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HANFORD LABORATORIES OPERATION
MONTHLY ACTIVITIES REPORT
SEPTEMBER, 1962

OCTOBER 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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HANFORD LABORATORIES OPERATION
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Operation Managers

October 15, 1962

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~~Date 73-5-8, DB~~
~~U. S. AEC Division of Classification~~

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRELIMINARY REPORT

This report was prepared only for use within General Electric Company in the course of work under Atomic Energy Commission Contract AT(45-1)-1350. Any views or opinions expressed in the report are those of the author only.

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TABLE I - HLO FORCE REPORT

DATE: September 30, 1962

	At Beginning of Month		At Close of Month		Total
	Exempt	Salaried	Exempt	Salaried	
Chemical R & D	136	138	131	133	264
Reactor & Fuels R & D	178	173	175	163	338
Physics & Instrument R & D	92	60	91	61	152
Biology	39	57	39	59	98
Operations Res. & Syn.	19	4	18	5	23
Radiation Protection	42	97	41	91	132
Finance and Administration	117	95	120	111	231
Programming	15	3	15	3	18
General	3	3	3	4	7
Test Reactor & Auxiliaries	50	186	58	299	357
TOTAL	691	816	691	929	1,620

BUDGETS AND COSTS

September operating costs totaled $2,422,000, an increase of $186,000 from the previous month; fiscal year-to-date costs are $6,795,000 or 23% of the $29,074,000 tentative control budget. Hanford Laboratories' research and development costs for September, compared with last month and the control budget, are shown below:

(Dollars in thousands)	COST Current Month	Previous Month	FY To-Date	Annual Budget	% Spent
HLO Programs					
02 Program	$ 78	$ 73	$ 214	$ 1 069	20%
03 Program	1	3	10	175	6
04 Program	1 060	961	2 909	11 418	25
05 Program	94	98	279	1 293	22
06 Program	270	240	743	3 154	24
	1 503	1 375	4 155	17 109	24
FPD Sponsored	106	88	279	1 370	20
IPD Sponsored	105	94	286	1 325	22
CPD Sponsored	146	150	417	1 369	30
Total	$1 860	$1 707	$5 137	$21 173	24%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Visual studies performed with an electrically heated test section in a glass tube revealed that considerable boiling can take place in the vicinity of the devices used to support fuel elements in the Hanford production reactors if the fuel elements are positioned more than 50% off center from a concentric position within the tube.

To further elucidate the consequences of reactor accidents, additional experiments were run to determine the flow rates that exist when high pressure steam-water mixtures are discharged from a pipe into atmospheric conditions.

Additional calculations concerning cooling of fuel elements in the PRTR in the event of a total power outage indicate that the installation of a four-inch vent valve in addition to the existing two-inch valve will result in fast enough primary system depressurization for light water injection to become effective before fuel temperatures become excessive.

Reactor kinetics studies, utilizing an analog computer, were completed for a uniformly enriched PRTR core consisting entirely of mixed crystal, UO_2-PuO_2 fuel elements. Preliminary analysis of the analog data indicates that nuclear excursions would be no more severe in a uniformly enriched core than in a spike enriched core.

Analysis of PRCF analog studies indicates that moderator void formation and resonance absorption in U-238 should provide sufficient negative reactivity to override the simulated nuclear excursions.

A feasibility study has been initiated to investigate the desirability of using hollow, meltable, plutonium driver fuel elements to provide a stronger inherent shutdown mechanism for the PRCF. The use of a low melting point (100 C) bismuth alloy containing plutonium is being considered.

A section of irradiated PRTR pressure tube was burst at 288 C (550 F) in a prototype test unit. Though having a neutron exposure of 3×10^{20} nvt (E > 1 mev), the tube swelled to almost twice its original diameter prior to failure. The maximum pressure before bursting was significantly greater than for unirradiated tubing.

Many of the solutions normally used for removing UO_2 after a rupture are ineffective in dissolving PuO_2. However, in ex-reactor, simulated rupture experiments a procedure yielding a satisfactory decontamination factor of 15 was found. This consisted of treatment with a mixture of oxalic acid, hydrogen peroxide and buffered peracetic acid, followed by an alkaline permanganate treatment and an oxalic acid rinse. The corrosion rate for carbon steel, stainless steel, and Zircaloy in the oxalic-peroxide solution is very low.

A high density UO_2-2.57 m/o PuO_2 capsule having an ETR exposure history of about 3×10^{20} fissions/cm^3, or about 10,000 MWD/T, showed a maximum cladding diameter increase of 0.014-inch.

Four of eight MgO-PuO_2 capsules successfully irradiated in ETR are being examined in Radiometallurgy; in the MgO-13.5 w/o PuO_2

[redacted]

material, which is calculated to have reached 2200 C, the PuO_2 second phase is clearly visible and distribution appears to be unaffected by columnar grain growth of the matrix MgO.

A sample of PuO containing a small amount of beta-Pu_2O_3 and 0.083 a/o carbon was prepared by reaction of PuO_2 and carbon at 1800 C in helium. The lattice constant of the PuO was observed at 4.961 ± 0.001 A.

A sample of PuC containing a small amount of Pu_2C_3 and which had a carbon content of 4.83 w/o was prepared by the reaction of PuO_2 and carbon. The lattice constant of the PuC was observed to be 4.971 ± 0.001 A.

A series of plutonium carbide samples varying from 12 to 52 a/o C have been given three different heat treatments, and the results illustrate the extreme sensitivity of the zeta phase to heat treatment.

Melting studies on PuN were conducted in atmospheres of purified argon, helium, and nitrogen with a tungsten ribbon furnace. X-ray diffraction analyses of the nitrogen melted specimens show a constant lattice parameter, thus indicating no constitutional changes. The melting point was determined to be 2750 ± 75 C. The melting point of Pu_2S_3 was found to be 1725 ± 5 C in vacuo on a tungsten filament.

Twenty-four fission product transient samples for Phillips Petroleum Company containing U-235-Al alloy cores have been completed through autoclaving; final assembly and end cap welding is in progress. Casting and sampling of U-233-Al core alloys is currently in progress. Extrusion of the U-235-lithium, plutonium-lithium bearing alloys is still being delayed due to analytical difficulties.

Previous observations of radially-nonuniform fission fragment and plutonium distributions were confirmed by final analyses of three additional irradiated UO_2 fuel capsules.

The surface tension of UO_2 was calculated from drop profiles recorded on high speed motion pictures. A value of 490 dyne cm^{-1} (roughly the same as mercury) was obtained.

Preliminary determination of the liquidus for the systems UO_2-ThO_2

[redacted]

was completed. Principal features observed were total miscibility and a melting minimum of 2730 C at 2.5 w/o ThO_2.

The photomontage "Reflection Electron Microscopy of Irradiated UO_2 Crystal" was selected for inclusion in the 1962-63 traveling exhibit of the Electron Microscopy Society of America.

Preliminary data from gamma irradiations with a Co-60 source (8.93×10^5 R/hr) of UO_2 specimens at approximately room temperature revealed a surprisingly high release of sorbed gases, comparable to that normally expected from thermal effects at high temperatures of 800-1000 C. Mass spectrometric analysis of released gases shows a high percentage of H_2, with N_2 and CO_2 also present. If confirmed, these data indicate that (1) release of sorbed gases in fuel elements may be increased by radiation, and (2) ionizing irradiation might provide a way of removing commonly troublesome sorbed gases from ceramic fuel materials prior to in-reactor use.

The hydrogen content of Zircaloy-2 cladding from four coextruded fuel samples irradiated to exposures up to 3600 MWD/T in the KER loops has been determined. The results show no significant pickup of hydrogen during irradiation, thus verifying the visual observation that cladding corrosion during irradiation was not significant.

The program to make Zircaloy-2 strip with high bend ductility resulted in the laboratory production of Zircaloy-2 sheet of sufficient ductility to meet the forming needs in the fabrication of N inner fuel supports. Pilot quantities of sheet have been prepared for evaluation.

A fuel tube equivalent to an N-Reactor inner tube, but with a fluted outer cladding, was charged into the ETR, P-7 loop, September 3, 1962. This element is operating with a maximum specific power and bulk water temperature equivalent to those anticipated in the N-Reactor.

Stainless steel has been successfully spot welded to zirconium by using a niobium foil interface. No protective atmosphere was required. Production of a satisfactory weld was accomplished by simply placing a small disc of niobium between the stainless steel and the zirconium. The weld was made directly through this sandwich and proved to be quite tough and strong. Samples are being made for testing in the autoclave. This development may permit the fabrication of all-steel supports for use on the N fuel element.

Microexamination of a graphite "fuel element end closure" completed by the magnetic force welding process revealed a weld of surprisingly high quality. The graphite in the weld zone recrystallized into a material of finer grain size and higher density than the base material.

A series of measurements were made of the activation energies of Zircaloy-2 during neutron irradiation. These measurements confirm values previously obtained and provide thermodynamic evidence that creep rates will be lowered during neutron irradiation, substantiating observations of reduced rates during irradiation.

A capsule has been charged into the 2A test hole at KE Reactor to measure the electrical resistance of the corrosion film on Zircaloy-2 during irradiation. Initial measurement in an oxidizing gas indicates the resistance of the corrosion film is reduced by a factor of 600 during irradiation. This reduction in resistance is indicative of a higher concentration of oxygen anion vacancies in the film during irradiation and is consistent with other observations of increased corrosion rate in a reactor environment.

A marked flux effect on the aqueous corrosion of Zircaloy-2 was noted from weight gain measurements on test coupons irradiated in the ETR, G-7, hot water loop at 540 F and at dose rates of 6.9×10^{13} nv and 1.2×10^{14} and on unirradiated specimens exposed to hot water under identical time-temperature conditions. The amount of weight gain was independent of cold work from 0 to 40 percent.

Results obtained in this laboratory are in disagreement with a recent British publication implying that neutron damage to molybdenum cannot be detected by electron microscopy regardless of post-irradiation annealing treatments.

Carbon black is often added to graphite mixes to increase density or lower permeability to gases. Irradiation of samples containing varying amounts of carbon black resulted in a general increase in contraction in both parallel and transverse directions and a decrease in anisotropy with increasing amounts of carbon black.

2. Physics and Instruments

Preparations continued for criticality studies on plutonium oxide-polystyrene mixtures. The concentration in the plutonium-plastic material being fabricated is about 1.15 grams per cc and the hydrogen to plutonium atomic ratio is about 15. Installation of the remotely operated critical assembly machine continued.

Corrections for the effect of stainless steel vessel wells have been applied to an old measurement of the limiting concentration of plutonium which is always safe in water solutions. The new value is 8.1 ± 0.3 grams/liter.

Criticality experiments continued with a 14" spherical vessel filled with plutonium nitrate solution. The experiments were done with a .030" cadmium shell separating the criticality vessel from a water reflector.

Suspected wall thickness variations in a number of replacement aluminum process tubes for the production reactors were verified with a precise ultrasonic measurement.

A special transistorized conductivity meter was developed to monitor the buildup of plutonium compounds in a 234-5 Building hood and to signal if the concentration exceeds limits.

Re-analysis of the data from the irradiated low-exposure Pu-Al PRTR element 5075 shows much better agreement with calculation than was previously reported. Improvements were obtained in the isotopic buildup as well as the Cs-137 burnup comparisons. The first of the low-exposure Pu-Al physics elements has been disassembled and delivered to Radiometallurgy Laboratory for preparation of burnup analysis samples.

A progress report on fuel re-use studies now in preparation indicates possible reductions in fuel cycle cost of up to one mill/Kwhr may be obtained by cycling fuel between fast and thermal reactors without reprocessing. In other studies, multi-group cross sections have been generated for carrying out the Phoenix fuel reactor studies and comparison of uranium and plutonium fuels in several small compact fast reactors is being made.

[redacted]

A theoretical study was made to determine whether very small diameter plutonium rods in light water moderator might produce a lower critical mass than homogeneously dispersed plutonium in water. The results of the calculation indicated that the homogeneous system has the lower critical mass.

In computer code development, the SUMMIT code for calculating scattering kernels in graphite has been made available on the 7090 computer. The transport theory program has been modified to permit flexible variation of the relative size of each region of the reactor as well as variation of over-all dimensions in determining criticality, thus saving the investigator time. Work is continuing on determining the parameters in the two mass gas models used in the neutron thermalization routine in RBU. Considerable improvements have been made in the cross sections on the RBU library in the recent updating of many of the important isotopes. Programs TEMPEST, GAM, SIGMA-3C, SIGMA-3H, and HFN have been loaded onto a chain tape so that any number of these programs can be run in almost any order. The straight burnup, recycle, and graded cycle options of the CALX burnup code have been run successfully.

Discrepancies between Nelkin's theoretical scattering kernels for water and experiment continue to become evident. Some difficulties in interpreting the results from the General Atomic's code which calculates the Nelkin's kernels has prompted the formulation of a Hanford water kernel code that calculates the Egelstaff S-function which is directly comparable with the usual presentation of experimental data.

Total cross section measurements were made on fourteen elements: calcium, strontium, barium, lead, cobalt, magnesium, yttrium, niobium, vanadium, zirconium, silver, molybdenum, tin, tungsten. The measurements covered the neutron energy range from 3 to 15 Mev., an energy range in which little experimental work has been done previously.

The development of equipment for neutron time-of-flight measurements with slow neutrons was advanced by the successful rotation of an aluminum crystal at 10, 000 rpm. The crystal produces two bursts of neutrons each revoltuion by Bragg diffraction.

[redacted]

A 1024-channel time-of-flight analyzer was completed and placed in use for slow neutron cross-section measurements. Cost estimates were developed for two possible modifications which would further increase its utility.

A series of analog computer runs were completed to determine the maximum power level excursions and the maximum fuel temperatures produced by various reactivity disturbances in the PRTR.

The performance of the developmental nondestructive heat transfer test for fuel core-to-cladding bond was improved through use of a new dual infrared radiometer which compensates for major variations in the emissivities of fuel elements.

Best results in the ultrasonic testing of fuel sheath tubing were obtained with rectangularly masked round transducers. In accompanying studies of the ultrasonic test, an unexplained anomaly was discovered between experimental results and the theoretically predicted Lamb wave propagation.

A new circuit was developed to automatically disconnect a storage battery from certain radiological instruments when the normal discharge end-point is reached.

Tests were completed on the prototype coincidence-count alpha air filter counter which was designed to provide immediate analysis of standard HAPO 4" x 8" air filters without the usual wait for radon-thoron decay. The instrument is now in use in 308 Building.

Measurements of the radioactivity of Alaskan Eskimos were completed and the scientists and equipment have returned. The equipment performance was trouble free and exceeded our expectations. Analyses of results are in progress and a preliminary report was presented at the Second Symposium on Radioactivity in Man.

Two successful field experiments were conducted as part of the continued investigation of the removal of particulates from the air. One experiment permitted measurement of the dry deposition of zinc sulfide particles on vegetation to a distance of 200 meters from the source. The second was designed to measure the relative scavenging of particulates from a diffusing plume by raindrops of different diameters.

[redacted]

3. Chemistry

Subsequent to the installation and shakedown tests of radiant-heat spray and pot calcination equipment in the High Level Radiochemistry Facility, both units were successfully used to convert actual Purex 1WW liquid waste to a calcined solid. The first of the full-level runs was with the radiant-heat spray calciner, operated with continuous melt-down in the solids receiver, and the second was a simple pot calcination run. The Purex 1WW used in the runs, although centrifuged at Purex, contained gross amounts of fine solids which, however, did not cause nozzle plugging or undue difficulty in feed control. Twenty-seven liters of feed, equivalent to about 0.1 ton of irradiated uranium, were used in each run and resulted in about one liter of solidified, calcined melt in each case. Operation of both calciners and the associated off-gas equipment, instrumentation, etc., was smooth and uneventful, and no detectable radioactive contamination was released to the atmosphere.

Experimental measurement of the emitted radiation from 25 curies of well aged promethium disclosed the unexpected presence of minute amounts of Pm-146. Although the abundance of the Pm-146 photons is very low, the energies are sufficiently high to require some small additional shielding for aged Pm-147 sources.

Thirty-nine grams of black, crystalline plutonium dioxide have been prepared by chemical precipitation from molten chloride salt solutions. This unusual form of PuO_2 has a tap density of 6.1 g/cc, about 50% of theoretical density.

Segments of non-irradiated Zr-4 clad UO_2 and UO_2-PuO_2 fuel rods were declad by the Zirflex process. In both cases the core materials disintegrated to a "mud" during the decladding step. The residue derived from the UO_2 fuel dissolved readily in dilute nitric acid; the residue from the mixed oxide fuel failed to dissolve completely upon prolonged treatment with boiling 5 $\underline{M}$ HNO_3.

Leaching studies of a sample of hard sludge from the Purex 103-A waste tank showed that about 80% of the cesium was removed by water washing; about 80% of the strontium and most of the remaining cesium was removed by a citric acid leach.

[redacted]

Hot cell development of a nickel ferrocyanide scavenging process for the removal of heat-generating cesium-137 from Purex tank supernates (in support of the CPD Waste Management Program) culminated in a successful plant test. Over 99.6% cesium removal was obtained in the plant test.

A process employing phosphotungstic acid for the scavenging of cesium-137 from acidic Purex 1WW was demonstrated in a hot cell test. Although 99.5% cesium removal was obtained, decontamination of the cesium from other fission products was poorer than expected.

The use of a new phenol-type extractant (4-sec-butyl-2,α methyl benzyl phenol, abbreviated BAMBP) is found to be competitive with dipicrylamine for the extraction of cesium from wastes of concern to the Hanford Waste Management Program. Although a higher operating pH is required with BAMBP than dipicrylamine, the former solvent has a lower density and thereby eliminates phase inversion problems previously encountered in flowsheet design.

The use of selected commercially available anti-foam agents was found to suppress foaming during the formaldehyde treatment of synthetic Purex 1WW in pilot scale facilities simulating the Purex plant equipment and operation.

Preliminary laboratory results show that a combined anodization and chemical etch of uranium metal surfaces produces an uniformly etched surface of excellent appearance and good nickel platability in a Thompson bath.

Promising results were obtained in the laboratory for the preparation of reactor process water by the treatment of raw water with mixed iron and aluminum hydroxides. In two tests, 96 to 97.7 percent of phosphate was removed and a very low rate of filter head loss occurred.

A 23-day operating period was completed during which an uninterrupted flow of deionized water was supplied to a new aluminum process tube with new aluminum-clad fuel elements. The concentrations of the radiologically significant radioisotopes were found to be 10 to 30 percent of those normally observed using conventional process water.

4. Biology

Fish sampled from the Columbia River again showed no columnaris infection. However, fish obtained from the Leavenworth Hatchery were infected, but most of these infections were slight. Evidence continues to point to the crowding of fish as being the cause for major outbreaks of the disease.

Interesting results are being maintained in the translocation of Zn-65 fed to trout. The element first deposits in the spleen, liver, and kidney, and then gradually translocates to the bone, eye, and gill filament.

A surprisingly large amount of Sr-90 (625 μc) fed daily to female miniature swine had no effect on litter size and birth weight of their offspring.

In the experiment where the coefficients for the transfer of I-131 from contaminated food to cows to milk to consumers are being measured, the I-131 in the thyroid of the cow is showing a half-life of about five days. In cow's milk, two half-lives are now apparent, one being less than one day and the other about four days.

In preliminary work with miniature swine, Ce-144 was found to deposit in highest concentrations in vertebral bodies and in the ends of the ribs. This suggests further work to determine whether part of the skeleton rather than all of it, should be considered in the future for hazard evaluation. The same situation seems to apply for Pm-147.

The effectiveness of DTPA is being tested in removing plutonium from the body after subcutaneous injection. Plutonium translocated to other organs is removed faster than without DTPA, but the chelating agent seems to have little effect on movement of plutonium from the injection site.

The number of agents being tested for their ability to remove internally deposited plutonium continues to increase. The most effective new agent studies is Tiron, a benzene-sulfonic acid derivative, which seems to operate differently than DTPA. It is not as effective as DTPA. Pluronics (an ethylene oxide derivative polymer) increased the excretion of plutonium in dogs that had previously inhaled plutonium dioxide.

This was better than DTPA, but still not effective in removing a significant per cent of the plutonium present in the body.

The transfer of potassium through cellular membranes appears dependent on the type of metabolism going on in the cell. Oxidative and fermentative strains of yeast have practically no influx of potassium, whereas a strain which is both oxidative and fermentative showed a very high influx of potassium.

Population counts of species of plankton from the Columbia River were made along with their concentrations of radionuclides. Mn-56 and Cu-54 were the most abundant gamma emitters found.

Concentrations of I-131 in deer contributed from State Game Departments are being measured. In California deer, thyroids increased steadily to about 8×10^{-4} μc/g and then decreased during the remainder of September. Deer from Maryland showed a similar increase, but were only one-half the concentrations of California deer. A HAPO deer thyroid was found to contain 10^{-2} μc I-131/g.

5. Programming

For nuclear reactor calculations, alpha for Pu-239 is defined as the ratio of the relative probability of neutron absorption in Pu-239 resulting in neutron capture to form Pu-240 over the relative probability of neutron absorption in Pu-239 resulting in fission. For thermal neutrons, the alpha value for Pu-239 is generally greater ($\alpha \approx .45$) than the value for U-235 ($\alpha \approx .22$). This difference is even greater in Pu-239 near thermal neutron resonance absorption regions ($\alpha \approx .75$). This characteristic has led some to criticize the use of plutonium in thermal reactors, especially in under-moderated machines such as boiling water reactors, and also, in high moderator temperature machines such as HTGCR's, because the resulting neutron spectrum tend to overlap the high α resonance regions. However, our studies indicate that the inherent self-shielding of these resonances can reduce the relative absorptions in the resonances, with the result that as plutonium enrichment is increased the α value for Pu-239 goes down and the heat and neutrons produced by Pu-239 increase. An over-all value of alpha of .54 is typical, which corresponds to an increased value for Pu-239 of 10 to 20 percent as compared to values computed when self-shielding is neglected. The study further indicates that an additional reduction in α of 5 percent or so is possible by placing the plutonium in the form of small diameter rods, although it is not clear that this is worthwhile economically.

Both the Salt Cycle Code and the Conventional Reprocessing Code now appear ready for use in power reactor fuel reprocessing economic studies. Some effort will be necessary to define the "best values" for the numerous cost variables in these codes before realistic evaluation of the salt cycle reprocessing economic advantages for specific reactor cases can be started.

TECHNICAL AND OTHER SERVICES

Two new cases of plutonium deposition were confirmed by bioassay analyses during September. One case was the result of a contaminated injury and in the other, inhalation was the mode of entry. The maximum estimated body burden resulting from the contaminated injury was estimated to be less than five percent of the permissible body burden. The inhalation case was estimated to have resulted in a deposition less than one percent of the permissible body burden. The total number of plutonium deposition cases that have occurred at Hanford is 299 of which 216 are currently employed.

Sampling equipment was set up at the 100-H, DR, B and C areas for the continuous collection of effluent samples from the outlets of the retention basins. This program is to be extended to all of the 107 basins in order to improve estimates of total quantities of radionuclides discharged to the river.

Relatively high concentrations of fallout materials were measured on air filter samples of our Pacific Northwest network during September. The peak concentration observed was 11 $\mu\mu c$ beta/m^3. This is comparable to the highest concentration observed in the fall of 1961 (12 $\mu\mu c$ beta/m^3). A milk sample collected at Moses Lake, Washington, September 18, 1962, contained 780 $\mu\mu c$ I-131/liter. Local milk collected on September 19, 1962, contained 90 $\mu\mu c$ I-131/liter.

A document was issued presenting the results of an extensive analysis of the measurement error structure associated with C-Basin Profilometer measurements. The conclusion was reached that positioning of the fuel element

[redacted]

prior to recording measurements is a major cause of measurement error.

An analysis was made of the effects of blending metal recovery feed on button and ingot density. Of the variables studied, the aluminum used in complexing the fluoride was found to exert the greatest influence on density.

Consultation services were afforded personnel concerned with inventory sampling plans to see if their procedures satisfied some of the general directions given in a cost accounting bulletin. The bulletin gave some rather arbitrary sampling percentages. Hanford makes use of a priori knowledge about the number of line items in different price strata and thus obtains a more-than-adequate sampling procedure with a smaller percentage of items sampled.

Consulting assistance was provided in the sampling of wire which is to be tested for uniformity and character of chemical composition after test reactor irradiation.

A complete finished exterior contour of a 1251 Weapon component has been machined by the magnetic tape controlled experimental δ-ω lathe. This prototype part has been dimensionally gauged and subjected to surface finish measurements, and has been found to be well within the required tolerances.

The detailed mathematical analysis of the propagation of ultrasonic waves in elastic plates continues. A hitherto unreported phenomenon has been discovered which it is hoped will explain some of the puzzling discrepancies that have been observed in experiments with nondestructive testing equipment.

A report was issued which describes the theory and application of a FORTRAN language program for calculating cubic crystal lattice constants at various temperatures. Work continued on the problem of indexing hexagonal and orthorhombic crystals.

Authorized funds for 13 active projects total \$2,338,100. The total estimated cost of these projects is \$8,732,000 of which \$1,329,000 had been spent through August 31, 1962.

[redacted]

SUPPORTING FUNCTIONS

Preparations for the decontamination of the primary system of the Plutonium Recycle Test Reactor continued throughout September. The reactor was discharged and the primary system D_2O drained. During the drain, foreign matter in the D_2O was removed by filtration and ion exchange.

After various means of decontaminating the system without the use of chemicals were investigated, efforts were directed towards chemical decontamination. Principle materials planned for use are oxalic acid and caustic permanganate. Hazards reviews, procedure preparations, systems preparations, and trial runs of the procedures using demineralized water were in progress at month-end. Plans were completed to truck all waste to a crib south of 200 East area. All chemicals were received on site.

A facility was set up in the discharge water pit to decontaminate the fuel elements. Ultrasonic vibration with a detergent solution was used. Nine fuel elements were cleaned in this fashion until the program was deferred because of interference with preparations for primary system decontamination.

Extensive effort in improving and marking flux monitor wiring circuits was initiated while the reactor was discharged, and similar work on other systems was outlined.

Work on procedural matters related to the Plutonium Recycle Critical Facility was emphasized in anticipation of AEC approval to operate. Revision of the safety rods to permit maintenance without requiring cell cover block removal was started. Work was completed to permit individual rod drops as needed for future physics tests. Additional relief was provided to the auxiliary moderator storage tank to prevent over-pressurization. Modifications to improve readability of the galvanometer was 50% complete. Installation of indicating lights to signify safety circuit trip causes was completed. An interlock was removed which will now permit moderator pumping without on-scale indication from instruments. Work was completed to protect rod magnets and circuit relays from excessive voltage.

Construction of the Fuel Element Rupture Test Facility is 99% complete. The water plant was tied-in and PRTR and Rupture Loop services are being supplied from it. The main loop was chemically cleaned, and substantial amounts of grinding wheel abrasive particles and cuttings were recovered.

Project work on the Gas Cooled Loop was 92% complete. The replacement heater was received. One of the gas blowers successfully passed preliminary tests. Several punch-list items were completed.

Most FPD maintenance functions performed as a service to HLO were transferred to Test Reactor and Auxiliaries Operation on September 1, 1962. This organization performs routine maintenance for HLO facilities and provides craft assistance to special fabrications in support of research and development work.

The September physical inventory of heavy water disclosed a loss of 2, 227 pounds valued at $30, 716. This loss actually applied to August operations and represents a correction to the August physical inventory which was made entirely upon instrumentation. Actual measurements could not be taken in August due to contamination of facilities. Scrap material amounted to $20, 793 and represented the heavy water from the primary loop. Total charges to operating costs were $51, 509.

Six Ph.D. applicants visited HAPO for employment interviews. Two offers were extended; one acceptance and two rejections were received. One offer is currently open. Six program offers and three direct placement offers were extended to BS/MS applicants. One program and one direct placement offer were accepted; four program and three direct placement offers were rejected. Current open offers are four program and two direct placement. No permanent assignments were effected from the Technical Graduate Program. Four new members were added to the rolls and one terminated. Current program members total 59. Twenty-three non-exempt employment requisitions were filled during September; twelve remain to be filled. Each requisition involves a single vacancy.

Carl A. Bennett

for Manager
Hanford Laboratories

HM Parker:CAB:mlk

REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATION

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 2000 PROGRAM

1. METALLURGY PROGRAM

Corrosion Studies

Corrosion Testing of NPR Fuel Elements. Additional corrosion data have been obtained on a series of NPR fuel elements to evaluate the brazed welded end closure and various gray areas caused by the following:

1. Scuff marks from Teflon rollers.
2. Marks on the ends of the fuel elements produced by bumping the elements against the ends of the basket during etching.
3. Constant contact points between the ends of the fuel elements and the wire baskets during etching.
4. Copper contamination produced by spatter during spot welding of the supports.

Following five weeks of exposure in 360 C deoxygenated pH-10 water, no accelerated corrosion from the gray areas has been noted. The welded areas, which had an acceptable appearance prior to testing, have continued to slough and reform a loose powdery oxide. Metallographic cross sections of one of these welded areas has shown that the actual metal loss is less than 0.1 mil. Corrosion tests are continuing to evaluate further brazed and welded end closures. The loose white oxide at welds is being analyzed for uranium and beryllium concentration.

Magnetic Effects on Electroless Nickel Films. It was previously reported that electroless nickel films on aluminum were strongly magnetic only after heating. X-ray analysis indicated that amorphous as-plated films are converted to a crystalline form when heated. An as-plated nickel film was isolated from the aluminum substrate and cut into sections. The sections were heated for 30 minutes at 100, 150, 200, 250 and 300 C. Heating at the three lower temperatures did not increase the magnetic effect appreciably. However, heating

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at 250 C indicated that the amorphous-to-crystalline transformation occurs rapidly at this temperature. A film section heated in a vacuum at 300 C became strongly magnetic, ruling out the possibility that the magnetic behavior of autoclaved specimens is due to iron or nickel oxides or compound formation at the nickel-aluminum interface.

Basic Metallurgy Studies

Notch Sensitivity of Zircaloy-2. The mechanical properties of Zircaloy-2 cladding are being investigated. In support of these studies notched tensile samples of Zircaloy-2 rolled sheet are being tested in the irradiated and unirradiated condition. Forty-eight sheet samples notched to varying depths plus twelve unnotched co-extruded cladding specimens have been exposed to approximately 1×10^{20} nvt ($>$ 1 Mev) in the ETR. Irradiation temperature was that of the bulk water coolant (120 F). The samples are being tested remotely by use of an Instron tensile machine. Tensile testing temperature is 280 C at three different strain rates. Load versus extension curves are being compared with unirradiated controls. Metallography of fracture areas will be undertaken.

Metallic Fuel Development

Fuel Irradiations. Radiometallurgical examination is nearly complete on the variable braze-thickness irradiation test, GEH-4-68, 69, and 70. Extensive cracking in the uranium has been found in both ends of GEH-4-70, which was irradiated as the bottom specimen in the vertical column of three elements. The cracking appears to be confined to the braze-heat-affected zones of the uranium, but the entire fuel element is being sectioned to be sure the cracking is confined to the ends. Both end closures are in good condition with no cracks in the braze.

An experimental single tube, dual-enriched metallic uranium fuel element completed a third cycle of irradiation in the M-3 pressurized water loop in the ETR. Since the third cycle was short (14 days), the element was not removed from the reactor for examination. The element has accumulated a total exposure of about 700 MWD/T and is currently undergoing the fourth cycle of reactor operation.

An 18-inch long fluted fuel tube, equivalent in cross section to an N-inner fuel tube, was charged into the ETR P-7 loop September 3, 1962. This element is being irradiated under operating conditions similar to those anticipated for N-Reactor. Irradiation, with interim measurement and examination, will be continued to high exposure in order to evaluate the swelling capabilities of a fluted Zircaloy-2 clad fuel element.

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N-Reactor Fuel Rupture Test. The inner component of an N-Reactor fuel assembly was intentionally ruptured in the P-7 position of the ETR. The purpose of this test was to evaluate a new method of intentionally rupturing fuel elements under N-Reactor operating conditions. The fuel cladding was failed using a rupturing device consisting of a hydraulic cylinder that moved a cutter along the outer surface of the element. The rupturing device successfully cut through the 40-mil Zircaloy cladding exposing the uranium core to the loop coolant. This device can be used to rupture any surface of an irradiated N-fuel element.

Hydrogen Analyses of Irradiated Cladding. Hydrogen analyses of the Zircaloy-2 clad from four coextruded fuel elements irradiated in the KER loops to exposures up to 3600 MWD/T show little hydrogen pickup by the cladding during irradiation. The fuel exposures and hydrogen content of the cladding samples were: 1200 MWD/T, 70 ppm; 2000 MWD/T, 47 ppm; 2300 MWD/T, 30 ppm; and 3600 MWD/T, 47 ppm. The pre-irradiation hydrogen content of the fuel cladding varied from 25 to 35 ppm. The low hydrogen pickup substantiates the observation that negligible cladding corrosion occurred during irradiation.

Fluted Single Tube Fuel Element. A coextrusion for experimental fluted single tube N elements was sectioned and examined. Dimensional measurements made on several sections from the extrusion indicated that design dimensions for cross-sectional area and clad thickness had been met. Peel tests indicated that the cladding bond was acceptable. Six elements about 10-1/2 inches in length are being fabricated from material in the control portion of the extrusion. Three of these elements will be used in an irradiation test in the ETR.

Copper Braze Closure. Two N inner fuel elements, braze closed with a Cu-Zr braze, have accumulated 100 hours of autoclaving in 300 C water. One of the elements was in the as-brazed condition and one was electron beam welded over the brazed closure. The braze-contaminated welds showed some discoloration but no severe corrosion. The as-brazed closures were corroded down about 0.020-inch, showing that the braze alloy will corrode but has sufficient resistance to corrosion to limit the penetration in case of a primary weld failure. Successful brazes have been at about 880 C with an 80 w/o Zr-10 w/o Ni-10 w/o Cu brazing alloy. The alloy is very brittle, however, perhaps too brittle for reactor use.

Self-Brazed Closures. A group of five test elements (10 closures) has been prepared for high temperature autoclave testing. These specimens were of the flat-faced, skirted, pre-"tinned" design described in last month's report. Inspection shows them all to have

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sound primary welds, and since they were electron beam welded prior to Sciaky pressing, no contamination of the weld beads is anticipated. These will be autoclave tested next week for confirmation. Later, intentional defects will be put in the primary weld bead and they will be re-autoclaved to determine the resistance of the secondary (braze) closure to high temperature water.

The test specimen described in last month's report was sectioned and examined metallographically. The braze filler was seen to be soundly adherent both to the Zr-2 cap and to the uranium face. It was free from bubbles and discontinuities. The heat affected zone was found to extend only about 1/16-inch below the interface. No thinning of the jacket wall by gripping in the spot welding chuck was apparent.

Another group of five elements (10 closures) has been prepared for application of the seam welding technique as described in last month's report. They will be Sciaky pressed and autoclave tested similar to those described above.

Hot Headed Closure Studies. Continued effort to projection weld and bond end caps to the hot-headed fuel element has yielded projection welds and bonding at the interface between the "two ring" projections. However, it has not been possible to prevent the uranium from moving into areas between the cap and the fuel element cladding. In an attempt to prevent the uranium from moving into the cap to clad interface, caps were made with "four ring" projections. The first weld made using the "four ring" projection design without the presence of an interface material indicated that the welds can be made and that the uranium can be prevented from moving into the cap to clad interface.

Attempts were made to modify the heading process on NPR inner tubes to aid in thickening the ID shoulders for the subsequent welding of the hot headed closure. The ID shoulders are thinner than the OD shoulders as presently headed due to the thinner ID cladding and the heading process which tends to further thin the ID cladding formed over the shoulders. The ends of the tubes were machined so that the ID cladding should thicken during upset prior to the cladding forming over the ID shoulders. The results were not encouraging. Thickening was noted only on one of four ID shoulders examined. Further work will include modifying the ID heading punch to aid in thickening the ID cladding prior to forming over the shoulders.

Double Ring Closure. The shape of the closure cap and the power settings on the 600 KVA resistance welder were altered to attempt to improve the integrity of the weld between the cap and the cladding walls and improve the bonding of the cap to the uranium. The uranium is recessed in this closure approach to accommodate the closure cap and an attempt is made to initially weld the base of the cap to the cladding wall through small intermediate rings wedged between the cap and the cladding around the OD and ID cladding. Examination of the closures indicated that a very good weld was established between the cap and the cladding when the element OD and ID cladding was fairly concentric. Eccentricity tended to allow uranium to "spike" by the weld during the following cap to uranium bonding. Cap to uranium bonding appeared good in all cases using a copper plate on the cap to aid in bonding.

NPR Fuel Element Charging Studies. Testing of the N-Reactor fuel element support continued in an attempt to charge the equivalent of one fuel charge through the test tube. All elements were charged singly. The process tube was rotated and elements 2 through 18 were charged with no evidence of scratching. The monofilament drawline broke (due to previously incurred damage) after charging the nineteenth element approximately 10 to 15 feet. It was then necessary to push the element out of the process tube with the tube cleaning device. The cleaning device had been idle for about 10 days before using it for this purpose. A probability exists that corrosion scale was flushed from the cleaning device during this operation. The element supports scratched the process tube from 19 feet to the discharge end of the tube.

The process tube was recleaned and rotated to expose new surfaces to the fuel supports. Music wire, 0.030-inch diameter, was substituted for the monofilament drawline. Element 21 induced a very light scratch in the last five feet of the tube. Element 22 induced a scratch 18 feet from the charge end of the tube. The scratch initiates at the bottom of the tube which indicates the element was rotationally unstable (rotated at least 45 degrees from gravitationally stable position) and examination of the supports shows wear areas on six supports. Detailed borescope examination has not yet yielded a definite cause of scratch initiation.

N-Reactor Fuel Element Supports. An interdepartmental task force was established by FPD to critically review N-Reactor fuel element support designs and criteria in view of modifications in the fuel production process and test results obtained since the original criteria were formulated. Assistance is being rendered in the design, fabrication, and testing of new support models. A series of outer support models having different longitudinal radii in the crown were made and tested in compression for two thicknesses of Zircaloy-2. The results indicate some variations of these models may meet the proposed new strength and impact resistance criteria.

In order to determine what designs are suitable and what criteria can be satisfied by the supports, a series of analyses are being performed. The analyses include: vibrational amplitudes for a two component fuel element with damping for various support spring constants; bending resistance for various crown radii and width; and maximum elastic energy absorption, deflection, and load for various thickness tubular flow through supports. The results of these analyses are being factored into the present study of support design and support design criteria.

The laboratory program to fabricate a demonstration quantity of N inner supports consists of three phases -- material development, forming method development, and the establishment of quality control tests. The material development phase of the program has resulted in the production of Zircaloy-2 sheet of higher bend ductility than that of commercially produced sheet. The fabrication schedules which result in the most ductile sheet are based on extrusion as the method of primary reduction from ingot to sheet stock for rolling or rod for turksheading. The crystallographic texture is the controlling factor in the amount of bend ductility which can be imparted to the finished sheet. Extrusion results in a favorable texture which can be partially retained through subsequent rolling stages by proper control of working and annealing temperatures. Present indications are that rolling and annealing temperatures should be much lower than those currently used in the commercial fabrication of Zircaloy-2 sheet. One fabrication schedule based entirely on rolling shows some promise and is being critically evaluated. This schedule involves cross-rolling and close temperature control in the final reductions. Pilot quantities of sheet from five candidate schedules have been prepared and test lots of N inner supports will be fabricated from each lot of material using present production methods.

The forming method development phase of the program has included the development of two different forming methods. The equipment for these methods has been designed and is being built. A series of annealing experiments is being conducted to establish the effect of intermediate anneals in the forming process.

Concurrent with the experimental work on sheet fabrication schedules, various test methods have been applied to each batch of sheet produced in order to establish an acceptance criteria. A bend test has been developed which suitably discriminates between acceptable and non-acceptable material. Tensile test results have been obtained on all experimental material and are being studied for possible use in quality control. Strain measurements on tensile specimens have provided qualitative information on texturing in the finished sheet products.

2. REACTOR PROGRAM

Gas Atmosphere Studies

Oxidation of Zirconium and Zircaloy-2 in Carbon Monoxide. Previous hydriding experiments have indicated negligible oxidation of Zircaloy-2 in a He, 2% CO, 4% H_2 mixture implying published data on zirconium oxidation in CO are in error. The corrosion rates of crystal bar zirconium and Zircaloy-2 have been measured in 1 atm CO carefully purified of oxygen and water vapor over a temperature range of 390 to 750 C. The measured rates are a factor of 20 to 30 lower than the published data, suggesting previous workers did not take adequate precautions to exclude water and oxygen from their test system. The new CO oxidation rates confirm the Zircaloy weight gains measured in the hydriding experiments and show why CO is an ineffective hydriding inhibitor.

Electrical Resistance Capsule. A capsule has been charged into the 2A test hole at KE to measure the effect of fast neutron radiation on the electrical resistance of the ZrO_2 corrosion film on Zircaloy-2. Immediately after reactor startup, the resistance at 450 C in an oxygen-helium mixture dropped from 60 megohms to 120 K ohms, which is indicative of a higher concentration of oxygen anion vacancies and interstitial defects in the corrosion film produced by the fast neutron flux. After three days, the reactor shut down and the resistance immediately returned to its pre-irradiation value at 450 C because the elimination of the neutron flux reduces the steady state concentration of lattice defects.

A plot of log resistance *vs* 1/temperature prior to reactor charging gave a slope of 10 Kcal for the activation energy of thermally activated electrons, whereas after reactor startup there was almost no variation of R with temperature.

Four days after pure helium was substituted for the helium-oxygen mixture the electrical resistance, in-reactor, of the Zr oxide began to decrease. The decrease in resistance is thought to be caused by the increase in the steady state concentration of oxygen vacancies resulting from the limited availability of oxygen to fill lattice vacancies.

The in-reactor measurements confirm the general picture of the electrical conductivity mechanism of ZrO_2 on zirconium. The total measured conductance is a function of the electrons associated with corrosion produced oxygen vacancies, radiation produced defects, and thermally activated electrons and holes. Holding any two of these environmental variables constant and varying the third produces the expected results in film resistance. Experimental verification of a radiation induced increase in ZrO_2 film conductivity implies creation of both oxide vacancies and interstitials. The increased Zircaloy corrosion rate measured in a previous capsule experiment is thought to result from accelerated oxygen diffusion via radiation induced oxide lattice defects.

Graphite Burnout Monitoring. Burnout rates on small specimens in channel 3580 at F Reactor continue to be low. During the latest test from July 2 to August 26, 1962, no sample was oxidized at a rate greater than 0.72 percent per thousand operating days (KOD). A plot of the rates against stack position showed a gradual decline from 0.72%/KOD at 60 inches to 0.40%/KOD at 190 inches. There was no distinct peak at 80 to 100 inches as would be expected if oxygen were present in the reactor gas atmosphere. It is possible that the burnout in this test was due to the radiation-induced CO_2-graphite reaction. The gradual decline of the rates is probably due to the inhibition of the reaction product carbon monoxide.

Burnout rates on small specimens were also measured at B Reactor in channel 3461. The results showed a rather sharp peak of 4.8 %/KOD at 140 inches into the graphite stack. Except that the peak height is a little lower, the results are the same as for the two previous test periods from April 7 to June 16, 1962, and from June 16 to July 23, 1962. The reaction responsible for this peak is not fully understood. The rate seems too high for the thermally induced CO_2 graphite reaction. Using Burton's data (R. E. Woodley, HW-74663, p. 8) and an estimated average of 15 percent CO_2 and a graphite temperature of 650 C (obtained from P. J. Zimmerman, B Reactor engineer), the calculated rate is not greater than 0.1%/KOD. The water-graphite reaction seems unlikely, too, because F-Reactor consistently operates wetter and yet shows lower burnout rates. Furthermore, the peak seems too far downstream for the O_2-graphite reaction; in tests at another reactor where there was a known air leak, the peak occurred in the area 80 to 100 inches into the graphite stack. Other possible explanations that involve radiation-induced oxidation are being checked. To shed more light on this problem two additional channels, 0373 and 1366, were charged with burnout monitors. The test is scheduled for discharge late in October.

Corrosion and Coolant Systems Development

Corrosion of Zircaloy Under Heat Transfer Conditions. The fifth heat transfer test was initiated during the month. A Zr-2 clad heater operating at a heat flux of 280,000 Btu/hr-ft^2 was equipped with thermocouples both in and adjacent to the clad and exposed to lithiated water at a pH of 10.0 and a pressure of 2000 psia. The loop temperature was adjusted to 628 F at which point the initial surface temperature was measured at 635 F or slightly below nucleate boiling. Measurement of the temperature drop across the water film, oxide film, and crud film is accomplished by comparison of the thermocouples in and out of the cladding. An initial value of 8 F was obtained on the uncorroded etched surface. During the first 50 hours the Δ T increased to 11 F and steadily increased to a value of 26 F up to 420 hours. At this time the test was interrupted to perform a 53-hour test at 80 C and a pH of 5.0-5.6 using CO_2 for pH control. This short-term test was in support of the PRTR decontamination effort and was conducted with the heater in place but not operating. Following flushing of the loop, operation of the heat transfer test was resumed. The Δ T had decreased to 23 F on startup and slowly started again; after 447 hours total exposure, the Δ T was 24 F.

Corrosion Studies in Decontaminating Solutions. Corrosion tests were completed to evaluate a commercial decontaminant based on ammonium citrate. The corrosion rates of carbon steel were about twice those experienced with a compound formulated at HAPO (0.08-0.09 mil/hr as compared to 0.04-0.05 mil/hr).

Dissolved Oxygen Analyzer. Laboratory evaluation of a commercial dissolved oxygen analyzer was completed. System performance is satisfactory. Response time measurements for detection of concentration changes in the coolant during on-line operation indicate that response is complete in 40 seconds. Analyzer response time during calibration, however, is nearly three mintues unless the electrolytic calibration cell potential is adjusted to just below the value required to initiate electrolysis just prior to calibration. In the latter case the response time is reduced to 50 seconds. This variation is due to transient conditions in the calibration cell when the potential is instantaneously increased from 0 to the value required to generate a known oxygen concentration.

Evaluation of Thermocouple Slugs for In-Reactor Tests. Three thermocouple slugs with brazed Zr-2 clad thermocouples were received and charged. Two elements were placed in TF-9 and one in TF-7; after $142\frac{1}{2}$ hours of operation, the Loop TF-9 was discharged. The upstream element, No. 4, was severely ruptured. The failure probably occurred at the welded end cap. Based on this, it has been decided that the elements for the reactor will have both end caps brazed, similar to those of a typical NPR fuel element. The upstream element No. 2 was visually examined; it was covered with a layer of brown crud but appeared to be sound.

A test of a Zr-2 clad thermocouple was completed after 1014 hours. The thermocouple surface was very rough. It was pitted (pits approximately 7-10 mils deep), and there were several patches of white oxide covering approximately 35% of the surface. There were wear marks where rubbing had occurred, but there was no evidence of fretting.

Evaluation of Low pH and Low Dichromate. One production test is in effect to evaluate corrosion of fuel elements in water at low pH (6.6) and low inhibitor concentration (1.0 ppm dichromate). One set of elements was discharged and is being examined for general and localized corrosion. Preliminary measurements indicate that total corrosion of aluminum is slightly greater in the water with reduced inhibitor than in the standard process water, at pH 6.6. Average corrosion values from two test locations were 1.95 mils and 1.99 mils after $14\frac{1}{2}$ weeks of exposure, while control location values were 1.62 mils and 1.75 mils. Additional measurements of aluminum corrosion, and of carbon steel corrosion, are being made.

Structural Materials Development

Prototype Irradiated Burst Test Equipment. The prototype equipment for burst testing irradiated tube sections has been modified to minimize specimen temperature variations. A new test assembly support base has reduced the convective and conductive heat loss from around the bottom of the specimen; rewiring has provided the option of varying the ratio of heat input to the bottom and top of the furnace; and a new inlet arrangement introduces the incoming high pressure water near the top of the specimen. Before the modification, the temperature variation over the length of the sample was 21 C at steady state, and 32 C when the sample burst. After the modifications the temperature differential was 5 C at steady state and 2 C at the time the sample burst with a temperature drop of 8 C in the interval.

Graphite Studies

NPR Graphite Irradiations. The third first-generation capsule, H-6-1, in the series of long-term irradiations of NPR graphite was removed from the General Electric Test Reactor on August 24, 1962. The capsule was successfully irradiated for four reactor cycles or a total of 96.6 effective days at full reactor power. The reactor cycle just prior to capsule removal was a non-operating cycle so the graphite received a radiation exposure of approximately 80 percent of that originally scheduled. The maximum sample exposure is estimated to be 2×10^{21} nvt, E$>$ 0.18 Mev, which is equivalent to approximately two years of exposure in the NPR. The lead which had been melted into the annulus between the inner and outer thermocouple leadout tubes apparently provided adequate shielding for the thermocouple connector as there was no evidence of failure of this seal.

The H-6-1 cpasule was disassembled in the hot cell and found to be in good condition. There was a slight amount of surface erosion on the samples in positions 4 and 5, which operate at the highest sample temperature, approximately 800 C. This erosion was in evidence only on the exposed external surfaces. The sample ends, which are used for the primary length measurements, were protected by the sample holders and were in excellent condition. A carbon and graphite deposit slightly thicker than is typical for these capsules was observed on the inner wall of the aluminum shell and inside of the bottom plug. This was probably a result of the sample erosion and redeposition. Some of this deposit is being returned to HAPO for analysis. All samples and flux monitors were recovered and are being returned to HAPO. Eight of the nine thermocouples in the capsule operated satisfactorily for the entire irradiation period; number 9 failed approximately 11 days prior to the end of the irradiation. The failure was caused by a break in one of the wires at the bend where the thermocouple enters the sample. Calibration will be checked on three of the thermocouples.

The second second-generation capsule, H-5-2, in the NPR series was installed in the GETR on September 1, 1962. The reactor has not started up on Cycle 37 so that capsule operation is not yet known. The status of the first second-generation capsule, H-4-2, is unchanged from last month.

Stored Energy. On September 8, Irradiation Testing Operation, IPD, obtained 19 samples of the moderator graphite from two process channels of D-Reactor. The neutron exposure to D-Reactor moderator has more than doubled since the last core samples were obtained in March of 1958. These core samples will therefore be used to determine the changes which have occurred in the total stored energy content since March 1958.

X-ray measurements of the crystal parameters are being obtained at 1/16-inch intervals along the samples. Eleven of these core samples will be sectioned into three parts and sent to NBS for total stored energy measurements. It is expected that the stored energy results will be available from NBS in about four months. At that time the X-ray measurements and total stored energy data can be combined to provide an estimate of the total stored energy distribution within the process tube blocks. This information will be available for the region between 7 ft-6 inches and 16 ft-4 inches of 2171-D and the region of maximum expansion in 2495-D.

The remaining eight samples will be used to obtain the stored energy release spectra and rate of release at temperatures below about 900 C. These data will make it possible to re-evaluate the conditions under

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which a stored energy release could occur and the effects of such a release.

The x-ray data obtained to date indicate that there probably have been increases in the total stored energy content of the graphite in spite of the increase in maximum graphite temperature at D-Reactor. However, the magnitude of the changes cannot be determined until the total stored energy data are obtained from NBS.

Thermal Hydraulics Studies

Visual Studies of the Effects of Fuel Supports on Boiling. Laboratory experiments were continued in the study of heat transfer conditions as affected by devices used to center fuel elements in the process tubes of the Hanford production reactors. The test section for these experiments consisted of a 1.340-inch OD electrically heated tube placed inside of a 1.504-inch ID glass tube with water coolant flowing in the annulus between the two tubes. The electrically heated tube had several different types of centering devices attached to the heated surfaces. These included BDF reactor type bumpers, a new submarine shaped bumper, a BDF type of self-support which had been intentionally crushed, and an aluminum oxide support often used for electrical insulation of test sections in the heat transfer laboratory.

The experiments were run at flows corresponding to two different conditions existing in the flow annuli of fuel elements in a reactor; these were 32.4 gpm representing a reactor central zone tube and 18.6 gpm representing a reactor fringe zone tube. Visual examination and high speed motion pictures were made of the test section while the heat generation was gradually increased. The following is a summary of the data.

Run	Conditions	Max. Heat Flux Btu/hr-sq ft	Comments
34	Fringe zone - test section with submarine shaped bumpers placed 40% off center toward wall of glass tube.	206,000	No visible boiling.
		306,000	A few very fine bubbles were formed in a scratch. They formed slowly and were swept away rather infrequently.
35	Same as Run 34 except central zone conditions.	458,000	No visible boiling.
		687,000	" " "
		918,000	A few fine bubbles were originating underneath one bumper. No general surface boiling.

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Run	Conditions	Max. Heat Flux Btu/hr-sq ft	Comments
36	Fringe zone - test section with aluminum oxide supports placed 30 to 40% off-center toward wall of glass tube.	310,000	A few small patches of bubbles were formed on the rod surface away from support. Bubbles were very fine, formed slowly, and were swept away infrequently.
		407,000*	An occasional bubble was formed in rod slot at downstream of one support.
37	Same as Run 36 except central zone conditions.	707,000	No visible boiling.
38	Fringe zone - test section with BDF type bumpers placed 50% off-center toward wall of glass tube.	206,000	A few fine bubbles originated under one bumper. Also, a small patch of rather stable bubbles existed in the wake of one bumper.
		305,000	Steady stream of bubbles issued from under sides and downstream of one bumper. A few bubbles formed slowly between bumper and glass.
39	Same as Run 38 except central zone conditions.	458,000	A few fine bubbles originated under downstream end of one bumper.
		599,000	A steady stream of bubbles were coming from under sides and downstream end of bumpers.
40	Fringe zone - test section with BDF bumpers and crushed self-supports alternated along the length placed 50% off-center toward wall of glass tube.	205,000	A few fine bubbles issued from downstream end of both type bumpers.
		305,000	Bubbles issued from under all bumpers steadily.
		453,000	Greater quantity of bubbles issued from under bumpers. No detectable difference in support type used.

Run	Conditions	Max. Heat Flux Btu/hr-sq ft	Comments
41	Same as Run 40 except central zone conditions.	460,000*	Small number of bubbles issued from under both bumpers.
		696,000	Steady streams of bubbles from under both type bumpers.
		910,000	No definite difference between two types of bumpers. Very severe wakes downstream of bumpers which were downstream of normal viewing position - appeared to be near burnout conditions. Steady bubble streams from bumpers in normal viewing position.

*High speed motion pictures were not taken of these runs.

It was concluded from the observations that considerable boiling would take place on the fuel elements in the vicinity of the centering devices when the fuel was allowed to become greater than 40% off-center toward the wall of the process tube. It was also concluded from visual observations that there might be a considerable difference between just one element being placed off-center and the case where all the elements in a tube were off-center. Additional experiments were planned to examine the case where the fuel elements were 100% off-center and resting against the tube wall.

Critical Discharge Rates of Steam-Water Mixtures. Additional experiments were run to determine the flow rates that exist when high pressure steam-water mixtures are discharged from a pipe into atmospheric conditions. Such information concerning critical flow is of value in reactor hazards calculations involving piping breaks. During the experiments seven constant pressure blowdowns of a 1000-gallon water tank resulted in 30 steady state data conditions. For the runs water was stored in the pressurized tank at the following temperatures and pressures.

Run No.	Temperature (^oF)	Pressure (psig)
1	450	1000
2	500	750
3	500	1500
4	500	1500
5	500	2000
6	550	1500
7	550	2000

This water was allowed to expand to the atmosphere through a 1/2-inch flow passage, and the maximum discharge rates were measured along with the axial pressure profiles. A valve was placed upstream of the test section to set the total flow resistance in the flow path, and therefore set the desired magnitude of the pressure and steam quality at the inlet of the test section. Flow rates from about 200 to 1500 lb/minute were observed and axial pressure gradients greater than 1000 psi per inch were measured near the discharge end of the flow passage. With such tremendous flow rates and pressure gradients, it is doubtful that there is sufficient time to allow the required transfers of heat and mass necessary for equilibrium expansions. Furthermore, no abrupt changes in the pressure gradients near the discharge were detected as was the case of critical discharge occurring at lower pressures studied earlier. These two factors suggest that the method of correlation used in the past by many investigators is inadequate to describe the occurrence of the phenomenon at high pressures.

B. WEAPONS - 3000 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 - 4000 PROGRAM

1. PLUTONIUM RECYCLE PROGRAM

Plutonium Fuels Development

Irradiation of Special Elements in the PRTR. Process tube activity records indicate that the failure of an $MgO-PuO_2$ element in the PRTR during August occurred after about three hours of 60 MW reactor operation, and the element was discharged after about 160 hours of full power operation due to excessive coolant activity. Radiometallurgical examination of the ruptured rod, now under way, showed the cladding had swelled in the immediate region of the failure and that the latter was of a ductile type. These observations suggest an internal pressure, perhaps from steam pressure as a result of waterlogging and/or from a volume increase of MgO as a result of hydration. Rod diameter measurements taken on each side of the failure were unchanged except for a localized diameter increase which occurred about two inches below the failure (an increase of 0.036-inch) and one which occurred about six inches above the failure (an increase of 0.013-inch). The swelled areas were sectioned longitudinally and showed that the core washout has proceeded to these points and that some hydration of the MgO had occurred.

Autoradiographs made by exposing the ruptured rod to glass show areas of high fission product concentration caused by plutonium segregation in the fuel material. It is possible, but most unlikely, that such an area of higher heat generation contributed to the failure; there were no indications of cladding burnout or nonuniform discoloration of the cladding surface.

The cladding near one end of the rupture is being examined metallographically. Results of this examination to date indicate that there were no cladding defects such as long internal cracks; there is no observable reaction between the $MgO-PuO_2$ core and the Zircaloy cladding; the zirconium hydride concentration is about normal for autoclave tested zirconium; the fracture was ductile; and the cladding structure in some areas is characteristic of a strained material. Examination of the cladding along the entire length of the rupture is continuing.

UO_2-PuO_2 Capsule Irradiations. Capsule GEH-14-85, containing high density UO_2-2.57 mole percent PuO_2 and having an irradiation exposure of about 3×10^{20} fissions/cm^3 or 10,000 MWD/ton of UO_2-PuO_2, was measured and shows a maximum cladding diameter increase of 0.014-inch

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or 2.4 percent. This capsule and GEH-14-86 (exposure - 5600 MWD/ton of UO_2-PuO_2) are currently being drilled for fission gas collection and analysis.

PRTR Prototype Irradiation Testing. Examination of the 42-inch long cosine enriched UO_2-PuO_2 seven-rod cluster is continuing in Radiometallurgy. Dark spots on the external surface of the fuel rods are associated with high plutonium concentration regions and result from localized areas of higher surface heat fluxes. A transverse cross-section was made through one of the dark spots indicated on the external cladding surface. A reaction layer had formed on the Zircaloy cladding and a high porosity particle (probably high in plutonium concentration) was bonded to the cladding. These observations are indicative of high fuel temperatures next to the cladding caused by plutonium segregation.

Phoenix Experiment. The plutonium burnup experiment designed to investigate the effect of burnup on the reactivity of high exposure plutonium is continuing. Current status of the samples is as follows: the sample which contains plutonium which initially had 6.25 percent Pu-240 (GEH-21-1) has received five cycles of irradiation and its reactivity has been measured in the ARMF; the sample containing plutonium with 16.33 percent Pu-240 (GEH-21-3) has been irradiated for its sixth cycle; and the sample containing plutonium with 27.17 percent Pu-240 (GEH-21-19) has been irradiated for six cycles and ARMF measurements have been completed. Goal exposure of these samples is seven MTR cycles. Continuous reactivity measurements were started on GEH-21-3 within six hours after the MTR shutdown following its sixth cycle of irradiation. The purpose of this transient measurement is to determine the reactivity effect of the short-lived fission products. A similar measurement was made on the same sample following its first cycle of irradiation.

Preliminary data indicate that the sample which contains 27.17 percent Pu-240 has about five percent more reactivity than the sample which contains 6.25 percent Pu-240 after about three cycles of irradiation.

MgO-PuO_2 and ZrO_2-PuO_2 Irradiation Capsules.

Four of the eight irradiated MgO-PuO_2 capsules are being examined in Radiometallurgy. The PuO_2 second phase is clearly visible in the MgO - 13.5 w/o PuO_2 material. The PuO_2 is randomly distributed in the MgO matrix in the equiaxed grain growth regions as well as the columnar grain growth regions where recrystallization and grain growth has occurred. The population density of the PuO_2 particles is essentially unchanged regardless of structure changes which have occurred in the surrounding matrix. There is a limited PuO_2 deficient region adjacent to the center void which suggests that some of the PuO_2 has vaporized at the very high central core temperatures and redeposited in cooler zones.

Extended Surface Plutonium Fuels. Contamination control is the principal problem preventing scale-up of the roll-cladding process. Bare ten-gram pieces of Pu-Zr alloy cores have been roll-clad successfully. These plates were sheared within 1/4-inch of the core without encountering contamination. However, when the size of the core was increased to 25 grams, widespread contamination occurred on the outside of the finished plate after roll-cladding. Smearable contamination could be removed, but nonsmearable contamination existed throughout the bond zone between the plates. To prevent this contamination, copper was electroplated on the surface of the plutonium-zirconium alloy. With the proper etching and plating conditions the alloy was completely coated so that no smearable contamination was detectable. Two copper plated cores which had very low level contamination on the surface were assembled in sandwich and roll-clad. In both cases widespread contamination existed on the plates after rolling. The contamination level was higher from the copper plated cores than from bare cores. The cause of the increased contamination is being investigated. In order to determine the effect of the copper on the core-clad bond, one of the plates was destructively tested. The bond appeared as strong and ductile as the parent metal in this qualitative test.

Uranium Fuels Development

Irradiation of a Large Diameter UO_2 Rod. Irradiation of a 2.33-inch OD UO_2 fuel rod (GEH-12-29) in the ETR-P7 loop was terminated when the element developed a cladding split at approximately one-third the design heat output rate. Preliminary post-irradiation examination revealed a bulge and a split in the cladding. A significant diameter increase (0.010 to 0.020-inch) and circumferential ridges on the Zr-2 cladding were observed with the height of the ridges (0.003-inch maximum) greatest near the mid-plane of the element.

Examination of the cross-section revealed a central void surrounded by large columnar grains. The appearance suggests that the central UO_2 reached a higher temperature than anticipated. The fuel structure indicates that a severe diametral variation in neutron flux existed during the short irradiation. Bowing of the element would be expected to develop a hot spot near the midplane, subsequently overheating the fuel and cladding to cause the bulge and split.

Remote Fabrication Studies. The fuel element remote fabrication cell now includes the following operating units:

a. A welding turntable with preset rotational speed control (± 1 percent).

b. A welding power supply with a control circuit to provide accurate synchronization with the turntable.

c. A two camera closed circuit TV installation for remote viewing.

d. An overhead beam vibrational compaction system modified to test the characteristics of different beam lengths and stiffnesses.

An experimental inert gas welding chamber, designed to provide accurate placement of the welding electrodes with a variety of tube sizes, was received.

Thermal Hydraulics Studies

Shutdown Cooling of the PRTR. Studies pertaining to cooling of fuel elements in the PRTR during a total power outage were continued. As discussed in the last monthly report, results of pressure-drop measurements during low flow through the primary system indicated the possibility that liquid natural convective circulation rates would be inadequate to prevent boiling and vapor binding in the system. In addition, calculations indicated that injection of light water into the system would not assure prevention of excessive fuel temperatures either since adequate cooling might not exist during the 15 minutes it would take to depressurize the system to where light water could be injected.

Additional calculations have been performed concerning the method of cooling by injection of light water. The time involved in depressurizing the primary system from operating pressure to the 500 psi available from the diesel well pump is influenced for the most part by the flow resistance through the vent valve as the primary coolant leaves the system. Preliminary calculations indicated that the installation of a four-inch vent valve in addition to the existing two-inch valve would result in the depressurization being accomplished in a time short enough that there would be little doubt of adequate cooling during the depressurization time.

Component Testing and Equipment Development

PRTR Mechanical Shim Rods. Detailed design on the second generation mechanical shim rod is approximately 85% completed.

Shim rod Environment Control Test Facility. Work on the shim rod environmental test facility was resumed following a period of inactivity due to a shortage of craft personnel and higher priority work. Fabrication and installation of the facility is approximately 40% completed.

EDEL-1 Renovation. Preparation was started of manual to aid new personnel in operating and maintaining the pressurized water loop used for equipment development. Installation of electrical power wiring, controls, and new safety circuit was completed. The repairs of the pump motor and stationary field adjustable speed drive was completed by the off-site vendor and the repaired unit was shipped on September 19. Fabrication and installation of new stainless steel deionized water piping and a steam condensate supply to the deionizers is 80% completed.

Fretting Corrosion Investigation. Investigation of equipment and test procedures to detect relative motion of a fuel element in a PRTR pressure tube mounted in the EDEL-I loop continued. Bench tests to determine the suitability of eddy current techniques to measure relative vibratory motions of a PRTR fuel element and pressure tube were continued by Physical Measurements Operation. Methods tested included (a) single sensing element circuits, (b) single-coil, dual-sensing-element circuits, and (c) double-coil, dual-sensing-element circuits. Additionally, the possibility for use of two-frequency excitation which would permit the added determination of relative vibration of the fuel element and shroud tube was given cursory study.

All of the circuits examined provided sensitivities (measured at the test coil terminals) of approximately 1.0 mv per 1 mil of relative movement of the fuel element with respect to the pressure tube. Sensitivity of the shroud tube motion was lower by a factor of about 6, however.

As yet, no choice has been made of the specific method to be pursued for development.

PRTR Shroud Tube Replacement Mockup. Excavation of the 32-foot deep pit and installation of the caisson for housing the shroud tube replacement mockup was finished. Earth backfilling and concreting around the caisson, fabrication of a top cover and painting with a prime coat will complete the caisson.

Inlet Bellows to Pressure Tube Gas Seal. Preliminary effort is under way to adapt the B-F ferrule seal design to the test section for the rupture loop. This seal was developed for the seal between the PRTR inlet bellows and pressure tube, and although quite expensive, is the only satisfactory seal thus far tested.

PRTR Rupture Loop In-Reactor Test Section. Repair of the EDEL-II circulation pump has been completed. This consisted of installing a new shaft and guide bearing, and opening a plugged hole in the casing cooling jacket so that sufficient cooling flow could be attained. After checkout of the repaired pump, the loop will be ready to perform the thermal cycling and stress tests on the Grayloc connector.

All work other than direct engineering coverage on this program was stopped September 17 as a result of lack of funds. A review of the remaining work to be done and a cost estimate to complete the work is being prepared for submittal to TRAO.

Rupture Loop Discharge Equipment. Work on the discharge equipment has been interrupted awaiting additional funds.

PRTR Gas Loop In-Reactor Test Section. A means of carrying the thermocouples for monitoring for excessive test section bowing in the reactor was worked out and will bring the couples out through the top shield centering flange. Installing the passages in the centering flange and finishing the "As-Built" flange drawing remain to be done. Special thermocouples for this installation are currently on order.

At month's end work on the gas loop components has ceased for lack of funds. A cost estimate for remaining work has been submitted to Maintenance and Equipment Engineering of TRAO.

Gas Loop Sample Charge-Discharge. Drawing of the sample holder installation and removal tool was issued for approval. A design for shipping cradles for the sample casks was issued for comments and comments received.

Hazards Analysis

Reactor Safeguards Reviews. The following information was presented at the eighth meeting of the General Electric Technological Hazards Council September 13, 1962.

a. Review of PRTR primary coolant system straightening vane failure.

b. Review of PRTR primary coolant pump check valve modification.

c. Review of PRTR fuel element rupture.

d. Control of excess reactivity in the PRCF, prepared in response to Technological Hazards Council suggestions.

Reactor Kinetics Studies. Reactor kinetics studies, utilizing an analog computer, were completed for a uniformly enriched PRTR core consisting entirely of mixed crystal, UO_2-PuO_2, fuel elements. Three different plutonium enrichment levels were considered in these studies, 0.43 w/o Pu (approximately 56 percent fissions in plutonium), 0.75 w/o Pu (approximately 70 percent fissions in plutonium), and 1.5 w/o Pu (approximately 83 percent fissions in plutonium). For each level of enrichment, assumed excursions were initiated by mechanism similar to those reported for spike enrichment cores in the PRTR Final Safeguards Analysis, HW-61236. Preliminary analysis of the analog data indicates that excursions involving a uniformly enriched core would be no more severe than excursions with a spike enrichment core containing natural UO_2 fuel elements and plutonium-aluminum spike enrichment fuel elements.

DC Electrical System. The adequacy of the backup provisions for the PRTR 125 V DC electrical system was investigated. This system has two functions: to operate the safety circuit relays and to provide the control power for the electrical switchgear. The system consists of a set of storage batteries and two 25 amp battery chargers, which cannot be operated simultaneously. Since the charging rate is only slightly greater than the predicted DC system load after startup of the PRCF, PRTR gas-cooled loop, and PRTR fuel element rupture test facility, it was concluded that the adequacy of the 125 V battery system will be marginal.

Analog Studies of Nuclear Excursions. Continued analysis of PRCF analog studies showed that the formulated models for moderator void formation by heat transfer and enhanced resonance absorption in U-238 provide sufficient negative reactivity to override the simulated nuclear excursions. The exponential power level rise was terminated after approximately 15 seconds. Average fuel temperatures of 300-400 F were required to override the excursions. The ramp additions of reactivity (10 cents per second for 11 seconds) were completed before the peak power level was reached. Analyses of the reactor transients for the "no scram" cases showed that the power level rose exponentially until the combined effect of the moderator void and negative fuel temperature coefficient was strong enough to turn around the excursion. The negative reactivity required to override the excursion is an amount which will reduce the system to delayed supercritical conditions and was observed to be approximately 1/3 of the positive reactivity added to initiate the excursion.

Meltable Driver Fuel Elements. The D_2O moderated PRCF does not have very strong inherent shutdown mechanisms. A study has been initiated to explore the feasibility of using thin wall cylindrical driver fuel elements which would melt at about 100 C. It has been suggested that these hollow, cylindrical fuel elements be fabricated of a low temperature meltable bismuth alloy containing plutonium. The reduction in K due to increased self-shielding and leakage after melting, the time required to melt the fuel elements, and the enrichment of the meltable fuel will be determined. A set of cross sections has been assembled for use in PRTR and PRCF calculations and attempts have been made at matching the reactivity of the meltable fuel element with plutonium-aluminum driver fuel elements.

PRCF Process Specifications. Preparation of comment issues of PRCF Process Specifications is approximately 90 percent complete. It is estimated that 25 percent of the comments have been reviewed. Preparation of final drafts for approval is approximately 40 percent completed.

Design Studies

PRTR Burnup Analysis. Calculations of fuel burnup transients in the PRTR were carried out for a number of fuel loadings between startup and the present time. Three-group perturbation theory was used to calculate reactivity coefficients from the different fuel types and locations. When the coefficients were combined, the net reactivity change per MWD of reactor output was found to be in good agreement with measurement. The method will be utilized to improve future estimates of the reactivity lifetime in PRTR. These results are summarized in a paper "Experience With the Plutonium Recycle Test Reactor" presented at the recent ANS topical meeting on the use of plutonium as a power reactor fuel.

Materials Development

Decontamination of PRTR. The previously mentioned failure of an $MgO-PuO_2$ fuel element deposited plutonium and fission product activities throughout the PRTR piping and radioactive particulate matter settled in many of the numerous dead-legs in the system. The activity was reduced to a considerable extent by flushing with filtration and by draining the dead-legs, but still remained too high for efficient operation. First, mechanical means and, when these were found inadequite, chemical means were prescribed for decontamination of the entire PRTR system.

A great deal of time was spent evaluating methods for removing activities by mechanical means. Hot water at high speed was flushed through a pipe section with: (a) no additions; (b) addition of air; (c) addition of detergent; (d) addition of CO_2; and (e) addition of diatomaceous earth. The results showed these methods were uniformly ineffective. Of more or less academic interest, an attempt was made to remove activity from piping sections by ultrasonics with and without detergent; this procedure removed about 25 percent of the activity.

The chemical decontamination problem was twofold: (1) to dissolve the crud, plutonium and magnesium oxides, and other particulate material wandering throughout the system; and (2) to remove the activities adsorbed on the piping. Previously it had been shown that it was impossible to remove the adsorbed activities without removing the adsorbed film; therefore, some film stripping procedure had to be developed.

A series of dissolution studies was started to find some method for dissolving particulate plutonium oxides. The two most effective reagents (HBr and HF) could not be used. Sulfuric acid with dichromate or permanganate and buffered oxalate-peroxide solutions were more or less effective. Because of the carbon steel components in PRTR, use of buffered oxalate-peroxide solution appeared preferable.

The choice of a buffered solution of oxalate and peroxide was checked out in the IRP facility. A test was completed using the following treatments: (a) oxalate-peroxide solution buffered with peracetic acid, (b) oxalate-peroxide solution buffered with gluconate, (c) a dilute (3%) nitric acid, (d) 10% nitric acid-peroxide, (e) a film-conditioning step with alkaline permanganate, and (f) a film removal step with an inhibited oxalic acid based proprietary compound. The buffered oxalate-peroxide solution was very effective, removing about 75% of the total activity. The reaction seems rapid, all occurring in the first ten minutes at a fairly low temperature (30-60 C). On heating, the rate did not increase. After the initial reaction, no further reaction occurred. The second oxalate-peroxide solution further decreased the readings from 5.0 R to 3.5 R. Some activity was deposited on the piping surfaces; after the two initial treatments, (a) and (b) above, the activity of the test section had increased from essentially background to 280 mr. Some of this activity was removed by treatment with the dilute nitric acid; the remainder was effectively removed by treatments with the alkaline permanganate and oxalic acid. The final residual activity was 1.5 R/hr on the filter and 5-25 mr/hr on the piping and test section.

The oxalate-peroxide solution has little effect on the high temperature films. It will remove some of the activity adsorbed on the film surface, but in order to do an effective decontamination, it will be necessary to follow the oxalate-peroxide mixture by the film stripping procedure (alkaline permanganate followed by some inhibited acid). Some PRTR jumper sections were decontaminated in the IRP. One-hour treatment with the buffered oxalate-peroxide solution reduced the activity by 50% (DF = 2). Two additional cycles, each consisting of treatment with alkaline permanganate followed by oxalic acid (proprietary) gave an over-all DF ranging from 6 to 15. Another set of jumpers was decontaminated using only the two cycles of alkaline permanganate and proprietary oxalic acid solution, but increasing the exposure to the acid step from one to two hours. This procedure was very effective in removing the film and adsorbed contamination, giving decontamination factors ranging from 10 to 40. The highest decontamination factors were obtained on the upstream piece where the turbulence was greatest, indicating some velocity effect.

Corrosion Effects of Decontamination. PRTR Operations made a complete list of all materials present in the PRTR. This list was checked to determine if any special corrosion problems would occur. Special precautions were specified for certain parts of the system. It was pointed out that the alkaline permanganate would tend to pit the Stellite valve seats and would be extremely corrosive to chromium plated parts.

The corrosion testing demonstrated that the buffered oxalate-peroxide solutions at 80 C were relatively non-corrosive to Inconel, stainless steel, Zircaloy-2, and carbon steel, the rate of uniform attack being less than 1×10^{-5} in/hr. Accelerated attack ($\sim 4 \times 10^{-5}$ in/hr) was noted on carbon steel at the vapor-liquid interface. Pitting did not seem to be a problem.

Carbonic acid solutions at 80 C and 100 psi were relatively non-corrosive to brass, Admiralty metal, and 400 series stainless steel. The attack on carbon steel averaged $\sim$ 0.2 mil/hour.

Fretting Corrosion. A test was performed in TF-2 at 300 C and a pH of 10.0 using LiOH to determine the benefits of spring-loading supports to prevent impact corrosion. A Zr-2 specimen was positioned on a vertically suspended bar and three flat 302 S/S springs were mounted tangentially 120° apart on the specimen. The springs were fixed at one end to provide a turbine-like design. This arrangement was positioned inside a Zircaloy-2 tube by manually compressing the springs, the tendency of the flat springs to unwind kept the Zircaloy-2 specimen from contacting the tube. After 263 hours, an examination revealed 9-10 mils deep fretting corrosion on the tube with little or no attack on the springs. A second test has been initiated at the same operating conditions using flat springs which are fixed at each end to form a circumferential bumper-type support.

Corrosion of Various Stainless Steel and Nickel Base Alloys. Corrosion results are presently being obtained on various stainless steel and nickel base alloys. Following 42 days of exposure in 550 C - 3000 psi deoxygenated water, the nickel and iron base alloys may be grouped in the following order of increasing corrosion:

Group 1. (5 to 9 mg/dm^2) Hastelloy "X", 446 stainless steel, Inconel "X", and Incoloy.

Group 2. (26 to 29 mg/dm^2) PDRL-102, and an experimental 24% Cr-5% Al-Fe alloy.

Group 3. (40 to 60 mg/dm^2) Hastelloy "N", 17-4 PH stainless steel, and an experimental 25% Cr-3% Al-0.5% Y-Fe alloy.

Group 4. (150 to 210 mg/dm^2) 316 stainless steel, 304 L stainless steel, 430 stainless steel, and two heats of 406 stainless steel (one from Carpenter and one from Allegheny Ludlum).

A comparison of these corrosion results with data previously obtained in oxygenated water (3-4 ppm) suggests oxygen may affect some corrosion rates. Oxygen reduced corrosion rates for the Fe-Cr-Al alloys, 430 stainless steel, and 446 stainless steel. In contrast, 304 L stainless steel and all of the nickel base alloys gained less weight in the deoxygenated system.

Pre-irradiated Zircaloy Tube Evaluation. Under the biaxial stress conditions of stress rupture tests, the annealed section of PRTR pressure tubes continues to exhibit creep characteristics superior to those of annealed Zircaloy-2 strip tested under uniaxial stress. To produce equal strain, the hoop stress on the tube sections must be about twice as large as the stress on flat strip specimens. The stress rupture data, summarized in the following table, also show that at 550 F the strain rate is independent of stress, except for one test, over the range of 35,000 to 40,000 psi hoop stress.

Stress Strain Data for Annealed PRTR Tube Sections

Hoop Stress	Percent Ultimate Strength	Temp (F)	Hours	Strain (in/in)	Strain Rate (in/in-hr)	Stress on Flat Strip to Produce Equal Strain
35,200	76	550	471	0.93	1×10^{-6}	15,000
36,200	78	550	2500(1)	1.29	1×10^{-6}	16,000
37,500	81	550	970	3.23(2)	3×10^{-6}	20,500
39,600	85	550	5590	3.77	1×10^{-6}	21,500
40,900	88	550	770	Burst	1.8×10^{-5}	23,000
36,000	88	650	2359	Burst	1×10^{-4}	

(1) Test terminated at 2500 hours.
(2) Test stopped and specimen burst tested.

A burst test was run on a sample after 1000 hours of testing under biaxial stress during which the sample had accumulated 3.23% strain. The ultimate hoop stress was 47,100 psi which is comparable to that for publication. This work will also be presented at the American Ceramic Society meeting in Seattle, October 17-19, 1962.

Plutonium Carbide Research

The series of plutonium carbide samples varying from 12 to 52 a/o C have now been given three different heat treatments:

1. A slow cool, 100 C/hour, from 600 C after holding there 124 hours.

2. A 168-hour treatment at 600 C followed by a 7.8 C/hour cool.

3. A quench from 600 C into liquid nitrogen.

The two samples of greatest interest (39.9 and 41.0 a/o C) were given two more heat treatments. They were held at 400 C for 10 days. In the first case they were quenched from this temperature into liquid nitrogen. In the second case they were cooled at 4.4 C/hour. These heat treatments have served to illustrate the extreme sensitivity of the zeta phase to heat treatment. In heat treatments "1" and "2" the zeta phase appeared both on x-ray and metallographic analysis in all samples below 41 a/o C. There was no evidence, though, of zeta in the 41 a/o C sample. Likewise, no zeta appeared in the alloys of higher carbon content.

After quenching from 600 C, there were no signs of zeta in any of the specimens. The x-ray lines were all sharp and none of the blade-like zeta structure was found in the micrographs. The lattice parameter of the PuC was constant from 12-14 a/o C at 4.955A. At this point it rose linearly to a maximum of 4.976A at 49 a/o C.

The 39.9 and 41.1 a/o C samples were held at 400 C for ten days. In one case they were quenched and in the other they were slowly cooled. While the slowly cooled samples have not been analyzed yet, the quenched ones definitely show signs of zeta on x-ray analysis. From this we have concluded that the sluggish formation of zeta below 40 a/o C is even slower above this composition. The reason for this is probably tied up with the mechanism of zeta growth. A possible explanation for this large difference in reaction kinetics may be that the 41 a/o C alloy is being slowly cooled from a single phase (PuC) region; the uncertainties in the diagram and equilibrating temperature certainly do not rule this out. If this is the case, the PuC would first have to transform to PuC plus epsilon plutonium and then to PuC plus zeta. The kinetics for this two-step reaction could easily be such as to inhibit the formation of zeta even on the slowest cooling.

The first dilatometric run on a sample of 11.7 a/o C showed all of the usual plutonium transformations. There were two strong anomalies noted which must be attributed to the carbide. The first of these occurred around 380 C on both the first and second runs. This was a sudden expansion in the middle of the delta plutonium contraction. The second break was around 515 C and represented an increase in the regular expansion of epsilon plutonium on heating. While the high temperature break was still apparent on cooling, the 380 C anomaly is suppressed. The sample has been run to 600 and back to room temperature twice. The second run still shows the 380 C break, but it is weaker than it was on the first run. The 515 C break is still strong. More cycles will be run to try to learn more about these transformations.

PuO_2-Carbon Reactions

A sample of PuO containing a small amount of beta-Pu_2O_3 and 0.083 weight percent carbon was prepared by reaction of PuO_2 and carbon at 1800 C in helium. The lattice constant of the PuO was observed to be 4.961 ± 0.001 A.

Plutonium Nitride

Melting studies were done in atmospheres of gettered argon, helium and nitrogen in a tungsten ribbon furnace. Temperature measurements were made with a brightness pyrometer corrected for emissivity and absorption in the furnace sight glass and lucite hood panel. Under one atmosphere of helium or argon the "apparent" melting point of plutonium nitride was 2600 ± 75 C. The term "apparent" is used because melts were not distinct but looked more like a decomposition or sublimation. This phenomenon was not observed under one atmosphere of nitrogen; melts were sharper, and there was no evidence of a decomposition or vaporization. A melting point value of 2750 ± 75 C was determined. X-ray diffraction analyses of the nitrogen melted specimens show a constant lattice parameter, thus indicating no constitutional changes.

Plutonium Sulfides

The melting point of Pu_2S_3 was found to be 1725 ± 5 C in vacuo on a tungsten filament.

3. UO_2 FUELS RESEARCH

Fission Fragment Migration

Final analyses of UO_2 samples removed from the cross sectional surfaces of three irradiated UO_2 fuel capsules confirmed previous observations of radially non-uniform fission fragment and plutonium distribution.

Capsules with high exposures (~ 14,000 MWD/T) show greater migration of cesium to the cooler, outer regions of the fuel capsules, while other fission fragments (Zr-Nb, Ce, Ru) and plutonium exhibited a more uniform distribution. The microstructure of the fuel suggests a continued slow change, after the initial formation of columnar grains, that would be expected to further modify the fission fragment distribution.

Thermal Conductivity of UO_2

Second cycle measurements of thermal conductivity of a large UO_2 single crystal were begun following a second irradiation to 10^{15} nvt. The measured values (to 780 C) appear inconsistent with data obtained after the first irradiation; conductivity is higher than previously observed and slope of the temperature dependent curve appears reversed. Experimental and instrumental difficulties may be responsible for the apparent changes. Measurements are to be repeated following instrument modifications.

Electron Microscopy of UO_2

Two papers discussing CFDO electron microscopy studies were presented at the Fifth International Congress for Electron Microscopy. They were:

1. Daniel, J. L., "Refelction Microscopy of Irradiated UO_2 Crystals."

2. McPartland, J. O., "A High Temperature Stage for Transmission Electron Microscopy."

A concurrently displayed photomosaic "Reflection Microscopy of Irradiated UO_2 Crystals," was selected for inclusion in the 1962-63 traveling exhibit of the Electron Microscopy Society of America.

Ultramicrotomy of UO_2 crystals appears to be feasible under properly controlled experimental conditions. This conclusion, reached following initial experiments by A. Persson of the LKB Company (Microtome Manufacturers), is supported by results of scattered experiments at Hanford and elsewhere. Careful adjustment of relative positions of crystal and cutting knife is essential to take advantage of the natural cleavage plane (111) of the UO_2. Investigation of the technique is continuing.

Modifications were made in the electron gun bias circuit of the JEM microscope to provide a continuously variable grid bias voltage to provide wider latitude and better illumination control for both reflection and transmission microscopy. The modified system is being tested for stability and brightness control.

Surface Tension of $UO_2(l)$

Surface tension of $UO_2(l)$ was calculated from drop profiles recorded on high speed motion pictures of explosive melting of UO_2. Analyses of three pendant drop profiles by two independent methods gave values between 340 and 760 dyne cm^{-1} with a mean of 490 dyne cm^{-1} (roughly the same as mercury or molten lead). The temperature of the liquid was assumed to be constant at 2800 C, the melting point of UO_2.

Explosive Melting of UO_2 and ThO_2

Direct resistance heating of UO_2 and ThO_2 was used as a means of generating high temperature fuel states typical of what might be expected in the core of high rated fuel elements, or during severe reactor power transients. A further goal is development of techniques for resistance heating semi-conductors for examination at high temperature in oxidizing atmospheres without the presence of foreign materials. Short, high current pulses were used to cycle samples from 1000 C to above the melting temperatures, to below 1000 C in less than three seconds. Heating time was controlled by varying the voltage drop across the specimen. High speed motion pictures (to 16,000 pictures per second) of the melting sequences that were made to study the melting cycle also provided data of more fundamental nature, such as the surface tension of $UO_2(l)$.

UO_2-ThO_2 Equilibrium Studies

The liquidus for the UO_2-ThO_2 system was determined. Principal features are total miscibility and a melting minimum of 2760 C at 2 w/o ThO_2. Solid solutions with compositions between 1.8 w/o and 54 w/o ThO_2 were prepared by Chemical Research Operation by electrolysis from fused salts. Melting temperatures of samples containing 1.8, 1.9, 2.4, 10, 14, 45 and 54 w/o ThO_2 were measured. A UO_2 - 7.5 w/o ThO_2 solution was obtained by in-reactor homogenization of a heterogeneously mixed fuel core. Samples with 95 w/o ThO_2 were prepared at ORNL by the Sol-Gel process. All melting points were measured under one atmosphere argon in a tungsten filament furnace.

Gross fractionation of uranium to the vapor phase was observed from each specimen studied, reflecting the high vapor pressure of UO_2 relative to that of ThO_2. X-ray fluorescence analyses of the fused residue and the sublimate from specimens with initial compositions of UO_2 - 2 w/o ThO_2, UO_2 - 12 w/o ThO_2, and UO_2 - 41 w/o ThO_2 showed residue compositions of UO_2 - 4.3 w/o ThO_2, UO_2 - 28 w/o ThO_2, and UO_2 - 51 w/o ThO_2, respectively, while the sublimates were reduced in thorium content to UO_2 - 0 w/o ThO_2, UO_2 - 8 w/o ThO_2, and UO_2 - 17 w/o ThO_2, respectively. The continuously changing composition introduces uncertainty into phase equilibrium studies. These results suggest a possible approach to U-Th separation and are valuable in interpreting the structures of previously irradiated UO_2-ThO_2 mixtures.

Plastic Deformation of UO_2

A single crystal of UO_2 was ductile under torsional stress at 2100 C. The crystal, 1/8-inch thick by 1/4-inch wide by 1-inch long, was twisted 120 degrees at approximately 10 degrees/second. It was deformed plastically much like the polycrystalline UO_2 tested previously.

The O/U ratio of polycrystalline UO_2 remained constant at 2.001 ± 0.0002 during torsional tests at 2300 C. The microstructures of deformed specimens showed some separation at the grain boundaries and little grain growth.

High Rate Densification Studies

Three-pound batches of minus 20 mesh electrodeposited $UO_{2.015}$(10.77 g/cc particle density) were consolidated by high rate densification at 1200 C and 325,000 psi impact pressure to yield a solid having density of 10.84 g/cc and an O/U ratio of 2.006.

Sintered UO_2 scrap was converted to a dense solid (10.80 g/cc) by two separate high rate densification techniques: (1) crush, ball mill for 48 hours, compact by high rate densification at 1200 C and 380,000 psi; (2) crush to minus 20 mesh, roast for 16 hours at 140 C, compact by high rate densification at 900 C and 200,000 psi, sinter for 12 hours at 1700 C.

A study of the effects of temperature, heating rate and soaking time was made on HRD UO_2 having a density of 10.17 g/cc. Increased densities were achieved by reducing the heating rates from 225 C to 75 C, by increasing temperatures from 1400 to 1750 C, and by increasing the soaking time from one to twelve hours. Density increased more rapidly during the first few hours of soaking time. A suitable sintering program for further densifying this type of HRD UO_2 is a heating rate of 100 C/hour to 1750 C followed by four hours holding at that temperature.

A 50 w/o tungsten - UO_2 mixture was compacted to a density of 13.44 g/cc (96 percent TD) by high rate densification. Compaction of blended and evacuated, -65 mesh fused UO_2 and -325 mesh tungsten powder at 1100 C and 250,000 psi impact pressure produced a cermet having a continuous tungsten phase.

Graphite Welding

Ceramographic examination of a graphite "fuel element end closure" completed by the resistance welding techniques employing the magnetic force butt welder revealed a weld of surprisingly high quality. The graphite in the weld zone had recrystallized into a material having finer grain size and higher density than the base material. A solid graphite cap was welded into the end of a 1/2-inch diameter by 1/32-inch wall thickness graphite tube. No intermediate material was used at the weld interface. An initial force of 250 pounds is applied between the parts and a 1/60 second welding current pulse of 50,000 amperes is employed to produce the weld. The high current density literally explodes the graphite at the weld interface to produce a temporary gap between the parts. An arc exists for 8 to 12 milliseconds during which the parts are rapidly brought together again to complete the weld. The described method produced the first known successful weld between two pieces of graphite.

4. BASIC SWELLING PROGRAM

Irradiation Program

A general swelling capsule is still under irradiation at a control temperature of 575 C. Another capsule that was successfully operated at 575 C was discharged after reaching its goal exposure of 0.15 a/o B.U. Two capsules previously discharged (#13 and #14) will be shipped to Radiometallurgy for disassembly. Assembly of one additional capsule is approximately seventy percent complete. This capsule contains two 1/16-inch OD heating elements in lieu of the 1/8-inch OD heating element that is normally required for this capsule design.

An ETR, MTR prototype swelling capsule previously used to determine the effects of thermal cycling on a metallographically polished specimen was reassembled using thicker heat transfer fins. The specimen was replaced with two halves of an axially split, hollow cylinder of high purity uranium. One of the samples had been beta heat treated and the other sample was in the as-extruded condition. The samples were subjected to a thermal treatment that simulated the in-reactor thermal history of general swelling capsule #11. A comparison of the irradiated and non-irradiated samples will be made in an attempt to determine the contribution of thermal cycling only to the appearance of the irradiated specimens. Concurrent with these tests, heat transfer data are being accumulated as a guide for the operation of the MTR capsule during irradiation. It is significant that identical internal capsule temperatures were obtained up to ~ 300 C with the same electrical power input to the capsule regardless of whether the heat transfer fins are 0.041-inch thick or 0.100-inch thick. Above ~ 300 C the same power input results in higher temperatures in the case of the thinner fins. At a power input of 1000 watts, the internal capsule temperature is approximately 120 C higher with the thinner fins. Apparently the contact area between the 0.100-inch fins and the capsule internal walls is increased and the contact resistance decreased as internal capsule chamber expands with increased temperature. Additional tests with this capsule are planned to determine the maximum power output of the 1/16-inch heater.

Post-Irradiation Examination

The specimens from capsule #12 (625 C, 0.16 a/o B.U.) have been polished, etched and examined with light and electron microscopy. The nature of the samples is such that they are extremely difficult to process and the microstructures are quite difficult to analyze. In the case of the as-extruded specimen irradiated adjacent to the control thermocouple sufficient grain structure was developed to have confidence in the appearance. As viewed in the light microscope this sample consists primarily of large grains with cracks outlining many of the boundaries. Superimposed upon this large grained structure is a second, much smaller grain boundary network that is identical in size with the original, preirradiation grain size. At higher magnification in the light microscope this smaller network is seen to consist of rows of pores. In the electron microscope these pores are seen to be from 0.2 to 1 μ in diameter. The etching characteristics of the surrounding matrix verify that these rows of pores bear no relationship to the presently existing grain structure. There is a zone on either side of these rows that is depleted in porosity. Beyond the zone are numerous pores < 0.1 μ in diameter. It is concluded that inert gas collected into pores at the grain boundaries during the initial stages of the irradiation at 625 C. The large grains now existing in the sample were brought about by a one-minute temperature excursion into the beta that occurred at about 0.1 a/o B.U. It is not clear when the cracks formed at the boundaries of the large grains, but they could have formed either during specimen handling or during cooling of the sample. It is significant that no depletion in the density of < 0.1 μ diameter pores was observed in the vicinity of the cracks. Very large holes are also present in this sample, but it is not clearly established whether these are pores or tears associated with cracking.

Two other specimens from capsule #12 contain cracks and large holes, some of which appear to be tears and others pores. In the electron microscope the samples appear extremely porous with pores varying from 0.1 to 10 μ in diameter. Of these two, the as-extruded specimen contains a few patches of very fine pores (~0.1 μ) surrounded by larger (0.5 μ) pores. These large pores may be outlining original grain boundaries but the integrity of the preparation was not sufficient to say for sure. The irradiation temperature of this sample was about 590 C due to the thermal gradient along the axis of the capsule. The third sample which was beta treated prior to irradiation (and, hence, large grained originally) operated at about 610 C. No hint of grain boundary segregation of porosity was observed but the general porosity seemed larger.

The samples from capsule #11 are being processed for metallography. Density values will also be determined for all specimens. The swelling and shape instability was much greater than was anticipated. This may be due to the geometry of the samples (1/2-inch OD x 0.030-inch wall hollow cylinder) or to the extreme high purity of the material. Additional studies are necessary to evaluate the relative importance of each variable.

Quantitative Metallography

As reported previously, attempts at unfolding pore size distribution information have been made. One approach, namely, that of determining the true diameter of spherical pores revealed on shadowed negative replicas, has been pursued. A replica of etched irradiated uranium containing pores was sprayed with polystyrene spheres of known size. This replica was then shadowed at 15 degrees incidence and processed for electron microscopy. Since measurements of shadow length and maximum pore diameter as revealed by the replica can be made, and since the angle of shadowing is known, it is possible from simple geometric considerations to establish which pores were cut above or below their centers by the specimen surface and the distance between pore center and the specimen surface. If, however, the specimen surface in the as-etched condition covers a considerable angular range, then the shadow lengths will vary. That this is actually the situation was established from measurements of shadows cast on the replica surface by the polystyrene spheres of constant size. The shadows of 240 spheres were carefully measured and a frequency plot versus apparent shadow angle was constructed. This plot shows a range of apparent angles of shadowing from 12 to 32 degrees associated with the reference spheres. This range of angles is due to the etching characteristics of the specimen. Consequently, a protrusion on the negative replica, which represents a pore on the specimen surface, cannot be accurately measured as to depth (shadow length) and maximum width. This would be possible, however, if all replicas were shadowed from two directions to determine the inclination of the reference surface. Such a technique, however, would lead to considerable masking of structural detail due to the additional shadows and does not appear to be too practicable. Another approach to obtaining three-dimensional distribution information from the areal distribution estimates involves assumptions as to the theoretical distribution which actually exists and subsequent statistical and mathematical transformation to yield a distribution which can be compared with the areal distribution data. This approach is being studied by Operations Research personnel.

A paper entitled "Swelling in Uranium - Quantitative Metallography" was presented at the Fifth International Congress for Electron Microscopy. The proceedings have been published by Academic Press.

5. IRRADIATION DAMAGE TO REACTOR METALS

Alloy Selection

Allegheny Ludlum Steel Corporation's R-27 alloy has mechanical properties which indicate that it may be useful at temperature to 1800 F. A cursory examination of the effects of various oxidizing environments and irradiation at high temperature is now in progress. Oxidation tests to 2000 F are in progress with tests at 1700 F and 1800 F completed. The effects of various heat treatments on the tensile properties of the material are also being studied.

Three additional alloys -- Hastelloy N, Haynes R-41 and Haynes R-235 -- are also being examined. Sheets 0.125-inch in thickness have been rolled and corrosion, oxidation and tensile specimens are being fabricated.

Hanford Laboratories are responsible for the procurement, storage and disbursement of structural materials to be used in the Irradiation Effects on Reactor Structural Materials Program, where use of such material is by more than one site in the coordinated program. Three alloys were procured this past month from United States Steel Corp. These alloys -- A212B, A302B and HY80 -- which are in plate form, are part of specially documented heats supplied by USS Research Laboratories for nuclear testing programs. They will be used to correlate data obtained from additional similar materials to be used in the present research program.

In-Reactor Measurement of Mechanical Properties

Analysis of in-reactor creep behavior and ex-reactor creep behavior of 20 percent cold worked Zircaloy-2 has led to a consistent theory for the in-reactor creep of Zircaloy-2. The theory is based on a number of mechanistic interpretations of experimental results. These are as follows:

1. The creep of Zircaloy-2 at 30,000 psi and temperatures including 250 C and above is controlled by the climb of dislocations over barriers of various types.

2. During irradiation, creep rates are less at 310 C and 250 C than in similar ex-reactor tests. The low creep rates during irradiation can be interpreted as being the result of an increase in the number of obstacles over which dislocations must climb. The new obstacles are produced by neutron bombardment.

3. During reactor outages creep rates increase with time to a value which is greater than expected in the early stages of similar ex-reactor tests. The increase in rates during outages is attributed to the removal of the obstacles through annealing. The fact that the rates attained after annealing are greater than ex-reactor rates is a very important result. This can only mean that the cold worked substructure loses much of its creep resistance during irradiation.

4. The number of irradiation produced climb obstacles under steady state conditions can be formulated in the following manner:

$$P\ (\phi) - K\ (Ni) = 0 \qquad (1)$$

where $P\ (\phi)$ is the rate of obstacle production and K is the reaction constant for obstacle removal through annealing. (Ni) is the number of obstacles per unit volume. K can be written as

$$k\ e^{\frac{-\Delta H_R}{RT}}$$

where ΔH_R is the activation energy for obstacle annealing and k is another reaction constant.

Solving (1) for N_i

$$N_i = \frac{P\ (\phi)}{k}\ e^{\frac{\Delta H_R}{RT}}$$

The creep rate ex-reactor can be expressed as

$$\dot{\epsilon} = A \frac{1}{N^{2/3}{}_h}\ e^{\frac{-\Delta E_C}{RT}}$$

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where A is nearly constant and accounts for the substructure of the material.

h is the height of the obstacle.

ΔH_C is the activation energy for dislocation climb.

In-reactor in the presence of annealing and production of obstacles the steady state creep rate can be written

$$\dot{\epsilon}\, i = A_i \left(\frac{1}{N_i^{2/3} h_i + N^{2/3}\, h}\right) \exp -\Delta H_C/RT$$

Since creep rates during irradiation are much lower than in the absence of radiation,

$$N_i^{2/3}\, h_i \gg N^{2/3}\, h$$

In this case the in-reactor rate can be written

$$\dot{\epsilon} i = A_i \left(\frac{k}{P\ (\phi)}\right)^{2/3} \exp - \left(\frac{\Delta H_C + 2/3 \ \ \Delta H_R}{RT}\right)$$

The foregoing reasoning has been verified by in-reactor activation energy measurements. The measurements were made using the temperature cycle technique in which activation energies are calculated from the creep rates just before and just after an abrupt change in temperature.

In the damage annealing range where dislocation climb controls creep, the activation energy calculated by the temperature cycle technique will depend upon the speed with which the steady state number of obstacles is established; that is, at the lower end of the annealing range where annealing rates are slow the obstacle concentration will be approximately the same before and after the temperature change. The observed activation energy in this case will be ΔH_C. At higher temperatures, annealing rates are fast and the equilibrium number of obstacles is obtained during the temperature change and the activation energy will be $\Delta H_C + 2/3\ \Delta H_R$.

Direct activation energy measurements in-reactor have shwon that below 325 C the activation energy is about 60,000 cal/mole. This is in excellent agreement with the ex-reactor

ΔH_C which is about 58,500 cal/mole. Above 350 C an activation energy of 86,000 cal/mole is observed. The 86,000 cal/mole value is $\Delta H_C + 2/3 \Delta H_R$. Between 325 C and 350 C the activation energy changes from 60,000 cal/mole to 86,000 cal/mole.

The low creep rates in-reactor are believed to be the result of irradiation produced obstacles to dislocation climb. In the temperature range where the in-reactor creep properties of Zircaloy-2 are being studied the irradiation produced obstacles can anneal. The steady state obstacle concentration is established when the rate of obstacle production equals that of obstacle removal. The increase in rate during reactor outages is the result of the decrease in the number of obstacles through annealing. Activation energy studies have also shown that in-reactor creep rates are controlled by climb of dislocations over obstacles and that the concentration of obstacles is an exponential function of temperature. In addition, the original creep resistance of Zircaloy-2 is markedly reduced by irradiation.

In view of the temperature dependence of the obstacle concentration and the loss of the pre-irradiation creep resistance, it must be concluded that in-reactor creep at higher temperatures would be greater than ex-reactor. In this temperature range the original creep resistance of the material will be decreased by irradiation and the creep resistance provided by irradiation produced obstacles will not be as great as the pre-irradiation creep resistance.

During the month an in-reactor creep test at 350 C and 30,000 psi was started. The test has been running 320 hours during which time one reactor outage has occurred. The creep rate in-reactor before the first outage was generally higher than in a similar ex-reactor test. A quantitative description of the in-reactor and ex-reactor rates cannot be given as both tests were in first stage creep where rates change rapidly. During the outage the in-reactor creep rate increased from the rate existing just before the outage to about 5.5 times that of the ex-reactor test. During startup the rate decreased to a steady state rate of 3.9 times the ex-reactor rate. The total plastic strain in-reactor was about 1.86 percent while the in-reactor strain was 0.76 percent. The 350 C - 30,000 psi in-reactor creep data tends to confirm the existence of the processes postulated above.

A contract is now being negotiated for the procurement of additional creep capsules. The purchase specifications call for ten capsules identical to those now being used in the program and unassembled parts for an additional ten capsules that can be tailored to particular environmental tests. The assembled capsules can be operated between temperature limits of 200 to 400 C, with stresses between 0 and 80,000 psi and strain recorded to one-half inch. The unassembled capsules can be altered to extend the above limits as a particular test or environmental condition may dictate.

Irradiation Effects in Structural Materials

The purpose of this program is to investigate the combined effects of radiation and reactor environment on the mechanical properties of structural materials. Special attention will be given to the determination of mechanical property changes produced in metals by irradiation at elevated temperatures.

During the month 72 irradiated bend test specimens of Zircaloy-2 were transferred to the 326 Building for testing. These specimens represent the 0, 10, 20, and 40 percent levels of cold work and the longitudinal and transverse directions with respect to rolling. Irradiations were conducted in the ETR, G-7, hot water loop (controlled at 540 F) to estimated fast neutron exposures ranging from 0.6 to 2×10^{20} nvt. The bend test specimens are also suitable for weight gain and hardness measurements, which are being accomplished prior to bend-to-fracture testing.

Prior to irradiation, the specimens were autoclaved at 300 C for 48 hours. This treatment develops a black, coherent oxide film resulting in an average weight gain per specimen of 7 mg/dm^2. After irradiation, weight gains were measured on three groups of specimens, each group containing 24 specimens representing all of the cold work and directional conditions given above. After irradiation, all of the specimens exhibited the same surface appearance as before irradiation. The first group was irradiated in a flux field of 6.9×10^{13} nv to an estimated total exposure of 6×10^{19} nvt (>1 Mev). The effective exposure time at 540 F was 19.2 days. The average weight gain for this group was 38.7 mg/dm^2. The second group was irradiated in a flux field of 1.2×10^{14} nv to an estimated total exposure of 2×10^{20} nvt. This group was irradiated concurrent with group one, and also had 19.2 days effective exposure at 540 F. The average weight gain for this group was 43.0 mg/dm^2. The third group was irradiated in a flux field of 6.9×10^{13} nv (same as group one) to an estimated total exposure of 1.2×10^{20} nvt. The effective exposure time at 540 F was 34.2 days. The average weight gain for this group was 52.4 mg/dm^2. The weight gain variation within each group of 24 specimens was within 4 mg/dm^2, and the over-all values were independent of cold work. In order to assess the contribution of neutron irradiation on weight gain, identically prepared specimens having the same autoclave treatment were exposed to hot water in an ex-reactor loop. The temperature history of this loop was programmed to duplicate that of the in-reactor loop, and water quality was duplicated as closely as possible. Consequently, the net difference in weight gains between the in-reactor and ex-reactor loop specimens should reflect flux effects only. The average weight gain on ex-reactor control specimens for groups one and two (19.2 effective days at 540 F) was 3.5 mg/dm^2. Continued exposure to 34.2 effective days (group three) would result in a nominal additional weight gain. The comparison of 38.7 mg/dm^2 in-reactor to 3.5 mg/dm^2 ex-reactor for group one exhibits a marked flux effect on corrosion rate. Under ex-reactor conditions such a high weight gain would indicate break-away corrosion. A flux effect is also apparent in comparing group one, 38.7 mg/dm^2 at 6.9×10^{13} nv, with group two, 43.0 mg/dm^2 at 1.2×10^{14} nv; both groups exposed under identical time-temperature conditions. The difference due to flux rate is greater than the total ex-reactor weight gain under duplicate time-temperature conditions. The high additional weight-gain for group three specimens over group two under the same flux rate but longer exposure time also confirms a high flux-induced corrosion rate.

Hardness measurements were made on the bend test specimens before and after irradiation using the 30 T scale of a Kentrall superficial hardness tester. In the unirradiated condition, hardness values varied from 77.3 to 83.2 over the range 0 to 40 percent cold work, corresponding to 92.4 to 99 on the Rockwell B scale. After a neutron exposure to 6×10^{19} nvt, the annealed specimens increased in hardness from 77.3 to 81.1, whereas the cold worked specimens did not change significantly in hardness. Specimens from groups two and three, irradiated to about 2×10^{20} and 1.2×10^{20} nvt, respectively, had comparable hardness values at each cold work level. The net change in hardness over the 6×10^{19} nvt exposure was about one point for each cold work level. Thus, no significant hardening occurred for the cold worked specimens after the lower exposure (6×10^{19} nvt), but a significant change of about the same amount for the 10, 20, and 40 percent cold work levels occurred after the higher exposures (1.2-2.0×10^{20} nvt).

Voids have been observed in the necked region of Zircaloy-2 tensile specimens tested at room temperature. The volume fraction of these voids as a function of distance away from the fracture surface was measured by quantitative metallography during the month. Furthermore, additional specimens tested at various strain rates and cold work levels have been examined metallographically. These studies reveal the following: (1) voids are not nucleated at the maximum load point (where strain is no longer uniform but unstable) as supposed but are nucleated after moderate necking has already occurred, (2) voids are aligned axially in the longitudinal specimens but exhibit no alignment in the transverse specimens indicating an influence due to mechanical anisotropy, (3) the maximum volume fraction of voids occurs at the fracture trace and equals about 1.5 percent, (4) the extent of void formation appears to be less at higher strain rates, higher cold work levels, in the transverse compared to longitudinal directions, and for fibrous compared to shear fractures, and (5) no association between voids and grain boundaries has been detected as yet. Since the hydrogen content of these specimens is less than 10 ppm, little or no influence from hydrides would be expected.

Damage Mechanisms

The objective of this program is to establish the nature of the interaction between defects present prior to irradiation and those produced by irradiation, and to investigate the possibility of neutralizing the effects of impurity atoms by chemical stabilization. Alloys of high purity iron with small amounts of carbon and nitrogen and alloys to which a chemical stabilizer such as titanium has been added will be studied.

A preliminary investigation of impurity effects in irradiated iron is continuing. Armco ingot iron, henceforth referred to as iron A, was used as the starting material. This material was decarburized and denitrogenized in a wet hydrogen atmosphere, designated iron B, and then treated in an ammonia-hydrogen atmosphere calculated to add 100 ppm nitrogen to the sample, designated iron C. Tensile testing of irons B and C irradiated to fast neutron exposures of 1×10^{17}, 5×10^{17}, 1×10^{19}, 2×10^{19}, and 5×10^{18} nvt has been completed. Examination of these results and those on iron A irradiated under identical conditions show the following trends:

1. The ultimate strength and fracture strength increase uniformly with dose while the ductility uniformly decreases with dose.

2. Unirradiated iron A showed a large drop in load yield point, iron B showed none, and addition of nitrogen caused a small one to reappear in iron C. This yield point, however, disappeared at a dose of 1×10^{17} nvt and did not reappear at the highest exposures studied.

3. The proportional limit, maximum load and fracture strain in irons B and C were approximately the same for all doses and were markedly lower than in iron A.

The results indicate that the levels of carbon and nitrogen content as well as neutron irradiation strongly influence the mechanical properties of iron. The residual impurities, both metallic and non-metallic, remaining in the iron after the hydrogen treatment are believed to have masked some of the effects of the nitrogen additions. For this reason samples of electron beam zone refined iron are being prepared from Ferovac E iron for use in perfecting the fabrication, carbon and nitrogen addition, and the various physical measurement techniques necessary before proceeding to experiment on the high purity material purchased from NRC and BMI.

The resistivity apparatus has been redesigned to permit measurement on 20-mil wire samples at helium temperatures. Measurements on a wire sample such as this will provide a larger e.m.f. for measurement as well as a more easily regulated current requirement. Measurements at 4.2 degrees K will eliminate the resistivity from phonon scattering and measure only that due to imperfections in the lattice. Thus the change in resistivity with irradiation should be several hundred percent, rather than the few percent found in room temperature measurements.

Neutron Spectra

The necessity for more detailed knowledge of the neutron spectra over a wide range of neutron energies became apparent from calculations which indicated one must consider neutrons of energies down to 50 kev in order to account for 95 percent of the damage in structural materials. The multigroup transport theory, (S_X) code has therefore been expanded from 19 groups to 33 groups. Previous fine-spectra structure was calculated only in the energy range 0.18 to 10 Mev. Group-average cross sections have been developed for a number of materials to enable calculation of neutron spectra expected in irradiation facilities to be used in the Irradiation Damage to Reactor Metals Program.

6. GAS GRAPHITE STUDIES

PRTR Gas Loop Samples

The design of the sample container and samples for the PRTR Gas Loop was completed. The samples are contained within three 3/32-inch tie rods attached to end rings of 406 stainless steel. Vibration of the samples is prevented by a Hastelloy-X spring assembly. Details of the sample container are given in drawing H-3-13689. Four types of graphite samples are loaded into each sample container: (1) solid cylinders,

(2) concentric cylinders, (3) nested tubular samples, (4) cluster of seven 0.42-inch samples. Sample details are shown in drawing H-3-14572. The oxidation of the steel sample-container material by CO_2 is being measured by induction heating and continuous weighing of specimens at various temperatures. The temperature distribution and weighing sensitivity were improved by the addition of a platinum susceptor.

Graphite Oxidation Studies

The rate of oxidation of graphite usually decreases with time during the initial phase of oxidation and then becomes approximately constant. British workers have reported that approximately one percent oxidation is needed before constant weight-loss rates are obtained. Oxidation rates on graphite usually refer to the constant region.

Because oxidation rates in the PRTR Gas Loop experiments are expected to be low, it may be desirable to pre-oxidize the samples. Studies have begun to determine whether any pretreatment will be required.

Three tests have been completed at this time utilizing a recording semi-microbalance to study the early stages of oxidation of a graphite sample. In the first test TSX graphite (sample No. 161-258C) was heated to approximately 875 C in CO_2 flowing at approximately 1 standard cu ft/hr. After 1, 2, 8, and 44 hours the sample had lost 0.09, 0.15, and 0.83 weight percent, respectively.

A second sample (161-258B) was outgassed for approximately 6 hours at 875 C and approximately 2×10^{-4} mm of Hg. The sample was then oxidized in CO_2 and after 1 and 2 hours, it had lost 0.11 and 0.13 weight percent, respectively. For the next 8 hours the weight loss rate (6.2×10^{-5} g/g/hr) was constant. From 10 to 53.5 hours the sample lost weight at a constant rate of 7.5×10^{-5} g/g/hr. The cause for the small increase in rate is not known. After 53.5 hours, the sample had lost a total of 0.51 weight percent.

A third sample (161-272A) was outgassed for 20 hours under similar conditions to the previous sample. It showed no decreasing rate of weight loss with time, but rather a slight increase in rate. After 4, 8, 12, and 16 hours, the sample had lost 0.011, 0.023, 0.035, and 0.050 weight percent, respectively. The unexpectedly large effect of prior outgassing on oxidation rate will be investigated further.

Irradiation of Coated Graphites

The second boat of silicon carbide coated graphite has been irradiated in K Reactor to an exposure of 4870 MWD/AT_K at 600 to 650 C.

The boat contained some previously irradiated coated samples, virgin samples that had been thermal cycled, and some different shapes such as rods, balls and rectangular parallelopipeds.

Thermal cycling in air from 250 C to 1200 C has been started to check the integrity of the irradiated coatings. Data obtained to date are given in the following table.

The weight changes are very small, indicating that the coatings were not sufficiently damaged by the irradiation and thermal cycling to cause the coatings to crack.

Sample No.	Geometry	No. Cycles	% Wt. Change	Remarks
M-12	Rod 1/2" OD, 4" long	1119	+ 0.022	2nd irradiation
M-3-1	1/2" x 1/2" x 4" parallelopipeds	568	+ 0.006	No prior thermal cycling
M-3-3	"	300	+ 0.011	
M-3-4	"	618	+ 0.006	
C-9	1/2" OD x 2" long	306	- 0.002	
C-13	"	300	- 0.031	
C-16	"	300	- 0.035	

EGCR Graphite Irradiation

Construction of the fifth capsule, H-3-5, in the series of long-term irradiations of EGCR graphite has been completed, and the capsule was installed in the GETR on September 1, 1962. The modification of the capsule design reported last month proved to be entirely satisfactory with no shifting of the central cooling rings occurring. A modification in the method of fluoroscopy provided a more positive indication of cooling ring location so that all swaging of the outer shell was done exactly on the respective rings. The capsule is currently awaiting reactor startup.

Irradiation of B_4C Graphite

Two capsules have been designed and are being built to irradiate 12 samples of graphite containing 6 to 9 percent boron as B_4C. These irradiations will take place in the Snout II facility and will be exposed to approximately 10^{19} nvt ($E > 0.18$ Mev).

The graphite to be irradiated will be production, high-grade material for the Fermi reactor inner-borated region. It is of two types, "Black" and "gray", which differ in the way the boron appears in the material. The "black" contains a relatively heterogeneous dispersion of B_4C, whereas the "gray" contains B_4C in solution.

7. GRAPHITE RADIATION DAMAGE STUDIES

Irradiation of Carbon-Black Graphites

A set of samples containing varying amounts of Thermax carbon black have been irradiated in K Reactor. The purpose of the irradiation was to determine the effect carbon black additions on dimensional stability. Carbon black is often included in mix formulations to increase density or lower permeability to gases. Previous tests have suggested that non-graphitizing fillers such as carbon blacks may increase the radiation-induced contraction of graphite at high temperatures.

Samples were prepared by the Pechiney Company of Chedde, France, under a cooperative exchange agreement with the Dragon Project in England. A filler of ground artificial graphite originally prepared from Texas Lockport coke was used. To attain a sound structure it was necessary to adjust the amount of pitch binder for formulations with differing amounts of carbon black. All samples were extruded as two-inch diameter rounds and heated together to a temperature of 3000 C. The composition and length changes observed after an exposure of 2639 MWD/AT_K at 600 to 650 C are given in the following table. This exposure is approximately equivalent to 6.2×10^{20} nvt ($E > 0.18$ Mev).

Hanford Ident. No.	Mix Comp., % filler by wt. Artificial Graphite Powder	Thermax Carbon Black	Pitch Binder	Length Change, % Transverse	Parallel
61-75	100	0	440	-0.014 ± 0.008	-0.035 ± 0.015
61-76	80	20	285	-0.044 ± 0.008	-0.039 ± 0.006
61-77	75	25	285	-0.047 ± 0.005	-0.046 ± 0.000
61-78	67	33	265	-0.043 ± 0.005	-0.060 ± 0.004
61-79	50	50	310	-0.068 ± 0.008	-0.061 ± 0.004
CSF (for comparison)				0.000	-0.015

With the exception of the mix containing 33 percent carbon black, a general trend of increased contraction is noted in both directions as the carbon black content is increased. There is also an apparent reduction in the anisotropy of contraction with increasing amounts of carbon black. The samples will be recharged in the irradiation facility to determine if the initial trends continue at higher exposure.

Irradiation of Pyrolytic Graphite

Irradiation at 600 C has been completed on production-grade pyrolytic graphite specimens. The peak exposure was estimated to be 2640 MWD/AT_K, equivalent to approximately 6×10^{20} nvt (E > 0.18 Mev). Samples were rectangular rods, 1/3 x 1/2 inch in cross section and 2 to 4 inches long. The average coefficient of thermal expansion (25 to 425 C) for these specimens prior to irradiation was 23.0×10^{-6} per degrees C in the transverse direction and 0.05×10^{-6} per degrees C in the parallel direction. The average change in length due to irradiation is -0.07 ± 0.02 percent in the parallel direction. This value is obtained from measurements at fifteen separate positions on four specimens. In the transverse direction an average expansion of 0.18 ± 0.10 percent was observed from 11 separate positions. Previous irradiations of laboratory samples deposited on cylindrical mandrels caused a slight expansion in the parallel direction. Because these samples were cleaved from the outer surface of the deposit, they likely contained complex compressive stresses parallel to the a axis which were probably partially annealed during irradiation.

In the current theory of high temperature radiation-induced contraction in graphite, a vestige of low temperature damage, that is, expansion in the c direction and contraction in the a direction, is believed to occur within the crystallites. The observation of such radiation-induced changes in pyrolytic graphite, which is analogous to a large crystallite, tends to confirm this aspect of the theory.

8. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

Dynamic Corrosion of Aluminum

Aluminum alloys (two lots of 1.8% Fe, 1.2% Ni; two lots of X-8001; and one lot of 10% Si, 1% Ni) are under test in 330 C deionized water, 25 fps flow, refreshment rate of 9 gal/hr. After ten days, penetrations were slightly higher for the silicon, nickel aluminum than X-8001 (1.0 vs 0.87 mil). However, the film on the 10% Si, 1% Ni alloy appeared somewhat more adherent and resistant to corrosion-erosion effects. After thirty days the appearance of the silicon-nickel aluminum alloy is at least as good as that of X-8001. The 1.8% Fe, 1.2% Ni alloy coupons show a black, adherent oxide, characteristic of good oxidation behavior by this alloy. Further tests on the silicon, nickel, aluminum alloy are contemplated to investigate corrosion resistance at relatively low temperatures in reactor process water.

9. USAEC-AECL COOPERATIVE PROGRAM ON DEVELOPMENT OF HEAVY WATER MODERATED POWER REACTORS

Thermal Hydraulic Studies

Heat Transfer Characteristics of 19-Rod Fuel Elements. Fabrication was completed of the test section designed to study the feasibility of steam generation when using bundle type fuel elements in a horizontal position. The test section is 76 inches long and consists of 19 rods spaced 0.050-inch apart in a 3.25-inch coolant tube. Twelve of the rods were wrapped with a wire on a 10-inch spiral pitch to maintain spacing between rods and to promote flow mixing. Thermocouples were installed in most of the rods and many of the water passages for detection of possible excessive temperatures due to stratification of the steam-water mixtures during boiling conditions.

The test section was installed in the high pressure heat transfer apparatus and initial isothermal runs were started.

Component Testing and Equipment Development

Dome Seal Type Nozzle Closures. Calculation of the edge seal pressures which would be developed at CANDU conditions of pressure and temperature and with the seal configuration and dimensions currently used in the gas loop nozzle seal was completed. Utilization of the general equations developed last month has proved to be quite feasible but also quite time consuming. Results of the analysis indicate the present design may be quite appropriate for these conditions.

10. REACTOR AND NUCLEAR SAFETY STUDIES

Advanced Reactor Concepts Studies

Plutonium Fueled Spacecraft Reactor. The concept of the Plutonium Fueled Spacecraft Reactor currently under study was modified by changing the basic design concept of the core from an array of fuel pins to an "inverted" core in which massive blocks of fuel material are pierced by circular coolant tubes arranged in a triangular lattice. The inverted core arrangement should provide considerably improved thermal hydraulic characteristics, minimizing the probability of "hot spots" caused by variations in flow at the heat transfer surface, and hopefully making more feasible the production of high-quality vapor at the high heat transfer rates envisioned for the reactor. The new core arrangement will also considerably reduce the complexity of schemes for venting of fission products from the fuel; such venting appears essential to obtain the high burnups envisioned.

Some effort during the month was applied to studies of design requirements for other components of the reactor cooling system, such as the radiator, turbine, and other auxiliary equipment. It is planned to carry such auxiliary equipment studies only far enough to obtain realistic values for "typical" sizes, weights, and performance characteristics of such equipment.

Application of Plutonium to Compact Reactors. In order to gain a broader view of the applicability and worth of plutonium fuel in compact reactors, and to augment and supplement studies of the Plutonium Fueled Spacecraft Reactor, a preliminary study was started of the effects of substituting plutonium for uranium as a fuel in a number of spacecraft power reactor concepts being developed at other sites. Two reactors selected for initial study are a boiling-potassium reactor proposed by ORNL and a closed-cycle gas-cooled reactor proposed by GE-NMPO. Possible benefits to be sought include reduction of reactor size and weight, extension of core lifetime, and reduction of control system complexity through flattening of reactivity.

Fuel Reuse. A report on "fuel reuse" studies was completed in rough draft form. This describes fuel cycle economic analyses for cases of direct interchange of fuel between thermal power reactors and fast power reactor blankets without processing and fabrication. The significant conclusions of this report are:

1. For economic parameters in the range of practical interest, "fuel reuse" offers a potential savings in fuel cycle cost of 0.5-1 mill/kwhr.

2. For the limiting case where the incremental cost of modifying the fast reactor blanket elements to fit the thermal reactor is equal to the total fabrication cost of enriched fuel for the thermal reactor, there is still a small potential economic incentive for fuel reuse of about 0.2 mills/kwhr.

Nuclear Safety Studies

Nuclear Health and Safety. A draft of the HAPO manual of nuclear health and safety requirements was distributed for internal Hanford Laboratories comment. This manual covers the governmental and company requirements for reviews and reports on nuclear health and safety.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

This study is aimed at establishing the combined effects of neutron irradiation and interstitial impurity atoms on the properties of metals. Most of the effort has been on molybdenum containing carbon in concentrations below 500 ppm.

A set of three capsules have been received in Radiometallurgy after a nominal irradiation of 10^{18} nvt (E > 1 Mev). The capsules contain the following Mo specimens: (a) 26 1/8-inch diameter single crystal tensile specimens of three classes (containing carbon in the range 10-20, 100-200, and 400-500 ppm); (b) six single crystal specimens of accurately determined lengths for length change measurements, two from each carbon class; (c) six single crystal x-ray diffraction specimens, two from each carbon class; (d) 21 polycrystalline one-eighth-inch diameter polycrystalline tensile specimens of which seven were annealed for 16 hours at 1050 C, seven at 1300 C, and seven at 1500 C; (e) eight polycrystalline 0.375-inch diameter tensile specimens annealed for 16 hours at 1050 C; and (f) two stored energy specimens, two length change specimens, and ten rods for hardness measurements, all prepared from material identical to (e). These capsules will be opened by a remotely operated lathe which is being installed in Radiometallurgy facilities. Similar capsules are currently under irradiation to goal exposures of 10^{19} and 10^{20} nvt (E > 1 Mev).

As reported previously, foil specimens of Johnson-Matthey molybdenum in the as-rolled state and in a stress relieved state (830 C anneal) show no damage discernible by electron microscopy after irradiation at 40 C to $\sim 10^{19}$ nvt (E > 1 Mev); however, defects in the form of spots and loops 25-150 A in diameter did form in the foil during post-irradiation annealing. That the results are valid and can be reproduced was established in a recent test. Non-irradiated and irradiated foils were annealed simultaneously at 600 C for two hours and after furnace cooling were thinned for transmission microscopy. Defects of the type described were observed only in the irradiated specimens. These results were of great interest to individuals at the Fifth International Congress for Electron Microscopy who were familiar with radiation damage studies being conducted under A. A. Johnson at the Imperial College, London. The latter has recently published an article in the August issue of Phil. Mag., in which he implies that neutron damage in irradiated and subsequently annealed molybdenum cannot be detected by electron microscopy. Additional information concerning Johnson's work will be obtained.

Analysis of the single crystal tensile tests is being continued. Data from four low-carbon samples (10-20 ppm C) have been studied in detail. The dominant deformation mode is slip on a [112] plane in a < 111> direction, regardless of initial specimen orientation. Failure occurs, leaving a chisel edge parallel to a [011]direction. The medium and high carbon crystals (100-200 and 400-500 ppm C) appear to deform in the same manner, but fail, leaving a (100) cleavage face. Completion of these analyses will permit the calculation of shear stresses on these slip planes.

A study of the diffraction of x-rays having wave lengths near the absorption edge in molybdenum has been proposed as a means for evaluating crystal perfection. In more nearly perfect crystals, extinction results in an anomalously large absorption coefficient. The dicontinuity in the normal absorption coefficient at the absorption edge is then partially masked by the anomalous absorption due to extinction. Tentative results indicate that this method will be useful in the study of crystal imperfections in molybdenum.

Polycrystalline molybdenum specimens annealed at various temperatures were subjected to tensile testing. The object of the annealing treatments was to produce a range of grain sizes for a Petch analysis of irradiation damage. The annealing treatments and resulting grain sizes were as follows:

Annealing Treatment	ASTM Grain Size (E 112)
16 hours @ 1050 C	No. $8\frac{1}{2}$
16 hours @ 1350	No. 8
1 hour @ 1800 C, followed by 16 hours @ 1550 C	No. $7\frac{1}{2}$

The results of room temperature tensile tests on these specimens prior to irradiation were as follows:

Annealing Treatment	U.Y.S. (psi)	L.Y.S. (psi)	0.2% Y.S. (psi)	U.T.S. (psi)	Elong. (%)	R.A. (%)
16 hrs @ 1050 C	49,000	42,700	--	70,700	50.8	72.2
16 hrs @ 1350 C	--	--	43,100	72,600	42.5	59.2
1 hr @ 1800 C followed by 16 hrs @ 1550 C	--	--	37,700	66,800*	19.7	9.7

*Also fracture stress.
All results average of three tests.

A paper, "Diffraction Effects from Irradiated Aluminum Single Crystals," was presented at the Eleventh Annual Conference on Applications of X-Ray Analysis, Denver, Colorado.

A paper, "Defect Structures Observed in Neutron-Bombarded Aluminum," was presented at the Fifth International Congress for Electron Microscopy, Philadelphia, Pa.

E. CUSTOMER WORK

1. RADIOMETALLURGY EXAMINATIONS

Metallographic examination has been completed on a series of stress corrosion coupons from the Purex storage tanks with no defects being found on the welded specimens or the control pieces.

The cause of the failure of an enriched production element from 2091 KE was determined to be a pin hole in the male end weld. Mechanical damage was found in the cladding of two overbore elements but no splits were found in the uranium under the damaged area. Samples of boron carbide were removed from a section of a horizontal safety rod to determine if there was any water damage.

2. EQUIPMENT PROJECTS

Project CGH-858 (High Level Utility Cell)

With the exception of the extended reach manipulators, all equipment is installed and operable. Training of operators will start the week beginning October 1, 1962, and first radioactive material should be introduced into the cell in the latter part of October.

Project CGH-857 (Physical and Mechanical Properties Testing Cell)

Cell Castings. All the panels and accessories have been received by Minor Construction. The base casting has been set and the side castings are being installed.

Instron Tester. The Instron tensile tester straining frame was removed from "I" Cell and correction was made in the clutch engagement. The unit is back in the cell and operable. Extensometer accessories have been installed and grips for both flat and curved tube section type tensile specimens are available.

3. METALLOGRAPHY LABORATORIES

Prototype fuels containing thorium - 2½% uranium and thorium - 2½% uranium - 1% zirconium have been received for metallographic examination. Considerable difficulty has been experienced in the past in performing metallography on thorium and these samples are no exception. A satisfactory grinding and polishing procedure consists of first grinding through 600 grit silicon carbide papers on Buehler Automet polishers followed by polishing on Syntron vibratory units. A brief, final lapping by hand on a rotating disc is usually necessary also. Several etchants were tried to reveal the microstructure satisfactorily, the best to date being an electrolytic etch in concentrated (85%) phosphoric acid at about 8 volts for 10 to 15 seconds. This etchant, unlike others, not only outlined the various phases and grains in the two alloys, but retained the inclusions as well.

A polyester resin has many advantages as a mounting media for all kinds of metallographic specimens and has been used routinely in the Metallography Laboratories for some time. It is often desirable or necessary to remove a metallographically polished specimen from the mounting media without applying undue stress or heat. Unfortunately, there is no solvent, to our knowledge, which will readily dissolve the polyester resin used. In tests of a number of solvents, three have been somewhat effective in removing samples from mounts, namely, ethylene dichloride, methyl ethyl ketone and ethyl acetate in descending order of effectiveness.

The effect observed in the tests was a flaking of small pieces from the surface of the resin over a period of two weeks to a month. The small flakes remained hard, and it is conjectured that swelling was responsible for flaking because little or no dissolution occurred. Flaking and subsequent removal of the specimen from the mount can be accelerated to some degree by warming the solvent in a reflux condenser if conditions permit the sample to be heated.

4. N-REACTOR CHARGING MACHINE

Modifications

The new transfer arm presence indicators were installed and tested. These new presence indicators caused the transfer arm system to operate unsatisfactorily. Further modification of the indicators was started.

Modifications to the new limit switch assemblies which control the vertical movement of the machine while loading or unloading magazines were completed for the assemblies on one side of the machine. These were tested and operated satisfactorily. Modifications to the assemblies for the other side of the machine were started.

Fabrication of the magazine piston removal equipment was completed. The equipment was installed on the machine. Fabrication of the rear pressure roller assembly was completed. The elevator control console was received and installed on the machine. The television monitor and associated equipment was received and installed. The front console was modified as required to accept this equipment.

Testing

Test reports were completed and issued for Design Test No. 4, which covers testing of the machine cross travel drive, and Design Test No. 13, which covers the hydraulic system. A draft of the plug conveyor functional test, Design Test No. 7, has been completed. Testing has been completed on the magazine support functional test, Design Test No. 18. The data are being evaluated and a report is being written. Portions of the transfer arm functional test, Design Test No. 9, and the filtered water system test, Design Test No. 19, have been completed.

All hydraulic connections on the charging machine were checked for tightness.

5. SPECIAL PLUTONIUM FABRICATIONS

Fission Product Transient Samples for Phillips Petroleum Company

Twenty-four fission product transient samples containing U-235 - Al alloy cores have been completed through autoclaving. Final assembly and end cap welding is in progress.

Casting and sampling of U-233 - Al core alloys is currently in progress.

Extrusion of the U-235 - lithium, plutonium-lithium bearing alloys is still being delayed due to analytical difficulties.

High Exposure Plutonium-Aluminum Fuel for Physics Instrument Research

The Zircaloy tubing has been ordered and is to be delivered the last week in September for the 1000 high exposure plutonium-aluminum rods for PRCF. End caps are to be delivered at the same time. The core material has been cast and will be extruded by October 1.

The PCTR-high exposure plutonium-aluminum fuel loading is 75 percent complete. Work is being held up by analytical problems with the remainder of the core material.

F. W. Albaugh

Manager, Reactor and Fuels Research
and Development

FW Albaugh:kb

PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATION

MONTHLY REPORT

SEPTEMBER 1962

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

N-Reactor Exponential Experiments

The experiments in the NPR mockup with control rods inserted are almost completely analyzed. It has been determined that the vertical extrapolation distances measured for the experiments with rods are the same as without rods; horizontal extrapolations were assumed to be the same.

Optimization of Retubed Lattices

The "C"-pile mockup has been changed to the overbore lattice by replacing tube-blocks and fuel; and measurements have begun.

Angular Distribution of Thermal Neutrons

A measurement of the angular distribution of thermal neutrons in and near a copper rod in graphite is under consideration. A set of transfer cross sections for 13 thermal groups has been calculated for use in multigroup S-X calculations for the experimental geometry. The derivation of cross sections for 5 fast groups is almost complete. The calculation will give the space-energy-angle distribution of neutrons for the PCTR with the copper bar in the center. This result can then be used to plan an experiment by indicating the radial positions and angles which should be studied as well as the spectral hardening which will determine the need for other detectors such as lutetium.

Code Development

Revisions to a special version of TRIP, the reactor kinetics code, have been satisfactorily run. Although no program errors have been found, PROBC, the collision probabilities code, gives probabilities greater than one in some cases; calculations are being held up pending further analytical work. SUMMIT, the GA code for calculating neutron scattering kernels in graphite, has been adapted to our monitor system and appears to be in running order. The punched card output has been revised to conform to SPECTRUM input requirements.

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Spatial Resonance Self-Shielding

Assistance to Programming Operation on heterogeneous cell calculations continued. A 1983 energy-level tape of Doppler broadened cross sections for U-235, U-238, Pu-239, Pu-240, and a 1/v absorber was made. The analysis of the UO_2 rod configurations mentioned last month was completed using the 19-energy level diffusion program G_2 with unbroadened and with 500 F broadened self-shielding factors. Using the broadened cross sections reduced k_∞ by about 40 mk in each case.

Instrumentation

The specifications for the multi-channel analyzer for use with the NPR Fuel Rupture Monitor were thoroughly reviewed and modified. A punched paper tape read-out and read-in unit will be used with the analyzer. This will facilitate its supplemental use in studies of changes in corrosion product activities or in determinations of loop clean-up factors, for example. All portions of the specification were updated to provide the latest possible circuit innovations.

Final drafts of test procedures for the NPR nuclear instrumentation were reviewed and comments were made at the request of both Instrumentation and Electrical Design, IPD, and Electrical and Instrumentation Design, CE&UO.

At the request of Testing Methods Engineering, FPD, an estimate was made of the fast neutron flux in an empty NPR process tube. The information will be used in planning a future test.

Design work continued on the experimental fuel failure detection instrumentation for the NPR fuels testing loop at the PRTR. The detector assembly, general sampling system, and ventilation and drainage arrangement designs neared completion. A number of purchase specifications for the electronic instrumentation were also drafted.

Systems Studies

Technical assistance was provided the NPR Project Section in evaluating transformers for isolation of the Flow Data Logging System from the Flow Monitor input lines. Recommendations also were made to reject several configurations of Temperature Monitor test set-ups proposed by the vendor.

Assistance was provided the IPD Reactor Design Analysis Operation in the failure sequence analysis of NPR instrumentation.

Vendor drawings for a single Bailey control system of the NPR type were reviewed. The system will be used for analysis and training purposes.

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Delivery is expected by November 1, 1962.

In the study of reactor transfer functions, analysis of the neutron flux vs. rod position data recorded in July has indicated the need for more accurate chamber calibration data and the use of a larger number of tests in a given test period. To date the outputs of only three of the five in-core chambers have been recorded simultaneously with rod position change due to equipment limitations. Repeated tests using different sets of three chambers were made in an attempt to overcome this limitation. However, because the number of rod movements which can be made during a given test period is limited to prevent undesirable disturbances to reactor flux distribution, future tests will be designed to record all five chambers and one rod position simultaneously. In addition, the rods will be selected to minimize the magnitude of rod motion necessary, thus allowing a larger number of movements to be made. Further tests will not be scheduled until calibration data have been obtained on the in-core chambers by IPD Instrument Development Operation.

An investigation of an application of the Parametric Expansion method of Dynamic Optimization to a simple position control servomechanism was conducted with use of an analog computer. The system feedback parameters were first determined in reverse time, then applied in forward time for specific input signals (steps and ramps) to the system.

A particular form of the time independent diffusion equation was submitted by IPD for analog solution. The boundary conditions were that the dependent variable was to be zero at two positions in space. It was required that a value for one of the constant coefficients be found such that the equation would fit the boundary conditions. For the analog simulation, the space variable, x, was replaced by time, t; the analog representation of the dependent variable was thus continuous over the range of t. The problem was also submitted to digital programmers for solution on the 7090; the 7090 program employed a ten-node finite difference approximation. The analog and digital solutions differed by less than five percent.

Two meetings were held with CE&UO Design personnel to firm up building and services requirements for a general purpose analog simulation laboratory proposed for HAPO. A study to determine the relative costs and advantages of various 300 Area sites is under way. One meeting was held with IPD personnel to firm up the design basis for an NPR plant simulation. A rough draft of the document is being prepared by IPD.

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SEPARATIONS

Experiments with Plutonium Solutions

Critical mass experiments were continued with plutonium nitrate solutions in the 14-inch diameter stainless steel sphere covered with a 0.03-inch cadmium shell and fully reflected with water. Further data were obtained for evaluating the effect of cadmium on the criticality of the water reflected unit.

The results of these measurements are summarized in Table I which follows. These data imply that for a $Pu(NO_3)_4$ solution containing ~ 48 g Pu/ℓ at an acid molarity of ~ 1.4 (total nitrate of ~ 140 g/ℓ), the vessel would be just critical when full (containing 23.22 liters). These numbers are based on a knowledge of the materials used in mixing the solutions and remain somewhat preliminary--until the detailed chemical analyses of the solutions have been received.

The effect of nitrate on the critical concentration of the cadmium covered vessel can be used in a qualitative manner to estimate the equivalent water reflected sphere (with no cadmium).

Considerable difficulties were encountered during the month as a result of equipment failure. The safety rod drive mechanism (not the rod itself) became jammed due to the slippage of one of the drive gears. The unit was returned to operation after repairs.

A leak developed in the liquid level manometer line where it entered the vessel through the pedestal support at the base of the assembly. As a result, Pu solution was leaking into the dump tank during the course of the experiments. In order to effect repairs, it was necessary to completely dismount the critical assembly vessel and to disassemble the dump mechanism. The faulty weld in the line was covered with epoxy resin and the unit reassembled. Although the units being repaired were highly contaminated, this work was accomplished without spread of contamination.

Experiments with Plutonium Oxide-Plastic Mixtures

Work has proceeded on the wiring and installation of the remotely operated split-table critical assembly machine for use in criticality measurements with PuO_2-plastic mixtures.

The Plutonium Metallurgy Operation has now completed the fabrication of 162, two-inch cubes of PuO_2-polystyrene for use in these experiments; this represents about 60% of the total Pu which is to be prepared for use in the

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TABLE I

CRITICALITY STUDIES WITH PLUTONIUM SOLUTIONS
IN 14-INCH DIAMETER STAINLESS STEEL SPHERE

(Measured Sphere Volume: 23.22 liters; Wall Thickness: 0.044-inch)

Experiment Number	Date	Reflector	Pu Conc. (g/l)	Acid Molarity	Sp.Gr.	H_2O (g/l)	Total NO_3 (g/l)	H/Pu Atomic Ratio	Critical Volume (liters)	Critical Mass (Kg Pu)*
1142118	8-30-62	Full Water + 0.03" Cd Shell	60.2	1.35	1.151	946	147	422.5	$21.9^{+.05}_{-.06}$	1.32
1142119	8-31-62	Full Water + 0.03" Cd Shell	56.7	1.27	1.148	957	137	453.4	$22.0^{+.04}_{-.04}$	1.25
1142120	9- 4-62	Full Water + 0.03" Cd Shell	55.0	1.49	1.144	942	150	461.1	$22.3^{+.02}_{-.02}$	1.23
1142121	9-10-62	Full Water + 0.03" Cd Shell	50.1	1.39	1.122	932	138	500.4	$22.8^{+.02}_{-.03}$	1.14
1142122	9-11-62	Full Water + 0.03" Cd Shell	50.2	1.39	1.123	938	139	502.6	$22.9^{+.05}_{-.07}$	1.15
1142123	9-27-62	Full Water + 0.03" Cd Shell	~48 **	~1.4 **					$23.3^{+.06}_{-.09}$	

* Pu^{240} content 4.6 percent.

** Chemical analysis incomplete and numbers subject to change.

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initial experiments. The concentration of the Pu in the plastic mixture is 1.15 g/cc; the H/Pu atomic ratio is ~ 15.

The problem of sealing the individual blocks for contamination control now appears to have been solved. In the method to be used, each block will be sprayed with a coating of rubber. The rubber coating has the following additional favorable qualities: (1) resistive to damage under normal handling, (2) surface texture conducive to assembly construction, (3) easily applied to cubes and (4) easily removed. This method of sealing originated with the Plutonium Metallurgy Operation.

Limiting Critical Concentrations of U^{235} and Pu^{239} in Aqueous Solutions

Stainless steel tanks were used for the containment vessels in previous PCTR experiments for determining the limiting critical concentration of a Pu-water mixture. The uncertainty in the measured value of 8.4 ± 1 g Pu/ℓ was largely the result of the uncertainty in the correction for the effect of the stainless steel on the measured value of k; subsequent measurements were then conducted to study the effect of the stainless steel (as used in the containment vessels for the Pu solutions). Dilute uranyl fluoride (UO_2F_2) solution was used in these experiments, since the results were expected to be applicable to the Pu experiments, and there would be no potential hazard to the PCTR from Pu contamination. An analysis of the experimental data from the measurements completed during July yielded the following results:

After correcting for the stainless steel, the value for the limiting critical concentration of U(93.2% U^{235}) was calculated to be 13.0 ± 0.1 g U/ℓ. This value is in excellent agreement with the previous value of 12.94 ± 0.03 g U/ℓ which was obtained with the aluminum vessels.

Knowledge gained from the more recent experiment for the effect of the stainless steel was used to correct the plutonium solution data. This resulted in a modification of the value obtained originally for the limiting critical concentration from 8.4 ± 1.0 g Pu/ℓ to 8.1 ± 0.3 g Pu/ℓ. The latter value is also in better agreement with Monte Carlo calculations which have given 7.71 ± 0.46 g Pu/ℓ.

The Effect of Fuel Lumping on the Criticality of a Moderated Plutonium System

A problem of interest from the viewpoint of criticality is the effect of resonance absorption in Pu^{239}. Since plutonium contains a large resonance absorption at ~ 0.3 ev, which has a particularly low value for η (~ 1.74) one might speculate as to whether the smallest critical mass for a moderated

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plutonium system is obtained with a homogeneous system.

To study this problem, a series of calculations were made for Pu rods with diameters of 0.2 cm, 0.02 cm, and 0.002 cm in a water lattice at the H/Pu ratio which gives the minimum mass for a homogeneous moderated system ($\sim$33 g Pu/ℓ). The thermal flux depressions with small fuel cylinders were determined by a S_4 transport calculation. The critical mass calculations were carried out with the HFN multigroup diffusion code using 13 group cross sections, two of which were below 0.625 ev. The results showed that the reduced capture in the 0.3 ev resonance, due to lumping, was far overshadowed by the smaller thermal utilization as a result of the thermal flux depression within the fuel rods. Therefore, it was concluded that the effect of lumping the fuel at the above concentration would not result in a decrease in the critical mass, but rather in an increase.

Effect of Composite Reflectors on Criticality - Comparison of Theory and Experiment for Cadmium Wrapped - Water Reflected Vessel

As a check on the computational methods for predicting the criticality of cadmium wrapped - water reflected vessels, a multigroup diffusion calculation was used to compute the critical volume as measured in one of the critical experiments at the Laboratory. For the experiment in question, the measured critical volume was $23.10^{+0.05}_{-0.06}$ liters for a $Pu(NO_3)_4$ solution with an acid molarity of 3.74 and a plutonium concentration of 69.4 g/ℓ.

The fast group parameters used in the calculation were obtained from the GAM-I slowing down code, and the thermal group parameters were obtained by averaging over a Wigner-Wilkins spectrum.

For the 23.1 liter cadmium wrapped vessel containing the above solution, the calculated multiplication factor was k_{eff} = 0.99. This one percent difference in experimental and calculated k_{eff}'s is equivalent to a 3.35% difference in critical volume.

Buckling of Partially Filled Spheres

A one-group , one-region, two-dimensional partial difference code is being written and debugged to solve the equation $\nabla^2 \phi + B^2 \phi = 0$ for a partially filled sphere. This calculation will yield (1) a more exact approximation to the buckling of an unreflected system, and (2) a finer calculation for the ratio of neutrons leaving the top and bottom surfaces, which is necessary for the reflected sphere calculation.

Transport Theory Development Work

Recent work on improving the range and reliability of criticality limits pertinent to plant safety led to extending the machine analysis of multi-energy transport theory (GE-HAPO Program S) to provide achievement of criticality by variation of the metric tensor. This improved analysis versatility eases the intricate task of delineating the critical mass envelope of a family of criticality curves, by permitting direct variation of size and shape, and thus reducing the need for pyramidal sequences of multiple runs.

The analysis permits flexible variation of the relative size of each region of the reactor as well as variation of over-all dimensions. In terms of an input-output characterization, each element of distance ds employed in the analysis is computed from its initial value dr by the metric transformation

$$ds = \left[1 + (1/k-1)\, g(r)\right] dr,$$

where g(r) is a flexible transformation generator provided as input, and K is an output eigenvalue specifying the extent to which the corresponding shape or size deformation must proceed to achieve criticality.

The analysis method for computing the deformation eigenvalue K is based upon chain-compounding transport perturbation theory progressively to high order, until three successively higher order representations show satisfactory coalescence. Formulation and programming work has been completed. Checkout work is proceeding satisfactorily and is near completion.

Instrumentation and Systems Studies

Work was started to replace the liquid level manometer at the Critical Mass Laboratory. Either an ultrasonic or a pressure measuring device will be used if a commercial unit is available with the required accuracy--20 inches plus or minus 0.01 inch.

Installation and testing of the new Hanford standard criticality alarm horns is now complete at the CML. The print-out and telephone dialing devices are being installed and should be completed by the first part of November. This unit prints out the time and identifies the alarm actuating monitor when a critical alarm occurs. Also, a device will dial a telephone and announce the alarm condition.

Development and installation of an aural monitor on channel two at the CML is now complete. This instrument permits the division of the pulse rate by 10, 100, or 1000 resulting in a more understandable aural representation of the flux level change.

A new Keithly Model 420 A log n-period meter has been received and is being evaluated for possible use in the CML safety circuit. It appears to work quite well except that the high frequency response cut-off of the period meter section will have to be decreased some.

All laboratory tests on the new control rod drive system for the CML were successfully completed and the system was delivered. A report describing the system was prepared.

Thirty-one runs were made on the EASE and GEDA analog computer simulation of a pot calciner. The object was to determine estimates of the safe operating conditions for a heated cylinder (called a pot) containing a heat generating waste material undergoing chemical decompositions and melting. It is desired to heat every portion of the cylinder to 850 to 1000°C without exceeding 1000°C by using a heater producing a wall temperature of 900°C, and then to cool the wall to 200°C until a steady state temperature is obtained. It was found that heating periods of 8 to 20 hours (depending on the heat generating rate and thermal conductivity) would be required. The thermal conductivity functions were 0.1, 0.2, and 1.25×10^{-4} T + 0.085 Btu/hr-ft °F. The heat generating rates were 0, 2500, and 5000 Btu/hr-ft^3 for 8 and 12 inch cylinder diameters. Only four finite difference sections were used to simulate the partial differential equation for the process. As a result an error of slightly less than 10% was introduced into the simulation. The addition of one or two additional sections probably would reduce the error to less than two percent.

Mass Spectrometry

The realignment and refocusing of the heavy element mass spectrometer for this program was completed. The mass resolution of the spectrometer is now within the design specifications, and satisfactory operation of the spectrometer was maintained during the month. A series of analyses of National Bureau of Standards uranium isotopic standards was performed in order to calibrate the accuracy and precision of analysis of the spectrometer.

Consulting Services on Nuclear Safety - Criticality Hazards

Nuclear Safety in CPD

Participation on the Recuplex Deactivation Hazards Review Committee and the Project 880 Hazards Review Committee continued throughout the month.

Reviews concerning Recuplex deactivation centered around methods of removing plutonium from the J-26A tank and the floor of the Reception-Blending Hood. J-26A is plugged with 4-6 Kg of plutonium solids. Efforts to dissolve these solids so that plutonium can be removed from the tank have not as yet been successful. A procedure using acid, heat, and slight pressure will be tried next month. In the hazard review of this procedure, attention was given to the nuclear safety of tanks into which the plutonium could be forced when it dissolves--particularly J-9, which is 7-3/4-in ID.

A procedure to remove the plutonium-organic waste materials from the floor of the Reception-Blending Hood by drying small (one liter) batches in a special drying hood and then storing the residues is being reviewed. The hood floor contains approximately 400 liters of this material which varies in plutonium concentration from 3-15 g/ℓ.

Two nuclear safety reviews were made for Advance Process Development. One review concerned the neutron counting of plutonium buttons, and the other the dissolving of PuO_2--both in 234-5 Building.

Nuclear Safety in FPD

A temporary specification covering the autoclaving of four hundred thirty-three 7% U^{235}-Al alloy fuel elements in the 313 Building autoclaves was reviewed and approved.

Nuclear safety specifications were also reviewed and approved for 1) the storage and shipment of uranium-oxide scrap, 2) the storage and processing of 1.6 w/o U^{235} enriched billets and fuel elements, and 3) the storage and processing of 2.5 w/o U^{235}-thorium alloy billets and extruded tubes.

NEUTRON CROSS SECTION PROGRAM

Quasi-Elastic Scattering of Neutrons from Water

Measurements were made at two scattering angles of the energy-broadening of the quasi-elastic component of the scattering of 0.15 ev neutrons from room temperature water. These measurements were made with significantly better neutron energy resolution than previously employed. Preliminary analysis of these data indicate a broadening which agrees within statistical errors with that previously measured at Hanford.

Inelastic Scattering of Neutrons from Water

Design work continued on water-sample holders for use in the measurement of inelastic scattering of neutrons from water at elevated temperatures using the triple-axis crystal spectrometer.

Rotating-Crystal Spectrometer

A spherical aluminum single crystal was oriented and mounted in a crystal rotator to test as a monochromator for inelastic neutron scattering measurements by the rotating-crystal time-of-flight technique. The crystal has been rotated at 10,000 rpm in test. Diffracted neutron spectra have been obtained, but the stability of rotational speed of the present system is inadequate for accurate measurement. Several difficulties have been encountered in the operation of the 1024 channel time-of-flight analyzer, and this equipment is still being debugged.

Fast Neutron Cross Sections

A series of measurements of neutron total cross sections from 3 to 15 Mev by the pulsed-beam time-of-flight technique with Van de Graaff neutron source was successfully completed during the month. The total cross sections of fourteen elements were measured to a precision of three percent or better. The elements which were measured were: Ca, Sr, Ba, Pb, Co, Mg, Y, Nb, V, Zr, Ag, Mo, Sn, and W.

In addition, a series of measurements was performed which was designed to study possible systematic errors caused by sample diameter, sample thickness, sample in-scattering, and the method of background determination.

The series of measurements was performed in approximately 150 hours of data-taking including a period of 123 hours of continuous operation of the Van de Graaff.

Instrumentation

The 1024-channel time-of-flight analyzer for slow neutron cross-section measurements is now in use at 105-KE. Some difficulties with poor solder connections are being remedied.

Two possible modifications of the 1024-channel analyzer were studied: a 6144-channel analyzer using the magnetic drum as a totalizing memory, and a magnetic tape storage unit that would use the IBM-7090 computer as a totalizer. The advantages of the latter scheme are faster acquisition speeds and an unrestricted number of channels. Costs would be about the same for either system.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Graphite Lattice Parameters for Low Exposure Pu-Al Fuel

Attempts to analyze the Lx Pu-Al fueled, poisoned lattices with the transport code have not been entirely satisfactory-- presumably due to lack of information about the thermal neutron flux spectrum in the cell. A solution to this problem would be to use a number of groups in the subcadmium region, say from 0.1 to 0.4 ev. However, there is no simple way at present to generate the cross sections required for such a group structure. In order to correct this limitation on the use of a number of groups in the thermal energy range, modifications to the code, SPECTRUM V, are under way such that cross sections for an arbitrary group structure can be obtained.

A simplified approach is also being taken to the problem to see if more satisfactory agreement with experiments can be obtained. IDIOT calculations of the thermal flux distribution have been obtained for the 10-1/2, 8-3/8 and 6-1/2 inch graphite lattices fueled with 19-rod clusters of Pu-Al fuel. The calculations have been done for both the poisoned and unpoisoned cells for each lattice spacing.

The 10-1/2 inch poisoned lattice is under study with the multigroup diffusion code, HFN. One case has been run in which only two groups were used in the calculation. The output from this case is presently being evaluated.

High Exposure Pu-Al Lattice Studies

The preparation and planning work for the experimental portion of these studies is now almost complete. Measurements on 19-rod clusters in graphite lattices with 10-1/2, 8-3/8 and 6-1/2 inch lattice spacing are planned. The 10-1/2 inch lattice will be measured first.

The PCTR will be used to determine the quantity of copper necessary to poison a unit cell to a k_∞ of one, and to activate Pu-Al, U^{235}, Al, Cu, Au and Lu foils, both bare and cadmium covered. The foil activations will be made in both the poisoned and unpoisoned lattices. An outline of the experimental procedure and hazards to the PCTR has been written.

A method of obtaining $\bar{\sigma}_a$, averaged over the lattice spectrum, by reactivity measurements is being investigated.

Low Exposure PuO_2-UO_2 Lattice Studies

Input has been prepared for an IDIOT run on a 6-1/2 inch unpoisoned lattice fueled with 19-rod clusters of both PuO_2-UO_2 and PuC-UC. The quantity of Pu and U was taken to be the same for both the oxide and carbide cases. An attempt is being made to evaluate the effects of the difference in the slowing down power on the thermal flux distribution.

The Critical Facility

The final draft of some of the process specifications which will govern the operation of the PRCF have been reviewed.

The preparation of the PRCF startup test procedures document (HW-71214 supplement) has continued through the month. The introduction and one of twenty-four tests remain to be completed. Reviews of these test procedures led to questions concerning the equivalence of paraffin and light water as a reflector. Their relative properties are being examined.

The calculations necessary for the absolute flux calibration of BF_3 counters have been completed. These calculations were made from data collected by observing the counting rates of the BF_3 chambers when placed in the Standard Pile.

The electronics equipment and counters have been moved to the PRCF area and checked for proper operation. Two of the amplifiers seem to pick up noise from electrical equipment operating in the area. This may be a problem.

Scope drawings of the modifications which are necessary for the experiments which use light water as moderator in the PRCF have been prepared by MEEO. These drawings have been reviewed and future work has been defined for the preparation of the experiments.

PRTR Fuel Irradiation Experiment

The lutetium activities which were obtained in an irradiation in the PRTR have been reanalyzed. The reanalysis was made with the assumption that the shape of the function which joins the slowing-down distribution to the Maxwellian distribution is that which has been calculated for a position in the outer ring of fuel rods in a 19-rod cluster.

The reanalysis has resulted in a value of 350 ± 10°K for the spectral index along the length of the Pu-Al fuel element. This value is not the final result, however. Parts of the analysis must be re-done on the basis

of this calculated spectral index since it is different from the one [redacted]
assumed when the joining function was calculated.

Neutron Spectrum Studies

Calculations of corrections which will be used for determining spectral indices from lutetium irradiations in 19-rod clusters of UO_2 and Pu-Al fuel elements have been completed. The corrections have been computed for foils at the surface and in the center of a UO_2 cluster and in a center rod, intermediate rod, and an outer rod of a Pu-Al cluster.

Status of PRTR Fuel Irradiation Experiments

Gamma Scanner

Due to the operational decision that only two fuel elements (or hangers containing irradiated rods or wires) can be stored in the discharge water pit during reactor operation, the use of the Fuel Examination Facility for gamma scanning will be in serious conflict with fuel element examinations and of limited availability. Installation of the gamma scanner in the basin is now being considered.

Burnup Analysis of PRTR Elements

Element 5075

Using the measured unirradiated plutonium isotopic composition (6.14 w/o Pu-240, 0.484 w/o Pu-241, 0.021 w/o Pu-242), the MELEAGER calculation now produces Pu-241 concentrations which agree with the measured values at the exposures which produce agreement with the measured Pu-240 concentrations. The MELEAGER Pu-242 composition is running somewhat high (about 16% at 27% burnup). The burnup from Cs-137 data fits the new MELEAGER results on the average for a fission yield of 0.0634, using $T_{1/2}$ = 29.15 years, and 326 Bldg. mass spectrometer data. Using 325 Bldg. mass spectrometer data the yield is .0655.

Element 5051

Fuel element 5051 (first of the low exposure Pu-Al for burnup analysis) has been disassembled and three rods delivered to the Radiometallurgy Laboratory for dissection.

Investigation of an isotopic dilution technique of analysis for burnup studies is under way. If successful, the cost of burnup analysis will be lowered and the time per sample analyzed cut down appreciably.

[redacted]

Code Development

Physics Chain Tape

Programs TEMPEST, GAM, SIGMA-3C, SIGMA-3H, and HFN have been loaded onto a chain tape. Any number of these programs can be run in almost any order. Successful multi-program runs as well as single program runs have been made.

RBU

Values of the average energy loss per collision, $\overline{\Delta E}$, and the average cosine of the scattering angle, $\overline{\mu_0}$, derived from the gas kernel, have been calculated for a representative range of initial neutron energies and scattering center masses. The results appear to describe the behavior of low energy neutron scattering in a gas in a satisfactory manner. Consequently, a machine program, MOMENTS, has been written and partially debugged to provide more extensive and accurate value. These values are required to determine the parameters in the two mass gas model used in RBU.

CALX

Debugging of the burnup portion of the CALX code continued. The straight burnup, recycle, and graded irradiation options are operating, but the initial enrichment option is not. The SIGMA subroutine, which calculates isotopic cross sections after a specified number of time steps, according to a nine-group, two-region, average flux model, is also not operating properly.

CLERK JAYNE

The CLERK JAYNE program has been modified to produce input cards for the NORMAL MODE burnup code. Additional input to CLERK JAYNE is region or core power level in megawatts, the number of burnup steps to be done, and the time interval in days between steps.

RBU Cross Section Updating

Further revision of the uranium and plutonium isotope cross sections in the resonance region are being put into the RBU Library. These changes are the result of a better treatment of the resonance region (particularly, the unresolved resonance region) based upon some theoretical considerations of Dresner[1].

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(1) Dresner, L., Resonance Absorption in Nuclear Reactors, Chapter 7. [redacted] Pergamon Press (1960).

Nelkin Scattering Kernel for Water

Investigation of the discrepancy between Nelkin's theoretical water kernel and experiment continued. Transport cross sections and total cross sections calculated at Hanford and KAPL were found to agree. However, since Hanford and KAPL are both using adapted versions of a GA code, there is still the possibility that none of these codes reproduce Nelkin's theory correctly. Consequently, a second Hanford water kernel code has been prepared, and is being debugged. This particular version calculates the Egelstaff S-function, and is directly comparable with the usual presentation of experimental data. The code is especially designed to be more flexible in changing details in the calculation.

Fast-Thermal Reactor Complexes: "Fuel Re-Use"

Evaluation studies of the "Fuel Re-Use" concept have been completed. An informal document, HW-74549, "Progress Report on Fuel Re-Use: Analysis of a Fast-Thermal Reactor Complex," is in preparation.

For the single reactor complex examined (Fast Oxide Breeder - D_2O Pressure Tube Machine), fuel re-use appears to offer an attractive approach for reducing fuel costs. Possible cost reductions seem to be just under a mill/Kwhr. Further reductions in fuel costs might be achievable by multiple fuel re-use cycles. It also appears that the ratio of thermal to fast reactors for a steady state economy could be maintained at a reasonable level.

The major questions that remain to be answered are engineering feasibility (particularly the question of non-uniform enrichment), and the extension of the fuel re-use concept to other reactor types. Partial answers to these questions could be obtained from further analytical work. However, concurrent demonstration experiments of fuel re-use would add considerably greater credibility to these results. If further analytical studies are carried out, a more detailed physics analysis--to account more accurately for spatial and spectral effects--would seem to be warranted.

Phoenix Fueling of Compact, Water Moderated Reactors

The first step, obtaining cross sections to be used, in the analysis of the three reactors is complete. This involved using the TEMPEST code to produce thermal group (0.0 to 0.683 ev) cross section characteristic of a homogeneous medium for each of the three trial fuel loads in each reactor. The input to the SHUSH-HFN combination which will be used to obtain cell disadvantage factors is about half complete. The multi-group cross section group sizes have been selected. The four groups have boundaries

[redacted]

that are common to some of the nine groups used in CALX; this will allow some comparison of results.

Plutonium Utilization Studies

A comparison of uranium and plutonium fuels in several small compact fast systems is being made. One of the systems is proposed to be a 1 MW power source for space applications. Another system under consideration might be used in a nuclear rocket. Also, a larger power system with a special core geometry, which is hoped to extend the core life, is being considered for a large space power station.

Instrumentation and Systems Studies

Further work was done to improve operation of the PRTR Fuel Rupture Monitor. The linearity of the preamplifiers and multichannel analyzer was tested, and appropriate adjustments were made. It was determined that the special transistorized preamplifiers needed new batteries after 1000 hours of use. The system linearity, using various radionuclides for test sources, was found to be within ± 1%.

Limited success was achieved in attempting to reduce the transient noise problem in the PRCF fission counter channel. A charge-sensitive preamplifier and a low-noise power supply were substituted for the commercial units in use to try to isolate the noise sources. Further work will be necessary.

Because of other work by PRTR personnel, no time was available to install the second generation, final model scintillation effluent monitor for gamma emitters. As a continued test, the instrument was kept in full operation in 329 Building for the month. Performance continues to be fully satisfactory. Installation at PRTR will be done as soon as possible.

Development continued on the eddy current instrument for measuring vibration of PRTR fuel elements both in the hot loop mockup and in the PRTR. The basic sensitivity of the method was improved a factor of two (to one mv/0.001" displacement) over that reported last month. It was also demonstrated that the fuel displacements relative to both the process tube and the shroud tube can be measured simultaneously by using two test frequencies. The performance of the testing coils at temperatures up to 500° F will be investigated next.

Analysis of the data obtained during PRTR Test No. 35 showed the presence of two predominant peaks in the frequency spectra of both the moderator level and neutron flux signals. The peaks occur at approximately four and nine cycles per second. The signals recorded during the test were considerably lower in level than that required for good precision, however, and some uncertainty exists as to their validity. The results indicate that the simultaneous recording of moderator level signal with the flux signal will be helpful in interpreting the results of future tests. The test also showed that the servomanometer is not a suitable source of signals for this type of test due to its limited dynamic response. It was shown that the significant frequency spectrum of moderator level variations extends to at least ten cycles per second. A method of using the Boonshaft and Fuchs transfer function analyzer in the analysis of moderator level-neutron flux-cross-power spectra is being investigated. Further progress will be delayed until the analyzer is returned from the factory, where it was sent during the month for modification.

Analog runs were completed to determine the maximum PRTR power level excursions and the maximum fuel element temperatures produced by various reactivity disturbances with the reactor status varying from subcritical to normal operating power level. Many of the runs made were startup cases and others assumed the complete failure of the "scram" mechanism. These runs resulted in large power excursions covering many decades of reactor power level. Therefore, "point storage" or analog memory was incorporated into the circuit to produce automatic scaling of all necessary variables. The use of this technique greatly reduced the time and effort required for this study.

HIGH TEMPERATURE REACTOR LATTICE PHYSICS PROGRAM

A preliminary estimate for the cost of the High Temperature Lattice Test Reactor has been obtained. Since this estimate is 50% above that given in the initial budget study in 1961, the scope must be given more careful study. A first draft of the Project Proposal requesting design funds has been revised.

NEUTRON FLUX MONITORS

Plans were partly completed to determine experimentally the neutron flux, temperature, and spectrum in a suitable test facility to be used for the testing of various foils made of plutonium and uranium radionuclides. The results of the experiments will be used for optimizing the composition of the regenerating detectors. Performed calculations indicate that neutron temperature can be approximated by determining the nuclide composition of Pu-239 and U-235 foils as a function of exposure. It is considered possible to determine the spectral hardness from the cadmium ratio of bining information from the several sources, it should be possible to obtain a consistent set of values for ø, r, and T which can be used as a basis for fabrication of the regenerating test units.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Electronic units of the prototype multiparameter eddy current testing equipment are being checked and adjusted as they are received from the shop. The investigation of diffusion and propagation of electric fields and currents in conductors by a combined analytic and empirical approach using solid and molten Woods metal is continuing. Resolution capabilities of the new graphical nulling unit were increased, and it is ready for a field test in which it will be used with an eddy current tubing tester in the testing of 3/16 inch diameter stainless steel tubing.

An estimate of cost of construction of the prototype multiparameter eddy current equipment was prepared. The five tuned amplifiers were tested. A modification of the 250 kc and the 3 Mc units is required to obtain higher Q circuits and to reduce the tendency to oscillate. These modifications are being made. Three of the L-C oscillators have been converted to crystal oscillators and are ready for final adjustments and testing.

The investigation of the diffusion and propagation of electric fields and currents in conductors is continuing. The equipment has been adapted to provide thermostatic control of the temperature in the liquid metal. With the thermostatic control present, oxidation of the surface of the liquid Woods alloy has been very low. The inert atmosphere mentioned in the August report will probably not be required.

Several plots of eddy current distribution vs. frequency have been made with steady state A.C. signals applied to the driving coils. Four different driving coils of varying size and shape have been used to produce these eddy currents in the liquid metal and in the pickup loop. In addition, these same driving coils have been driven with square wave signals. Thus plots of eddy current magnitude as a function of time have been obtained for various depths in the metal. These plots are compared with plots derived from theoretical considerations. Additional plots will be obtained for different size pickup loops in the metal. Preliminary study indicates the eddy current depth of penetration may be less than expected.

Tubing testing equipment is being prepared to test type 304 3/16 diameter stainless steel tubing having a wall thickness of about 0.036 inch. The complete test arrangement consists of (1) a Hanford Inconel Tubing Tester

Model 1002, modified for this application, (2) amplifier and discriminator units in a commercial rod and tubing tester, (3) a commercial tube feeder, and (4) a time delay and tubing marker unit.

It is desired to detect the presence of defects which are greater than 1% of the wall thickness, including cracks and wall thinning. The assembly of equipment is complete, and it is ready for final checking out with standards which are being prepared. The Inconel Tubing Tester Model 1002 previously developed was modified for the present application. A 100 kc crystal oscillator, frequency quadrupler, and a graphical null balance unit were added. A differential type test coil assembly is used with each of the two windings consisting of 100 turns of No. 36 copper wire, operating at 400 kc. A phase discrimination technique is used to discriminate against small diameter variations and shallow surface irregularities. As expected, samples of the stainless steel tubing show varying amounts of "noise" or background signals. Placing the test coil and tubing in a magnetic field of about 3 kilogauss did not affect this noise, thus it is tentatively concluded that it is caused by tubing electrical conductivity variation. The graphical nulling unit has facilitated the identification of two types of tubing noise whose signals are characterized by different phase angles. The phase angles indicate that the sources of some small noise signals may be near the tubing surface, and the sources of some of the larger noise signals may be at a greater depth. These conditions need further study, and the effect of annealing the tubing is now being made.

The paper, Detection of Anisotropic Conditions Using Eddy Currents, by H. L. Libby and R. L. Brown, Jr., was published in Nondestructive Testing, Vol. XX, No. 5, September-October, 1962.

Zirconium Hydride Detection

Neutron attenuation studies indicate that this method could be used to detect 5000 ppm ± 2500 ppm hydride content in Zircaloy-2. Sonic damping studies indicate further correlation between damping and hydride content. Focused ultrasound methods appear more promising than plane transducer methods for determining the effect of hydride on the stress dependence of ultrasonic attenuation. Powdered Zircaloy-2 hydride samples have been sent to the University of Washington and Varian Associates for nuclear magnetic resonance studies.

Zircaloy-2 plate containing 100, 300, 700, and 10,000 ppm (weight) hydrogen, were sent to Argonne National Laboratory for neutron attenuation measurements. Attenuation taking place as the neutron beam passed through the 10,000 ppm sample indicated that 2500 ppm changes around a 5000 ppm concentration level could be detected. No attenuation of neutrons passing through the 100, 300, and 700 ppm samples could be detected.

Sonic attenuation measurements on 3 x 5 x 1/16 inch plates of Zircaloy-2 containing 0, 1500, 3000, 5000, and 7000 ppm hydrogen by weight were made by the Fuels Preparation Department. The damping was found to increase with increasing hydrogen content. This agrees with results from a different set of samples, which were reported last month.

Ultrasonic attenuation measurements using focused transducers should be less affected by changes in surface contour than measurements using flat transducers. Changes in apparent attenuation, due to interference and scattering caused by slight curvature of flat reflecting surfaces which occurs during application of stress, have thus far prevented positive identification of changes in actual attenuation which may be due to stress. A jig required to apply focused transducers, and thus minimize the effect of surface curvature, is being designed.

Powdered Zircaloy-2 samples (100 and 200 mesh) containing 500, 710, 2200, and 16,400 ppm hydrogen were sealed in evacuated capsules and sent to Varian Associates and the University of Washington for nuclear magnetic resonance studies. Determinations of the spin-spin and spin-lattice time constants, the effect of particle size, and an estimation of the practicality of applying NMR to hydrogen analysis in massive Zircaloy-2 samples were requested.

<u>Heat Transfer Testing</u>

The new dual radiometer designed to compensate for surface emissivity variations improved the heat transfer test results for a fuel element which was used to demonstrate the instrument.

It was necessary to manually take the ratio of signals from the radiometers, since equipment required to electronically record and delay the signals was not yet available. This method was too time consuming to allow demonstration of the dual radiometer system on a number of fuel elements. Results from an aluminum clad uranium fuel element containing 1/2, 3/8, 1/4, and 1/8 inch diameter mica produced defects, showed that the 1/4 inch diameter defect could easily be detected above the noise produced by surface emissivity variations. This particular defect was detected once before using a reflective emissivity compensation scheme. However, the reflective method (first type mentioned in HWIR-68574) requires that the fuel element be held at a particular temperature. The main advantage of the new ratio method is that the fuel element temperature during the test is not critical. It also should be less affected by small differences in surface contour. Thus, it should be more applicable to detection of over-all differences between bond conductances of different fuel elements, as well as local variations in a given fuel element. Tape recording equipment required to record and delay the radiometer signals has been assembled so that the capabilities of the ratio heat transfer test concept can be more fully evaluated.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Work continued on standards and calibration bench-marks aimed at expanding the information obtained to date on variations of ultrasonic response as a function of notch depth. Evaluations of rectangularly masked round transducers were completed. Use of such transducers appears to offer the best compromise in transducer selection. Material for correlation studies was received, along with recordings and data from the tests made in a manufacturer's plant. Fundamental studies of Lamb-wave propagation at V_L showed an anomalous condition existing between experimental results and predicted mathematical behavior. A progress report, HW-72802, "Ultrasonic Testing of Zircaloy Sheath Tubing for Fuel Elements" was issued.

Refinements in the preparation of standards and calibration bench-marks were studied during the fabrication of notches for additional defect depth studies. (The standards fabrication is 80% complete.) Careful observation of the prepared notches by means of direct microscopic observation and replication techniques disclosed the need for close control of the electromachining operation. The greatest difficulty appeared to be in the maintenance of electrode shape. In severe cases, the corners of the rectangularly shaped electrode burn away at too high a rate, creating a semi-circular notch instead of the desired rectangular shape. Normally, the electrode can be expected to make a notch having all radii of a 1.0 to 1.5 mil size. Thicker and longer electrodes are being used to overcome the above difficulty.

One of the most important parameters in establishing a suitable ultrasonic test for fuel element sheath tubing is the selection of a transducer. Flat circular transducers masked with rectangular openings have been found to form good ultrasound beam shapes for the transverse tubing test. The resulting beam shape is better because it is formed into an elongated uniform cross section which present a favorable condition for Lamb-wave generation.

In order to determine the conditions under which superior beam structures are generated a number of rectangular masks were studied using ball bearing reflectors. Though the mathematical model for rectangular beam shapes has not been developed, the theory for flat piston drivers was found to be a sufficiently close approximation. The resulting sound pattern from a masked rectangular beam was found to be predictable by considering that the short and long dimensions function as flat piston drivers having diameters equal to these dimensions. At a given operating frequency this means that the focal point, Y_o^+, (the last peak in the near field) for the short dimension is closer to the transducer than the Y_o^+ for the long dimension. Therefore, if the water path for a tubing test is chosen to be at Y_o^+ for the short dimension, the desirable elongated beam results. At this water path, however, the beam contains the near field lobe structure of the long dimension. Past experience has demonstrated that near field structure tends to cause confusing test results. The ratio of the long and short dimensions determines the extent of the near field effects. By proper choice of this ratio the near field effects of the long dimension can be partially subdued.

Alternately, a more uniform elongated lobe can be obtained by operating the rectangular masked transducer at the Y_o^+ distance for the long dimension. While the elongation is not as great, the near field structure is eliminated. Also the longer axis of elongation is produced by the short dimension of the mask due to beam divergence beyond Y_o^+ for this dimension. Therefore, the long dimension of the mask would be perpendicular to the tube axis for tubing tests at Y_o^+ for the long dimension.

Additional studies will be made dynamically to determine which condition of operation offers the optimum results.

Correlation and evaluation of test results using the ultrasonic test as established by this program are proceeding along two fronts. In one instance, about 250 tubes have been tested at a manufacturer's plant under the prescribed test conditions within the limits of his capability. These tubes have now been received on plant along with the pertinent test data and recordings. The first step in the evaluation of these tubes is to rerun the ultrasonic test using similar standards and optimum test conditions. To do this some modifications in the Hanford equipment are being made to improve the tube feed and drive mechanism. It is desirable that the ultrasonic data be obtained first to avoid effects of surface mars that may arise in handling in subsequent correlation tests (eddy current, fluorescent penetrant, X-ray). In the second part of the correlation program, use is being made of fuel element sheath tubing used routinely at Hanford. Defective tubes encountered in our present inspection scheme have been set aside to be examined with the proposed ultrasonic test. Additional material, in the thinner-wall sizes particularly, will be obtained from other interested sites (ANL & CGE) to round out the tubing available for evaluation.

The continuing fundamental analytical studies have revealed some anomalous conditions. Contrary to previous work, the symmetrical mode frequency equation has been found to have real analytical solutions at a phase velocity, V, equal to the longitudinal velocity, V_L. The asymmetrical mode, as was assumed previously, analytically vanishes at V equal to V_L. Experimentally, the situation was demonstrated to be exactly reversed. That is, the symmetric modes appear to vanish at V_L while asymmetric modes do not. No explanation of this anomaly has been found as yet.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

The third in a series of experiments to measure directly the dry deposition of zinc sulfide tracer material on vegetation was completed at the new experimental area near the old Hanford Townsite. Zinc sulfide was released into the air from a source near ground level and was sampled at a height of 1.5 meters on arcs at distances of 25, 50, 100 and 200 meters downwind. Samples of grass and organic ground surface material were taken from which the areal distribution of the grass stand and the deposition of zinc sulfide per unit area can be determined. Assaying of samples was in progress at month end.

Analyses of the data from 17 of the "30 Series" diffusion experiments at Hanford have yielded some positive results that support our earlier conclusions on the relationship between vertical diffusion, deposition and the stability of the atmosphere. The 17 tests were selected on the basis of quality as determined from containment of the diffusing plume within the sampling grid, lack of contamination from previous experiments and minimal equipment failure. The effect of horizontal dispersion was removed from the analysis by computing the crosswind integrated exposure normalized to a unit emission for each arc distance to 3200 meters. Of the 17 experiments, 5 were moderately stable, 4 slightly stable and 8 unstable as determined from the atmospheric stability ratio in the first 100 feet above the surface. Significant findings in this selected set of experiments follow:

1. The normalized crosswind integrated exposure stratified according to stability ratio with no overlap between stability groups. Thus, every experiment in the moderately stable group had higher values of exposure at a given distance than any of the slightly stable group, which, in turn, were higher than any in the unstable group.

2. The decrease of crosswind integrated exposure with distance cannot accurately be represented by a power function in this

set of data. The added curvature results from vertical diffusion and deposition.

3. Within a given stability group, the decrease of crosswind integrated exposure with distance is similar, but appreciable differences between groups are noted. The unstable experiments show a greater decrease with distance than the stable experiments.

The above observations relate directly to the vertical diffusion and deposition processes. Finding such a consistent set of data is encouraging. Screening of additional data according to the stringent quality criteria was started.

In Air Force-supported work, the Series III diffusion data from Vandenberg Air Force Base were transmitted to the Air Force. Analysis of all the Vandenberg data continued.

In our precipitation scavenging work, another experiment was conducted to investigate the relative scavenging of particulates in a plume by different size raindrops. The artificial rain fell through the zinc sulfide plume at a distance of 200 meters from the source and droplets were collected on the Hanford flat-sheet samplers. First analysis of the data showed that drops 0.4 mm diameter were 1.6 times as efficient at scavenging as 0.82 mm drops, and drops 0.36 mm in diameter were 1.5 times as efficient as 0.64 mm drops. An earlier experiment in natural rain had yielded values of 1.8 and 1.4, respectively. In contrast, the Langmuir theory generally used in washout predictions gives values in the range 0.26 to 0.92, depending upon the density and size of the scavenged material.

Dosimetry

Measurement of the radioactivity in Alaskan Eskimos was completed. Slightly more than 700 people were counted. A first analysis was made of the data and a preliminary report was presented at the Second Symposium on Radioactivity in Man. The principal results were:

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Cs-137 Body Burdens in
Alaskan Eskimo Villages

Village	Number of Subjects	nc Cs-137 Minimum	nc Cs-137 Maximum	nc Cs-137 Average
Point Hope	107	3	119	16
Diomede	12	8	35	22
Barrow	259	8	166	52
Kotzebue	132	17	518	138
Anaktuvuk	52	83	790	421

A total of 68 white people were counted that ate very little native food. Their average Cs-137 body burden was the same as that of the people in Richland. Seventy Eskimo children who had been away at boarding schools for nine months were counted. Their average body burden was about 70% of that of their parents. Fifty-two Eskimos who came from villages near Kotzebue were counted. Though significant statistical uncertainties exist in these data, they tend to indicate different average body burdens in these villages and hence different dietary habits.

Twenty-four-hour urine samples were obtained from 15 Eskimos with Cs-137 body burdens between 36 and 684 nc. A few of the data were suspect because of the difficulty in making the Eskimos understand what was wanted. With little weight given these data, the urinary excretion rate was 0.6% of the body burden per day. In all of the data, the ratio of this relative excretion rate to the urinary excretion rate of potassium was quite constant at 0.46% per gram of potassium in the urine.

An investigation was made of a reported change in calibration of the whole body counter. It was found that the resolution had changed. During studies to determine what caused the change, the change disappeared. Probably a poor connection caused the difficulty.

A method of cutting scintillator crystals was developed in order to continue certain aspects of our study of CsI scintillators at liquid nitrogen temperature.

The success reported last month in getting the Van de Graaff back into operation was illusory. Removing the ion-getter pump had a beneficial effect, however, and the pump has now been completely removed from the system until we learn how to make it behave. Early in the month, the troubles with the accelerator essentially disappeared. We do not know why. The machine is in operation again.

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The neutron double moderator dosimeter was used to measure background dose rates to cosmic ray neutrons at elevations of 400 and 6400 feet.

The gamma ray calorimeter was moved into a room where it should be less subject to temperature fluctuations during the seasons. It was put back into satisfactory operation. An experiment is in progress to measure more accurately the heat loss effect found this summer.

Radiation Instruments

Tests continued on the ionization chamber sensors fabricated off-site for use in the miniature personnel signaling dose meters. Of 44 sensors received, tests of samples to date indicate that about 75% will be satisfactory after minor adjustments; whereas only about 20% of the initial shipment appeared to operate correctly. Because of a difference in the size of the pulse caused by recharge from sensor to sensor, the transistor amplifier used in the dose meters was redesigned to permit correct operation with all detectors. The major problem in adjustment of the sensors for proper operation is that of properly positioning the extremely small fibers. An input shorting circuit also was designed to prevent failure of the input transistor due to rapid charge or discharge of the sensor storage capacitor power supply. The amplifier was changed to permit use of the five volt magnitude, five μs rise-time pulses typically obtained from the new sensors. The modified circuit performed satisfactorily with a pulse generator, and tests are in progress to check all working sensors with the new circuit.

Technical assistance was provided the Radiation Protection Operation regarding the establishment of functional specifications for the signaling dose meters. Further development effort will relate closely to the specifications as established.

All circuit modifications were incorporated into the experimental prototype logarithmic response radiation area monitor. Calibration tests over the first 3 1/2 decades were satisfactorily completed. Meter reading errors were determined to be less than about ± 10% from 1 mr/hr to 4 r/hr. The upper scale tests, to 500 r/hr, will be completed as soon as the Van de Graaff accelerator becomes available. The monitor was operated continuously for three weeks with the probe exposed to a dose rate of about 200 mr/hr; no reading changes occurred.

A solid state circuit was developed and tested which will automatically disconnect the load to a standard storage battery when the battery has reached its normal discharge end-point. The circuit prevents complete battery discharge, and it is applicable to certain radiological instruments and telemetry systems. A descriptive report was written.

Experiments started on a scintillation dose rate meter circuit of an all solid-state type especially suitable for use in portable radiation monitoring instruments. With the use of five transistors, including one in a controlled oscillator circuit to provide high voltage for the phototube, a reasonable logarithmic response from 1 mr/hr to 10 r/hr was obtained in initial tests. Four transistors are used in a dc amplifier configuration with the output used to control the oscillator circuit. In essence, the variation of phototube output current with changing dose rates serves to change the magnitude of the high voltage applied to the phototube. Since the phototube gain is logarithmically related to the applied high voltage over four or more decades of gain, the over-all instrument response is such to provide the desired logarithmic response on the readout meter. Several different circuits are being experimented with and the major problems to be resolved are those of the dependence of circuit performance on the applied battery voltage, typically 9 to 12 VDC, and of the temperature dependence effects. The general approach appears promising.

Investigations were made regarding apparent drift problems of a scintillation probe used for thyroid uptake studies of cattle. Tests showed the problem to result from an inadequate amount of magnetic shielding around the phototube; thus, orientation changes caused output signal variations. Substitution of appropriate shielding solved the problem.

Further circuit developments were completed for use in the experimental Atmospheric Physics portable mast instrument system. A special power supply was designed and tested for use in powering all system neon lamps. The control panel was completed in both design and fabrication, the four printed circuit card files were assembled and framed, and interwiring commenced. The design was completed on a transistor driver circuit used to drive the relay coils, and the circuit performed correctly.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples as received during the month. In addition, three uranium samples were analyzed for Testing Methods Engineering Operation, FPD, in order to provide the sensitivity of analysis required for these samples.

TEST REACTOR OPERATIONS

The PCTR was operated intermittently during the month with one unscheduled shutdown due to electronic failure.

Measurements were made during the month to aid in the design of the proposed High Temperature Lattice Testing Reactor. A set of gadolinium, dysprosium and cadmium foils was irradiated. A molybdenum sulfide heater was tested to determine the reactivity loss due to using this type of heater in the HTLTR.

The PCTR was made available two nights during the month to the University of Washington Graduate Center.

The TTR was operated one week during the month to irradiate gold and lutetium foils for calibration purposes. There were no unscheduled shutdowns.

Part of the month was devoted to minor maintenance items.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

Consultation service was given on meteorological and climatological aspects of 1) low altitude sampling of fallout to CR&D, 2) electrical systems audit to CE&UO and 3) environmental sampling for iodine to RPO.

Meteorological services, viz., weather forecasts, observations and climatological summaries, were provided to plant operations and management personnel on a routine basis.

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	90	86.8
24-Hour General	60	85.1
Special	132	82.6

September averaged a little warmer than normal. Precipitation, although largely confined to one day, slightly exceeded normal.

Instrumentation

Detector assembly mounting modifications were made to one of seven scintillation transistorized beta-gamma continuous air monitors which were designed for use at NPR and were fabricated off-site. Confusion regarding the specifications had led the fabricator into incorrect placement of the detector assembly with the result that filter replacement became a problem.

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The one modified unit, satisfactory to IPD personnel, will be used as a model by IPD for their similar modification of other units.

Advice and assistance was rendered to Security and to Instrument Maintenance, HLO, regarding the special barricade monitoring system. Suggestions were made regarding improvement to the present system and also for replacement of certain portions.

Corrective changes were nearly completed on the garment loading portion of the Automatic Conveyor-Type Alpha, Beta, Gamma Laundry Monitor designed for use at the HAPO Laundry Facility. Changes are being incorporated to permit positive re-positioning of the loading rack relative to the monitor. All other system components, both electronic and mechanical, continue to perform satisfactorily. The master drawing of the control and switching portion was completed, and the general operation and maintenance manual is about 75% completed.

All necessary laboratory tests are essentially completed on the field model continuous coincidence-count alpha air monitor as developed and fabricated for Radiation Protection Operation. Adequate tests, under high radon-thoron background conditions, have been satisfactorily completed, and an alarm-point sensitivity for airborne Pu^{239} has been established for a 1 MPC level (2×10^{-12} $\mu c/cc$) continuous for about 15 hours; thus, the equivalent sensitivity for airborne Pu^{239} is 15 MPC-hours under the highest radon-thoron background conditions. The final report was started.

A special transistorized conductivity monitor, developed to provide suitable alarming for powdered plutonium buildup, was completed, tested, and was delivered to Finished Products Chemical Technology, CPD. Personnel of FPCT will install the instrument. The instrument, although specifically developed for the Hood Nine problem, can also be easily used as a liquid level monitor.

The two-section alarm system, designed for use with the scintillation transistorized Columbia River Monitor which was developed for RPO, was completed, tested, and installed. All portions are working satisfactorily.

The five scintillation, transistorized beta-gamma continuous air monitors, designed for use in 327 Building Radiometallurgy Laboratory, were adjusted to provide alarming in about 32 minutes for continuous airborne beta-gamma concentration of 1×10^{-9} $\mu c/cc$. Performance of the instruments has continued to be satisfactory except for occasional minor troubles with the commercial chart recorders used with the instruments.

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All laboratory tests were completed on the prototype coincidence-count alpha air filter counter, which was designed for Radiation Protection Operation to provide immediate counting of standard HAPO 4 inch x 8 inch air filters without waiting for radon-thoron decay. The instrument can detect on the filter Pu^{239} equivalent to a continuous air concentration of 1 MPC (2×10^{-12} μc/cc) existing for 24 hours with a flow rate through the filter of 10 CFM. This amount can be detected immediately after filter removal from the building air system even though the radon-thoron concentration had exceeded 1×10^{-10} μc/cc (a high HAPO level) during the sampling time. The instrument is now in satisfactory operation in the 308 Building. A complete report is being prepared.

Calibration of micro-displacement readout systems to be used by Physical Metallurgy Operation for in-reactor creep measurements continued. The basic calibration of the transducer system used in the third generation creep capsules has been completed. The basic calibration was actually performed twice using two different models of D-C, D-C translators in order to determine the interchangeability of the various models. Further calibrations are now in progress to determine the effect, on linearity, reproducibility, etc., of varying the zero control settings on the translators. The IBM 7090 program for analyzing the calibration data was received in its final operational form this month. Initial runs indicated that the precision of the reference system has been slightly improved and is now of the order of ± 0.000010 micro-inches for two of the instruments and ± 0.000030 micro-inches for the third. Since this computer program is expected to find other usage in similar calibration problems around the plant, a detailed set of instructions and a duplicate program deck will be made available to interested groups.

Chemical Effluents Technology Operation requested engineering assistance in constructing a passive resistor network that will simulate ground water conditions in the Hanford area. A 5000-node, three-dimensional network simulation will be built with capability for expansion to 15,000 nodes at a later time. An automatic readout system with IBM compatibility is to be used. At the present time, the project is being studied for initial recommendations.

Components are being procured for a tape-punch readout device for a tensile-strength testing machine for Physical Metallurgy.

Design of the 100-KW Radiation Effects Facility creep capsule solid state scanner-programmer is 80% complete. Testing of mercury wetted relays and specifying of the solid state digital equipment to be used in the scanner will be completed this month. The Minneapolis-Honeywell data logger to be used in the expansion capsule tests will be shipped the first week of

October. During checks of the system at the vendor's plant, it was noted that the system did not have the versatility or flexibility required of test facility equipment to avoid early obsolescence. The feasibility and cost of adding a program board to the system were investigated. The findings were favorable and a program board for the expansion capsule data logger will be ordered separately and installed at HAPO.

A routine preventative maintenance schedule was written for the Radiation Effects Facility. It consists of log sheets describing the tasks to be performed and space for initials upon completion. It is expected that this procedure will definitely increase the operating reliability of the test facility instrumentation. Control problems are still prevalent in the expansion capsule temperature control system. Installation of all-electronic precision temperature controllers to replace the electro-mechanical recorder-controllers is expected to improve control. However, the basic problem lies in the non-linearity of the closed loop transfer function coupled with the large perturbations thrown into the system during reactor transients. A non-linear compensator has been developed which will compensate for the non-linear closed loop gain as determined from qualitative data. Upon completion of laboratory tests, the compensator will be installed in the control loop. It is anticipated that this device may eliminate the control problems.

Optics

Some work was done on the design of an operational unit of an electrical readout process tube traverse mechanism for NPR and a similar unit for the old reactors. This work was stopped temporarily pending receipt of work orders, but has now been resumed on a unit for the old reactors.

Data taken at 105-D reactor using the 12-inch model optical readout traverse mechanism was analyzed. The indicated displacements of the process tubes looked very good although we have nothing to compare them with since no mercury manometer traverse was run.

An achromatic high power lens was designed for the Purex crane periscopes. This lens uses non-darkening glass and will replace the lenses which are now used.

At the request of Testing Methods, FPD, the problem of photographing the inner bore of tubular fuel elements was considered. A work order was received to design and build such a camera.

The fabrication of the elliptical mirror to be used as a 30 power objective of a high temperature microscope has become a very challenging problem.

The process involves selective grinding and polishing in five zones of different radii and blending the zones at final polish. We have brought the foci of the five zones to within three percent of their final value and expect to get to within 0.5%.

During the three-week period (September 2-September 23) included in this report a total of 252 man-hours shop work was performed. The work included:

1. Fabrication of several profilometer probes for FPD.
2. Repair of two crane periscope heads for Redox and two for Purex.
3. Fabrication of ten glass bearings for CPD.
4. Repair of the PRTR Fuel Exam Facility Borescope.
5. Repair of the PRTR basin periscope.
6. Complete overhaul of the large Lenox borescope for Irradiation Testing Operation, IPD.
7. Fabrication of a lens adapter for Applied Physics, HLO.
8. Fabrication of a quartz dilatometer standard for Radiometallurgy Operation.

Physical Testing

Requests for testing service remained steady with a decided trend toward jobs which require several complementary tests. A total of 6,142 tests were made on 5,757 items representing some 50,293 feet of material. As in the past months, tubular components continue to dominate the type of material tested. Twenty-seven different components were serviced this month; these represent all of the HAPO operating departments and service organizations and other AEC contractors. Advice was given on forty-five different occasions on general testing theory and applications. FPD was supported in making arrangements for a course on vibration and balancing measurements through an instrument manufacturer which seven personnel attended.

Return of the tube shop facilities at White Bluffs to HLO and the associated problems of equipment control, maintenance, and landlord responsibilities have been largely resolved.

A number of aluminum ribbed process tubes were tested at White Bluffs to determine uniformity of wall thickness down the length of the tube. An ultrasonic resonance method was used to examine the tubes at four angular locations at one-foot intervals. Both uniform-wall tubes and thin-wall tubes (thinner between the ribs) were represented in the work. By means of the ultrasonic thickness measurements it was possible to confirm IPD's suspicions that the tubes were not as uniform as previously thought. In many cases, a decided non-uniformity in the form of a thin-wall was encountered, usually 90° displaced from the ribs.

Field testing activity proceeded routinely and in one area, advantage was taken of recently developed capabilities in vibration and motion testing. In response to a request for a study of ground vibration conditions in the vicinity of the 234-5 Building, seismographic readings were taken. CPD was interested in potential sites for location of a secondary standards laboratory having the requirement of satisfactory ground stability to avoid unnecessary "noise". With the assistance of Geophysics, instrumentation was assembled and measurements taken. A geophone was used with a dynograph for recording purposes.

Six NPR Parker Ridglok connectors were cut, mounted, polished and photographed for evaluation of the seal fitting. Two of the six fittings had been made up 15 and 20 times, respectively, while the other four had been made up only once. The photographs displayed the side slip of the ferrule, cracks in the bottom of the ferrule and the seal fit. With this information, the customer was able to make a suitable decision as to acceptability of the fittings.

Tests were made on a transportation pool car rear axle which had failed in service. Magnetic particle test, hardness tests, and metallographic examination were used. Graphs were plotted illustrating the hardness changes in the longitudinal and transverse axis and circumferentially around the axle at different locations near the failed area. Samples were cut and micro polished, etched and examined. No evidence was found of material defects that could have initiated the fracture. A stress riser due to a change in hardness in the longitudinal axis may have been present and combined with a possible mis-match of hub to axle contour could have been a contributing factor to the failure.

Efforts continued on the study of NPR pipe fatigue specimens aimed at developing potential in-service monitoring schemes. The limitations of conventional ultrasonic shear testing were demonstrated in following the progress of cracks generated in fatigue tests made by IPD. Though sensitivity appears to be more than adequate for the initial stages of crack growth, following the penetration of macro cracks will require either two-transducer techniques, or some modification of transducer design. The latter would appear to be the most promising in retaining simplicity of application. Failures occurring in the base metal were also studied metallographically. Samples cut from failed areas were polished, etched, examined for decarburization and photographed. Micro-hardness traverses were made to determine the depth of decarburization. Metallographic examination indicated the failures initiated from small areas of corrosion which

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appeared to be present in the pipe prior to fatigue testing. This conclusion was warranted on the basis that the decarburization layer was of a uniform depth with the spot corrosion contained within the decarburized layer.

Analog Computer Facility Operation

The major problems considered during the month were:

1. Analog solution of the diffusion equation.
2. Pot calciner.
3. Dynamic optimization.

Eighty percent of the GEDA equipment and ninety-six percent of the EASE equipment were in good operating condition during the month.

Computer utilization was as follows:

GEDA	EASE	
98	91	Hours Up
31	31	Hours Scheduled Downtime
0	0	Hours Unscheduled Downtime
7	14	Hours Idle
136	136	Hours Total

Over-all maintenance has been good. Most of the trouble appearing on the computers has been traced to the problems being solved, and not to the equipment being used.

Instrument Evaluation

The Model II Scintrans (of 65 fabricated in Seattle) which are now in general HAPO field use have been performing satisfactorily. Only a few minor changes were necessary to optimize the performance. The only field condition requiring maintenance is an occasional pin-hole light leak in the scintillation alpha probe covering.

Two on-site fabricated scintillation transistorized alpha hand counters successfully passed thorough evaluation testing and were delivered to the field for routine use. Signal to background count ratios of about 3 to 1 were obtained for the HAPO standard, distributed 500 d/m Pu^{239} sources. The instruments also have both alpha and beta-gamma probes which are cable-

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connected to the instrument for use as clothing and shoe checking purposes.

All tracings for the alpha-only and for the combined alpha-beta-gamma hand and shoe scintillation transistorized monitors are in the process of being updated in order to provide the latest circuits and mechanical features.

Except for the on-site fabrication of the required miniature BF_3 proportional counter tubes, the thirty transistorized portable neutron monitors, which were fabricated off-site in Seattle, are ready for use. One unit, complete with miniature detector and all drawings, was sent to SRL by RPO personnel. Comments from RPO indicate that SRL is interested in obtaining a number of the instruments. With use of the small counter tube and an appropriate moderator for fast neutrons, the instrument provides thermal-slow neutron ranges of 2 mrem/hr, 20 mrem/hr, and 200 mrem/hr and fast neutron ranges of 50, 500, and 5000 mrem/hr. These ranges match the stated requirements received from IPD.

One prototype scintillation transistorized portable alpha monitor (poppy) is being slightly modified to permit incorporation of alkaline batteries in place of the originally-planned mercury batteries. This change will permit instrument operation to well below $0^{o}F$ although the weight will be increased slightly.

Evaluation tests continued on the transistorized, line-operated alpha monitor as designed for use with HAPO air proportional type alpha probes. It appears that some input amplifier circuit changes will have to be made to secure a better counting efficiency and lower background counting rate.

A further test was conducted on the Nuclear Measurements Corporation Model GA-2A scintillation, vacuum-tubed radiation area monitor. The detector was exposed to a dose rate of 1×10^5 r/hr for one hour. The meter correctly stayed solidly off scale during the exposure; however, a calibration check performed immediately after the exposure showed calibration reading errors from -40% to +300% over the three-decade range from 5 mr/hr to 4 r/hr. The instrument was recalibrated to provide the usual -15% to +15% reading errors. It appears that the phototube performance was permanently degraded by the high exposure although calibration could still be achieved.

Evaluation tests continued to be satisfactory on the experimental transistorized portable G.M. monitor which employs rechargeable batteries to power the circuits. Tests indicate that a fully charged battery will operate the instrument for 70 to 80 hours continuously.

F. G. Dawson for P. F. Gast
Manager
PHYSICS AND INSTRUMENT RESEARCH
AND DEVELOPMENT

PF Gast:mcb

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CHEMICAL RESEARCH AND DEVELOPMENT OPERATION

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Ground Water Temperature Studies

A ground water mound beneath 100-B Area was found to have the same general shape as the isotherms established by temperature measurements. The highest part of the mound, about 10 feet above the normal water table, is located within the area enclosed by the 80 C isotherm. In the north part of the area the mound is caused primarily by the leakage of reactor coolant effluent into the ground beneath the retention basins. From the retention basins to the south the mound is created by leakage from reactor coolant effluent lines leading to the basins. It was calculated that approximately 1.5 million cubic feet of reactor coolant effluent per day enter the Columbia River from sources within the 100-B Area.

Treatment of NPR Decontamination Wastes

Additional chemical and physical properties of simulated NPR decontamination wastes were determined.

Studies on scavenging Co-60 from NPR citrate decontamination waste containing triethanolamine continued. An iron concentration of 200 ppm was found to be sufficient to precipitate 95 percent of the Co-60 from the neutralized waste at 85 C. Excess caustic reduces the scavenging slightly. The presence of a silicone antifoaming agent had no detectable effect on Co-60 scavenging.

Effluent Monitoring

The radioiodine correction sequence for the As-76 monitor was checked on eight random samples during the month. The ratio of correction versus error was found to be 1.43 ± 8 percent (standard deviation). When extracted radioiodine fractions were counted in the monitor, zero values were obtained. These results indicate that the correction being made is correct and reasonably stable for samples measured at reactor equilibrium. Cross check of results indicated good agreement between Purex laboratory and the

As-76 monitor. Decay measurements indicate that significant amounts of interfering isotopes are not present in extracted samples.

Iodine monitoring was uneventful during the month. Satisfactory operation is being obtained from the recycling of the carbon tetrachloride through the system.

Reactor Studies

In several of the in-reactor silicate addition tests it was not possible to adjust the pH of the water after addition of silicate to bring the pH to that of the reactor as a whole. This resulted in an increased pH from 6.6 to 7.1. In order to determine if this pH change caused any of the effects observed, sodium hydroxide solution was added to a reactor tube to maintain a pH of 7.0 for three weeks and then increased to maintain a pH of 7.5 for two weeks. The change in pH to 7.0 produced no significant change in most of the effluent radioisotope concentrations, including As-76 and P-32, but did cause an increase of about 50 percent in the Np-239 concentration. It also caused a reduction of the Cr-51 concentration by a factor of 3. These results indicate that the silicate itself and not the accompanying pH change was responsible for the more important reductions observed in the reactor effluent concentrations.

A 23-day operating period was completed during which an uninterrupted flow of deionized water was supplied to a new aluminum tube with new aluminum-clad fuel elements. Compared with process water-supplied reactor tubes the radioisotope concentrations produced using deionized water were 10 percent of the normal for P-32, Np-239, Cu-64 and Sc-46; 15 percent of normal for Mn-56; and 30 percent of normal for As-76. In addition, the Cr-51 was only 2 percent of normal since it was not added as a corrosion inhibitor. The Na-24 concentration was twice normal. Tests are under way to determine if this increase is due to residual sodium from the ion exchange process or from the n,α reaction on aluminum. If the Na-24 comes from the aluminum it would indicate an increased corrosion rate.

Promising results were obtained in the laboratory through the use of mixed iron and aluminum hydroxide coagulation treatment of raw water. By using 2 ppm Fe $(NO_3)_3 \cdot 9H_2O$ and 14 ppm alum to adjust the zeta potential of the floc to 0 to 1 mv, 96 percent removal of added phosphate was obtained. During a second test the alum feed was reduced to provide a zeta potential of 0 to -2 mv and an improved phosphate removal of 97.7 percent was obtained. [redacted] both tests a very low rate of filter head loss occurred, about one-half the rate of loss when alum alone was used.

FUEL PREPARATION PROCESSES

Electrodeposition of Nickel on Uranium

Preliminary results have been obtained in a program directed at the development of electrochemical techniques for characterizing uranium metal surfaces in terms of their suitability to receive an electrodeposited nickel coating. In work on a base-line procedure for reproducibly yielding a satisfactory surface, difficulty was encountered in the use of either a chemical etch or an anodic etch alone. With the particular uranium samples used, neither etch yielded a surface which could be successfully plated from a Thompson bath. A surface preparation treatment which combined anodization and a chemical etch, however, produced a uniformly etched surface of excellent appearance and good platability in a Thompson bath.

Scouting studies of the electrochemical characterization of uranium surfaces were started with measurements of the initial EMF of uranium metal, versus a saturated calomel electrode, upon immersion in 0.1 M $KClO_4$. The results obtained in air-saturated potassium permanganate offer hope of this technique becoming quite useful, since significant differences in potential were noted with different uranium surface treatments. One marked trend was that with a brief nitric acid etch just prior to the potential measurement, an increase in nitric acid concentration through the range from 8 M to 13 M was accompanied by a corresponding increase in the surface activity (increase in negative potential). The relationships between nitric acid concentration, EMF, and suitability for plating will be studied.

SEPARATIONS PROCESSES

Denitration of Purex 1WW with Formaldehyde

The analytical procedure used to determine residual formaldehyde in the formaldehyde-treated waste was found to be unsatisfactory. Based on results from a new method, only about 0.7 percent of the formaldehyde escaped to the product without reaction as compared to two or three percent reported last month (HW-74813 C) for similar operating conditions.

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In an equilibrium run made to obtain additional information on nitric acid destruction and formaldehyde utilization, the nitric acid concentration in the product was 0.97 M and about 2.1 moles of nitrate were destroyed per mole of formaldehyde fed. A feed ratio of 2.67 moles of free acid per mole of formaldehyde was used.

The use of commercially available anti-foam agents to suppress foaming during formaldehyde treatment of Purex 1WW was tested in laboratory-scale equipment simulating the Purex plant equipment. Synthetic 1WW prepared from reagent grade chemicals produced relatively little foam; however, foaming tendency was induced in the 1WW by refluxing it with 30 percent TBP-Soltrol. The addition of one or two grams of General Electric Antifoam 60 per liter of 1WW containing degraded solvent reduced foaming to about the level obtained with 1WW without degraded solvent. Dow-Corning AF Emulsion at the three grams per liter level reduced foaming significantly. Other antifoaming agents tested, Dow-Corning Antifoam A and B, and General Electric Antifoam 10 and 66, did not reduce foaming.

Disposal to Ground

A laboratory evaluation test of waste discharged to the Hot Semiworks 216-C-6 condensate crib during the initial Sr-90 recovery program was completed. The limiting Sr-90 breakthrough volume was determined to be 0.5 column volumes, about 220,000 gallons. For this specific waste and at the present rate of discharge, 5000 gallons/month, the crib has an estimated remaining life of 2.5 to 3 years.

Assistance was provided Research and Engineering, Chemical Processing Department, in investigating potential techniques for "mining" several feet of plutonium-bearing soil from the bottom of the Recuplex 216-Z-9 crib for possible plutonium recovery operations. The most promising dry mining equipment appears to be either a "slusher" (horizontal, cable-reel operated scraper) or a small, remotely-operated mine tractor with scraper bucket. Both of these equipment pieces are used extensively in conventional mining operations.

WASTE TREATMENT

Removal of Cesium from Purex Waste by Precipitation

Laboratory and hot-cell development of a nickel ferrocyanide scavenging process for the removal of heat-generating cesium-137 from Purex tank supernates (in support of the CPD Waste [redacted] Program) culminated in a highly successful plant test. Over 99.6 percent cesium removal (vice only 95-98 percent required for waste management heat control) was obtained in the 3800 gallon run. The precipitate was successfully metathesized with silver nitrate and the cesium product, which had been decontaminated with respect to sodium by a factor of 910, transferred to the 325-A storage vault for future studies on absorption on inorganic ion exchange media for storage purposes. Significance of the test is twofold: (1) it proves conclusively that plant-size centrifuges can remove the precipitate as effectively as laboratory filters and centrifuges and to the degree required by the Waste Management Program, and (2) capability is demonstrated to produce a low-sodium solution from which cesium could be loaded to very high capacity on inorganic ion exchangers for long term storage or shipment in an improved, compact, high-integrity, across-country shipping cask (to replace the present STT's). Details of the test are given in a report, HW-75051, "Purex Plant Test of the Cesium Nickel Ferrocyanide Processes," by S.J. Beard and W.C. Schmidt.

Tracer-level experiments and a hot-cell run were made on the phosphotungstic acid flowsheet for precipitation removal of cesium from acidic 1WW (as contrasted to ferrocyanide treatment of alkaline stored supernates). The 99.5 percent removal obtained in the hot-cell run agreed well with laboratory predictions; however, decontamination from other fission products was poorer than expected. Tracer-level experiments showed little difference between a technical grade phosphotungstic acid, which it is proposed to use in a Purex plant test, and CP reagent. The technical material appears quite adequate for plant use.

Economic studies by the CPD Advance Process Development Operation show that the major essential material costs for precipitation processing are for silver nitrate and phosphotungstic acid in the alkaline and acidic processes, respectively. Alternate ways to metathesize ferrocyanides and ferricyanides (which precipitate cesium from acidic solution and could be substituted for phosphotungstate) are accordingly being sought.

Cesium Solvent Extraction

Oak Ridge has recently reported upon a new phenol-type cesium extractant (4-sec-butyl-2, α-methyl benzyl phenol, abbreviated BAMBP) which gives substantially higher cesium extraction coefficients than the best previous phenol extractant, o-phenyl

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phenol. Experiments performed during the month verified the ORNL results and extended the evaluation of BAMBP to waste compositions peculiar to the Hanford Waste Management Program. BAMBP was found to be very comparable to dipicrylamine in cesium extractability, selectivity and ease of stripping. A major potential advantage is that the density of the solvent (about 0.9) would be always less than that of the aqueous feed, scrub and strip streams, eliminating any possible problems of phase inversion in flowsheet design. The only apparent drawback is the requirement of a somewhat higher pH than with DPA. (Modification of the molecule to increase ionization of the phenol group or use of a different diluent may well extend the effective pH range.)

Using a 1 M solution of BAMBP in the ORNL recommended diluent, diisopropyl benzene (flash point 170 F), cesium extraction coefficients (E_a^o) of about 18 were obtained from synthetic 103-A supernate (9 M Na) to which 0.2 mole/liter of additional sodium hydroxide had been added to bring the pH to > 11. Without NaOH addition, E_a^o was only 0.05. High extraction coefficients (up to 80) were also obtained with caustic-adjusted, tartrate-complexed synthetic 1965 Salt Waste (effluent from D2EHPA recovery of Sr and rare earths). Extracted cesium was readily stripped with 0.05 to 0.5 M HNO_3, and separation from sodium was very nearly complete (single stage Na DF's of ~ 3000). Hot-cell experiments with actual 103-A supernate and Purex 1WW showed no detectable extraction of any fission product other than cesium and no appreciable solvent decomposition, at least in two weeks of contact. Soltrol-170, which is used as diluent for D2EHPA in strontium purification, was found to be an even more effective diluent for BAMBP, yielding cesium extraction coefficients two- to three-fold higher than diisopropyl benzene. Soltrol diluent would permit use either of a lower concentration of BAMBP or a somewhat lower pH. Laboratory and hot-cell experimentation with BAMBP and related compounds is continuing, and an adequate quantity of BAMBP has been ordered for 321 Building cold semiworks tests. The significance of the discovery of BAMBP is that two practical cesium solvents are now available for potential plant use.

In-Tank Solidification

Results of measuring the residence time distribution function of the in-tank solidification model indicate the extent of mixing is more nearly complete than is predicted by the ideal-fluid model. The same conclusion is reached qualitatively from data obtained from grab samples taken from twelve positions throughout the tank following introduction of a step-function input. These data show

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that flow to the periphery of the tank is very pronounced. [redacted]rmore, large eddy currents are induced in between the regions of pronounced flow. These results indicate the flow region is turbulent despite the low velocities. If it is sufficiently turbulent, the flow pattern will be essentially independent of the Reynolds number. "Mixing time" or the degree of homogeneity of the model and prototype can then be best compared on the basis of equal rate of momentum transfer per unit mass of the two systems. Rate of momentum transfer is determined by the circulation rate and velocity of issuing stream and consequently can be varied by changing circulation dimensions of the model. Thus, one can design the model to give both the correct rate of momentum transfer at the circulator and the correct Froude number throughout the rest of the tank. The present model does this within the probable limits of accuracy of the concept.

Leaching of Purex Stored Waste Sludge

A sample of hard sludge from the Purex 103-A waste tank was leached with water, then citric acid and finally aqua regia. About 80 percent of the cesium and 5 percent of the antimony (Sb-125) were removed in the water leach. Maximum concentration of citric acid used was one molar. About 82 percent of the strontium was removed by the citric acid leaching. The citric acid leach also removed 15 percent of the iron, 90 percent of the Zr-Nb-95, 50 percent of the Ce-144, 70 percent of the Sb-125, 20 percent of the Ru-106 and most of the remaining Cs-137.

Fission Product and Waste Packaging

Installation and shakedown of equipment for studying hot-gas throuth-drying of synthetic inorganic ion exchange media was completed. The equipment includes a system for preparing up to 30 SCFM of air at 800 F dry bulb and -120 F wet bulb temperatures, a zeolite bed 8 inches in diameter and 36 inches high, and provision for close temperature and pressure control of the drying cycle. Gas humidity instrumentation is currently being installed.

In an initial run with 1/16-inch pelletized zeolite, an air purge at 25 SCFM was successfully used to remove 90 to 95 percent of the interstitial water in about two minutes. A 20 SCFM purge with dry air at 800 to 850 F removed the majority of the bound water in about 3-1/2 hours. Degree of dryness achieved was not determined in this run.

A sample of Linde 13-X synthetic zeolite was loaded to 1.96 meq/g dry resin (about 50 percent of theoretical exchange capacity) with

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inert cesium. Portions of this loaded zeolite were then heated to 800 C with varying proportions of LiF, SiO_2 and B_2O_3. Weight fraction of zeolite in the mixtures was one-fourth or one-half. Several of these combinations appeared to form true melts and one formed a transparent glass on cooling. This latter contained zeolite, LiF, SiO_2 and B_2O_3 in the proportions 1:1/2:1/2:2 by weight. Cesium-loaded 13-X traced with Cs-137 has been prepared and will be incorporated into melts similar to these for leaching studies.

Cesium Recovery by Ion Exchange

Equilibrium distribution coefficients (K_d's) have been determined for several isotopes in two synthetic waste solutions with Linde AW-400 synthetic zeolite. These data are shown in the following table:

Equilibrium K_d's for Several Isotopes with Linde AW-400

	K_d, $\frac{\text{c/m/g sorbent}}{\text{c/m/ml solution}}$				
Waste	Cs	Sr	Ce	Zr-Nb	Na
103-A Tank Supernate	91	1.6	4.4	1.8	1.5
"1965 FTW" (adjusted to pH 3.5)	313	1.2	0.6	0.4	2.5

Column volumes to 50 percent breakthrough can be estimated by multiplying the K_d by the bed bulk density of about 0.75 g/ml. As shown, AW-400 is highly selective for cesium. K_d's were also determined for cesium in FTW at various pH levels; the date indicated that cesium values were constant down to pH 2.4, but the K_d decreased significantly at lower pH levels. The data for zirconium-niobium may be questionable due to the use of tracers rather than actual waste zirconium-niobium.

A column of AW-400 has been recycled through a cesium recovery flow sheet with synthetic 103-A supernate waste five times with no measurable loss of capacity. The recycling is continuing.

Optimum concentration of oxalic acid for sodium and zirconium-niobium removal from clinoptilolite has been studied. Dilute solutions are more effective per mole of acid for sodium removal. Thus, for 90 percent sodium removal 23, 14, 9 and 8 column volumes were required for 0.05, 0.1, 0.25 and 0.5 M solutions, respectively. An alternate solution for sodium removal in place of nitric or oxalic acids has been found to be 0.1 M to 0.2 M $(NH_4)_2CO_3$ at room temperature. Cesium losses were the same or less than those

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with acids, and sodium was removed to greater than 90 pe[redacted] 7 and 4 column volumes for 0.1 and 0.2 M $(NH_4)_2CO_3$ solutions, respectively. Less tailing was also observed than with the acids. However, where zirconium-niobium removal is needed, the dilute ammonium carbonate solutions would not be effective. The most useful application would probably be for aged, alkaline supernate wastes.

Strontium Packaging

The results of laboratory column experiments indicate a variation in the strontium loading rate between different lots of Linde 4A synthetic zeolite. Strontium breakthrough curves for 4A received recently were not as steep as the curves for material received three years ago. The same particle size, temperature, base ion, flow rate, and feed solution were used in these experiments. A difference in color was noted between the two lots of 4A; this may be caused by different clay binders. Information concerning possible variations in the manufacture of 4A and other zeolites has been requested from the vendor.

Observations in the laboratory and pilot plant indicate that 4A does not wet readily. The dry 4A will evolve gas for several days or weeks when submerged in water. The gas is believed to be air from the interstices of the bonded 4A. A 15 minute vacuum treatment on submerged 4A generally removes sufficient gas so that few, if any, bubbles form during column operation.

TRANSURANIC ELEMENT AND FISSION PRODUCT RECOVERY

Recovery of Np and Pu from Purex Waste

Neptunium(V) in simulated 1WW or FTW solutions reduced to Np(IV) when the solutions were made 0.02 M in N_2H_4 and allowed to stand 30 minutes at room temperature. Under the same conditions Pu(IV) was not reduced to Pu(III) in 1WW but significant reduction occurred in FTW presumably due to the lower hydrogen and nitrate ion concentrations.

Two techniques for extraction of both neptunium and plutonium from FTW into the same D2EHPA-Soltrol phase were developed. In one, Pu(IV) is extracted first leaving Np(V) in the aqueous phase. Addition of ferrous sulfamate and hydrazine to the aqueous phase reduces neptunium to Np(IV) without serious reduction of the extracted plutonium. Extraction of the Np(IV) occurs on

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recontact of the organic and aqueous. In the second, neptunium is reduced to Np(IV) and plutonium to Pu(III) by addition of hydrazine and ferrous sulfamate to the FTW. Contact with D2EHPA-Soltrol extracts Np(IV). Sodium nitrite is added to the aqueous to oxidize Pu(III) to Pu(IV). Under proper conditions only a small fraction of the extracted neptunium is oxidized to non-extractable Np(V). Recontact of aqueous and organic results in extraction of Pu(IV). Extraction of greater than 90 percent of both neptunium and plutonium from simulated FTW was achieved with both techniques.

Neptunium(IV) was extracted from simulated 1WW with tri-isooctyl amine (TIOA) in xylene as diluent. Relatively high concentrations of the amine are required to obtain satisfactory extraction - 0.2 to 0.5 M TIOA was required to obtain E_a^o's from 10 to 40.

Recovery of Strontium and Rare Earths from Purex FTW

In one flowsheet for extraction of strontium and rare earths by D2EHPA from Purex FTW, citric acid is used to complex and prevent extraction of iron, chromium, nickel and aluminum. Chromium complexes with citrate are formed quite slowly at room temperature. The formation rate is increased with increased temperature; one hour at 60 C after addition of citrate is sufficient to reduce the chromium E_a^o to a very low value (0.00008 versus 0.02 after one hour at 25 C). Other experiments showed the desirability of keeping "free" citrate (excess over that complexed) as low as possible consistent with feed stability. Cerium and europium distribution coefficients are reduced markedly as "free" citrate is increased.

Pulse Column Performance - DPA-NB System

Investigation of the use of 0.02 M dipicrylamine (DPA) to extract cesium from synthetic Purex 1965 FTW (formaldehyde-treated waste) was continued. The FTW was first butted with about 0.1 M citrate in excess of that required to complex the multi-valent cations and adjusted to a pH of 7.3 to 8.5. A nine-foot tall, three-inch diameter glass pulse column was used with an aqueous-to-organic flow ratio of about two. In several of the runs, a dilute acid scrub stream was also added to the column, dividing the column into either a three-foot extraction and six-foot scrub sections or six-foot extraction and three-foot scrub sections.

The results of these runs were highly encouraging. A stable capacity of 540 gph/ft^2 was readily obtainable with H.T.U.'s in the range of 1.0 to 1.3 feet. Cesium waste losses were on the order of 0.5 percent with a six-foot extraction section. There was no evidence of

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gross iron precipitation, although the feed was quite hazy, presumably from colloidal, hydrated iron oxides.

The substitution of citric acid for dilute nitric acid (0.125 - 0.5 M) scrub solution improved decontamination from sodium and potassium without causing any significant cesium reflux.

Stripping studies with 0.5 M HNO_3 demonstrated that the A/O ratio could be reduced to 0.15 without exceeding a one percent cesium loss in the nine-foot column. An A/O ratio of 0.25 was required using 0.125 M HNO_3. These results suggest that a higher Cs/HNO_3 ratio can be obtained in the product by using a more dilute acid.

In conjunction with the DPA-process runs, an attempt was made to operate the column without pulsing it. Air at 0.9 to 1.8 scfh was supplied to the bottom of the three-inch diameter column to provide agitation and the motive power for countercurrent flow of the dispersed phase through the 3/16-inch hole, 23 percent free area nozzle plates. The capacity of the column under these conditions was approximately half that obtained with the pulser (310 gph/ft^2 in the extraction column, 370 in the scrub and strip columns). The H.T.U.'s were surprisingly low--1.1 ft. was obtained in the extraction column (near flooding) and 1.9 in the strip column, using 1.8 scfh of air.

The input of air at modest rates may make it feasible to consider the use of conventional packed columns through the improved swirling action supplied by the air.

<u>Pulse Column Performance - D2EHPA System</u>

Previous results (HW-74522 C) showed that cerium waste losses of one percent or less were obtained readily from a citric acid-complexed feed, using a nine-foot extraction section. However, recent attempts were only marginally successful. At room temperature, only 50 percent of the cerium was extracted. This loss was reduced to seven percent by increasing the operating temperature to 50 C. The reason for the discrepancy is being investigated.

Studies with the ten-foot high 1B column (strontium-cerium partition) show that excellent cerium DF's (660 to greater than 4000) can be obtained using 0.03 M HNO_3 as the strontium stripping reagent.

Cerium stripping with 2 M HNO_3 in the ten-foot 1C column has been satisfactorily accomplished by organic-continuous operation at 50 C. Under these conditions, the cerium waste loss was reduced to 1.7 percent, compared with about 20 percent under previous operating conditions at lower temperatures with an aqueous continuous column. This value is still relatively high from a solvent degradation standpoint and suggests that a subsequent batch wash, perhaps using oxalic acid, may be required for additional solvent cleanup.

Cesium Recovery with Clinoptilolite - Scale-Up

Investigations with 20-50 mesh clinoptilolite in a four-inch diameter bed to remove cesium from Purex plant waste were continued. Experiments during this period were confined to simulated FTW which had a cesium concentration of 1.35×10^{-3} M. It is concluded that a feed flow rate at least one gpm/ft^2 may be employed. In addition, the tendency of the height of the absorption band to increase with flow rate indicates that a bed of given capacity should be made as long as practical.

Data obtained in washing and eluting cesium from a four-inch diameter column support the laboratory data which indicates that improved purity from sodium and potassium is obtained by washing the column with oxalic acid solution.

Elution of the 23-inch bed was accomplished in 8 to 12 column volumes with a mixture of 1 N NH_4OH and 3 N $(NH_4)_2CO_3$ solution at a flow rate up to 2 gpm/ft^2. Typically, the cesium was concentrated more than three-fold. The ammonium carbonate may be subsequently removed and recovered by vaporization and condensation, thus concentrating the product stream prior to final loadout of cesium.

The clinoptilolite displayed good physical stability through six cycles of FTW treatment with no apparent loss of capacity. However, erosion of the mineral was estimated to be about two percent for 60 to 100 column volumes of FTW processed.

Purex Strontium Flowsheet Studies

The disappointing results of B-cell tests of the peroxide-tartrate process (which was designed to separate strontium from total rare earths in fewer steps than the present plant process) were reported last month. Two further runs were made this month, the first at pH 3 (versus 2 previously) and the second (also at pH 3) with hydrazine added. Strontium recovery was excellent in both cases but the decontamination factor from cerium was only about 2. A plant test gave similar results. Extensive supporting laboratory experiments (with synthetic 1WW) have not fully resolved the failure of the process with full-level 1WW, but do suggest two plausible explanations: (1) radiation damage or destruction of the rare-earth complexant (however, hydroxylamine and hydroazine were ineffective when used as chemical protection agents in hot-cell experiments), or (2) catalytic destruction of the complexant by some material present in actual 1WW but absent in the synthetic. It was observed that copper in concentrations greater than 0.0002 M is detrimental, and it may well be that copper or some similar catalyst is present in the plant waste. Continuing experimentation is aimed at finding the cause of the poor performance and also at determining whether peroxide-tartrate generated complexants could be employed, alternately, as a selective wash to remove rare earths from a strontium sulfate precipitate.

Hydroxyacetic Acid Complexing of Iron

The use of hydroxyacetic acid as an inexpensive substitute for tartaric acid in the Purex strontium process has been suggested by an outside vendor and tested successfully with tracer-level synthetic solutions by CPD Process Chemistry Operation who found that iron was effectively complexed and held in solution by hydroxyacetate and that strontium recovery was not adversely affected. B-cell tests with actual 1WW were made at pH 2 and 3. Normal appearing strontium precipitates (and little or no iron precipitation) were observed in both cases. Analyses of supernatants indicated satisfactory precipitation of both cerium and strontium; however, strontium material balances were very low (probably implying failure to completely dissolve the precipitate). An additional run will be made at pH 1, to determine the effect of lower pH and to determine conclusively the fate of the strontium.

Aged Promethium Dose Rate Measurements

Calculations of the gamma dose rate from Pm-147 and Pm-148, and the associated shielding requirements, were reported in the July report (HW-74522 C). These showed that 2-year promethium would require 1.25 inches of lead and 3-year promethium only 0.2 inches of lead to shield a 250 (thermal) watt source to an allowable radiation level of 200 mr/hr at one foot from the source. Similar Sr-90 or Cs-137 sources would require 4.5 inches of lead.

Because of uncertainties in the method of calculation, an experimental measurement was undertaken of the Brehmstrahlung from 25 curies of well-aged ORNL promethium (recovered from ARCO waste). The results showed that the Pm-147 Brehmstrahlung calculations were, if anything, conservative (i.e., somewhat overestimated the required shielding); however, two low abundance gammas were discovered whose energies and intensities indicate the presence of minute amounts of Pm-146. Although the intensity of these photons is very low, the energies are sufficiently high (0.75 and 0.45 Mev) to require some additional shielding for aged Pm-147 sources. Abundance of the 0.75 Mev gamma was 1.7×10^{-7} per Pm-147 decay and of the 0.45 Mev gamma 1.4×10^{-7} per Pm-147 decay, or a total of 3.1×10^{-7}. The latter figure is in excellent agreement with a value of 2.1×10^{-7} calculated from theory. (The predicted fission yield is uncertain by at least a factor of two.) Even allowing for the detected amount of Pm-146, only 0.33 inches of lead would be required to shield a 250 watt promethium source to a radiation level of 200 mr/hr at one foot. Therefore, although the Pm-146 radiation will limit some potential applications of a promethium, for many applications the presence of Pm-146 should make little or no difference.

EQUIPMENT AND MATERIALS

Corrosion of Boron Stainless Steel in HNO_3-HF-$Al(NO_3)_3$ Solutions

Boron-304 stainless steel containing one weight percent boron corroded at rates of less than 0.05 mils/mo when exposed at 50 C to nitric acid solutions ranging from five to 60 w/o HNO_3. Pyrex Raschig rings exposed to 13 M HNO_3-1.0 M HF at room temperature corroded at a rate of about 100 mils/mo. When exposed at room temperature to 2 M HNO_3-1.25 M $Al(NO_3)_3$-0.25 M HF, Pyrex Raschig rings and 1 w/o boron-304 stainless steel corroded at rates less than 0.05 mils/mo.

PROCESS CONTROL DEVELOPMENT

Gradient Control System

A key problem in the development of the Gradient Control System for the new Plutonium Reclamation Facility (PRF) is measurement of plutonium concentration gradient along a given column length. It is planned to measure this gradient by neutron count rates external to the columns. A series of experiments has been completed to determine appropriate geometries and shielding techniques applicable to the PRF. Both vacuum tube and transistorized instrumentation were used in the experiments. The noise pickup problem previously encountered with the transistorized equipment was solved by the use of double shielded cable between the pre-amp and the detector. Measurements were made in both organic and aqueous solutions containing up to 100 gm/liter of plutonium. The geometry of primary interest was two four-inch columns, four feet apart. In this case, 85 cpm/gm/liter was obtained from the low concentration (0-10 gm/liter) column, when the high concentration column contained 100 gm/liter. Background from the 100 gm/liter column was reduced from 80 to 20 cpm by a shield consisting of 3 inches of paraffin and 20 mils of cadmium. It is concluded from these tests that neutron counting provides a suitable means of obtaining the required concentration gradients, although a single BF_3 tube, as used in the tests, may provide insufficient count rates for use with the control system. Means of increasing the count rate such as using multiple tubes or the more sensitive helium-3 type detectors are being evaluated.

Test of Orifice Flow Meter with Linearized Signal

The applicability of an orifice meter in flow control loops which are a part of more complex control systems has been enhanced by the availability of inexpensive special purpose analog computing units capable of extracting the square root of pneumatic signals. A flow metering system composed of an orifice, differential pressure cell, a pneumatic square root converter and recorder were tested to determine the overall linearity of the system. The results show a linearity within 1 percent, from 15 to 100 percent of range. However, below 15 percent performance was erratic due to the zero drift of the square root converter.

C-Column Studies

The experiment design for the next series of C-column runs has been completed. The design consists of 48 experimental runs in which the effects of six variables upon column performance, characterized by waste losses and concentration profile, will be determined. The six variables selected are: (a) capacity, the sum of aqueous and organic flowrates, (b) flow rate ratio, (c) extractant acid concentration, (d) extractant temperature, (e) feed acid concentration, and (f) pulse frequency.

This design neglects the influence of six other variables which can affect the performance of the column, namely: (1) pulse amplitude, (2) feed uranium concentration, (3) temperature of the feed, (4) fixed uranium in the organic phase (such as DBP complexes), (5) cartridge and column geometry, and (6) unknown factors such as "do badders". The elimination of these latter variables for study was justified for the following reasons (numbered as to variable):

1. The initial block of 20 runs performed previously showed that pulse amplitude effects are negligible after all effects due to amplitude-frequency products are removed.

2. The feed uranium concentration effects can be eliminated by using dimensionless variables ($[U]/[U_f]$) over the feed concentration range of interest, 70-95 gms/liter.

3. The temperature of the feed only enters in its effect on the temperature profile of the column; hence, this effect will be determined by varying the extractant temperature.

4. The influence of the non-extractable uranium appears as a reduction of the driving potential for mass transfer and can be factored directly into the model. It will be determined for each run.

5. The influence of these macroscopic effects on the form of the differential mass transfer model should be negligible.

6. No systematic method of measuring this variable is known. However, a study is contemplated by a straightforward extension of the 48 runs, which could investigate the influence of surfactants.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Flowsheet Development - To date, 39 grams of black, crystalline plutonium dioxide have been prepared by precipitation from molten chloride salt solutions. The PuO_2, which is to be incorporated in PuO_2-UO_2 fuel elements for irradiation-testing, which precipitated in seven gram batches from LiCl-rich LiCl-KCl melts at 550 C, using a 45 percent O_2 - 55 percent Cl_2 gas sparge. Prior to precipitation, the melts were dried by treating with carbon and chlorine at 700 C. The PuO_2 product had a tap density of 6.1 g/cc, about 50 percent of theoretical density. Spectrochemical analysis indicated potassium and lithium contents of 1000 and 500 ppm, respectively.

Electrochemistry of Uranium in Molten Chloride Salt Solutions - EMF measurements for the cell reaction

$$UO_2 + Cl_2 = UO_2Cl_2$$

were reported last month for the LiCl-NaCl salt system. In an endeavor to test the effect of changes in chloride activity upon the reaction thermodynamics, this work has been expanded to include the measurement of EMF as a function of NaCl/LiCl ratio at several different temperatures. The results of this study were that the data, when plotted in the form of EMF as a function of mole percent NaCl in the melt, gave a typical "S"-shaped, titration-type curve. The steep part of the curve occurs in the neighborhood of the eutectic, about 25 m/o NaCl - 75 m/o LiCl. At 850 K, the horizontal sections of the curve occur at about 0.50 volts, in the melts containing less than 20 m/o NaCl (low chloride activity), and at about 0.52-0.53 volts, in melts containing more than 30 m/o NaCl (high chloride activity). The implication is that on increasing the NaCl/LiCl ratio in the melt, one is essentially "titrating" between two complex forms of UO_2Cl_2 in the solution.

Cerium Stand-In for Plutonium in UO_2 Deposition - The conditions which are most favorable to the production of stoichiometric UO_2 (dry, oxygen-free melt) are those which suppress the formation of Pu(IV) necessary for PuO_2 deposition. If CeO_2 can be co-deposited with non-stoichiometric UO_2 which can later be hydrogen reduced to give UO_2 with suitable characteristics, PuO_2 may behave similarly. Thus, the behavior of cerium on a semi-engineering scale is under study.

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Electrolyses were carried out with graphite electrodes in 2-liter quartz crucibles at 530 C and 1.5 volts with equimolal LiCl-KCl initially containing 20 w/o uranium and 1 w/o cerium. Cerium decontamination factors of 35 and 25 and oxygen-to-uranium ratios less than 2.01 were obtained with chlorine and hydrogen chloride sparges during electrolysis. An intermittent electrolysis with cycles of one minute on and one minute off was conducted with a chlorine gas sparge. The decontamination factor was no less than for continuous electrolysis under similar conditions, but the deposit surface was much smoother.

Decontamination factors in the range 6 to 10 were obtained with air-chlorine sparges. Duplicate runs with air sparges gave feed-to-deposit decontamination factors for cerium ranging from 1.1 to 2.6, with the lowest values being obtained nearest the air sparge. Oxygen-to-uranium ratios ranged from 2.008 to 2.078. Hydrogen reduction of the UO_2 is planned as part of the product evaluation. Most of the cerium was unaccounted for in material balances, probably due to precipitation.

Induction Heating Controller - The Pyrochemical Test Facility uses a 30 KW, 9600 cps generator for induction heating of a 15 KW load. An independent control system to utilize the excess capacity in a second work station has been devised and laboratory tested. The new system uses a solid state controller to vary the firing angle of silicon controlled rectifiers (SCR's) to maintain the desired temperature of the salt bath. In the original circuit design difficulty was experienced in obtaining firing pulses of sufficient power from the transistor pulse circuit. It was thus necessary to amplify these pulses, and a multi-vibrator using small (one ampere) SCR's was built for this purpose. The multi-vibrator adequately fires the small SCR's which in turn fire the large power-controlling SCR's rated at 70 amperes.

Fission Product Release During Salt Cycle Reprocessing - Oxidation tests of UO_2 fuel to establish reaction times for fission product release studies related to PRTR fuel reprocessing were completed. Bare sintered UO_2 cylinders 1/2-inch diameter by 3/4-inch long were completely oxidized to U_3O_8 in 1-1/2 hours at 400 and 500 C, 23 percent oxidized at 300 C for 2 hours, and 33 percent oxidized at 600 C in 2 hours. The slower oxidation rate at 600 C is a function of oxygen diffusion rates through the adherent U_3O_8 layer prevalent at temperatures of ~ 600 C and above. Three-quarter-inch long sections of canned specimens required 2 hours at 400 C for complete oxidation while lying in a horizontal position. The U_3O_8 primary particle size at 300 C was approximately 2 μ diameter

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compared to the approximately 4 μ diameter primary particles formed at temperatures above 400 C. Also, as the temperature increased the number of larger agglomerates as well as their diameters increased.

Dissolution of Irradiated PRTR Pu-Al Fuel

A five-inch-long segment of PRTR, Pu-Al spike fuel, irradiated to goal exposure (80 MWD/assembly), was declad and dissolved to test proposed Redox flowsheet conditions. Plutonium loss to the Zirflex decladding solutions was 0.043 percent. The declad rod activated readily in 1.0 M HNO_3-0.015 M $Hg(NO_3)_2$ and remained activated (rapid dissolution) when 12 M HNO_3 was added to bring the dissolver solution to 6.0 M HNO_3 in one cut and 4.0 M HNO_3 in another. The presence of fluoride is not necessary for activation at 1.0 M HNO_3. These results parallel results obtained earlier with Pu-Al alloy non-irradiated and irradiated to 60 percent of goal. It appears probable that this activation and dissolution procedure can be used in Redox dissolution of the spike fuels to achieve a dissolution time cycle much shorter than could be achieved with nitric acid only as dissolvent.

Dissolution of PRTR UO_2 and PuO_2-UO_2 Fuels

Segments of non-irradiated Zr-4 clad UO_2 and PuO_2-UO_2 PRTR fuel rods were declad by the Zirflex process. The UO_2 was high density fused oxide; the PuO_2-UO_2 was of the oxide mixture type. Both materials were compacted by swaging. In both cases, the core materials disintegrated to a "mud" during the decladding step. The UO_2 mud was readily soluble in dilute ($\leq$5.0 M) nitric acid; a green-brown residue (not identified but presumed to be PuO_2) remained after prolonged treatment of the PuO_2-UO_2 mud with boiling 5 M HNO_3.

RADIOACTIVE RESIDUE FIXATION

Calcination of Radioactive Wastes

Two tracer-level spray calciner runs and one tracer-level pot calciner run in the A-cell calcination equipment were completed early in the month and were followed by a successful full-level run of each type. The first of the full-level runs was with the radiant-heat spray calciner, operated with continuous melt-down in the solids receiver, and the second was a simple pot calcination run. The Purex 1WW used in the runs, although centrifuged at Purex, contained gross amounts of fine solids which, however, did not cause nozzle plugging or undue difficulty in feed control.

Twenty-seven liters of feed, equivalent to about 0.1 ton of irradiated uranium, were used in each run and resulted in about one liter of solidified, calcined melt in each case. Operation of both calciners and the associated off-gas equipment, instrumentation, etc., was smooth and uneventful, and no detectable radioactive contamination was released to the atmosphere.

Although analytical results are not complete, preliminary indications are that less than one percent of the feed ruthenium was volatilized in the spray run and perhaps 3 or 4 percent in the pot calcination run. Volatilization of other fission products was negligible. Most of the evolved ruthenium was found in the condensate from the condenser which constitutes the first unit in the off-gas train. The remainder was trapped successively in the caustic scrubber, the electrostatic bubble scrubber, silica gel bed, and the Cambridge absolute filters, for an overall decontamination factor in excess of 10^9 (limit of detection). Both pots were capped and set aside for observation of possible pressure build-up. None has been observed to date. Complete details of the runs and the results obtained will be presented in the July-September "Radioactive Residue Fixation Quarterly Report," HW-75290.

Zeolite Properties

Work on equilibrium loading of cation exchangers continued. Cesium, sodium, and hydrogen loading on clinoptilolite was determined at 25 C in a ternary system. Sections in the ternary system with 1, 5, 15, 30, 60 and 80 percent cesium and various ratios of sodium and hydrogen were used to outline equilibrium loading relationships.

Five-pound samples of clinoptilolite, pelletized by three different techniques, were received from Minerals and Chemicals Phillip Corporation. These pellets are stable in boiling water and in boiling 20 percent nitric acid. The cesium kinetics of the clinoptilolite pellets were compared to naturally cemented clinoptilolite by obtaining loading curves on shallow beds of the same grain size range. Cesium loading rates with a 0.1 N CsCl influent ranged from 0.0133 sec^{-1} for the naturally cemented clinoptilolite to 0.0100 sec^{-1} for the clinoptilolite pellets with the slowest cesium loading kinetics. Within the experimental error the above loading rates are not significantly different. The cesium loading kinetics of the various clinoptilolite samples were not significantly affected by the addition of a wetting agent (Dupanol or Aerosol OT).

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

Stream functions were derived for general three-dimensional flow of fluids through heterogeneous porous media. These expressions are a needed step in order to deduce the spatial permeability distribution beneath the project for use in the electrical analog of ground water flow. Also, the stream function will be used to deduce flow paths and travel times from potential measurements later obtained from the analog. The general stream function should also have application in solving unsaturated flow problems.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Environmental Studies

During a recent period of relatively high I-131 emission from the Redox stack a series of air samples were taken using the AEC Beechcraft airplane. Samples were taken using IPC filters and an activated charcoal backing to determine the fraction of the I-131 activity associated with particulate material. A pattern of collecting flights were flown as suggested by Hanford meteorologists so as to sample the plume at distances of 1, 3, 5, 10 and 25 miles from the stack. In addition, flights were made from Othello and from the Hanford ferry to the stack in an effort to fly in the plume for longer periods. Although the efficiency of the IPC filters for removing particulate material is not known exactly for the conditions under which these samples were taken, they are quite efficient and the data showed that near the stack the I-131 is largely gaseous but is more completely associated with particulate material as it moves further from the stack. Laboratory studies on the particulate material show that the I-131 is generally difficult to remove even in dilute acid solutions. This finding may be important in explaining the uptake relationship observed with plant material.

Radiation Chemistry

Erioglaucine dye in aqueous solution was found to exhibit a long-lived electron spin resonance signal following irradiation, indicating the presence of free radicals. The observed signal is only about four times the noise level, but is not present in unirradiated dye. The signal appears to contain five lines.

This experiment will be repeated on highly purified dye when obtained so as to affirm that the source of this signal is due to the erioglaucine. This work is being performed in a search for materials which will produce on irradiation relatively long-lived free radicals which will not rapidly dismutate but which will react with the primary free radicals formed in the water by radiation. This reaction of the compound with the primary free radicals inhibits their reaction with vital biological components and hence "protects" biological systems from the harmful effects of radiation.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Computer techniques were used to predict zinc sulfide deposition for expected experimental conditions for the three tube diameters under investigation. Parameters investigated were flow rates and tube lengths for one particle size distribution.

Experimental equipment was completely assembled in 271-U Building and is being proven for the 1-1/4 inch diameter tube. Five initial runs at low air flow rates were completed to determine the reproducibility of sample weights for two samples simultaneously withdrawn from the distribution chamber and to determine the pressure drop characteristics of the system. Entirely satisfactory performance was noted.

Aerosol Generation and Characterization

The spinning disc generator was operated to obtain control parameters for immediate usage. Primary testing was to determine the effects of ionizer voltage, liquid feed rate, and disc rpm on the generated particle characteristics. It appears that the device will generate adequate quantities of uranine-methylene blue particles in the size range 2 to 10 microns.

An 80 percent EtOH-20 percent H_2O dye solution containing 0.3 g/l methylene blue and 0.6 g/l uranine was found to give nearly uniform particles at disc speeds of 30,000 - 50,000 rpm. Particle size is roughly inversely proportional to disc rpm. At 20,000 rpm and below inhomogeneity resulted, possibly due to instability of the disc rotor. Liquid feed rates from 1 to 3 ml per minute had little effect on the resulting particle size. The sonic jet ionizer had no noticeable effect on the shape of particles generated in the particular cases investigated.

The recently received Royco 210 particle counter was calibrated, and an attempt was made to make size measurements of aerosols produced with the spinning disc generator. Although satisfactory sizing of 1.17 μ polystyrene spheres was achieved, methylene blue-uranine particles of the order of 3 - 6 microns were not correctly sized by the instrument. Calibration with larger size particles appears to be necessary.

W. H. Reas
Manager
Chemical Research and Development

WH Reas:cf

[redacted]

BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

Dr. L. L. Eberhardt was assigned to Radioecology Operation on August 31, 1962.

Dr. Bruce O. Stuart was assigned to Pharmacology on September 17, 1962.

Dr. Harvey A. Ragan was assigned to the Experimental Animal Farm on Sept. 19, 1962.

David O. Wilson hired in as a Biological Scientist in the Plant Nutrition and Microbiology Operation on September 10, 1962.

Dr. Harold V. Koontz, a summer professor from the University of Connecticut, assigned to Plant Nutrition and Microbiology, terminated on Sept. 7, 1962.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

Summary of the growth and mortality of the pooled data for the period 8/28 through 9/30 monitoring with rainbow trout is shown in the table below:

Per cent effluent	Dead	Live	Total	Mortality	Average weight (g)
0	3	235	238	1.3	25.3
1	0	238	238	0	23.6
2	1	237	238	0.4	23.4
4	0	119	119	0	22.4

All treatment groups with the exception of the 4 per cent group were replicated by duplicate troughs with 119 fish each. No effluent effect is apparent at this time.

Columnaris

To obtain data on the incidence of columnaris infection at a time when no disease outbreak exists, fish from the Leavenworth Hatchery were sampled. From 99 sockeye salmon, 12 positive identifications were made. In each case the incidence of infection was very slight with only 1 to 4 colonies present on plates taken from the infected gills.

Samplings from 12 adult river fish taken from the 300 Area showed no infection from columnaris.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY AND TRANSFER OF RADIOACTIVE MATERIALS

Zinc-65

After a single oral administration of 9 μc Zn^{65} to yearling trout, the blood, spleen, liver, and kidney showed a sharp increase in activity within 24 hours and decline thereafter. However, the bone, eye, and gill filament have shown increase in activity with time, and at the end of 21 days post administration, the bone, eye, and gill filament showed 25, 18 and 120 nc/g tissue, respectively.

Strontium-90

The relationship between Sr^{90} in the operculum and body burden in trout can be expressed by $y = 80x + 4.7$ where y equals μc/fish and x equals μc/operculum. This first approximation is based on 64 trout fed Sr^{90} daily for 3 to 14 weeks. Greater reliability may be anticipated when a total of about 150 Sr^{90} trout are completely processed. These results were obtained from fish that were killed in the fish incident last April 30.

Using litter size and birth weight of offspring as a measure of reproductive performance of female miniature swine, no differences were found between controls and animals receiving up to 625 μc Sr^{90}/day. It appears that levels of Sr^{90} which markedly shorten the animal's life span do not restrict the early reproductive performance of the female large animal.

Iodine

The study designed to simulate a single contamination event on forage for dairy cattle was completed. After the cessation of I^{131} feeding, the effective half-life in the thyroid was about five days. The effective half-life of I^{131} in the milk was more complex, with an initial half-life of less than a day, followed by a half-life of about four days.

Reasonable monitoring methods were developed for utilization in quantifying and differentiating I^{125} and I^{131} deposition in the thyroid of sheep. (The double tracer method will be utilized in order to detect the effectiveness of various routes of administration on I^{131} uptake. Simultaneous studies are indicated, since the uptake and effective half-life changes with age and season of the year.)

Cerium

Ten days following intravenous administration of Ce^{144} chloride in a citrate buffer to a two-year-old miniature pig, approximately 40 per cent of the administered dose was found in the skeleton. Highest concentrations of Ce^{144} in skeletal tissues were found in vertebral bodies and the distal end of the ribs. Soft tissues are currently being analyzed.

Promethium

Promethium-147 chloride in a citrate buffer was administered intravenously to two miniature swine approximately two years old. At sacrifice, ten days after administration, approximately 85 per cent of the administered dose still remained in the body. Approximately one-half of the recovered Pm^{147} was found in the skeleton and one-half in the liver, with only about 1 per cent in other soft tissues. Concentrations within the skeleton were rather uniform, excepting vertebral bodies, which were approximately two times the average for skeleton.

Plasma and milk concentrations of Pm^{147} were followed in female sheep for ten days following a single intravenous dose of Pm^{147} nitrate in a citrate buffer. Milk concentrations increased rapidly, with peak concentrations being observed at four to seven hours post-injection. The milk to plasma ratio at seven hours post-administration and later was approximately five.

Neptunium

Efforts continue to describe quantitatively the fatty liver symptoms in rats induced by Np^{237}. Results are not yet available.

Efforts continue to determine the effect of neptunium valence state on absorption and distribution in the rat. No completely satisfactory results are yet available although there is an indication that neptunium secreted in the urine is predominantly tetravalent.

Plutonium

Studies of the comparative toxicity of Pu^{239} and Pu^{238} are somewhat confused by the fact that solutions of the two isotopes may not be strictly comparable since they involve such widely differing concentrations of plutonium. Citrate containing solutions prepared in the same manner result in different depositions of the two isotopes with about twice as much Pu^{238} depositing in the bone as Pu^{239} and correspondingly lower deposition of Pu^{238} in the soft tissues. Dialysis studies will be made in an attempt to solve this problem.

A number of additional substances have been tested for their ability to promote the excretion of plutonium from rats. None of these are as effective as DTPA, but several have appreciably increased excretion rates as compared with untreated control animals. 8-hydroxyquinoline and β-methylumbelliferone, which might be expected to precipitate plutonium in body fluids, both decrease bone deposition and increase liver deposition. 8-hydroxyquinoline increases fecal excretion of plutonium by a factor of 2. The most effective new agent studied was Tiron (1,2-dihydroxybenzene-3,5-disulfonic acid) which reduced bone deposition to 7 per cent of the administered dose (controls 51 per cent). In this respect, it was essentially equivalent in effectiveness to DTPA; however, it left 3.6 per cent of the injected dose in the liver (DTPA-treated animals 0.6 per cent) and resulted in an over-all excretion of only 42 per cent of injected dose as compared with 73 per cent excreted in DTPA-treated animals. While these studies may not lead directly to a new therapeutic agent superior to DTPA, they may direct attention to new types of therapeutic agents or suggest possibilities for combined therapy.

Analyses on the liver, regional lymph nodes, femurs, and urine samples from pigs injected subcutaneously on each foreleg with 5 μc of Pu^{239} reveal that a single IV injection of Na_3Ca DTPA effects a reduced deposition of plutonium in the liver and femur, but no significant reduction in the regional lymph nodes. An animal which had a portion of the plutonium removed surgically four hours after injection evidenced a reduction in the amount of plutonium deposited in the tissues, as did the animals who were surgically treated and either received Na_3Ca DTPA or had a tourniquet applied. It is difficult to evaluate the effects of these treatments because of the variation among the animals in the amount of plutonium surgically excised. The most significant observation was the elevated urinary plutonium content seen in the pigs receiving the DTPA solution.

External monitoring of the sites of subcutaneous injections of 5 uc of Pu^{239} indicate that suffusing the tissue around the site with Na_3Ca DTPA immediately after injection or as long as four hours later, either alone or coupled with an IV injection of the DTPA solution, causes no gross movement of plutonium from the injection site. Analyses of the tissues and urine (now underway) will reveal whether or not more subtle changes are effected.

Approximately 80 per cent of the administered dose of Pu^{239} was recovered from the tissues of a male sheep ten days after a single intravenous dose of Pu^{239} nitrate in a citrate buffer. Forty per cent of the recovered Pu^{239} was found in the skeleton, 58 per cent in the liver, and the remaining two per cent in other soft tissues, chiefly kidney, spleen, and lung. As previously reported, no acute effects on liver function were observed in this animal, as measured by I^{131} labeled rose bengal dye plasma clearance.

Radicactive Particles

Three dogs were exposed to $Ce^{144}O_2$ aerosols to study the acute radiation syndrome. Only one curie Ce^{144}-Pr^{144} was required to produce an aerosol of sufficient concentration to cause deposition of 2 mc in each dog in a half-hour exposure.

Pluronics (polypropyleneglycolethylene oxide polymer) and, to a lesser extent, DTPA increased the rate of excretion of plutonium in dogs following exposure to $Pu^{239}O_2$ aerosols. At 15 days after exposure about 1 per cent of the body burden was excreted in feces and 0.05 per cent in urine with Pluronics treatment compared with 0.1 per cent in feces, and 0.002 per cent in urine without treatment. Since the per cent of the body burden excreted each day was still

Two dogs died and two were sacrificed two and one-half years after exposure to $Pu^{239}O_2$ aerosols. Preliminary tissue distribution data are given in the Table below:

Distribution of Pu^{239} in Dogs

	Lung		Bronchial lymph nodes	
Dog	(μc) Pu^{239}	Per cent of total	(μc) Pu^{239}	Per cent of total
182*	2	85	0.4	15
184*	2	93	0.1	5
158	0.04	50	0.05	50
79*	0.3	-	-	-

*Lymphopenia occurred in these dogs

Cellular Studies

The relationship between metabolic activity and membrane permeability are being studied utilizing various yeast strains. One of the strains carries on only oxidative, another only fermentative, and the third carries on both oxidative and fermentative metabolic activities.

Although cellular potassium content of the three strains was very similar, uptake experiments showed that influx of potassium into the oxidative and fermentative strains was nearly zero, whereas the strain which utilizes glucose either way shows a very high influx.

Initial data show a cellular phosphate content in the oxidative strain which is only about a third of the content for the other two strains.

The effects of X-rays on potassium and phosphate movement through the cell membranes will be studied as soon as the normal movements of these ions are known.

Plant Studies

Literature reports show that benzimedazole enhanced the accumulation of ions by plants. Data obtained in this laboratory do not show such enhancement but instead show a toxicity effect of the benzimedazole. To more fully define the difference between the results obtained in this laboratory and results reported in literature, the effect of benzimedazole on the respiratory system of barley roots was examined. Under the conditions used, this drug did not alter oxygen uptake. Absence of a respiration change makes any effect on ion uptake even more interesting.

Soil from the outdoor plots contaminated nine years ago with strontium-90 and last year with cesium-137 was brought into the laboratory and mixed with a fresh contamination of Sr^{85}. This triply contaminated soil was then placed in pots, planted with beans, and successively cropped for four plantings. The per cent of cesium taken up by the plants was less by a factor of 100 than the strontium. Uptake of all isotopes increased with each harvest. Analysis for the strontium-90 uptake is incomplete, so no comparison can be made between the presently applied strontium and that which was added to the soil nine years ago.

Columbia River Limnology

Plankton counts were made on samples collected from January to June, 1962. Among sixteen common forms enumerated, Asterionella was the dominant genus. Tabellaria, Melosira, and Synedra were also abundant.

Concentrations of potassium and phosphate increased while boron and nitrate decreased in the river.

Mn^{56} and Cu^{64} were the most abundant gamma emitters measured in plankton, being 8.4×10^{-2} and 2.7×10^{-2} $\mu c/g$ wet weight, respectively.

Rattlesnake Springs Limnology

Recording thermographs were installed and placed in operation in the stream and impoundment. Phosphate and nitrate concentrations in water from the impoundment decreased during the past two months. The greater decrease was for phosphate which dropped from 0.04 ppm to 0.003 ppm.

Plant Ecology

Studies of the environmental characteristids associated with different natural plant communities of the Rattlesnake Springs ecological reserve continued. Soil sodium (soluble, extractable, and exchangeable) was considerably higher from greasewood community areas than from sagebrush areas.

Project Chariot

On the basis of samples analyzed to date, Cs^{137} is the dominant gamma emitting isotope in fresh water fish collected during the past summer in arctic Alaska.

Fallout

I^{131} concentrations in California deer thyroids increased steadily to a maximum of 8×10^{-4} $\mu c/g$ wet weight during early September and then decreased gradually during the remainder of the month. Maryland deer thyroids showed a similar pattern but contained about one-half the I^{131} concentrations in California deer. One HAPO deer thyroid obtained near the end of the month contained 1×10^{-2} μc I^{131}/g wet weight, about 15 times the concentration in California deer thyroids collected on the same day.

Radiation Effect on Insects

Pupae of the flour moth Ephestia were exposed to X-radiation. Up to 60 kr did not increase mortality or inhibit life stage development. At higher dosages, there were abnormalities in wing morphology and ecdysis.

HA Kornberg
Manager
BIOLOGY OPERATION

HA Kornberg:es

C. Lectures

a. Papers Presented at Society Meetings and Symposiums

None

b. Off-Site and Local Seminars

L. K. Bustad. July 5 and 6, 1962. Physio-pathological effects of radioiodine. Summer Institute on Radiation Biology, University of Washington, Seattle, Washington.

c. Seminars (Biology)

None

d. Miscellaneous

None

D. Publications

a. Documents (HW)

None

b. Open Literature

Mraz, F. R. 1962. Intestinal absorption of Ca-45 and Sr-85 as affected by the alkaline earths and pH. Proc. Soc. Exptl. Biol. and Med. 110: 273-275.

McClellan, R. O., W. J. Clarke, J. R. McKenney, and L. K. Bustad. 1962. Preliminary observations on the biologic effects of Sr^{90} in miniature swine. Am. J. Vet. Res. 23: 910-912.

Marks, S., L. J. Seigneur, P. L. Hackett, R. J. Morrow, V. G. Horstman, and L. K. Bustad. 1962. Effects of the administration of single doses of iodine-131 to sheep of various ages. Am. J. Vet. Res. 23: 725-730.

APPLIED MATHEMATICS OPERATION
MONTHLY REPORT - SEPTEMBER, 1962

ORGANIZATION AND PERSONNEL

R. L. Buschbom was granted an educational leave of absence to continue his studies toward a master of science degree in biostatistics at the University of California.

OPERATIONS RESEARCH ACTIVITIES

Work continued on the study of HAPO relationships to the Tri-City Area. Contacts were made with Washington State people for the purpose of uncovering and acquiring State data relevant to the research. Not all the State data desired will be available. It will be necessary to bridge the gaps by recourse to local data sources and analytical techniques. To date, certain basic background data have been acquired and analyzed, but key HAPO data have been slow in coming in.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

A prototype linear programming model of the FPD Production Forecasting Process was constructed which incorporates several additional and more flexible characteristics.

The analysis of production test IP-310 data was completed. This test was designed to evaluate the effectiveness of the UT-2 tester in predicting dimensional distortion during irradiation. Although the original post-irradiation measurements were made in early spring, results were not completely available until the present time since the initial data analysis disclosed the desirability of remeasuring the fuel elements. The final analysis was therefore based on the averages of two measurements.

Assistance was provided in analyzing data from a previously designed welding experiment conducted to examine the effects on weld quality for projection fuels of six process variables.

Data from the first experiment designed and conducted to evaluate certain process variables in the hot die sizing cladding process are being analyzed. The intent is to select the set of process variables leading to best overall quality of the jacketed fuel element.

A small amount of data from a pilot plant test aimed at developing a canning cycle for CVII cores was analyzed. Lead preheat and can-sleeve preheat times were the variables under study. The design utilized did not permit a separation of the quadratic effects for these two variables.

Data from another pilot plant test are currently being analyzed. This test was conducted to determine the effects of lead plugging ingot and dingot cores, as well as varying lead preheat time and also time in the duplex furnace, on internal bond quality. The analysis will be completed shortly.

An initial analysis was performed on NPR inner tube warp data in attempting to see if identified process variables significantly affected the amount of warp observed. A more complete study will be made when requested computations are completed.

Irradiation Processing Department

Work continued on the problem of estimating defect frequency and size distributions in connection with welded primary piping for the NPR Project. A mathematical model of the process of distribution of defects and their lengths has been constructed, and a document describing the theory has been partially completed. Estimation of the parameters of the model is now under way.

On September 14 the master conversion listing to permit identification of the classes of work designated by work order numbers was received, along with the new delineation of outage causes for the period January 1, 1961, to the present time. Minor debugging of file maintenance and report programs in compliance with these data was performed. Initial analytical reports based on these data indicate a lack of sufficiently detailed recording of manpower usage to permit accurate model building at this time. Further evaluation of these data will be required before reliable conclusions can be made.

A document was issued presenting the results of an extensive analysis of the measurement error structure associated with C-Basin Profilometer measurements. The conclusion was reached that positioning of the fuel element prior to recording measurements is a major cause of measurement error. A recommendation was made that in single-shot production tests, fuel elements be measured twice to reduce the measurement variance. Further, it was recommended that changes in the criteria used to classify a fuel element by profile type be incorporated in the EDPO Program.

A preliminary investigation has been made of ledge corrosion data taken from the Visual Attributes report for Quality Certification monitor charges. A more detailed analysis of these data is under way. Comparisons will be made between reactors and within reactors over "time".

Assistance was provided in a determination of what amount of dingot material, assumed to be "worse" than ingot by a factor F, could be accepted such that the probability of incurring two or more additional ruptures per month is kept below a specified level.

A complicated expression exists theoretically relating "R" values (a measure of temperature imbalance in a process tube) to fuel element and process tube dimensions. An attempt is being made to find a simpler expression for this which will permit an easier identification to be made of the effect on "R" values of changes in any or all of the dimensions.

Suggestions were given to aid in selecting NPR fuel elements and charging them in the K loops in a manner best designed to obtain a rough evaluation of certain process variables, such as billet type, position in extrusion, and whether or not a given piece has been straightened.

Chemical Processing Department

An analysis was made of the effects of blending metal recovery feed on button and ingot density. Of the variables studied, the aluminum used in complexing the fluoride was found to exert the greatest influence on density.

A comparison was made of three different gauges used to inspect cutting tools. No significant biases between the gauges were found. Indications are that an increase in the optical gauge magnification may lead to greater reproducibility of results.

A comparison was made of an experimental alpha counter with those in current use. This comparison was made over a wide range of counting rates.

Several general regression analyses were preformed in studying the relationships between edge wall thickness and other part measurements.

Minor consultation was provided in answering a query concerned with particle size measurements.

Relations and Occupational Health Operation

Work is being done in preparing the management report conveying results of the recently conducted HAPO Attitude Survey.

Contract and Accounting Operation

Consultation services were afforded personnel concerned with inventory sampling plans to see if their procedures satisfied some of the general directions given in a cost accounting bulletin. The bulletin gave some rather arbitrary sampling percentages. Hanford makes use of a priori knowledge about the number of line items in different price strata and thus affords a more-than-adequate sampling procedure with a smaller percentage of items sampled. The objective of the inventory sampling used here is to minimize the variance of the dollar discrepancy estimate subject to some constraint as to the number of line items sampled. More samples would have little effect on reducing the variance.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HL

2000 Program

Pulse Column Facility

The factorial experiment to study the extraction characteristics of the pulse column as a function of six independent variables, extractant stream temperature, extractant stream acid concentration, feed stream concentration, feed stream and extractant stream flow rates, and the pulsing frequency was completed. The design is a 1/6 replicate of a $3^3 2^3$ factorial with 12 duplicate runs -- a total of 48 pulse column runs. For each run the column is started up at one nominal setting of the six independent variables and the aqueous uranium concentration monitored at a single port for a period of several hours to obtain a measure of the uranium concentration variability under supposedly equilibrium conditions. Following this a profile of the column aqueous and organic concentrations is taken by sampling the two phases at ports ranging over the entire length of the column.

An experiment was designed to estimate the organic zero shift of the gamma absorbtiometer to be used for analyzing the feed stream concentrations. The experiment was conducted and the data are currently being analyzed.

3000 Program

Machining Development

A complete finished exterior contour of a 1251 component has been machined by the magnetic tape controlled experimental δ-ω lathe. This prototype part has been dimensionally gauged and subjected to surface finish measurements, and has been found to be well within the required tolerances.

A tape is now being prepared to machine the corresponding interior contour.

Work is continuing on an EDPM Program which will generate the design data for Floturn blanks which are to be shear-spun into certain preselected complicated shapes. This work represents an extension and generalization of previous programs which were prepared to generate much simpler geometric configurations.

4000 Program

Nondestructive Testing

The detailed mathematical analysis of the propagation of ultrasonic waves in elastic plates continues. A hitherto unreported phenomenon has been discovered which it is hoped will explain some of the puzzling discrepancies that have been observed in experiments with nondestructive testing equipment.

Other

Assistance was given in developing a regression equation for swaging operation variables. The formula has been applied in practice with good results.

Consulting assistance was provided in the sampling of wire which is to be tested for uniformity and character of chemical composition after test reactor irradiation.

Methods used in interpreting round-robin test data from a nested design were provided for use in a forthcoming article in which such data are to be presented.

5000 Program

Actinide Element Research

A report was issued which describes the theory and application of a FORTRAN language program for calculating cubic crystal lattice constants at various temperatures. Work continued on the problem of indexing hexagonal and orthorhombic crystals.

Division of Research

The debugging of the Monte Carlo program for testing program GEM appears to be completed. A number of cases have been run successfully. Several revisions in the input routine for GEM were made during the month.

Program ZERO was reprogrammed and debugged during the month. The new version provides for simpler preparation of the input and has flexibilities not present in the former version. More machine plotting of the program data using PICTURE was done to obtain gross data characteristics.

A paper, "Quantitative Analysis of Sets of Multicomponent Time Dependent Spectra from Decay of Radionuclides", was written jointly with Chemical Research and Development for presentation at the Symposium on Application of Computers to Nuclear and Radiochemistry at Gatlinburg, Tennessee, October 17-19, 1962. Work continued on the preparation of tables and graphs for inclusion in the formal report, "Fixed Time Count Rate Estimation with Background Corrections".

The final report on the search pattern problem was issued as HW-74936, "Search Patterns and Detection Grids".

6000 Program

Biology Operation

Work continued on a problem of confidence interval estimation using multichannel analyzer data from an Alaskan fallout study.

General

Work continued on the analysis of mass spectrometer data on three gas standards. Three independent analyses of approximately 50 runs each are being done using a components of variance model to resolve the total variation into a between run component, individual peak components, and experimental error.

Carl A. Bennett
Manager
Applied Mathematics

CA Bennett:dgl

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Reactor Dimensionality

MELEAGER is a nondimensional burnup code useful in reactor survey work. To extend its usefulness a two-group diffusion code has been written and debugged which will take reactor size into account and consider interactions of the regions of a multiregion reactor. The diffusion code is coupled to MELEAGER and each region (up to 24 regions) is treated as a separate MELEAGER case for burnup analysis. The usual MELEAGER end point controls are bypassed and the burnup continues so long as the diffusion code finds the reactor to be critical regardless of the reactivity of a particular region. Therefore, the diffusion code has been named ALTHAEA, after Meleager's mother in Greek mythology, since the diffusion code has control over MELEAGER's life or death as did Althaea.

Briefly MELEAGER-ALTHAEA is a chain code system which alternates between the two codes. A reactor is described in the MELEAGER input using up to twenty-four successive MELEAGER cases, each representing a region of the reactor. The sequence of events follows:

Step 1. The MELEAGER code computes Westcott cross sections for the region utilizing the thermal flux depression and resonance self-shielding corrections provided by HLO. Thermal and epithermal macroscopic cross sections are derived for the region using the Westcott formulation. These two-group cross sections are saved on tape along with all the other data necessary to continue the MELEAGER burnup for this region.

Step 2. Step 1 is repeated for each region of each reactor described and the ALTHAEA code is called into the computer.

Step 3. Thermal and epithermal fluxes, normalized to the desired over-all power of the reactor, are computed for each region using two-group diffusion equations. The two-group fluxes are used to compute the modified Westcott flux and epithermal index (r) for each region. If the reactor is subcritical, no further computations are made. If control rod action is requested in the input, control poison is added until the reactivity is reduced to 1 and new values of Westcott flux and r are obtained. A complete set of MELEAGER cases describing the reactor are then saved on tape so that MELEAGER can continue the burnup of each region.

Step 4. Step 3 is repeated for each reactor. If no reactors are critical, the next chain computation is called (QUICK, PROTEUS, PLOTTER, etc.). If at least one is critical, the MELEAGER code is called into the computer.

Step 5. MELEAGER computes new Westcott cross sections based on the new value of r and using the new flux value computes burnup for a region. When burnup has proceeded for a time interval given in the input data, new two-group cross sections along with all the other data required to continue the burnup of the region are saved on tape.

Step 6. Step 5 is repeated until all regions of each reactor have been processed. The ALTHAEA code is called into the computer and the sequence is repeated beginning with Step 3 above.

The ALTHAEA code uses the iterative difference equation method to solve the two differential diffusion equations for the two unknown fluxes in the following sequence:

1. Westcott flux and r values from the preceding burnup period or as input guesses are used to compute initial thermal and epithermal fluxes.

2. Source terms for fission neutrons to the epithermal group are computed from the flux and fission cross section as computed by MELEAGER in each group.

3. The epithermal group diffusion equation can now be solved for the epithermal flux.

$$D_1 \nabla^2 \theta_1 - \Sigma_1 \phi_1 - \Sigma_{12} \phi_1 + S = 0 \qquad (1)$$

4. The thermal group diffusion equation can now be solved for the thermal flux.

$$D_2 \nabla^2 \phi_2 - \Sigma_2 \phi_2 + \Sigma_{12} \phi_1 = 0 \qquad (2)$$

5. The fluxes and fission cross sections are used to compute a new value of S, the fission source term in equation (1). The reactivity is the ratio of the present to the previous fission source terms.

6. If any of the new flux values vary from those of the preceding iteration by more than a permissible value given as input, the iteration is continued from Step 3 above.

The Westcott formulation assumes a slowing down density independent of energy which does not exist in any real system since some epithermal neutrons are lost. The value of the flux per unit lethargy interval is less at near thermal energy than at near fission energy. Either value could be used to compute r, the spectral index, but it is probable that the lower value is most appropriate since the low energy resonances are usually most important in the calculation of the Westcott cross section. The lower value is generated if the true value of the slowing down power per unit cell volume is used to compute the removal cross section. If harder spectra are adjudged as more appropriate in some circumstances, a reduced value of slowing down power may be given as input. For example, if it is desired to use an epithermal index typical of flux per unit lethargy interval near fission energy, the slowing down power could be multiplied by the resonance escape probability as a correction factor. In general, a correction factor closer to one should be appropriate.

The ALTHAEA code also requires input values of the transport cross section for the thermal and the epithermal groups in each region. If it is desired to cause a greater spatial variation of the flux (and lesser leakage) than the estimated value of the transport cross section produces, an increased value may be used. Similarly, a reduced value of the transport cross section will reduce flux variation and increase leakage. Like MELEAGER, the ALTHAEA code must be "calibrated" by a more authoritative source.

Once calibrated, it appears that the ALTHAEA code will provide adequate comparisons of alternate fuel cycles involving various reactor region loadings. Examining a seven-region example of the APWR with ALTHAEA indicates that the single region zero dimension approximation of the APWR as evaluated by MELEAGER is a satisfactory approximation. This result is due to the fact that the APWR is a large enough machine to have very little leakage and, in fact, constitutes several critical masses. Consideration of the reactivity of smaller reactors will require use of a code such as ALTHAEA. For large reactors ALTHAEA can add much to a fuel cycle analysis by supplying detailed data with respect to the relative heat generation in each region. The usefulness of a fuel load in a large reactor may be limited by maximum heat transfer and minimum power generation considerations well before a reactivity limit is reached. Usually, moving the fuel about the reactor will suffice to correct the differences, and MELEAGER presumes that this can be satisfactorily accomplished. This is a good assumption for batch and many, but not all, graded fuel cycle loads. For graded cycles, MELEAGER records the variations in heat transfer that would exist if the neutron flux were constant over the reactor. Thus, cycles that merit examination by ALTHAEA can be identified.

Generalized Reactor Parameters Study

A study to determine the effect of certain key reactor design parameters on the value of successively recycled plutonium was begun. Three parameters are considered (1) the moderator index, (2) the parasitic absorption index, and (3) the effective moderator temperature. These parameters effectively characterize the five simulated reactor types in the basic plutonium value study.* The contribution of certain second order variables, such as the estimated plutonium price required by the minimization calculation, must be evaluated, however.

* These are APWR, HWR, BWR, GCR, and OMR. For a full description of the study see HW-72217.

Each experiment will consist of a uranium enriched step and one self-produced plutonium recycle step, each minimized with respect to the total fuel cost so that the value calculated will be a simple indifference value. (That is, the PUVE two-step equilibrium solution.) This type of value is not as accurate as the seven-step value used in the basic study. However, since the reduction in the computer time of a two-step experiment allows a much greater number of experiments and, if the calculated values and fuel costs are interpreted only in comparison with one another, the inaccuracy of the two-step solution can be tolerated.

Some preliminary conclusions obtained from the study are as follows:

1. There is an optimum moderator index for which the fuel cost is a minimum and the plutonium value is a maximum. The optimum point depends on the parasitic absorption index.

2. Increasing the parasitic absorption index invariably increases the fuel cost but has little effect on plutonium value.

3. Varying the moderator temperature has a negligible effect on the fuel cost but may significantly affect the plutonium value. Additional experiments are planned to verify and extend these conclusions.

It is important when noting these conclusions to recognize that the method of solving for these values involves comparison of two plutonium fueled modes with different amounts of U-235 present in each mode. The lack of decisiveness is inherent, of course, because U-238 is the fertile fuel for both modes and, although the first mode begins with U-235 enrichment, 50 percent of the heat may be generated by in situ fission of the plutonium that is grown in. Thus, changing a reactor variable that adversely affects plutonium fuels will increase the cost of both modes and, yet, the indifference value of plutonium for these two modes may be scarcely altered. This is distinctly different from comparing U-235 spike fuel with plutonium spike fuel, as might be done for a burner reactor.

Combined Cycles Studies

Plutonium of three isotopic ratios has been analyzed in three reactor types selected to show the interactions occurring between reactor type and plutonium composition. Uranium from the diffusion cascade has been analyzed in the same reactor types in order to provide a basis of comparison. Three types of cladding (zirconium, stainless steel, and Hastelloy) were used for the uranium and plutonium cases. Two specific powers (15 and 30 megawatts per ton) were analyzed for most of the above combinations of reactors and cladding types. A total of approximately forty optimizations have been carried out with each optimization involving five or six lattice spacings and with each lattice spacing requiring five to eight levels of enrichment in order to obtain minimum fuel costs.

The following preliminary observations are made:

1. Fissile fuel values are enhanced when the correct lattice spacings are chosen to best utilize the specific fuel composition.

2. Mature plutonium has a higher value (based on $/gram fissile) than low exposure (95 percent Pu-239, 5 percent Pu-240) plutonium. This is based on a comparison of low exposure plutonium with plutonium containing as much as ten percent Pu-242.

3. Plutonium is worth at least $10/gram fissile in almost all instances; and is worth considerably more at higher specific power in fuel elements providing geometrical shielding of the low lying resonances of the neutron cross sections, and in reactors having considerable parasitic absorption of thermal neutrons.

4. D_2O moderated reactors can give very low fuel costs if one ignores the capital expense uniquely associated with D_2O. Adding the capital expense, which amounts to as much as one mill/kwh_e makes the D_2O machine have relatively high fuel costs with today's economics and fuel prices.

5. Numerous graded cycle fuel costs were obtained with optimized total fuel costs of less than 1.0 mill/kwh_e at standard conditions (15 MW/T specific power) $30/pound uranium fuel element fabricating and jacketing (FEFJ) charges, 12.5 percent economic interest, 4.75 percent AEC use charges, assumed plutonium price $10.24/gram fissile, $10/pound uranium separations charges, 33.3 percent thermal-to-electrical conversion efficiency, and natural uranium available at $9.47/pound of uranium as a fuel grade UO_2.

6. The sodium-cooled graphite-moderated reactor gave batch loading fuel costs of less than one mill/kwh_e. Initial alpha values of 0.53 are obtained for Pu-239 at optimum lattice spacing and fissile enrichment in this reactor. Alpha values of 0.58 are typical for H_2O moderated reactors at optimum conditions. The lower alpha value for the graphite moderated case is to be expected because of the geometrical shielding of the plutonium resonances afforded by the large fuel lumps. Future work should be carried out to better define this parameter.

7. Increasing the price of source plutonium in plutonium-enriched natural uranium fuels increases the fuel cycle cost up to a point where the fuel cycle minimum costs are obtained with enrichments close to natural uranium. Increasing the fuel price for the plutonium further, then, results in a decrease in fuel costs. Under these conditions more plutonium is being produced than is being consumed.

8. The data at hand will provide insight into the effects of the various economics parameters on optimum lattice spacings and their interactions with fuel type.

9. Adjoint solutions are possible to determine the impact of reactor size on fuel costs and lattice optimization. These solutions are to be obtained from the use of data fitted with polynomials which were stored on magnetic tape from the initial runs. The block of economics data presently being analyzed is for very large reactors where xenon and samarium plus leakage do not exceed 45 milli k.

These and other results will be summarized in a progress report. After completion of this report it is planned to emphasize the obtaining of comparable data for thorium enriched with plutonium of varying compositions, as well as with U-233 and with U-235. Data will also be obtained with U-233 as an enrichment for UO_2 with naturally occurring enrichment.

Phoenix and Reduced Density Fuels

Work on the Phoenix Study, Reduced Density Study, and Tailored Phoenix Study continued during September.

Minimized fuel costs have been obtained for a range of water-to-fuel volume ratios varying from 1.5 to 4.0.

The plutonium was of a composition "rich" in the Pu-240 isotope. Its composition was 45 percent Pu-239, 40 percent Pu-240, 10 percent Pu-241, and 5 percent Pu-242. This was diluted with varying amounts of tails uranium to provide a range of fuel densities from normal oxide density to one-tenth normal oxide density.

Optimum fuel costs were obtained with the lowest moderator-to-fuel ratios and under conditions where the Pu-240 cross section was maximized by using a fuel element of large surface-to-volume ratio. Optimum density was about 40 percent of normal oxide density. This combination of conditions yielded a fuel cost of about 1.35 mills/kwh.

As the moderator-to-fuel ratio increases, the spectrum is softened and the effective Pu-240 cross section decreases. It is then necessary to increase the spatial concentration of the U-238 in order to obtain the most economical amount of fertility for the system. Fuel costs will be higher due to the increased neutron capture in the moderator and U-238, and to less effective utilization of the Pu-240 fertility. Fuel costs for water-to-fuel ratios of about 4 were 1.75 mills/kwh or about 0.4 mills higher than at optimum conditions.

In Situ Value Calculations

The investigation of the contribution made by the plutonium bred in a thermal reactor to the performance of the reactor was continued. Previous work was reported in the July monthly report as "Premature Discharge Price Calculations." In this work, it was assumed that the fuel cost is a function only of the uranium burnup costs, the plutonium credit, and the jacketing costs. With this model, it was shown that an unforeseen plutonium market price that would be sufficient to interrupt a planned irradiation in order to sell the bred plutonium is in the limit:

$$X = \frac{D_o + C - D_e + E\frac{dD_e}{dE}}{P_e - E\frac{dP_e}{dE}} \qquad (3)$$

where:

X = plutonium price \$/gram

D = uranium price, \$/cm^3 fuel

C = jacketing costs, \$/cm^3 fuel

E = exposure, MWD/Ton

P = plutonium concentration, grams/cm^3

All of the above, with the exception of the jacketing cost, are considered to be functions of the exposure and, hence, are subscripted e or o. The latter denotes an initial property.

The plutonium price given by this equation, which has been termed a premature discharge price, does not represent the value of the plutonium; but, rather, the value of the additional exposure. In the vast majority of situations, only an extremely inflated market price for plutonium could induce the reactor operator to prematurely discharge the fuel elements from his reactor in order to recover the plutonium. This cannot be construed as uniquely enhancing the value of plutonium as a similar "value" could be allocated to the partially spent uranium or to special fission products.

An expression that directly reflects the contribution of the bred plutonium can be derived by considering the hypothetical situation wherein minute amounts of plutonium can be added to the fuel. If this plutonium has the same composition as that existing in the reactor, the value of the plutonium will be proportional to the additional exposure so obtained. Furthermore, if the worth of the additional exposure is proportional to the fuel cost and the depletion of the uranium during the extended exposure is taken into account then, the *in situ* value of the plutonium is in the limit:

$$V = \frac{D_o + C - D_e + E \frac{dD_e}{dE}}{P_e + E \frac{dP_e}{dE}} \qquad (4)$$

Note that this expression is identical with that obtained for the premature discharge price except for the sign of the second term in the denominator. The results calculated with this expression are somewhat higher than the values given by the PUVE solution. The premature discharge prices for the same cases are much higher than the PUVE solution values.

Uranium Price Schedules

Different versions of a computer code designed to calculate uranium price schedules have been combined into one program. This program has been named UCOST. A number of changes were made in order to make the program more efficient, to simplify the required input data, and to generalize the calculation. Additional generality was obtained by removing the restrictions in the previous codes of natural uranium feed, of the optimum tails composition, and of nonzero unit feed and separative duty costs. It is recognized that most cases of interest will retain these restrictions so that, for simplicity, the program will impose them unless specifically directed otherwise.

Briefly, the program can be operated with any reasonable feed composition by specifying any three of the four schedule parameters (1) the cost of separative duty, (2) the cost of the feed, (3) the optimum tails composition, and (4) the value of the tails composition. In addition, all of these can be specified at once, but the tails composition will then not necessarily be the optimum composition. The printout includes, for each enrichment specified, the cost of the uranium per kilogram, per pound, and per gram of contained U-235; the flow rates of feed and of separative work required to produce a unit weight of the enriched uranium; and the feed and separative work portions of the total cost in $/kg of uranium.

A description of this program will be published as HW-74762, "UCOST - A Computer Code to Calculate Uranium Price Schedules Based on the Diffusion Cascade."

Code Development

OPTIMIZER. A new code called OPTIMIZER which will optimize costs from MINIMIZER against any independent variable has been written and debugged, and is being used daily. MINIMIZER is used to choose the enrichment that will yield minimum fuel cost in a given reactor. OPTIMIZER then uses the MINIMIZER results from a group of reactors with a single property (say lattice spacing) varied to calculate the optimum value of that independent variable to achieve minimum fuel cost.

PLOT. A generalized data plotter written in conjunction with the OPTIMIZER code. This code uses curve fits of data from OPTIMIZER, chooses a set of coordinates, and plots the fitted data curve. The coordinates usually are fuel cost and some reactor parameter.

MINIMIZER. An additional calculation is now being made in MINIMIZER to detect when operating on natural uranium from U_3O_8 may be cheaper than the cascade uranium at some low enrichment. The use of natural uranium from ore will eliminate the conversion to, and from, UF_6 as the AEC pricing regulations stipulates. This saves about $8/kg on natural uranium cost. (Natural uranium as UF_6 is $23.50/kg minus $3 credit for reclaiming fluorine minus $13.25 for natural uranium as U_3O_8 leaves a balance of $8.25/kg.)

PROTEUS. A routine was programmed to integrate the instantaneous k_{oo} calculated by MELEAGER (with respect to in-reactor time with importance weighting) to represent graded irradiation. The routine has been incorporated into the PROTEUS code as an alternative to the "graded k_{oo}" calculated by MELEAGER. This optional calculation was programmed to facilitate a flux-volume weighting modification when results become available from codes accounting for reactor dimensionality such as ALTHAEA.

MELEAGER. The several existing versions of MELEAGER are being consolidated in a single deck. These versions follow:

1. Standard MELEAGER
2. Burnable poison MELEAGER
3. Zoned Spectrum MELEAGER
4. Diffusion MELEAGER
5. RECYCLE CHAIN MELEAGER

Plutonium Self-Shielding Studies

Comparison of plutonium and U-235 fueling systems is complicated by the unique low energy plutonium resonances. The 0.3 ev Pu-239 and the 1.0 ev Pu-240 resonances are particularly important. The Pu-239 resonance is predominantly a capture resonance and the corresponding alpha value is approximately 0.75, and the value of eta is 1.67; while the alpha value for 2200 meter per second neutrons is about 0.39 and the value of eta is 2.08 -- the same as U-235's. It would appear that as the thermal flux is hardened or elevated in temperature, the effective alpha value for Pu-239 would increase well beyond the 2200 meter figure, and the associated fuel value would decrease. It is of extreme importance to note that the resonance self-shields so that as the concentration of plutonium is increased, the 0.3 resonance absorptions tend to decrease per atom of Pu-239 present. This effect is often encountered, but it is usually overshadowed by the corresponding spectral hardening due to the increased blackness of the fuel as the plutonium concentration is increased. This effect clearly shown by considering addition of increasing amounts of plutonium to a lattice cell already well hardened by a 1/v absorber.

Such computations have been done with two models. The first and simpler model considers the reactor to be homogeneous, with the fuel and moderator intimately mixed. The second and more complex model considers the reactor to be heterogeneous which involves a moderator media in which lumped fuel is placed. In the heterogeneous case, the self-shielding of the resonances is increased.

Earlier work using SPECTRUM V (HW-71953) code had indicated that the alpha values for Pu-239 decreased at higher plutonium concentrations due to self-shielding in homogeneous fuel moderator mixtures. It was considered likely that heterogeneous reactors would attain this degree of self-shielding at lower concentrations due to the fact that resonance shielding effects are accentuated by heterogeneous fuel geometries.

Accordingly, SPECTRUM V, which is a multigroup zero dimensional slowing down model, was altered so that shielding factors based upon the geometrical configuration could be applied to the cross sections at each energy.

A two-dimensional array of the shielding factors was supplied to SPECTRUM V from an external source for each specific geometry to be considered. They are supplied as a function of Σ_a and Σ_s for a homogeneous equivalent of the cell being analyzed. The factors are derived by a transport code for each neutron energy considered in SPECTRUM and the specific geometry being considered and is defined as the ratio of the average effective fuel cross section with flux depression divided by the total cross section which would be encountered without flux depression. The appropriate shielding factor is selected in spectrum from the two-dimensional array in memory by means of an interpolation extrapolation subroutine. The SPECTRUM code then develops the flux and effective cross sections as though a homogeneous case were being run. The results of such studies are shown in the following table for one-half-inch diameter fuel rods from the heterogeneous case (in parenthesis) as compared to a homogeneous reactor. An alpha value of 0.473 corresponds to an eta of 1.96 computed by the relationship:

$$\eta = \left(\frac{1}{1+\alpha}\right)\left(2.88\right) \frac{\text{neutrons}}{\text{neutron absorbed}}$$

An alpha value of 0.575 corresponds to an eta of 1.82. The boron concentration of 2.25 x 10^{19}/cc of fuel is used to approximate the parasitic absorptions encountered in a stainless steel jacketed water moderated reactor.

TABLE I

Pu-239 ALPHAS AS A FUNCTION OF Pu-239 AND 1/v ABSORBER B_{10} CONCENTRATION COMPARED FOR HOMOGENEOUS AND HETEROGENEOUS CASES

Grams of Pu-239/cc	0.04	0.20	0.40	1.00	2.00
Atoms of B_{10}/cc	ALPHAS				
0	0.537 (0.529)	0.566 (0.533)	0.565 (0.525)	0.537 (0.501)	0.493 (0.473)
1.25×10^{19}	0.547 (0.538)	0.569 (0.536)	0.566 (0.527)	0.537 (0.501)	0.493 (0.473)
2.25×10^{19}	0.553 (0.545)	0.571 (0.537)	0.567 (0.528)	0.537 (0.502)	0.493 (0.473)
5.25×10^{19}	0.569 (0.559)	0.575 (0.543)	0.569 (0.530)	0.537 (0.503)	0.493 (0.474)

Note: The heterogeneous alphas are designated within the parentheses.

Power Reactor Fuel Reprocessing Economic Studies

Checking and debugging of the Conventional Reprocessing Code was continued during the month and is now essentially complete. Both the Salt Cycle Code and the Conventional Reprocessing Code now appear ready for use. Some effort will be necessary to define the "best values" for the numerous cost variables in these codes before realistic evaluation of the Salt Cycle Reprocessing economic advantages for specific reactor cases can be started.

E. E. Eckl[?]

for Manager,
Programming

WK Woods:jm

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF SEPTEMBER 1962

A. ORGANIZATION AND PERSONNEL

Transfers within the Section during the month included John M. Selby transferring from Radiological Development and Calibrations to Composite Dose Studies and Records, and Bernhardt V. Andersen transferring from Composite Dose Studies and Records to Radiological Development and Calibrations. Charles N. Anderson transferred from the X-ray Department, Pinellas Plant, to External Dosimetry. Lee A. Bond transferred from Internal Dosimetry to the Analytical Laboratories. The following temporary summer employees resigned from the Company: Harold E. Ransom and James L. Beecroft (teachers), P. O. Anderson, D. R. Naught, D. F. Adams, G. M. Stephens, E. D. Britch and L. R. Kincheloe.

B. ACTIVITIES

Occupational Exposure Experience

Two new cases of plutonium deposition were confirmed by bioassay analyses during September. The total number of plutonium deposition cases that have occurred at Hanford is 299 of which 216 are currently employed. The two new plutonium deposition cases were both CPD employees. The first case was estimated to be less than five percent of the permissible body burden and resulted from a scratch accidentally self-inflicted with a contaminated stainless steel wire. The second case was estimated as less than one percent of the permissible body burden and was the result of accidental exposure to air contaminated because of a hood glove rupture.

Two employees were examined with the wound counter in the Whole Body Counter laboratory as a result of a contamination incident and a contaminated injury. The first, a J. A. Jones craftsman, was found to have plutonium skin contamination amounting to 2.9×10^{-3} μc. The skin area was intact. Decontamination was affected by conventional procedures. The second, a CPD process operator, received a contaminated puncture wound while working in a 234-5 hood. Wound counter examination detected 2×10^{-4} μc of plutonium in the employee's finger. No medical action was required. Urine sampling for bioassay was ordered for each of these cases.

There were five incidents at the 234-5 Building and one incident in HLO facilities which required special plutonium bioassay sampling for 16 employees.

A Hanford Laboratories PRTR employee was found to have nasal contamination amounting to 10,000 counts per minute during a personal survey. Examination in the Whole Body Counter showed a transient internal deposition of $Ru^{103-106}$ equivalent to approximately two percent of the maximum permissible body burden.

A radiation incident which occurred near the end of August was investigated with the following results. The hand of an IPD employee received radiation dose exceeding the Hanford operational control for a four-week period. The radiation exposure resulted when an employee disconnected a flexible hose from a vacuum line. Contamination which had accumulated at the joint resulted in a surface dose rate of 60 rads/hour measured by a CP-TP survey instrument and 2 r/hour at 18 inches. Evaluation of the radiation exposure indicates the radiation dose to the hands may have been as great as 9 rads. The whole body dose estimated from the film dosimeter was 0.7 rad including 0.2 r.

Tubing decontamination at the 242-B Building involved radiation levels to 60 r/hour and radiation dose rates to personnel to 4 r/hour.

The maximum dose rate encountered during two simulated waste calcination runs conducted in A cell of the High Level Radiation Facility was 250 mrads/hour. One milliliter of 1WW solution diluted to 33 liters was used for the trial. Two subsequent "hot" runs were completed without difficulty.

The hands of sixteen employees with plutonium body burdens estimated as greater than five percent of the permissible acquired through circumstances unknown were examined with the wound counter. No plutonium was detected in any of these employees' hands.

Environmental Experience

A comparatively heavy influx of fallout materials occurred during the week ending September 7, when the average concentration in air as measured on filter samples of our Pacific Northwest network reached 11 μμc beta/m^3. This is essentially the same as the highest value of 12 μμc beta/m^3 observed following USSR testing in the fall of 1961. The fallout was also evidenced by increased concentrations of I^{131} in pasture grass, cattle thyroids, and milk. The peak level in local milk was about 90 μμc/l on September 19.

High concentrations of radioactive material were observed in the exhaust duct from the 325-A hot cell during a waste calcination run. Surveys conducted at the base of the stack and in the 300 Area revealed no detectable amounts of contamination.

A total of 225 biological and produce samples were obtained for radiochemical analysis. They include: 63 samples of milk, 31 samples of pasture grass, 2 samples of meat, 39 sets of beef thyroids, 2 samples of Willapa Bay oysters, 80 samples of Columbia River fish, and 8 samples of baby food.

Two aerial surveys were made; one confirmed that slightly higher radiation readings exist over the mud and sand bars near the mouth of the Columbia River than are characteristic of the lower section of the river as a whole. The other survey was associated with special tests at Redox.

Studies and Improvements

Sampling equipment was set up at the 100-H, DR, B and C Areas for the continuous collection of effluent samples from the outlets of the retention basins. This program is to be extended to all of the 107 basins in order to improve estimates of total quantities of radionuclides discharged to the river.

The 308 Building ventilation system review was completed. The ventilation system performed exceptionally well and does, in fact, operate as designed. Two deficiencies were discovered as a result of tests performed on August 25. These were: (1) the exhaust capacity is so great that under extreme emergency conditions the supply plenums would rupture; and (2) the atmospheric reference lines are undersized for the length of line needed. The following recommendations were made: (1) install backflow dampers in the supply and exhaust plenums (these dampers in the supply will allow air to be pulled through them with the supply fans off); (2) disconnect the motor-operated dampers in the supply but not remove them; and (3) install larger lines between the reference sensor and the flow regulators.

Two features were added to the film dosimeter processing machine. The system used to actuate a buzzer was modified to shut off the sequence timer of the machine if the X-ray does not go on or if the anode current of the X-ray tube does not come up to 9 milliamperes or greater.

Ten BF_3 tubes were received and evaluated. These tubes are constructed from extruded copper rather than machined brass. The sensitivity of the new tubes is the same as those made of brass. Very little variation was found between tubes.

It was determined that multiple collision theory should have been used to determine the dose, and thus the fast scale calibration of the new BFQ instruments. The meter scales were redrawn to reflect the change in dose.

Two 0.075 inch silicon diodes were exposed to plutonium fluoride neutrons in the 234-5 Building for a two-week period. Both diodes detected a dose from these neutrons. Positive ion accelerator irradiations are continuing in an effort to determine the energy dependence of each diode. Using the constant voltage readout technique, it appears that there may be some energy dependence. With the diode neutron irradiations, a lithium-6 sandwich solid state spectrometer was exposed for the purpose of determining the linearity and sensitivity of the spectrometer to monoergic neutrons.

Calibration data for the alpha-beta-gamma air sample counters was examined by an IBM 7090 computer program. The program indicated the calibration could be done with a one-point fit. When tried, the slopes of the curves varied between instruments. It is, therefore, necessary that each instrument has its own calibration obtained by the one-point fit.

A new radiation protection regulation issued by the Department of Labor (as 41 CFR 50-204) was reviewed, and comments provided.

A compilation of background material on occupational exposure standards, working limits and administrative control practices was completed. This material was prepared for the use of Union Relations personnel.

Eight members of the Radiological Emergency staff participated in the 1962 Civil Defense exercise, SPADE FORK, to the stated satisfaction of HOO-AEC. Three shifts participated during the four-day exercise.

I^{131} studies continued in the Whole Body Counter laboratory during September. Twenty-three children were examined a total of 27 times. The maximum quantity measured in a thyroid was 90 μμc I^{131}.

A liquid scintillation technique for determining C^{14} in urine was studied. The counting efficiency of the method was determined to be 41 percent. Sample volume was critical because of quenching effects of quantities of water greater than one milliliter. The effect of carbonate ion on counting efficiency was studied and found to be negligible.

Procurement and assembly of components for the mobile Whole Body Counter continued during the month. The electrical system was rewired, a new distribution panel installed, and the slide door installation was completed.

C. VISITS AND VISITORS

Visitors consulting with members of the Radiation Protection Operation during the month included:

S. Auerback) - Oak Ridge National Laboratory, Oak Ridge, Tennessee
J. S. Olson)
Lt. Col. R. Kyner) - Division of Biology and Medicine, AEC-Washington, D.C.
Dr. M. Schulman)
Prof. M. A. Rollier - University of Italy
N. Ketzlack - Atomics International, Canoga Park, California
J. P. Vanne - EURATOM Transuranium Institute, Karlsruhe, Germany
J. M. Grunke - Department of Agriculture, Boise, Idaho
Dr. D. Ferret - United Kingdom Atomic Energy Authority, Harwell, England

Members of the Radiation Protection Operation visiting off-site during the month included:

F. Swanberg - Attended symposium on Radioactivity in Man at Northwestern University, Chicago, Illinois. Consulted with Argonne National Laboratory.

E. C. Watson - Attended Fallout Symposium sponsored by Defense Atomic Support Agency in San Francisco, California.

L. C. Rouse - Consulted with Pacific Gas and Electric Company, San Francisco, California, on radiation records systems.

A. R. Keene - Conducted ASA N2.2 Subcommittee meeting in San Francisco, California.

D. RELATIONS

Three suggestions were submitted by personnel of the Radiation Protection Operation during September. One suggestion was adopted; five were rejected. Five suggestions are pending evaluation.

Safety meetings were held throughout the Section during the month of September. A safety and housekeeping inspection was also conducted.

Six radiation protection orientation lectures were presented to Biology Research employees. Fifty IPD personnel attended Disaster Level Monitoring courses presented at 1760-D Building. To date a total of 303 persons have attended the course.

E. SIGNIFICANT REPORTS

HW-74307 8 - "Radiological Status of the Hanford Environs for August, 1962" by R. F. Foster.

HW-74398 - - "Evaluation of Radiological Conditions in the Vicinity of Hanford, April-June 1962" by Environmental Studies and Evaluation Staff.

HW-75008 - - "Dose Versus Shield Material and Thickness from a Nuclear Excursion" by L. G. Faust and L. L. Carter.

HW-75108 - - "Monthly Report for September 1962 - Radiation Monitoring Operation" by A. J. Stevens.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

	September	1962 to Date
External Exposure Above Permissible Limits		
Whole Body Penetrating	0	3
Whole Body Skin	0	3
Extremity	1	3
Hanford Pocket Dosimeters		
Dosimeters Processed	2,867	26,336
Paired Results - 100-280 mr	16	83
- Over 280 mr	2	10
Lost Results	0	0
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	9,521	86,131
Results - 100-300 mrads	160	2,677
- 300-500 mrads	18	242
- Over 500 mrads	10	96
Lost Results	31	241
Average Dose Per Film Packet - mrad (ow)	13.28	13.50
- mr (s)	43.77	29.18
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	3,104	14,427
Results - 50-100 mrem	15	25
- 100-300 mrem	12	41
- Over 300 mrem	0	2
Lost Results	19	80
Fast Neutron		
Film Read	778	3,730
Results - 50-100 mrem	50	422
- 100-300 mrem	171	659
- Over 300 mrem	55	66
Lost Results	9	31
Hand Checks		
Checks Taken - Alpha	51,227	306,537
- Beta-Gamma	49,921	467,086
Skin Contamination		
Plutonium	30	223
Fission Products	66	435
Uranium	0	12
Tritium	0	0

Whole Body Counter	Male	Female	September	1962 to Date
GE Employees				
Routine	0	0	0	136
Special	4	0	4	169
Terminal	5	0	5	90
Non-Routine	10	2	12	207
Non-Employees	16	18	34	69
Pre-Employment	0	0	0	8
	35	20	55	679

Bioassay	September	1962 to Date
Confirmed Plutonium Deposition Cases	2	16*
Plutonium - Samples Assayed	179	3,068
- Results Above 2.2×10^{-8} μc/Sample	28	174
Fission Product - Samples Assayed	122	3,631
Results Above 3.1×10^{-5} μc/Sample	0	15
Uranium - Samples Assayed	114	1,414
Biological - Samples Assayed	0	247
Strontium - Samples Assayed	0	299

Tritium Samples	Maximum	Count	September Total
Urine Samples > 5.0 μc/l	17.3	25	
< 1.0 μc/l		79	
Samples Assayed			224
D_2O Samples			
Moderator	782.3 μc/ml	4	
Primary Coolant	5.50 μc/ml	2	
Reflector	768.4 μc/ml	2	
Samples Assayed			8
Other Water Samples			
No. 185 EFD	.287 μc/ml		56
			288

Calibrations	Number of Units Calibrated	
	September	1962 to Date
Portable Instruments		
CP Meter	944	9,050
Juno	269	2,494
GM	503	4,976
Other	164	1,674
Audits	96	939
	1,976	19,133

*The total number of plutonium deposition cases which have occurred at Hanford is now 299, of which 216 are currently employed.

	Number of Units Calibrated September	1962 to Date
Personnel Meters		
Badge Film	1,158	14,182
Pencils	-	12,670
Other	453	3,786
	1,611	30,638
Miscellaneous Special Services	229	8,813
Total Number of Calibrations	3,816	58,584

F. Swanberg, Jr.
lw
for the Manager
RADIATION PROTECTION

AR Keene:FS:ljw

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

A tentative financial plan from HOO-AEC covering the 04 Program indicates a substantial reduction from the Hanford Laboratories FY 1963 request. A comparative summary follows:

(Amounts in thousands)	FY 1963 Request	DRD Planning	Increase (Decrease)
Research and Development	$11 182	$ 9 068	$(2 114)
Test Reactor Operation	2 340	2 350	10
Equipment	1 245	1 041	(204)
Irradiation Units	2 673	2 020	(653)
Total	$17 440	$14 479	$(2 961)

At the end of the first quarter, 25% of currently available funds has been spent. Available funds on the Experimental Gas Cooled Reactor subprogram have been overspent. Other subprograms (Gas Loop Project, Uranium Fuels Development, Plutonium Ceramics Research) will be curtailed or discontinued in the near future in order to stay within allotted funds.

A request for preparation of the FY 1963 Midyear Budget Review was received from Contract Accounting, indicating a full-scale review with completion scheduled in November. Section managers within Hanford Laboratories have been requested by letter to supply needed information in accordance with a time schedule provided.

Activities for which special accounting codes were established during the month are described below:

Accounting Code	Activity
.6E	Perform PuC, Pu_2C_3 Synthesis for United Nuclear Corporation - estimated costs of $500 cover sample preparation, chemical analysis and X-ray analysis.
.6G	University of Washington use of Laboratory Facilities - the University will be billed for materials and supplies in connection with the use of facilities in 306 and 326 Buildings.

The operating cost reports for all Hanford Laboratories' components were prepared by EDP equipment for the first time in September covering August business. This represents a major milestone in the mechanization of cost accounting for Hanford Laboratories.

An audit was made of signature authorizations on store withdrawals for the week ending September 2, 1962 for Finance and Administration, Programming, Test Reactor & Auxiliaries, Radiation Protection and Applied Mathematics Operations. Discrepancies were reported to managers concerned.

One suggestion award with over $500 annual savings was reviewed during the month and granted financial approval.

Visual aid data were prepared for the Manager of the Biology Operation for use during a recent visit of personnel from the Division of Biology and Medicine, Washington, D. C.

Assistance is being given Mr. B. B. Field of Applied Mathematics Operation on a study of the economic impact of HAPO on the Tri-Cities Area.

An annual report on HAPO Technical Personnel Recruitment results for FY 1962 was prepared at the request of HOO-AEC. Pertinent information is summarized below:

Personnel Status	Applicants Interviewed	Offers Made	Hires
Professional	303	168	86
College Recruitment	902	202	95
Total	1 205	370	181

General Accounting

Following is a summary of the status of letters or agreements covering specific actions requiring AEC concurrence:

AT-256	Participation in Standardizing Activities - (S. H. Bush)	AEC considering
AT-262	Washington State Scientific and Advisory Committee - Selective Service System - (I. H. Dearnley)	In preparation

1236891

Travel activity this fiscal year is running slightly higher than in FY 1962, but somewhat lower than in FY 1961, as shown below:

Number of Trips Started - 1st Quarter

FY 1961	314
FY 1962	258
FY 1963	290*

* FY 1963 statistics do not include the attendance of 43 Hanford Laboratories' people at the ANS meeting in Richland in September.

During the month of September, billings to completed plant from Work in Progress accounts amounted to $182,759.

Hanford Laboratories' material investment at September 1, 1962 totaled $25.1 million as detailed below:

	(In thousands)
SS Materials	$23 830
Reactor and Other Special Materials	933
Spare Parts	333 -1)
	$25 096

(1- Includes a reserve established at September 1, 1962 amounting to $80,319.

Nuclear materials consumed in research during the fiscal year to date totaled $2.07 million of which $2.00 million are applicable to Hanford Laboratories, and $ .07 million to Fuels Preparation Department. The following is a detail by program for the Hanford Laboratories' portion:

	(In thousands)
2000 Program	$ 604
3000 Program	314
4000 Program	1 083
	$2 001

The annual physical inventory of Reactor and Other Special Materials in the custody of 97 holders was taken on September 26, 27, and 28, 1962. In accordance with established practice, this inventory was witnessed by financial personnel. Reconciliation of the inventory findings is in progress, and upon completion a report of results will be issued.

Survey 20 - Part I, witnessed verification of HAPO inventories of normal uranium, thorium, and U-233 as of the end of September 1962 by a team of HOO-AEC personnel accompanied by a member of the Nuclear Materials Operation, was in progress at month's end.

An inventory of unused One Trip Property Passes in the possession of 53 authorized holders accounted for all except two passes which had been inadvertently destroyed. Nine passes were not located during the previous inventory, indicating that pass control has improved. Two missing passes represents a minor fraction of the 2,197 passes processed since March 31, 1961, the date of the previous inventory.

Laboratory Storage Pool activity for the month of September 1962 is summarized below:

Equipment	Current Month Quantity	Current Month Value	FY to Date Quantity	FY to Date Value
Items received	63	$46 493	243	$182 796
Items withdrawn by custodians	6	3 016	28	19 175
Equipment reassigned (purchase eliminated)	7	1 909	52	26 658
Items disposed of by excess	6	434	6	434
Equipment on hand at 9-30-62			1 251	703 107 -1)

(1- Includes 140 items valued at $58,181 which were on loan at 9-30-62.

The material inventory in the Laboratory Pool at month end was comprised of the following:

Material	Quantity	Total Value
Beryllium	1 035 grams	$ 631 -2)
Gold	2 205 grams	3 175
Palladium	2 224 grams	2 713
Platinum	2 322 grams	6 687
Clean scrap	441 grams	1 120
Contaminated scrap	6 703 grams	15 551 -2)
Silver	6 815 grams	477
Hafnium	2 939 grams	500 -2)
Zirconium	5 252 pounds	104 706
		135 560
Additional material held for convenience of others		148 758
Total material held at the Pool		$284 318

(2- Reflects price change applied this month

September heavy water inventory disclosed a loss of 2,227 pounds valued at $30,716. Scrap material increased during the month an estimated 27,868 pounds due primarily to the classification of all moderator in the primary loop as scrap. The charge to cost for scrap generated during September ($20,793) and loss ($30,716) totaled $51,509.

Action during the month on projects is indicated below:

New Money to HL

CAH-977 Facilities for Radioactive Inhalation Studies $3,000

Physical Completion Notice Issued

CAH-888 Biology Facility Improvements, 108-F Building (AEM Services only)

Construction Completion and Cost Closing Statement Issued

CAH-842 Critical Facility (AEM Services only)

Contracts processed during the month included:

CA-350	Lewis W. Seagondollar
DDR-110	Battelle Memorial Institute
DDR-138	Battelle Memorial Institute
DDR-160	Battelle Memorial Institute

OPGs issued in September are shown below:

OPG No.	Title
7.14	Foreign Travel
55.6.2	Membership in Trade & Professional Societies and Associations
5.2	Supplement - Automatic Data Processing Equipment
22.1.12	Test Reactor and Auxiliaries Organization

Personnel Accounting

During September, 131 nonexempt and 10 exempt employees of FPD Plant Facilities Operation were transferred to Hanford Laboratories. Of these, 122 were assigned to Test Reactor and Auxiliaries and 19 to Finance and Administration, Facilities Engineering Operation.

TECHNICAL ADMINISTRATION

Employee Relations

Twenty-three nonexempt employment requisitions were filled during September with twelve remaining to be filled.

Professional Placement

Advanced Degree - Six Ph.D. applicants visited HAPO for employment interviews. Two offers were extended; one acceptance and two rejections were received. One offer is currently open.

BS/MS - Six Program offers and three direct placement offers were extended. Offers accepted: one Program and one direct placement. Offers rejected: four Program and three direct placement. Current open offers: four Program and two direct placement.

Technical Graduate Program - No permanent assignments were effected. Four new members were added to the rolls and one terminated. Current program members total 59.

Technical Information

The first Quarterly Progress Report: Irradiation Effects on Reactor Structural Material was published this month.

ECONOMIC EVALUATIONS

A cursory review and evaluation was made of actual cost data reported or otherwise available on the complete fuel cycle for plutonium enriched fuel elements.

FACILITIES ENGINEERING

At month's end Facilities Engineering Operation was responsible for 13 active projects having total authorized funds in the amount of $2,338,100. The total estimated cost of these projects is $8,732,000. Expenditures on them through August 31, 1962 were $1,329,000.

The following summarizes project activity in September:

Number of authorized projects at month's end	13
Number of new projects authorized	1
CAH-977, Facility for Radioactive Particle Inhalation Studies	
Projects completed	1
CAH-888, Biology Laboratory Improvements	

Pages appended to this report provide detailed project status information.

Services

Satisfactory progress was made in the engineering services provided on the following jobs:

Equipment procurement valued at $310,000 including the issuance of two requisitions totaling $33,000. Material and equipment lists for two projects were revised.
321 Tank Farm alarm system
340 Emergency valving sequence study
108-F source handling facility
141-M proposed building addition
300 Area process simulation facility
Split-half machine
Controlled environment facility
Third-party inspections of three elevators

Pressure system assistance was provided on:

Dynamic materials test apparatus
Quartz glass UO_2 test section
TF-23 test loop
C-25 Building autoclaves
EDEL-1 loop
PRTR ion-exchange vessel
189-D hydraulics laboratory
189-D electrical preheaters
Third-party inspections of thirteen pressure vessels

Plant engineering effort was expended on:

325 vacuum pump replacement
308 vent studies
3702 ventilation modification

325 Analytical laboratory modifications
321-A circuit breaker panel installation
3705 electrical service panel installation
231-Z alarm system standardization
108-F increased transformer capacity and bus duct split

Facilities Operation

Landlord costs for August were $118,240 which represented 77% of the anticipated expenditure. Improvement maintenance costs were $8,128.

Waste disposal operations are summarized below:

	August	July
Concrete Barrels	2	2
Loadluggers	2	2
Crib Waste	270,000 gal.	270,000 gal.

Building Operation during September included continuation of the filter plant (315 Building) "run in." Basic training of the operators was completed. The plant was physically tied into the 309 Building on September 21, 1962. Minor revisions in the system are being made.

Drafting

The equivalent of 135 drawings were produced during the month for an average of 24.1 man-hours per drawing.

Major jobs in progress are: PRTR as-builts, PRTR shim rod control, electrical resistivity sample holder, cladding cutter assembly for PRTR, glove box and vacuum system, high temperature furnace, "C" Cell Equipment-327 Building, process tube and fuel handling carriage, fuel element spacer former and PRTR corrosion test facility.

Construction Supervision

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories components is given below:

	Unexpended Balance
Orders outstanding at beginning of month	$205 686
Issued during the month (inc. 5 suppl. and adj.)	50 268
J. A. Jones Expenditures during month (incl. C.O. Costs)	126 231
Balance at month's end	129 723
Orders closed during month	87 745

Maintenance work orders total seven with face value totaling $42,122.

Construction and maintenance activities completed during September included:

108-F Electrical service change
108-F Repair fire damage in rooms 2202 and 2202-A
Rattlesnake Spring Radioecology facility - install laboratory tables and electric recorders
309 Critical facility startup items
309 Rupture loop electrical and mechanical modifications
309 Gas loop electrical and mechanical modifications
328 First floor office revisions
3718 Office addition
307 Retention basin VCP line repair

D Parsley
for Manager
Finance and Administration

W Sale:whm

1236898

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75127

DATE 9-30-62

PROJ. NO.	TITLE	FUNDING
CAH-822	Pressurized Gas Cooled Loop Facility	4141 Operating

AUTHORIZED FUNDS			COST & COMM. TO 9-16-62	$ 1,131,761(GE)
$ 1,170,000	DESIGN $ 43,000 / CONST. $ 1,127,000	AEC $ 15,000 / GE $ 1,155,000	ESTIMATED TOTAL COST	$ 1,170,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	8-19-59	7-25-62*	DESIGN	4-29-60
CONST.	10-17-60	DIR. COMP. DATE 12-31-62	CONST.	12-31-62

ENGINEER

TR&AO-MEEO - DP Schively

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100	97	92
PF	1.4	0	0
CPFF	22.3	97	91
FP	6.6	100	100
Gov.Eq.	69.7	100	93

SCOPE, PURPOSE, STATUS & PROGRESS

New heater has passed shop tests and was shipped 9-20-62.

First gas bearing blower rescheduled for test on September 30, 1962. Earliest completion date of both units is now estimated to be October 30, 1962.

Vendor has been troubled with axial oscillations on start and stop attributed to shaft unbalance. New balancing parts for both rotors have been completed.

*Initial authorization date was December 18, 1958.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS			COST & COMM. TO	$
$	DESIGN $ / CONST. $	AEC $ / GE $	ESTIMATED TOTAL COST	$

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN			DESIGN	
CONST.		DIR. COMP. DATE	CONST.	

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

1236899

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75127

DATE 9-30-62

PROJ. NO.	CGH-857
TITLE	Physical & Mechanical Properties Testing Cell - 327 Bldg.
FUNDING	0290
AUTHORIZED FUNDS	$ 460,000
DESIGN $	45,000
CONST. $	415,000
AEC $	--
GE $	460,000
COST & COMM. TO 9-16-62	$ 344,148
ESTIMATED TOTAL COST	$ 460,000
STARTING DATES — DESIGN	11-2-59
STARTING DATES — CONST.	2-12-62
DATE AUTHORIZED	9-22-61*
DIR. COMP. DATE	12-15-62
EST'D. COMPL. DATES — DESIGN	3-15-61
EST'D. COMPL. DATES — CONST.	12-15-62
ENGINEER	FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE	7	310
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION	.5	888
GE FIELD ENGINEERING	.4	30

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	46	48
PF			
CPFF	18	50	50
FP			
Equip.	82	45	48

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities for determining physical and mechanical properties of irradiated materials, and involves the installation of a cell in the 327 Building.

Current estimate of Title I and II costs - $60,000. Detailed design started 4-1-60. Procurement and construction authorized 9-22-61.

Number of purchase orders required	19	Value (Est.)	$253,000**
Number of purchase orders placed	19	Value	206,345

Final shipment of cell was received September 18, 1962.

Installation of the cell is in progress. All cell casting exclusive of plugs and door have been placed.

Fabrication of storage rack is in progress.

Fabrication of ventilation exhaust filter holder has been delayed pending receipt of materials.

* Original authorization for design was October 1, 1959.

** Includes delivery charges, inspection and contingency.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-75127

DATE 9-30-62

PROJ. NO.	TITLE	FUNDING
CAH-866	Shielded Analytical Laboratory - 325-B Building	61-a-1

AUTHORIZED FUNDS			COST & COMM. TO	
$ 700,000	DESIGN $ 60,000	AEC $ 546,500	9-16-62	$ 137,250(GE)
	CONST. $ 640,000	GE $ 153,500	ESTIMATED TOTAL COST	$ 655,000

STARTING DATES				EST'D. COMPL. DATES	
DESIGN	9-5-59	DATE AUTHORIZED	5-31-60*	DESIGN	11-14-60
CONST.	6-15-61	DIR. COMP. DATE	11-15-62	CONST.	10-30-62

ENGINEER

FEO - RW Dascenzo

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	6	2571
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	10	100	100
AE-TIT. II	90	100	100
CONST.	100	100	99
PF	3	1	1
CPFF	2	0	0
FP	95	100	100

SCOPE, PURPOSE, STATUS & PROGRESS

This project will allow greater capacity for analytical work involving today's more highly radioactive solutions and consists of adding a shielded laboratory to the 325 Building.

Final inspection and acceptance of the Contractor's work was performed on 9-21-62. Several minor items were cleaned up by the Contractor on 9-26-62. Only two exceptions remain for the Contractor - 1) substituting six vacuum gauges for six low pressure gauges on the Air Sampling Lines, and 2) cleaning up a stain between the cover glass and lead glass on Viewing Window No. 3.

J. A. Jones forces will make the tie-in of a distilled water line, shorten all hinged cell trays, install minipulators purchased by G.E., replace a rheostat, and several minor items of work.

The Contractor was scheduled to complete his work on 9-7-62.

* Original authorization for preliminary design was August 12, 1959.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75127

DATE 9-30-62

PROJ. NO. CAH-867	TITLE Fuel Element Rupture Test Loop					FUNDING 58-e-15	
AUTHORIZED FUNDS $ 1,500,000	DESIGN $ 130,000	AEC $ 820,000		COST & COMM. TO 9-16-62		$ 571,683(GE)	
	CONST. $ 1,370,000	GE $ 680,000		ESTIMATED TOTAL COST		$ 1,500,000	
STARTING DATES	DESIGN 8-1-60	DATE AUTHORIZED 8-17-62	EST'D. COMPL. DATES	DESIGN 3-15-61			
	CONST. 11-2-60	DIR. COMP. DATE 10-31-62		CONST. 10-31-62			

ENGINEER: TR&AO-MEEO - PC Walkup

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	0	2475
COST PLUS FIXED FEE	2	2400
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	91	100	100
AE-TIT. II			
CONST.	100	100	99
PF	2	100	50
CPFF	57	100	98
FP (1)	10	100	100
(2)	31	100	100

SCOPE, PURPOSE, STATUS & PROGRESS

(1) G. A. Grant Company
(2) Lewis Hopkins Construction Company

This facility is to be used for fuel rupture behavior studies with respect to physical distortion and rate of fission product release.

Construction work has been suspended for Loop Design Tests.

* Initial authorization was on 10-1-59.

PROJ. NO. CAH-888	TITLE Biology Laboratory Improvements					FUNDING 60-h-1	
AUTHORIZED FUNDS $ 420,000	DESIGN $ 44,000	AEC $ 359,500		COST & COMM. TO 9-16-62		$ 60,178(GE)	
	CONST. $ 376,000	GE $ 60,500		ESTIMATED TOTAL COST		$ 415,000	
STARTING DATES	DESIGN 8-8-60	DATE AUTHORIZED 4-18-61*	EST'D. COMPL. DATES	DESIGN 3-31-61			
	CONST. 7-10-61	DIR. COMP. DATE 3-31-62		CONST. 8-20-62			

ENGINEER: FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		2,660
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	NS	100
TITLE I			
GE-TIT. II	17	NS	100
AE-TIT. II	83	NS	100
CONST.	100	100	100
PF	1	100	100
CPFF	10	NS	99
FP	89	100	100**

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides additional space for biological research supporting services, and involves an addition to the 108-F Building.

* Original authorization for design was May 3, 1960.

** General Electric has been advised by the AEC that the vinyl floor covering replacement will be shipped on September 24 and should arrive by October 1, 1962. Also, it should be completed by the end of the week of October 1, 1962. As-built work is progressing. The physical completion notice was completed and published showing a small underrun.

This project will not be reported further.

1239102

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75127

DATE 9-30-62

PROJ. NO.	TITLE	FUNDING
CAH-916	Fuels Recycle Pilot Plant	4-62-d-3

AUTHORIZED FUNDS			
$ 465,000	DESIGN $ 465,000	AEC $	COST & COMM. TO 9-16-62 $ 464,500
	CONST. $ -0-	GE $ 465,000	ESTIMATED TOTAL COST $ 5,460,000***

STARTING DATES			EST'D. COMPL. DATES	
DESIGN	3-15-61	DATE AUTHORIZED 6-29-62**	DESIGN	10-15-62
CONST.	11-1-62*	DIR. COMP. DATE	CONST.	11-15-64

ENGINEER

FEO - RW Dascenzo

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION	7	6913
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	97	96
TITLE I	11	100	100
GE-TIT. II	89	97	96
AE-TIT. II			
CONST.	100	0	0
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project is to provide a facility to perform a full scope of engineering tests and pilot plant studies associated with fuel reprocessing concepts.

325 of the 331 drawings have been issued for comment, 270 for approval and 175 approved for construction. The specifications are 80% complete.

A new project proposal revision, to permit demonstration of the waste calcination program in FRPP has been submitted to the General Manager's office for approval.

* Estimated construction starting date for removal of burial ground fill.

** Original authorization for initiation of design was February 9, 1961. June 29, 1962 is the authorization date for the last design supplement.

*** Including transferred capital property valued at $100,000.

PROJ. NO.	TITLE	FUNDING
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes	62-k

AUTHORIZED FUNDS			
$ 29,600	DESIGN $ 29,600	AEC $	COST & COMM. TO 9-16-62 $ 29,600
	CONST. $	GE $ 29,600	ESTIMATED TOTAL COST $ 289,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN	11-7-61	DATE AUTHORIZED 10-23-61	DESIGN	5-31-62
CONST.	12-15-62	DIR. COMP. DATE	CONST.	11-1-63

ENGINEER

FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER - Bovay Engineers		260
DESIGN ENGINEERING OPERATION		260
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	57	100	100
AE-TIT. II	43	100	100
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities to permit deliberate destructive testing of irradiated zirconium tubing. This will provide operating and tube life data not available because of the limited operating history of Zircaloy-2 pressure tubing in reactors.

The project proposal was submitted to the Commission on July 2, 1962 and is awaiting approval.

A meeting was held with the Commission on September 20 to review the project and more specifically, project costs. A detailed letter has been sent to the Commission explaining increases in project costs.

SEMI-MONTHLY PROJECT STATUS REPORT GENERAL ELECTRIC CO. — Hanford Laboratories			HW- 75127 DATE 9-30-62
PROJ. NO. CAH-927	TITLE Additions to the 271-CR Building Waste Treatment Demonstration Facility		FUNDING 62-[illegible]
AUTHORIZED FUNDS $ 92,000	DESIGN $ 11,000 CONST. $ 81,000	AEC $ 76,300 GE $ 15,700	COST & COMM TO 9-16-62 $ 14,888(GE) ESTIMATED TOTAL COST $ 92,000
[illegible]	DATE AUTHORIZED [illegible]-15-61	EST'D. DESIGN 2-5-62	PERCENT COMPLETE

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75127

DATE 9-30-62

PROJ. NO.	TITLE	FUNDING
CGH-951	A-C Column Facility - 321 Building	0290

AUTHORIZED FUNDS	DESIGN $ 5,000	AEC $ -0-	COST & COMM TO 9-16-62	$ 32,801
$ 55,000	CONST. $ 50,000	GE $ 55,000	ESTIMATED TOTAL COST	$ 55,000

STARTING DATES	DESIGN 1-30-62	DATE AUTHORIZED 1-12-62	EST'D. COMPL. DATES	DESIGN 4-1-62
	CONST. 3-15-62	DIR. COMP. DATE 10-31-62		CONST 10-31-62

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		197
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	90	87
PF	100	90	87
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide a closely integrated "A" Column in series with the relocated "C" Column to permit the development of a mathematical model for the mass transfer of uranium, as well as the exploration of the possibilities of computer optimization of a combined "A-C" extraction battery.

Relocation of "C" Column is complete. Instrument line gutters are installed. Miscellaneous interconnecting piping work is continuing. "A" Column fabrication and installation work is continuing.

Bids have been reviewed and accepted for process stream temperature control, flow control, pH control and variable speed drive speed control instrumentation systems.

PROJ. NO.	TITLE	FUNDING
CGH-955	Reactivation of the H-1 Loop - 105-H Building	0490

AUTHORIZED FUNDS	DESIGN $ 10,000	AEC $	COST & COMM. TO	$
$ 10,000	CONST. $	GE $ 10,000	ESTIMATED TOTAL COST	$

STARTING DATES	DESIGN 4-15-62	DATE AUTHORIZED 3-29-62	EST'D. COMPL. DATES	DESIGN
	CONST.	DIR. COMP. DATE		CONST.

ENGINEER

FEO - DM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		11
TITLE I			11
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project would have provided the primary test facility for determination of the feasibility of using aluminum-clad fuel elements in high temperature water by studying improved alloys and corrosion inhibitors. However, an alternate course of action has been selected and design work has been terminated. A project proposal requesting cancellation of this project has been routed for signatures.

* The $2614 in charges which were incurred for preliminary scoping have been transferred to operating costs.

1236905

HW- 75127

DATE 9-30-62

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

PROJ. NO.	TITLE	FUNDING
CGH-957	Small Particle Technology Laboratory - 325 Building	62-k

AUTHORIZED FUNDS			COST & COMM. TO 9-16-62	$ 35,050
$ 40,000	DESIGN $ 2,000	AEC $ --	ESTIMATED TOTAL COST	$ 40,000
	CONST. $ 38,000	GE $ 40,000		

STARTING DATES		DATE AUTHORIZED	3-21-62	EST'D. COMPL. DATES	
DESIGN	4-23-62	DIR. COMP. DATE	11-1-62	DESIGN	5-31-62
CONST.	7-6-62			CONST.	11-1-62

ENGINEER

FEO - DS Jackson

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE	9.1	192**
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION (HLO)	0	32.6
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	85*	52
PF	100	85*	52
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides laboratory space for research and development in small particle technology related to the generation, control, and disposal of radioactive wastes.

Painting, sheetmetal, electrical and mechanical work are continuing. Laboratory equipment has not been received on-site. Mechanical progress will accelerate upon receipt of this equipment.

* Project Planning Schedule.

** Corrected to include shop fabrication time.

PROJ. NO.	TITLE	FUNDING
CAH-958	Plutonium Fuels Testing & Evaluation Laboratory - 308 Bldg.	62-k

AUTHORIZED FUNDS			COST & COMM. TO 9-16-62	$ 2,000
$ 150,000	DESIGN $ 15,500	AEC $ 148,000	ESTIMATED TOTAL COST	$ 150,000
	CONST. $ 134,500	GE $ 2,000		

STARTING DATES		DATE AUTHORIZED	6-22-62	EST'D. COMPL. DATES	
DESIGN	10-15-62	DIR. COMP. DATE	5-15-63	DESIGN	1-15-63
CONST.	1-15-63			CONST.	6-30-63

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides for the extension of plutonium research laboratories on the second floor of 308 Building by erection of plastered ceilings and walls to provide contamination control barriers. It also includes laboratory service extension and fabrication of a metallography hood.

Work Authority CAH-958 (1) dated July 3, 1962 authorized the General Electric Company $2,000 to review the project scope and design and to submit a cost estimate in sufficient detail to assure the Commission that costs for the project are reasonable - CE&UO is presently preparing this review.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75127

DATE 9-30-62

PROJ. NO.	CAH-962
TITLE	Low Level Radiochemistry Building
FUNDING	05-1-63-H-001-23
AUTHORIZED FUNDS	$ 113,000
DESIGN $	113,000
CONST. $	
AEC $	82,000
GE $	31,000
COST & COMM. TO 9-16-62	$ 16,500 (GE)
ESTIMATED TOTAL COST	$ 1,200,000
STARTING DATES: DESIGN	7-23-62
STARTING DATES: CONST.	8-1-63
DATE AUTHORIZED	6-28-62
DIR. COMP. DATE	--
EST'D. COMPL. DATES: DESIGN	5-15-63
EST'D. COMPL. DATES: CONST	8-1-64
ENGINEER	

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION	1	35
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	-0-	-0-
TITLE I		92	35*
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides a building in which extremely sensitive radioanalyses and methods development can be performed in an atmosphere protected from the environs. It consists of designing and constructing a building housing approximately 22,000 square feet of floor area including the basement.

A schedule for the design criteria progress has been approved by the AEC.

Location of the facility has been established.

* Higher priority work has delayed progress on this design work.

PROJ. NO.	CAH-963
TITLE	Geological & Hydrological Wells - FY-1962
FUNDING	62-k
AUTHORIZED FUNDS	$ 80,000
DESIGN $	1,400
CONST. $	78,600
AEC $	68,500
GE $	11,500
COST & COMM. TO 9-16-62	$ 10,317 (GE)
ESTIMATED TOTAL COST	$ 80,000
STARTING DATES: DESIGN	5-18-62
STARTING DATES: CONST.	7-6-62
DATE AUTHORIZED	5-9-62
DIR. COMP. DATE	4-1-63
EST'D. COMPL. DATES: DESIGN	6-1-62
EST'D. COMPL. DATES: CONST.	4-1-63
ENGINEER	FEO - HE Ralph

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	8	274
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	52	37
PF			
CPFF	2	1	1
FP	98	51	37

SCOPE, PURPOSE, STATUS & PROGRESS

This project involves the continued drilling of special hydrological research, test and monitoring wells.

The Contractor is operating 2 rigs on a double shift basis.

Seven of the nineteen wells have been completed.

1600 feet of drilling have been completed to date.

Contractor is beginning to recover lost time since adding a fourth drilling crew.

1260571

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75127

DATE 9-30-62

PROJ. NO.	TITLE	FUNDING
CGH-974	Analog Simulation Facility	62-a-1

AUTHORIZED FUNDS	DESIGN $	AEC $	COST & COMM. TO	$
$	CONST. $	GE $	ESTIMATED TOTAL COST	$ 1,600,000*

STARTING DATES	DESIGN	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN
	CONST.	DIR. COMP. DATE		CONST.

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide an appropriately sized and consolidated analog computer simulation facility for the Hanford complex. Initial application will be associated with startup programs for the NPR.

A preliminary project proposal requesting design money in the amount of $160,000 was submitted to the Commission June 12, 1962.

* Approximate estimate.

PROJ. NO.	TITLE	FUNDING
CAH-977	Facilities for Radioactive Inhalation Studies	62-k

AUTHORIZED FUNDS	DESIGN $ 13,500	AEC $	COST & COMM. TO	$
$ 13,500	CONST. $	GE $	ESTIMATED TOTAL COST	$ 140,000

STARTING DATES	DESIGN 1-1-63	DATE AUTHORIZED 9-24-62	EST'D. COMPL. DATES	DESIGN 2-15-63
	CONST. 5-1-63	DIR. COMP. DATE		CONST. 2-15-64

ENGINEER

FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide additional facilities essential to the conduct of Biology research programs involving the effects of inhaled radioactive particles. It will comprise an addition to the 144-F Building consisting of approximately 2000 square feet of indoor dog pens and supporting facilities and approximately 2200 square feet of outside dog runs.

The proposal was submitted to the AEC on 6-29-62.

Directive No. AEC-211, dated September 24, 1962, authorizing design funds in the amount of $13,500, has been issued.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75127

DATE 9-30-62

PROJ. NO.	TITLE	FUNDING
CAH-982	Addition to Radionuclide Facilities - 141-C Bldg. 100-F	62-1

AUTHORIZED FUNDS $ | DESIGN $ | CONST. $ | AEC $ | GE $ | COST & COMM TO $ | ESTIMATED TOTAL COST $ Not Established

STARTING DATES: DESIGN 12-15-62* | CONST. | DATE AUTHORIZED | DIR. COMP. DATE | EST'D. COMPL. DATES: DESIGN 6-15-63* | CONST.

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide an addition to the 141-C Building in 100-F Area to supplement the present radionuclide study facilities.

The building addition will comprise approximately 2500 square feet for laboratory facilities and controlled feeding plus for swine.

The project proposal requesting $14,000 for Title I and II design was submitted to the AEC September 14, 1962.

* Based upon AEC approval by October 15, 1962.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS $ | DESIGN $ | CONST. $ | AEC $ | GE $ | COST & COMM. TO $ | ESTIMATED TOTAL COST $

STARTING DATES: DESIGN | CONST. | DATE AUTHORIZED | DIR. COMP. DATE | EST'D. COMPL. DATES: DESIGN | CONST.

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

TEST REACTOR AND AUXILIARIES OPERATION

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for September was 0. Efforts to decontaminate the primary system continued throughout September. The reactor was discharged and the primary system D_2O drained. During the drain, foreign matter in the D_2O was removed by filtration and ion exchange.

Various means of decontaminating the system without the use of chemicals were tried with little or no success, and efforts were then directed towards chemical decontamination. Principle materials planned for use are oxalic acid and caustic permanganate. Hazards reviews, procedure preparations, systems preparations, and trial runs of the procedures using demineralized water were in progress at month-end. Plans were completed to truck all waste to a crib south of 200-East Area. All chemicals were received on site.

A facility was set up in the discharge water pit to decontaminate the fuel elements. Ultrasonic vibration with a detergent solution was used. Nine fuel elements were cleaned in this fashion until the program was deferred because of interference with primary system decontamination preparations.

Equipment Experience

Three outlet jumpers were replaced to provide test material for decontamination work. Considerable galling occurred between stainless steel threaded fittings. Two of three required refabrication.

Extensive effort in improving and marking flux monitor wiring circuits was initiated while the reactor was discharged. The work was 90% complete. Similar work on other systems was outlined.

One river pump was using excessive amounts of oil. Pump characteristics were referred to the vendor.

Preventive maintenance required 494 manhours or 9.5% of total.

Improvement Work Status (Significant Items)

Maintenance and Engineering personnel worked extensively on decontamination facilities and system preparations and very little progress was made on improvement activities.

Work Completed:

Outlet nozzle cap modification

Work Partially Completed:

Safety circuit ground and low voltage detector
Fueling vehicle hoist modification
Primary oxygen analyzer installation
Flanges for safety relief valves in helium system - 85%
Position indicating lights for convection cooling assist valve
Decontamination facility
Third exhaust air activity channel

Design Work Completed:

Enlarge chemical feed system
Primary pump recording ammeters
High pressure helium compressor inter-after cooler relief
Outlet nozzle bracing
Interlock between charge-discharge machine, shroud seat and discharge hoist
Control room ventilation scope
Fuel transfer system modifications
Modification to temperature alarm and data reduction system
Compressed air supply revisions

Design Work Partially Complete

Additional fuel storage and examination
Oil storage building
Boiler feed pump seals

Process Engineering and Reactor Physics

Computational studies of the change in K_{eff} of the PRTR with change in the primary coolant purity have been completed. The results are that a 1% increase in the D_2O purity, in the range from 90% to 100% will increase K_{eff} by 0.1%. Therefore, improving the quality of the primary coolant from the current nominal 94.5% to about 99.5% will increase K_{eff} of the PRTR by about 5 milli-k.

A technique has been devised for the computation of the rate of reactivity burnup in the PRTR. The results of the calculations are in qualitative agreement with observed burnup rates. Further work is planned in an effort to improve this agreement.

Procedures

Revised Operating Procedures issued		5
Revised Operating Standards issued		9
Temporary Deviations to Operating Standards issued		1
Revised Process Specifications accepted for use		0
Maintenance Manuals issued		0
Maintenance Inspection Sheets issued		4
Drawing As-built status	September	Total
Approved for as-built	98	837
Ready for approval		30
In drafting		32
Voided		76
		972 (corrected)
Scheduled for review		325
		1 299 (corrected)

Personnel Training

Qualification subjects	460 manhours
Specifications, Standards, Procedures	107
Fueling Vehicle	0
Maintenance Procedures	41
	608 manhours

Status of Qualified Personnel at Month End

Qualified Reactor Engineers	8
Provisionally Qualified Reactor Engineers	1
Qualified Technicians	6
Qualified Technologists	17
Provisionally Qualified Technologists	2

Plutonium Recycle Critical Facility

Work on procedural matters was emphasized in anticipation of AEC approval to operate. Revision of the safety rods to permit maintenance without requiring cell cover block removal was started. Work was completed to permit individual rod drops as needed for future physics tests. Additional relief was provided to the auxiliary moderator storage tank to prevent overpressurization. Modifications to improve readability of the galvanometer was 50% complete. Installation of indicating lights to signify safety circuit trip causes was completed. An interlock was removed which will now permit moderator pumping without on-scale indication from instruments. Work was completed to protect rod magnets and circuit relays from excessive voltage.

A total of 62 operator and maintenance manhours were devoted to training.

Fuel Element Rupture Test Facility

Project Status (Project CAH-862)

Construction is 99% complete. The water plant was tied-in and PRTR and Rupture Loop services are being supplied from it. The main loop was chemically cleaned. Substantial amounts of grinding wheel abrasive particles and cuttings were recovered. During design tests problems were encountered with oscillations of pressure under automatic control. Project progress was delayed by diversion of personnel to PRTR decontamination activities and by the fact that the loop demineralized water system was being used by PRTR.

Operation

Operation of the filter plant was satisfactory in all respects.

Operating procedure issuance was 15% complete. The rupture loop Technical Manual Chapter was issued. Total training time was 74 manhours.

GAS COOLED POWER REACTOR PROGRAM

Gas Cooled Loop

Project Status (Project CAH-822)

Project completion was 92%. The replacement heater was received, but installation work was hindered by PRTR decontamination activities in B Cell. One of the gas blowers successfully passed preliminary tests. Several punch list items were completed.

Operation

Issuance of operating procedures is 83% complete. All operating tests and job hazards breakdowns were completed and issued. Total training time was 16 manhours.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 18,286 hours. This includes 13,682 hours performed in the Technical Shops, 3,617 hours assigned to Minor Construction, 930 hours assigned to off-site vendors, and 57 hours to other project shops. Total shop backlog is 20,091 hours, of which 70% is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 4.2% (779) of the total available hours.

Distribution of time was as follows:

	Manhours	% of Total
Fuels Preparation Department	4,255	23.27%
Irradiation Processing Department	2,554	13.97%
Chemical Processing Department	397	2.17%
Hanford Laboratories Operation	11,080	60.59%

LABORATORY MAINTENANCE OPERATION

Most FPD maintenance functions performed as a service to HLO were transferred to Test Reactor and Auxiliaries Operation on September 1, 1962. This organization performs routine maintenance for HLO facilities and provides craft assistance to special fabrications in support of research and development work. Activities for the month were essentially routine.

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:bk

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
C. A. Ratcliffe	Remote Depth Measurement of Compacable Materials, HWIR-1552
D. E. Wood	A Directional Thermal Neutron Detector
J. J. Cadwell	Quick Disconnects for Tubing
J. J. Cadwell	Opening Mechanism for Sealed Containers

Manager, Hanford Laboratories