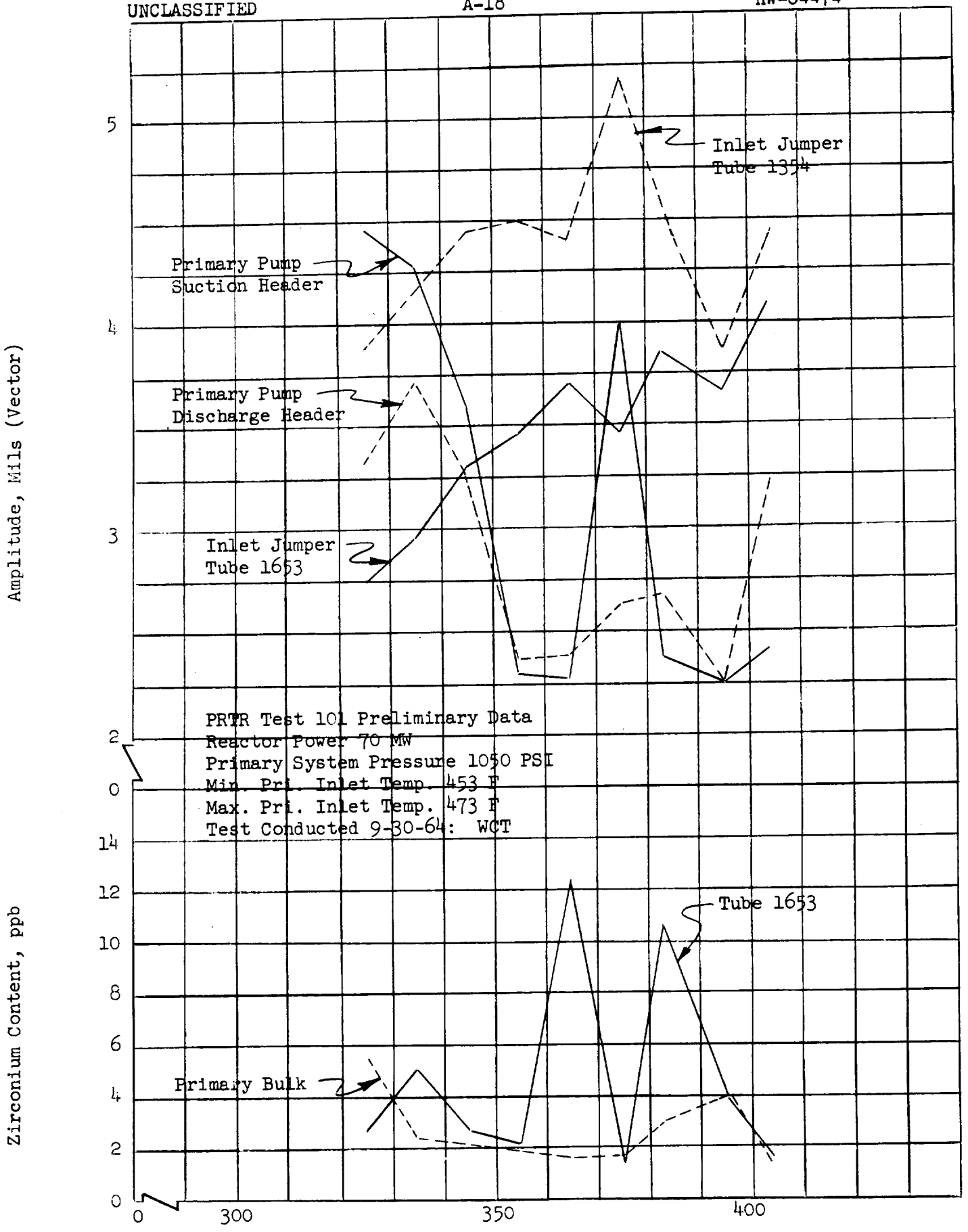


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(4) Transient operation increases the zirconium content of the coolant by a factor of 20 or more. A few hours before and after the test the zirconium content of the coolant was less than 0.1 ppb. During the test sequence the steam generator pressure was increased in 10 psi steps in about 10 minutes and held at pressure for 20 minutes. The coolant samples and vibration measurements were taken in a 5-minute period following the pressure-hold period.

Gasket and Closure Tests. Of two sets of PRTR gaskets tested in the PRTR Gasket Test Station, all three cap gaskets have leaked, whereas the three nozzle-to-tube and inlet gaskets did not leak. All were spiral wound asbestos-stainless steel type except for one special nozzle-to-tube gasket. In this latter gasket the silicone rubber portion of the seal failed, but the soft iron portion maintained a seal.

PRTR-HPD Core. A document, "Technical Design Criteria - PRTR High Power Density Core," HW-83932, 10/1/64, was issued. The criteria set forth the general guidelines for design of the high power density core which strives to meet the fuel development and physics program objectives for the next few years. Some of the specific items are subject to revision, depending on the results of further experiment or analysis.

A draft document outlining the transition from the present PRTR core loading to the proposed short core has been written. Included are the bases for the transition, three proposed schedules and a summary of supporting research and development work. The three schedules illustrate, respectively, a 15-month transition period terminating in a batch core loading, a 10-month transition to a batch loading, and a transition to a graded fuel cycle reaching equilibrium in about 23 months. A comment draft, HW-84449 RD, is being distributed.

To refine the previous preliminary estimates, a revised analysis of the fuel temperature, using two models and power distributions and three values of $UO_2$ thermal conductivity, is being made on the high specific power core for the PRTR. It is expected that revised fuel temperature predictions will be higher than the previous ones.

Fuel Re-use. The second draft of the progress report on fuel re-use (HW-81282) is being completed. Graphical presentation of the economic results was reorganized for clarity. The Value Analysis subsection is being reworked in an attempt to obtain a better correlation and understanding of the effect (or lack of effect) of use charge.



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Thermal Hydraulics Studies. An analytical method was developed to allow rapid calculation of the surface temperature of the shroud tubes surrounding the PRTR pressure tubes. A knowledge of these temperatures is desired to determine conditions of stress, corrosion, and local boiling of the moderator surrounding the shroud tubes.

Using the method, it was found that subcooled boiling of the moderator will not occur under the proposed operating conditions for a short core (70 Mw power with a reactor core of 55 tubes and fuel elements 58 inches long, coolant temperature at 505 F, and moderator temperature at 150 F), provided the pressure tube remained nearly concentric within the shroud tube. If the pressure tube were eccentric such that the helium gap were reduced to approximately 25% of the nominal gap, the shroud tube temperature will reach the boiling point of the moderator. Should eccentricity cause the pressure tube to physically contact the shroud tube, local subcooled boiling will occur, but the shroud temperature will not rise appreciably above the moderator saturation temperature because of the large improvement of the heat transfer coefficient once boiling begins.

Also, an analysis was performed to predict the shroud tube surface temperature during a scram, when the heavy water moderator in the reactor calandria is displaced by helium gas. Results were presented graphically which relate the shroud tube temperature as a function of the helium atmosphere temperature within the calandria. A document, HW-84445, covering the PRTR shroud tube temperature was prepared and issued.

Calculations were also made to determine the heat transfer capabilities of the PRTR when operating as an organic-cooled, heavy water moderated reactor. It was found that for a 750 F bulk outlet coolant temperature, the PRTR could operate at 40 Mw with a full length, 85-tube core or 20 Mw with a 58-inch long, 55-tube core. These powers of the reactor are established by the fuel cladding temperature which was assumed to be limited to 850 F.

2. Plutonium Ceramics Fuel Research

An irradiation capsule for wafer specimens which will allow closely controlled sample temperature has been designed. The method of temperature control consists of directing the thermal energy from the sample through well defined insulated paths of thermal resistors into large heat sinks. Interface heat transfer



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temperature gradients are minimized by thin malleable metal foils between the sample and the thermal resistors.

The thermionic emission apparatus for measuring work functions of plutonium compounds has been successfully operated for the first time. The work function of the tantalum heater is being measured as a standard check prior to measurements on other samples. Present measurements are being made in a vacuum of about $5 \times 10^{-9}$ Torr.

Studies are continuing on characterization of potential fast reactor fuel material obtained by ballmilling and impacting $UO_2$-20 wt% $PuO_2$ powders. Preliminary data show that after four hours of ballmilling, the $UO_2$-$PuO_2$ powder mixture has an average particle size of ~3 microns with 95% of the particles less than 7 microns and 99% of the particles less than 11 microns.

In an effort to determine whether beta-$Pu_2O_3$ exists over a composition interval, samples of this compound containing 10% $PuH_x$ and 10% $PuO_2$ were reacted at 1450 C for four hours in high vacuum. The lattice parameters of the hexagonal beta-$Pu_2O_3$ for all samples were essentially constant, indicating very little or no oxygen deficiency or excess exists.

Stainless steel-20 vol% $PuO_2$ cermets showed no changes in structure when heated to 900 C in a vacuum for 400 hours. Metallographic examination of specimens held at test temperature for 64, 154, and 409 hours showed no evidence of reaction between the $PuO_2$ and the 304 stainless steel. No suboxides of plutonium were noted from x-ray diffraction scans of the specimens.

3. Ceramic (Uranium) Fuel Research

Gas release data were obtained on the last three thoria elements irradiated in the PRTR. An element containing Hanford processed Sol-gel thoria released 0.003 $cm^3/gm$ after $2.2 \times 10^{20}$ nvt exposure, an element containing low-fired gel material released approximately 0.6 $cm^3/gm$ after $2.2 \times 10^{20}$ nvt exposure, and an element containing Nupac denitrated material released 0.16 $cm^3/gm$ after $1.1 \times 10^{20}$ nvt exposure.

The surface roughness test element which has been rescheduled for irradiation in the ETR P-7 loop is being remade. It has been found necessary to reduce the U-235 enrichment of the $UO_2$ to 1.5%. If the original element with 4% enrichment had been irradiated in the P-7 position, the higher neutron flux (as opposed to the flux



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in the MTR) would have caused fuel rod rupture by thermal expansion rather than by burnout.

A technique for eliminating gross dimensional changes in pneumatically impacted stainless steel-$UO_2$ cermet fuel pins was demonstrated. Cermet cores and powdered stainless steel end caps are pre-compacted into pellets of similar bulk density by cold pressing. The pellets are then loaded into thin wall stainless steel tubing and pneumatically impacted to bond and seal the pins.

The effects of particle size, shape, and density are being evaluated with regard to Nupac process variables, such as particle fracturing from cold pressing and pneumatic impaction, cold pressing, and metal matrix and particle density after pneumatic impaction.

Nine stainless steel-$UO_2$ cermet fuel pins were fabricated to evaluate the following variables: effect of binders, particle density, metal matrix density, cold pressing technique, and ceramic distribution throughout the matrix. Post-pneumatic impaction evaluation of these effects is in progress.

Two stainless clad, stainless-30 wt% $UO_2$ cermet pins (0.640" OD x 3-1/2" long) fabricated by pneumatic impaction were reduced by hot extrusion to 0.200" OD x approximately 36" long fuel elements. This method of fabrication is to be evaluated as a means for fabricating fast reactor fuel elements using fuel pins with lengths approaching three feet.

Dimensional variation along the length of the as-extruded pin is ± 0.004 inch except the rear-most three inches where tailing of the extrusion caused a diameter reduction of 0.015 inch below nominal. Plans for further processing and evaluation of this material include: (1) a light (0.001") swage pass for polishing and straightening purposes, (2) cross-section and metallography of the pin length for evaluation of clad thickness variation and processing effects on the $UO_2$ fuel particles.

The impaction compacts were extruded inside a 2-3/8" diameter mild steel billet can at 1200 C through a 0.750" die. The outer mild steel can was then removed chemically.

The Model 1220-B Dynapak machine was completely overhauled during the month. This was the first time the machine was dismantled in



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the 2-1/2 years since it arrived at Hanford. It had made approximately 1000 impaction shots during that time. The machine was out of operation for 10 days while it was being worked on. Sixty shots have been made since operation was resumed.

The extent to which packed-particle fuels support their cladding against external pressure collapse was investigated. One-foot long rods of $UO_2$, vibrationally compacted and clad in 6061-T4 aluminum were hydrostatically pressed at room temperature. No observable distortion was produced in the cladding of any rod, at pressures to 3150 psi. Empty tubes with identical welded closures were pressed simultaneously, with the results summarized in the following table.

EFFECT OF PRESSURE ON UNSUPPORTED ALUMINUM CLADDING

| O.D. | Cladding Thickness | Collapse Pressure |
|---|---|---|
| 0.520 | 0.030 | 660 |
| 0.303 | 0.026 | 1375 |
| 0.340 | 0.045 | 3150 |
| 0.371 | 0.060 | ~4000 (est'd) |

4. Basic Swelling Studies

Irradiation Program

The first controlled pressure-temperature swelling capsule, containing high purity uranium, U + Fe-Al and U + Fe-Si specimens, reached goal exposure and was discharged. It operated successfully at 450 C and 1000 psi irrespective of the operating conditions of the reactor.

A second controlled pressure-temperature capsule was assembled, bench-tested, charged into a reactor, and is operating successfully at goal conditions of temperature and pressure, namely 575 C and 1000 psi. It, too, contains high purity uranium, U + Fe-Al and U + Fe-Si specimens, and will provide insight into the effect of pressure on "growth" induced, crystallographically aligned tearing.

Post-irradiation Examination

Metallographic examination of the high purity uranium specimens irradiated to 0.05 at% B.U., in a capsule controlled at 700 C is in progress. The specimens were in the form of solid cylinders 1/8 inch,



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1/4 inch, and 1/2 inch in diameter and split tubes 1/2 inch in diameter with 0.030 inch, 0.060 inch, and 0.120 inch thick walls. Severe temperature gradients in this capsule caused portions of the end specimens to operate in the alpha phase and the bulk of the specimens in the center of the capsule to operate in the beta phase. A small region of the 1/2 inch diameter rod may also have operated in the gamma phase. Those regions which operated in the alpha phase have retained their original, small, equiaxed grain structure while those regions which operated in the beta phase now consist of large, irregular alpha grains. Superimposed on the large alpha grain structure is a network of porosity which apparently outlines a prior set of grain boundaries. These boundaries are probably the beta grain boundaries that existed during the irradiation. Optical metallography is continuing, and electron metallography will be initiated shortly.

Three additional general swelling capsules were opened in Radiometallurgy and the specimens were recovered. Each capsule contained four high purity uranium specimens, two U + Fe-Al specimens, and two U + Fe-Si specimens. Several different heat treatments were given the specimens prior to irradiation. Capsule No. 22 was irradiated at 525 C until 0.037 at% B.U. was attained, and then the temperature was dropped to 200 C for an additional 0.05 at% B.U. Capsule No. 23 was irradiated at 625 C to 0.11 at% B.U. in the specimens. Capsule No. 27 was irradiated at 500 C to 0.11 at% B.U. Macrophotography and density have been completed on these specimens, but metallographic examination will be delayed for at least a month. The general behavior of these specimens corroborates the previous observations and conclusions in that (1) specimens from both alloys behave better than high purity uranium, (2) the U + Fe-Al alloy specimens are superior to the U + Fe-Si alloy specimens, (3) gamma annealing of the alloy specimens followed by either a beta quench or an alpha anneal results in still further improvement in specimen behavior.

## 5. Irradiation Damage to Reactor Metals

### Alloy Selection

Several nickel base alloys are being studied to determine the effects of irradiation and environment upon their mechanical properties. Ninety specimens irradiated at 280 C to exposures varying from $5.4 \times 10^{19}$ to $8 \times 10^{20}$ nvt (E>1 Mev) were weighed, measured, and tested for hardness. Eight of these specimens were also tested in tension. Hardness values for the precipitation hardenable alloys, Inconel 625, Inconel 718, and Hastelloy R-235,



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showed no change as compared to the control specimens. Hardness values for the solution hardenable alloys, Incoloy 800, Hastelloy N, and Hastelloy X-280, showed a slight increase in hardness with increasing exposure.

One specimen of Inconel 600 has been examined by electron microscopy to determine changes in the microstructure due to irradiation. This specimen was irradiated at 740 C to an exposure of $9 \times 10^{19}$ nvt (E>1 Mev). As previously reported, substantial differences in microstructure were found in the irradiated specimen when compared to the nonirradiated control. No pores were found in the specimen which might have been formed by helium being formed from the presence of boron in the material. The specimen was subsequently annealed at 1050 C for 24 hours and re-examined for pores which may have formed through coalescence of helium atoms due to the heat treatment. Again, no such pores were found. Similar investigations will continue to determine the cause of mechanical property changes in nickel-base and refractory metal alloys.

Chromium alloys have properties which may make them suitable for applications in future high temperature reactors. Quantities of a Cr alloy (Cr-0.7% Y-1.4% Zr-0.7% Ti-0.25% C) were received and are presently being evaluated to determine mechanical properties and heat treating characteristics. Specimens of this alloy will be irradiated and the effects of environment and irradiation on its mechanical properties determined.

In-Reactor Measurements of Mechanical Properties

A capsule containing an annealed 304 stainless steel specimen has been charged and will be tested at 650 C and 20,000 psi stress. An ex-reactor test at these conditions has been running about 200 hours and shows a creep rate of about $5 \times 10^{-4}$ in/in/hr.

During the past month additional activation energy tests of annealed 304 SS during irradiation have been completed at 550, 575, 601, and 621 C, 30,000 psi stress; and 655, 680, and 704 C, 20,000 psi stress. The tests completed to date indicate a stress and temperature dependency of the activation energy. The high value of the activation energy is around 140,000 cal/mole and decreases with decreasing temperature and stress to around 100,000 cal/mole. Indications, from an Arrhenius plot of in-reactor creep rates, are that activation energies may decrease to as low as 60,000 cal/mole with decreasing temperature and stress. In the absence of any irradiation, the activation energy for annealed 304 SS appears to be about 100,000 cal/mole.



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An irradiated in-reactor test capsule has been dishcarged into a cask and is awaiting shipment to Radiometallurgy. General inspection of the capsule components and metallography studies of the annealed 304 stainless steel specimen is planned.

Capsule Development

A high temperature creep machine has been developed with a capability of remote operation. The apparatus can measure specimen elongation to $10^{-4}$ inches. The stress is applied to the specimen through a bellows-type loading system. A Lincoln welder is used as the power supply for the 1/8 inch diameter molybdenum rod wound heater.

To date six tests have been run with the following results:

| Material | Temp. | Stress | Minimum Creep Rate | % Elong. | Time to Rupture |
|---|---|---|---|---|---|
| Haynes 25 | 2100 F | 4000 psi | $1.9 \times 10^{-1}$/hr | 15.9 | 38 min |
| " | " | " | $1.9 \times 10^{-1}$/hr | 12.3 | 33 " |
| " | " | 3000 psi | $6.7 \times 10^{-2}$/hr | 10.0 | 78 " |
| " | " | " | $7.2 \times 10^{-2}$/hr | 9.5 | 63 " |
| " | " | 2000 psi | $<2.6 \times 10^{-3}$/hr | >1.3 | (Thermocouple failure, stopped test after 3 hrs) |
| Hastelloy X | " | 4000 psi | $7.7 \times 10^{-1}$/hr | 13.8 | -- |

The Haynes 25 data agree well with the results obtained earlier using a dead weight loading system and induction heating. The Hastelloy X test is the first of its kind. More data will be obtained on both Haynes 25 and Hastelloy X at various temperatures and stresses.

Irradiation Effects in Structural Materials

Four special quadrants containing 24 tensile specimens fabricated from Inconel 600 - Zircaloy-2 tubes have been sent to the ETR for irradiation in the G-7 loop. The specimens are formed so that the transition joint is contained within the gage length of each of the samples. The over-all length of these specimens is 8-1/4 inches.

These special irradiations are in support of the "Advanced BONUS Core Development Program" being conducted by General Nuclear Engineering Corporation. Goal exposures for these irradiations are



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$1.0 \times 10^{20}$ and $5.0 \times 10^{20}$ nvt (>1 Mev) at temperatures near 287 C. The test includes specimens from two different sized tubes, one with a 1.5 inch OD and one with a 3.3 inch OD. The smaller diameter tubes are approximately 0.030 inch thick and were fabricated by a coextrusion process. The larger diameter tubes are approximately 0.90 inch thick and were fabricated by an explosive welding technique. Tests will be performed on specimens fabricated by each method in the as-received condition, after discharge from the out-of-reactor loop, and after irradiation in the G-7 loop. Tensile tests will be conducted at room temperature and at 287 C on specimens in each condition.

The dead-weight loading frame described in the last report was used to compare the stress-to-rupture lives of two Inconel X-750 specimens, one irradiated at 50 C to $1.8 \times 10^{20}$ nvt (E>1 Mev) and the other an unirradiated control. Both specimens had been given the standard heat treatment [1177 C/2 hrs, AC; 843 C/24 hrs, AC; 704 C/20 hrs, AC]. The conditions of the stress-to-rupture test were 48,000 psi stress at a temperature of 1350 F (732 C). The results are as follows:

| Specimen | | Hours to Rupture |
|---|---|---|
| Control: | A007 | 175.2 |
| Irradiated: | A001 | 9.5 |

Specimens in two experimental heat treatments will also be tested.

Environmental Effects on In-Reactor Zircaloy Corrosion

On the assumption that aqueous Zircaloy ocrrosion in-reactor is accelerated by oxygen dissolved in the water, hydrogen additions (to repress free oxygen) were made to the coolant of ETR G-7 loop during cycle 65. Zirconium alloy samples exposed to the hydrogenated, low oxygen coolant have now been weighed and corrosion rates calculated. Corrosion rates were reduced at least a factor of two by hydrogen additions at all flux levels, but still were greater than out-of-reactor. The beneficial effects of hydrogen addition to the coolant were greater at the lower flux levels. Determination of hydrogen pickup fractions and confirmatory determination of corrosion rates by oxide thickness remain to be done.

ATR Gas Loop Studies

Model Loop Studies. Subsequent to the 2000 F run on Sept. 17, 1964, two bellows in the primary piping were replaced with pipe sections,



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and, in addition, a new flow tube was installed. During the period of Oct. 20-23, the model loop ran continuously at various flows, pressures, and temperatures. Additional test equipment was installed on Oct. 26, and more test runs were made during the following three-day period. Much of the operating time was with 2060 F helium temperature at the heater outlet and 1850 F helium flowing into the test section.

One of the purposes of the sustained tests was to evaluate the behavior of layered metallic foil insulation and to determine if this type of insulation meets the design requirements of the ATR gas loop. During the tests the thermal flux through the metallic insulation was determined by three independent means. The preliminary thermal conductivity values calculated from the test data verified that the thermal conductivity under the test conditions was independent of helium flow rates through the insulation. Conductivity values ranged from 0.3 at 800 F to 0.5 at 1600 F.

Preliminary indications based on comparisons of heat loss through the fiber-frax insulated piping and the metallic foil insulated test section show the metallic foil to be a better insulation than cast fiber-frax.

In order to obtain thermal conductivity data without delay and to develop data on amounts of impurities to expect during loop startup, the loop was operated without a bypass cleanup system. Helium purity was a problem during the period of Oct. 20-23. Although the loop was evacuated, purged, and heated to maximum temperatures with 2060 F helium and the cycle repeated many times, impurities remained high. At shutdown on Oct. 23, these impurity levels were measured in the helium:

| | | |
|---|---|---|
| Hydrogen | 1500 | vpm |
| Oxygen | 0.2 | " |
| Nitrogen | 40 | " |
| Methane | 20 | " |
| Carbon monoxide | 1300 | " |
| Carbon dioxide | 170 | " |
| Water | 700 | " |

On Oct. 27, the loop was evacuated, purged, and pressurized with Grade A helium for startup. A temporary helium bypass cleanup system consisting of a liquid nitrogen cooled trap was installed across the circulator. The combination of bake-out during the previous week's high temperature operation and the bypass cleanup system reduced the impurities to a significantly



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lower level. At shutdown on Oct. 29, these impurity levels were measured in the helium:

| | |
|---|---|
| Hydrogen | 780 vpm |
| Nitrogen | 10 " |
| Carbon monoxide | 58 " |
| Water | 30 " |

Since initial loop startup in July, the gas bearing circulator has been run with helium pressure varied from 0-350 psig and the inlet helium temperature varied from 60-650 F with speed variations from 3600-25,000 rpm. To date the circulator has run a total of 133 hours with 46 starts. No operational problems have been noted and no maintenance has been needed.

Although more detailed studies are to be made in the near future, preliminary evaluation of the attemperating principle shows it to be a very effective way to reduce gas temperature.

$ZrO_2$-$O_2$ Probe. The $ZrO_2$-oxygen probe described last month has been shown to be extremely sensitive and appears capable of following $O_2$ concentration from one atmosphere air to below one part per billion $O_2$. It was found that $H_2$ in the gas stream invalidated the reference $O_2$ concentration (NiO-Ni). The design was modified to permit air flow through the reference side, thus sweeping out traces of $H_2$ and providing a stable $O_2$ concentration. A new device was built and installed in 314 Building and is now being calibrated.

Gas Chromatograph. The CMO gas chromatograph is now operational in the 314 Building. Several devices were tried to resolve the $O_2$-Ar interference problem, including variation of collector voltage and use of a Cu getter to eliminate the $O_2$ peak. Neither was successful. It appears the best approach will be to cool the column to -100 C where $O_2$ and argon separate on molecular sieve.

High Temperature Oxidation of Candidate ATR Materials. Thermal cycling tests are in progress on Haynes Alloy 25 in a 25 torr static oxygen atmosphere and in a 0-020 torr replenished oxygen atmosphere at 1120 C. The tests have initially been scheduled for ten cycles consisting of 3000 minutes at temperature and 1200 minutes cooling time per cycle. In the fourth cycle for Haynes Alloy 25, in 25 torr $O_2$, the oxidation rates are comparable to those in isothermal tests. The continuously recording microbalance indicates that although spalling of the oxide occurs during cooling, the residual scale appears protective.



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Corrosion of Haynes Alloy 25 in High Velocity, Contaminated Helium. A sample of Haynes Alloy 25 was exposed to high velocity, contaminated helium (~500 fps) at 2050 F for 111 hours, to determine if erosion by high velocity gases is an important consideration in the ATR loop. The system was initially charged with Grade A helium to a pressure of 350 psig. No attempt was made during the run to clean the gas. The oxygen level rapidly diminished to undetectable levels, while the Co, $CO_2$, and $H_2$ levels gradually increased to 10, 3, and 32 ppm, respectively.

The sample was stripped of its oxide film in a molten salt bath and the metal loss computed to equal a penetration of 0.25 mil. This penetration is in good agreement with that calculated from weight gain data obtained in oxygen at 18μ or 42μ pressure and a temperature of 2050 F. These preliminary data indicate that gas velocity effects are not important.

Gas Loop Materials. The evaluation of Hastelloy-X and Hastelloy-W as candidate weld metals for dissimilar metal welds between AISI 316 stainless steel, Hastelloy-X, and Haynes Alloy 25 is proceeding.

6. Nuclear Graphite Studies

Kinetics of the Reaction of Carbon Dioxide with Hydrogen. Certain solid surfaces strongly catalyze the reaction between carbon dioxide and hydrogen, and we have used TSX graphite to determine the effect of graphite on the reaction rate. It was reported earlier that no enhancement of rate was observed at 920 C. The study has now been extended to cover: (1) a range of temperatures from 680 to 970 C; (2) a range of hydrogen partial pressures from 0 to 400 torr; and (3) a range of carbon dioxide partial pressures from 0 to 500 torr. Over this wide range of conditions, we observe no substantial difference between the homogeneous gas-phase reaction and the reaction in the presence of graphite.

The graphite rod appeared to be oxidized homogeneously since there was no evidence of pitting. It should be emphasized, however, that if the rod becomes contaminated by handling an enhancement of the reaction rate of carbon dioxide with hydrogen is observed.

Inhibition of Air Oxidation of Graphite. The inhibiting effect of $CF_2Cl_2$ on graphite-air oxidation has been explained as a result of preferential chlorine adsorption on the graphite. The study of this effect disclosed complications which are only partially understood. An initial and temporary accelerated oxidation



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of EGCR graphite occurred while vanadium and other impurities were being removed. This effect is not exhibited by purer graphites such as CSF, SP-6, and spectrographic graphite. It is probably that $CF_2Cl_2$ initially enhances the catalytic effect of certain impurities, most probably vanadium, which is present in EGCR graphite. Surface pitting which is ultimately produced by vanadium and other catalytic impurities is eliminated by $CF_2Cl_2$ or chlorine probably by removal of the impurity, and thus for an equal weight loss a larger surface area is produced. The larger surface area leads to an increased oxidation rate after the inhibitor is removed for a given amount of burn off.

The effect of the related compound, $CF_3Cl$, on EGCR graphite oxidation has also been determined. $CF_3Cl$ has greater thermal stability than $CF_2Cl_2$ and does not provide enough chlorine for inhibition; however, the accelerating effect is about a factor of five. The results with $CF_3Cl$ suggest that part of the effect of $CF_2Cl_2$ on graphite oxidation may be due to traces of $CF_3Cl$ formed by pyrolysis.

Thermal Conductivity of EGCR and TSX Graphite. Thermal conductivity measurements were completed for samples of EGCR and TSX graphite irradiated at high temperature to high exposures. Data were obtained by the thermal-pulse method in the temperature range 25 to 500 C. The EGCR sample was irradiated at a controlled temperature of 625 C to $3.33 \times 10^{21}$ n/cm$^2$, E>0.18 Mev. The thermal conductivity transverse to the extrusion direction was 0.045 cal/cm-sec °C, increased to 0.055 at 300 C, and then decreased to 0.050 at 500 C. The TSX transverse sample was irradiated at 800 C to $1.25 \times 10^{21}$ n/cm$^2$, E>0.18 Mev, and its thermal conductivity was 0.065 cal/cm-sec °C at 25 C, 0.072 at 275 C, and 0.070 at 500 C. Additional measurements are planned for TSX and CSF samples irradiated to greater than $10^{21}$ n/cm$^2$.

EGCR Graphite Irradiation. The ninth capsule, H-3-9, in the series of long-term irradiations of EGCR graphite has operated successfully for one cycle in the GETR. During capsule fabrication the end was broken off the 800 C high exposure, NC7, transverse sample. This was probably due to stresses built up in the sample which had caused measurable warpage during the two previous irradiations. The amount of warpage was indicated by an increase in the standard deviation of the post-irradiation sample lengths from a value of 0.001 to a value of 0.004%. By modifying the sample assembly with a small graphite collar it was possible to re-install the sample in the H-3-9 capsule. All thermocouples in the capsule are functioning properly and sample temperatures duplicate those of previous capsules.



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A report, HW-71500B, "The H-3 Irradiation Experiment: Irradiation of EGCR Graphite Interim Report No. 2," has been completed.

Annealing of Irradiated Graphite. Three capsules each containing two parallel and two transverse 4" long x 3/8" quarter-round CSF samples were irradiated at 400 C in the ETR to an accumulated exposure of 6 x $10^{20}$ n/cm$^2$, E>0.18 Mev. Prior to irradiation, samples were annealed at 1500 C for one hour and then thermal expansions from 25 to 700 C and lengths were measured. After irradiation the thermal expansions and lengths were remeasured prior to annealing. The samples in two capsules were annealed at 1100 C and 1500 C, and those in the third capsule were held as standards.

During the annealing it became apparent that the samples in one capsule were irradiated at 300 C or less. The c-spacing changes on annealing were also determined from the (002) diffraction line with a high temperature x-ray diffractometer. All of the radiation-induced transverse length changes and half of the parallel length changes were annealable for the lower irradiation temperature (<300 C). Annealing of the transverse length changes occurred in two temperature ranges of 400 to 700 C and 1100 to 1500 C.

For samples irradiated at 400 C only an additional shrinkage was observed transverse to the extrusion direction during annealing between 1100 to 1500 C. Neutron irradiation increased the CTE transverse to the extrusion direction by about 5 to 7%, but the annealing treatment eliminated most of this expansion. Also, the c spacings increased from irradiation, but all of this increase was annealed during heating between 1100 to 1450 C. No significant changes in CTE were observed parallel to the extrusion direction.

These results support the theory that high-temperature radiation damage includes an unannealable contraction concentrated parallel to the extrusion direction as well as effects due to crystallite dimensional changes.

Effect of Impregnation on High Temperature Contraction of Graphite. The sixth irradiation has been completed on a series of samples which are intended to determine the effect of impregnation with furfuryl alcohol on high temperature radiation-induced contraction. The test involves TSX (NPR) graphite and HLM-85, used in HTGR fuel elements. Carbon added by the reimpregnation process amounted to 8% for TSX and 4% for HLM-85. Irradiation temperatures were near 1000 C. Although exact exposure data for the sixth irradiation are not yet available, the cumulative exposures will range from 4 to 6 x $10^{21}$ n/cm$^2$, E>0.18 Mev.



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The most striking results were obtained from transverse samples of TXS. During the latest irradiation all transverse samples, including the unimpregnated control, expanded by amounts ranging from 0.05 to 0.33%. Previously only two of the four samples had exhibited positive length changes during an irradiation. The impregnated sample heat treated to 2600 C now has returned to its original length after having displayed a maximum contraction of -0.5%. These data are consistent with results from GETR irradiations.

Parallel samples of both TSX and HLM-85 continue the behavior established in previous tests. These results still support the theory that carbon from the impregnant contracts at a much faster rate than the base graphite.

In-Reactor Creep of Graphite. The test that permits the irradiation of three samples under constant load of 600 psi is in progress at the Engineering Test Reactor. In 66 effective days of reactor operation, the test has accumulated an exposure of approximately $9.8 \times 10^{20}$ n/cm$^2$, E>0.18 Mev. The break detector indicates that none of the samples have broken. The test will continue until there is an indication of failure in one of the samples. The purpose of this test is to define the strain limit of graphite more precisely. A new test at 800 psi is ready for installation when the present capsule is removed.

7. Boronated-Graphite Studies

2-C Irradiation Tests. The 2C-3 capsule continues to operate satisfactorily. Measurement of the helium pressure in the primary blanket-gas system indicates a gradual increase proportional to exposure indicating a leak-tight capsule.

A small lathe enclosed by a glove box has been readied so that samples of irradiated boronated graphite can be taken at various radial depths for isotopic analysis of the $B^{10}/B^{11}$ ratio. This will yield a quantitative measurement of boron burnout as a function of depth to compare with theoretical calculations.

A comparison has been made of the length-change results of transverse boronated-graphite samples from the 2C (Hanford) irradiation with those from the ETR irradiations. For a given exposure, the ETR irradiations produce 4 to 6 times the length change observed in the 2C tests, when the sum of thermal and fast neutrons are used as a measure of exposure. The difference is even greater if only those neutrons of E>0.18 Mev are used to index exposure. It is thus apparent that a total neutron count is preferable to index the results



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of the two facilities, but there still remains a problem of exact rectification of results.

Temperature-Controlled, Shielded Capsule. The temperature-controlled thermal-neutron-shielded capsule continues to perform satisfactorily in the M-6 position of the ETR. The capsule contains 26 samples of graphite, impregnated with $B_4C$ particles at 5 and 7 wt% concentrations. The reference temperature 1000 F ± 20° was maintained for 25.8 effective days during cycle 66. The temperature of the sample-train column follows a modified cosine curve, ranging from approximately 700 F at the lower sample to 1000 F at the reference sample, to 930 F at the upper sample. Due to inherent ETR flux shifts, the lower and upper samples varied from the above averages by approximately ± 50° and ± 25°, respectively.

The boron-steel shield tube that surrounded the capsule during cycle 66 was removed and replaced by a new shield tube. Calculations indicate that the effective thermal-neutron shielding power of a 1.85 wt% B-steel tube decreases by approximately 13% for each 28 effective-day cycle.

The test will be discharged at the end of November. The estimated peak neutron exposure at discharge will be approximately $0.43 \times 10^{20}$ (thermal) and $8.15 \times 10^{20}$ (>0.18 Mev).

8. Aluminum Corrosion and Alloy Development

Weight loss measurements after chemical cleaning to remove oxide were obtained for the last charge of aluminum-clad fuel elements exposed in C-1 Loop for 58 days at 260 C loop outlet temperature and pH 4.5 ($H_3PO_4$). The corrosion rates so measured ranged from 0.3 mil per month on the upstream element (which operated at the lowest temperature and heat flux) to 4.9 mils/mo on the second to downstream (X-8001) element. Most other elements had rates of 2 to 3 mils/mo.

The corrosion rates of the X-8001, A-288, and KYZ alloys tested were not significantly different from each other, except for the X-8001 element noted above. The two downstream elements (X-8001 and KYZ) showed extensive nonuniform attack with many shallow pits about 2 to 10 mils deep. No nonuniform corrosion was evident on the other six pieces.

9. Metallic Fuel Development

Irradiation of Thorium-Uranium Fuel Elements. Irradiation of three tubular Zircaloy-2 clad, thorium-2.5 wt% uranium-1.0 wt%



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zirconium fuel elements continued successfully in the ETR-P7 loop. The fuel elements are currently operating at a maximum temperature of 460 C, a surface heat flux of 51 cal/sec-cm$^2$ (6.8 x 10$^5$ Btu/hr-ft$^2$) and a specific power of 45 watts/gm (138 kw/ft). The integrated exposure is 3.1 x 10$^{20}$ fissions/cm$^3$ (9300 Mwd/ton). Fuel swelling continues to be no more than that expected from the solid fission products. The measured volume increase at 9300 Mwd/ton was 0.7% which is 0.2% less than that reported last month. The difference is presumably within the experimental error of measurement.

Irradiation of Metallic Uranium with Sub-Micron Uranium Carbide Dispersions. The irradiation performance of fuel rods containing a sub-micron dispersion of uranium carbide in uranium is being evaluated in the ETR. Of the three irradiation test capsules, two contain fuel rods identical in uranium compositions, but with one having a uranium carbide size of 2-5 microns produced from arc-melted uranium and the other a uranium carbide particle size of less than 0.5 micron produced from the uranium shot. The third capsule contains two fuel rods with the fine carbide.

One capsule has achieved goal exposure of 0.3 at% burnup and has been shipped to Radiometallurgy for examination. Maximum temperature at the center of the uranium rod with the sub-micron carbide particles was 500 C and on the rod with 2-5 micron-sized carbide particles was 450 C. The remaining two capsules operated at a maximum fuel temperature of 600 to 625 C during the first cycle of irradiation. A drop in power in that region of the reactor resulted in a decrease in fuel temperature to the 475 to 500 C range during the second and third cycle of irradiation. The exposure at the end of the third cycle in these two capsules was 0.17 at%.

Examination of the two rods shipped to Radiometallurgy is under way. Optical metallography shows extensive tearing porosity in the uranium with fine carbides but none in the uranium with the coarse carbides. The tearing porosity may be related to the cladding failure (described last month) and the consequent loss of restraint.

Samples from both rods have been declad and post-irradiation annealed at 630 C for 100 hours and 610 C for 10 hours. Density measurements made before and after annealing were as follows:



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| | Rod | Annealing Temp, °C | Time hr | %Δρ Resulting from Annealing | %Δρ Change During Irradiation |
|---|---|---|---|---|---|
| | 1 | -- | -- | -- | -2.3 |
| Coarse | 1 | 630 | 100 | -3.2 | -- |
| Carbide | 1 | 610 | 10 | -3.2 | -- |
| | 31 | -- | -- | -- | -6.3 |
| Submicron | 31 | 630 | 100 | 0 | -- |
| Carbide | 31 | 610 | 10 | 0 | -- |

Although the as-irradiated uranium density decrease in rod 1 was almost half that in rod 31, after the high alpha temperature anneal the coarse carbide uranium continued to increase in volume (density decrease), but the uranium with fine carbide showed no further volume increases.

Optical metallography and electron microscopy will show the extent to which carbide particles influenced the density or porosity changes that occurred during annealing.

10. USAEC-AECL Cooperative Program on Development of Heavy Water Moderated Power Reactors

Two-Phase Flow in Multirod Fuel Bundles. In many multirod fuel bundles an imbalance exists between the mass flow rate (lb/hr) and heat transferred to the individual flow channels formed between the rods. This means that the enthalpy of the coolant in one channel will, in general, be different from that of the neighboring channel at any cross section and a condition of two-phase flow in one channel and single phase flow in the neighboring channel might occur. In fact, two-phase flow might occur in each channel, but with unequal steam qualities. In either event flow diversion from the channel with a higher steam quality to the channel of lesser steam quality or of single phase liquid, will take place. The pressure drop in each channel of the system will be the same; therefore, one may establish the theoretical mass flow rate in each channel through the use of two-phase and single phase pressure drop relationships.

A computer program has been partially written which will predict the local enthalpy rise in a system of two parallel connected channels. This program uses the Martinelli-Nelson method for



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calculating two-phase pressure drops. However, an article written by J.R.S. Thom, entitled "Prediction of Pressure Drop During Forced Circulation of Boiling Water" in a recent edition of the International Heat Transfer Journal, presents a method for predicting two-phase pressure drops that is claimed to be in closer agreement with experimental work than the relationship given by Martinelli and Nelson. For this reason, Thom's procedure, instead of Martinelli and Nelson's, will be used to establish the theoretical flow diversion in the present analysis.

Thus far, a very simple analytical relationship for the two-phase to single phase pressure drop ratio has been developed using Thom's work, and is valid for the pressure range from 250 psia to 2100 psia. This expression was found to predict the pressure drop ratio within a few percent of that which has been found experimentally at Hanford using electrically-heated test sections.

11. Critical Flow at High Pressures and Temperatures

The investigation of maximum discharge rates from reactor piping breaks was continued with an experimental study at low temperatures and pressures of two-phase critical flow from the branch side of a tee to the atmosphere. Earlier, an investigation was performed studying critical flow from elbows to the atmosphere. It is believed that the chances of a piping break at fittings such as tees and elbows are increased because of maximization of bending moments at these points. However, predictions of maximum two-phase fluid losses cannot be accurately performed because the technology is not able to cope with the alteration of flow patterns caused by the fitting. Even if it were possible to calculate critical flow rates with the present available methods based, as usual, on a pressure at the downstream end of the piping system, this information would not be of much use in the practical prediction of the critical flow characteristics of piping systems with a break immediately adjacent to a fitting. This is because the pressure drop-flow behavior of fittings is not known at the extremely high two-phase velocities which will be encountered. In the present study, the combined critical flow-pressure drop behavior of several types of pipe fittings is being studied with the intent of developing theories and correlations which will enable simple and complete analysis of piping breaks at such fittings.

The test section considered in the present investigation consisted of a 1/2-inch weld-type pipe tee which was welded into the approximate center of a two-foot long piece of 1/2-inch schedule 40 stainless steel pipe. Steam-water mixtures were supplied to the tee



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from one side of the pipe and the other side of the pipe was capped. Flow to the atmosphere was through the branch side of the tee. Wall pressure taps were provided in the tee and along the pipe to determine the distribution of pressure in the test sections. Tubing from these pressure taps were routed to a Heise gage through a rotary pressure-scanning valve. Steam-water mixtures of the desired enthalpy and flow rate were made by the mechanical mixing of saturated steam at 225 psig and water at 385 F.

The pressure distributions along the test section and the associated critical flow rates were determined for mixture enthalpies of 350, 500, 600, 700, 800, 900, and 1000 Btu/lb. In general, the flow rates attained were between 0.5 to 3 lb/sec and were limited by the capabilities of the hot water supply system. Preliminary analysis of the data and comparisons of this data taken with the tee test section to that obtained earlier with an elbow test section reveal the following:

(a) If the pseudo-critical pressure is defined upstream of the tee, the slope of the pseudo-critical pressure-critical flow lines are essentially constant at flow rates below approximately 2 lb/sec. This corresponds well to the behavior of elbows in critical flow.

(b) Comparing the pseudo-critical pressures obtained with the tee-test section to that obtained with an elbow-test section revealed that higher pressures were obtained with the tee for corresponding flow rates and enthalpies.

The ratio of tee to pipe critical flow rates was found to be constant for all corresponding pseudo-critical pressures and enthalpies and equal to 0.713 ± 3.9%. This regular behavior is surprising considering the complexity of the nonequilibrium two-phase expansion process characteristic of the critical flow process.

12. Phoenix Fuel Program

MTR Experiment. A parametric study to identify engineering operating requirements for an MTR-Phoenix experiment has continued. The analyses were concerned principally with methods which would result in increased plutonium fuel loading through increased metal-to-water ratios. The relationships of metal-to-water ratio, fuel loading and heat transfer characteristics were examined by (1) reducing the number of plates with coolant channel and velocity



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remaining the same as present MTR design (0.117" channel width at 30 fps velocity), and (2) decreasing the coolant channel to 0.0845" and velocity to 25 fps (thereby maintaining the core pressure drop at about 40 psi).

These preliminary calculations indicate that high power (40 Mw), high specific density (kg/l) cores desired for Phoenix program will require improvements in the present indicated horizontal power distributions based on physics calculations.

13. Nuclear Rocket Fuels Studies

Research and development in the field of nuclear fuels and alloy development in support of NASA programs continued. Details of these activities are reported separately via distribution directly to the sponsors.

D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

This program is directed toward establishing the combined effect of impurities and neutron irradiation on the properties and structure of specific metals, and deducing from thermally activated recovery processes how the damage state can be altered. Present studies involve single and polycrystalline specimens of molybdenum, nickel, and rhenium.

Examination of transmission electron micrographs of molybdenum foils which were irradiated to $10^{20}$ nvt (E>1 Mev) disclosed a nonuniform distribution of defect clusters. Many of the spot defects are associated with interconnecting dislocation networks or rings. An estimate of the number of spots gives an approximate concentration of $10^{16}$ cc, about the same or less than the concentration observed after an exposure of only $10^{19}$ nvt in the same material. This discrepancy can be explained if we assume the spot defects to be unresolved prismatic loops; as more defects are formed and migrate to the loops, the loops grow until adjacent loops become near enough so that the attractive force between them results in coalescence of the loops into a single larger one. Interactions between growing loops of unlike Burgers vectors result in complex network formation. These processes account for the reduction in the extrapolated density of spot defects. The Burgers vectors of the observed dislocation rings have been determined to be $\frac{a}{2}$ <111> , the rings being uniformly distributed on all {111} planes.



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Dislocations having other Burgers vectors have not been observed.

X-ray line broadening and lattice parameter changes have also been studied in molybdenum foils irradiated to $10^{20}$ nvt and subsequently annealed. Foils containing less than 10 ppm carbon, 100-200 ppm carbon, and 400-500 ppm carbon were identical in line breadth and lattice parameter after $10^{20}$ nvt, all showing pronounced broadening and a lattice parameter slightly less than that measured after $10^{19}$ nvt. After annealing at 750 C (1382 F) for two hours, the line breadths of all foils decreased, those with the higher carbon content being slightly broader. Lattice parameters of the 100-200 ppm C and 400-500 ppm C foils decreased to near pre-irradiation values, while the lattice parameter of the high purity (<10 ppm C) foil remained unchanged. Apparently the recovery of lattice expansion after irradiation is accelerated by the presence of excess carbon. Line broadening, on the other hand, is increased by the presence of carbon.

A complete Fourier analysis of the broadening in a molybdenum foil containing 400-500 ppm C irradiated to $10^{20}$ nvt has been performed for comparison with the earlier analysis for similarly-irradiated high purity molybdenum. The x-ray particle size is slightly greater (1350 A compared to 1000 A), and the strain distribution in the lattice is quite similar. Further analyses of irradiated and annealed samples are in progress.

The defect configuration in the molybdenum foils irradiated to $10^{20}$ nvt, as seen in electron micrographs, suggests an agglomeration or coalescence of defects. If the resulting arrangement of clustered defects is such as to produce local variations in electron density, the technique of small-angle x-ray scattering may be applicable. This possibility is being explored for possible use.

2. Plutonium Physical Metallurgy

The objective of this program is the derivation of fundamental information relative to (1) the kinetics and mechanics of the phase transformation in plutonium, and (2) the mechanisms by which monoclinic plutonium deforms.

Because considerable conjecture exists on the mechanism of the beta to alpha transformation of plutonium, experiments were conducted to establish whether applied stress markedly affects this transformation. Decreasing the maximum transformation rate from 25% per minute to 0.2% per minute increased the unit length change only 20% during transformation under compressive stresses of 2000



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and 5000 psi. If the transformation proceeds by a diffusion process, the concentration of defects and their mobility should be high, and, as a consequence, the unit length change should be much greater at the lower transformation rate. These data support the thesis that diffusion is not the major mechanism for the beta to alpha transformation.

Experimental equipment has been designed and fabricated for preliminary and final metallographic polishing of beta plutonium in the temperature range 100 to 140 C.

X-ray goniometer investigations of alpha-rolled plutonium, as well as the columnar grained material resulting from the transformation of beta plutonium to alpha under a compressive load, have not, as yet, yielded sufficient data for the construction of definitive pole figures. This is, in part, due to the complexity of the alpha structure. There is evidence also to indicate that grain size continues to be a problem, particularly in the case of the columnar grained specimen.

Preparations are under way to roll new specimens. The stock will be the second half of the ingot of electro-refined metal previously used to produce the present samples. This portion of the casting has been stored in vacuum far in excess of 120 days and will provide further data on the observation that self-damage radically affects the working characteristics of alpha plutonium.

An attempt is being made also to produce evidence of slip or twinning, or both, in alpha plutonium by deforming a previously polished surface by compression. By using a longitudinal section of the columnar grained material, it is hoped that crystallographic details which have been derived from the x-ray goniometric studies can be applied to determine the slip and twinning systems. Some work along this line has been reported by the French but without corroborative x-ray evidence.

## E. CUSTOMER WORK

### 1. Radiometallurgy Laboratory

#### Examinations

The results of routine examinations and measurements are, or will be, reported as part of the sponsoring research and development programs. During the period September 27 to October 22, 1964, Radiometallurgy Laboratory output included:



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| | |
|---|---|
| Metallography samples processed | 61 |
| Photomosaics | 6 |
| Autoradiographs | 10 |
| Replicas | 17 |
| Burnup dissolutions | 32 |
| Decladding dissolutions | 6 |
| Fission gas samples | 9 |
| Tensile Tests (rm. temp.) | 27 |
| Rockwell hardness tests | 95 |
| Microhardness tests | 7 |
| Densities | 28 |
| Induction heated gas sampling | 53 |
| X-ray | 3 |

Equipment

Project CAH-136, Service Addition - 327 Building. Vitro Engineering completed detailed design of the 327 Building service addition. Detailed drawings were received and comments submitted. AEC approval was obtained for revision of the design criteria to permit a change in the locker room arrangement.

Project CGH-857, Physical & Mechanical Properties Testing Cell. Final calibration of the remote impact tester to Watertown Arsenal testing standards is in progress at Testing Machines Incorporated. Initial calibration attempts were not successful, and further modification of the tester hammer was needed.

"E" Cell Metallographic Facilities. All of the in-cell metallographic preparation equipment has been completed and installed except the orbital grinder. Fabrication of the grinder was completed, and final testing is in progress. The remote metallograph fabrication was completed, and final testing and alignment is in progress.

"B" Cell Modification. Machining of the large viewing window opening in the "B" cell door is in progress at 100-H shop. Shipment of the viewing window was delayed by dimensional errors made by the window frame fabricator.

"I" Cell Microhardness Blister. Final testing of the remote microhardness tester was completed, and the tester was placed in operation.

Remote Belt Sander. Final design of the remote belt sander hood and baffle for coolant containment was completed, and fabrication is in progress.



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2. Metallography Laboratories

Routine metallography work will be reported as part of the sponsoring research and development component's work. However, items of unusual interest or representing departures from routine operations will be reported here.

A technique that permits characterization of tungsten powders in the electron microscope by extending the Mathews and Buthala carbon-film preparation technique has been successful. A benzene slurry of the powder is dropped onto the surface of a glass slide. The dispersed powder is shadowed with palladium at 45° and uniformly covered with an evaporated layer of carbon at 90°. The deposited film is scribed into a grid of small squares, properly sized for examination screens. The slide is immersed in 25% aqueous potassium hydroxide, and the squares float to the surface. Two changes into fresh KOH completes dissolution of the tungsten powder. The film is transferred to several water wash-baths and is collected on examination screens. Dissolution of the tungsten powder results in a replica displaying surface contours of the particle.

During the report month 308 samples were processed, a total of 487 micrographs and macrographs taken, 1871 negatives printed, and 6905 prints processed.

3. High Temperature Lattice Test Reactor (HTLTR)

Environmental and Mechanical Test Data on HTLTR Materials. Compression strength data at room and elevated temperature were obtained on candidate fuel and control rod materials for the HTLTR. Dispersions of $UO_2$ in graphite $B_4C$ in graphite were tested in the form of 0.3 inch diameter by 0.6 inch specimens at RT in air and at 1000 C in an argon atmosphere. Average test values obtained for these materials and pure graphite for comparison were as follows:

| | Compression Strength (psi) | |
|---|---|---|
| | RT | 1000 C |
| $UO_2$ - graphite (HL) | 2,390 | 2,970 |
| $UO_2$ - graphite (Vendor) | 16,900 | 17,600 |
| $B_4C$ - graphite (Vendor) | 3,960 | 3,430 |
| Graphite (305) | 3,470 | 3,680 |
| Graphite (AGSX) | 8,920 | 8,650 |

The above properties for fuel and control materials are in the transverse direction. Testing will continue in order to get longitudinal



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data and to evaluate additional lots of HL material which is still under development.

An environmental test for a detailed evaluation of the compatibility of $UO_2$ and TD nickel in the reactor atmosphere was started. In this test, simulated fuel element assemblies are exposed in a nitrogen-graphite environment at 1100 C for 100 hours. Similar tests in separate test chambers were started for an evaluation of possible $B_4C$ interactions with TD nickel and K23 insulating brick.

HTLTR Mockup. Leak checking of the large HTLTR mockup was completed. Seven room temperature evacuation cycles were completed, after which each of the four heating elements were operated at approximately 2 kw at a maximum indicated temperature of 500 F. Best sustained vacuum was 10 mm Hg, absolute. Gas samples taken during initial pump down indicated about 21 vol% $H_2O$. Typical water collection data were as follows:

| Date | Time | Vol.(ml) |
|---|---|---|
| 10/15/64 | 0930 | 320 |
| | 1030 | 550 |
| | 1130 | 320 |
| | 1330 | 560 |
| | -- | 120 |
| 10/16/64 | 0810 | 2850 |
| | 1010 | 390 |
| | 1100 | 120 |
| 10/17/64 | 0800 | 2720 |
| | 1535 | 1110 |
| 10/19/64 | 0800 | 5220 |
| | 1030 | 130 |
| 10/20/64 | 0815 | 1910 |
| 10/21/64 | 0800 | 2100 |
| 10/22/64 | 0815 | 2350 |
| | | 2260 |
| 10/23/64 | 0815 | 2650 |
| 10/24/64 | 0910 | 2370 |
| 10/26/64 | 0800 | 4950 |



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All parts for the horizontal control rod have been fabricated, the uranium-bearing pieces being made of plain graphite. Wiring of the drive subassembly has started, and the control elements have been assembled satisfactorily.

$UO_2$-Graphite Dispersions. Twelve HTLTR control rod sleeves were fabricated from 69 wt% $UO_2$-graphite. The fabrication process consisted of: (1) blending $UO_2$-graphite flour and a phenolic resin binder, (2) mechanical rolling and shredding, (3) drying at 100 C, (4) pressing and screening to -100 mesh size, (5) cold pressing, (6) heat treatment in a helium atmosphere after being packed in a graphite powder, and (7) final machining.

Two dies were designed and fabricated for use in forming the inner and outer fuel rods. Special heat treated tool steel was used. A special holding fixture was made for use in grinding the $UO_2$-graphite sleeves to final dimensions.

Additional sleeves are being fabricated to complete the quantity required for prototypic HTLTR mockup tests.

4. EBWR Fuel Elements

Irradiation Testing of EBWR Prototypic Fuel Rods. A maximum exposure of $3.4 \times 10^{20}$ fissions/cm$^3$ (12,600 Mwd/ton $UO_2$) was achieved for the 12 capsules still under irradiation. Performance continues to be excellent, and no failures have occurred. Low fuel densities and reactions between capsule components and fuel are probable reasons for anomalously high fission gas release values for two capsules examined after discharge.

5. Other Customer Work

$PuO_2$-62 wt% (80 Ni/20 Cr). A new technique for preparation of NiCr-clad $PuO_2$-NiCr compatibility samples is being evaluated. Pressed cermet cores are loaded into 1/4-inch NiCr cans, end caps welded on, the cans buried in $Al_2O_3$ within a double stainless steel Nupac assembly and impacted at 1200 C.

I&E Pu-Al Fuel Elements. Twelve, 3.3 wt% PuAl, aluminum clad, I&E fuel elements were fabricated for IPD test purposes. The elements were fabricated using standard IPD, I&E components and machined PuAl cores to achieve a slip fit. Essentially 100% core-clad contact resulted from hydrostatic pressing (15,000 psi and 100 C) after assembly and closure welding. Standard supports were attached to the elements by ultrasonic welding.



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Boiling Burnout Studies for the Advanced Test Reactor (ATR). Calculations were made to modify the design of the second electrically heated test section which will have a variation of heat flux across the flow channel as well as the cosine variation along the length. This modification will fit the latest heat flux pattern predicted by the Phillips Petroleum Company for a fuel element in a distorted flux area in the reactor.

The heat generation curve for the reactor varies by a factor of 2 across a 3.78-inch coolant channel. Since the test section coolant channel is about one-half the width of the reactor coolant channel, the lateral heat flux variation across the test section was chosen to correspond to the heat flux profile across about half of the fuel plate. It was found from the calculations that a reasonable approximation of the reactor heat flux curve can be obtained with a test section which has a smooth lateral taper. The taper angle will vary with axial location on the test section in order that the lateral flux distribution be the same along the entire test section length. Hand removal of very small amounts of heater bar material might be done to enhance the lateral heat flux profile near the channel edge.

Some additional calculations were performed to investigate the coolant channel edge heat flux between fuel plates in the ATR. Two effects were investigated. First, it was found that the temperature difference between the coolant stream between two fuel plates and the coolant stream between the fuel elements support plates greatly affected the edge heat flux. The edge heat flux varied from +21% of the channel center flux with 0° $\Delta T$ to -35% with 180° $\Delta T$ between coolant streams. In other words, with the fuel plate coolant 180° above the temperature of the support plate coolant, heat will be removed from the fuel plate coolant at the channel edges. Since the total heat input to the support plate coolant streams is low, a large $\Delta T$ will exist, and, thus, negative heat fluxes at the channel edge will be common in the ATR.

Second, at certain locations on ATR fuel assemblies, bumpers are provided on the outside of the support plates to assure positive spacing between fuel assemblies. As presently designed, there will be no support plate coolant stream at bumper locations. This not only means that the fuel coolant channel edge flux will be psoitive, but, also, it means it will be relatively high--about 44% of the center channel heat flux. The edge heat flux at bumper locations could easily be reduced by machining longitudinal slots in the bumpers. If this were done, the bumpers would act as fins to conduct heat away from the fuel plate channel edge rather than as heat insulation to the channel edge. However, since the bumpers are only about two inches long, they would probably not cause burnout on the edge with these heat fluxes.



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Critical Mass Physics. The horizontal safety and control rods for the split-table machine were completed and delivered to the Critical Mass Laboratory for installation. These rods move plutonium-bearing plastic blocks towards a known near critical mass of similar construction. On receipt of a scram signal, the rods are withdrawn automatically by a spring triggered by interrupting current to a holding electromagnet. High speed movies show that the release time of the magnet is 0.02 sec, and the rod is fully withdrawn in 0.12 sec. Excessive vibration of the units during the scram cycle was solved by adding telescoping adjustable braces to the units.

DOD Support. Development of special fabrication methods for DOD programs continued. Details of these activities are reported separately via distribution directly to the sponsors.

F W Albaugh

Manager
Reactor and Fuels Laboratory



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# PHYSICS AND INSTRUMENTS LABORATORY

## MONTHLY REPORT

## OCTOBER 1964

## FISSIONABLE MATERIALS - 02 PROGRAM

### REACTOR

#### NPR Utilization Studies

The NPR Phase III fuel cycle analysis using uranium metal fuel has essentially been completed. A supplemental work order has been received from NRD to extend the Phase III fuel cycle studies to include $PuO_2$ enrichment in $UO_2$ and perhaps in $ThO_2$ if time permits.

A two-dimensional analysis of the NPR lattice using the code 9-ANGIE has been started to resolve the problem of reactivity averaging in determining the reactor multiplication for a graded fuel loading.

#### NPR Coproduct Experimental Program (1.95% U-235)

The scope of the experimental program on the 1.95 NPR coproduct lattice has been expanded. Originally the measurements were to include $k_\infty$ determinations on the wet and dry lattice plus determinations of thermal utilizations and spectral indices for both lattices and conversion ratios on the wet lattice. To assist in the hazards evaluation of a target melting incident, additional measurements will be made of $k_\infty$ of the dry lattice without target. To support the control rod enhancement effort differential, $k_\infty$ measurements will be made on a few strong absorbing samples placed in the N-Reactor neutron spectral environment set up in the core of the PCTR.

The N-Reactor control rod system can be strengthened by replacing the natural boron with $B^{10}$, and by increasing the surface area of the control rod. Some degree of verification of these predicted effects can be obtained in scaled experiments performed in the sample of the N-Reactor lattice used in the PCTR experiments. There, in the correct neutron spectral environment, various composition and shape samples of neutron absorbers can be compared in simple reactivity measurements. To this end, an attempt is being made to locate samples of $B^{10}$ and Hf. To date, approximately one pound of amorphous $B^{10}$ has been located in the Equipment Development Subsection of IPD. Plans are for NRD to oversee the fabrication of this $B^{10}$ into a rod, encased in aluminum for a test sample. These tests are expected



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to take place in mid-November.

NPR Block Test

The expected fabrication yield of reactor grade 1.95% driver fuel for the N-Reactor 1.95% Block Test is 21 to 28 full N-Rreactor columns. NRD Chemistry and Metallurgy requires as many as four of these columns, to be removed one at a time at various stages of exposure. NRD Reactor Physics interests require approximately five full columns for conversion ratio determinations. A 21-tube block load (a 5 x 5 array with the four corner tubes removed) has been proposed as a means of fulfilling these requirements. The reactivity, flux flattening, and neutron spectral conditions in a block of this size, placed in the center of N-Reactor, are being evaluated. Techniques being employed are those of the two-group diffusion model used in earlier studies of flux flattening in the spike loading studies. Present work is somewhat more complicated, however, because of the neutronically black target element, which is difficult to analyze in the diffusion approximation. Transport theory code (THERMOS) is being employed in an attempt to obtain an acceptable solution. The results of these analyses are expected to clarify the question of flux peaking in the 1.95 block, and to show that a 21-tube block is sufficient to meet neutron spectral requirements.

Coproduct Experiments for NPR

Experiments are starting in the PCTR to measure the infinite medium multiplication factor $k_\infty$ and conversion ratios of both driver tube and target rod for the coproduct lattice. All components have been fabricated and loaded into the PCTR, and the first critical approach is starting.

Utilization of Plutonium Fuel

Multi-group computations completed for I&E Pu-Al production reactor elements now include a set containing a greater degree of space and energy detail than in previous calculations. The integral transport code THERMOS was used to obtain cross sections for fission and absorption in the Pu-240 resonance energy group and the thermal group. These data were obtained for cells whose fuel contains 1.0, 1.8, and 2.2 w/o high exposure plutonium. The computations were repeated with $H_2O$ regions voided to obtain void coefficients. Four-group cross sections were generated for these cases using GAM and TEMPEST in ZODIAC, which also produced HFN solutions of fluxes and $k_\infty$ in the heterogeneous model. Comparison (in a zero-dimensional model) was made between modified GAM-TEMPEST results, and a four-group cell-averaged cross section set incorporating THERMOS results. Fluxes were then computed using a few group (infinite, homogeneous) diffusion model and solutions for $k_\infty$ were obtained. These different methods give grossly different $k_\infty$ results.



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However, the calculated void coefficients are not greatly different for the three cases.

Subcritical Experiments with Enriched N-Fuels for Nuclear Safety Guidance

Exponential and neutron multiplication experiments were continued with enriched N fuels in light water that provide data for nuclear safety guidance. Measurements have now been completed with two fuel assemblies in each of two lattices. The fuel assemblies are the 1.25% enriched outer tube (2.4 in. o.d., 1.8 in. i.d.), and the tube-in-tube assembly consisting of the outer tube and an 0.95% enriched inner tube (1.25 in. o.d., 0.44 in. i.d.). The length of the zirconium clad fuel elements is 26 in. The cylindrical arrays were 52 in. in height with each fuel column being comprised of two fuel elements. A hexagonal loading pattern was used for the lattices, for which case the lattice cell is an equilateral triangle with equal separation between adjacent rods. Results of the measurements completed during the month are presented below.

Tube-in-Tube Assembly
(1.25 wt% outer tube, 0.95 wt% inner tube)

| Lattice Spacing | $H_2O/U$ Volume Ratio | Critical Buckling | Computed Critical No. of 52 in. Assemblies from Buckling | Computed Critical No. for 26 in. Assemblies in Cylindrical Array |
|---|---|---|---|---|
| 2.8 in. | 1.30 | 3244 μB | 101.2 | 173.7 (~9710 lb U) |
| 3.1 in. | 1.86 | 3884 μB | 67.4 | 105.5 (~5894 lb U) |
| | | Outer Tube Only (1.25 wt%) | | |
| 2.8 in.* | 2.53 | 4620 μB | 62.0 | 88 (~2870 lb U) |
| 3.1 in. | 3.36 | 3789 μB | 67.9 | 106.5 (~4036 lb U) |

* Results for 2.8 in. lattice were given in previous month.

Further data are required to determine the maximum buckling in either case and the minimum number of 26 in. fuel elements (or assemblies) that can be made critical in water. Preliminary results suggest the critical number of tube-in-tube assemblies may not differ too significantly from those of the outer tube only.



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## Instrumentation

Two more hybrid preamplifiers were completed for use with the regular fission detectors of the N Reactor subcritical neutron flux monitoring system. These units plus a similar one completed earlier replace proportional counters. The preamplifiers were required to achieve proper performance with the fission detectors.

Design and development effort was initiated on a system for measuring cooling age of irradiated reactor fuel. Circuit configurations were established and appropriate purchase specifications are being prepared. The system will be assembled in a manner to permit easy transport from one reactor area to another for the measurement work. Scintillation gamma spectrometry, with all solid state circuits, will be the measurement technique.

Neutron flux monitoring techniques and heat generation rates in cadmium were discussed with members of Irradiation Testing, IPD.

## System Studies

N Reactor's pressurizer spray control system was studied in detail to determine the cause of malfunctions which gave rise to excessive pressure transients in the primary coolant system during reactor scrams. It was found that the improper limiting of the proportional-plus-derivative controller unit caused saturation of the system. The addition of limiters to the controller corrected the difficulty.

Equations for an N Reactor three-surge-tank analog model were derived and programming was started.

# SEPARATIONS

## Critical Experiments with Plutonium Solutions

A series of criticality experiments were completed with plutonium nitrate solutions in a 15.2 in. diameter stainless steel sphere (30.2 ℓ volume). The data obtained from these experiments have provided critical mass values for Pu solutions in the vessel bare, and with reflectors of water and concrete. Plutonium concentrations in the nitrate solutions have ranged from 24 to 170 g Pu/ℓ. The most recent experiments (water reflected sphere) provide the following information on criticality in the full sphere with high acid molarity.



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| Critical Pu Conc. | Acid Molarity | Total Nitrate | H/Pu Atomic Ratio | Critical Mass |
|---|---|---|---|---|
| 38.7 g/ℓ | 7.70 | 520 g/ℓ | .556 | 1.17 kg Pu |

Neutron flux measurements were also made for determining the critical buckling of the solution. The variables affecting criticality (such as the nitrate, type and extent of reflector, etc.) are needed for nuclear safety guidance and also for the development of reliable theoretic methods for computing criticality.

Neutron Multiplication Measurements on Large Arrays of Fissile Material

Existing nuclear safety specifications in CPD restrict the number of SN and PR cans of Pu solution that may be stored together to 15. The number of these cans currently in use poses problems in available storage areas. A series of subcritical neutron multiplication measurements were performed on arrays of SN and PR cans filled with plutonium nitrate solutions to determine if the nuclear safety limit could be increased. The principal difference between the PR and SN cans is that the inner cylindrical container (5.8 in. i.d. with 10 ℓ volume) of the PR can is surrounded by a lead-cadmium alloy about 1/2 in. thick, while the SN can does not contain this shielding. Sixty-four SN cans containing 109 kg of Pu solution were positioned in close contact in a planar array at the Critical Mass Laboratory. The number for criticality was estimated to be $\geq$ 350 cans, and the shape of the multiplication curves indicated an infinite number would probably be subcritical. The same cans were then stacked in a double tier planar array. The results showed the critical number to be $\geq$ 230 in this geometry. Layers of plywood were placed on top of the array and on the three sides away from the wall to simulate additional reflection of proximate room walls; the results did not significantly affect subcriticality. The effect of placing materials in the space between cans of the array was also investigated and found to be small. Exchanging 15 SN cans for PR cans in the array indicated the latter to be slightly more reactive; the lead shielding in the PR cans acts as a reflector resulting in a higher $k_{eff}$ for each can.

Although some further experimental checks are indicated, preliminary results of the array studies show that SN and PR cans may be safely handled in much greater numbers than previously permitted.

Theory and Analysis

1. Minimum Critical Volume of Plutonium Nitrate Solutions

For a hypothetical homogeneous mixture of Pu atoms in water, the critical



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volume continues to decrease with Pu concentration reaching a minimum value for Pu metal (19.6 g/cc) of ~ 0.3 ℓ. However, analyses of critical experiments with $Pu(NO_3)_4$ solutions indicate that a minimum volume of about 7.2 ℓ will be obtained with nitrate solutions in spherical geometry at a Pu concentration of 175 g Pu/ℓ. This represents the limiting case for zero acid molarity, with the volume and mass both being larger for higher concentrations of Pu in the form of $Pu(NO_3)_4$ in water. The latter information is of practical interest for nuclear safety guidance in handling $Pu(NO_3)_4$ solutions.

2. Comparison Between Measured and Calculated Criticality for $PuO_2$-Polystyrene Compacts

Favorable comparisons have recently been obtained between computed and measured criticality of $PuO_2$-polystyrene compacts. Criticality has been calculated for these compacts (H-to-Pu atomic ratio of 15 and 1.12 g Pu/cm$^3$ density) by both multi-group diffusion (HFN) and $S_N$ transport (DTF) methods. The multi-group parameters were computed by the GAMTEC code utilizing the Sher normalized thermal group data. In the $S_N$ calculations, neutron scattering was approximated by a spherical harmonic approximation through the $P_1$ term. The diffusion theory calculations gave values of 16.0 and 6.0 cm as the critical thicknesses for the bare and Plexiglas reflected infinite slabs, respectively. These correspond to the experimental values of 15.9 ± 0.4 and 6.0 ± 0.4 cm. A critical slab thickness of 15.71 cm was calculated by the DTF code for the bare slab which also agrees with experiment within the error of measurement. The $S_N$ calculation for the reflected slab has not been completed. A time-dependent calculation with DTF gave a mean neutron lifetime of 1.67 μ sec which compares favorably with the measured value of 1.38 ± 0.017 μ sec from pulsed neutron source experiments.

3. Estimated Critical Mass for $Pu^{238}$ Metal

$Pu^{238}$ is currently being produced at Savannah River for use in radioisotopic heat sources. This Pu is made from irradiation of $Np^{237}$ shipped from Hanford. In view of cross section measurements which show an unexpectedly high fast fission cross section for $Pu^{238}$, there has been some concern about the criticality of this isotope. A Monte Carlo calculation has been made (based on somewhat uncertain cross sections) which indicates the critical mass of a bare sphere of $Pu^{238}$ metal may be as low as ~5 kg at a density of 19.6 g/cc (this is half the value for $Pu^{239}$ metal which is 10 kg). The estimated critical mass of $Pu^{238}$ is less than the amount to be used in some of the radioisotopic heat sources under consideration which shows the concern about criticality of $Pu^{238}$ is well justified.



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Critical Mass Laboratory Instrumentation

Installation and checkout of an on-line reactor noise analysis system for the Critical Mass Laboratory is essentially complete. The system continuously monitors the fluctuating signal from a neutron detector. After the signal is passed through active filter and rectifying circuits, the resulting frequency spectrum is displayed, plotted or digitally recorded. The analyzer covers a frequency range from 0.1 cps to 3.16 kc/sec. The system is ready for operational tests with noise of a known spectrum. An analysis of reactor noise provides information on the neutron lifetime and reactivity of the system under study.

Consulting Services on Nuclear Safety--Criticality Hazards

1. Nuclear Safety in HL

Specification K-1, which covers $PuO_2$-$UO_2$ Mark I fuel elements, was revised for PRTR.

2. Nuclear Safety in NRD

Temporary specification No. 3-64, covering the processing of a "scrap ingot" was reviewed and approved for N Reactor Fuels Engineering.

3. Nuclear Safety in Shipment of Fissionable Materials

Shipments of 5 kg of plutonium metal and 602 kg of 3 wt% enriched $UO_2$ were reviewed and approved.

Drop tests were performed on two of the Class I HL-designed shipping containers for fissile materials. The maximum test to simulate accident conditions was a drop from 30 feet onto a concrete surface. Also, a puncture test was made by dropping the containers from a height of 40" onto the end of a 6" diameter steel post. The containers survived these and other less stringent tests in fair order. As a test of a container's resistance to fire following an "accident," one of the containers will now be subjected to a standard one-hour fire test by Underwriter Laboratories. The inner container of a Class I shipping container is surrounded by materials which isolate the fissile material contained therein and prevent neutron interaction.

4. Nuclear Safety Training and Education

A course has been organized in nuclear safety for personnel in N-Fuels, NRD. Course emphasis is being placed on criticality of slightly



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enriched uranium. Nuclear safety of the fuel element fabrication operations in the 333 Building will be specifically covered. There will be about seven sessions. Classes began October 20 with 15 persons in attendance.

Another series of lectures in nuclear safety is currently being given to personnel of CPD. These lectures are designed to introduce technically oriented personnel to the technical bases and computational methods used to prevent criticality accidents. The material on nuclear safety is presented in sixteen $1\frac{1}{2}$-hour lectures. Lectures began September 14 with 25 persons attending.

Separations Instrumentation and System Studies

Investigations were initiated to incorporate an $Am^{241}$ detector and associated electronics into the $Pu^{239}$ liquid sample counting system which measures $Pu^{239}$ concentration by the detection of the 17 keV X-rays. The added $Am^{241}$ detection will be used to subtract the minor, yet troublesome, contribution to readout from the $Am^{241}$. The subtraction technique, if successful on the liquid sample counter, will be applied to some in-line monitoring problems.

Discussions were held to outline another problem regarding the measurement of $Am^{241}$ distribution in a separations column. It appears that multiple scintillation detectors and spectrometry techniques can be utilized to solve the problem.

METALLURGY - Nondestructive Testing

N Fuels Testing

1. Clad Thinning Test

Development is continuing on an ultrasonic test to detect clad thinning over the brazed regions of N Reactor fuels (dog-bone test). Following laboratory studies of critical angle response to clad thickness changes, a prototype probe assembly was fabricated. The technique consists of directing the ultrasound at an incident angle approximately equal to the Rayleigh angle in order to generate boundary waves. Changes in the metal characteristics result in beam displacements which can be detected by proper location of the receiving transducer. The effective depth of penetration can be conveniently controlled by selection of the ultrasound frequency. A prototype probe was constructed using 2 Mc $BaTiO_3$ transducers; preliminary results were encouraging. A controlled test on production fuels will next be performed.



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2. End Cap Thickness Test

The prototype end cap thickness tester is in continued use as a monitor of manufacturing process. A proposal for the design and construction of a production model is being delayed pending the delivery of necessary laboratory equipment.

3. Bond Test - Lithium-Aluminum Target Elements

A prototype ultrasonic tester for detecting the bond quality of lithium-aluminum target elements has been returned to the laboratory for minor circuit revisions. Upon completion of this work, the unit will be set up in the 306 Building for use by Physical Testing personnel during initial fuel production runs.

4. Braze Contamination Test

An initial evaluation of the X-ray fluorescence tester, designed to detect uranium contamination in the end cap braze closure, has been completed. Analysis of available data indicates that the test is incapable of production testing in the range of 0 to 3% contamination. Additional laboratory work will be required to improve the performance characteristics of this system. Modifications under consideration include: (a) Changes in the geometry of the pickup aperture, (b) replacing the X-ray unit with an isotope source, (c) closer source to target coupling, and (d) increased detector size.

5. Surface Contamination Test

An improved system to detect uranium contamination on finished fuel surfaces has been designed. Fabrication of the mechanical portion has begun and is about 5% complete. The electronic portions of this unit have been designed, fabricated, and partially checked out in the laboratory. The remainder of the checkout will be completed as soon as the mechanical portions are available.

6. Enrichment Test

A Hanford-developed tester which monitors the percentage enrichment of as-received uranium billets has been in use for some time by N Fuels Quality Control. The operating and maintenance manual is being written and a spare test head is being fabricated on site. The single channel analyzer is being modified by the addition of a switch and two reference circuits to simplify its operation.

7. Irradiated Fuel Test

Design of the mechanical portions of the remotely operated, irradiated fuel, testing station (UT-10B) for use in the 105-N fuel examination basin has been completed and fabrication of this unit has begun. Incompatibilities between the basin structural design and the support requirements of the underwater fuel handling system were brought to the attention of NRD Projects personnel, and necessary structural design changes are being made to resolve this problem. Upgrading of the electronic equipment, to be transferred from 105-KE to 105-N as part of this project, is about 75% complete.

Destructive examination of NIT fuel #1093-12 in Radiometallurgy confirmed the presence of cracks detected ultrasonically in the 105-KE fuel examination facility. One radial crack extending through the entire wall thickness was observed along with several others which extended through about one-third of the wall thickness. The latter defects emanated from the inside surface in a radial direction. One circumferentially oriented crack was also noted. The cause of these cracks is unknown at this time.

IPD Fuels - Tester Development

1. Improved Bond Test

Preliminary development work has been completed on a prototype bond tester to monitor the internal bond quality of AlSi fuels from the outer surface. If successfully applied on the production line, this approach could eliminate the troublesome internal probe. A prototype unit is being developed, and production line evaluation tests are scheduled to commence on April 1, 1965.

2. Thorium Element - Surface Contamination Test

An alpha counting system to monitor surface contamination on a special thorium target fuel was discussed with several IPD personnel. The basic requirements of this system are quite similar to those of the present contamination monitoring system being used routinely by N Fuels. Design of the new unit is scheduled to begin during January, 1965.

3. Warp Correlation Test

The SORT tester, used to measure the mechanical vibration resonance of AlSi fuels for correlation with warp data, has been upgraded and



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the electronic portion repackaged. The unit has been checked out and is ready for delivery to AlSi Fuels. An operating and maintenance manual has been prepared.

## NEUTRON CROSS SECTION PROGRAM - 02/04 PROGRAM

### Triple-Axis Spectrometer

The study of systematic effects in the determination of absolute differential-scattering cross sections has been continued. This study has included a detailed determination of the resolution elements of the monochromating and analyzing spectrometers. Experimental determinations have also been made of the dependence of the analyzing spectrometer on the spatial distribution of neutrons incident on a scattering sample. A theoretical analysis of these effects is in progress. Experiments are being continued to define the absolute scattering cross section of neutrons from $H_2O$.

### Time-of-Flight Spectroscopy for Slow Neutrons

The TOF spectrometer has been reassembled with a copper crystal for the rotating-crystal monochromator. Experiments were made to determine the resolution elements of the spectrometer using the Cu [$\bar{2}20$] reflections. The measured resolution elements are more favorable than those obtained with the aluminum monochromator used in the first test runs. The phased mechanical-chopper with a newly-designed slit system has now been reassembled and installed. Measurements were completed to determine the relative efficiencies of the scattered-neutron detectors.

### Fast-Neutron Cross Sections

The results of the 3- to 15-MeV total cross-section measurements made during September were analyzed. This analysis revealed that the data obtained in the energy region of about 8- to 10-MeV were not valid due to a malfunction in the TOF analyzer. These measurements were repeated in a measurement series during the month. Analysis of these results indicates that the anomalous effect was reduced but still present. A sample of $Tc^{99}$ was obtained and prepared for future measurement.

## REACTOR DEVELOPMENT - 04 PROGRAM

## PLUTONIUM RECYCLE PROGRAM

### PRTR Burnup Experiments

The recalculation of plutonium cross section ratios is continuing



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$\left( \text{e.g., } \frac{\sigma_c^{241}}{\sigma_a^{239}}, \frac{\sigma_a^{241}}{\sigma_a^{239}}, \text{ and } \frac{\sigma_c^{239}}{\sigma_a^{239}} \right)$ in an effort to exhaust all possible methods of calculation. The latest method is based on integration of polynomials which are constructed as least square fits of the experimental data. The polynomials have been fit as functions of N-239 for N-240, N-241, N-242, $\frac{N\text{-}241}{N\text{-}239}$, $\frac{N\text{-}241}{N\text{-}239}$, and $\frac{N\text{-}242}{N\text{-}239}$. The results are to be included in a formal report now being prepared to be submitted to Nuclear Science & Engineering.

Experimental data are now being received from the dissolution of element 5213, a 1 w/o $Pu_2$-$UO_2$ fuel element.

## $PuO_2$-$UO_2$ Graphite Lattice Studies

The initial experiment using 0.5" diameter zirconium-clad $PuO_2$-$UO_2$ rods has been completed in the PCTR. The fuel contained 0.9 w/o $PuO_2$ with a Pu-240 content of 7.2 a/o, and was arranged in 19-rod clusters. Measurements were taken on a 3 x 3 cell array in the PCTR with an 8-3/8" square pitch lattice.

The goals of the experiment were to determine the mass of copper required to poison a unit cell in an infinite lattice such that the neutron multiplication factor of the cell is unity, and to collect sufficient flux data for both the poisoned and unpoisoned infinite lattices to determine the average absorption rates in all cell components, and to determine the cadmium ratio ($R_{cd}$) in all cell components.

Flux spectrum adjustments to obtain a flux match, i.e., equivalent ($R_{cd}$) at corresponding cell positions throughout the lattice were made using combinations of Pu-Al and natural uranium rods outside the 3 x 3 test array. Reactivity measurements taken during the flux matching process indicate that the mass of copper required for unity multiplication is insensitive to the cadmium ratio in the test cell. The mass of copper required for the test cell in the infinite lattice was measured to be 1148 grams.

Four irradiations were performed to obtain bare and cadmium covered foil activation data for the poisoned and unpoisoned infinite lattices. Detector materials used were Cu, Au, Pu, and U-235. Radial traverses were taken in both poisoned and unpoisoned lattices through the fuel and graphite. A three-dimensional traverse of the flux in the poison was taken using 0.443 inch square copper foils in the poisoned infinite lattice. All the foil activation measurements have been completed using scintillation counting techniques, and the activities are currently being analyzed using the APDAC I computer code.



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PRCF - 1.8% Pu-Al Fuel, 0.80" Lattice

A tape recording of reactor noise was made for 30 minutes at a power level of 40 watts. The recorded signals were obtained from a $BF_3$ ion chamber positioned near the edge of the core on the top template.

The drop method was used to measure the worth of safety and control elements. All drops were made from a nominal 40 watt level. This level was held for sufficient time to allow the delayed neutrons to reach equilibrium. The changes of neutron flux were recorded by a digital voltmeter at five second intervals. Reactivity worths obtained are:

| | | | | |
|---|---|---|---|---|
| Control Rod #3 | | = | 15.2 | mk |
| 3 C.R. | together | = | 57.1 | mk |
| Safety Sheet #1 | | = | 4.46 | mk |
| 3 S.S. | together | = | 15.0 | mk |
| 3 C.R. and 3 S.S. | together | = | 76 ± 3 | mk |

Control Rod #3 was calibrated accurately in the region from 100% to 72% out by reactor period measurements. Three fuel rod worths were measured and added to give the required excess reactivity. The measured worth over this range is 1.147 mk.

Preparations were completed for neutron spectrum measurements using lutetium.

Radial traverses of film void and tube void were made from the center of the core to beyond the edge of the fuel. These voids extended from a point at the bottom of the fuel elements to beyond the top level of the moderator. The film void consisted of an annular ring of air, 0.030" thick, with the inside edge formed by the fuel element cladding. The outer edge was formed by a lucite thimble with a wall thickness of 1/16". The tube void was the same as the film void with the fuel element removed. This formed a void 5/8" in diameter. The worth of these voids were the same in the center of the core (-0.071 mk). The worth of the film void approached zero as it was moved along the radius. The worth of the tube void increased (negatively) to a maximum of -0.22 mk at a point 70% of the radius from the center. The worth then approached zero as the void was moved beyond the fuel elements.

A traverse of the worth of a fuel element replacing $H_2O$ and another with 0.030" void around the fuel element were made. A fuel element with 6% Pu-240 was used in these traverses. This fuel element was worth slightly more (~1%) when loaded in a void tube than when it displaced $H_2O$ in the center of the lattice. However, at a point, 84% of the core radius from



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the center, the fuel element in the void is worth 2.8 times what it is worth when displaced by $H_2O$.

This experiment is now approximately one-half finished. There is enough fuel available to reduce the moderator level to below the top template. It is not certain that this is enough to allow the moderator to be lowered to the authorized limit derived from safety considerations (15 mk).

Two rods of $HfO_2$ in aluminum, containing 0.54 g/cm$^3$, and 1.08 g/cm$^3$ of $HfO_2$, have been fabricated. Radiographical inspection reveals no noticeable variation of density along the length of the rods. Reactivity measurements of samples taken from the ends of the rods indicate that the concentration of hafnium is uniform throughout the rods to within 0.5%.

PRCF Spectrum Measurements

Measurements of the departure of the epithermal spectrum from the $\frac{1}{E}$ form has been measured for the EBWR core in the PRCF. The measurements were made using gold and palladium foils at the center of the core, 10-1/4" above the center. Assuming the epithermal spectrum has the form $\frac{1}{E(1+\beta)}$, values of $\beta$ equal to 0.018, 0.017, and 0.010, respectively, were measured. A rough draft of a report has been prepared to describe this measurement.

Data obtained from lutetium foils irradiated in the EBWR core of the PRCF have been analyzed. Calculations of resonance integrals for Lu-175 and Lu-177 have begun. The results of the calculations will be used to correct the spectral index measurements for a depletion of neutrons with energies near the Pu-239 resonance.

Kinetics of Plutonium-Fueled Cores

The reactor noise from the fourth measurement on the EBWR - $H_2O$ core, and a 1.8 w/o Pu-Al core in the PRCF have been extracted from a tape recording of the power noise for each core. The data at this stage of the analysis are the time averages of the signal power analyzed at given frequencies. The average power vs. the corresponding frequency is the locus of a transfer function. The best fit of the analytical transfer function to these data will provide a value of $\beta/\ell$, for the reactor loading.

A boron arrow was inserted into a 1.8 w/o Pu-Al loading in the PRCF to determine the ratio $\beta/\ell$ from a step change in reactivity. Data which describes the resulting flux transient have been recorded for the first two seconds after the insertion. The parameters of a series of two exponentials which best fit the data have been determined using a subroutine written



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for program GLEX. These parameters will yield the ratio $\beta/\ell$. An attempt is being made to determine whether a sum of three or four exponentials will better fit the data.

Code Development

1. HFNTOO

A user's report that HFNTOO "true" power density differed from that computed by the old HFN decks in use for several years triggered an investigation of the relationship between the old binary deck and the only available symbolics, from which HFNTOO was compiled. From comparison of core dumps from the two versions, it was determined that differences between symbolic and binary decks existed in four subroutines. Perusal of the symbolic listings led to the discovery of a logical error in one of the symbolic routines, introduced at the time the power option was added. Correction of this error eliminated the original problem. Further study led to the observation that the "true" power density option is actually non-existent in the HFN binary decks which have been in use for several years. Still pending is the question of the influence exerted by the three other non-corresponding routines, none of which have had any effect in test cases thus far run.

2. ZODIAC2

The CMBOZ2 option, which utilizes LISTIN input for TEMPEST and GAM, is now operative. With the inclusion of this feature and the latest HFN changes, the ZODIAC2 tape is pronounced ready for general use. Some minor polishing, which will not affect the working of the code, is still scheduled.

A final ZODIAC2 descriptive document has been released, superseding all previous documentation. This report gives a more adequate description of the chain's operation and provides comprehensive input instructions for all codes on the chain, including those elsewhere described.

3. ZODIAC3

The advent of Fortran IV will necessitate revision of nearly all utility routines now in use. Since these must be available for future codes whether or not ZODIAC is rewritten, preliminary work is being done in this area while approval is pending on the ZODIAC3 proposal.



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4. Two-Dimensional Multi-Energy Transport Code

The LASL two-dimensional transport code, DDK, has been given further tests. So far the package seems to be reliable in regards to its x,y (slab) and r,z (cylinder) options. The $(r,\theta)$ cylinder option has yet to be run successfully.

5. Composite Library Tape

Revisions have been made to the TEMPEST and SIGMA-3 libraries to utilize currently available cross section data. These libraries have been placed on a CLT and are available for PHYSICS CHAIN and ZODIAC CHAIN runs. It should be noted that these two libraries recognize ID number 380, as the TEMPEST isotope Unit M, instead of the old ID number 300.

Modified 710 Reactor Study

Reactor physics studies concerning the use of Pu-239 or U-233 in the 710 Reactor are under way.

Two transport theory codes, the S-N code and LASL's DDK code, have been successfully run on test problems. The one-dimensional S-N code will be used for most of the burnup and reactivity coefficient calculation. The two-dimensional DDK code will only be used for special problems.

In addition to the statics calculations, preliminary kinetics calculations (accident analysis) were also started. Comparison of U-235, U-233, and Pu-239-fueled reactors were made.

For large reactivity ramps, and for very large reactivity steps, the energy release is roughly independent of the delayed neutron fraction. A large delay fraction is of most value from the viewpoint of making prompt criticality more difficult to achieve. This is to say, a given reactivity step could cause prompt criticality in a plutonium reactor, but only a delayed critical transient in a U-235 reactor. Similarly, for a reactivity ramp, a large delayed neutron fraction allows the control system more time to act before the systems become prompt critical. However, once the reactivity is appreciably over one dollar, the value of $\beta$ is of little significance.

Shutdown Characteristics of U-233, U-235, and Pu-239 Propulsion Reactors

To minimize post-shutdown cooling problems in nuclear rocket reactors, it is advantageous to be able to reduce the reactor power as rapidly as



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possible. A brief investigation was carried out to compare shutdown rates in U-233, U-235, and Pu-239 reactors for specified reactivity variations.

All kinetic calculations were performed using a lumped parameter neutronics model with six groups of delayed neutrons. Only short term shutdown was considered ($0 \leq t_o \leq 60$ sec.), hence, the effect of fission product heating was neglected.

The power density as a function of time during shutdown was computed for representative U-233, U-235, and Pu-239-fueled reactors. Several reactivity variations were considered. For a two-second shutdown with a maximum reactivity decrement of 0.02, for example, the power decreases rapidly during the first two seconds, then slowly decays in an exponential manner. The shutdown power density in the Pu-239 reactor is roughly a factor of three less than that in the U-235 reactor because the delayed neutron fraction of Pu is only one-third that of U-235. It is interesting to note that even though U-233 and Pu-239 have approximately the same total delayed neutron fraction, the group distributions are significantly different. The net result is that the exponential power decay in a U-233 reactor is not as fast as that in a Pu-239 reactor.

From the viewpoint of shutdown cooling, the energy release, E, is of more importance than the power density. Tables I and II give the normalized energy release

$$E = \frac{1}{P(o)} \int_o^t P(t')dt'$$

for two and five seconds shutdown, respectively. The integral is taken over the first sixty seconds after the initiation of shutdown. The most important point to note from these tables is that the fractional energy saving, $\frac{E^{25}-E^{49}}{E^{25}}$, is approximately independent of both shutdown time and shutdown magnitude.



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Table I

Normalized Energy Release for a Two Second Shutdown

| ρ | (1) U-235 | (2) U-233 | (3) Pu-239 | (4) Saving Col.(1)-Col.(3) | (5) % Saving $100 \frac{Col.(4)}{Col.(1)}$ |
|---|---|---|---|---|---|
| -.01 | 6.955 | 4.077 | 2.987 | 3.968 | 57.0 |
| -.02 | 3.823 | 2.248 | 1.670 | 2.153 | 56.3 |
| -.04 | 2.099 | 1.249 | 0.947 | 1.152 | 54.9 |

Table II

Normalized Energy Release for a Five Second Shutdown

| ρ | (1) U-235 | (2) U-233 | (3) Pu-239 | (4) Saving Col.(1)-Col.(3) | (5) % Saving $100 \frac{Col.(4)}{Col.(1)}$ |
|---|---|---|---|---|---|
| -.01 | 7.867 | 4.786 | 3.629 | 4.238 | 53.9 |
| -.02 | 4.548 | 2.771 | 2.132 | 2.416 | 53.1 |
| -.04 | 2.633 | 1.616 | 1.268 | 1.365 | 51.8 |

Plutonium reactors appear to require significantly less propellant for after-cooling than U-235 reactors. Whether this propellant savings is of any practical significance, depends strongly upon the mission. For missions involving many startups, such as satellite inspection or interception, it appears that the propellant savings would be appreciable.

## Isotopic Analysis of PRTR Samples

Isotopic analyses were performed on 17 plutonium samples of PRTR-irradiated fuel elements. Of these, 15 were macrodrill samples from Al-Ni-Pu fuel element number 5103, and 2 were burnup samples from $PuO_2$-$UO_2$ fuel element number 5187.

## Instrumentation and System Studies

Development work was initiated on a reactor time-to-power instrument, which will combine neutron flux and reactor period data into a single meter reading to continuously and instantaneously indicate the proper



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control action to be taken by the reactor operator. The resulting signal could also be used for the automatic control of reactor startup. An experimental circuit was designed to include the use of analog multipliers, and tests of several multipliers will be initiated when they are obtained. It appears the most difficult segment of the experimental work will be that of obtaining compatible period and log-N signal information from the reactor instrumentation.

Experiments were conducted on the PRTR underwater gamma scanner, which is used to measure relative fuel burnup along fuel rods and to measure relative activity along irradiated wires in the development of reactor fuels. The experiments determined that the loss of instrument stability was caused by inadequate regulation of the power supplies. Modifications were incorporated and the system drifts fell into acceptable limits after a one-hour system warmup. Further tests on the detector were outlined. In addition, the detector shielding was modified to permit use of $Co^{60}$, $Cs^{137}$, and $Am^{241}$ calibration sources in an accurate and reproducible fashion. Fabrication was initiated on a signal blocking mechanism which will close the aperture in the lower collimator shielding. This will permit determination of the shield-scattered photons.

A meeting with members of Test Reactor and Auxiliaries Operation, HL, firmed details for the forthcoming test on the PRTR automatic controller in the period and startup modes of operation. The test will incorporate one of the large analog computers in the 3707-C Building to control the moderator level control system at the reactor through telephone lines. A new pressure transducer for moderator level indication was installed for use in the test. The transducer is connected to a line with 3-15 psi pressure and will provide a signal input proportional to the moderator level.

An ultrasonic test has been developed to determine the location of transition regions between as-extruded and annealed grain structures along the length of PRTR process tubes. This application consists of directing 5 Mc ultrasonic shear waves along the longitudinal axis of the tube. Since the test is intended for use on process tubes in the PRTR basin, a follower type transducer assembly was designed. Test technique and preliminary results have been documented.

## PHOENIX FUEL PROGRAM

One of the possible disadvantages of the MTR as a test bed for a Phoenix experiment is its low operating temperature. From a physics point of view, this low temperature may not necessarily be a shortcoming, since the reactivity time curves ("hot" and "cold") do not differ markedly for the MTR

geometry. The "hot" and "cold" cores studied are highly idealized. If proper account of plate thickness is taken, the reactivity swings are much more pronounced.

A survey of existing methods for treating neutron absorption in materials with granular structure ($PuO_2$-BeO systems, for example) has been completed. Recommendations for future work on this problem have been made.

Analysis of the recent U-Al/polyethylene experiment is essentially complete; a report on the experiment will appear in the next Physics Research Quarterly Report. The critical system ($k_\infty = 1$) contained the following:

| Isotope or Element | Concentration (atom/barn-cm) |
|---|---|
| U-235 | $2.36 \times 10^{-4}$ |
| U-238 | $1.61 \times 10^{-5}$ |
| H | $5.05 \times 10^{-2}$ |
| $B^{10}$ | $2.80 \times 10^{-5}$ |
| $B^{11}$ | $1.15 \times 10^{-4}$ |
| Al | $1.08 \times 10^{-2}$ |
| C | $2.52 \times 10^{-2}$ |

Cadmium ratios for 0.0005" Au and 0.002" Mn were 1.69 and 5.36, respectively. The determination of the critical boron concentration was found to be relatively insensitive to variations in the neutron spectrum; an 11% reduction in the boron concentration produced a 7% increase in cadmium ratio, but only a 0.9% change in the predicted critical boron concentration.

The first of the Pu-Al/polyethylene experiments is scheduled to begin in the PCTR during late November.

EBWR PROGRAM

Calculations using the unit cell model which assumes the thermal neutron flux distribution to be separable in space and energy have been made for the EBWR-fueled approach-to-critical lattices. The calculated lattice parameters using this model were then compared to the values calculated using THERMOS. For the space-energy separable model, the energy variation of the thermal flux was calculated using the Wigner-Wilkins spectrum in TEMPEST. Average one-group cross sections taken from the spectral calculation were utilized in a one-dimensional S4 calculation to determine the spatial variation. The S4 derived flux depression factors ($\phi_{mod}/\phi_{cell}$, $\phi_{clad}/\phi_{cell}$, $\phi_{fuel}/\phi_{cell}$) were then used as density weighting factors in the spectral calculation. The process was continued in an iterative



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manner until the flux depression factors did not change from the previous case (i.e., converged). A comparison of the $S^4$-Wigner-Wilkins ($S^4$-WW) with THERMOS calculated thermal parameters $\bar{\eta}f$ shows the $S^4$-WW $\bar{\eta}$'s and f's both to be higher than the THERMOS values. The $S^4$-WW calculated $\bar{\eta}f$'s range from 0.35% to 1.35% higher than THERMOS values going from tight to loose lattices (0.55 to 0.93 inches, respectively). Average one-group cross sections from THERMOS were also utilized in an $S^4$ calculation, and the resulting thermal utilizations differed by at most 0.15% (higher) when compared to the THERMOS multi-group values. Hence, THERMOS multi-group and $S^4$ one-group calculations yield near identical solutions to the transport equation. This correction due to mathematical differences in solving the transport equation when applied to the above calculations of f, leaves the $S^4$-WW values higher by 0.2 to 0.4%. This is contrary to uranium-fueled systems, where the calculation of $\bar{\eta}$ does not affect f, because of the 1/v cross section variation, the accurate computation of the thermal utilization for plutonium systems is dependent on the accurate computation of the energy variation of the thermal neutron flux.

## HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

The initial run with the HTLTR mockup has been started. The system contains four graphite heating elements mounted in a graphite stack that is surrounded by insulation and an external gas envelope. The insulation, stack, and heaters are in an arrangement similar to that planned for the HTLTR. Before raising the system to high temperature it is being cleaned of most of the gaseous contaminants held in the graphite and insulation. This is being done by a combination of evacuation, gas circulation, and heating up to 500°F. Throughout this process the composition of the effluent from the system is being measured periodically. This information will be of value as a check on the method that has already been used for predicting gas contaminant levels to be expected in the HTLTR. Samples of graphite and other materials have been placed at various locations in the stack. A device for inserting and removing samples of ceramic and metallic materials has been constructed. It will be mounted on the mockup prior to the start of high temperature operation.

The component parts of the prototype control rod have been constructed and assembled. Graphite cylinders are at present being used in place of the $UO_2$-graphite matrix fuel cylinders. The fuel cylinders will be installed when they become available in November. The control rod is being coupled to the rod actuator system in preparation for room temperature operational tests.

A pair of $UO_2$-graphite matrix cylinders, of the dimensions required for the HTLTR control rods, have been fabricated. The density of contained $UO_2$ is



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3.0 g $cm^{-3}$. The compressive strengths of samples cut from earlier batches of the matrix material indicate the strength at 1000°C to be about that of graphite (6000 psi). Samples for further strength tests have been cut from the most recently prepared matrix material. Off-site preparations have indicated that strengths 2 to 3 times that of graphite can be achieved.

## FUEL CYCLE ANALYSIS PROGRAM

### Plutonium Value Studies

At request of Washington-AEC, a study to determine plutonium values under private ownership conditions is under way to be completed early in November 1964. Four reactor types and a broad set of conditions are to be examined. About 60 hours of 7090 computer time are involved.

The four reactor types to be used are simulations of the boiling water reactor, pressurized water reactor, heavy water reactor, and sodium graphite reactors. The conditions to be examined are shown in Table I.

In the self-produced recycle scheme, only the plutonium produced in the fuel is recycled, and in the 50 percent self-produced recycle scheme, 50 percent of the plutonium produced in the fuel is recycled. (The balance is sold.) Unlimited plutonium recycle refers to recycle of self-produced plutonium in natural uranium enriched with any amount of plutonium required. Meeting the early delivery date required for this study requires by-passing use of several advanced analytical methods and examination of more optimum fueling systems that were to be developed during this fiscal year. An effort is being made to estimate the impact of the following modifications when they are applied: optimum lattice spacing; fuel density selection; and advanced analysis using RBU, THERMOS, and other codes.

### A Comparison of Bred Fuel in a Pressurized Water Reactor With Optimized Lattices

For every fueling system (such as plutonium in natural uranium, $U^{233}$ in thorium), there exists a degree of moderation for minimum fuel cost. This degree of moderation is determined by the nuclear characteristics of the fuel and the economic situation which includes fuel price, interest rate, fabrication cost, separation cost, and other items. Since for any given lattice spacing there is an optimum initial fissile enrichment which will produce a minimum fuel cost, selection of an optimum lattice requires: (1) Choosing a number of lattice spacings, (2) determining the minimum fuel cost enrichment for each, and (3) selecting the lattice spacing with the least fuel cost.



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TABLE I

PLUTONIUM VALUE STUDY ECONOMIC CONDITIONS

| | | Interest Rates | | | | Uranium Price Schedule $/kg ($30 Separative Duty) | | Fabrication Charges $/kg | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | 6 | 8 | 10 | 12 | 17.30 | 23.50 | 125 | 90 | 50 | 100 |
| Self-produced recycle | BWR | x | x | x | x | x | x | | x | | |
| | PWR | x | x | x | x | x | x | x | | | |
| | SGR | x | x | x | x | x | x | | | | x |
| | HWR | x | x | x | x | x | x | | | x | |
| 50 percent Self-produced recycle | BWR | | | x | | x | x | | x | | |
| | PWR | | | x | | x | x | x | | | |
| | SGR | | | x | | x | x | | | | x |
| | HWR | | | x | | x | x | | | x | |
| Unlimited plutonium recycle | BWR | | | x | | x | x | | x | | |
| | PWR | | | x | | x | x | x | | | |
| | SGR | | | x | | x | x | | | | x |
| | HWR | | | x | | x | x | | | x | |





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Figure 1 shows the result of analysis for four different fueling systems utilizing bred fuels and a slightly enriched uranium case for reference. The reactor simulation used in this study has 3/8-inch OD, zirconium clad rods with varying amounts of water which change the degree of moderation.

Table II shows the sensitivity of the moderator index to the amount of surrounding water.

TABLE II

MODERATOR INDEXES FOR VARIOUS AMOUNTS OF WATER

| Moderator Index (SDPV) | Radius of Water Region (cm) | Volume Water/Volume Fuel |
|---|---|---|
| 1.0 | 0.65 | 0.75 |
| 2.0 | 0.84 | 1.50 |
| 3.0 | 0.93 | 2.25 |

From the data shown in Figure 1, it appears that, under these economic conditions, the following conclusions can be drawn:

1. Natural uranium which contains some fissionable material is a better fertile fuel than thorium when used in fueling systems with either plutonium or $U^{233}$ enrichment.

2. The natural uranium fueling systems are more sensitive to the moderator index. This is mainly due to the effect of spectral hardening causing an increase in the alpha of $Pu^{239}$, although this effect may be somewhat overstated by the present computing system which usually yields higher alpha values than other computing systems (RBU) when the moderator index (SDPV) is less than about 1.5.

3. Although the $Pu^{239}$ alpha effect exists, systems using $U^{238}$ in a fertile fuel tend to optimize with less moderation than with the thorium system due to the fast effect available in the uranium system. Table III shows the $U^{238}$ fast effect ($\epsilon$) versus moderator index.

4. Plutonium (76% $Pu^{239}$, 18% $Pu^{240}$, 5% $Pu^{241}$, and 1% $Pu^{242}$) priced at \$10/gram for $Pu^{239}$ and $Pu^{241}$ is a less costly fuel than a comparable $U^{233}$ sequence (76% $U^{233}$, 18% $U^{234}$, 5% $U^{235}$, and 1% $U^{236}$) priced at \$14/gram of $U^{233}$ and \$12/gram $U^{235}$.



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MINIMUM FUEL COST AS A FUNCTION OF MODERATOR INDEX FOR FIVE FUELING SCHEMES USING STANDARD AEC FUEL PRICES

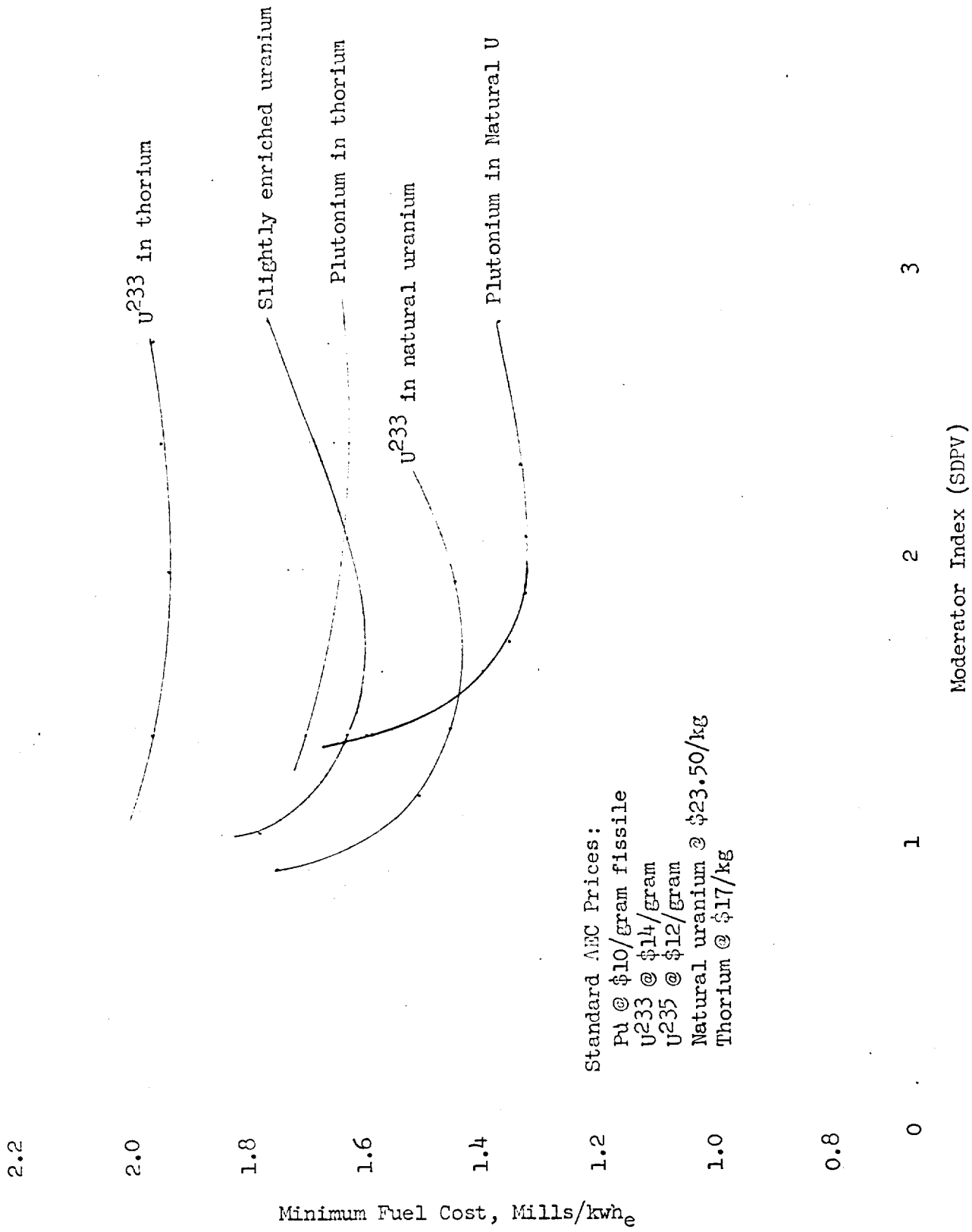




FIGURE 1

TABLE III

FAST EFFECT VERSUS MODERATOR INDEX

| Moderator Index (SDPV) | Fast Effect ($\epsilon$) |
|---|---|
| 1.0 | 1.068 |
| 1.5 | 1.053 |
| 2.0 | 1.044 |
| 3.0 | 1.032 |

5. Uranium-233 is much less sensitive than is plutonium to moderator index which allows the $U^{233}$ in natural uranium to take special advantage of the $U^{238}$ fast effect.

MELEAGER Calibration

Monte Carlo calculations in the RBU code have been used to calculate effective cross sections of $Pu^{239}$ as a function of concentration and geometry. Enrichments of 1 percent and 4 percent in natural uranium were compared in two configurations. In one case the $PuO_2$ was uniformly dispersed in the $UO_2$ while in the second case it was in an annulus equal to 1.3% or 5.2% of the total fuel volume. The plutonium isotopic composition was 60% $Pu^{239}$, 18% $Pu^{240}$, 17% $Pu^{241}$, and 5% $Pu^{242}$. The fuel element was 0.354-inch OD on 0.456-inch-square pitch. The moderator region contained a mixture of iron at 0.93 gram/cc and water at 0.75 gram/cc. Preliminary results are presented in Table IV representing 30 minutes each of 7090 green light time in Monte Carlo. Convergence was poor and each case has been given an additional 30 minutes of 7094 green light time, but the results have not yet been analyzed. Comparable calculations have been made in THERMOS and JASON. These values and the resultant values from MELEAGER, based on normal JASON input, are also included in Table IV. Further comparisons with other codes and experimental data will be attempted in order to determine the proper corrections for MELEAGER since MELEAGER appears to be far too pessimistic.



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TABLE IV

ESTIMATES OF ETA FOR PLUTONIUM$^{239}$ IN HARD SPECTRA

| Computer Code | 1% Uniform | 1% Annulus | 4% Uniform | 4% Annulus |
|---|---|---|---|---|
| RBU | 1.926 | 1.880 | 1.903 | 1.896 |
| THERMOS | 1.948 | | 1.923 | 1.910 |
| MELEAGER | 1.839 | | 1.753 | |

Cross Section Library

Cross section data have been derived from a recent version of the RBU Basic Library using the GANDS program to make up three cross section data tapes for various temperature ranges for use in the MELEAGER Economics Chain system. The two higher temperature tapes permit calculations pertinent to reactors operating at higher temperatures than was previously possible. A summary of the data is presented in Table V. The error in G is the deviation in the thermal-group cross section as used by MELEAGER from that represented by the data on the RBU tape. The error in S + bG is the deviation of the infinite dilution resonance integral as used by MELEAGER from that represented by the data on the RBU tape. The table is for illustrative purposes only because, in MELEAGER, the resonance integral cutoff energy is a function of the neutron temperature which, in turn, is dependent (among other things) upon fuel concentrations which change throughout the burn up.

FORV Catalog

A catalog (designated FORV) of reactor fuel costs and fuel cost curves is being compiled for AEC-Washington. The catalog containing 6400 fuel costs and 1600 curves, covers 20 price schedules and various fabrication, separation, and interest charges for five simulated reactor types. Plotting routines have been debugged and the program is in production.

Code Development

1. PROTEUS

The PROTEUS code (curve fits the output from the MELEAGER burnup code to the proper form for the QUICK economics code) has been revised to accurately treat private ownership of nuclear fuel. As originally written, all of the accounting information necessary to calculate fuel costs according to the present AEC regulations was written at 6-month



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TABLE V

GANDS DATA FOR MELEAGER

| Isotope | | 2200 m/s $\sigma_a$ / $\sigma_f$ | Resonance Integral with 0.45 ev Cutoff | Largest Percent Error of Fitting 0°-600° G | 0°-600° S + bG | 500°-1100° G | 500°-1100° S + bG | 1000°-1600° G | 1000°-1600° S + bG |
|---|---|---|---|---|---|---|---|---|---|
| $U^{233}$ | --- | 575 | 853 | .0004 | .006 | .0005 | .001 | .0008 | .001 |
| | | 526 | 729 | .0004 | .005 | .0005 | .001 | .0008 | .001 |
| $U^{235}$ | --- | 678 | 408 | .003 | .01 | .001 | .04 | .001 | .002 |
| | | 579 | 269 | .003 | .03 | .001 | .04 | .001 | .003 |
| $Pu^{239}$ | --- | 1008 | 581 | .23 | .10 | .01 | 2.6 | .001 | .10 |
| | | 754 | 348 | .19 | .08 | .01 | 2.8 | .001 | .12 |
| $Pu^{241}$ | --- | 1372 | 526 | .09 | .99 | .007 | .81 | .001 | .01 |
| | | 1020 | 429 | .08 | .84 | .006 | .60 | .001 | .008 |
| $Th^{232}$ | --- | 7.56 | 99 | * | .02 | * | .001 | * | .001 |
| $U^{238}$ | --- | 2.72 | 281 | .002 | .0002 | .0017 | .001 | .001 | .001 |
| $U^{234}$ | --- | 95 | 341 | .01 | .03 | .0009 | .001 | .0007 | .001 |
| $Pu^{240}$ | --- | 280 | 8440 | .02 | .02 | .18 | .001 | .002 | .001 |
| $Pa^{233}$ | --- | 47 | 586 | .003 | .02 | .003 | .002 | .003 | .001 |
| $U^{236}$ | --- | 6 | 322 | .002 | .02 | .0003 | .001 | .0008 | .001 |
| $Pu^{242}$ | --- | 19.2 | 1041 | .0003 | .02 | .0009 | .001 | .002 | .001 |
| $Np^{237}$ | --- | 169 | 1318 | .06 | .01 | .02 | .05 | .002 | 2.2 |
| $Am^{241}$ | --- | 628.5 | 1638 | .18 | .27 | .005 | 2.1 | .003 | 1.6 |
| $Am^{243}$ | --- | 138.5 | 1163 | .16 | .02 | .01 | .001 | .02 | .0009 |
| $Am^{242m}$ | --- | 7997 | 3793 | * | ** | * | ** | * | ** |
| $Cm^{242}$ | --- | 20 | 9.5 | * | ** | * | ** | * | ** |
| $Cm^{243}$ | --- | 1400 | 664 | * | ** | * | ** | * | ** |
| $Cm^{244}$ | --- | 15 | 7.1 | * | ** | * | ** | * | ** |
| $Pu^{238}$ | --- | 420 | 212 | * | .0006 | * | .00009 | * | .0001 |

* G value of 1.0 fits the data with negligible error.
** S value of 0.0 fits the data with negligible error.

intervals, and this specific time period was fixed internally in the code. Now, one can enter any specific time between accounting calculations to simulate private ownership of nuclear fuel. If no change from the normal 6-month period between accounting calculations is desired, the position on the input card designated as the accounting time period may be left blank.

Provision has been made to permit the adjustment of the amount of reactor control necessary to account for non-interaction of the neutrons between old and new fuel on the graded fuel cycle by multiplying the internally calculated control requirement by a factor. At the present time, the factor is set equal to 1, which assumes no interaction. At some later time, the ALTHAEA code will be used to determine factors to account for the degree of interaction of the neutrons in the fuel.

2. VESTA, Fuel Utilization Code

A special data set for the final debugging of the VESTA code has been devised and is being run. Simulation of large thermal reactors and advanced converter reactors operating on fractional plutonium recycle status is nearly complete for the pressurized and heavy water reactors. Work is in progress for the sodium graphite and gas cooled reactors. Two fast reactors are also being simulated: One is plutonium carbide fueled and has a 7-year doubling time, the other is plutonium oxide fueled and has a 15-year doubling time. Both are sodium cooled. These simulations are being used with the VESTA code for various projections of economic growth and reactor development rates.

NEUTRON FLUX MONITOR PROGRAM

The two uranium regenerating neutron flux detectors, which employ ionization chamber coatings of approximately 90% U-234 and 10% U-235, were installed in a test facility of the KW Reactor. Prior to installation, one chamber was carefully radiographed to determine the chamber construction. Measurements of chamber background current, due to alpha particles from the uranium coating, indicated one chamber had about three times more coating material than did the other unit. In addition, gamma sensitivity was measured to be about 70% greater than had been predicted. Since the chambers were just installed in the facility, time is now being spent in assembling the instrumentation for recording the data.

The boron-11 chamber and a number of test coaxial cables were prepared in an assembly and taken to KW Reactor in preparation for irradiation. Considerable radiographic testing of the several different cables was con-

ducted, and the tests indicated that a number of the cables had been incorrectly labeled as to the center wire size. One of the metal-sheathed cables was opened and is being pumped down in an attempt to improve the cable leakage resistance, since such leakage can cause loss of signal and added false current due to the offset (input) voltage of the associated amplifier. Gamma irradiations were conducted on several cables and on the boron-11 chamber to establish the relative merit of the cables and to aid in predicting background currents during the planned reactor tests. For a gamma ($Co^{60}$) exposure rate of $7 \times 10^4$ R/hr, the cable currents could not be measured but were estimated to be less than 1 pA. The boron-11 chamber current, at the same exposure rate, provided about 1 to 2 pA current.

Progress was achieved on the microwave method of measuring reactor in-core neutron flux with the design completed on two more gas cavities. Fabrication was partly completed on the cavities. These units should permit the determination of the mode of the signal being studied, and it is hoped that observation of the shift of one mode can be achieved, as has been successfully demonstrated with a gas discharge tube in the laboratory. No further reactor tests were undertaken.

## NONDESTRUCTIVE TESTING RESEARCH PROGRAM

### Electromagnetic Testing

Earlier methods of displaying eddy current test data super-imposed on a cross section view of tubing under test was replaced by a technique which uses a radar type position indicator with a linear radial sweep. As before, the sweep rotates in synchronism with the position of the test probe. In the newer method, the rotating sweep provides a "field" upon which the pertinent tubing thickness and flaw location information will be imposed. Data are presented by blanking and unblanking the cathode ray beam at the appropriate time by using a video signal derived from the eddy current test coil output. Although all details of the new method have not yet been worked out, the circuit has been assembled in a breadboard manner and a fabricated defect on the inner surface of a process tube sample has been successfully displayed.

A "double boiler" heating container for melting Wood's metal has been fabricated. This metal bath is used to study optimum eddy test coil design and the effect of field geometry on test results. A traversing arrangement with a one-inch stroke has been incorporated to translate the test coils through the bath. A potentiometer also was added to give horizontal sweep voltage for use in displaying the eddy current testing device output signals on a cathode ray oscilloscope.



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Several eddy current test coils have been fabricated for use in this test assembly. A mold to hold a ring of molten Wood's metal to simulate a section of tubing was designed and is now being fabricated. In addition to evaluating general tubing tests, the Wood's metal bath facility will also be used to further analyze the multiparameter tester and the recently developed dynamic display unit for viewing tubing cross sections.

Heat Transfer Testing

The surface thermal impedance of one material in contact with a semi-infinite layer of a second material has been analytically determined as a function of the thermal properties of the layers, the thermal contact conductance between them, and the frequency of the applied sinusoidal thermal wave. The analytical general solution was evaluated for a range of specific materials, frequencies, thicknesses, and contact conductances.

Graphic presentation of the computed surface impedance of a 0.1 inch thick layer of inconel in variable resistive contact with various second layer materials has been completed. For this experiment, a thermal test frequency of 1 cycle per 5.9 minutes was used. Variations in contact conductance between the two layers caused the first layer surface impedances to vary along straight line loci. An attempt was made by Applied Mathematics, HL, to reduce the analytical general solution to the form of a straight line using variations in the contact conductance as the independent variable. However, this did not appear feasible in view of the large number of terms, and the possibility of small valued high order terms in the equation. Thus, it will be necessary to complete the graphical presentations of the values that have been obtained for a number of the other specific cases to get an indication of how generally the straight line relationship holds.

Practical methods of implementing the thermal wave testing method are being developed. A thermoelectric heat pump has been obtained, and equipment is being designed for the initial experimental work.

Fundamental Ultrasonic Studies

The analytical and numerical studies on wave propagation at interfaces between acoustically dissimilar, attenuating media are complete except for minor modifications. This work is being conducted by Applied Mathematics Operation, HL. A wave model has been developed which appears to adequately explain the changes in shape of broad-frequency-band pulses propagating in media having a nonlinear relationship between attenuation and frequency. The model assumes a linear differential equation of motion and linear boundary conditions which permits the use of the super-



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position principle in obtaining a solution.

The problem of nonlinear attenuation of multifrequency pulses is solved by numerically integrating over the frequency spectrum by use of a computer program. Pulse shape changes can be predicted to the accuracy of measurements, provided the material constants are known. The program has been tested with representative pulses and attenuation coefficients and appears to behave properly.

Oblique angle shear-wave attenuation measurements on 304-L stainless steel are complete except for the reduction of data. The tests on some of the prepared samples of aluminum and Zircaloy-2 showed large variations in shear-wave propagation, especially in the as-worked samples. It is expected that heat treating will provide adequate standards to obtain the measurements necessary for the theoretical studies.

Fatigue Detection

The eddy current electronic equipment has been modified to optimize the signals obtained from three aluminum alloys. Specimens of each alloy are being measured by conventional techniques for their respective conductivities. The samples will then be used to calibrate the eddy current tester for variations in conductivity. The usual technique of monitoring the temperature gradient to make these measurements proved difficult to apply to copper and aluminum because of their high thermal diffusivities.

Further eddy current tests have been conducted on samples which had been fatigued in earlier work. These tests substantiate that the eddy current test is most sensitive to fatigue in steel and steel alloys. Some reproducible fatigue signals were observed in stainless steel but these were much less in amplitude than those obtained for steel.

To determine the mechanism of fatigue detection by eddy currents in A-212 B steel samples, the specimens were tested, demagnetized and retested a total of five times. Each time a decrease in the fatigue signal was noted with the final signal amplitude being about 80% of the original value. This information indicates that some of the fatigue signal in steels may be due to the formation of permanent magnetization.

The mechanical manipulating system for the ultrasonic fatigue detection work has been assembled and aligned. Initial experiments with this equipment have shown that the mechanical movements are sufficiently reproducible for fatigue studies. Although the mechanical system proved to be satisfactory, transducer variables were a problem. A group of twelve transducers purchased for use in fatigue studies were evaluated and only four



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to be usable for this work. Evaluation of several transducers fabricated at Hanford show qualities that are superior to those available in transducers purchased offsite. These data indicate that it might be necessary to specially fabricate transducers to meet the stringent requirements of the current tests.

Current ultrasonic critical angle tests have substantiated earlier indications that there were differences in the reflected beam null signals prior to fatigue. These were most pronounced in the 304L stainless steel samples. The null signal profile of the samples has been recorded for use in comparing null signal variations after fatigue application.

Since fatigue is believed to initiate at the sample surfaces, it was of interest to determine if conventional surface wave (Rayleigh wave) techniques may also be useful for fatigue detection. The usual procedure involves a contact test rather than immersion techniques. Fatigue destructed aluminum specimens were studied using a 5 Mc/sec Rayleigh transducer.

Although various types of couplant, transducer pressure and coupling spreading were used, the test was not successful in detecting fatigue. Further work is planned to evaluate the applicability of surface waves to this problem.

Ultrasonic Transducer Development

A mechanical system, metal test rods, and dielectrics (mica) were prepared for the evaluation of capacitor characteristics for generating and detecting ultrasonic pulses. This device consists of mica dielectrics held against each end of the rod by metal electrodes. As a voltage pulse is applied to one capacitor, the deformation in the dielectric sends an ultrasonic wave propagating in the test rod. Upon reading the opposite end, the pulse again deforms the dielectric of the second charged capacitor. This results in a net change in charge stored which is recorded as an output voltage signal.

Improvements have been made in the mechanical system and dielectric preparation such that the signal from detected pulses is well above the noise level of the electronics. Test rods of aluminum, copper, brass and stainless steel were examined by this method. Since ultrasonic pulses are generated at the end of the sample, the ultrasonic energy travels entirely within the test rod which makes the test independent of transducer characteristics. Detected pulse shapes were repeatable and were shown to be characteristic of the metal composition of the test rod. Thereby, properties such as tensile strength of metal may be examined by this method without interference of other variables. One significant advantage of this



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system is its ability to function at high temperatures. Because of this characteristic, properties of the metal rods may be examined at high temperatures.

Imaging Ultrasound

Alternate methods for imaging ultrasound were investigated in the continuation of this project. This effort has been primarily theoretical to date due to the time delay in material procurement. Some experimental work has been completed using electroluminescent phosphor in an effort to obtain some familiarization with their properties. It is anticipated that these phosphors will eventually provide a visual readout for an ultrasonic imaging system.

The difficulty in applying the previously mentioned thermoplastic recording technique to imaging ultrasound arises from the inability to convert the oscillating charge pattern of the crystal to a uni-directional force on the film. One method currently being considered which might overcome this limitation is to coat the rear face of a piezoelectric crystal with a photo-electric emitter, e.g., caesium-antimonide. By brightly illuminating this surface an abundance of electrons is created. With a closely placed control grid, the necessary diode action may be achieved. At each cycle, some charge will overcome the grid "bias" and be deposited on the film placed behind the grid. In this manner, charge will continue to accumulate on the film over a number of cycles which will increase the sensitivity of the device.

Semiconductor Ultrasonic Transducers

Experiments on piezoelectric semiconducting junction transducers were continued. Work to date has centered on the more common piezoelectric insulating materials since single crystals of the II-VI Group compounds have not yet been received from commercial sources. Lithium sulfate monohydrate, which is currently being used, is a good insulator (wide forbidden band gap). However, to produce semiconductor junction transducers, this material must be made into n or p like semiconductors while still retaining its piezoelectric properties.

Attempts to produce this condition in lithium sulfate monohydrate are being carried out along two approaches:

1. Lithium sulfate monohydrate is water soluble and can be grown from aqueous solution. Attempts will be made to grow these crystals and add various dopants to produce the desired effects. To date, "seed" crystals approximately 1/8 x 1/8 x 1/4 inch have



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been grown in this effort.

2. Various defect centers caused by radiation damage may produce the desired effect. To investigate this problem, two single crystals of $LiSO_4$ - $H_2O$ have been irradiated by $Co^{60}$ $\gamma$ sources, receiving total doses of approximately $5 \times 10^7$ and $5 \times 10^8$ R. The production of defect centers will be revealed by additional bands occurring in the optical absorption spectrum of these crystals.

Single crystal $LiSO_4$-$H_2O$ and $BaTiO_3$ have been received from commercial sources and are being investigated for their piezoelectric, electrical and optical absorption properties.

Nondestructive Testing of Isotope Heat Sources

Further evaluation of ultrasonic inspection for measuring wall thickness revealed that both the cladding and core material rapidly attenuate ultrasound. Since the attenuation increases as the frequency increases, a lower frequency signal must be used to adequately penetrate the cladding.

Since commercially available transducers are not sufficiently damped at the frequency required for these tests, several ultrasonic transducers have been fabricated to evaluate methods of achieving highly damped transducers. Using various amounts of tungsten in the backing member, different damping factors were obtained. To determine the applicability of these transducers to heat source inspection, these transducers will both be evaluated for their ultrasonic properties and will be placed in an irradiation zone to determine the effect of radiation.

Recent thermal studies have revealed the importance of detecting cracks present in the compacted core material. In order to penetrate the core material with ultrasonic waves, transducers and equipment capable of generating and detecting 100 kc ultrasound are being developed. Once developed, this test will also yield information as to the core to cladding bond and the liquid or solid state of the core center.

The surface thermal impedance of an Inconel layer in contact with a semi-infinite $SrTiO_3$ layer has been analytically determined as a general function of thickness of the Inconel, frequency, contact conductance, and thermal properties of the $SrTiO_3$. Computer evaluation of the analytical results has provided values that are being graphically presented to obtain general conclusions about the surface thermal impedance characteristics. These conclusions are being used as guide lines in the development of a thermal testing method for isotope heat sources.



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## BIOLOGY AND MEDICINE - 06 PROGRAM

### Atmospheric Physics

Analyses were made of two atmospheric diffusion experiments in which fluorescein and zinc sulfide tracers were emitted from a common source. A technique of discrimination of these two tracers had previously been demonstrated. The objective of these experiments was to achieve generation of fluorescein particles which have aerodynamic properties similar to those of the zinc sulfide. Because fluorescein is soluble in water, the size of the particulate can be varied through controlling the concentration of the source solution or by changing the size of the slurry droplet which is generated. The results obtained in previous tests were used to determine the slurry concentration and drop size setting necessary to produce fluorescein particles of the same size as zinc sulfide.

Comparisons of the masses of the tracers collected near the ground at 200 meters, 800, 1600 and 3200 meters downwind are favorable. In the mean, the ratio of the fluorescein mass to the zinc sulfide mass was 1.07. The average value for a given arc distance ranged from 0.88 to 1.19. For individual filters, the fluorescein to zinc sulfide mass ratio varied from 0.52 to 2.01. The initial attempt to pre-select fluorescein particle size is indeed encouraging. Further improvement may be possible with finer adjustments of generation procedures.

The first quantitative verification of the calibration of the real time sampler has been made. The sampler was operated in the field detecting puffs of zinc sulfide. The data collected by the real time sampler were compared with those of the bulk filter system. These filters were exposed adjacent to the real time sampler intake. The computed concentration was within the standard error range of the RTS calibration in six of the seven trials. Comparisons of the concentrations determined from RTS data and those of the bulk filter system follow:

| A | B | |
|---|---|---|
| Concentration (gms/m$^3$) determined from filter | Concentration (gms/m$^3$) determined from RTS response | A/B |
| $4.78 \times 10^{-6}$ | $5.05 \times 10^{-6}$ | 0.95 |
| 6.44 | 7.25 | 0.89 |
| 10.5 | 10.6 | 0.99 |
| 20.7 | 13.7 | 1.51 |
| 43.9 | 36.0 | 1.22 |
| 56.7 | 45.5 | 1.24 |
| 106.0 | 84.0 | 1.26 |



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One elevated source diffusion experiment was successfully completed in unstable conditions this month. Exposures were measured near the ground out to one mile from the source. The real time sampler was also used to measure concentrations 300 meters from the source. The tri-level wind system and the new portable mast system were operational for this test.

In Air Force work, Atmospheric Physics will receive $75,000 for a 12-month period beginning April 1, 1965, for tracer development and studies of the diffusion in the atmosphere from instantaneous releases. It is expected that this support will be continued for three to five years.

Radiological Physics

Another isotope has been discovered in the Alaskam caribou. Analysis of a blood sample from a caribou produced an X-ray peak at 5.9 keV due to $Fe^{55}$. This isotope is not a fission product, so it must have been produced by neutron activation.

Whole body burdens of $Cs^{137}$ for 12 Eskimos at Anaktuvuk Pass are lower than the body burdens measured last month. This was expected. Hair samples from these 12 Eskimos are being taken and analyzed monthly.

A summary of the data obtained from the Alaskan urinalysis study at two villages shows the potassium content is not a good index of urinary excretion. The average potassium content decreased by 30 percent during the months of April and May. The cause of this reduction is not known, but may be due to a change in diet or physical activity. The average potassium concentration of monthly urine samples at Richland, Washington, for the past year has been constant at 2.34 g/liter ± 7%. Using the volume of the urine sample rather than the potassium content as an excretion index gives an average body burden estimate for the Eskimos that agrees with the whole body counting measurements made at the same time.

A Victoreen r-meter from the U of W spermatogenesis laboratory was compared with Hanford's standard r-meter. They agreed very well. A minor error in read-out procedure was uncovered and the U of W personnel have been informed so the error can be corrected.

A dog phantom was used to obtain calibration information for measuring plutonium body burdens in dogs. Several cases of interest were investigated. The standard case was one in which the dog phantom rested on its back and the plutonium source was distributed at the periphery of the lungs. The standard case was compared to (1) dog phantom rotated 180° and resting on its sternum; (2) dog weight increased by 2 kg of tissue; (3) dog weight increased by 2 kg of fat; (4) the detectors moved 2 centi-



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meters away from the standard counting position; and (5) plutonium source centrally located in the lungs. The change in counting efficiency for the above five cases was -6%, -33%, -37%, -14%, and -29%, respectively.

The positive ion accelerator performed satisfactorily during the month.

The $He^3$ neutron spectrometer was successfully tested using the anti-coincidence circuit. This circuit rejects pulses from neutrons which do not lose all of their energy inside the sensitive volume of the counter. The efficiency of the counter for 0.35 MeV neutrons was measured as $3 \times 10^{-4}$ counts per neutron per $cm^2$.

A pulse rise time discriminator built for use with proportional counters was successfully tested. The circuit is capable of rise time discrimination over a dynamic range of about 50.

The unexplained variation in the output of the thermopile between the two cells of the small calorimeter has been traced to the galvanometer amplifier. This amplifier appears to be sensitive to room temperature changes, although the room temperature is controlled to $\pm 1^{\circ}$ centrigrade. A temperature controlled box is being fabricated to hold the critical parts of the system.

A basic problem when using thermistors for sensitive temperature measurements is the heat due to the energy which must be added to the system when the thermistor is read. To reduce this heat, a pulsed read-out system was designed and built. Preliminary results indicate an order of magnitude increase in sensitivity can be achieved by refinement of this idea.

## Instrumentation and System Studies

Successful operation was achieved on the installed sliding valve mechanism used in Biology inhalation studies. A linear actuator is used to drive the cylindrical valve and thermistor transducers are employed for control. Two more valves will be assembled for future studies. In addition, field testing was continued on the improved dog masks used in smoke inhalation studies. Larger thermistors are being used to secure a more rugged system.

The organic scintillation detector, used in the measurement of gamma dose rates to fish at a Biology facility, was increased in size to improve the system sensitivity. Calibration tests are being conducted.

Discussions were held with Biology personnel regarding the development of an instrumentation method which will permit measurement of $P^{32}$ and $Zn^{65}$ uptake in algae cultures under controlled conditions.



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The Biology field analyzer and monitor, to be used in field and laboratory radionuclide measurements with experimental animals, was completed, tested, and delivered to the Biology Laboratory. Operation appears to be fully successful.

Experimental solid state circuitry development was continued on the animal physiological function and radionuclide uptake measurement and telemetry system and a number of necessary components were received. In addition, the techniques established for temperature measurement and data transmission are being considered for application to humans working under relatively high temperature conditions at Hanford. From the conducted experiments in a reactor heat exchanger, it appears that a modified system will provide the desired temperature measurement in addition to possibly supplying human pulse rate information. The measured data will be telemetered to a central receiver and recorder.

Circuit modifications were incorporated in the ion chamber peak pulse reading instrument which will be employed in radiation dosimetry studies. The modifications now permit reading accuracies to ± 1%.

The indicator lamp logic and driver circuits and a modified pulse generator tester were completed. The instrument is essentially completed and is being used in dosimetry studies on a test basis. A high-current power supply, for use with the instrument, has been designed and will be fabricated.

Circuit development work was continued regarding a single electron counter which is to be used in radiation dosimetry studies. Difficulties with the fast tunnel diode discriminator are still apparent as the pulse amplitude continues to vary with the pulse rate. An attempt will be made to develop a fast avalanche transistor circuit to circumvent the problem.

Progress was achieved on the development of a low noise, charge sensitive preamplifier using field effect transistors. Such units will be particularly useful with solid state diode radiation detectors. Fabrication was initiated on several promising designs using both a single field effect unit and two cascode-connected field effect transistors in the input circuit. Detailed tests regarding temperature effects, gain, and frequency response will be conducted on the preamplifiers when fabrication is completed.

Calibration information was provided by Atmospheric Physics Operation to update the coordinate transformation data for the wind turbulence analog computations. An additional function generator was added to the simulation with associated circuits to increase the usefulness of the simulation. The



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analog circuit was checked out and found to be operating correctly and is now ready to accept data for transformation.

Analog computer runs were completed which defined the distribution and time transients of $I^{131}$ in various parts of a human body after ingestion. The parts of the body include the lungs, stomach, small intestines, upper and lower large intestines, skin, blood, kidneys, thyroid, bone, muscle, and other organs. The transients were observed after ingestion of a single dose of $I^{131}$ and after the ingestion of multiple doses once a day and twice a day from a decaying source. In addition to defining the transients, it was found that when a biological decay constant of four hours is used for $I^{131}$ in the blood, the analog computation matches experimental results. The simulation was improved by adding a logarithmic chart readout capability which causes pure exponentials to be displayed as straight lines. The simulation is capable of handling parent and daughter radioisotopes in animals as well as in man.

## WASHINGTON DESIGNATED PROGRAM

### Isotopic Analysis Program

Isotopic analyses were provided on program samples during the month in accordance with current schedules.

Testing of the new ceramic vacuum-lock sample changer for the mass spectrometer has continued. Reworking the mechanical components has eliminated the problem of seizure due to thermal expansion caused by the heated filaments. The electrical contact system is being modified prior to testing at high voltages.

## EXPERIMENTAL REACTOR PHYSICS FACILITIES

### PCTR Operation

Operation of the PCTR continued routinely during the month. There were no unscheduled shutdowns. The experiment to measure $k_\infty$ and neutron flux distribution for the 19-element $PuO_2$-$UO_2$ cluster in an 8-3/8 inch graphite lattice was completed during the month.

### TTR Operation

The TTR was operated on an intermittent basis during the month. There was one unscheduled shutdown due to an electronic failure. One set of foils was irradiated for the Materials Testing Operation, and one set of foils was irradiated for the Radiological Development Operation. The TTR was



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operated on a one-night-a-week basis for the University of Washington Graduate Center.

Critical Approach Facility

The critical approach facility was not operated during the month.

PRCF Operation

Experiments on the PuAl core were continued. Calibration of 75¢ worth of Rod #3 was completed. Drop tests were performed to determine the worth of the safety mechanisms. A total of five drops were made; Rod #3, Sheet #2, all Rods, all Sheets, and the total system. A $D_2O$ type (shutter) shim rod was found to have no reactivity effect in the reflector.

Reactor kinetic tests were completed successfully. Two noise recordings and three Boron Arrow firings were made at a reactor power of 43 watts.

Worth measurements were made as a function of radius for fuel elements, "tube" voids and "film" voids. "Tube" voids and "film" voids were simulated by using a plexiglass thimble, first voided and then with a fuel element inside the thimble. The same fuel element was used at each radius to eliminate fuel element relative worth discrepancies. Measurements were also initiated on fuel elements containing various percentages of $Pu^{240}$.

Moderator sensitivity and worth measurements are in progress.

COMPUTER FACILITIES

The General Electric 412 process control computer was received and wired into the 321 Building solvent extraction process. System check-out went smoothly, and computer programs are now being loaded and checked out.

The analog computer utilization was as follows:

| EASE 1132 | EASE 2133 | |
|---|---|---|
| 170 | 156 | Hours Up Time |
| 0 | 16 | Hours Scheduled Down Time |
| 6 | 4 | Hours Unscheduled Down Time |
| 176 | 176 | Hours Total |



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Problems considered during the month were:

1. Columbia River flood routing.
2. Pot calciner.
3. Meteorology study.
4. Biological distribution of radioisotopes.
5. Crack propagation.

Detailed descriptions of analog computer applications have been prepared and documented in HW-84448-10. This is the first of a series of reports designed to give more background and details on various analog computer applications at Hanford.

Troubles with the IDACAS iterative control units in the new analog computer have necessitated the ordering of replacement units.

Five of the ten evening sessions of the analog computer course were completed. Course material included a comparison of analog and digital computers along with the fundamentals of analog computer programming. Twelve of the fifteen students also have solved simple problems on the Donner 3500 computer.

Continued investigation of automatic parameter optimization in dynamic systems produced good results. The bases for a generalized approach to problems of parameter optimization have been defined. Further studies are required to develop general methods of measuring parameter sensitivity. The approaches used in automatic optimization are expected to be useful in design as well as model fitting studies.

## CUSTOMER WORK

### Weather Forecasting and Meteorological Services

Meteorological and climatological services included: 1) Advisories to Radiological Chemistry on fallout sampling during aircraft flights, 2) transmission of climatological data to AEC for use by potential bidder, 3) calculations to Biology Section on cooling rates of water droplets in irrigation sprinkler systems, and 4) for Research and Engineering, IPD, an evaluation of the potential consequence of a stack release of radioactive materials.



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Weather Summary

| Type of Forecast | Number Made | % Reliability |
|---|---|---|
| 8-Hour Production | 93 | 83.8 |
| 24-Hour General | 62 | 84.7 |
| Special | 114 | 92.1 |

Temperatures during October averaged near normal. Precipitation of 0.28 inch was only one-third the normal amount.

Mass Spectrometry

Isotopic analyses were provided during the month on 8 uranium samples. Of these, 6 were in support of HAPO $U^{233}$-production studies, and 2 were in support of the Phoenix fuel program.

Instrumentation and System Studies

To assist Health and Safety, HL, initial measurements were completed to determine the conductivity of "Hi-X" fire-smothering foam. The conductivity instrument will be used to measure foam level in tests to be performed at a later date.

For Waste Solidification Engineering, HL, developmental work was initiated on a liquid level detector for use in the glass waste solidification process. The instrumentation uses a fast pulse (less than 1 nsec) propagated down a transmission line with a sampling oscilloscope measuring the time required to reflect the pulse from an impedance discontinuity. Experiments were conducted with both stainless steel and aluminum probes, with good results obtained with various probe lengths. Two prototype probes, one of 14 inch length for use in a glass furnace and the other of 8 ft. length for larger vessels, are being fabricated for direct application.

General electronic circuitry and main chassis wiring was nearly completed for the beta-detection assault mask monitor, which is being developed for use at the Laundry Facility at Hanford. Design of the dual-probe support mechanism was initiated.

Detailed laboratory testing was initiated on a commercial scintillation detector and multichannel system of Geochemical and Geophysical Research, HL. The tests were undertaken to determine the cause of minor but troublesome drift problems which occurred in the field.



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Field tests were completed, including several aerial surveys over Hanford, on the developed scintillation detection and solid state circuitry aerial gamma monitor. General operation appeared to be successful, according to Environmental Monitoring, HL.

Circuit modification work was continued on a personnel dosemeter obtained from AECL by Radiological Engineering, IPD. The required commercial components were finally received.

Two silicon controlled rectifier stacks and associated gating devices were installed in the 105 KW Reactor test facility for the swelling program instrumentation. Final wiring and testing of the circuit on a dummy capsule remain to be completed before an N Reactor test is started. The silicon controlled rectifier systems replace obsolete magnetic amplifier saturable core reactor units which were used to provide a controlled amount of power to the heaters in the test swelling capsules.

Development work continued on a method of remotely measuring the displacement of metal specimens in a high temperature oven. Several tests of an optical system indicated the need for better optical apparatus and a different optical arrangement to obtain the resolution required.

Modification and debugging of the creep data logger at 105 KW continued during the month. The micropositioner probe contact circuit and print/command circuit were modified to permit proper cycling. The manual micropositioner control and measurement system were removed and brought into the lab to be modified for installation in the PRTR. Manual control and data acquisition at the KW test facility will be accomplished with the creep data logger. The automatic operation of the creep data logger has been under test for approximately a week. Its operation was satisfactory with a dummy capsule when programmed for a logging cycle of every half hour.

Development work on a method of detecting the propagation of a crack in a process tube or tubing in a containment vessel was started. The method utilizes accelerometers mounted on the vessel at different locations. As pressure increases in the containment vessel when a crack propagates, a characteristic 40 kc signal occurs which is sensed by three or more accelerometers. The output of the accelerometers is compared and a triangulation procedure is used to locate exactly the position of the crack in the vessel. The electronic equipment and necessary transducers are being obtained for preliminary tests on zirconium process tubing.

A proposal was made to substitute a digital control and instrumentation system in place of a conventional analog control and instrumentation system



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in the High Temperature Lattice Test Reactor. Study and evaluation of the requirements for a digital system were started in an effort to have recommendations available at the earliest possible date. Both the reactor requirements and available digital systems are being studied.

Effort continued on the computer program for the Columbia River flood routine problem which involves fitting of three parameters of a mathematical model to river flow data. An automatic steepest descent method was developed for optimizing the parameters so that the model fits actual flow data. The method is general and is applicable to any optimization problem involving the fitting of a mathematical model to measurement data. It is believed that this optimization method is a major breakthrough in the use of analog computers. The method was used with success on six different sets of river flow data for the Environmental Studies and Evaluation Operation of Hanford Laboratories.

Transient solutions for a mathematical model of a pot calciner were made for the Waste Solidification and Engineering Operation of Hanford Laboratories. Several of the process parameters were varied and their effects determined while monitoring the high speed solutions on the oscilloscope.

Optics

During the five-week period covered by this report, the following work was performed:

1. The eddy current scanning probe was completed.
2. Components were fabricated for an ultrasonic image demonstrator unit. The unit was assembled.
3. Six ultrasonic standards were fabricated for attenuation measurements.
4. Two crane periscope heads were repaired for Purex.
5. A coating of $CaF_2$ was applied to tantalum Van de Graaff accelerator targets.
6. Fifty bell jars were tested for window flatness.
7. A laser rod was silvered for Ceramic Fuels.
8. A crane periscope head was repaired for Redox.
9. A spotting scope was modified for use with a camera in taking pictures in the PRTR basin.
10. A camera was fabricated for Radiometallurgy for use in creating large mosaic pictures.

Physical Testing

Sonic detection methods were used to locate a leak in a newly installed



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underground water line at the request of Vitro Engineering. Accurate location of the leak (within 2 inches) minimized excavation required for repairs.

Radiography revealed 16 out of 160 fittings were acceptable for KER Loop test use.

Requests for high resolution ultrasonic tests on small diameter or thin walled tubing is maintaining a work load requiring two-shift operation. Zircaloy tubing over three inches in diameter was inspected for flaws less than four mils deep, and a sample of stainless tubing for the Saxton reactor fuel, only a third of an inch in diameter, for flaws less than two mils deep. A test was completed on thin walled Zircaloy tubing for flaws less than one mil deep.

A successful solution to the poor performance of an induction heater in CPD was achieved after the control section was rewired, the work coil redesigned, the load balanced, and the grade of graphite changed in the susceptor. Desired temperature of 2300°C is now achieved.

A surveillance program, resulting from a study of the kinetic behavior of aluminum control rods, is now being pursued by periodically measuring the hardness along the length of the horizontal control rods in the reactors. These measurements indicate the condition requiring preventive maintenance.

Engineering developmental assistance has been concluded on a new extensometer system to establish quality control criteria from a tensile test. The extensometer, load cell, and recording equipment has been installed and is operating satisfactorily.

Simulated nuclear heating of an oxide fuel element was observed by gamma radiography and image intensification. The sample was heated electrically and the resultant molten area was magnified on the television monitor screen. A movie camera recorded the image.

Voids in oxide fuel elements, simulated with various sized polyethylene balls, were successfully detected by gamma ray photometry. The maximum scanning rate and assembly line setup have yet to be determined.

## INSTRUMENT EVALUATION

Further circuit modifications were incorporated in one developed, and offsite fabricated, scintillation detection and solid state combined alpha-beta-gamma hand and shoe monitor. This unit was one of three fabricated offsite on one purchase order by Radiation Protection, HL. Seven other



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monitors, fabricated offsite on a prior order, are in service at Hanford. Drawing tracings are now being revised to reflect the various modifications incorporated during the extensive evaluation and acceptance testing.

Instructional training was given to personnel of Radiation Monitoring, HL, regarding the use and calibration of the combined alpha-beta-gamma hand and shoe monitors.

Drawing tracings were reviewed and corrected on the latest model hand and shoe monitors, which represent improved circuitry over the units mentioned above. Eight of these latest monitors are now being fabricated offsite.

Fifteen of 25 purchased Juno portable radiation survey instruments have now passed detailed acceptance tests.

Engineering assistance and calibration tests were completed on a developed scintillation and solid state circuitry drinking water monitor which was designed for use by Radiation Protection, HL.

RS Paul

Manager
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RS Paul:mcs



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# CHEMICAL LABORATORY

## RESEARCH AND ENGINEERING

### FISSIONABLE MATERIALS - 02 PROGRAM

### IRRADIATION PROCESSES

#### Plating of Nickel on Uranium by Vapor Deposition

It has been reported[1] that a nickel coating is deposited on glass heated to 350 C in the presence of bis(hexafluoroacetylacetone) nickel(II) vapor and hydrogen at one atmosphere pressure. The film, when stripped from the glass, is ductile, bright and pure. Using similar techniques nickel films have been produced on copper and uranium. The film on copper was soft, adherent and shiny. The film on uranium was adherent, but badly blistered by underlying oxide (or hydride) deposits. Thicknesses were estimated to be 0.1 to 1 micron. This system may have considerable merit for a variety of applications and deserves more detailed investigation.

#### Overheated N-Reactor Fuel Study

An unirradiated N-Reactor co-product element was heated within an inductively-heated steel tube representing the "driver" fuel element. After about one minute of heating at 1100 C, the Zircaloy cladding of the target element failed at a point allowing the target material to flow to the wall of the hot steel tube. An exothermic reaction ensued between the two materials. This result was unexpected; further elements will be tested (e.g., with a Vycor liner in the steel tube) before proceeding to the irradiated target.

### SEPARATIONS PROCESSES

#### Disposal to Ground

Piezometer tubes for depth sampling and head measurement were installed in the five wells monitoring the 1301-N crib at 100-N Area. A head difference of ten feet noted in one well indicates that there may be confining strata in the local flow system. Further sampling and head measurements have been scheduled to obtain a more accurate evaluation of the flow patterns.

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1. Lyle B. Spendlove, Thesis, School of Engineering, Air Force Institute of Technology, Wright-Patterson Air Force Base, Ohio, GNE/Phys/63-16.

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Polonium Volatility Studies

The volatility of polonium from bismuth oxide has been examined in the temperature range 585 to 727 C using a modified transpiration method. Examination of the transport rate of polonium as a function of carrier gas flow rate showed a linear dependence, indicating that equilibrium between gas and solid was attained.

The temperature dependence of the volatility is approximated by

$$\log \pi \text{ (mm Hg)} = \frac{-1.63 \times 10^4}{T} + 15.62 \qquad (1)$$

where the volatility, $\pi$, is defined as the partial pressure of polonium found in the gas phase divided by the mole fraction of polonium in the bismuth oxide. The range of conditions under which equation (1) was determined are as follows: mole fraction polonium from $1.6 \times 10^{-6}$ to $1.92 \times 10^{-6}$, temperature from 585 to 727 C, removal of polonium from solid from $10^{-2}$ to 15 percent. Tank argon dried by passage through a magnesium perchlorate column was used as the gas with flow rates between 24 and 118 ml/min.

The vapor pressure of polonium over the metal has been reported to be

$$\log P \text{ (mm Hg)} = \frac{-5377.8}{T} + 7.2345. \qquad (2)$$

In comparing the vapor pressure of polonium as calculated from the above equations, one finds that the ratio

$$\pi_{Po(Bi_2O_3)}/P_{Po(metal)}$$

increases by almost a factor of 100 as the temperature is increased from 585 to 727 C. This is perhaps due to the increasing rate of diffusion of polonium from the oxide particles.

Polonium Chemistry

The volatility of polonium from solutions containing diphenylcarbazide (DPC) was studied by measuring the amount of polonium found in 14 milliliters of distillate recovered from 25 milliliter samples of acidified bismuth nitrate containing trace quantities of polonium. This series of experiments was done in improved equipment and material balances for polonium were excellent. Conclusions are as follows:



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1. In the absence of bismuth and at acidities ranging from 0.25 to 1 M $HNO_3$, over 99 percent of the polonium distilled.

2. With 0.25 M Bi, 0.25 M $HNO_3$ and 1 g/l DPC, about 80 percent of the polonium distilled. The lower recovery is probably due to the effect of precipitated bismuthyl nitrate which carries polonium. A red color indicates the formation of a Bi-DPC complex which may also suppress polonium volatility.

3. Volatility of polonium was strongly suppressed at an acidity of 2 M $HNO_3$ as only three percent of the polonium distilled. In these experiments sulfamic acid was used to suppress nitrite formation and prevent oxidation of the DPC.

4. In one experiment starting with 0.5 M $HNO_3$ and 0.5 M $Bi^{+3}$ rapid oxidation of the DPC occurred just before the distillation was completed.

In other experiments it was confirmed that polonium is almost quantitatively carried on bismuthyl nitrate, a phenomenon now being examined as a possible process step.

U-233 Support Studies

A short-cooled, IPD-irradiated thoria target element was dissolved in the B-Cell one-slug dissolver to provide feed solution for full-level evaluation of $MnO_2$ scavenging of Pa-233. Dissolution was uneventful; the rate was about as experienced for nonradioactive $ThO_2$ and there was no indication of "blue thoria" formation.

Using the above solution, which contains more Pa than expected Purex thorium feed, several scavenging runs were made. The proposed flow sheet (based on tracer-level studies and Purex plant equipment and operating limitations) involves precipitation of 0.04 M $MnO_2$ by addition of manganous nitrate, followed by permanganate, to a solution 1 M in both $Th(NO_3)_4$ and $HNO_3$. With full-level feed (actually about two-fold "hotter" than Purex will receive) no precipitate was formed and, therefore, no protactinium scavenging resulted. A second strike yielded some precipitate, but two-thirds of the protactinium had returned to solution after five hours. Two similar experiments were run with radiolysis inhibitors (sulfamic acid or urea) added. Either 0.1 M sulfamic acid or 0.1 M urea protected the precipitate for periods of over 20 hours, a length of time considerably longer than expected plant time cycles. Adequate Pa scavenging was also obtained with ten-fold "hotter" feed when protected with 0.1 M sulfamic acid. Successful plant performance accordingly appears assured with respect to this factor.



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Several reagents have been evaluated for redissolution of the Pa-bearing $MnO_2$ for load-out into a storage cask. Sucrose was found to be very satisfactory. Stoichiometry of the reaction apparently corresponds to the equation

$$12\ MnO_2 + C_{12}H_{22}O_{11} \rightarrow 12\ Mn^{++} + 12\ CO_2 + 11\ H_2O.$$

The dissolution was carried out in 0.5 M $H_2SO_4$ and resulted in a sulfate solution suitable for anion exchange recovery and purification of the daughter U-233.

Thoria Dissolution

It has been reported (SRL) that the dissolution of thoria in nitric acid-hydrofluoric acid solutions is accelerated by having metallic aluminum present and undergoing dissolution simultaneously with the thoria. Investigation of this procedure has produced no evidence for any catalytic effect of dissolving aluminum on the dissolution rate of thoria. Any increase in thoria dissolution rate appears to be due to increased agitation caused by the off-gases produced by dissolving aluminum.

Extraction of Thorium with 1,5-Dimethoxypentane

Distribution ratios ($E_a^o$) for the extraction of thorium by 1,5-dimethoxypentane from 1 M $HNO_3$ - 1.5 M $Al(NO_3)_3$ and 0.2 M $HNO_3$ - 0.5 M $Al(NO_3)_3$ decreased markedly with decreasing thorium concentration in the aqueous phase. When extracting thorium from 2 M $HNO_3$, distribution ratios were essentially independent of aqueous phase thorium concentration. Aqueous phase solubility of 1,5-dimethoxypentane increased markedly with increasing $HNO_3$ concentration in the aqueous; equal volumes of 3 M $HNO_3$ and 1,5-dimethoxypentane were completely miscible.

Thorium Solvent Extraction

Miniature mixer settler runs testing a study flow sheet employing 5 per cent TBP-Soltrol for Purex plant recovery and purification of thorium were continued. Thorium loss to the 1CW was 2.7 percent in a run (12 mini stages) simulating the proposed strip column (1C) conditions. This was reduced to 1.1 percent by decreasing the aqueous to organic flow ratio from 0.7 to 0.5 and the 1CX nitric acid concentration from 0.2 to 0.01 M.

Bismuth Dissolution Studies

A potential method for large scale processing of irradiated bismuth will involve nitric acid dissolution followed by solvent extraction of

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polonium-210 with dibutyl carbitol. Two batch dissolutions of bismuth target slugs were carried out to provide feed stock for subsequent solvent extraction studies and to provide dissolution experience. In the first of these, 96 percent of the bismuth charged was dissolved in four hours with a total nitric acid to bismuth mole ratio of 5 to 1. Nitric acid was added incrementally to maintain the acidity constant at 4 ± 1 M. In the second run, a fresh charge of bismuth was completely dissolved in five hours. The average penetration rate in both runs was about 0.13 in/hr and was the same both radially and axially. The final product from both runs contained 1.9 M bismuth and an estimated 3.8 M $HNO_3$. The overall nitric acid consumption using a downdraft reflux condenser was about 3.3 moles/mole bismuth dissolved. The reaction proceeded very smoothly and did not appear to be exothermic. About one to two feet of foam was observed in the second run.

## Ion Exchange Process Hazards Studies

Studies of the critical temperatures at which runaway chemical reactions can occur in ion exchange processes systems have been discussed in earlier reports. More recently, studies have been devoted to determining mechanisms by which these temperatures can be reached. As an example, 4-inch diameter fixed bed experiments with anion ion exchange resin partially loaded with dichromate, have shown that a step increase in temperature of a 7 M $HNO_3$ feed stream results in a moving reaction zone with rapid increases in bed centerline temperatures as the zone moves through the bed. In one experiment a step increase in feed temperature from 24 to 80 C resulted in bed centerline temperatures of 88, 97 and 107 C at 1, 2 and 3 feet downstream of feed inlet as the temperature front moved through the bed. Work is continuing with calorimetric studies and preparation of a computer program linking reaction kinetics, thermodynamics and flow conditions with expected temperature rises.

## In-Tank Concentration of Wastes

In support of the program on in-tank concentration of wastes using immersion heaters, vapor pressure versus temperature and concentration data were obtained for three waste compositions. Two of the compositions were essentially sodium hydroxide-sodium aluminate-sodium nitrate-sodium nitrite; the third was sodium carbonate-sodium nitrate-sodium nitrite. For all cases, except the third composition highly concentrated, curves showing log vapor pressure versus temperature were parallel over the pressure range studied (ca. 300 - 750 mm Hg). For the sodium hydroxide containing solutions, the ratio of latent heat of vaporization of the solution to that of water at the same temperature was near unity for the unconcentrated solutions and increased to about 1.25 at the maximum concentration studied. This ratio was about one for all concentrations of the sodium carbonate-sodium nitrate-sodium nitrite solution.



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Strontium Semiworks Solvent Studies

Experiments to compare the effects of irradiation and chemical degradation on Ce(IV) extraction by D2EHPA-TBP-Soltrol 170 and D2EHPA-TBP-normal paraffin hydrocarbon solvents were continued. During 10 extraction-strip cycles (simulated solutions) in which the solutions received about 27 watt-hours of Co-60 gamma radiation per liter per cycle Ce(IV) $E_a^o$'s for both solvents decreased from greater than 2000 to about 70. An additional five cycles did not further decrease the Ce(IV) $E_a^o$ for the solvent containing normal paraffin hydrocarbon. Similar data for solvent containing Soltrol 170 are forthcoming.

In separate studies it was found that caustic washing of solvent prepared from chemically degraded Soltrol 170 appears to markedly decrease extraction of Ce(IV) in present Strontium Semiworks procedures.

WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

Purification of Strontium-90 from Magnesium

Although it has been suspected that purification from magnesium is not obtained in the D2EHPA (Semiworks) strontium purification process, this has not been a matter of concern in the past for two reasons: (1) magnesium is probably removed in the lead sulfate precipitation process (magnesium sulfate is highly soluble), and (2) there is very little magnesium in the uranium metal or other essential materials currently used. Both situations are expected to change. The lead carrier process will not be used in the Waste Management program and alterations in the metallurgical processes used by the uranium suppliers will probably result in the introduction of significant quantities of magnesium.

The D2EHPA solvent extraction behavior of magnesium was measured and found to be identical to that of strontium. Addition of complexing agents (such as DTPA) also failed to result in a significant change in the relative extractabilities of the two, apparently precluding magnesium separation by this scheme. A precipitation process, utilizing the low solubility of magnesium hydroxide, was much more successful. Less than 5 percent of the magnesium initially present remained in solution at pH 10 and less than 2 percent of the initial strontium was lost below a pH of 11. Strontium losses increased to 15 percent at pH 13. Conditions used for magnesium hydroxide precipitation included heating to 80 - 100 C, addition of NaOH to the desired pH, digestion and centrifugation. The precipitate occupies a volume less than 8 percent that of the original solution and is somewhat gelatinous, but centrifuges well.



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CSREX Process Chemistry

The reaction of 0.37 M BAMBP in cyclohexane with nitric acidified 0.2 M $NaNO_2$ was examined. Essentially no reaction occurred when the system at pH 6.5 was stirred overnight. When nitric acid was added to bring the acidity to pH 3.5, reaction occurred with evolution of brown fumes ($NO_2$). After three minutes the organic layer was removed and washed with water. Spectrophotometric analysis disclosed a strong peak at 371 mμ indicative of nitration (or nitrosation). A little more than 50 percent of the BAMBP was apparently destroyed and the extraction coefficient for cesium had dropped from 1.8 to 0.4. When BAMBP-cyclohexane was exposed to 0.2 M $NaNO_2$ adjusted to pH 1 with acetic acid, rather than nitric acid, the same vigorous reaction was observed thus demonstrating that the chemical attack was actually by nitrous acid. The ease of nitration (or nitrosation) is not surprising because of the activating action of the phenolic group from ortho and para ring positions.

EQUIPMENT AND MATERIALS

Non-Metallic Materials

Three six-inches square pieces of Corning No. 8362 shielding window glass (6, 2-3/4, and 2 inches thick) have received a total Co-60 gamma dose (incident on one face) of $7 \times 10^7$ r at a rate of $7 \times 10^4$ r per hour. No spontaneous discharge has occurred and color development has not been marked. This type glass has been specified for cell-side sections of the windows in the 324 Building high-level cells.

Slag and Crucible Dissolver

Following the successful completion of glass mockup dissolution studies, design and fabrication of the full scale plastic-lined slag and crucible dissolver were completed. Shakedown operation of the dissolver was performed on a round-the-clock basis. Initial tests were directed toward determination of optimum acid concentration and entry point; next, the ability of the dissolver to maintain circulation under conditions of high solution feed rates was satisfactorily demonstrated. Subsequent operation has consisted of the addition of 190 cans of simulated slag and crucible material at the rate of two cans per hour. Performance has been generally satisfactory; the operating problems that arose were readily surmounted. At month's end the system was shut down for inspection and to determine the composition and location of solids deposits.

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## PROCESS CONTROL DEVELOPMENT

### Control System for Electrorefining Process

A control system for use in plutonium electrorefining process studies has been designed, fabricated, tested and delivered to CPD. The system controls either electrolysis voltage or current, in the ranges 0-1, 3, 5 and 10 volts, and 0-100 amps, respectively. Selectable high and low limits are provided to terminate the process in the event pre-set voltage or current conditions are not met.

### In-Line Plutonium Detection

Performance of the recently installed scintillating glass and X-ray plutonium detectors in Purex has been very satisfactory. The improved response time compared to the alpha printer (90 seconds versus 15 minutes) plus the simplicity of the new detectors has resulted in a decision to remove the alpha printer from service. In a related study, the effects of varying Pu-240 content on plutonium detector performance are under investigation. Activity measurements, using neutron, alpha and X-ray detectors, were made of samples containing plutonium with 6 percent and 10 percent Pu-240 content. Significant differences in measured activity were obtained with the various detectors. Comparison and analysis of experimental and calculated results is now in progress.

### Computer Process Control Study

A GE-412 process computer was delivered for use in the development of advanced process control and data handling technology. An extensive series of acceptance tests was completed satisfactorily. Connection of input/output signal cables between the computer and the experimental pulse column has also been completed. A preliminary check verified the continuity of the input signals to the computer and revealed an encouragingly low noise level on these signals. The ability of the digital output converter system to successfully convert the computer output information into equipment starting and stopping functions, graphic panel alarm display functions, and control option functions was verified. Previously prepared programs, assembled on EDPO's GE-225 computer, are now being debugged on the 412. To date, 7 of the 16 completed process control programs have been debugged.

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## REACTOR DEVELOPMENT - 04 PROGRAM

### PLUTONIUM RECYCLE PROGRAM

#### $PuO_2$ Precipitation in the Salt Cycle Process

Essentially complete (99.7 percent) recovery of plutonium from a Salt Cycle salt melt was achieved by a two-step precipitation process. The KCl-2.5 LiCl melt was first sparged with an oxygen-chlorine stream at 575 C, precipitating 95 percent of the plutonium as $PuO_2$, then with a hydrogen-argon stream, precipitating the balance of the plutonium along with about five percent of the uranium as $UO_2$.

#### Electrochemistry of Noble Metal and Rare Earth Ions in Molten Chloride Salts

An investigation has been started on the electrochemistry of the noble metal and rare earth ions under Salt Cycle conditions. As a first step in the work on noble metals, attempts were made to develop a micro-electrode more noble than platinum. Micro-electrodes of gold prepared in several ways all showed about the same characteristics - characteristics which, unfortunately, were not much improved over those for platinum at anodic potentials and were much worse at cathodic potentials. The differences noted in the current-voltage curves for platinum and gold electrodes in LiCl-KCl melts were extremely interesting and are considered to have an important bearing on double layer structure at solid electrodes in melts. It was also of interest to note that the over-voltage for $UO_2$ electrodeposition was about 0.2 volt greater on gold than on platinum at a polarization rate of 5 mv/second.

Electrode potential data were obtained with nonporous graphite electrodes for $5 \times 10^{-3}$ molar ruthenium, rhodium and palladium solutions. The decomposition and rest potentials measured in this study show, as expected, that these metals will codeposit with $UO_2$ in the $UO_2$ electrodeposition steps of the Salt Cycle process. They also indicate little chance of achieving a separation from uranium by electrodeposition of the noble metals in a pre-$UO_2$-deposition step.

The electrochemical studies bore unexpected fruit, however, in observations of chemical behavior of the noble metals which may prove helpful in devising uranium-noble metal separation schemes. It was found, for example, that at temperatures above 700 C, ruthenium, rhodium and palladium precipitated at a significant rate from chloride salt melts containing their ions. Precipitates were also formed from these solutions at lower temperatures with exposure to oxygen or upon the addition of $Li_2CO_3$. The only precipitate characterized was rhodium which deposited as the metal at 700 C.



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Studies of rare earth electrochemistry were also complicated by electrode difficulties. For reasons not understood, platinum and graphite electrodes did not show metal deposition waves, but a gold electrode did (perhaps because of alloy formation between the rare earth metal and the gold). Three reduction waves were found for samarium ($5 \times 10^{-3}$ M in 1:1 KCl-LiCl): the reduction of Sm(III) at $E_{p/2} = -1.01$ v; a metal deposition wave at -1.40 v; and a third wave (perhaps due to the melt) at -1.43 v. Only one reduction wave was seen for $5 \times 10^{-3}$ M neodymium(III) in 1:1 KCl-LiCl, the metal deposition wave at -1.22 v.

In the course of this work it was noted that addition of carbonate causes immediate precipitation of $Nd_2O_3$ and $Sm_2O_3$ from Nd(III) and Sm(III) melts. Addition of carbonate to Sm(II) (in a dark red-violet solution prepared by electrolysis of a Sm(III) solution) oxidized the Sm(II) to Sm(III) and precipitated $Sm_2O_3$. Sparging with HCl quickly dissolved the precipitates. Sparging these solutions with air or oxygen caused a slow precipitation and considerable changes in the current-voltage curves.

Spectrophotometry of Plutonium in Chloride Melts

Equilibria involving plutonium ions in the KCl - 3 LiCl melt system have been studied spectrophotometrically. As expected, this melt exhibits less complexing strength than those previously studied (KCl-1.4 LiCl, LiCl-KCl, and CsCl-1.4 LiCl). The ratio (R) of plutonium(IV) to plutonium(III) in the $Cl_2$-saturated melt was found to be about four-tenths that in LiCl-KCl and only about one-tenth that in CsCl-1.4 LiCl (at a given temperature). The data for KCl-3 LiCl at 500 to 700 C can be expressed as:

$$\log R = (2660/T) - 3.77$$

Thus, about 32 percent of the plutonium is present as plutonium(IV) at 500 C and about 8 percent at 700 C.

The ratio of plutonyl(VI) to plutonyl(V) in KCl-3 LiCl sparged with $O_2$-$Cl_2$ was also found to be lower than in the melt systems studied previously; in fact, plutonyl(V) is the predominant plutonyl species in this melt. The equilibrium

$$PuO_2 + 1/2\ Cl_2 = PuO_2Cl$$

was studied to learn how low a plutonium concentration can be attained by precipitation of $PuO_2$ by $O_2$-$Cl_2$ sparging. The results at 600 C can be expressed as:

$$\text{w/o Pu} = 0.028\ X_{Cl_2}^{0.5} \quad \text{or} \quad \underline{M}_{Pu} = 0.0018\ X_{Cl_2}^{0.5}$$

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where $X_{Cl_2}$ is the fraction of chlorine in the sparge gas. Equilibrium was approached from both directions to assure validity of the results. The effect of temperature on this equilibrium has not yet been determined.

The attainment of equilibrium by precipitation of $PuO_2$ during the $O_2$-$Cl_2$ sparge is not a very rapid process. For example, three hours after all the plutonium(III-IV) had been oxidized and precipitation had begun, the plutonium concentration had decreased only about 90 percent of the way to the equilibrium concentration. About five hours were required for equilibrium to be obtained.

The results obtained here are in good agreement with Benedict's results on precipitation of $PuO_2$ from KCl-2.5 LiCl containing 20 w/o U and 0.2 w/o Pu. Using 50 percent $O_2$ - 50 percent $Cl_2$ at 575 C, he precipitated 90 percent of the plutonium. From the above equation for KCl-3 LiCl at 600 C, with an initial Pu concentration of 0.2 w/o and $X_{Cl_2}$ = 0.5, 10 percent of the plutonium would be left in the melt at equilibrium.

## Salt Cycle Process Engineering Development

In August it was reported that sparging the molten salt bath with pure $O_2$ resulted in the precipitation of 92 percent of the plutonium initially in solution. Recently received radiochemical analyses revealed that the Ce-144, Y-91, Pm-147 and Eu-154 were precipitated to the extent of 92, 18, 5 and 2 percent, respectively.

## Non-Metallic Materials

A cellulose graphite crucible produced by the Hirst Research Center of the General Electric Company, Ltd., has been exposed for 96 hours to a NaCl-KCl melt at 900 C under an argon atmosphere. There has been no apparent damage to the crucible; none of the melt has percolated through the crucible.

## Dissolution of PRTR $UO_2$-$PuO_2$ Fuels

Conversion of uranium to $UF_4$ was less than two percent during Zirflex decladding of irradiated PRTR $UO_2$-$PuO_2$ fuel when the initial decladding solution composition was 2 M $NH_4F$-0.17 M $NH_4NO_3$ and terminal zirconium concentration was 0.1 M. Fluoride concentration during subsequent core dissolution to 0.84M uranium was well under 0.1 M. This may represent a practical approach to processing the $UO_2$-$PuO_2$ fuels in the Redox stainless steel equipment.



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## RADIOACTIVE RESIDUE PROCESSING DEVELOPMENT

### Spray Calciner

Experiments were made to determine the effects of a draft tube and colemanite addition on the drying capacity of the 14-inch diameter calciner with simulated 1WW feed. The drying capacity is somewhat arbitrarily defined as the feed rate at which the centerline temperature inside the calciner and 30 inches below the nozzle is just 250 C.

The addition of colemanite had no detectable effect, whereas the addition of a draft tube considerably increased drying capacity: 10.5 gph compared with 6.5 gph at the same atomizing steam rate and furnace temperature (700 C).

### Spray Calciner-Melter

Three coupled 14-inch spray calciner-melter runs without a draft tube were made with simulated 80 gal/ton U Purex waste. Results were:

1. Simulated 1WW with a borax concentration of 85 grams/liter was calcined and melted with no difficulty during a three-hour run.

2. Feed for one run contained sulfate plus colemanite. Excess iron was added to a simulated Purex waste containing sulfate to suppress sulfate corrosion of the 310 stainless steel melter. Visual corrosion was not evident.

3. Operating capacity for the prototype scale spray calciner-melter is between six and seven gallons per hour of an 80 gal/ton U Purex waste with added colemanite. This rate represents solidification of waste generated by processing from 1.8 to 2.1 tons of uranium per day. The capacity was determined with the calciner wall temperature at 700 C, no draft tube, and using a Spraying Systems Company set-up No. 5 atomizing nozzle with a 7/32-inch air orifice. The melter wall temperatures were maintained from 950 to 1000 C.

   At ten gallons per hour, the calciner could not completely dry the feed. Increasing the calciner wall temperature to 750 C started melting of powder on the wall. The calciner and melter capacity are well matched since the melter progressively filled with powder at the 10 gph rate.

4. An electrical conductivity probe for detection of liquid level in the melter proved satisfactory for these runs.



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Melter Materials

Corrosion rates for coupons of 50 Cr-50 Ni alloy, 35 Cr-65 Ni (Corronel 230) and 25 Cr-20 Ni (310 SS) during 90 hours' exposure to a Purex waste with added colemanite melt at 900 C were 0.4, 1.2 and 36 mils/mo, respectively. An attempt to cathodically protect 310 stainless steel in this melt was not successful.

Continuous Phosphate Glass Experiment

The first full-level run in the equipment installed in A-Cell of the High Level Radiochemistry Facility for testing the Brookhaven continuous phosphate glass process was described last month; however, the melt pot had not been sectioned at that time nor was any analytical data available. When the pot was sectioned (in the Radiometallurgy Laboratory) it was found that a grossly crystalline, rather than glassy, product was produced. In addition, a boil-over during the run contaminated the condensate and off-gas samples resulting in acquisition of little or no valid data on fission product behavior.

A second and much more successful run has now been made. The run was much smoother, the quantity of phosphoric acid added was better controlled (ratio of $P_2O_5$ to metal oxides is quite critical in determining whether a stable glass is produced), and the melter was operated at a somewhat higher effective temperature (by filling the pot less full, and by turning off inlet line cooling and continuing furnace heating after cessation of feed addition). Result was a uniform, obsidian-like glass which showed no trace of crystalline structure under "metallographic" examination. Most surprising finding of the run was that apparently little or no ruthenium (or any other isotope) was evolved to the off-gas or condensate streams, and that essentially all of the fission product activity was retained in the glass. If verified by additional runs, this retention of ruthenium would be a significant advantage for the Brookhaven process. Samples of the glass are being retained for long-term observation. Leaching rate measurements also have been initiated.

Laboratory studies of the effect of composition on properties of phosphate glasses showed that glasses prepared by melting at 1200 C exhibit much lower aqueous solubility, devitrify less easily, and are less affected by the $P_2O_5/M_2O$ ratio than are glasses prepared at lower temperatures. It is hypothesized that these effects may be due, in part at least, to loss of sulfate at the higher temperature. This theory is being tested by chemical analysis of the glass samples and by preparation of glasses from sulfate-free synthetic waste.



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In anticipation of future hot cell glass runs in which massive fission product spikes will be added to better simulate the specific activity of high-burnup power reactor wastes, ISOGEN computer code calculations have been made of the isotopic content of a typical power reactor fuel (Dresden). The code was modified somewhat to simplify handling of output data and to include summation in terms of atoms, grams, curies, beta watts, and gamma watts for each element and each time interval of interest.

Waste Solidification Engineering Prototypes

Testing of prototype equipment was initiated to confirm design values and check operability of the system prior to installation in the Radiochemical Engineering Cells of the Fuels Recycle Pilot Plant. Pot calcination is the first process being tested. Initial results indicate:

1. The six zone induction heated furnace system operates at 50 percent efficiency (power input versus water vaporized).

2. Operation of the 8-inch diameter pot calciner with a Darex type feed (typical of Purex type without sulfate) essentially reproduces the results experienced at ORNL. Overall average capacity was 10 liters per hour for the 62-hour run using a constant net power setting of 18 kw throughout the run. The equipment performed quite well throughout the run.

3. Water boil-up rates can be increased to 60 liters per hour without difficulty.

Intermediate Level Waste Treatment

Laboratory experiments were conducted with a thin bed of 100-230 mesh clinoptilolite, to measure the differential pressures and cesium decontamination factors across the bed for recent steam-stripped Purex Tank Farm condensate. The transfer tank containing the condensate had previously been treated with hypochlorite to insure a low bacteria count. The differential pressure across the 1.8 cm bed increased from 1 psi to 3 psi after 8300 bed volumes at 5 gallons per minute per square foot flow rate. The run was terminated after 9300 bed volumes when the pressure increased to 10 psi due to a different batch of condensate that had been fouled by residual material in a storage tank. The cesium decontamination factors varied from 100 to 200 throughout the run. These results compare favorably with previous results with thin precoat ion exchange beds prior to the recent difficulty with solids in the condensate.

## CONTAINMENT SYSTEMS EXPERIMENT

### Major Facilities and Equipment

The footing was poured and anchor bolts were installed for the containment vessel in T-Plant to complete preparations for erection work by the vendor, now scheduled to begin in mid-November. Negotiations for the reactor simulator vessel were completed and final contract approval is expected within two weeks.

### Fission Product Simulation

The Aerosol Development Facility in Building 271-CR has been provided to develop suitable aerosols and the necessary aerosol generating and sampling equipment for the CSE program.

One run was made to demonstrate the adequacy of the facility for test purposes. About 0.06 μg of carrier-free iodine-131 was transmitted to the tank through a heated 18-foot long, 1-inch diameter pipe. The atmosphere in the receiver was air at ambient temperature and pressure. The iodine was generated in essentially pure elemental form, but apparently reacted rapidly with organic vapors in the tank to a form which penetrated silver-plated screens and absolute filters but which was adsorbable on one layer of charcoal-impregnated filter paper. Fifteen minutes after delivery to the tank, 70 percent of the iodine was in this penetrating form; after four hours 90 percent of the airborne iodine was in this form. The time-integrated deposition velocity constant for iodine deposited on copper was 0.012 cm/sec. Surfaces of SS 304 and epoxy paint had deposition velocities of 0.002 cm/sec.

An attempt to heat $UO_2$ inductively using a frequency of 380 KC was unsuccessful, even when the $UO_2$ was preheated to 1800 C. No difficulty in inductively coupling to $UO_2$ with 4 MC frequency has been experienced. A tungsten boat with 60 mil walls was penetrated by molten stainless steel in about five minutes at about 1600 C, showing that tungsten will not be a suitable crucible material for clad $UO_2$ heating in oxidizing atmosphere.

### Program Planning

The system of equations relating simulator vessel pressure, rate of flow from a rupture, and duration of blowdown was solved by an analog computer program. These results will either be used as input data for the overall solution to the pressure transient problems, or the computer program will be made an integral part of the general program, depending on the number of computer components required for the



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general program. A simplified case of a containment system involving water spray pressure suppression, and air recirculation and cooling in a single containment vessel (as opposed to vapor suppression in a wet well-dry well) is being prepared for computer solution next.

The CSE program document has been submitted to Technical Publications for formal issue. An informal memo was prepared to outline a program for development of simulants for CSE, by comparing the transport behavior and characteristics of simulant aerosols and those of the fission product aerosols released from large samples of highly irradiated $UO_2$ in several different conditions affecting transport.

OFF-GAS DISPOSAL TO GROUND

The "Steady-State Flow in Soils" computer program for analyzing liquid flow through porous media has been revised to form a new program which will calculate gas flow through soils. The program will handle compressible gas conditions in soils with relatively low moisture contents. The full capacity of the original program (8000 grid points) is still available to calculate in one, two or three-dimensional, or axisymmetrical systems.

SHIPPING HAZARDS

The initial effort on this new program is to identify those highly radioactive material shipment accident situations for which a better understanding of airborne release phenomena would lead to more realistic determination of accident consequences. Emphasis in this program will be placed on gaining an understanding of the behavior and disposition of the shipped radioactive materials in very serious accidents which breech the primary container. Subsequent release and transport of gases and particles due to high temperatures, air currents, or other energy sources will be studied.



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BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

The statistical analysis of numerical errors in the permeability calculation methods was completed. The calculated values from the "Stream Computer Program" were compared with the known permeabilities solved in mathematically closed form from the test model system. The errors found showed that the numerical methods were far more accurate than had been expected. The average error in permeability due to evaluation of the potential integral was << 1 percent of the true permeability. Additional analyses showed that the error is essentially independent of the length of the streamline generated. Accordingly, remarkably accurate results can be expected in using the Stream Program to analyze ground water flow systems. This adds additional emphasis to the work under way on obtaining accurate functional fitting of tabular data which are inputs to the streamline-generating program.

It was found that the extension of Philip's numerical methods to transient, partially-saturated, radial flow does not give a unique solution. This is because of a singularity at $r = 0$; i.e., the point source. With Philip's methods, the point source must be used to apply Boltzmann's transformation which reduced the partial differential equation to an ordinary differential equation. The transformation cannot be used for a source of finite radius. A better boundary value problem can be written by not requiring a point source. That is, actual physical leaks from tanks probably define a source of finite radius. Solutions of these boundary value problems will be pursued.

Soil Chemistry

Equilibrium distribution coefficients ($K_d$) were determined for phosphate adsorption on a Hanford soil. Values range from 4.8 ml/g for P-32 traced distilled $H_2O$ to 1.3 ml/g for adsorption from a 100 ppm P solution. Results indicate that the relationship between phosphate adsorbed and equilibrium solution concentration fit the Langmuir adsorption isotherm at concentrations below 50 ppm P. A $K_d$ of 11 ml/g was determined for P-32 adsorption from traced tap water.

Soil dispersion treatments which will aid in the separation and identification of the different mineral series in Hanford soils have been determined. Dispersion is accomplished by removing exchangeable divalent cations and free carbonates with a slightly acid sodium acetate buffer. Soil organic cementing matter is removed with hydrogen peroxide.



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Chemical cements, such as colloidal silica and alumina, are removed with 2 percent $Na_2CO_3$. Final dispersion is in dilute sodium carbonate solution. Mineral concentration was achieved by particle size fractionation. Montmorillonite, mica and a chlorite-like mineral have been identified in the clay fraction of a Hanford soil. The minerals were identified by X-ray diffraction, using both random powder and parallel oriented specimens.

## RADIOLOGICAL AND HEALTH CHEMISTRY

### Counting Methods

Gamma spectrometric analysis utilizing a lithium-drifted germanium detector has been demonstrated to be suited for high level work where good energy resolution is important, and in the X-ray region of 20-100 kev where high efficiency is obtained together with excellent resolution. Good germanium detectors have resolutions a factor of 10 better than sodium iodide detectors. This makes analyses possible for gamma emitters in mixtures which have energy peaks too similar to be sufficiently resolved using sodium iodide detectors. For example, a fission product mixture whose gamma spectrum showed only Cs-137 and Zr-Nb-95 peaks incompletely resolved with the sodium iodide detector, showed seven distinct peaks from Zr-Nb-95, Ru-106, Cs-134, Cs-137 and Ce-144 with the germanium detector.

### Radiation Chemistry

Evidence was obtained this month to confirm the mechanism by which erioglaucine is decomposed in irradiated aqueous solutions containing dissolved oxygen. The utility of erioglaucine as a test material for studying radiation effects in biological systems lies not only in its lack of toxicity but also in its response to only one primary irradiation produce formed from water, namely hydroxyl radical. Direct measurements during the past years have shown that erioglaucine does not react with molecular hydrogen, oxygen or hydrogen peroxide. While it does react with hydrated electrons, these reagents are in short supply in aerated systems due to their rapid reaction with $O_2$. This leaves two reactants as possible causes for bleaching of the dye, hydroxyl and perhydroxyl radicals. It has been shown that 2, 5-dichlorohydroquinone does not react with perhydroxyl radicals but does react with hydroxyl radicals. Irradiation of erioglaucine in the presence of an excess of this reagent gave an effective measure of the reaction rate of erioglaucine with perhydroxyl radicals. The rate was found to be too small to be significant in the relative rate constant studies made to date.

p-Nitrobenzoate ion was found to form a radical with a half-life of several minutes in aqueous solution at pH 12 when irradiated in the



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gamma cell flow system. Again the hydrated electron appears to add to the pi-electron structure of the solute as indicated by the strong triplet splitting in the spectrum. The exact structure of the radical-ion is not known, but the hyperfine structure within the triplet band envelopes does not correspond to that from electrolytically reduced p-nitrobenzoate or p-nitrobenzaldehyde. Some more complicated structure is suspected which may prove interesting in pointing out the difference berween electrolytic and radiolytic reduction.

In order to detect organic radicals formed by reaction of solutes with hydroxyl radicals in the gamma irradiated flow system, $N_2O$ was bubbled through a reservoir in the liquid stream to convert all hydrated electrons to hydroxyl radicals. A solution of tryptophan treated in this way did not give rise to an ESR signal at pH 5; however, at pH 12 a singlet absorption peak was obtained. Due to the lack of hyperfine structure in the spectrum the chemical structure of the radical is unknown. Since tryptophan itself does not decompose in basic solution it seems probable that the radical observed in the irradiated stream is a radiation-produced radical of tryptophan which is either stabilized in base or decomposed further in base to give a more stable radical.

## ATMOSPHERIC RADIOACTIVITY AND FALLOUT

### Aerosol Sampling Studies

The filter retaining ring of the filter supports used extensively in meterological trace dispersion studies and washout measurements was shown to account for as much as 40 percent of the total collected ZnS particles. This error will be particularly noticeable at low sampling velocities with high windspeeds. This collection by impaction on the filter support ring may well represent a major source of nonreproducibility in sample evaluation. Further investigation is planned.

Nonuniformity of particle concentrations at points of sampling and the somewhat random errors introduced by filter support geometry contribute to the uncertainty in the factor by which a sub-isokinetic sample should be corrected. Data obtained to date are being evaluated and organized for a report.

### Aerosol Generation and Characterization

Further study of the electrical charge carried by particles from the spinning disc aerosol generator showed that virtually every particle was positively charged. The apparent prominence of negatively charged particles stated in last month's report was attributed to inadvertent reversal of collector polarity. Filtering the sheath air to the charge



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spectrometer confirmed the presence of positive charges on the generated particles by excluding extraneous charged particles.

Improved flow patterns were achieved by providing multiple outlets for a horizontal settling chamber for separating polydisperse particles into size fractions. Small temperature differences between inlet water and the bulk settling chamber water, however, were found to create nonuniform velocity profiles which must be rectified for optimum performance.

## ISOTOPES DEVELOPMENT - 08 PROGRAM

### Promethium Isotopic Composition

Work continued on the determination of the isotopic content (Pm-146/Pm-147 ratio) of VBWR and Yankee fuel materials. Preliminary analyses were obtained on two samples from the first Yankee core, discharged May 18, 1962. One sample was from the center of a centrally located fuel element assembly, exposed to 13,200 Mwd/metric ton with a fast-to-thermal neutron flux ratio of 3.1. This contained a Pm-146 to Pm-147 activity ratio of $5.2 \times 10^{-6}$. The second sample was from the edge of the same fuel element assembly, exposed to 17,200 Mwd/metric ton with a fast-to-thermal neutron flux ratio of 1.8. This contained a Pm-146 to Pm-147 activity ratio of $4.7 \times 10^{-6}$. Preliminary calculations demonstrated that these values agree well with the 5.9 millibarn (n,2n) cross section for Pm-147 for fission spectrum neutrons, and fission yield value of $1.2 \times 10^{-8}$ previously used in Hanford work.

As the exposure increases at the same fast-to-thermal flux ratio, the Pm-146 to Pm-147 ratio is expected to increase approximately linearly with exposure. For a more thermal flux, such as the Dresden boiling water reactor, less Pm-146 is expected. Hanford-produced promethium will continue to have an activity ratio of about $5 \times 10^{-7}$.

The VBWR and second Yankee core samples require additional time for Pm-148 decay before conclusive results can be obtained.

### Promethium Re-Entry Burnup Studies

Debugging of the isotopic heat source atmospheric re-entry burnup code continued, results of the first runs of the code on the computer indicating certain needed changes in the mathematical model used. In the meantime, collection of experimental data for use in the calculations continued. The surface tension of molten neodymium (stand-in for promethium metal) has been calculated from the dynamic behavior (measured from high speed motion pictures) of drops and filaments produced in the plasma jet experiments (described last month). Calculated results were



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in the range 49 to 70 dynes/cm, in good agreement with the theoretical Zadumkin formula (which relates surface tension to crystal structure, heat of sublimation, temperature, coefficient of linear expansion, and characteristic Debye temperature). Because of uncertainties in the experimental determination, the calculated value from the Zadumkin relationship is probably the more reliable. The thermal conductivity of neodymium oxide, needed both for ablation calculations and for generator design purposes, has been measured at low temperature with a hot wire technique and is being measured (in Graphite Research and Development Operation) at higher temperatures by a heat pulse diffusivity method.

Pneumatic Impaction

Use of radiant-heat spray drying for the preparation of strontium titanate powders suitable for pneumatic impaction was investigated with the laboratory spray dryer. Solutions prepared from strontium nitrate and titanium nitrate in Sr/Ti ratios of 1:1, 2:1, 3:1 and 4:1 were processed at temperatures of 600 to 1200 C. Resultant powders are being examined by X-ray diffraction to determine whether $SrTiO_3$, $Sr_2TiO_4$, etc., were indeed produced. If favorable results are obtained, larger quantities will be prepared for trial impaction.

Americium-241 Shielding Requirements

Shielding requirements of Am-241 were calculated to determine whether americium could be handled in the lightly-shielded promethium glove boxes. Shielding requirements were found to be like those of Pu-238 on the basis of literature data. A recently purified sample of Am-241 was examined, however, and two unreported gammas found at 0.6 and 0.64 Mev. These are in about equal abundance with the 0.33 and 0.37 Mev gammas. That these are indeed discrete gammas and not coincidences of lower energy gammas was shown by counting through a lead absorber, which reduced the 0.33 and 0.37 Mev gammas appreciably but had only a slight effect on the higher energy gammas. Examination of americium from the leading and trailing fractions of a recent ion exchange purification run showed the same relative abundance of these gammas in all fractions, strongly suggesting that they are indeed from americium. Additional experiments will be required to determine whether they are from Am-241 or Am-243. The new gammas will increase the shielding requirements for americium somewhat, especially for large sources. The shielding will be comparable to that required by promethium containing 0.5 ppm Pm-146.



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Engineering Program

Fabrication of the 1220F Dynapak machine was completed in vendor shops and performance tests successfully completed. The machine was shipped October 26. Design of punch and die tooling was completed and procurement initiated.

The chemical processes for the production of solid materials suitable for high energy impaction of the various candidate strontium, promethium, cerium and cesium compounds have been integrated to a simplified precipitation-filtration process. The process equipment has been designed and is currently being constructed. A complete compilation of equipment needs for installation in the Fuels Recycle Pilot Plant was completed.

E. R. Irish

for Manager
Chemical Laboratory

MT Walling:cf

[redacted]

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# BIOLOGY LABORATORY

## A. ORGANIZATION AND PERSONNEL

L. R. Richardson, Biological Analyst, transferred from C.P.D. to Metabolism Operation on October 5.
J. L. Dilley, Biological Scientist, joined the Inhalation Toxicology Operation on October 12 as a new employee.
P. C. Lee, Secretary, transferred from C&A Operation to Aquatic Biology Operation on October 19.
J. C. Pekas, Biological Scientist, joined the Experimental Animal Farm Operation on October 26 as a new employee.
P. H. Ray, Biological Analyst, was temporarily removed from the payroll due to illness absence (injuries sustained in an automobile accident).

## B. TECHNICAL ACTIVITIES

### FISSIONABLE MATERIALS - 02 PROGRAM

Sera from 40 adult white fish, captured at Priest Rapids Dam, all tested negative for columnaris antibody, but 15 of the 20 scrap fish (14 suckers, 5 squaw fish, and 1 carp) were positive. These results suggest that scrap fish have an immune response to columnaris.

Adult chinook salmon (20 fish) exposed by the contact method for 90 minutes to a virulent strain of Chondrococcus columnaris died three days post exposure. To our knowledge, this is the first laboratory demonstration of acute death of adult salmon from columnaris infection. Previous laboratory work was with juvenile salmon.

### BIOLOGY AND MEDICINE - 06 PROGRAM

#### METABOLISM, TOXICITY AND TRANSFER OF RADIOACTIVE MATERIALS

Zinc

Autoradiograms of the GI tract of a rainbow trout with a body burden of 5.7 mCi $Zn^{65}$ indicated $Zn^{65}$ deposition principally in the fundic glands, the mucosal epithelioid cells (especially the basilar portions) and the smooth muscle of the mucosa. There was no tissue damage in the stomach and pyloric caeca, however, the hind-gut showed an increased number of goblet cells in the surface epithelium and a loose vacuolated appearance in the laminar propria.

Strontium

A 2½-year-old female miniature pig in the 125 μCi/day group became acutely ill, and because of her deteriorating condition was killed five days later. The body burden at death was 1000-1250 μCi and the



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accumulated radiation dose to the skeleton was ~15,000 rads. At first observation the leukocyte count was elevated and the erythrocyte count slightly depressed. During succeeding days, the leukocyte count increased to over 200,000 leukocytes/$mm^3$ and erythrocytes decreased to a packed red cell volume of 17%. (The normal leukocyte count at this age is ~10,000 cells/$mm^3$ and the normal packed red cell volume is ~40%.) The increase in leukocytes was due primarily to an increase in neutrophils including many very immature ("blast") forms.

The spleen was approximately five times normal size, apparently due to neoplastic involvement. The hepatic lymph nodes as well as other regional lymph nodes were markedly enlarged. Numerous foci of apparently neoplastic tissue were observed in the liver. On the basis of the hematologic and gross pathologic findings, the condition was tentatively diagnosed as "acute myelogenous leukemia". A final diagnosis will be made based on microscopic examination of the tissues.

This is our fifth observation of neoplasia of the hematopoietic tissue in miniature swine ingesting $Sr^{90}$. The four previous cases included three cases of "lymphosarcoma" in animals ingesting 125 μCi $Sr^{90}$/day and one case of myelogenous leukemia in an animal ingesting 25 μCi $Sr^{90}$/day. We have not observed any similar cases in control animals or animals ingesting 1 or 5 μCi $Sr^{90}$/day. Six similar cases have been observed in other laboratories in dogs with chronic feeding or injection of $Sr^{90}$. In all these cases $Sr^{90}$ administration started early in life. Dogs or miniature swine given single doses of $Sr^{90}$ have shown no neoplasia of hematopoietic tissue, but have, in contrast, shown an increased incidence of bone neoplasms. No bone neoplasms have yet been observed in miniature swine chronically ingesting $Sr^{90}$, and only two cases of bone neoplasms have been seen in dogs chronically ingesting $Sr^{90}$. The suggested relationship between type of effect observed and the mode of $Sr^{90}$ exposure will be evaluated further as these continuing studies progress.

## Plutonium

Partially hepatectomized rats were injected with $Pu^{239}$ one day or ten days following the surgical procedure. Deposition of plutonium in the liver was reduced to one-fourth normal levels in the animals injected one day after partial hepatectomy and was reduced to two-thirds normal levels in the animals injected ten days after partial hepatectomy. This was a larger and longer-lasting effect than noted in previous similar studies with $Ce^{144}$-$Pr^{144}$.

## Gastrointestinal Radiation Injury

An effort was made to measure directly the fate of $C^{14}$-labeled bile salts injected intravenously three days after 1500 r abdominal X-irradiation. These were preliminary studies involving considerable analytical difficulty - material balances were not good. However, there was a clear indication that about twice as much of the bile salts reached the large intestine in

the irradiated animals as compared to unirradiated controls. There was also a many-fold higher concentration of bile salts in the small intestine of control animals at 24 hours post-injection. These data provide direct qualitative support for earlier, more precise but less direct indications of a defect in bile salt reabsorption in the irradiated small intestine.

Inhalation Studies

Results of analyses for $U^{234}$, $U^{238}$, and $Th^{230}$ in rat lungs after multiple (two times weekly for eight weeks) exposures to uranium ore dust show a progressive accumulation of thorium, reaching levels 5 or 6 times that of uranium at two months post-exposure. Bronchial lymph nodes also accumulate twice as much thorium as uranium. These results confirm those obtained in earlier single exposure studies.

Two types of 3-way respiratory valves (a linear and a rotary action valve) have been designed and constructed for use in aerosol inhalation experiments. The valves, actuated by pressure transducers, allow collection of exhaled aerosols. The electronics were designed by the Instrument Research and Development Laboratory, HL. One of the valves has been tested successfully in the measurement of deposition and pulmonary clearance of $Cr^{51}$ oxide aerosols in dogs.

Studies on the biosynthesis of fatty acids by lung tissue are being extended to determine more precisely the experimental conditions for optimum *in vitro* synthesis and to obtain further evidence for the synthetic pathway.

Dimethyl sulfoxide (DMSO) and Triton (a wetting agent) were tested for their effectiveness in removing inhaled $Ru^{106}O_2$ from rats. DMSO reduced the $Ru^{106}$ lung burdens about 10% and will be tested further.

Plant Nutrition

The effect of concurrent addition of chloroamphenicol (CAP) and iodide upon rubidium and iodine transport in beans was tested. Results showed: (a) CAP had no effect upon transpiration, (b) reduced Rb absorption, and (c) enhanced iodide absorption. This indicates transport (not transpiration) is of major importance in governing salt absorption by the plants; the difference in response of I and Rb to CAP, as well as other observations, indicates different types of mechanisms regulate their absorption and these mechanisms are sensitive to other kinds of inhibition.

## Microbiology

Efforts to isolate indolepyruvic acid from filtrates of Neurospora were only partly successful. Mycelia grown at 20C failed to accumulate the compound for unknown reasons, although the specific activity of the transaminating system in 20 C cells was at least as high as that observed in cells grown at 30 C.

## Radiation Effects on Insects

Initial tests were performed with DDT to determine experimental levels at which to stress flour beetle populations under conditions of both insecticide and radiation. Results indicated: (a) wild types had comparable sensitivities to DDT; (b) sooty mutants were about ten times more sensitive to DDT than were wild types; and (c) older larvae placed on 0.02% DDT rations had a higher survival rate than did younger larvae.

## Columbia River Limnology

Concentrations of $P^{32}$ and $Zn^{65}$ averaged 42 and 69 nc/g organic matter, respectively. This is a threefold increase from August values in both cases. Organic matter per cubic meter of water decreased by the same factor during the period.

Concentration factors for $P^{32}$ and $Zn^{65}$ in plankton were about 83,000 and 19,000, respectively. Compared to last month's values, these represent a threefold increase for $P^{32}$, but an identical situation for $Zn^{65}$.

## Terrestrial Ecology

Barley seedlings grown in soils from beneath hopsage canopies without soil moisture stress contained 72-100 mg/g dry weight as compared to values of 52-60 mg/g for barley grown in soil taken from adjacent intershrub areas.

Alfalfa seeded into $Sr^{90}$ plots during late summer of 1963 was harvested six times during the 1964 growing season. Strontium concentrations increased slightly during the season, with no apparent effect of plant maturity upon the concentration. Current values are nearly identical to those found during 1958.

## Alaskan Studies

Preliminary calculations of biological half-life of $Cs^{137}$ in Anaktuvuk Pass adult Eskimos were made using whole-body burdens determined by the shadow-shield counter and $Cs^{137}$ intake calculated from dietary information. Substantial variation was found in the values derived, but generally indicated a half-time substantially shorter than the 100-150 days reported in the literature.

Whole-body counts of 12 Anaktuvuk Pass adult men were obtained by Radiological Physics Operation on 13 October, utilizing the three-inch crystal. Cesium-137 body burdens were found to be about 20% lower than those measured on September 9-10.

Fallout Studies

Collection of plant and animal samples continued. Partial summaries of past collections in several locations showed: (1) no reduction of fallout radionuclide concentrations with time; (2) an effect of seasonal patterns of precipitation; and (3) a striking change in radionuclide concentrations in animal samples associated with a seasonal shift to new plant growth in the animals' diet.

R. C. Thompson
BIOLOGY LABORATORY

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TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposia

Society for Experimental Biology and Medicine (Northwest Section Annual Meeting) - Seattle - October 24, 1964:

J. D. Berlin and C. C. Bowen - Flagellar initiation and centrioles in Albugo.

J.J.C. Hsieh - The mechanism of iodine gas absorption by leaves.

W. H. Matchett - Utilization of tryptophan by Neurospora.

M. P. Fujihara - The effects of radiation and $D_2O$ on cichlid fish.

D. H. W. Liu and W. J. Clarke - Sites of zinc deposition in the gastro-intestinal tract of trout.

B. J. McClanahan - The zinc-cadmium interrelationships.

V. H. Smith - Cooperative action of desferrioxamine-B with diethylene-triamine-pentaacetic acid in removal of plutonium from rats.

b. Seminars (Off-Site and Local)

L. K. Bustad and R. O. McClellan. Radionuclide toxicity in domestic animals. College of Veterinary Medicine, Washington State University, Pullman, Wash. October 30, 1964.

Radiation Protection Course - Richland, Washington:

B. O. Stuart - Effects of radiation on living tissue. October 21, 1964.
J. F. Park - HAPO Programs in Radiobiology. Review of Inhalation Toxicology programs. October 28, 1964.
R. C. Thompson - Current Research in the Metabolism Operation. Oct. 28.

c. Seminars (Biology)

Mary Dell Stienhardt, Department of Genetics, University of Washington - An inquiry into the mechanism of information transfer from DNA to messenger RNA. October 5, 1964.

C. E. Cushing - Productivity and radionuclide accumulation by periphyton. October 14.
J. R. McKenney - Transport of Ca and P in the Isolated Rat Intestine. Oct. 20.
H. E. Erdman - Dose Ratio of 250 kvp X-rays and 4.6 mev fast neutrons for induction of 50% dominant lethals. October 28.

d. Miscellaneous Lectures

None

II. Articles Published

a. Open Literature

Palmer, R. F. and R. C. Thompson. Strontium-calcium interrelationships in the growing rat. American J. Physiol. 207: 561-566 (1964).

Bair, W. J. Meeting - Inhaled radioactive particles and gases. Science 146: 440-444 (1964).

Noyan, A. Water absorption from the intestine via portal and lymphatic pathways in rats. Proc. Soc. Exptl. Biol. Med. 117: 317-320 (1964).

b. HW Documents

None

III. Visits and Visitors

a. Visits to Hanford

Mary Dell Steinhardt, Dept. of Genetics, University of Washington, Seattle, toured Biology facilities and presented a seminar. 10/5/64.

Seven members of the Irrigation Experiment Station, Prosser, toured facilities and attended the seminar. 10/5/64.

The Bio-Medical Program Directors Meeting was held in the 300 Area on October 12 and 13. Several attendees visited Biology on days following the meeting to review research programs with various investigators.

Dr. W. T. L. Neal, Atomic Energy Branch, Ministry of Agriculture, Fisheries and Food - United Kingdom, discussed food and agriculture problems of civil defense and effect of radiation on farm animals with the staff of the Experimental Animal Farm and others in Biology. 10/21/64.

Dr. G. H. Orians (Assoc. Professor of Zoology), William House and Henry S. Horn (Graduate Students), of the University of Washington discussed mutual research with members of Radioecology. 10/21/64.

Dr. and Mrs. O. Biddulph, Department of Botany, Washington State University, Pullman, discussed mutual research with F. P. Hungate and R. L. Uhler. 10/23/64.

About 25 members of the Western Washington Experimental Station, Puyallup, Washington toured the Biology facilities on October 26.



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b. Visits Off-site

10/1-3 - H. A. Kornberg and R. C. Thompson attended a Space Nuclear Propulsion Meeting at USNRDL in San Francisco.

10/3-12 - V. H. Smith attended a Symposium at National Institute of Health, Bethesda, Md. and discussed chelating agents with H. Kroll at Eltex Corporation in Providence, Rhode Island and the SNAP program with Wittenberg and Powers at Monsanto Research Corporation, Miamisburg, Ohio.

10/3-6 - L. K. Bustad visited Cornell University at Ithaca, New York at the request of the AEC to review program in progress.

10/13 - V. G. Horstman inspected feed at Pendleton Grain Growers, Hermiston, Oregon.

10/16 - L. K. Bustad presented a talk to the Washington Science Teachers Conference in Cheney, Washington.

10/23-25 - L. K. Bustad performed library work and attended the Meeting of the Soc. for Exptl. Biol. and Med. in Seattle.

10/23-25 - The following personnel from Biology attended the Eighth Annual Meeting of the Soc. for Experimental Biology and Medicine (Northwest Section) in Seattle: (papers presented are asterisked) -
H. A. Kornberg, M. P. Fujihara*, W. H. Hatchett*, J.J.C. Hsieh*, B. J. McClanahan*, D. H. W Liu*, V. H. Smith*, J. D. Berlin*, J. L. Palotay, F. P. Hungate.

10/21-24 - R. C. Thompson visited Ames Laboratory, Moffett Field, Calif. to discuss SNAP program with D. Kochendorfer and USNRDL, San Francisco for the same purpose.

10/26-30 - W. C. Hanson attended a Fallout Meeting in Washington, D.C. and the Information Meeting of the Health Physics Division in Oak Ridge, Tennessee.

10/26 - F. P. Hungate attended a Fallout Meeting in Washington, D.C.

10/26-31 - J. R. McKenney attended the Health Physics Division Review Meeting in Oak Ridge, Tennessee and discussed radiation studies with Dr. Fish.

10/27-30 - W. J. Bair attended the Health Physics Division Review Meeting in Oak Ridge, Tennessee and the Nominating Committee meeting of the Health Physics Society.

10/30-31 - L. K. Bustad and R. O. McClellan presented annual lectures in the course in Toxicology at the College of Veterinary Medicine, Washington State University, Pullman.

IV. Achievements

None

V. Honors and Recognitions

Capt. Robert Shannon (USAF), assigned to the Biology Laboratory, received the Air Force Commendation Medal for "superior professional ability, leadership, and devotion to his profession . . " in providing the Otis Air Force Base with an "outstanding veterinary medicine program".

VI. Professional Group or Organization Assignments

None

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# APPLIED MATHEMATICS OPERATION

## MONTHLY REPORT - - OCTOBER, 1964

### ORGANIZATION AND PERSONNEL

No change.

### ACTIVITIES FOR OTHER HAPO COMPONENTS

#### Contract and Accounting

A sampling plan was devised for obtaining an estimate with prespecified precision of the "percent utilization" of file cabinets within HL as of December 15, 1964.

#### Chemical Processing Department

Attempts to construct more realistic yet solvable mathematical models for the proposed counter-current crystallization process for recovery of aluminum nitrate from Redox acid wastes continues.

#### Irradiation Processing Department

During the installation of project CGI-966, Safety Circuit Trip Identification System, problems involving the interaction of row and column ground detectors were resolved through the study of a model assembled at the central maintenance shop in 100-H. Project acceptance tests were successfully completed at D reactor on October 30 following adjustment of the ground detector according to the instructions of "Invariance of Panellit Ground Detection Functions Under a Change of Supply Voltage," a letter from R. L. Richardson to the project engineer, dated July 20, 1964.

A procedure was developed for determining optimal numbers of process tubes to be measured in the continuing Alternate Tube-Corrosion Monitor Program.

An extensive set of power curves was developed to assist in the design of an acceptable run-to-rupture experiment to compare the HDS and AlSi fuel processes.

An analysis of data was carried out to compare two fuel-end bonding processes.

Work was continued on the reliability study of reactor pressure monitoring systems.

#### N-Reactor Department

The detection capability of a prototype X-ray fluorescent test was evaluated relative to uranium contamination in fuel end-closures.

N-Reactor Department (continued)

A screening study was carried out to eliminate outliers from the pre-irradiation fuel measurements which are to be used in a study of in-reactor fuel swelling.

Work was completed on a model and associated procedures appropriate to comparing pre-extrusion lubricating treatments of uranium billets.

A procedure (with examples) appropriate to the construction, maintenance, and interpretation of process control charts for fractions defective relative to several fuel characteristics was formulated.

## ACTIVITIES WITHIN HANFORD LABORATORIES

### 2000 Program

Flow System Models

The double precision version of the "shoe-box" program used in soil permeability studies has been modified to obtain the same information as previous versions. This additional information can be used in examining computational errors incurred in the various calculations performed by these programs.

C-Column Computer Control

Installation of the GE 412 process control computer in the Hanford van next to the 321 Building was completed. The computer is now operational, and machine debugging of functional programs is in progress. Work continues on the coding of the general diagnostic (equipment failure and peripheral failure routines) program.

Work continued on the statistical analysis of data from the calibration experiment for the organic photometer to be used to monitor organic uranium stream concentrations in the A-column test facility. Statistical analysis was completed of the calibration experiment for the new set of gamma absorptiometer sample cells for aqueous uranium stream concentration estimation in connection with the operation of the C-column.

### 4000 Program

Radiation damage to fissionable materials

Several meetings were held with personnel of Physical Metallurgy to consider the best way of implementing the use of newly developed quantitative metallographic methods of analyzing electron microscopy photographs. The problem of characterization of heterogeneous distributions in a matrix is also being considered.



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4000 Program (continued)

Analytical Laboratories

The initial mass spectrometer simulation study of the propogation of measurement errors was concluded. A more comprehensive study was requested and has been initiated.

Sonic Transmission in Visco-elastic Materials

The first draft of the mathematical analysis of the differential equation of particle motion in media having attenuation losses has been completed. This model has been employed as the basis for an EDPM program for numerically determining the spectral distribution of rebounding pulses caused by an initial pulse impinging on the plane boundary separating dissimilar media. Continuing efforts will be directed toward verification of the analysis and the EDPM program with experimental data.

Particle Packing Studies

R. L. Richardson, Mathematics Operation, was co-author of "Principal Variables in Vibrational Compaction," a paper presented at the U.S.-Japan International Exchange Meeting on Ceramic Nuclear Fuels.

The general solution to a mathematical model describing theory used in thermal neutron detection using electro-magnetic wave guides has been obtained for a specific geometry. Generalizations of the problem are now being investigated.

HW-84388, "Finding Precision Lattice Constants of Hexagonal Crystals by Using a Computer," has been issued as partial documentation of work done on $Pu_2O_3$ crystals.

5000 Program

Materials Research and Service

A statistical study related to the consistency of samples of graphite was initiated.

Computation and Statistical Analysis

A Fortran 4 program was written to perform double precision calculation of the critical points of the statistical test to determine whether or not the waiting time distribution between successive counts arises from a decaying radionuclide or from pure background. A Fortran program was also written and used to calculate the expected values of the $I^{th}$ order statistics for samples of size n from a normal distribution. The BLU system is nearing completion, and debugging efforts have increased. Data for the final system are being prepared and checked out. Statistical analysis of program data was done in preparation for the results presented at a program meeting here at Hanford during the month.

A new IRA program to plot decay curves was specified during the month. This program will provide additional information in analyzing samples and significantly reduce the work of hand plo-ting these curves.

Several modifications were made to the GEM program. The edit of raw spectra cards was improved with the result that fewer cases will be rejected and fewer input cards will be needed. The second modification provides for punching the results onto cards for entry in the IRA file.

Efforts continued to develop mathematical models explaining the diffusion of metals into metals.

6000 Program

Ecological Models

Revisions and additions to the EDPM atmosphere diffusion program are in progress.

Biology

The analysis of the data from the DNA study of mouse livers of radiation chimeras was completed.

An EDPM program was written to perform calculations on swine plasma data relative to the $SR^{90}$ toxicity study.

R. Y. Dean for
Manager
Applied Mathematics

CA Bennett:RYD:lg

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# RADIATION PROTECTION OPERATION REPORT FOR THE MONTH OF OCTOBER 1964

## A. ORGANIZATION AND PERSONNEL

R. W. Meisinger transferred from ES&EO to Technical Administration effective October 1. J. N. Morgan accepted permanent assignment in RMO effective October 5. R. W. Clark and G. T. Hewitt transferred from RMO to IPD effective October 26. C. Bruinekool and K. E. Hinrich transferred from IPD to RMO also effective October 26. C. A. Grosclaude terminated from RMO effective October 23. J. J. Jech transferred from RMO to ES&EO effective October 1. L. W. Brackenbush accepted a rotational assignment in RDCO effective October 1. J. N. Paglieri began a rotational assignment in Nuclear Health & Safety October 12.

## B. ACTIVITIES

### Occupational Exposure Experience

There was one new plutonium deposition case confirmed by evaluation of bioassay data during the month. The evaluation of additional urinalysis data resulted in fourteen employees and one past employee being reclassified as nondeposition cases. The total number of individuals who have received internal depositions of plutonium at Hanford was consequently reduced from 347 to 333, of which 233 are currently employed at Hanford.

There were nine incidents involving 12 employees which required special bioassay sampling for plutonium analysis; however, the results of these analyses are not yet available for evaluation. The following are brief descriptions of the more significant incidents:

> A CPD employee received a puncture wound at the base of the small finger, right hand, while working at the Plutonium Incinerator Building (232-Z). The employee was changing an exhaust filter in a hood when the metal edge of the filter penetrated the glove and skin of his finger. An examination of the wound, using the plutonium wound counter, indicated 0.03 μc plutonium at the wound site. A recount, following surgical excision at GE First Aid, indicated less than the detection limit of $1 \times 10^{-4}$ μc plutonium.
>
> A CPD employee cut his left thumb on a piece of broken glass while working in the Plutonium Purification and Fabrication Facility (234-5). The employee was performing analytical work on plutonium chloride. The contamination in the thumb was reduced from 400,000 d/m to ≈10,000 d/m and the employee sent to the Whole Body Counting Facility. An examination

using the plutonium wound counter, detected $3.5 \times 10^{-3}$ μc plutonium at the wound site. Re-examination, following surgical excision at Kadlec First Aid, indicated $1.5 \times 10^{-3}$ μc plutonium. The wound was flushed in DTPA solution.

A CPD employee received a puncture wound on his right index finger as a result of work in the Plutonium Purification and Fabrication Facility (234-5). The employee felt a slight pain in his finger as his gloved hand brushed against a piece of wire mesh screening. Skin contamination of 1000 d/m was reduced to 500 d/m and the employee sent to the Whole Body Counting Facility. An examination, using the plutonium wound counter, indicated $6 \times 10^{-3}$ μc plutonium.

Two CPD operators were exposed to air-borne concentrations of plutonium while working in the Plutonium Purification and Fabrication Facility (234-5). One employee sealed equipment and tools into a hood by way of a hood glove. A personnel survey at the end of the job indicated extensive skin contamination up to 10,000 d/m and nasal contamination of 13,500 d/m, smearable on this employee. Nasal contamination of ≈100 d/m was received by the second employee working nearby.

Two CPD employees were exposed to air-borne concentrations of plutonium while working in the Plutonium Purification and Fabrication Facility (234-5). The seal on a plastic bag containing plutonium oxide failed, resulting in skin contamination of 1,000 d/m and nasal contamination of 2,000 d/m and 350 d/m, smearable.

There were three plutonium contaminated injuries this month, all three requiring excision. These cases bring the total number of plutonium contaminated injuries for the year to 12, with 10 requiring excision. During the corresponding period in 1963 there were 20 plutonium contaminated injuries with 14 requiring excision.

In addition to the incidents involving plutonium, there was one incident involving four persons that required evaluation for possible intake of other radioisotopes. Four HL employees were exposed to air-borne concentrations of graphite dust while decontaminating a cask in the Radiometallurgy Facility (327 Building). Nasal contamination of 10,000 to 35,000 c/m was detected by nasal smears of the employees involved. Examinations, using the Whole Body Counter, detected trace amounts, much less than 1% of the maximum permissible body burden of 30 μc $Cs^{137}$.

There were three incidents requiring special evaluation of personnel exposures. Two of these involved localized exposures resulting from wearing contaminated clothing. The third incident involved dose

rates up to 70 R/hr, observed in the High Level Radiochemistry Facility (325A) when 9 liters of highly radioactive thorium nitrate entered a vacuum line leading out of the cell. The vacuum line was flushed, reducing dose rates from the line to 45 mR/hr; however, the dose rate at the shielded basement trap increased to 200 R/hr. Personnel exposures of the four persons involved were all less than 150 mR.

Environmental Experience

Work on the Gable Mountain Swamp to control contamination that entered the swamp on June 12 was essentially completed. Backfilling of the edges of the swamp buried the contamination in the near shore area and resulted in a rise in water level of about one foot. This action reduced radiation levels to a few mR per hour. The concentration of beta emitters in the swamp water had previously declined to about $10^{-5}$ μc/cc. Food plants for water fowl were either buried or burned. Ducks sampled during the latter part of the month had external contamination of less than 1000 c/m and $Cs^{137}$ concentrations in their flesh of about $10^{-4}$ μc/gm. The bottom of the swamp is still highly contaminated, however, and will likely remain so for a year or two.

The concentration of fallout materials in the air of the Hanford environs was about 0.3 - 0.4 pc $\beta/m^3$ during the early part of the month, but rose to a peak of 4 pc $\beta/m^3$ on October 22-23, following the atomic test carried out by the Chinese on October 16. The level had declined to about 1 pc/$m^3$ by the end of the month. No significant increase in the $I^{131}$ concentration in local air or milk was evident by the end of the month.

Nuclear Safety

Safeguards studies of PRTR operation at power levels up to 120 Mw are essentially complete. Assumed reactor nuclear excursions were simulated on the analog computer. These studies demonstrated that nuclear excursions during reactor operation at 120 Mw should not increase the hazard to the public above the present level. As would be expected, the energy release in excursions was greater with safety circuit trips set for 120 Mw operation than with the trips set for 70 Mw. In no case did the calculated fuel element surface temperature approach the melting point of the Zircaloy cladding. The only core melting predicted was near the centerline of the fuel element rods in the highest neutron flux region. Centerline melting of this type would not lead to cladding failure and, therefore, should not result in the release of fission products from the fuel elements.

Assistance was given the Engineering Development Operation in the preparation of a study of the conversion of the PRTR to an organic-cooled facility. This study was prompted by a request from the Commission. A preliminary evaluation of the effect on nuclear safety of changing

to organic coolant indicated that this could be done safely. Advice was requested later on safety aspects of installing an organic-cooled loop in the PRTR.

A meeting with personnel of the Research and Development Branch, Division of Reactor Development, Oak Ridge National Laboratory, Phillips Petroleum Company, Brookhaven National Laboratory, Argonne National Laboratory, and Hanford Laboratories was attended at Germantown on October 22. The meeting was called for the exchange of information on fission product release and transport, the Containment Systems Experiment, and Loss of Fluid Test. A summary of this information was presented to the Research and Development Subcommittee of the Advisory Committee on Reactor Safeguards on October 23.

Discussions with personnel of the Richland Operations Office of the AEC led to agreement on the content of the Operating Safety Limits for the PRCF. A document, HW-82329 REV, "Operating Safety Limits for the Plutonium Recycle Critical Facility," was forwarded to RLOO for approval and re-issuance.

A major revision of the Chemical Processing Department's "Hazards Evaluation Study", HW-71918, was initiated. At month's end all contributions by Nuclear Health and Safety to this study were completed and are now in the process of being reviewed. The major change in the revision of this study is in the area of accidental processing of short-cooled fuel. The previous study used 30-day cooling, the present study assumes two-day cooling, for similar tonnage.

A hazards review of the shipment of irradiated N-reactor fuel elements to an offsite processor was initiated. A hazards review is in progress on the in-tank solidification and the storage of chemical processing wastes.

Consultations

Assistance was provided to laboratory groups in the preparation of Building Emergency Plans for the 308, 328, 3702 and 3705 Buildings.

Personnel exposure data were reviewed with management members of Occupational Health of Relations and Occupational Health Operation, Ceramics Research and Development of Hanford Laboratories, and Manufacturing Section of Irradiation Processing Department.

The dose rate in the control room of N-reactor was calculated at the request of Nuclear Health and Safety, when an adjoining room was filled with a specified concentration of fission gases.

Consultations were held with members of Non-Destructive Testing and Methods on beta-gamma fission product ratios as a function of reactor discharge time.

Assistance was given to Biology personnel in the use of the thermoluminescent reader for readout of TLD's used in animal exposures.

Several meetings with IPD design personnel were held concerning air sampling and storage problems in the $ThO_2$ target element hood in 3732 Building.

Studies and Improvements

Efforts to identify the source and eliminate contamination from the 308 Building air exhaust system are continuing. The final building exhaust filter was inspected and the same conditions were found as reported for the other two filters. Filter samples of the air exhausted from the building indicate a concentration of $5 \times 10^{-15}$ μc Pu/cc or a weekly release of about 0.2 μc Pu. The hood exhausts from Rooms 131 and 138 are presently considered to be the source of the contamination. The maximum concentration measured in these exhaust lines during the month was $3.9 \times 10^{-11}$ μc Pu/cc. A particle size analysis was completed on samples of the air exhausted from the hoods in Rooms 131 and 138. The mass median diameter of the contamination on the two samples was 0.39 μ and 0.57 μ. Contamination on a filter sample taken in April of 1963 had a MMD of 0.46 μ. An alpha energy analysis of the recent samples indicated 5% of activity of $Am^{241}$ in one sample and 16% by activity in the other sample. Since the two samples were definitely different in source, the theory that contamination from one source has diffused through the exhaust system was rejected.

A number of plutonium samples were used for surface dose rate measurements. The samples contained different compositions of plutonium with the amount of $Pu^{240}$ being stated as 3%, 8%, 9%, 16% and 26%. Four samples of $PuO_2$-$UO_2$ were also measured. All measurements were made through 12 mils of plastic.

In conjunction with these measurements, a sample of plutonium metal (503 gm) was obtained. The metal measured 1.3 rad/hr with a 12 mil plastic bag. The 12 mils of plastic attenuates by about 30%. Applying this correction factor yielded a dose rate of 1.7 rad/hr. This is in excellent agreement with the calculated dose rate (1.657 rad/hr). This same piece of metal is to be oxidized and returned for further measurement. A summary of these results is given in Table I, "Surface Dose Rates of Some Plutonium Materials." All dose rates given in Table I were corrected for the absorption of the 12 mils of plastic.



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TABLE I

Surface Dose Rates of Some Plutonium Materials

| Material | % $Pu^{240}$ | % $PuO_2$ | Rad/hr |
|---|---|---|---|
| PuAl-20 w/o | 8 | 0 | 0.62 |
| | 16 | 0 | 2.0 |
| | 26 | 0 | 3.4 |
| $PuO_2$-$UO_2$ | 8 | 1 | 0.21 |
| | 8 | 3 | 0.26 |
| | 8 | 5 | 0.33 |
| | 8 | 10 | 0.42 |
| | 3 | 8 | 0.25 |
| | 3 | 16.5 | 0.33 |
| | 3 | 27.3 | 0.40 |
| Pu | 9 | 0 | 1.7 |

All process radiation monitoring equipment drawings for Project CAH-916, Fuels Recycle Pilot Plant, were reviewed and approved for construction. While these are primarily process control instruments, they will be extremely useful to radiation monitoring personnel particularly during in-cell maintenance and decontamination. RDCO was also requested to engineer an additional $I^{131}$ sampling system for installation in the process off-gas system.

Two CP's were exposed to high energy gamma (~6 Mev) generated by the positive ion accelerator using the F(p,α)0 reaction. The purpose of the experiment was to determine whether or not the CP fulfilled the Bragg-Gray requirements at these energies. To perform the experiment, sleeves of phenolic were placed over the CP with readings taken with each successive sleeve. It was concluded that the CP does fulfill the stated requirements to within the +10, -5 per cent calibration accuracy.

The single channel gamma spectrometer was calibrated using the X-ray diffraction machine in the 326 Building. The energies used were: Cu, $K_{av}$ - 8.05 Kev; Ag, $K_{av}$ - 22 Kev; Mo, $K_{av}$ - 17.5 Kev. The resolution was about 40% for full width half maximum. The resolution is the same for energies from 5 Kev to 0.50 Mev. There may be a malfunction in the analyzer since the resolution is expected to improve as the energy hardens.

The K-fluorescent sources in Calibrations were analyzed and the energies now assigned appear to be very reasonable and the stated energy is the

predominant energy in each case. The Cu K-fluorescent source was found to have some impurity which generates an appreciable radiation in the 20-30 Kev energy region in addition to the primary emission at 8 Kev.

Background buildup due to radionuclide contamination of the detector was effectively controlled by continuous addition of NaEDTA solution to the water stream entering the scintillator. This is accomplished with a small oscillating pump operating at a pumping rate of about 1 cc per minute.

The "final design" electronics for the monitor was installed at the Richland Water Plant, and calibrated early in the month. It has performed satisfactorily. The installed system was found to be about 20% more sensitive than the prototype system.

An improved electron beam sweeping device was designed and fabricated for the electron Van de Graaff. Faster switching of the plus and minus 5,000 volt beam sweeping potentials is accomplished through use of magnetically closed reed switches in place of the mechanical relays previously used. This modification provides better control over the equipment when short bursts of high intensity radiation are desired.

Particle size analysis was completed on two routine filter samples taken at the Weapons Fabrication Facility (234-5) during an internal exposure incident on October 6-7. The mass mean diameter was found to be about 10 microns with a number mean diameter of about 0.2 microns. The gross amounts of activity would not permit detection of particles below about 0.2 microns. About 90% of the activity was found on particles that were 3 microns or larger.

The positive ion accelerator was used for several low level neutron exposures to thermoluminescent dosimeters. Multiple sets of thermoluminescent dosimeters were exposed to 0.2, 0.4, 0.6 and 0.8 rads of fast neutrons of 0.8, 1.0 and 1.4 Mev. The dose was determined from the first collision RBE dose curve. The exposures to 0.8 and 1.0 Mev neutrons were done with the tritium (p,n) reaction which has a very low gamma yield. The exposure to 1.4 Mev neutrons was accomplished with the carbon (d,n) reaction which has a very high gamma yield.

The results after readout of the TLD's were not as good as expected. After exposure to the $PuF_4$ and PuBe neutron sources, the TLD's seemed to be more sensitive. The minimum sensitivity seemed to be about 0.1 rads. For the accelerator exposures, the minimum sensitivity appears to be about 0.5 rads, using the present dosimeter configuration. The problem may be the proper separation of the neutron and gamma responses. Results from the 0.2 and 0.4 rad exposures were not positive for the exposures to 0.8 and 1.0 Mev neutrons. An exposure to 0.4 rads of 1.4 Mev neutrons yielded a positive result. In all cases, 0.6 and 0.8 rad exposures yielded positive results though the magnitude of the results were not always in the correct proportion.

1235103

The scheduling by job code and suffix of personnel for examination in the mobile whole body counter was instigated during the month. This scheduling includes the printing out by Data Processing of a punched card for each man who is to be examined in a particular manufacturing or laboratory area. The card format includes name, payroll number, suffix, job code and date that the employee will report for examination. Of the 200 personnel in the 100-D Area who were scheduled for examination during the month, 80% were examined as scheduled. Make-up examinations are planned during the first week of November for the remaining personnel. Nine personnel had small transient depositions of $Sc^{46}$, $Fe^{59}$, $Co^{60}$, and $Ar^{41}$.

The mobile whole body counter was exhibited in the 300 Area for visiting scientists attending the meeting of the Bio-Medical Program Directors of the U. S. Atomic Energy Committee.

All equipment was ordered for building two new field wound counting systems. Most of this equipment was delivered with the exception of the electronic gear and detectors using 1/4" x 1 mm crystals. The system for use in the 700 Area First Aid should be ready for use before year's end. This includes a counting system consisting of a "large" (1-3/4" x 1 mm) and a small (1/2" x 1 mm) wound counter for low-energy X-rays from plutonium or americium and two small geiger-type probes for higher energy $\beta$-$\gamma$ detection in wounds. The smaller crystal-type counters (1/4" diameter) will be added as they are received from the vendor.

## Research Studies

### Effect of Reactor Effluent on the Quality of Columbia River Water

Measurements continued on river water temperatures and of meteorological parameters that influence them. A small barge, supporting a temperature recorder, was anchored in the river just upstream from the 300 Area at a location which previous surveys have shown to give a temperature representative of that of the cross-section of the river at this point.

### Mechanisms of Environmental Exposure

Eleven volunteers continued to partake of Columbia River fish in a controlled experiment to study the uptake and retention of $P^{32}$ and $Zn^{65}$ from fish. The body burden of $P^{32}$ has leveled off at approximately 170 nc at an ingestion rate of 200 grams of fish per week, although there are significant variations among the subjects.

As part of the Columbia River fish ingestion experiment, creatine analyses were made on the subjects' urine to ascertain the utility of this substance in establishing the metabolic activity represented

by a single sample of urine. The results indicated that, for a urine sample from an individual for whom no previous excretion data are available, a creatine analysis is a somewhat more reliable index of metabolic activity than is sample volume.

Nuclear Facility Monitoring Guide

Progress on the monitoring guide was reviewed with other staff members, and new suggestions are being incorporated. Members of Chemical Effluents Technology are collaborating on the preparation of a section that deals with effluent monitoring.

C. TRAINING

A field emergency entry and rescue problem was set up for the 300 Area firemen. The training included the use and reading of dose rate instruments in an actual gamma field.

Four more classes were held in the course, "Radiation Protection Training for Exempt Personnel." This makes a total of 7 classes completed in the 11-class series. Attendance was excellent. Comments of the attendees were most favorable.

In the "Second Radiation Monitoring Operation Refresher Course," all of the sessions on "Effects of Radiation on Living Tissue" were held, and four of the five sessions on "Elementary Nuclear Physics." At the request of K. R. Heid, an additional subject, "Radiological Civil Defense," was added to the refresher course. The course was rearranged to permit completion of the Civil Defense sessions before the end of the year.

A four-day orientation program was organized and completed for S. W. Porter of the Armed Forces Radiobiological Research Institute, Bethesda, Maryland. Mr. Porter was given the opportunity to see as much as possible of the radiation protection activities in the production departments and Hanford Laboratories.

D. SIGNIFICANT REPORTS

HW-80892-8, "Radiological Status of the Hanford Environs for September 1964," R. F. Foster.

HW-84482, "Radiation Monitoring Operation Monthly Report, September 1964," A. J. Stevens.

AR Keene
Manager
RADIATION PROTECTION

# FINANCE AND ADMINISTRATION OPERATION

## ACCOUNTING

### Cost Accounting

The Hanford Laboratories' control budget was adjusted during the month as follows:

1. Control budget for the Maintenance and Equipment Engineering of Test Reactor and Auxiliaries Operation was increased $15,000 to more accurately reflect the level of service being provided other components of Hanford Laboratories.

2. The increased funds for the Containment Systems Program included in the Financial Plan were reflected as R&D in September. This was changed in October and will be reflected in the expense project - $131,000.

3. The increase of $100,000 for General R&D - Civilian in the Financial Plan was incorrectly included in the Conceptual Reactor Design Program in September. This was corrected to show the increase in Specific Fuel Cycle Analysis.

The FY 1965 Midyear Budget Review call letter was received from Contract Accounting on October 26. Schedule provides for budget review to be completed and submitted to Contract Accounting by November 23, 1964.

Equipment funds allocated to Hanford Laboratories for FY 1965 can be summarized as follows:

| | |
|---|---|
| 02 Program | |
| General Purpose | $400 000 |
| HL Programs | 50 000 |
| 03 Program | 105 000 |
| 04 Program | 790 000 |
| 05 Program | 300 000 |
| 06 Program | 275 000 |
| 08 Program | 40 000 |

The HL allocation for miscellaneous capital work orders aggregates $200,000.



1235106

New organization codes established during the month were:

7140 - Purchasing Operation
7352 - Maintenance Standards & Procedures

The following 04 Program codes were established during the month:

.31 - Gaseous Effluent Studies
.53 - Reactor Safety Analysis & Evaluation

Special accounting codes for customer work were established during October as follows:

| | | |
|---|---|---|
| .5K | - Saxton Plutonium Fuels | $ 70 400 |
| .5L | - Development for Picatinny Arsenal | 100 000 |
| .2V | - $UO_2$ Compaction for APED | 300 |

Additional authorizations were received for the following special requests:

| | | |
|---|---|---|
| .3P | - Cermet Fuel Fabrication | $125 000 |
| .3R | - Cermet Joining Studies | 25 000 |

General Accounting

Approval letter No. AT-364, AEC Monograph on Plutonium - Its Industrial Hygiene Aspects - Dr. R. C. Thompson, is in process.

OPG 7.9, Control of Secret Documents, Page 17, was revised and OPG 3.1.3, General Electric Educational Assistance Program, was canceled.

Hanford Laboratories' net material investment at October 1, 1964 totaled $27.9 million as detailed below:

| | | | (In thousands) |
|---|---|---|---|
| SS Material | | | $26 311 |
| Reactor & Other Special Materials | | | 1 296 |
| Spare Parts | | | 366 |
| Exotic Materials | | | 56 |
| Yttrium | | | 26 |
| Subtotal | | | 28 055 |
| Reserve: | Spare Parts | $86 | |
| | Yttrium | 26 | (112) |
| Net Inventory Investment | | | $27 943 |



1235107

The cumulative value of nuclear material consumed in research by Hanford Laboratories during FY 1965 (at October 1, 1964) is shown below:

| | |
|---|---|
| 02 Program | $ (1 581) |
| 03 Program | 172 942 |
| 04 Program | (327 290) |
| Total | $(155 929) |

The credit in the nuclear material consumed in research account is due primarily to return of material to Redox at full value.

A report of findings in connection with the quarterly inventory of Reactor and Other Special Materials in the custody of 120 Hanford Laboratories' material holders was issued during the month. This was a certification type inventory except for the holders of materials within Chemical Laboratory (7600), which was a witnessed type inventory by HL Financial personnel. The inventory revealed only minor variances attributed to rounding. An adjustment was made for quantity only.

The status of Hanford Laboratories' heavy water inventory at October 31, 1964 is as follows:

| | Current Month | | FY to Date | |
|---|---|---|---|---|
| | Pounds | Value | Pounds | Value |
| Beginning Balance | 59 041 | $800 660 | 32 762 | $447 572 |
| Acquisitions | | | 27 445 | 379 015 |
| Scrap Returns (SROO) | | | | |
| Consumption -1) | | | | |
| PRTR: Loss | (2 157) | (29 816) | (3 323) | (45 940) |
| Scrap | | (7 158) | | (16 961) |
| Ending Balance -2) | 56 884 | $763 686 | 56 884 | $763 686 |

(1- Consumption - Scrap reflects amount charged to Cost only.

(2- Includes 15,398 pounds of heavy water scrap valued at $190,294.

Loss amount shown above (2,157 pounds, $29,816) represents actual for September and October based on physical inventories taken on October 16, 1964 and October 26, 1964 covering period September 1, 1964 through October 26, 1964. Loss amount shown on September 30 report was an estimated amount since no inventory was taken on September 30, 1964.



1235108

Laboratory Storage Pool activity is summarized as follows:

| | Current Month | | FY to Date | |
|---|---|---|---|---|
| | Quantity | Value | Quantity | Value |
| Beginning Balance | 1 946 | $1 556 640 | 2 027 | $1 493 571 |
| Items Received | 82 | 244 167 | 479 | 411 036 |
| Items Reclaimed by Custodians | (27) | (498 454) | (237) | (521 333) |
| Equipment Transfers | (43) | (28 375) | (153) | (76 145) |
| Items Disposed by PDR | (7) | (5 441) | (38) | (11 290) |
| Items Disposed by Excess | (22) | (70 547) | (149) | (97 849) |
| Ending Balance | 1 929 | $1 197 990 | 1 929 | $1 197 990 -1) |

(1- Includes 149 items valued at $81,025 on loan at October 31, 1964.

| Loans & Transfers in Lieu of Purchases | Current Month | | FY to Date | |
|---|---|---|---|---|
| | Quantity | Value | Quantity | Value |
| Loans | 17 | $ 7 624 | 62 | $ 32 622 |
| Transfers | 43 | 28 375 | 153 | 76 145 |
| Total | 60 | $ 35 999 | 215 | $ 108 767 |
| Operating Costs (9-30-64) | | $ 1 325 | | $ 4 296 |

Laboratory Storage Pool material and equipment at October 31, 1964 totaled $2.1 million as detailed below:

| | |
|---|---|
| Equipment | $1 197 990 |
| Reactor & Other Special Materials | 260 035 |
| SS Material | 154 800 |
| Other Material | 471 127 |
| Exotic Material | 57 506 |
| Total | $2 141 458 |

Transfers from Work in Progress accounts to completed Plant accounts totaled $347,525 for the month of October. Building 3746-D, valued at $3,481, was transferred from IPD and eleven 614 Monitoring buildings, valued at $33,433, were removed from record by means of a PDR. A total of 1,887 input documents were prepared and submitted to EDP during October for additions and changes to HL Plant and Equipment records.



1235109

The following contracts were processed in October:

| | |
|---|---|
| CA-472 | Robert W. Albrecht |
| CA-473 | Kermit L. Garlid |
| CA-474 | Lewis W. Seagondollar |
| CA-475 | Selmer W. Peterson |
| CA-477 | James G. Knudsen |
| SA-370 | Jonas Ahgook |
| SA-371 | Pyromet Company |
| SA-374 | PUD No. 1 of Franklin County |
| MRO- 80 | Fisher Scientific Company |

Personnel Accounting

Suggestion Awards for October, which amounted to $890, were paid to 29 employees.

Patent Awards were made to the following:

| | | |
|---|---|---|
| W. W. Schulz | HWIR 1614 | A Process for Dissolving Irradiated Pu-Al Alloy Fuel |
| H. L. Libby | HWIR 1548 | Eddy Current Nondestructive Testing Device with Graphical Nulling Feature |
| H. L. Libby<br>C. R. Wandling | HWIR 1650)<br>HWIR 1650) | An Eddy Current Tubing Tester with Signal Pattern Display and Graphical Nulling Features |

Helen F. Knight and Carl H. Hemphill retired effective November 1, 1964.

Personnel statistics follow:

| Employee Changes | Total | Exempt | Nonexempt |
|---|---|---|---|
| Employees at beginning of month | 1 811 | 790 | 1 021 |
| Additions and transfers in | 79 | 17 | 62 |
| Removals and transfers out | 42 | 6 | 36 |
| Employees at end of month | 1 848 | 801 | 1 047 |



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| Overtime Payments During Month | October | September |
|---|---|---|
| Exempt | $ 4 979 | $ 4 183 |
| Nonexempt | 29 442 | 32 152 |
| Total | $ 34 421 | $ 36 335 |

| Gross Payroll Paid During Month | | |
|---|---|---|
| Exempt | $ 833 657 | $ 816 549 |
| Nonexempt | 601 525 | 726 187 |
| Total | $1 435 182 | $1 542 736 |

| Participation in Employee Benefit Plans at Month End | October | | September | |
|---|---|---|---|---|
| | Number | Percent | Number | Percent |
| Pension | 1 751 | 99.3 | 1 706 | 99.3 |
| Insurance Plan - Personal | 388 | | 384 | |
| - Dependent | 1 451 | 99.6 | 1 419 | 99.7 |
| U. S. Savings Bonds | | | | |
| Stock Bonus Plan | 161 | 33.7 | 158 | 34.5 |
| Savings Plan | 71 | 3.8 | 70 | 3.9 |
| Savings & Security Plan | 1 198 | 86.9 | 1 185 | 87.5 |
| Good Neighbor Fund | 1 330 | 71.7 | 1 297 | 71.5 |

| Insurance Claims | Number | Amount | Number | Amount |
|---|---|---|---|---|
| Employee Benefits | | | | |
| Life Insurance | -0- | $ | 1 | $15 502 |
| Weekly Sickness & Accident | 8 | 639 | 10 | 835 |
| Comprehensive Medical | 49 | 2 874 | 45 | 3 486 |
| Dependent Benefits | | | | |
| Comprehensive Medical | 83 | 7 437 | 90 | 8 863 |
| Total | 140 | $10 940 | 146 | $28 686 |

## TECHNICAL ADMINISTRATION

Forty-three nonexempt employment requisitions were filled; 65 remain to be filled.

Suggestion Plan activity:

| | |
|---|---|
| Suggestions received | 40 |
| Suggestions adopted | 27 |
| Suggestions rejected | 35 |
| Suggestions in process | 87 |



1235111

Visitors Center activity:

| | |
|---|---|
| October attendance | 1 569 |
| Average attendance per day open | 58 |
| Cumulative attendance since 6-13-62 | 81 107 |
| Conducted groups | 12 (totaling 227 people) |

Plant tour activity:

| | Number | Total People |
|---|---|---|
| General Public Relations Tours | 8 | 124 |
| Special Tours | 3 | 85 |

Overall recruiting results for October are as follows:

| | |
|---|---|
| Offers extended | 6 |
| Offers accepted | 5 |
| Offers rejected | 4 |
| Added to roll | 6 |

Advanced Degree - Five Ph.D. applicants visited HAPO for employment interviews. Five offers were extended; three acceptances and three rejections were received. Three offers are currently open.

BS/MS (Direct Placement) - One offer was extended. There were two acceptances and no rejections received. There are no offers currently open.

BS/MS (Program) - No offers were extended. No offers were accepted, and one offer was rejected. There are no offers currently open.

Technical Graduate Program - Five Technical Graduates were placed on permanent assignment. One new member was added to the roll and none terminated. Current Program numbers 35.

The Hanford Nuclear Program will be divided beginning November 1. Of the 28 who have not yet selected off-program placement, 18 will remain under the Laboratory Program and 10 under the new GE Program operated by NRD.

The transfer of the GE-AEC business records is proceeding aided by the efforts of eight temporary employees. Two-thirds of the transfer is now completed.



1235112

## FACILITIES ENGINEERING

At month end, Facilities Engineering Operation was responsible for 13 active projects having total authorized funds in the amount of $10,729,000. The total estimated cost of these projects is $11,880,000. Expenditures through September 30, 1964 were $5,066,000.

The following summarizes project activity in October:

| | |
|---|---|
| Authorized projects at month end | 13 |
| New projects authorized | 0 |
| Projects completed | 0 |
| New projects submitted to the AEC in October | 0 |
| New projects awaiting AEC approval | 2 |
| CAH-123, Laboratory Fire Protection System | |
| CAH-155, Services for Biology Laboratory Facilities - 300 Area | |
| Project proposals being prepared | 5 |
| Geological and Hydrological Wells - FY 1965 | |
| Power Supply - Million Amp Welder | |
| PRTR Increased Power Level | |
| Shielded Creep Test Facility - 3707-C Building | |
| Experimental Facility Addition - 309 Building | |

The status of active projects follows:

### CAH-100 High Temperature Lattice Test Reactor

Detailed design is 94 percent complete compared to a scheduled 100 percent. Several Phase II drawings have been reviewed and commented on. Phase II drawings of the PMARS package and the two heat exchangers were approved. The Phase I construction contract was awarded to George Grant on October 23. The Commission considered a proposal that the Company design and fabricate a completely computerized instrument system. The Commission decided that Vitro should perform the design and a vendor fabricate the control system.

### CAH-116 PRTR Decontamination and $D_2O$ Cleanup

Construction is about 1 percent complete. The Commission authorized J. A. Jones Construction Company $72,000 for procurement and assembly of the $D_2O$ cleanup system. J. A. Jones plans to fabricate and assemble the entire



1235113

package. The Company requested early procurement of some components of the PRTR decontamination system to permit decontamination of the primary heat exchanger. Construction funds have not been authorized for the PRTR decontamination portion of the project.

CAH-119 PRTR Storage Basin and Experimental Facilities Modifications

Construction is approximately 16 percent complete and is on schedule. Construction work is progressing on that portion of the work for which approved design exists (all phases except heating and ventilation). J. A. Jones submitted a cost estimate of $267,000 for construction of the building and basin addition portion of the project, not including underwater equipment.

The construction specification was completed and approved on October 12, 1964. Comments of the electrical and heating and ventilating drawings were transmitted to the Commission on October 28. Procurement specifications for the underwater equipment were prepared by the Company and are awaiting Commission approval. Excavation for the new storage basin has progressed to the minus nineteen (-19) foot level.

CAH-123 Laboratory Fire Protection System

No action has been taken by the Commission.

CAH-126 Waste Transport System

Design is 87 percent complete compared to a scheduled 90 percent. Three drawings were received for review and comment.

CAH-136 Service Addition - 327 Building

Design is approximately 95 percent complete compared to a scheduled 100 percent. Eighteen drawings were received for review and comment. Comments were transmitted to the Commission.

CAH-137 Temporary Physical Sciences Center

Design is about 85 percent complete. The Commission authorized Vitro an additional $2,000 for a new total of $9,500 for performance of design.

Construction is 65 percent complete. Boiler room demolition and removal work is complete. The new electric boiler and hot water heater are in place. New electrical and piping services are being installed. The PABX telephone switchboard was received by the Company.

CAH-146 Atmospheric Physics Building

The Commission approved the design criteria document on October 8, and are negotiating a design contract with the engineering firm "Bouillon, Christofferson and Shairer."

CAH-151 Office Addition to the 308 Building

Design is 93 percent complete compared to a scheduled 100 percent. The Company returned its design comments to the Commission. The specifications have not been issued for comment.

CAH-153 Plutonium Recycle Critical Facility, Irradiated EBWR Fuel Handling

Detailed design is approximately 40 percent complete and on schedule. Comment prints were issued on the structural steel and the fuel cask on October 28.

CAH-155 Services to Biology Facilities - 300 Area

The project proposal is being revised to coincide with changes made in projects CAH-977, CAH-982, and this project by the Commission.

CAH-916 Fuels Recycle Pilot Plant

Construction is 82 percent complete compared to a scheduled 73 percent. The upper wall sections of the Metallurgical, High Bay and Mechanical Cells, and the ceiling of the Low Bay Cell, are being formed and poured. Shielding and piping are being placed in the ceiling of the Metallurgical Cell. Interior walls in building areas 2 and 3 are being plastered. Building cranes have been installed in the Cold Canyon and mock-up area.

CAH-962 Low Level Radiochemistry Building

Construction is 32 percent complete compared to a scheduled 36 percent. Approximately half of the precast concrete wall panels were ruined when they stuck to the casting slabs. Replacement slabs have been poured. Basement and outside piping are being installed. Sheet metal ventilation ductwork is being installed in the basement and main floor. The exhaust stack was erected and metal roof decking installed. Insulation is being placed on the roof. Interior concrete block walls are being constructed in the basement. The contract was extended 39 days as a result of the pipefitters strike.



1235115

CAH-977 Facilities for Radioactive Inhalation Studies

A revised design criteria document was submitted to the Commission on October 8 for review and approval. The Commission has questioned the propriety of services being provided by project CAH-155. The project scope is being revised.

CAH-982 Additional Radionuclide Facilities

Preparation of a revised design criteria document has been deferred pending a decision by the Commission on the scope of this project and projects CAH-977 and CAH-155.

Engineering Services and Plant Engineering

Engineering work was performed in support of design and construction on active projects, project proposals, preliminary planning and design criteria for new projects. Principal work items included: (1) review of Vitro's design on heating the reactor, the Phase II electrical drawings and computer design for Project CAH-100, High Temperature Lattice Test Reactor; (2) review of drawings for Project CAH-116, PRTR Decontamination and $D_2O$ Clean-up, Project CAH-136, Service Addition - 327 Building, and Project CAH 151, Office Addition - 308 Building; (3) completed specification for the procurement of the PRTR fuel element underwater cutting and examination equipment for Project CAH-119, PRTR Storage Basin and Experimental Facilities Modifications, and reviewed the heating and ventilation systems on this project; (4) developed a criterion for the power sources and control of computer room air conditioning for Project CAH-137, Temporary Physical Sciences Center; (5) issued five drawings for comment on the design for the on-site shipping cask on Project CAH-153, PRCF Irradiated Fuel Handling System; (6) prepared a design change to modify the building alarm systems on Project CAH-916, Fuels Recycle Pilot Plant, and revised the locations of motor control centers, and reviewed vendor submittals for miscellaneous mechanical items for Project CAH-962, Low Level Radiochemistry Building; and (7) approved 18 Vitro drawings on the Containment Systems Experiment and reviewed 34 Chicago Bridge and Iron drawings for the containment vessel.

Facility planning and budget study work included: (1) a draft of a design criterion for a shielded creep test facility at 3707-C Building has been prepared; and (2) a scope drawing and criterion have been issued to the on-site A-E for the design of a maintenance shop addition to the north side of the 314 Building.

Engineering and consulting work was provided to research and development personnel as requested. Major work items included: (1) a request has



1235116

been received for engineering assistance in developing equipment for use in fuel element studies for the Hot Cell in Building 292-T; (2) a drawing has been made and other drawings revised to show head tank piping for the Waste Calcination program in FRPP, also a sample load-out facility for the Low Bay Cell has been started; (3) at the request of AEC, additional information and cost estimates are being prepared for alternate installations of a million amp welder being purchased by Hanford Laboratories; (4) design and purchase specifications for an air receiver and pressure switches have been completed for the modification of the 141-F breathing air system to provide storage capacity in the event of a power failure; and (5) a drawing was completed for the installation of a helium storage balloon installation in the 314 Building.

Plant engineering work during the month included: (1) a review was made of methods of improving the air flow conditions in the 306 Building Pilot Plant area. The health hazards involved are variable but could be improved by complete isolation of the area in question; (2) the Vitro drawing was approved for the 300 Area classified scrap incinerator. The specifications were reviewed and minor comments submitted to Vitro; (3) electrical power source and control scheme were developed for the new standby air compressor recently installed in the 340 Building; (4) the requirements for a public address system for 314 Building were reviewed. A purchase requisition and specification for this equipment were completed; (5) proposed modifications to the unused 2709-E were scoped and estimated to convert this building into a maintenance facility; (6) the laboratory portion of 189-D Building is being studied with the intent of improving the arrangement of existing equipment and of providing change room, lunch room, and office facilities; and (7) a scope and estimate were prepared for additional office space for the 3718 Building.

Consulting services for other departments included assistance on: studied means of accomplishing crib waste effluent monitoring of FRPP; provided design for an electric motor drive on an existing overhead door in the 306 Building; further work was performed on a study to provide an alternate source of power for N-Plant; recommendations were made to CPD concerning the new Proportional Control System for the Hood Dry Air Supply System; and consulting service is continuing on a study of the excessive heat problem in 109-N.

Pressure Systems

Three bids were received for a High Pressure Furnace Prototype to be installed in the 308 Building, and the bids require further negotiation to complete the purchase.



1235117

Fabrication has started by J. A. Jones on the Liquid Sodium Loop to be installed in the 314 Building. Plant forces will fabricate the piping and instrumentation. The third-party inspector reviewed the design of the loop and inspected some of the equipment.

Fabrication was also started by J. A. Jones on the PRTR Corrosion and Film Loop to be installed in "B" Cell. It is 10 percent complete. Facilities Engineering is performing field inspection and providing welding and fabrication records for the documentation of the system.

Containment Systems Experiment

Field work is on schedule and within estimated cost amount. The canyon space is ready and accessible to the lump sum contractor for field erection of the containment vessel. The lump sum contractor will be on site during the latter part of November.

Design is continuing by Vitro for utilities, laboratory, and associated equipment used as a part of this experiment. The design effort is being controlled to best suit field or procurement requirements.

Containment vessel design and fabrication is progressing with slight delays incurred by procurement difficulties in steel delivery; however, this is minor to the overall schedule. Design is essentially complete and shop fabrication has started.

Simulator vessel negotiations with Struthers-Wells Corporation have been completed to the satisfaction of General Electric Company and J. A. Jones Company. The AEC is presently reviewing and approving the formal papers. It is estimated that Struthers-Wells Corp. will be sent the contract for design and fabrication of the vessel in the near future.

Expenditures to date appear in the following order of magnitude:

| | |
|---|---|
| Design (Vitro) | $ 33 000 |
| Scope, Administration, Estimating, etc. | 26 000 |
| Construction | 231 000 |
| | $289 000* |

*Excludes Containment and Simulator Vessels.

Facilities Operation

Costs for the month of September were $179,698, which is 102 percent of the forecast for the month. The costs for the first quarter of FY 1965 are



1235118

$458,944, which is 85 percent of the predicted year-to-date. As part of the transition to Battelle Memorial Institute, the costs and rental must balance at mid-year. The budget was consequently divided into the first and second half-year. The rents are being adjusted accordingly to assure a mid-year balance. It appears that the budget for the first half of FY 1965 will not be exceeded.

Grounds improvements are proceeding. This includes road patching, grade and fill of blow areas, and walkways.

Waste Disposal

The following table summarizes the Waste Disposal Operation:

| Item | August | September |
|---|---|---|
| Concrete waste barrels disposed to 300-wye burial ground | 10 | 6 |
| Concrete waste barrels disposed to 200-W burial ground | 0 | 12 |
| Loadluggers of dry waste disposed to 300-wye burial ground from 300 Area sites other than the 325 building | 47 | 45 |
| Loadluggers of dry waste disposed to 300-wye burial ground from 325 bldg. | 7 | 6 |
| Loadluggers of dry waste disposed to 200-W burial ground from 300 Area sites | 6 | 7 |
| Containers of high level dry waste disposed to 300-wye burial ground waste tanks | 75 | 83 |
| Crib waste volume, gallons | 245 000 | 280 000 |

None of the retention basins exceeded the Class II activity levels during the month.

Sixteen thousand gallons of liquid waste were dumped during the month in the 216 BC trench. To date, 42,000 gallons have been disposed to the 270,000 gallon capacity trench.

Two sizes of core samplers were ordered and have been received for taking samples from the bottom of the 15,000 gallon underground storage tanks. Maintenance has completed fabrication of part of the equipment.



1235119

Decontamination of the interior of the three 5,000 gallon trailers reduced the dose rates by 50 percent to 75 percent. The total cost was about $435.00.

Building Operations

The instrument air compressors and the alum pumps were overhauled at the filter plant. Maintenance is revising turbidity stations to give more accurate, reliable readings.

Reheat coils are on order for #4 supply unit in 308 building. The efficiency of the four air drying towers in this building is being checked to maintain optimum drying of compressed air for hood use.

Revisions of air flows between 306 and 306-A buildings were made to reduce objectionably high pressure differentials.

The sump pumps in 326 building have been troublesome. Increasing flows to these pumps have resulted in minor floods when any delay occurred in pump operation. A review of the pumps and controls is being made.

The shop fan motor in 328 building failed during the week of October 19. This motor was sent to the "H" area shops for repair and is now back in service.

The 3760 building fan motors were serviced the first part of October. The bearings of number one supply motor are being replaced.

The low pressure annunciator for No. 3 and 4 wells and the high tank level signal for 340 building have been removed from the 384 building and reinstalled in the equipment room of 325 building.

Air balance operators have been assigned to operator training. Four new operators were transferred to HL on October 19 as part of the transition plan. Air balance routines are completed as time permits, and essential air balance is being done on overtime.

Drafting

The equivalent of 107 drawings was produced during the month for an average of 35.8 man-hours per drawing.

Major designs completed or in progress are: (1) a powder processing glove line; (2) a high temperature furnace for the NASA program; (3) FRPP service piping and electrical work to waste solidification equipment; (4) underwater



1235120

measuring equipment for calibrating elements in N Area; (5) layout of new equipment for scope of increased power level of PRTR; (6) a sodium pump test facility in PRTR; (7) a critical mass expansion tank for building 209-E; (8) scope for a polonium hood complex; and (9) a Picatinny Arsenal Ordnance projectile design.

Drafting service is also supplied in support of other laboratory engineering programs.

Construction

| | Unexpended Balance |
|---|---|
| Orders outstanding beginning of month | $751 665 |
| Issued during month (incl. suppl. & adj.) | 91 718 |
| J. A. Jones expenditures during month (includes CO cost) | 197 175 |
| Balance at month end | 646 208 |
| Orders closed during month | 89 454 |

Maintenance Work Orders active - 7, Face Value - $21,816.

Major nonproject jobs in progress during the month were: (1) 100 Areas - construction of an electron microscope in building 108-F, remodeling of men's and women's rest rooms in building 141-M, remodeling of dog pens and dog runs and installation of an intercom system in building 144-F, relocation of stairway and installation of additional electrical services in building 189-D, completion of fire detection systems in five buildings in 100-F Area; (2) 300 Area - installation of 4" stainless steel drain line in building 306, recoating of exterior of the containment vessel has been completed in 309, construction started on an enclosure for a gas loop in building 314, performing electrical and pipe work in the 321 building and installed a metal building for an air conditioner for the 321 building, completion of five offices and women's lounge in 325 building, modified the entrance and enlarged gas bottle dock for the 329 building - also installed hoods and services in laboratory 11-B of this building, constructed a personnel change and survey building near the 340 building, constructing partitions and change doors in the 3717-B building, constructed an office and enlarged the rest rooms in the 3718 building, installed a metal partition in the 3718-AX building, constructed an addition to the 3718-C warehouse; and (3) General Area - continued work on the contingency maintenance items for the 3201 building, patched miscellaneous blacktop roadways and installed guard rails in the 300 Area, and fabricated racks for the waste solidification engineering prototype.



1235121

Well Drilling Program - FY 1965

The revised rough draft of the project proposal and a new unit cost summary are being prepared.

GENERAL

There were no reports of invention or discovery during the month.

DS Parsley
Acting Manager
Finance and Administration

DS Parsley:RDT:whm

## REACTOR DEVELOPMENT - 04 PROGRAM

### PLUTONIUM RECYCLE PROGRAM

#### Plutonium Recycle Test Reactor

##### Operation

Reactor output for October was 1,312 MWD for an experimental time efficiency of 76% and a plant efficiency of 60%. There were seven operating periods during the month, four of which were terminated manually, two were terminated by scrams, and one continued through month-end. A summary of the fuel irradiation program as of October 31, 1964, follows:

| | Al-Pu | | $UO_2$ | | $PuO_2$-$UO_2$ | | Other | | Program Totals | |
|---|---|---|---|---|---|---|---|---|---|---|
| | No. | MWD | No. | MWD | No. | MWD | No. | MWD | No. | MWD |
| In-Core | 0 | | 7 | 1850.9 | 78 | 14128.0 | | | 85 | 15978.9 |
| Maximum | | | | 349.7 | | 376.2 | | | | |
| Average | | | | 264.4 | | 181.1 | | | | |
| In Basin | 7 | 572.5 | 26 | 2769.0 | 48 | 5884.3 | | | 81 | 9225.8 |
| Buried | | | | | | | 1 | 7.3 | 1 | 7.3 |
| Chem. Process. | 68 | 5465.8 | 35 | 1965.8 | | | | | 103 | 7431.6 |
| Program Totals | 75 | 6038.3 | 68 | 6585.7 | 126 | 20012.3 | 1 | 7.3 | 270 | 32643.6 |

(Note: MWD/Element x 20 $\simeq$ MWD/TU for $UO_2$ and $PuO_2$-$UO_2$.)

Estimated heavy water loss and indicated helium loss for the month were 975 pounds and 131,596 scf., respectively.

##### Equipment Experience

Only 50 outage hours were charged to repair work. Valve and leak repairs constituted the main repair items.

Preventive maintenance utilized 237 manhours, or ~ 5% of the total maintenance effort.



1235123

Improvement Work Status (Significant Items)

Work Completed

Primary Water Sample Station
Repair and Recoating of PRTR Containment Vessel Roofing
Thermistor Probe Installation in FEEF
Bypass for LWI Flow Switches

Work Partially Completed

Corrosion Loop Installation
Shim Rod Shroud to Top Cap Modification
Air Lock Door Operators
Modification to PRTR Warehouse 3718-C
Vibration Snubbers for Earthquake Protection
Supplemental Emergency Water Addition
Voltage Off-Normal Detection and Alarm for 24-Volt Battery System
PRTR Steam Utilization
Additional Fuel Storage and Examination Facility
Mark III Shim Prototype
Creep Test Facility
Flux Wire Scanning System
"C" Cell Instrument Tubing Penetration
Installation of Second Generation Shim Rod
$D_2O$ Cleanup Facility

Design Work Completed

Alarm Annunciator - High Helium Flow to RLT-1
Thermal Barrier Seal and Hoist Improvements
Tube #1354 Flow Test of Short Fuel Element

Design Work Partially Completed

Instrument Power Supply
Decontamination Building
Pneumatic Irradiation Facility
PRTR Experimental and Building Facility Addition
PRTR Increased Power Level

Process Engineering and Reactor Physics

PRTR Test 105, Irradiation of a Short $UO_2$ - 2 w/o $PuO_2$ Fuel Element, was issued and the first short 2% element was charged on October 26. The element has been operating at an indicated tube power slightly in excess of 1000 KW.



1235124

The absolute activity measurements on samples of the wires activated under PRTR Test 89 were completed and absolute thermal flux values have been determined from these activities. The results follow:

| Position | $\phi_{Thermal}$ |
|---|---|
| 0" | $4.66 \times 10^{12}$ |
| 3" | $8.3 \times 10^{12}$ |
| 48" | $9.25 \times 10^{13}$ |

The positions listed above refer to the distance below the bottom of the top shield.

The maximum leak rate experienced during the month from HX-1 was approximately 4 lbs/day. Average leakage was 1/2 to 3 lbs/day.

PRTR Test Number 98, Emergency Air Cylinder Station Use Rates, was completed. Test data showed that individual instrument use rates were within limits on that portion of the instrument air system served by the Control Room Backup Station. It was concluded that the use rate for this station could not be reduced and the higher storage capacity requirements must remain in effect.

Experimental Reactor Services

The status of the various test elements at the end of October 1964, is shown below. Those elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to October 1, 1964, have been deleted from this table.

| Test No. | Channel Location | Element Number | Description | Date Initial Charge | Date Dis-charged | Approximate Accumulated MWD |
|---|---|---|---|---|---|---|
| 14 | 1956 | 5097 | Moxtyl-Swaged | 4/2/62 | -- | 229.2 |
| 14 | Basin | 5098 | Moxtyl-Vipac | 5/8/62 | 10/26/64 | 340.2 |
| 14 | 1758 | 5099 | Moxtyl-Vipac | 5/8/62 | -- | 238.7 |
| 48 | 1156 | 5150 | Moxtyl ($\frac{1}{2}$" x $\frac{1}{2}$" pads) | 8/1/62 | -- | 237.7 |
| 54 | 1542 | 5116 | Moxtyl (clip on pads) | 5/8/62 | -- | 252.5 |
| 54 | 1554 | 5118 | Moxtyl (clip on pads) | 5/8/62 | -- | 376.2 |
| 61 | Basin | 5185 | Moxtyl-Physics | 5/28/63 | 10/6/64 | 245.7 |
| 61 | 1249 | 5186 | Moxtyl-Physics | 5/28/63 | -- | 251.5 |
| 61 | 1445 | 5192 | Moxtyl-Physics | 6/13/63 | -- | 249.6 |
| 67 | 1047 | 5117 | Moxtyl (Repaired Wire) | 10/20/63 | -- | 207.9 |
| 80 | 1746 | 5214 | Moxtyl (1% $PuO_2$ Swaged) | 11/18/63 | -- | 209.7 |



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| | | | | | | |
|---|---|---|---|---|---|---|
| 85 | 1855 | 5230 | Moxtyl (1% $PuO_2$, Vipac) | 1/30/64 | -- | 144.9 |
| 37 | 1548 | 1098 | $UO_2$-Physics | 5/12/62 | -- | 233.9 |
| 37 | 1550 | 1097 | $UO_2$-Physics | 5/12/62 | -- | 251.5 |
| 37 | Basin | 1099 | $UO_2$-Physics | 5/12/62 | 10/6/64 | 206.6 |
| 72 | 1253 | 5253 | Zircaloy Coupons (1% $PuO_2$) | 9/1/64 | -- | 53.2 |
| 105 | 1255 | 6000 | $UO_2$-2 w/o $PuO_2$, 58" long | 10/26/64 | -- | 5.2 |

Fuel element 5185 was found to be a leaker and was discharged from the reactor. Subsequently, it was disassembled in the basin.

A total of 32 irradiated fuel elements were inspected during the month.

Fuel Element Rupture Testing Facility

Corrections to the Rupture Loop during the last outage have made a significant improvement in vibration level. Following loading of an intentionally defected element the loop was placed back into service. Some control problems continue at the higher temperature which are attributed to the low fuel power generation and the oversized trim in the heater bypass valve. New trim has been ordered for this valve.

$UO_2$ fuel element #1039, which had been tested with a 1-1/2 inch longitudinal slit during August and September, was inspected and photographed in the storage basin using a new optical viewing instrument. It was found that the 1-1/2 inch slit only penetrated the cladding for a distance of 5/8 inch. The slit was then lengthened to 3-1/4 inches for further testing in the loop. The reactor was started up with this test element on October 27. The test element was cycled through three startups in which the reactor reached full power. After the third startup, the reactor stayed at full power through the month's end, for an accumulated exposure during October of 1.45 MWD or 23.7 MWD/Ton. The coolant activity, as measured by both the delayed neutron and Gieger Muller instrumentation, did not increase when the slit was lengthened.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 30,767 hours. This includes 18,971 hours performed in the Technical Shops, 7,246 hours assigned to J. A. Jones Company, and 4,550 hours assigned to off-site vendors. Total shop backlog is 18,144 hours, of which 90% is required in the current month with the remaining hours distributed over a three-month period. Overtime worked during the month totaled 1,289 hours or 5.8% of the total available hours.



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Distribution of time was as follows:

| | Man Hours | % of Total |
|---|---|---|
| N Reactor Department | 2 933 | 9.5 |
| Irradiation Processing Department | 6 137 | 20.0 |
| Chemical Processing Department | 458 | 1.5 |
| Hanford Laboratories | 21 239 | 69.0 |

Nine craftsmen (carpenters, painters and sheetmetal workers) were transferred from IPD to Technical Shops.

## LABORATORY MAINTENANCE OPERATION

Total productive time was 18,100 hours of 20,200 potentially available. Of the total productive time, 95% was expended in support of Hanford Laboratories components, with the remaining 5% directed toward providing service for other HAPO organizations. Craft overtime worked during October was 1,200 hours or 5.9% of total available hours.

Manpower utilization (in hours) for October was as follows:

| | | |
|---|---|---|
| A. Shop Work | | 2 000 |
| B. Maintenance | | 6 300 |
| 1. Preventive Maintenance | 1 800 | |
| 2. Emergency or Unscheduled Maintenance | 1 400 | |
| 3. Normal Scheduled Maintenance | 3 100 | |
| C. R&D Assistance | | 9 800 |

Three standards cells were hand-carried to the National Bureau of Standards in Boulder, Colorado, for calibration and certification.

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:JGZ:bk



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## INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

| INVENTOR | TITLE OF INVENTION OR DISCOVERY |
|---|---|
| F. B. Quinlan | A Process for Making Wire from Brittle Metals and Alloys (HWIR-1765) |
| R. E. Burns and J. F. Phillips | Increase in the Dissolution Rate of Thorium Oxide |
| J. D. Kaser | Use of Colemanite as Fluxing Agent for Melting Dried Radioactive Waste |
| L. A. Bray | Dissolution of $MnO_2$ with Sugar-Protactinium Recovery Program (HW-84468) |

Manager, Hanford Laboratories



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