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HANFORD LABORATORIES OPERATION MONTHLY ACTIVITIES REPORT

JULY, 1962

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AUGUST 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

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HANFORD LABORATORIES OPERATION

MONTHLY ACTIVITIES REPORT

JULY, 1962

Compiled by
Operation Managers

August 15, 1962

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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: August 7, 1962

	At Beginning of Month		At Close of Month		Total
	Exempt	Salaried	Exempt	Salaried	
Chemical R & D	139	134	142	137	279
Reactor & Fuels R & D	175	170	176	169	345
Physics & Instrument R & D	97	60	97	60	157
Biology	37	57	38	57	95
Operations Res. & Syn.	19	4	19	4	23
Radiation Protection	41	101	42	101	143
Finance and Administration	120	96	119	98	217
Programming	16	3	15	3	18
General	3	4	3	3	6
Test Reactor & Auxiliaries	52	184	52	181	233
TOTAL	699	813	703	813	1,516

BUDGETS AND COSTS

July operating costs totaled $2, 137, 000 or 8% of the $28, 049, 000 tentative control budget. The budget is based on data contained in the initial AEC financial plan and the proposed departmental allocation of funds by the General Manager - HAPO.

Hanford Laboratories' research and development costs for July, compared with the tentative control budget, are as follows:

(Dollars in thousands)	July Cost	Annual Budget	% Spent
HLO Programs			
02 Program	$ 63	$ 1 069	6%
03 Program	6	175	3
04 Program	888	11 150	8
05 Program	87	1 293	7
06 Program	233	3 154	7
	1 277	16 841	8
FPD Sponsored	85	1 300	7
IPD Sponsored	87	1 325	7
CPD Sponsored	121	1 354	9
Total	$1 570	$20 820	8%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

In an effort to reduce burnout of the graphite, all Hanford reactors recently began operation under provisions of a Process Change Authorization. The effective reduction of air in-leakage was demonstrated by monitoring samples charged into three channels at C Reactor on June 3 and removed July 3. Weight-loss rates of all samples were less than 2% per 1000 operating days, whereas between March 16 and June 3 the weight-loss rate was as high as 25%.

It was concluded from visual studies performed with an electrically heated test section in a glass tube that little or no boiling would take place between the process tube ribs and the fuel elements in the Hanford production reactors under normal flow and heat generation rates up to 1-1/2 times normal.

The burnout heat flux for an NPR fuel element in which the inner fuel tube was positioned 80% eccentric toward the outer fuel tube was found to be approximately 50% of that for concentric placement of the inner fuel tube.

A PRTR Mark I fuel element was installed in a transparent process tube and was examined for vibrations induced by flow of cooling water past the element. No vibrations were detected with a stroboscopic light or with high speed motion pictures at flows up to 125 gpm. When vibrations were induced in the outlet nozzle a small movement of the fuel relative to the process tube was detected in the high speed motion pictures.

An investigation of the nuclear safety of the PRTR loaded with mixed uranium-plutonium oxide fuel was completed. It was found that the negative fuel temperature coefficient, which inherently operates to limit excursions, is nearly twice as large as the effective value for the present zone enriched core; the effective delayed neutron fraction is substantially reduced and ranges from 0.0046 to 0.0033 for the loadings studied; and temperature and void coefficients of the moderator appear to be slightly more negative than for the present loading.

Calculations of reactivity and fuel burnup for a plutonium-fueled fast reactor for spacecraft applications indicated that core life or control span are not significantly benefited by varying the isotopic composition of plutonium.

A preliminary study was completed of PRTR power level potential without major modification. It was estimated that PRTR power level could be increased to approximately 100 Mw at a cost of $100,000 or less and 2-6 weeks' shutdown time, depending primarily on what modifications are required to light water injection and primary system pressure relief equipment, and that approximately one year's time would be required for design and safeguards analysis, equipment procurement and modification, and processing the safeguards analysis.

Zircaloy-2 clad uranium rods having intentional striations of up to 30% of the cladding thickness have withstood cladding strains up to 3.5% without failure. Cladding temperatures during irradiation were 350 C or higher. These data are consistent with previous observations on Zircaloy-2 cladding irradiated at temperatures above 350 C.

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The experimental single tube, dual-enriched fuel element is nearing completion of its second cycle of irradiation in the ETR M-3 Loop. Peak specific power is calculated to be 130 KW/ft with a corresponding maximum fuel temperature of 410 C.

Metallographic examination of N-inner tubes irradiated to 3000 MWD/T is in progress at Radiometallurgy. Cross-sections of the tubular component do not show visible inner clad buckling and no deficiencies have been observed in the braze closure. Small cracks, typical of high exposure fuel, are evident in the fuel adjacent to the closure.

The high temperature ductility of sintered UO_2 plates was demonstrated by twisting them more than 180 degrees without fracture at temperatures above 2000 C.

Welds produced by laser radiation were evaluated. A continuously operating laser beam appears necessary to achieve results comparable to those of high voltage electron beam welding.

Photomicrographs of Zr-2 cladding samples from two irradiated PRTR fuel rods revealed what appeared to be excessive hydriding (150-200 ppm). However, vacuum fusion analyses of cladding from the same rods indicated 37-55 ppm. Non-irradiated Zircaloy having the same fabrication history contains approximately 35 ppm hydrogen.

High energy impaction of a powder mixture having the composition UO_2 - 40 mol percent ZrO_2 produced particles having a density of 9.35 g/cc (94.4% T.D.). A dispersion having an over-all composition of $UO_{2.005}$ was prepared by high energy impaction of a mixture of uranium powder and $UO_{2.04}$.

Two hundred 0.050-inch diameter, high density yttrium oxide spheres, fabricated by grinding high energy impacted yttria powder, were delivered to Irradiation Processing Department for preparing radiation sources for cancer therapy.

Uranium mononitride powder was high energy impacted in vacuum, at 1100 C and 500,000 psi, to yield a bulk material having density of 14.25 g/cc (99.5% T.D.).

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The University of Washington received assistance from HLO in the fabrication of test fuel specimens for experiments at the University, consisting of rolling of seven buttons of an alloy of aluminum and uranium-235 to 0.050 and 0.025-inch sheet, the shaping of six sheets to rectangles with a drilled center hole, and the forming of 0.025-inch discs to fit the center hole of each such sheet.

Tensile tests, interrupted at intervals for X-ray photographs, have been performed on five unirradiated single-crystal molybdenum specimens. The results will be utilized to determine deformation modes and to calculate resolved shear stresses for molybdenum.

Two general swelling capsules that have reached their goal exposure were discharged this month and two new capsules were charged. Two previously irradiated capsules are being disassembled in Radiometallurgy for specimen recovery and examination.

Flux monitors have been irradiated and discharged from the in-reactor creep capsules. New flux monitors were recharged to continue the flux measurements. An existing computer code is being adapted to calculate the actual flux affecting the creep of the Zircaloy-2 specimens.

Preliminary tests have shown that the alkaline permanganate, now recommended for the first step in decontamination of stainless steel, is very effective in removing chromium plate. It is important that chromium plated parts not be used in reactors to be cleaned by procedures using this reagent.

Hydrogen analyses have been received on zirconium samples exposed to simulated NPR gas atmosphere, in-reactor, for 167 days. Corrosion product hydrogen pickup in irradiated samples was approximately the same as for comparable samples exposed ex-reactor. Corrosion weight gains in the reactor were about twice as much as comparable ex-reactor samples.

A PRTR, UO_2 fuel element, exposed to average reactor flow rates in a single tube mockup, caused no detectable corrosion at points of contact after three months of operation with the fuel in the same location. Earlier tests had caused 1-5 mils deeps marks in the tube wall at points of contact with the fuel exposed as short as two weeks. Apparently some undetected change in conditions has occurred between this and previous tests.

Failure initiated at a 17-mil deep fretting corrosion mark in a room temperature burst test of a section of irradiated, 45% cold worked, PRTR pressure tube. No evidence of hydriding was found under this mark which was produced by contact between the tube wall and a single rod wire wrap.

Twenty-eight ZrO_2-PuO_2 rods were fabricated by cold swaging for a special irradiation cluster for PRTR irradiation. Two additional zirconium-clad, plutonium-zirconium alloy fuel plates were roll-clad for an extended surface fuel element. $PuO_{1.62}$ samples were heated to selected temperatures and cooled at various rates to study possible phase transformations. Fast cooling from 1000 C at room temperature yielded FCC alpha-Pu_2O_3 containing ~20% BBC alpha-Pu_2O_3. Slow cooling to room temperature in a vacuum from 500 C gave a two-phase mixture of PuO_2 and what appears to be PuO. Also, $PuO_{1.62}$ was observed to have a melting of ~2400 C in a helium atmosphere, compared with the apparent melting point of 2280 C for PuO_2 under the same conditions. The room temperature resistivity of PuO_2 has been determined to be 2×10^{10} ohms-cm.

Density plots of as-cast PuC alloys between 52 and O a/o C indicate a two-phase region between 52 and 49 a/o C. The density stays essentially constant over the region from 49 - 40 a/o C.

Single phase PuN which shows no metallic plutonium has been synthesized for the first time. Construction of a high temperature vacuum X-ray diffractometer was completed to carry out high temperature diffraction measurements of plutonium nitride.

A seven-gram sample of plutonium hydride was reacted with one atmosphere of hydrogen sulfide at 500 C for one hour and then heated one hour at 1500 C for homogeneization. The resultant produce had reacted with the tantalum crucible. X-ray examination of product material scraped from the remaining pieces of the crucible indicated that PuO_2 was the major constituent.

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2. Physics and Instruments

Analysis of buckling values of the NPR lattice has now been completed. Exponential pile measurements were performed in previous months on a mockup of the N-Reactor lattice and on a condensed version of this lattice. The condensed lattice, as the name implies, removes the voids which are present in the mockup, but retains the same amount of graphite per lattice cell. The buckling value for the condensed lattice is 2.1 times the buckling value for the mockup lattice. The difference in buckling is a measure of the effect of the voids in the mockup lattice.

Instrumentation activities in support of the NPR project included: sensitivity and shielding calculations for the top and bottom shield ionization chambers; reviews of specifications for all the nuclear instrumentation systems; further testing of the vendor prototype gamma spectrometer for the rupture monitor and recommendation of modifications; and calculations of shielding for the Traveling Wire Flux Monitor.

Eddy current testing of more than 28 miles of small diameter Inconel tubing for NPR was nearly completed. Preliminary correlations of the eddy current indications with burst tests indicate that up to 4% of the tubing may need to be rejected.

The White Bluffs equipment used for NPR process tube testing was transferred to HLO. This facility will be used for remaining NPR work and for limited testing of Zircaloy tubes for the K reactors.

Studies of the effect of overboring C-Reactor continued with a buckling measurement on a C pile mockup containing one control rod in the center of the exponential pile.

A program to measure the effect of neutron irradiation on K-Reactor control splines continued. One set of splines reached its goal exposure. PCTR measurements on the control splines will be made later in the year when all sets have accumulated the desired exposures.

Methods were recommended for improving the fission counters for the subcritical monitors and for applying fission counters as poison spline monitors at the existing reactors.

[redacted]

Useful data for the studies on reactor automatic control were obtained from tests at KW-Reactor, and progress was made in the development of a satisfactory analog model for the control and transfer function analyses.

Preparation, operation, and revision of various versions of the reactor kinetics code TRIP have continued in support of hazards studies on the production reactors.

Criticality experiments continued on plutonium nitrate solutions in a 14" spherical vessel. The effect of acid molarity on criticality was studied for solutions in the concentration range of 30 to 45 grams of plutonium per liter. An increase in the acid molarity from 2 to 4 increased the critical mass of the 14" sphere by about 15%.

The first load-out of dilute plutonium solutions from the Critical Mass Laboratory for transfer to a concentrator in 234-5 was accomplished without spread of contamination or other incident.

The effect of newly installed shielding walls around a storage hood in the 234-5 Building was measured using neutron multiplication techniques. The walls are flat tanks containing water. The results of the measurements show that it will not be necessary to reduce the number of plutonium pieces which may be stored in the hood.

A training course in nuclear safety was begun during the month at the request of CPD personnel. About 100 CPD Supervisors are to attend the course in groups of about 25. Each group will receive 10 lectures on nuclear safety.

The analysis of the criticality problem associated with dissolving plutonium continues and a theoretical study to compute the buckling of partially filled, reflected spheres was begun. The analysis method for calculating photon heating from Cs-137 in shipping containers was developed for Chemical Development Operation. An analog study of radioactive waste disposal heat transfer had to be suspended until sufficient computing equipment is available to obtain a solution with the necessary detail.

Analysis and evaluation of fuel burnup data from PRTR Pu-Al element 5075 continues. Possible problems associated with Pu and Cs-137

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concentration measurements are being investigated. Burnup analyses analogous to those performed on Pu-Al element 5075 have begun on UO_2 element 1041, which was exposed to about 2000 MWD/T, for checking laboratory methods for analyzing UO_2 and UO_2-PuO_2 elements.

Approach to critical measurements were made using Pu-Al rods in a light water lattice with a .85" spacing between rods. The inner zone of the loading was made up of 211 rods containing 5% Pu-240. The outer zone was made up of 240 rods containing 6% Pu-240. The extrapolated number of rods for criticality was 486. This number is two percent smaller than the number obtained for a uniform mixture of the two types of rods.

The analyses of the experiments of 5.0 w/o and 1.8 w/o Pu-Al rods in water have been completed. Excellent agreement was obtained in most cases. A paper on this work, entitled, "Plutonium-Aluminum Alloy Rods in H_2O: Comparison of Eigenvalue and Perturbation Analysis with Experiment," by J. J. Regimbal was submitted to the American Nuclear Society for presentation at the Winter Meeting.

The neutron temperature in a 19-rod fuel cluster was determined by activation of lutetium foils in the PRTR. The measurement was made between adjacent fuel rods in the outer ring of rods. The neutron temperature was found to be about 25^{o} C higher than the temperature of the moderator and coolant water which were near room temperature at the time of the irradiation.

Analyses of PCTR data on low exposure Pu-Al fueled graphite lattices continued. New calculations on k_∞ for a 10-1/2" poisoned lattice gave k_∞ = .974 as compared with the experimental value of 1.00.

Preparations for PRCF startup included work on test procedures, flux traversing rigs, and neutron counter assemblies. Assistance was also provided on hazards evaluations for two different PRCF loadings.

Code development on the two fuel cycle burnup codes, CALX and RBU, continued. The cross sections of selected isotopes in the RBU basic library are being updated. The library is used as a source of basic cross section data for other reactor physics codes to insure uniformity in these data used in analyses. Assistance is being provided to Programming Operation in analyzing a fuel element design having spatial

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self-shielding as well as assisting in fuel cycle analysis work through code development.

To test a basic concept in reactor theory, a measurement was made on the angular distribution of neutron current flowing into a strongly absorbing rod. Preliminary comparison of S_n transport theory predictions with the measurements has provided a contradictory evaluation of the level of reliability of the S_n analysis method. While good agreement with experiment is obtained on the angular flux distribution in one direction, non-physical flux oscillations are obtained in the opposite direction. Although it is felt that this instability may have only localized second-order effects upon the calculation of the important reactor parameters, until it is cleared up, future S_n analysis in curvilinear geometry should be carefully scrutinized.

Major portions of the work on "Fuel Re-Use" have been redone, using more realistic compositions and more detailed calculational procedures. The fuel concept still appears attractive.

The program for measuring the scattering of low energy neutrons from water molecules continued. Measurements were made to determine the effect of multiple scattering in the water sample on the experimental results. The data obtained to date are being analyzed and reports are being written for presentation at an IAEA Conference at Chalk River.

Initial tests were satisfactorily completed on an improved system for determining the power-density spectrum of reactor flux "noise". Work with this is expected to be applicable to design of an analyzer for determining the correlation between moderator-level and power-level fluctuations at the PRTR.

Cobalt-60 radiography at PRTR confirmed that a vane had broken out in the flow-straightening section of the primary loop. Equipment is also being installed for vibration measurements on PRTR piping.

Tests confirmed the anticipated advantages of a new graphical alternating current nulling device being developed for use with either conventional or multiparameter eddy current nondestructive testing equipment.

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In the development of ultrasonic methods for the nondestructive testing of fuel sheath tubing, good correlations were obtained between experimental results and theoretical predictions based on postulated Lamb-wave propagation.

One of the new prototype shirt-pocket sized dose meters which audibly alarms was completed for testing. Another unit failed due to sticking of the ionization chamber fiber after an integrated dose of 1200 r. Research is continuing towards relieving this problem.

The study of radioactivity in Alaskan Eskimos is proceeding satisfactorily. Measurements are still to be made at Anaktuvuk Pass and Point Hope and a few more at Kotzebue in hope of finding some Diomede Islanders.

Off-site work in the atmospheric diffusion program at Vandenberg Air Force Base, California, was completed. One hundred eight experiments were completed in the three series at this site under a variety of meteorological conditions. These data provide not only the basis for the formulation of prediction methods in support of advanced missile firings, but are an important addition to the studies in basic diffusion as related to topographic and climatic parameters.

Promising results were obtained from preliminary experiments on use of silicon surface barrier detectors for monitoring airborne alpha emitters.

Tests of a new experimental scintillation transistorized, $1\text{-}10^6$ mr/hr, logarithmic response area radiation monitor led to modifications to reduce the sensitivity to temperature changes.

Installation of the recently developed Automatic Conveyor Type Alpha-Beta-Gamma Laundry Monitor was nearly completed and testing was started.

A special groove-depth microscope, with an accuracy and repeatability of $\pm$ 80 microinches, was assembled, tested, and delivered to the Lawrence Radiation Laboratory for use through a hood wall.

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3. Chemistry

Addition of silicate to a tube in KE reactor at a concentration of either 10 or 20 ppm results in a reduction of phosphorus-32 and arsenic-76 in the effluent by a factor of two to three. Laboratory studies indicate radioisotope reductions of a factor of 10 are potentially possible. The difference between the plant and laboratory results is yet to be resolved.

The water treatment plant for supplying "tailor-made" process water to two tubes in the KE reactor is now operational.

On the basis of laboratory studies, Purex process equipment was modified to permit alkaline permanganate washing of the second cycle solvent. Almost immediately, improvement in 2D-2E column performance was noted; higher pulse rates without flooding were achievable and waste losses improved.

Engineering studies have failed thus far to define the source of operating difficulties that are being experienced with the Purex plant formaldehyde reactor for the denitration of the 1WW waste stream.

Preliminary studies show that about 80% of the plutonium present in Purex 1WW can be extracted from the as-received waste by batch extraction with a D2EHPA-Soltrol solvent. After feed adjustment, about 80% of the neptunium present was extracted.

In behalf of waste management studies, the extraction of cesium and strontium from citrate complexed Purex waste by D2EHPA-Soltrol and the subsequent partition and recovery operations continue to show technical feasibility.

Although near quantitative precipitation of cesium, as cesium 12-tungstophosphate, from simulated Purex formaldehyde treated waste can be readily effected, phase separation efficiencies obtained in engineering scale centrifuge equipment are found to be only 70 to 80%.

The extraction of cesium from simulated Purex formaldehyde treated waste by clinoptilolite results in fission product decontamination factors of 2×10^3, 9×10^4, 1.4×10^5 and 7 for strontium, cerium, ruthenium and niobium, respectively. Complementary studies showed that clinoptilolite lost only 16% of its original cesium absorption capacity after eight process cycles over a 60-day period.

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Dissolution rates of fused, cesium bearing clinoptilolite in tap water at room temperature were found to be approximately equal to those of the most insoluble glasses.

Two hundred gallons of current, centrifuged Purex 1WW waste was obtained for use as feed to the radiant heat spray calciner installed in the High Level Radiochemistry Facility. Although the waste was found to contain about 13 volume percent solids, the particulate material is easily suspended and is not expected to cause excessive operational problems.

Miniature pulse column studies showed that cesium, after extraction into a dipicrylamine-nitrobenzene-tetralin solvent, could be back extracted readily into dilute nitric acid with a high (99.9%) recovery efficiency.

Characterization studies of sludge in the 108 SX tank revealed that the 14-inch-thick solid deposit consists of relatively hard and thin (2 - 3 inches) top and bottom layers with a softer center layer.

Glass-like solids can be produced by fusing Linde Zeolite 13X with a mixture of SiO_2, B_2O_3 and LiF. A volume reduction of about three is obtained.

Studies to determine the cause of low extraction efficiencies of radio-iodine from Redox stack gas by charcoal beds revealed the presence of two radio-iodine bearing components. The major component (98%) absorbs with a high efficiency ($\sim$ 90%); the minor component (2%) absorbs with a very low efficiency ($\sim$ 7%).

Engineering studies show the diffusion of molten chloride salts through walls of graphite containers can be prevented by the application of low external gas pressures.

An electrolytic technique was examined for the purification of contaminated molten chloride salts. Preliminary results show that 99.99, 97.5, 98 and 60 percent of the uranium, iron, plutonium and europium, respectively, can be removed readily.

The nuclides, Zr^{95} - Nb^{95} and Ru^{106} have been present in surface air all through this year at steady levels of about 0.1 d/m per cubic foot. Air samples in July showed the Zr^{95} - Nb^{95} concentration at the 12,000-foot level to be about 100 times the surface contamination.

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Calculations were made to compare the shielding requirements of fission product promethium relative to other radionuclides when used in a heat source generator producing 250 watts (thermal) with an allowable radiation level of 200 mr/hr at one foot. Three-year-old promethium was found to require about one-twentieth the lead shielding necessary for either Sr-90 or Cs-137 (0.2 inches vs. 4.5 inches).

4. Biology

Three hundred nineteen people from Point Barrow and 62 from Anaktuvuk Pass, Alaska, were counted in the whole-body counter. The Anaktuvuk people, who are nomadic and utilize much caribou, contained the highest levels of Cs-137 found to date. Total body burdens of permanent Eskimo residents of Anaktuvuk averaged 400-440 nc with a maximum of about 790 nc. The permanent Eskimo residents of Point Barrow averaged about 35-50 nc.

Thyroid adenoma was observed for the first time in a sheep that had received a single dose of I-131. The animal had been given 3 mc of I-131 orally in 1958 when it was six months old.

Following exposure of rats to I-131 aerosols, the maximum level of I-131 in thyroids was about 20% of the total deposited. Maximum thyroid levels of 2.5% of the total deposited were obtained when the I-131 was diluted with 10^6 atoms of stable I-127 per atom of I-131.

Gross examination revealed fatty livers in rats intravenously injected with 6, 12, or 24 mg citrated Np-237/kg body weight. Animals receiving the higher doses did not survive past 48 hours. Some of the rats which received 6 mg/kg died within 48 hours, but others survived to the 72-hour sacrifice.

Survival studies of rats which were treated with 1.5 mM/kg EDTA as the zinc, manganese, calcium, or nickel salt before receiving 800 r whole-body X-ray suggested that the protective effects of treatments employed may be dependent upon the cation rather than on the EDTA moiety.

X-radiation caused yeast cells to lose potassium to washing solutions. The capacity to retain potassium was restored, however, by placing

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the cells in glucose solution, indicating repair of the damaged K-absorptive mechanism.

The biological half-life of Ce-144-Pr-144 retained in the whole body from inhaled administration was 400 days in two dogs at 100 days after exposure. Aerosol and intraperitoneal injection administrations of DTPA were equally effective in clearing Ce-144-Pr-144 from the lungs and whole bodies of dogs after exposure to $Ce^{144}O_2$. Treatment was most effective when started immediately after exposure. Excretion of Ce-144 removed by DTPA is via urine.

No statistical differences were found in swimming performances of young Chinook salmon that had been reared in zero, three, or five percent concentrations of reactor effluent water.

Gamma emitters found in plankton from near Hanford, listed in order of decreasing abundance, were Mn-56, Cu-64, Na-24, Cr-51, La-140, Zn-65, Zn-69, Sc-56, Sb-122, Mn-54 and Co-60.

The fish disease, columnaris, was noted for the first time this season in fish troughs at both 100-F and 100-K Areas.

5. Programming

Revision of the MELEAGER physics code to improve its accuracy has been completed. In particular, the code can now be applied to evaluate reactors having higher plutonium enrichment than before. The revised code is being used not only to re-compute the plutonium values previously reported in HW-72217, but also to determine plutonium values when plutonium from an unlimited stockpile is used to enrich natural or tails uranium. It is intended that these new results can be obtained in time to be presented at the American Nuclear Society National Topical Meeting in Richland on September 13.

TECHNICAL AND OTHER SERVICES

One new case of plutonium deposition was confirmed by bioassay analyses during July. The total number of plutonium deposition cases that have occurred at Hanford is 292, of which 210 are currently employed. The new deposition case resulted from a plutonium oxide contaminated injury received by a CPD process operator in the 234-5 Building. The injury

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occurred when a plutonium casting was unintentionally flung from a lathe and sliced through the hood glove and a surgical glove on the man's hand, causing the injury. After two excisions, the plutonium in the wound was reduced to about one-tenth of the maximum permissible body burden. A total of about 1.9 μc Pu was removed from the injury by both excisions (the maximum permissible body burden for Pu-239 is .04 μc). It is not yet possible to provide a reliable estimate of the magnitude of the plutonium deposition because of the effect of treatment with DTPA which was administered.

There were 12 incidents at the 234-5 Building and 4 incidents in HLO facilities which required special plutonium bioassay sampling for 25 potentially involved employees.

Two employees were exposed momentarily to high gamma dose rates on the front face work platform at the 105-B reactor during the charging of a poison column control facility tube with the reactor operating. The charge machine broke free from the ball valve assembly and allowed reactor cooling water to flush several poison and aluminum pieces from the tube. Although the men left the work platform immediately, evaluation of the dosimeters which they were wearing indicated gamma doses of 0.78 r and 0.19 r. The exposure was believed to be due to three aluminum pieces that had been irradiated about three minutes before the backflow occurred.

Levels of I-131 in local milk returned to normal values of 3 to 4 μμc/l in early July following the sharp rise noted last month (maximum of 68 μμc/l on about June 20, 1962). Concentrations of fallout materials in air filter samples for the Pacific Northwest decreased steadily in July. The monthly average was 3 μμc/m^3 of air compared to the average of 6 μμc/m^3 noted in June.

The mathematical analysis and computer program which determines the steady-state nonviscous flow pattern of a fluid in a cylindrical tank has been completed. These developments will enable numerical studies of the effectiveness of various axially-located circulating devices in connection with the concentration of suspended waste products.

In connection with the preparation of metal blanks to be shear-spun on a Floturn machine, a computer program was written which specifies the precise geometry of the blank, the lathe coordinates required to machine the blank, and the lathe coordinates required to machine mold surfaces for casting blanks. The program is designed to specify blanks which (1) have

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uniform or minimum shear during the turning process, and (2) meet the practical considerations involved in casting.

An extensive analysis is being made of the relationship of warp to hot spots on fuel elements in the various reactors. Preliminary results indicate differences in the relationship between reactors, probably due to the differences in annuli. The latter hypothesis is now being examined.

As a result of significant differences noted in the mass spectrographic measurements by two different laboratories, a modification of the routine monthly measurement procedure has been adopted to aid in the elimination of bias and/or to identify unexplained variance.

A study of statistical properties of railroad accidents is being made based on data received from the ICC. The ultimate objective is to obtain the probability that a given freight car will sustain a given impact.

A joint paper was written which describes the data logging system for and calibration of the gamma absorptiometer used to assay the uranium concentration of pulse column aqueous and organic phases.

An experimental design was devised for the estimation of the radial void distribution in tubular fuel elements from micrographs of cross sections of the elements.

A mathematical model has been fitted to data associated with liver damage in sheep. Further analysis is being done to provide tolerance bands for the curves obtained.

Statistical analysis of pencil data for the first four months of 1962 was completed. A technique is being developed whereby individual urinalysis samples will be tested for Pu content only if a composite sample fails to meet acceptable criteria.

Authorized funds for 13 active projects amount to $2, 744, 600. The total estimated cost of these projects is $9, 117, 000 of which $1, 484, 000 has been spent through June 30, 1962.

[redacted]

SUPPORTING FUNCTIONS

The PRTR was shut down the entire month to effect repairs to the damaged flow straightening vanes noted late in June. Complete reactor discharge and draining of the primary system was accomplished without incident. During various stages of the outage, advantage was taken of system conditions to inspect and clean the primary system of all foreign objects, and to conduct a total power failure test, containment tests, and special process evaluation tests.

One of the flow straightening vanes which consisted of 10 gage sheets, 2 feet long, with unsupported widths of approximately 6 inches was found to have come loose. Failure was attributed to fatigue. Five major pieces of the vane were found and determined to comprise all of the vane. One short section was still intact in the vane assembly, one large section was lying in the vane assembly, two triangler (~4" x 6") were located in the lower ring header, and one small (~1" x 2") piece was located in a process tube jumper. The replacement assembly was made up of a bundle of 2 inch stainless steel tubing, 2 feet long.

About 50% of a planned wiring improvement program in the PRTR control room was completed during the month. The improvement program involves installing a better quality wire marker on about 1500 wires (old gum type labels were becoming illegible or falling off), removing about 70 unused wires, and rerouting others for easier and quicker trouble shooting.

The 2400 volt cable to primary pump motor #3 (single speed) shorted during startup, destroying a section of the cable. The short occurred where the molded rubber plug is vulcanized to the cable insulation. The cable had been in use only a short time. A spare new cable was installed and all three motors and the wiring between the motors and breakers were "megged" to ground as well as phase to phase. No defects were found. No damage occurred to the motor, its connection box, or its breaker.

After reactor discharge special flow tests were conducted to determine the flow required to backseat the primary pump discharge check valves and the primary pump bypass check valve to ascertain that light water injection flows would provide adequate cooling under certain postulated emergency conditions. It was found that over 700 gpm was required and that this was not satisfactory. The valves were modified during the month such that the bypass valve would be positively closed by an external actuator and the other valves would be more nearly closed under no-flow conditions. Subsequent tests resulted in 30 - 50 gpm flows closing the check valves.

A total power failure test was conducted at PRTR. Emergency lighting, DC power and compressed air were found to be satisfactory. Instrumentation was also observed to function as needed. Air supply was shown to be adequate for approximately two hours.

Total productive time for the Technical Shops was 21, 572 hours. This includes 16, 104 hours performed in the Technical Shops, 3975 hours assigned to Minor Construction, 1303 hours assigned to off-site vendors, and 190 hours to other project shops. Total shop backlog is 21, 929 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 4.8% (1137) of the total available hours.

Four Ph. D. applicants visited HAPO for employment interviews. Four offers were extended; four acceptances and nine rejections were received. Current open offers total five. Eight direct placement offers were extended to BS/MS applicants; seven acceptances and two rejections were received. Five program offers were made. At month's end, four direct placement and eight program offers were open.

Six Technical Graduates were placed on permanent assignment; eight members were added to the roll and two terminated. Current program members total 56.

The AEC's Inhalation Toxicity Group met at Hanford on July 5 and 6. Eighteen scientists from AEC-supported laboratories attended.

On July 29, at approximately 4:15 p.m., a fire occurred in the climatizer room of the 108-F Building. The fire was attributed to a defective ballast serving the light system. The fire was restricted to one room, and total damage to room and equipment was estimated to be about $20,000. Radioactive isotopes (tracer quantities) were contained and no spread of contamination was detected. There was no reported injury associated with the incident.

W. K. Woods
for Manager
Hanford Laboratories

HM Parker: WKW: mlk

REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATION

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 2000 PROGRAM

1. METALLURGY PROGRAM

Corrosion Studies

Corrosion of Titanium Bonded Fuel Element End Closures. Coextruded, Zircaloy-clad, fuel elements with titanium bonded end closures are currently being corrosion tested. One group of elements is being exposed to 400 C, 1500 psi steam while the second group is being subjected to 360 C water. Two weeks of exposure in either environment indicates the fuel elements have excellent corrosion resistance with no signs of accelerated attack.

Autoclave Testing of NPR Fuel Elements. Following the etching and autoclaving of several batches of NPR production fuel elements in the 333 Building facility, grey areas were found on some of the fuel elements. Investigation disclosed that these grey areas were due to several causes:

1. Scuff marks from Teflon rollers.
2. Marks on the ends of the fuel elements produced by bumping the elements against the basket during etching.
3. Contact 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.

To evaluate the effect of these grey areas on the corrosion integrity of the fuel, representative fuel elements are currently being autoclaved in 360 C deoxygenated pH 10 water. Following seven days of exposure, the copper contaminated areas were slightly greyer in color. Other grey areas were unchanged; however, the brazed and welded end closures were white and flaking. These welded areas had an acceptable black corrosion film when the elements were placed in the autoclave. Further evaluation of these fuel elements will be made following additional exposure.

Metallic Fuel Development

Fuel Irradiations. Five N-inner fuel tubes (NIE), which were successfully irradiated to 3000 MWD/T under N-Reactor operating conditions in a KER loop, are being examined. The superficial appearance of these elements was good, with no evidence of outer surface bumps, crud, clad striations, or warp. The brazed closures were free of whitish deposits. The measured average fuel swelling was 1.7 ± 0.5 u/o and the average OD change was +0.007 inch, accounting for about three-fourths of the measured volume increase. The average length change of the 23-inch elements was +0.005 inch. Visual examination of the 0.44-inch diameter bore of the tubular element indicated a slight degree of inner bore buckling. A few small cracks, typical of highly exposed uranium, were found in the fuel. The closure appeared sound and unaffected by its irradiation service.

The experimental single tube, dual enriched fuel element which had previously received one cycle of irradiation in the ETR was recharged into the M3 loop of the ETR for resumption of the irradiation test. Temperature and flow measurements of the loop water indicate that the element currently is operating with a maximum uranium temperature of 410 C.

Radiometallurgical examination is continuing on the variable braze thickness irradiation test, GEH-4-68, 69 and 70. Two of the three fuel elements have been sectioned. All closures examined appear to be in good condition. The uranium is extensively cracked in the braze-heat-affected zones of some of the closures, but none of the cracks propagate into the cladding and only one crack extended into the braze between the cap and the core. No indications of incipient failure have been observed.

Fluted NPR Single Tube Element. The die and billet components for the fluted NPR single tube element are in fabrication. Components for a dummy of this shape are also being fabricated. The dummy will serve the purposes of checking out the die, providing flow test material and, it is hoped, cap material for the fuel element. The billet is designed with a Zr-2 core with unbonded Zr-2 cladding which is thicker than that on the fuel element. After extrusion and stripping the clad, the Zr-2 core should fit the inside contour of the fuel element clad.

Cladding Deformation Studies. Diameter measurements have been completed on 12 irradiated fuel rod specimens which had intentional striations machined in the Zircaloy-2 cladding prior to irradiation. Three of the fuel rods, each containing U-2 w/o Zircaloy increased in diameters approximately 3.6 percent. No indications of necking or plastic instability of the cladding could be observed. Cladding surface temperatures were approximately 360 C for these elements and fuel exposure was approximately 1500 MWD/T. Initial calculations from the diameter measurements indicate that the greatest total cladding strains took place in the specimen with the deeper striations.

The other nine fuel specimens, which contained unalloyed uranium, increased in diameter a maximum of 1.4 percent. Again, the fuel rod with the deeper striation had the largest total cladding strain. Although only a cursory visual examination was made on these last elements, no indications of cladding instability could be seen. Cross-sectional metallographic specimens will be cut from the rods to determine if the cladding strain is associated with the machined striations.

Fuel Element Swelling Model. A mathematical model for the swelling of uranium is being empirically evaluated using test data from fuel element irradiations. The model accounts for burnup and temperature distributions and considers the constraint of the uranium to be equivalent to a pressure. The restraining pressure is a power function of temperature and burnup and has a factor which may be physically identified as a cut-off temperature. Thus far, data from twelve fuel irradiations have been used to statistically evaluate the parameters in the model. The model is strongly dependent on the rate of increase of the fuel temperature above the cut-off temperature.

Heat Treatment Studies. The as-extruded length of fuel elements from tube 1024 was compared with their beta heat-treated length and these data were compared with the as-extruded thermal coefficient of expansion of the fuel. The change in length of a two-foot element due to heat treatment was +0.015-inch for the front of the extrusion, -0.008-inch one-third of the way down the extrusion, and +0.040-inch at the rear of the extrusion. Thermal expansion of the as-extruded fuel elements showed similar trend. The coefficient was 13 x 10^{-6} per degree C at the front, decreased in the central section, and then increased to 14.5 x 10^{-6} per degree C at the rear. Samples from four tubes extruded under conditions similar to those for tube 1024 provided thermal expansion data to determine if 1024 was typical of all extrusions made at 1170 F, 14-15 in./min ram speed, and 800 F die temperature. Only one of the four tubes showed as much variation in expansion coefficient as tube 1024, thus indicating that 1024 is not typical. This study does indicate, however, that change in length on heat treating provides an easy means of detecting large variations in as-extruded texture. The small amount of data available also indicates that when there is large variation in length change on heat treating, the tendency to warp on heat treating is also large.

Four NIE and one NOE uranium coextrusion billets were beta heat treated as a final fabrication step before coextrusion. The treatment consists of a single beta phase heating in chloride salt and water quenching.

A second NON Fe-Al additive ingot was beta treated, then alpha soaked three hours and water quenched prior to primary extrusion. N-outer tube coextrusions from the previous Fe-Al ingot are currently being heat treated in a variety of conditions to observe the effects on compound size and dispersion, grain size, clad-core bond and hardness. These results are intended to provide data for both element heat treatment and future billet heat treatment.

The effect of nitrate salt quench bath circulation and temperature on the quench rate of NIE fuel was determined. There is no appreciable effect at standard quench temperature (300 C), but an increasing effect on cooling rates is observed at lower temperatures.

<u>Fuel Component Development</u>. The susceptibility of N-Reactor fuel assemblies to vibration was examined. It was found that the tendency to vibrate in the lowest frequency mode dominated. The measured natural frequencies of the fuel components agreed well with calculated values based on spring constants of the supports measured statically. The fuel assembly is now being analyzed as a two-degree of freedom system to determine the displacement of the supports as a function of process tube motion, support spring constants, damping, and forcing frequency. The object of this study is to establish a basis for specifying spring constants for both the inner and outer fuel tube supports.

<u>Fuel Deformation Studies</u>. A series of laboratory tests designed to evaluate the effects of surface irregularities on the mechanical behavior of Zircaloy-2 cladding was completed. Burst tests of extruded tubing with internal notches of controlled geometry indicate that any notch producing more than 10 percent reduction in wall thickness results in a pronounced reduction of the total elongation accompanying fracture of the tube. Sharp notches (0.005-inch radius) were more effective in this regard than notches having a one-eighth inch radius.

<u>KVNS Self-Support</u>. The proposed KV self-support is a "suitcase handle" type rail with the bridge section forged in an inverted channel section to provide adequate strength. The die has been proofed on 2s aluminum, and approximately 300 supports of X-8001

aluminum alloy have been fabricated. Inspection of these units indicates a need for some die modification to eliminate some of the metal flash and produce better dimensional control. The present supports are satisfactory for testing the basic support properties of strength, collapsibility, and welding stability. A sample lot (approximately 100 supports) has been provided FPD to examine welding behavior and furnish FFDO with welded supports for strength and collapsibility tests. No modifications will be made on the die until preliminary testing indicates the present design to be a satisfactory approach.

N Outer Fuel Element Support. Testing of the N outer fuel element support is continuing in an attempt to determine the cause of process tube scratching in the 314 Building mockup. Testing is proceeding as follows:

Test IV-A

a. Process tube length	5 feet
b. Lubricant	Static process water
c. No. of fuel elements	22
d. Charging	One at a time
e. No. of charging passes (total)	135.

Periodic inspection (approximately every 11 elements) showed no damage for the first 66 charging passes. Between the 66th and 77th pass some light scratching was noted but did not increase in the 78th through 135 passes.

Test IV-B

a, b, c.	Same as Test IV-A
d. Charging	Short column
e. No. of element passes	92.

Test IV-B differs from IV-A in method of charging. A short column of three elements was in the tube continuously, the downstream element being displaced by charging a new element into the tube. Some light damage was noted in this test also. The light scratching began at the "charge" end of the tube and disappeared three feet down the tube. This slight damage does not appear to resemble the severe scratching encountered in the 314 tests. Additional testing is planned to determine if scratching is length sensitive before another test is run in a full length tube. This interim test will be carried out in a 17-foot length of process tube in the 306 Building.

Hot-Headed Closure Studies. During July 1962, effort was shifted from the joining of end caps to discs to the joining of end caps to hot-headed fuel element sections. Three approaches were investigated:

1. Caps to fuel elements without interface material.

2. Caps to fuel elements with thin vapor-plated copper at the interface.

3. Caps to fuel elements using copper shim as interface material.

Although the search for the optimum bonding conditions is not complete, the most significant results seemed to occur with the use of three-mil thick copper shims. Preliminary metallographic examination of one of the best samples at 100x and 400x showed good encapsulation of the copper shim and good contact at the Zircaloy-to-Zircaloy and uranium-to-Zircaloy interfaces. Metallographic examination of the bond is proceeding.

Resistance-Brazed Closure. Use of a modified "self-brazing" process, in which the plated caps are "tinned" by heating in vacuo prior to assembly and electron beam welding, has resulted in greatly improved metallurgical bonding in the closure zone. However, there is a tendency for the uranium to be extruded, or dissolved in the alloy and carried with the molten braze material into the annuli between the cap and sidewalls of the fuel element. This is thought to be detrimental from the standpoint of corrosion, although it would offer the advantage of announcing an incipient rupture during irradiation by evolving radioactive corrosion products. Cap modifications designed to prevent flow of uranium into these annuli are being studied.

Glass Extrusion Lubricants. The possible use of a low temperature softening glass as a lubricant for coextrusion work is presently being evaluated. The glass would be used to eliminate the need for the coextrusion billet to be canned in copper, requiring only that the Zircaloy-2 components be sealed to protect the uranium. The glass would be applied to the billet prior to preheat to provide a protective covering during preheat as well as provide the necessary lubrication for extrusion.

Three lead glasses are being considered for a suitable glass lubricant. Glass #1 had a chemical composition of PbO, 59.2 percent; SiO_2, 28.2 percent; CuO, 3.3 percent; Na_2O, 9.3 percent. This glass gave indications of having good lubricating properties around 650 C with a softening point of approximately 350 C. Glass #2 had an addition of five percent PbO and a five percent reduction of SiO_2. This change had an effect of lowering the softening point of the glass to about 330 C. Glass #3 had a further increase of five percent PbO, giving a composition of PbO, 69.2 percent; SiO_2, 18.2 percent; CuO, 3.3 percent; NaO, 9.3 percent. Glass #3 had a softening point of about 310 C, but upon quenching the molten glass, small particles of lead separated from the melt as the glass system had apparently reached its lead saturation point.

Each glass batch was hand mixed and melted at 1050 C in an induction furnace with a total firing time of 40 minutes. After a water quench the glass was ball milled for six hours, screened to -325 mesh, and mixed with a 50 percent H_2O, 50 percent ethyl alcohol solution to give a mixture suitable for brush application. The glass was then applied to a zirconium billet and fired in a muffle furnace to determine its softening point and lubricating potential in the 500-700 C range.

The lead glasses have to date only been tried on three solid Zircaloy-2 tubular billets at approximately five to one reductions. It was necessary to heat the billets to 700 C to insure extrusion on the 700-ton press; therefore, only glasses #1 and #2 were used. The billets were prepared as follows:

Billet #1 - Billet initially copper displacement plated and painted with glass #1 emulsion.

Billet #2 - Billet vapor blasted and painted with glass #2 emulsion.

Billet #3 - Billet vapor blasted and painted with glass #1 emulsion.

Billet #1 had the lowest extrusion constant and the least galling to the die with billet #2 having the highest extrusion constant and the most galling to the die. The mandrel did not gall in any of the three extrusions. The glass appeared to readily vapor blast from the extrusions.

Rolling of Cerium. A cast cerium billet was rolled from a one-half inch thick slab to 0.065-inch sheet. The first attempts to roll this apparently ductile (lead-like) metal failed. The cerium cracked into many pieces before a 20 percent reduction had been made. The metal may be successfully rolled if a 500 C vacuum anneal is made after each 10-12 percent reduction. The cerium was placed in a long quartz tube which was connected to a "Megavac" pump. One end of the tube, containing the cerium, was placed in a muffle furnace at 500 C. Cooling one end of the tube allowed it to act as a condenser for any volatile matter that might be given off by the cerium.

2. REACTOR PROGRAM

Gas Atmosphere Studies

Hydriding of Zircaloy-2 in Hydrogen-Carbon Monoxide-Helium Mixtures. Zircaloy-2 coupons have been exposed up to six months to a simulated reactor helium gas mixture contaminated with hydrogen and carbon monoxide and a dew point of about -30 C (0.1 to 0.5 mm H_2O partial pressure). No gas phase hydriding was observed at temperatures up to 400 C for a variety of surface pre-treatments including vapor blasting, etching and pre-autoclaving. However, it was not completely clear from these data whether the observed inhibition was due to the water vapor or the carbon monoxide or both.

To resolve this point, Zircaloy-2 coupons with the same three surface pre-treatments (vapor blasting, etching, pre-autoclaving) were exposed at 325, 375 and 425 C to a helium-hydrogen-carbon monoxide mixture. Water vapor and oxygen were quantitatively removed from the gas stream by passing the He-H_2 mixture over 650 C calcium metal and the C.P. carbon monoxide over 1200 C graphite. The gases were then mixed to make a He plus 4% H_2 and 2% CO mixture.

After two weeks of exposure, gas phase hydriding was found in all the 425 C samples, the etched and vapor blasted 375 C samples and the vapor blasted samples at 325 C. Almost no protective ZrO_2 film was formed during the two-week exposure and most of the observed weight gains could be accounted for by the hydrogen absorbed.

The hydriding observed and the lack of protective film formation in the He-H_2-CO mixture is in sharp contrast to the previous experiment where water vapor was present. This indicates carbon monoxide is definitely inferior to water vapor as a hydriding inhibitor and is inadequate as an oxidant in the absence of hot graphite. However, CO may still contribute to inhibition by reacting with H_2 in the presence of hot graphite, $CO + H_2 \rightleftharpoons H_2O + C$, to form the required water vapor.

Effect of Heat Treatment on Electrical Resistance of ZrO_2 Films. Previous data have been reported which showed beta quenched Zircaloy-2 to have a lower hydrogen absorption rate than the as-received material. Electrical resistance measurements have been made on the 70-day corrosion films of samples in these two conditions. At 450 C in low pressure (25 mm) water vapor, the ZrO_2 film on the as-received material had a resistance of 5 meg ohms compared with 200 K ohms for the beta quenched material. The energies determined from a log R versus 1/T plot give 17 K cal for the as-received material and 12 K cal for the beta quenched oxide. This very interesting result offers an explanation as to why beta quenching reduced hydrogen pickup. If the conductance and electron mobility of the oxide is increased by beta quenching, then the two negative charges brought to the metal by the oxide anion (O^{--}) during corrosion can be neutralized by electron flow back out from the metal through the oxide rather than by proton (H^+) diffusion inward to neutralize this negative charge. Increasing electron flow relative to proton flow would reduce hydrogen pickup. One possible explanation as to why beta quenching could increase electrical conductivity would be that beta heat treating (1010 C) takes second phase alloying ingredients into solution. The resulting more uniform distribution of foreign atoms in the ZrO_2 film increases the number of active sites to provide electrons to the conduction band.

In-Reactor Hydriding Corrosion Capsule. Eighteen Zircaloy-4 and 18 Zircaloy-2 samples exposed to simulated NPR gas atmosphere (He, 0.5% CO, 0.5% H_2, 0.05% H_2O) in the reactor for 167 days have been recently analyzed for hydrogen. Two analyses of each sample were made, and a third analysis was made on eight samples where agreement between the first two analyses was poor. Agreement between the hydrogen analyses was generally very good. In-reactor samples showed corrosion weight gains about twice that of comparable ex-reactor samples. The percent corrosion hydrogen pickup was approximately the same for irradiated and ex-reactor samples for the same temperature and environment. Corrosion rates increased with increasing temperature with samples at 400 C, especially the Zircaloy-4, showing signs of spalling oxide. Fourteen of 18 irradiated Zircaloy-4 samples showed somewhat lower hydrogen pickup than comparable irradiated Zircaloy-2. The four cases where Zr-4 hydrogen pickup was higher were at the higher temperatures where the Zr-4 was corroding more rapidly. The fact that the hydrogen pickup fractions were generally in the 10-50% range found for ex-reactor corrosion data, indicates there was negligible gas-phase hydriding from the 0.5% H_2 in the gas. The water vapor inhibited gas-phase hydriding in-reactor in the same manner as was previously observed ex-reactor.

Graphite Burnout Monitoring. Recently all Hanford reactors began operation under provisions of a Process Change Authorization (PCA). The purpose of the PCA is to minimize air in-leakage and to reduce the inlet dew points. It is expected that these precautions will reduce the graphite burnout rates.

The effectiveness of reducing air in-leakage at C Reactor was shown by burnout monitoring samples inserted into Channels 1889, 1960 and 2780 for 14 operating days between June 3, 1962, to July 3, 1962. No sample lost weight greater than 2 percent per 1000 operating days, whereas between March 16, 1962, and June 3, 1962, the greatest rate loss was 25 percent. Some weight gains (up to 3.5 percent per 1000 operating days) were noted on samples from Channel 1889 downstream from the point where the maximum weight loss occurred.

Burnout rates from small graphite samples in 3580-F from June 24, 1961, to July 2, 1962, were measured. The highest weight-loss rates were approximately 0.6 percent per 1000 operating days both at 40 and 160 inches into the graphite stack; however, the region between 50 and 100 inches where it is now known that the reaction between oxygen and graphite is likely to occur was not monitored in this test. Also, these rates may not be representative of the lower portions of the stack. Experience at C Reactor with monitoring channels in the top and bottom portions of the graphite stack indicates that highest burnout rates occur in the lower regions. Therefore, to gain confidence in burnout monitoring at F Reactor a lower facility is needed.

At 1880 KW between May 5, 1962, and July 3, 1962, maximum burnout rates were approximately 4 percent per 1000 operating days; these were measured about 4 foot downstream from the 0.130-inch coring zone where the highest temperature would be expected. The channel was recharged with enough samples to thoroughly monitor this zone and to help clarify these unpredicted measurements.

Burnout rates in 3478-D between May 28, 1962, and July 10, 1962, were reduced by one-half from those between February 8, 1962, and May 28, 1962; nevertheless, the maximum rate was still 20 percent per 1000 operating days in this last test period under the PCA. Again, a monitoring facility in the lower portion of the reactor might indicate an even higher burnout rate.

Corrosion and Coolant Systems Development

Corrosion of Chromium Plate in Decontamination Solutions. Samples of chromium-plated stainless steel, presumably the same as portions of the NPR primary pumps which may be chromium-plated, were exposed to an alkaline permanganate solution for 2 hours at 105 C. Each of six samples lost about one mil of plate thickness in this solution, whether previously defected or not, and considerable dark film was formed on the samples. Exposure in a 90 g/l solution of Bisulf-16 at 85 C removed the dark film.

Fuel Element Rupture Tests. Two NPR inner tube elements were tested in TF-9 to determine if there was any difference in the manner and degree of rupturing with 100 ppm Fe and 50 ppm Si added to the U core of one of the elements. Both elements were defected close to the Zr - 5 w/o Be braze with a 0.025-inch pinhole. Three runs of one hour each at 300 C, 1800 psi and 20 fps were made. After one hour, the uranium core element had ruptured considerably worse than the alloy core element. A blister about 1-inch long by $1\frac{1}{4}$-inch wide with a weight loss of 13 grams was present on the uranium core element and on the alloy core element the blister was about 3/8-inch in diameter with a weight loss of 5 grams. After three hours, both elements were ruptured around the entire circumference for about 2 inches down the element. Weight loss of the uranium core element was 236 grams and the alloy core element was 327 grams.

Long-Term Corrosion Tests for NPR. A new test has been initiated in TF-1 to determine the corrosion rates of A212 carbon steel, 304 stainless steel, and Zircaloy-2 in 300 C deionized water with the pH adjusted to 10.0 using ammonium hydroxide. One-half the carbon steel samples were polished before charging, the other half were polished and then soaked in inhibited HCl for 10 minutes before charging. The purpose of the acid soak is to evaluate the effect of surface treatment on corrosion.

Samples of A212 carbon steel which had previously been exposed to six cyclic decontaminations using alkaline permanganate followed by Wyandotte-5061 (one week exposures to pH 10, 300 C water followed by a decontamination) were also charged in TF-1 to determine the uniform corrosion rate following a series of decontaminations. This test will determine if carbon steel returns to its normal pre-decontamination corrosion rate of 0.1-0.2 mil/year, or if the rate remains high as indicated from a plot of corrosion penetration vs number of cyclic decontaminations.

A low-temperature corrosion test at NPR graphite cooling system conditions (but without the hydrogen addition) is in progress in TF-4. Corrosion rates of Zircaloy-2, A212 carbon steel and 304 stainless steel are being determined. After 1200 hours, the rates of the stainless steel and Zircaloy are undetectable ($<$ 0.01 mil/yr); the corrosion rate of the carbon steel is less than 0.05/mil/year; crud release is very low.

A test to determine corrosion rates of steels and Zircaloy in water adjusted to pH 8.0 with LiOH is continuing in TF-5. After 869 hours of exposure, the corrosion rates of the carbon steel (A212, 1051 and 313472) are less than 0.1 mil/year. The corrosion rates of all materials are comparable to those obtained at pH 9 or 10.

Corrosion Testing at 900 F. The results of the first test at 900 F in TF-15 were obtained during the month. After 1900 hours in neutral water, Hastelloy-X and D979 are corroding at very low rates ($\sim$1 mil/yr); stainless steels (304, 316, 406) are corroding at rates $<$12 mils/year; and carbon steel rate is about 24 mils/year. The Zr-2 samples corroded at a low rate for about 40 days and then underwent a rapid acceleration in rate.

Non-Uniform Corrosion. Several inlet and outlet sections of KER-1, 3 and 4 main loop heat exchangers were descaled and examined for corrosion. Numerous pits up to 1/16-inch deep were found on the carbon steel surfaces exposed to the raw cooling water. All other surfaces, including weld and crevice areas, appeared to be in excellent condition. No stress cracking was found.

A stainless steel tubing line which failed on TF-5 was examined by the Metallography Laboratory and definite stress cracking was found. The loop has operated the past few years at pH 8-10, using LiOH except for one 3-month period at neutral pH. The history of the loop operation will be examined to determine if the presence of chloride ion in the loop may have been possible.

Sodium Ion Detection. Provisions are being made to analyze the NPR secondary coolant for sodium ion as a method to detect leakage into the system. The standard method is the flame photometer, but another system, the sodium electrode, is being evaluated as a check method. The sodium sensitive electrode measures sodium ion concentrations in the same manner, and with the same equipment, as standard electrodes measure the hydrogen ion concentration. Data obtained to date indicate that the electrode does not give reproducible readings for identical solutions at approximately neutral pH levels. The recommended procedure involves adjustment of the solution pH to 9-11 with gaseous ammonia or morpholine before the sodium concentration is measured. This minimizes the interference caused by hydrogen ions present in neutral solutions and gives much more stable operation. Solutions adjusted to pH 10 with morpholine gave readings that were reproducible for sodium ion concentrations between 10^{-1} and 10^{-5} molar. Evaluation of the effects of using different pH conditioning agents and different solution pH values is now in progress.

Analyses of Dissolved Oxygen. In previous testing a great deal of trouble was experienced with the dissolved oxygen analyzer, to be used at NPR. The instrument consists of a galvanic cell with silver and zinc electrodes. Oxidation of the zinc electrode releases zinc ions and reduction of the dissolved oxygen produces hydroxide ions in the flowing water stream. An electrolytic cell is an integral part of the analyzer and is used to generate oxygen gas for calibration of the analyzer. In previous tests the analyzer did not correctly measure the oxygen concentration generated during calibration.

From the tests completed this month it was demonstrated that the measuring cell had been passivated by exposure to too much oxygen ($>$250 ppb). Additional tests demonstrated that the analyzer will work satisfactorily at water temperatures from 70-85 F, that calcium carbonate is a satisfactory material for adjusting the inlet water conductivity, that the accuracy is within $\pm$ 2 ppb in the 0-100 ppb range and within $\pm$ 1 ppb in the 0-20 ppb range, that water flow rate variations of $\pm$ 5% of the desired rate can be tolerated without decreasing the accuracy (except during calibration), and that the response time for sensing oxygen concentration changes is only 1-2 seconds.

It is possible that this same instrument may also be used for the continual analysis of hydrazine.

Structural Materials Development

Stress Rupture Tests of NPR and KER Tubing. Under biaxial stress conditions, NPR tube sections continue to exhibit creep characteristics superior to those of 15 percent cold worked Zircaloy-2 strip tested under uniaxial stress. To produce comparable secondary creep rates, the hoop stress on the tube section is 10,000 to 15,000 psi greater than the stress in the uniaxial test. Further, under these same conditions, the primary creep strain of the tube is less than that of the conventional creep specimen. The primary strain and secondary creep rate of a KER tube containing 30 percent cold work was greater by a factor of four than NPR tubing with 15, 18 and 35 percent cold work. Data are not available for a direct comparison of the KER tube and strip material. Since the majority of the tubes for the NPR were fabricated by essentially the same process as this KER tube, some concern was felt that they might also exhibit this creep behavior. Tests have started on a 30 percent cold worked NPR tube and after 64 hours the primary creep strain is slightly less than for any of the other types of NPR tubing under comparable test conditions, and better than the above KER tube by a factor of six. Both pre- and post-irradiation data will be obtained in an effort to explain the unpredictable behavior of the Zircaloy-2 in this KER tube.

Graphite Distortion

NPR Graphite Irradiations. The series of long-term irradiations of NPR graphite continues to progress satisfactorily. The two capsules, H-6-1 and H-4-2, currently being irradiated in the General Electric Test Reactor are operating satisfactorily with all thermocouples functioning. Construction has started on the second of the second generation capsules, H-5-2. This capsule will contain eight samples previously irradiated in H-5-1, four samples previously irradiated in H-4-1, and 12 new samples.

Thermal Hydraulic Studies

Visual Studies of the Effects of Fuel Supports on Boiling. Further experiments were performed 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 latest experiments consisted of a 1.304-inch OD electrically heated tube placed inside of a 1.504-inch ID glass tube. Ceramic coated ribs fastened to the inside of the glass tube were used in some of the runs to simulate the reactor process tube ribs. The heated tube was positioned on the ribs in a manner similar to the way the fuel elements normally rest on the ribs in a process tube. Experiments were also run without ribs or centering devices to obtain data for comparison.

The experiments were run at flow conditions corresponding to those in the annulus of fuel elements in K Reactor tubes. Visual examination and high speed motion pictures of the test section were made while the heat generation was gradually increased.

From the experiments it can be concluded that little or no boiling is initiated between the process tube ribs and the fuel elements and that away from the ribs with a full annulus width there is no surface boiling at normal or even $1\frac{1}{2}$ times normal heat flux conditions. The extensive and vigorous boiling which has been observed previously with pressed down self-supports or bumpers coated with aluminum oxide electrical insulation cannot be readily explained unless the presence of aluminum oxide initiates boiling at a much lower temperature than would otherwise occur.

Similar studies will be made with the following devices for centering the fuel elements:

a. BDF "suitcase handle" bumpers.
b. BDF elliptical bumpers.
c. C overbore self-supports.
d. 0.50-inch long alumina ribs.

<u>Hydraulic Tests</u>. Failure of the threads on the outlet elbows from the C-Reactor rear nozzles has prompted the design of a new fitting which clamps to the nozzle without the use of integral threads. Flow tests were conducted using this new elbow design to determine the pressure drop and compare it with the pressure drop across previously used outlet elbow and pigtail assemblies. The results of these tests show that the new elbow has a higher pressure drop than the previous assembly but by reaming the ID of this elbow to enlarge the flow area around the temperature measuring elements the pressure drop of the new elbow can be reduced to an acceptable value. No evidence of flashing or critical flow conditions was detected under normal reactor operating conditions.

<u>Heat Transfer Characteristics of NPR Fuel Elements</u>. The studies to determine boiling burnout conditions for the NPR tube-in-tube fuel elements were continued. Six boiling burnout points were obtained with an electrically heated model of the fuel element with the inner fuel tube positioned 80 percent eccentric toward the outer fuel tube. (Percent eccentricity is that portion of the normal annulus thickness that the inner fuel piece is displaced from a coaxial position toward the wall of the outer fuel piece.) This degree of eccentric placement approximates that expected should the center portion of the support pieces break off leaving only the weld tabs to separate the two fuel tubes.

The experiments were conducted at 1500 psig. The data were plotted and compared with previous data for this same annular case but having concentric placement of the inner tube. For mass flow rates from 1,000,000 to 4,000,000 lb/hr-sq ft, the burnout heat flux for the eccentric annulus was consistently between 42 and 50 percent of the burnout heat flux for the concentric case. However, for a mass flow rate of 500,000 lb/hr-sq ft, the burnout heat flux determined for the eccentric case corresponded very closely to that determined previously for the concentric case. For mass flow rates of 500,000 and 1,000,000 lb/hr-sq ft upstream burnout was obtained for all three data points Upstream burnout was not obtained with mass velocities of 2,000,000 to 4,000,000 lb/hr-sq ft. Failure of the test section prevented continuation of the experiments.

B. <u>WEAPONS - 3000 PROGRAM</u>

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.

C. REACTOR DEVELOPMENT - 4000 PROGRAM

1. PLUTONIUM RECYCLE PROGRAM

Thermal Hydraulic Studies

Calculations of Temperatures for PRTR Fuel Elements. Calculations were made of Mark-II-C tubular fuel element jacket temperatures for the case of a maximum UO_2 core temperature of 2500 C. Members of Ceramic Fuels Development Operation are conducting a study of exposures to which such fuel elements could be irradiated before pressures from fission gases and sorbed gases might cause the inner jackets to collapse. To perform their calculations it is necessary to know the jacket temperatures.

Two limiting cases were chosen: (1) a maximum core temperature of 2500 C, and (2) a maximum surface heat flux of 800,000 B/hr-sq ft. For the 2500 C limit case equations were developed for determining the heat generation density and the fraction of heat flowing to the inner and outer surfaces of the fuel tube. The calculations indicated that temperatures at the water cooled surfaces of the inner jackets of both fuel tubes would be about 640 F, and that the temperature rise across the jacket would be about 200 F. Heat fluxes calculated for this case were in the order of 600,000 B/hr-sq ft. As this was a more limiting case, temperatures were not calculated for a heat flux of 800,000 B/hr-sq ft.

PRTR Flow Distribution. During reactor operation it has been found that an unequal flow distribution exists in the various quadrants of process tubes in the PRTR. In an attempt to explain this situation it was assumed that the inlet-tee in the bottom ring header was completely blocked off on one side such that all the flow proceeded in one direction around the header. In the top ring header it was assumed that the flow divided evenly at a point opposite the outlet and proceeded in two directions from that point toward the outlet. The pressure drops around the bottom and top ring header were then calculated to determine if they would be of such magnitude to cause the unequal flow distribution that was being encountered during operation of the reactor. The results of the calculations, however, indicated that the pressure drops resulting from such a blockage of the inlet-tee in the bottom ring header would not be of sufficient magnitude to account for the flow distribution being obtained during reactor operation.

In a further attempt to explain the distorted flow distribution in the PRTR, it was suggested that the fluid entering the inlet-tee in the bottom ring header had a warped velocity profile. Previous work had shown that a 10 psi pressure drop between the fluid leaving one side of the inlet-tee and the fluid leaving the other side was required to cause the flow distribution that was being obtained during reactor operation. Results of the calculations indicated that a relatively small distortion in the velocity profile at the tee could cause an unequal flow split resulting in the unequal flow distribution in the reactor process tubes.

A further possibility for a cause of the unequal flow distribution could be a variation in the size of the orifices which are installed at the inlet of each process tube. This possibility has not yet been thoroughly investigated.

Vibration Tests of PRTR Fuel Elements. A PRTR UO_2 Mark I fuel element was installed in a transparent "pressure" tube and was examined for flow induced vibrations. Flows of room temperature water up to 125 gpm were used. Visual examination with a stroboscopic light and high speed motion pictures were used to detect vibration. No flow induced vibrations were seen in any of the tests, indicating that if any vibrations exist they have a very small amplitude.

Further tests were made by inducing a vibration on the outlet nozzle at a frequency of 3500 cycles per minute. In this case small movement of the fuel element relative to the process tube could be detected in the high speed motion pictures. The motion was in a horizontal direction, but it could not be determined if this was a torsional or side to side motion of the fuel element within the tube.

Component Testing and Equipment Development

EDEL-1 Renovation. Overhaul of EDEL-1 continued to ready the facility for the study of PRTR pressure tube fretting corrosion. The boiling water pressurizer and its control circuits have been removed. Installation of the gas loaded back-pressure valve will be completed upon delivery of the valve. Replacement of seals and bearings in the Byron-Jackson circulation pump has been completed. The motor and variable speed magnetic drive for this pump have been returned to the factory for repair of cracked welds between the clutch poles and magnetic barrier, replacement of all bearings, and dynamic balancing. Overhaul of the 12 gph injection pump is complete pending the delivery of two bearings. All open electrical wiring on the back of the control panel has been removed. All electrical controls will be located in one panel section, the back of which will be enclosed. Design of the new safety circuit is complete and fabrication of the relay rack has been started. Installation of the new thermocouple system is 25 percent complete.

Design work for long-term modifications to EDEL-1, which includes changing the loop piping to stainless steel, has been halted for lack of funds.

Fretting Corrosion Investigation. Techniques to determine vibration characteristics of a PRTR pressure tube assembly in the prototype EDEL-1 test facility are being studied. Surface mounted velocity type pickup elements and accelerometers with appropriate amplifying accessories are suitable means for determining the vibration amplitudes and frequencies of pressure tubes and jumpers. Detecting and defining the vibrations of fuel elements will require development of appropriate techniques, however. Methods being considered include differential transformer, eddy current, electromagnetic, and acoustic types of signal transmissions.

Shroud Tube Replacement Mockup. Work on the mockup pit was stopped all month by a strike of construction ironworkers. Work on the shield and calandria mockup crates has been delayed pending availability of funds. Design layout of special tools for removing a faulty shroud tube continued.

Inlet Bellows to Process Tube Gas Seal. Additional leak rate tests were conducted on a pressurized stainless steel "O" ring seal. The B-F seal was tested an additional three thermal cycles in addition to that reported last month. Leak rates were as follows:

	Max. Leak Rates Liters/Hr @ 2" Hg						
Thermal Cycle	1	2	3	4	5	6	Bolt Torque
(1) Existing Copper "O" Ring	0.4	11.8	21.2	111	132		30 ft-lb
(2) Zirconium "O" Ring	200	260					30 "
(3) B-F Ferrule (Zirconium)	3.2	1.5	1.4				15 "
(4) Pressurized SST "O" Ring	0.4	0.4	36	50	49		15 "
(5) B-F Ferrule				0.6	0.6	0.5	15 "

Test results for Items (1), (2), and (3) were reported in the June monthly report. The B-F seal, Item (5), was the same seal as previously tested under Item (3). The additional three cycles were run after the seal had been disassembled and reassembled on the same section of the process tube.

An apparently satisfactory gas seal has been demonstrated in the B-F seal. However, due to its intricate shape and high cost, further testing of less expensive seals is contemplated.

Outlet Nozzle to Top Shield Gas Seal. A purchase order was placed for a compressible metallic gasket for the seal. Shipment is promised for early August, 1962. Drawing H-3-14547 has been issued for comment and shows a more rugged nozzle hold-down design.

Hazards Analysis

An investigation of the nuclear safety of the PRTR loaded with mixed uranium-plutonium oxide fuel was completed. A range of plutonium enrichment from 0.43 to 1.5 w/o Pu in the fuel was assumed.

It was found that the negative fuel temperature coefficient, which inherently operates to limit excursions, is nearly twice as large as the effective value for the present zone enriched core. The effective delayed neutron fraction is substantially reduced and ranges from 0.0046 to 0.0033 for the loadings studied. Delayed photoneutrons from the D_2O contribute significantly to these values. A decrease in the prompt neutron lifetime from the present value was also computed with lifetimes ranging from 0.5 to 0.3 milliseconds. Temperature and void coefficients of the moderator appeared to be slightly more negative than for the present loading. A detailed study of coolant void effects was also undertaken for coolant qualities from 0.25% to 10% H_2O. The coolant void coefficient changes from negative to positive as the amount of H_2O impurity increases. The cross-over point varied from about 6% H_2O for the lowest enrichment to about 16% H_2O for the 1.5 w/o Pu case. These values are believed to be pessimistic because of deliberate conservatism in the calculations. Some reduction in both moderator level coefficient and present shim rod strength can be expected with the mixed oxide loading.

Design Studies

PRTR Power Level Study. A document summarizing the results of this study was completed in rough draft form and issued for comments. It was estimated that PRTR power level could be increased to approximately 100 Mw at a cost of $100,000 or less and a two- to six-week shutdown time, depending primarily on what modifications are required to light water injection and primary system pressure relief equipment. It is estimated that approximately one year's time would be required for design and safeguards analysis, equipment procurement and modification, and processing the safeguards analysis.

Calculations were refined to determine the adequacy of existing equipment to permit reactor operation at power levels above 70 MW. In general, the results of the calculations confirmed earlier conclusions that the present cooling systems for the moderator, reflector, and shields would be adequate for 100 MW operation but not for 125 MW and that shim and shroud tube bellows temperatures would be acceptable at either of the power levels. The study disclosed, however, that additional pressure relief capacity would be required for the steam generator as well as for the primary coolant system at either 100 MW or 125 MW.

Preliminary calculations were made of the effect of higher power operation on adequacy of cooling following a total loss of power to the primary pumps. These calculations indicate that the cooling provided by forced circulation during pump flywheel rundown, and by natural convection after the pumps stop, should be as adequate at the higher power levels as at 70 MW. Although operation at higher power levels would increase the heat load following reactor scram, such operation requires a lower normal reactor inlet temperature and, thus, provides higher increases in coolant temperature through the reactor and thus higher convective heads.

Plutonium Recycle Critical Facility

Hazards Analysis. Reactor kinetic studies for typical PRCF core configurations were formulated. Analog simulated reactor excursions will be terminated by scram or inherent shutdown mechanisms. The inherent shutdown mechanisms used in these studies are the prompt negative fuel temperature coefficient and the moderator void coefficient. Moderator voids were assumed to have formed from both steam generation in the immediate vicinity of the fuel and radiolytic decomposition of D_2O.

Several analog studies terminated by reactor scram at a power level of 150 watts have been completed. The excursions were initiated by reactivity addition of 10 cents per second for 10 seconds with the reactor operating at 100 watts. Core configurations studied included cores with 100, 80, 60 and 35 percent of the fissions occurring in plutonium. In none of the studies was a measurable fuel temperature increase noted.

PRTR Rupture Loop

Component Testing and Equipment Development. New retainer flanges of 300 series stainless steel have been ordered to replace the present 17-4 PH retainers on the inlet valves. The leakage rate of the inlet shutoff valves at 20 feet of water back pressure is less than 1 cc/hr. This valve retains the water in the process tube during installation of the inlet cooling hose prior to fuel element discharge.

EDEL-II test facility was prooftested at 2100 psig and an apparent 600 F in preparation for rupture loop Grayloc connector thermal cycling test. Loop flow instability was encountered in the initial operation; subsequent testing showed the existing temperature readout to be faulty and the actual loop fluid temperature was approximately 640 F (near saturation temperature). Fluid flashing in the pump probably occurred in the initial test.

EDEL-II has been shut down to correct a high pump seal leakage rate. The seal shows only slight damage; however, slight eccentricity in the shaft exists. Since the shaft had been out-of-line and straightened previously, a new shaft has been ordered.

Dismantling the pump also disclosed a plugged condition of the cooling passage to the seal chamber exterior cooling jacket. This condition apparently existed at the time of manufacture. It is being investigated further.

Discharge Equipment. Hydrostatic pressure testing of the process tube end closures and hose assemblies has been completed. Results were satisfactory at 100 psig; this included the outlet elbow snap plugs, special three-way plug valve, and all hose assemblies and quick-disconnect fittings. Flow tests have yet to be conducted.

The shielded viewing cart has been completed and delivered to the 314 Building.

Tool fabrication is approximately 90% complete. Completion of the powered cutoff tool is pending receipt of a flexible drive shaft. Delivery of this item is expected July 24, 1962.

Modifications to the quick-disconnect coupling tool and valve operator tool are required to eliminate a physical interference. Individual tools have been given a preliminary test as they were completed.

Hazards Analysis. A proposed new limit on the release of noble gas fission products from the rupture loop was analyzed for the effect on the loop design and operation. This limit would allow up to 850 curies per day of noble gases to be exhausted from the PRTR stack. The total noble gases anticipated to be exhausted in a rupture test is of the order of 600 curies, within the proposed limit. Therefore, no change in the facility is necessary to meet the new limit.

PRTR Gas Loop

Component Testing and Equipment Development. "As built" work on the in-reactor test section drawings was approximately 70% completed. Various minor modifications to the discharge casks for the loop sample holders were completed. Design of a new transition piece between the loop nozzle and the sample casks was completed. Redesign of the nozzle closure to eliminate installation of the closure without locking it in place was completed, fabrication of one new closure is in progress, and the existing closure is being modified. Examination of the sample holder installation and removal tool discloses that it needs complete redesign. The sample container has been redesigned and a prototype is being fabricated. The spiral shielding plugs for the in-reactor test section have been shipped off-site for application of an aluminizing coating.

The first samples installed in-reactor in the Gas Loop will be a special assembly to determine the temperature distribution in the test sample as a result of gamma and neutron heating. Drawing H-3-14568 has been issued for comment showing a proposed design for this special test assembly.

Plutonium Fuels Development

Plutonium-Bearing Fuel Elements for PRTR. UO_2-PuO_2 fabrication during this period has consisted largely of continued swaging experiments to obtain further information on the control of final length. Small groups of tubes were selected from several tubing weight categories and cut to various lengths ranging from 62 inches to 63-3/4 inches. These groups were then filled with standard loads of UO_2-PuO_2 and cold swaged. Collection of data and evaluation of the information received is not yet complete. The aim is to be able to compensate for the variances in tubing weight and yet fabricate a rod to a length tolerance of $\pm$ 1/16-inch within a diameter tolerance of $\pm$ 0.002.

The incremental loading procedure for vibrational compaction was simplified by combining the coarse and medium UO_2 fractions so that each increment was composed of two additions rather than three. The loader binds when feeding -20 +60 mesh UO_2; however, -35 +100 mesh UO_2, mechanically mixed with -6 +20 mesh UO_2 does not jam the loader so that each loader cycle now feeds one complete increment. Fuel core densities of 86 to 88 percent of theoretical density were obtained with the altered size fractions when added in the same proportions as previously (70 percent coarse, 15 percent medium, and 15 percent -325 mesh mixed UO_2-PuO_2).

Two loaders were mounted on a sliding rack so that two tubes may be loaded at once and four rods may be compacted simultaneously.

PRTR Fuel Element Examination. Fifteen irradiated plutonium-bearing fuel elements were examined in the PRTR storage basin in July. The two main objectives of the examination were to gage the end bracket diameters and to determine the general condition of the clusters. Measuring the end bracket diameters over the spacing gussets was necessary to determine the amount of wear that has taken place between the fuel element and the process tube. Ring gages were used to measure the end bracket diameter. The original diameter over the end bracket spacing gussets measured 3.216 inches to 3.226 inches. All eleven aluminum-plutonium cluster end brackets inspected show some wear with the lower bracket being worn more than the upper bracket. Six of these also show the start of wear on the individual rod wire wraps, although no wire wear was found on the element which had the maximum end bracket wear. One of the four UO_2-PuO_2 clusters inspected also had maximum end bracket wear even though it had been under irradiation only two weeks. The general condition of the clusters was good, but some showed slight corrosion of the spot fusion welds.

Fuel element 5061 was found to have a broken wire at the end cap. This will be examined further to determine the cause of failure.

Experimental Elements for PRTR Irradiation. Twenty-eight ZrO_2-PuO_2 rods were fabricated by cold swaging for a special irradiation cluster. The core material consisted of fused, stabilized ZrO_2 in the following size fraction: -20 to +60 mesh (47 percent); -60 to +100 mesh (13 percent); -100 to +325 mesh (24.5 percent). This material was incrementally loaded into the tube along with a master blend of -325 mesh ZrO_2-PuO_2 containing 8.8 w/o PuO_2. Final composition of the fuel rods is 1.3 w/o PuO_2. Ttoal plutonium for the cluster will be ~322 grams. Three swaging passes plus several finishing passes were used to achieve the final length of 90-1/2 inch ± 1/16 inch. Total reduction in area is 42 percent. The rods are currently being grit-blasted and etched prior to final assembly steps.

Two additional zirconium-clad, plutonium-zirconium alloy fuel plates for an extended surface fuel element were made by roll-cladding. As in the previous plate, the aluminum foil wrapping used around the core for contamination control was eliminated. To minimize contamination which might occur during heating and early compaction of the "loose" assembly, the assembled sandwich was cold-rolled until the sandwich was tightly compacted. The plutonium core was thus "locked" in place to prevent movement of minute plutonium particles, which cause contamination. After cold-rolling, the sandwiches were hot-rolled in the usual manner. The cold-rolling was partially successful. One plate was not contaminated and one plate had slight contamination on the edge. Both plates were sheared to within one-fourth inch of the core without encountering further contamination and the plates appeared to be well bonded.

A metallographic examination showed the core thickness to vary approximately twenty percent except at the edges and ends where the core was thinner. The thinning results from the extreme softness of the core alloy. It is not believed that the thinning will present a serious fabrication problem. A suitable fuel assembly is being designed to irradiation test this fuel concept.

Several Al - 7.5 w/o Pu tubes have been hot extruded for use in the segregated plutonium test cluster. These tubes have a 0.610-inch OD and a nominal 0.460-inch ID. They were extruded at a 7 to 1 reduction in area over a tapered floating mandrel at 550 C. They will be drawn down to a final OD of 0.5 inch with a 0.030-inch wall. Earlier attempts to extrude the tubes at a greater reduction, to lessen the number of drawing steps, resulted in several mandrel failures. No mandrel or other tooling problems have occurred using the present extrusion conditions.

Irradiation Testing. Examination of the 42-inch long cosine enriched UO_2-PuO_2 seven-rod cluster is continuing in Radiometallurgy. Autoradiographs made from the irradiated fuel rods show areas of high fission product concentration due to plutonium segregation within loaded increments. Dark spots on the external surface of the fuel rods are associated with high plutonium concentration regions and result from localized areas of abnormally high surface heat fluxes. Transverse and longitudinal sections have been cut from selected areas on the fuel rods. Columnar grain growth and central void formation has occurred in some of the "hot spots". The voids are not necessarily located in the geometric center of the core indicating radial displacement of the PuO_2 enrichment during fuel loading.

The longitudinal sections taken through these areas are interesting in that the areas of columnar grain growth are not symmetrical, the void is not centrally located in the columnar grain growth region, the center void is elliptical and not circular, and the columnar grains in the radial direction are smaller than the columnar grains in the longitudinal direction. Considering the fact that the element was irradiated in the vertical position, the appearance of the longitudinal section suggests that the "hot spot" had been molten and the columnar grains and void formed during freezing of the liquid.

Autoradiographs made from the polished sections show no fission product migration in those areas which had equiaxed grain growth; however, some localized fission product migration has taken place in those regions where central void formation and columnar grain growth has occurred.

One capsule (GEH-14-85) containing high density UO_2 - 2.57 mole percent PuO_2 is being irradiated in the MTR. The present exposure is about 3×10^{20} fissions/cm^3 or 10,000 MWD/ton of UO_2-PuO_2. It was decided to discharge the capsule at the end of MTR Cycle 177 (about August 6, 1962) and not recharge the second capsule (GEH-14-86) because of the time delay to complete the required hot cell leak test prior to reinsertion. Tentative arrangements for return of the pair to HAPO for metallurgical examination have been made.

The bellows assemblies for the HELIOS Experiment were received from the vendor. Still no word from the vendor as to possible delivery of the modified thermocouple for the same experiment.

Uranium Fuels Development

High Energy Impaction. A modified Bridgman anvil technique for high energy impaction of powders in vacuum at 1100 C and 500,000 psi was successfully scaled up from one-inch to two-inch diameter specimen containers. Uranium mononitride, UO_2, UO_2-ZrO_2, and UO_2-U powder mixtures were successfully compacted. The particle density of the impacted UN was 14.25 g/cc (99.5 percent TD). The density of the UO_2 was 10.92 g/cc. A powder mixture having the composition UO_2 - 40 mole percent ZrO_2 produced 94.4 percent TD material that consists of ZrO_2 particles in a UO_2 matrix. A mixture of uranium powder and micronized produced impacted particles, comprising metallic uranium in a matrix of $UO_{2.04}$, with a density of 10.86 g/cc and an over-all composition of $UO_{2.005}$.

Tungsten carbide dies are useful in high energy impaction of powders, as they can be employed at pressures greater than those possible using conventional tool steel dies. Previously reported results indicated that a 12 percent cobalt-bonded tungsten carbide punch was satisfactory at impact pressures of 300,000 to 350,000 psi, but failed at 390,000 psi. A new, 25 percent cobalt-bonded tungsten carbide punch was used successfully to compact three pounds of UO_2 in a 2.5-inch diameter can at 408,000 psi. The micronized UO_2, which had an O/U ratio of 2.09 before it was heated and impacted had a particle density of 10.78 g/cc. The O/U ratio is not yet determined. It is encouraging that a technique is now available for high energy impaction of UO_2 to ultra-high density without the subsequent sintering treatment required when tool steel dies are used, and without using expendable die parts, as required with the Bridgman anvil technique. While further experiments are aimed at defining conditions necessary for preparing a final product of near-stoichiometric composition, previous experiments with similar UO_2 compacted by Bridgman anvil techniques showed that the O/U ratio could be reduced to less than 2.01 by hydrogen reduction at 800 C.

Ten pounds of ultra-high density, six percent enriched UO_2 particles were prepared by high energy impaction for use in an AEC-sponsored irradiation test at another site. The six percent enriched UO_2 powder was loaded into a four-inch diameter stainless steel can, heated to 1100 C, and compacted by high energy impact at 150,000 psi to a density of 10.63 g/cc. The compacted UO_2 was crushed to minus four mesh screen size and sintered 12 hours in 1750 C hydrogen to achieve an O/U ratio of 2.001 and a density of 99.3 percent TD (10.87 g/cc), based on the calculated density of 10.947 g/cc for UO_2 containing six percent U-235.

Yttrium oxide powder was compacted by high energy impact in an evacuated stainless steel container at 1100 C and 200,000 psi to a density of 4.36 g/cc (90 percent TD). The compacted material was ground to spheres 0.050 and 0.035 inch in diameter for irradiation experiments in a production reactor.

High Power Laser. The "Assistance to Hanford" contract with GEL on the evaluation of a high power laser for welding and cutting was completed. Individual laser pulses having durations between two and 12 milliseconds were directed into metal surfaces. A high peak power, short duration laser beam produced cavitation and evaporation of the metal and little retention of molten metal to effect a weld. Such a beam has application to cutting and drilling. A network storage system provided a laser pulse of 12 milliseconds maximum duration with lower peak energy values. Spot welds produced on the metal surface were similar in shape to spot welds produced by arc welding. A continuously operating high power laser (not yet developed) appears necessary to produce welds having high depth to width ratio comparable to high voltage electron beam welds.

High Temperature Ductility of UO_2. Permanent deformations (twists) of 180 degrees were introduced in sintered UO_2 fuel plates during torsional testing between 2000 and 2500 C. Strain to fracture varied greatly because of physical instability of the apparatus and non-uniform strain rates. These conditions are being corrected. Tantalum reacted too extensively with the UO_2 plates to serve as a replacement for tungsten as a heater material.

Thermal Etching of UO_2. At 1700 C, under one atmosphere of argon, preferential vaporization from UO_2 grain boundaries intersecting a polished surface yielded an excellent ceramographic etch. This treatment also caused a change in void shape from random, isometric to rectangular and triangular. Surface diffusion in the voids to approach a minimum energy shape (as dictated by the orientation of a grain) probably causes the change in void morphology. A five-minute vacuum anneal at 1700 C caused severe grain boundary etching and enlarging of many of the voids.

Thermal Conductivity of UO_2. The last group of a series of sintered UO_2 thermal conductivity specimens was successfully removed from the capsule in which it was irradiated during the past four years. Sample temperatures were maintained less than 100 C during the irradiation to avoid thermal annealing of irradiation effects. Exposure of the UO_2 is approximately 3×10^{20} nvt.

Burnup analyses, ceramography and annealing studies were begun. Two specimens (one low and one high density) were shipped to Battelle Memorial Institute for detailed thermal conductivity measurements.

The large UO_2 single crystal used earlier for thermal conductivity studies was irradiated at low temperature ($\leq$ 200 C) for the second time, gaining an additional exposure of 1×10^{15} nvt (total exposure, 2×10^{15} nvt). One end of the specimen was broken during pre-irradiation handling; subsequent squaring of the end (by ultrasonic sawing) reduced the specimen length to 1.00-inch long (compared with 1.77 inch originally). Post-irradiation examination of the UO_2

revealed several small surface cracks not previously evident. Such defects, that would affect thermal conductivity if they extended into the interior of the crystal, may have caused some of the conductivity decrease noted after the first irradiation. The crystal was returned to BMI for further thermal conductivity measurements at temperatures to 1200 C.

Irradiation Effects in Single Crystals. A single crystal of UO_2, the second in a series to evaluate physical property changes resulting from irradiation, was irradiated in the Hanford Snout Facility to an exposure of 1 x 10^{15} nvt.

Materials Development

Effect of Partial Dissolution of the Oxide Film on the Hydriding of Zircaloy-2. Zircaloy-2 coupons were autocalved to form a 25 mg/dm^2 oxide film, followed by heating in vacuum to partially dissolve the film. Samples were then autoclaved to a total exposure of 92 days. As reported previously, vacuum heating at 600 C for one hour resulted in loss of the protective character of the film as evidenced by a large increase in oxidation rate immediately after heating. Autoclaving for 92 days resulted in three corrosion cycles (115 mg/dm^2) for heat treated specimens, compared to two cycles (73 mg/dm^2) for specimens autoclaved but not heat treated. The corresponding hydrogen pickups were 248 (36% of theoretical) and 139 (30% of theoretical) ppm, respectively. Analysis of an unautoclaved Zr-2 specimen showed 16 ppm hydrogen. Normal hydrogen pickups are 30-50% for autoclaved Zircaloy-2 at the above exposures. Partial dissolution of the oxide film did not significantly increase the percent of hydrogen pickup.

Creep of Zircaloy-2 Pressure Tubes. Stress rupture tests were continued on three sections of the annealed portion of the PRTR pressure tubes. The test data are tabulated below and compared to that obtained by conventional creep tests of annealed Zircaloy-2 rolled strip samples. To produce comparable strains a hoop stress considerably larger than that of the uniaxial stress in the conventional creep tests is required at both 288 C and 343 C.

% Ultimate Strength	Hoop Stress 1000 psi	Hours	Temp. °C	% Uniform Strain	Stress for Equal Strain in Creep Samples from Strip, 1000 psi
85	39,600	4279	288	3.6	21,000
88	36,000	1571	343	9.0	23,000
75	36,200	2500*	288	1.3	17,000

*Terminated

Fretting Corrosion. A PRTR, UO_2 fuel element, exposed to average reactor flow rates in a single tube mockup, caused no detectable corrosion at points of contact after three months of operation with the fuel in the same location. Earlier tests had caused 1-5 mils deep marks in the tube wall at points of contact with the fuel in lengths of time as short as two weeks. Apparently, some undetected change in conditions has occurred between this and previous tests. A vibrator was received for use in tests to measure the effect of external vibrations on the rate of fretting corrosion.

In-Reactor Pressure Tube Monitoring. The present equipment for visually inspecting and measuring depths of fretting marks in the PRTR pressure tubes is being renovated as a result of the extensive in-reactor monitoring in June.

Radiation resistant lenses are being installed in the one extender section of the M-2 borescope presently equipped with standard lenses; the complete borescope protective tube arrangement is being redesigned for greater ruggedness and sized to permit inspection of the PRTR Rupture Loop tube in the future; a tube dryer is being incorporated into the probe; and fabrication has started on a more rugged and maneuverable dial gage assembly for measuring depths of fretting marks.

The Mark III in-reactor tube monitoring equipment, a refinement and combination of the two present probes, is 80 percent complete. The omniscope objective section has been returned to the vendor for recementing of the objective lenses which separated when overheated by the illuminating light. The light mounts are being redesigned to avoid a recurrence of this problem and to provide a sturdier unit. The gas gap instrument has been assembled. Experience with the new light mount and depth measuring device being developed for the present equipment will be used in the Mark III unit.

Post-Irradiation PRTR Pressure Tube Evaluation. Metallographic examination of a 17-mil deep fretting corrosion mark through which failure of a PRTR pressure tube occurred in a room temperature burst test indicated no increase in hydride content. In a previous sample, a 3-mil deep mark under which a 2-mil layer of hydride was found, altered the failure characteristics of the tube. Apparently, in the absence of hydriding, corrosion grooves up to 17 mils deep do not affect the strength of the tube.

Autoradiographic Examinations. Some orifices and inlet jumpers from PRTR were autoradiographed to determine the pattern of deposition of the radionuclides. The pictures taken during the examination indicated the possible presence of a long crack on the inside surface of one jumper. The jumper was decontaminated for non-destructive and destructive testing. To date, non-destructive tests have revealed no evidence of a crack.

Zirconium Concentration in the Primary Coolant. A program was established to obtain more frequent measurements of the zirconium concentration in the primary coolant in an effort to detect the onset of fretting corrosion. Daily samples will be analyzed chemically with approximately the same sensitivity obtained by radiochemical analysis (0.1 ppb).

Failure of Flow Straightening Vanes. The discovery of a small piece of stainless steel plate (1-inch x 2-inch x 0.140-inch) in the primary system led to the removal of the flow straightening vane immediately upstream of the main venturi. An investigation of the failed straightening vane is under way to determine the cause of failure.

All four pieces of the straightening vane were recovered, decontaminated in alkaline permanganate followed by ammonium citrate, analyzed, examined and metallurgically sectioned. The failed vane (5.5-inch x 24-inch x 0.140-inch) had failed by four main cracks with little or no deformation and one crack with extensive deformation. Two of the four cracks occurred at the two weld-vane interfaces. The other two cracks went diagonally across the plate from side to side. The fractured edges were all heavily coated with the black oxide normally associated with extended exposure to high temperature water. Metallurgical sections of fracture surfaces and adjacent areas reveal some cracks with a little branching. Although the transgranular cracks resemble stress corrosion cracks, the number of cracks and the extent of branching were less than is usually seen in this type of corrosion. The plate is exposed to a very turbulent flow of 230 C, pH-10, D_2O which contained some fluoride and chloride contamination for a short period about a year ago. Thus, several mechanisms of failure suggest themselves, namely, stress corrosion, corrosion fatigue, fatigue or any combination of these. More metallurgical sections are being processed in an attempt to determine the initial cause of failure.

A non-destructive examination of the remaining five vanes and the pipe wall revealed only one more short crack in an adjacent vane. Destructive examination of the remaining vanes will be started shortly.

2. PLUTONIUM UTILIZATION STUDIES

Plutonium Oxides

Several samples of $PuO_{1.62}$ were heated to selected temperatures and annealed at various cooling rates in order to study the possible phase transformations. Cooling to room temperature in a observed sign of the current carriers. All samples examined to date have shown p-type conductivity whereas oxygen-deficient PuO_2 should be n-type.

Plutonium Carbides

Density plots of as-cast PuC alloys between 52 and 0 a/o C indicate a two-phase region between 52 and 49 a/o C. The density stays essentially constant over the region from 49-40 a/o C. This can be explained by a defect NaCl type structure which becomes successively more deficient in carbon. The loss in carbon atoms is very nearly balanced by the cell shrinkage to yield a constant density. At 40 a/o C an unexplained sharp rise in density occurs in which the value jumps from 13.75 g/cc to 14.75 g/cc. From here the density values go almost linearly to the density of the starting metal. This indicates the appearance of a high density phase at 40 a/o C. The linear nature of the plot from that point to zero a/o C suggests a two-phase system in this region.

The x-ray patterns of these compositions also show a marked change at the 40 a/o C point. The PuC phase gives a well defined pattern in the 41 a/o C and higher samples, but in all those below 41 a/o C the high angle lines are very diffuse and difficult to read. Alpha plutonium lines appear in this region, but there are several lines which are not indexed either by alpha plutonium or the zeta lines reported by Mulford, et al.

The 40 a/o C structure reveals a needlelike substructure in the grain boundary phase. This may be the beginning of zeta formation. If so, it could help explain the break in the density curve (zeta is believed from dilatometric data to be a high density phase); it could also explain the sudden diffuseness of the high angle PuC lines. The acicular phase has become quite prevalent at 25 a/o C.

In an effort to obtain a more nearly equilibrium room temperature structure, the samples were heat treated 124 hours at 600 C in vacuo. They were cooled at 100 C per hour from 600 C to room temperature. These will be examined by x-ray and metallography.

Samples have been machined from different alloys for dilatometer and resistivity studies.

Plutonium Nitride

Effort this month was concentrated on the synthesis of high purity plutonium mononitride and the construction of an apparatus with which to carry out high temperature x-ray diffraction measurements in vacuum. The thermal expansion curve for PuN will be obtained from high temperature lattice expansion data. The plutonium mononitride to be used for these experiments was prepared by reacting -325 mesh plutonium hydride with gettered nitrogen at 600 C. The product is single-phase PuN by x-ray diffractometer analysis. The high temperature vacuum apparatus is complete and requires only leak checking prior to operation.

Plutonium Sulfide

Initial attempts to synthesize PuS will be by the reaction of plutonium hydride and hydrogen sulfide at elevated temperatures.

A seven-gram sample of plutonium hydride was reacted with one atmosphere of hydrogen sulfide at 500 C for one hour and then heated one hour at 1500 C for homogeneization. The resultant product had reacted with the tantalum crucible. X-ray examination of product material scraped from the remaining pieces of the crucible indicated that PuO_2 was the major constituent. The hydride reactant was in the furnace bell jar for several days while the vacuum system leaks were being repaired and was probably thoroughly oxidized during this time nullifying any further sulfiding attempts.

3. UO_2 FUELS RESEARCH

PRTR Thermocoupled Element

Examination of two fuel rods from PRTR thermocoupled test element #1072, TC-1 (200 MWD/T_U) was completed. Examination of fuel cross-sections from the center of each eight-foot long rod revealed microstructure nearly identical to that of nonirradiated, cold-swaged UO_2. Photomicrographs of the Zr-2 cladding from both rods revealed what appeared to be excessive hydriding (estimated 150-200 ppm). However, vacuum fusion analyses of cladding from the same rods showed only 37-55 ppm hydrogen. Nonirradiated Zr-2 having the same fabrication history contains approximately 35 ppm hydrogen.

Prototypic PRTR Fuel Element

Two additional single rod fuel elements were fabricated. High energy impact formed UO_2 was vibrationally compacted in three-foot long, 2.328-inch OD x 0.060-inch wall Zr-2 cladding. The proposed irradiation of these elements in the ETR P-7 loop includes three parts: (1) one element will be taken slowly to full power, held for three hours and discharged; (2) the second element will be taken to full power at a normal reactor startup rate and immediately discharged; and (3) the third element will be taken to full power at normal reactor startup rate and will be left in-reactor for the remainder of the cycle. These tests are intended to reveal information about fuel core melting during rapid reactor startup, and about the effect of columnar grain growth on subsequent central fuel temperatures.

Remote Fabrication

A welding turntable and welding power supply for remote fabrication of PRTR Mark II nested tubular fuel elements were received. The circuit of the latter is being modified to adapt the control sequences to the needs of the fabrication system. A smaller, less sophisticated, experimental version of the remote fabrication facility welding chamber is being constructed. The smaller chamber will allow tests of new design concepts before construction of the large chamber is undertaken.

4. BASIC SWELLING PROGRAM

Irradiation Program

Two general swelling capsules containing axially split, hollow cylinders of uranium were charged into individual test holes. Each is being irradiated at a constant temperature of 575 C. Two capsules that had reached their goal exposures were discharged and are in the reactor discharge basin to permit some radioactive decay before being shipped for disassembly. Two previously irradiated capsules are now being disassembled for recovery and examination of the specimens. The condition of these specimens will influence the choice of the type of samples to be used in future general swelling capsules. The construction of additional capsules of this type has been delayed due to the fact that the one-eighth inch electrical resistance heaters employed in the general swelling capsules have not yet been delivered. As there is no assurance of a definite shipping date, an attempt will be made to use heaters of one-sixteenth inch diameter that are on hand. Two heaters will be connected in parallel and wound in a double helix on the specimen chamber in lieu of the former single helix for the one-eighth inch heater. Minor design changes have already been made in the remaining capsule components and external connections to accommodate the different sized heaters.

An MTR prototype capsule containing a single cylinder of uranium has been assembled and is now being thermally cycled from 75 C to 600 C. It is planned to subject this specimen to 100 thermal cycles to determine their effects on a metallographically prepared surface. Depending on the condition of the surface the sample may be re-inserted into a capsule and be re-examined after additional cycles.

Post-Irradiation Examination

An axially split, hollow uranium specimen irradiated to a burnup of 0.4 a/o at a nominal control temperature of 575 C has been polished, etched, and replicated for electron microscopy. As previously reported, this specimen had increased markedly in volume and had assumed a lacy microstructure. Unfortunately, the replicas had picked up particulate radioactive particles from the specimen surface and due to their high level of radioactivity could not be processed. Additional attempts at preparing suitable replicas will be made.

Examination of replicas prepared from Zircaloy-2 clad U (enriched) - U (depleted) diffusion couples irradiated at low temperatures and then annealed at various temperatures has continued. Two levels of burnup in the enriched uranium layer of the coaxial couples are under study - 0.2 and 0.4 a/o. Replicas prepared from couples annealed at 700 C were unsatisfactory due to insufficient relief on the etched surfaces of the specimens. Accordingly, the specimens will be etched and replicated again. Specimens of the same type etched after annealing at 800 C have been processed for electron metallography. Porosity is far greater (greater number and size of pores) than was the case with similar couples annealed at lower temperatures. The microstructure in the specimens is typical of gamma-treated uranium. Porosity is present in the depleted uranium core section of the couples, where the burnup is estimated to be 0.02 and 0.01 percent. Porosity in some regions close to the original U-U interface is greater than in the central core region. Due to the non-uniform distribution of pores in the depleted uranium section, no reliable estimates of gas concentration gradients by quantitative metallography can be made. Pores very frequently appeared in a linear array in the depleted uranium section which may mark the position of some prior grain boundary. The specimen will be reground and reprocessed for metallography in order to substantiate the results obtained. No evidence of gas porosity in the Zircaloy-2 clad or Zr-U intermetallic zone was found. A gap between the uranium and the intermetallic zone immediately adjacent to the uranium, however, did exist. At the present time it is believed that this gap is due to pull out during polishing of a very brittle intermetallic between U and Zr.

A new approach to the quantitative determination of pore size distribution is being pursued to supplement the information already obtained with 0.29 and 0.41 a/o burnup specimens subjected to post-irradiation annealing. Additional measurements have been undertaken to corroborate and extend the existing data. The new method involves the assumption that the empirical distribution can be represented by a general log-normal distribution function with two adjustable parameters. These two parameters can be estimated through a quantitative metallographic analysis from simple counting measurements made upon the electron micrographs. The resulting distribution of three dimensional pore diameters can then be converted to the expected distribution of diameters obtained on a random section through the structure, compared with the distribution of section diameters that have been previously obtained empirically and tested for goodness of fit. The advantages of this approach are (1) increased statistical accuracy, (2) reduction of the three dimensional distribution from an empirical histogram to a mathematical formula which may be more easily manipulated, and (3) unbiased estimates of the total pore volume and total pore surface area in the structure.

5. IRRADIATION DAMAGE TO REACTOR METALS

Alloy Selection

An alloy whose mechanical properties indicate that it may have potential use in temperature environments to 1800 F is alloy R-27, developed by the Allegheny Ludlum Steel Corporation. A cursory examination of the effects of various oxidizing environments and irradiation at high temperatures is now in progress. Corrosion and oxidation specimens have been prepared with one specimen presently being run in 1700 F CO_2. Initial results from this test indicate that the alloy has good oxidation resistance in this environment. Additional specimens will be run during the next month to check the results of these initial tests. Tensile specimens of alloy R-27 are being prepared and will be irradiated in a gaseous environment at about 1200 F.

Four new high-strength, nickel-base alloys have now been received. These alloys include Hastelloy Alloy N, Haynes Alloy No. R-41, and Haynes Alloy No. R-235. These materials were received as 0.125-inch hot-rolled sheet. Cursory examination of their oxidation and corrosion properties will be made as well as a determination of the effects of irradiation upon their tensile properties.

Hanford Laboratories is to be responsible for the procurement, storage and disbursement of structural material samples to be used in the Irradiation Effects on Reactor Structural Materials Program, in cases where the use of the material is by more than one participating site. Steps were initiated during the past month to provide for storage and disbursement facilities for these materials. In addition, tentative specifications were written for Zircaloy-2 plate, Zircaloy-2 rod, types 348 and 304 stainless steel sheet and rod, and Inconel Alloy 600 sheet and rod. These specifications will be reviewed and submitted to the Materials Selection Committee for approval.

In-Reactor Measurement of Mechanical Properties

The in-reactor creep test conducted on a twenty percent cold worked Zircaloy-2 specimen at 250 C and 30,000 psi stress was terminated after 1407 hours. The test was concluded upon the failure of one of three specimen heaters. During the test four reactor shutdowns occurred. In two of these, creep rates were found to be greater than just after reactor startup. Experimental difficulties precluded creep rate measurement during the other two shutdowns. This behavior is similar to that of a test conducted on the same material at 310 C and 30,000 psi.

In the 250 C test transient creep was observed about 265 hours after startup in two reactor operating periods. The other operating periods were less than 265 hours in length. The transient creep curve was sigmoidal. Initially, the creep rate increased with time, then after about 40 hours the rate decreased. The first transient produced a plastic strain of about 0.03 percent, the second 0.15 percent. The second transient occurred at 1252 hours and nearly equaled the creep strain which had occurred before the onset of the transient. No experimental condition has been revealed which might account for the transient. Explanation may be related to the buildup of neutron-induced defects which decrease creep strength at 250 C. The effect appears to occur with regularity 265 hours after the startup of the reactor. The nature of such a defect cannot be accurately defined; however, various theoretical approaches to radiation damage have predicted vacancies will be produced by irradiation. The buildup of vacancies could enhance diffusion and therefore increase diffusion controlled creep. The "transient" effect will be studied further as part of the specimen can still be maintained at 250 C. The results will not be directly correlatable to isothermal tests; however, the existence of the transient can still be confirmed.

At 1408 hours the total plastic creep strain in the in-reactor test conducted at 250 was 0.280 percent. Nearly 0.18 percent of this occurred during the two transient periods. A parallel ex-reactor test conducted in an Instron creep machine accumulated 0.116 percent strain in the same time period.

Two flux monitors were discharged from creep capsules and new ones charged during a reactor shutdown this month. The monitors are presently being analyzed for calculation of flux values. Until this time monitors have not been successfully discharged. The monitors will be changed at regular intervals in the future.

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 51 tensile specimens irradiated in the G-7 ETR hot water loop were tested at room temperature. These specimens were of Zircaloy-2 and types 304 and 348 stainless steel with varying amounts of cold work. The raw data from these tests were programmed for electronic data processing. A shipment of an additional 226 irradiated specimens to Radiometallurgy from the ETR was made during the month. These consisted of tensile, notched-tensile, and bend test specimens of Zircaloy-2 irradiated in the range 10^{19} to 10^{21} nvt (> 1 Mev).

Initial comparisons of tensile data for specimens irradiated at ~280 C in the G-7 loop with data for specimens irradiated in an adjacent tube in ambient ETR water revealed only nominal differences in room temperature properties. For example, at an exposure of about 1.2 x 10^{20} nvt the yield strengths for 0, 20, and 40 percent cold worked specimens, representing the rolling direction and irradiated at about 280 C, were 70.8, 93.3 and 92.2 ksi, respectively. Corresponding values for specimens irradiated in ambient water to approximately the same exposure were 71.5, 96.5 and

101.0 ksi. The directional effects observed for the two irradiation temperatures were also about the same. For the annealed specimens the rolling direction exhibited a yield strength greater by about 10,000 psi than the transverse direction. On the other hand, yield strengths for the 20 percent cold worked specimens varied less than 2000 psi due to direction, which is within experimental error. The correspondence in uniform elongation between the two temperatures of irradiation was also excellent, and directional effects were consistent for both series of tests. In general, the annealed specimens exhibited about 4.5 and 0.8 percent uniform strain for the longitudinal and transverse directions, respectively. The values of uniform strain for the 20 and 40 percent cold worked specimens were about 1.2 and 1.5 percent, respectively, with little difference due to direction.

Radiation Effects on Tensile Properties of Nickel-Base Alloys

At the start of the alloy evaluation program, the four nickel-base alloys Hastelloy X, Hastelloy R-235, Inconel, and Inconel 702 were selected for a study of the effects of neutron exposure upon the tensile properties.

Series of duplicate subsize tensile specimens were prepared from the solution heat treated alloys. The unirradiated control specimens were heated on graphite supports in sealed, evacuated, and helium refilled stainless steel containers at 650 C (1200 F) for 500 hours to simulate reactor environment conditions. According to the original schedule, series of each alloy were to be irradiated for three and six months, respectively, prior to their being tensile tested at atmospheric temperature and at 700 C (1292 F). Instead of the planned six-month irradiation, however, the alloy specimens of the last test series accumulated a $16\frac{1}{2}$-month neutron flux exposure.

A comparison of the atmospheric temperature tensile test results of the irradiated and the unirradiated alloy specimens does not indicate any pronounced property changes due to radiation. The precipitation hardening alloys Hastelloy R-235 and Inconel 702, however, did lose considerably in ultimate strength and elongation after one year of exposure, while the property changes of the other alloys are moderate.

Testing at 700 C (1292 F) reveals more severe effects upon the alloy properties, particularly with increasing neutron exposure. In general, testing at this temperature, compared with the room temperature tests, results in lower yield and ultimate strengths and an increase in ductility, as expected, except for the simultaneous, very severe reduction in elongation of the Inconel 702 alloy. Exposure to neutron radiation appears to have a cumulative damaging effect upon the tensile properties of these four alloys when tested at 700 C (1292 F) which is especially pronounced in the Hastelloy X, Inconel, and Inconel 702 alloys, with losses of over 50% in ultimate strength and up to 80% in elongation. It is hoped that additional hardness and structural examinations of these alloy specimens will help to clarify their behavior at 700 testing temperature after the long term exposure.

Since the high nickel-base alloys are known to be sensitive to cold working and tool mark notching effects, such as result from the conventional preparation and machining of test specimens, effects of this kind may have contributed to the considerable scatter of the test results.

Damage Mechanisms

The objective of this program is to establish the nature of interactions between defects present prior to irradiation and defects introduced by irradiation, and to investigate the possibility of neutralizing the effect of impurity atoms by chemical stabilization. High purity iron containing interstitial impurities, such as carbon and nitrogen, and a chemical stabilizer such as titanium will be studied.

If specimens irradiated to various exposures are to be tested and compared so as to yield information on the role of interstitial impurity content and the role of the stabilizer, it is essential that specimens have a common and constant grain size. Experiments are therefore being performed to establish a procedure which will produce a controlled grain size in iron. The size of recrystallized grains obtained by annealing at suitable temperatures is being surveyed and an appropriate working and annealing procedure is being determined. Working by swaging originally appeared promising. However, metallography on transverse sections disclosed a swirled, spiral flow pattern in the metal; microhardness values varied from 92 to 148 D.P.H. This structure is unsatisfactory. Drawing as a method for introducing cold work in the iron will be investigated next. Dies for this operation have been ordered.

Precision x-ray measurements of the lattice parameter of the as-received high purity iron are desired; however, the grain size is such that conventional methods have failed. A modified single crystal technique is being employed, in which a single large grain in the polycrystalline sample is isolated by a fine x-ray beam. This technique has been only partially successful because of distorted peak shapes; however, further experiments are under way.

The swaged rod of iron has been annealed at 600 C and will be rolled into foils for electron microscope and x-ray diffraction studies. Techniques are being developed for thinning iron foils for subsequent examination in the electron microscope. Preliminary results indicate that only small iron specimens can be satisfactorily examined in the electron microscope equipped with an immersion objective lens. Large specimens due to their magnetic field distort the electron beam to such an extent that transmission electron microscopy is impossible. An electrolytic thinning technique which will produce many small specimens from a large specimen is currently being developed, and a jet electrolytic thinning apparatus is being designed for fabrication.

Ingot iron specimens which were decarburized in wet hydrogen at 1000 C, nitrogenized by annealing in an ammonia-hydrogen atmosphere, and then irradiated to a number of different exposures have been tested. Very little difference has been noted between the mechanical properties of the two materials after irradiation (decarburized versus decarburized and then nitrogenized). It is believed that impurity effects due to added nitrogen have been masked by the presence of other impurities. The uniform elongation and total elongation of the decarburized iron decrease with exposure at a slower rate than do the same properties of the nitrogenized material. The radiation hardening in both materials is essentially the same, i.e., increase in yield stress (0.2 percent offset) from 44,000 psi in the unirradiated condition to 62,000 psi at an exposure of 5×10^{18} nvt. Strain-aging experiments are now in progress on other specimens and may be expected to yield information on the more subtle changes, due to the addition of nitrogen, in the accumulation of radiation damage.

6. GAS GRAPHITE STUDIES

EGCR Graphite Irradiation

The fourth capsule, H-3-4, in the series of irradiations of EGCR graphite is scheduled for removal from the General Electric Test Reactor at the end of the current reactor cycle. The capsule continues to operate satisfactorily with all thermocouples in operation.

Flux Intensity Test

The irradiation capsule, GEH-13-8, designed to study the effect of flux intensity on property charges of graphite at a controlled sample temperature will be removed from the Engineering Test Reactor at the end of the current reactor cycle. At that time it is estimated that the maximum sample exposure will be approximately 1.5×10^{21} nvt ($E > 0.18$ Mev). The temperatures of the samples in the four positions having electrical heaters are being controlled at 650 C. The thermocouple in the #1 position failed (became open) during a recent reactor scram. Consequently, the temperature at that position is being controlled manually. All other thermocouples are functioning properly.

Speer Carbon Company Contract DDR-136

The first shipment of graphites produced by Speer Carbon Company under Contract DDR-136 was received. This contract is a continuation of the study of the effects of additives on radiation-induced dimensional changes in graphite. Additives used in the graphites just received include metallic aluminum and Al_2O_3 at the 5 percent level and three mixes containing 4.5 percent purified Fe_2O_3 plus 0.5 percent of one of the principal impurities found in technical grade Fe_2O_3. These are SiO_2, Al_2O_3 and Co_2O_3. Under the previous contract the purified Fe_2O_3 appeared to have less effect during processing than the commonly used technical grade Fe_2O_3.

A standard mix containing no additive was again prepared. Mix formulation was the same as previously used; however, processing was altered at several steps. A slower baking rate was used to reduce cracking; the impregnation step was eliminated as a variable which might mask additive effects; and to insure full additive effect the purifying gas was not admitted until the temperature had reached 2900 C. Other mixes containing a fluid-coke filler and several additives are currently being processed. These will give an indication of effectiveness of additives in altering dimensional behavior of graphites having high radiation-induced contraction rates.

Boronated Graphite Irradiations

A new capsule design has been completed which is capable of irradiating samples in the temperature range of 400-600 C. The principal heat source is the (n, alpha) reaction of thermal neutrons with the boron additive. This reaction generally encumbers capsule design since heating levels of 40 watts/g per 10^{14} nv are attained. This capsule design, once proven, will enable wide variation of heat input conditions and temperatures to be handled. The samples of boron-containing graphite are axially mounted in a graphite tube, and a set of graphite rings of proper width and radius control the radial heat flow and thus set the temperature of irradiation for the samples. Capsule venting is provided to prevent excessive pressure buildup of helium gas during irradiation.

In-Reactor Tensile Creep of Graphite

The magnitude of radiation-induced dimensional changes produced in graphite depends upon the temperature and neutron exposure. Since both flux and temperature gradients will occur in large blocks of moderator graphite, differential stresses are expected to develop caused by the differential radiation-induced strains. There is some concern that the stresses developed could exceed the strength and result in rupture of the graphite blocks.

If the rate of thermal or radiation-induced creep were sufficiently large, rupture of the blocks would be prevented or delayed. Thermal creep of graphite has been measured at temperatures above 2000 C, but little information is available for the temperature range 400 to 1000 C of current interest in gas-cooled reactors.

Calculations for conditions in the Experimental Gas-Cooled Reactor have indicated that a creep rate of 10^{-11} in/in/sec at a stress of 1000 psi would be necessary to prevent cracking of the graphite. A program to study thermal and radiation-induced creep is in progress at Hanford.

At present tensile creep is being studied on EGCR graphite over a load range of 800 to 1000 psi and at temperatures of 475 to 600 C. To date tests have shown an initial primary creep, which is small and occurs over a period of a few hours. No secondary tensile creep has been found. Tests to date have been run for 2000 hours. From the sensitivity of the equipment it is estimated that the secondary creep rate must be less than 10^{-12} in/in/hr.

An in-reactor test was recently installed in the F-6 position of the ETR. The capsule uses a bellows and draw bar assembly to produce the desired tensile load on a 4-inch specimen. The temperature is attained by gamma heating. In the first capsule of this type, GEH-13-9, the specimen is loaded to 800 psi in a helium atmosphere. The mean temperature is 480 C. An unloaded reference sample is being irradiated concurrently in the same capsule. The capsule design is such that a strain-rate device can be installed if the initial results of pre- and post-length measurements indicate that radiation-induced creep occurs.

Air Oxidation Studies

A series of experiments is in progress to determine the effect of variations in gamma intensity on oxidation of graphite by air. Previous data had indicated a decrease in activation energy caused by a gamma flux of 1×10^6 r/hr. However, no influence of radiation on air oxidation in the temperature range 500 to 700 C has been detected in recent experiments at 6×10^6 r/hr. Experiments are continuing at lower temperatures to determine the range where radiation will influence the reaction.

7. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

Ex-Reactor Corrosion Testing

Testing of A288 and X-8001 aluminum, A212 C/S, Zr-2, and 304 S/S continued at 300 C and a pH of 4.5 adjusted with chromic acid. The loop was down during the month for removal of a valve that developed a leak in the valve body. Examination of the 304 S/S valve revealed extensive carbide precipitation with accompanying intergranular corrosion and cracking. Adjacent weld areas on the 304 S/S piping were free from this attack. Another valve will be removed to determine if this condition is general in the loop or was particular to the one valve.

Nickel-plated 1100 alloy aluminum bars were exposed in TF-3 at 580 F and a pH of 10.0 using LiOH. A one-mil thick plate treated for 100 hours at 400 C and a two-mil thick plate treated for 100 hours at 400 C were evaluated. After approximately 100 hours of exposure, examination of the one-mil material revealed extensive intergranular corrosion attack through portions of the nickel-plate. The two-mil material is still under test.

The test to determine if 200 ppm each sodium dichromate and sodium benzoate would inhibit the corrosion of carbon steel and aluminum in neutral pH, 300 C water has been discontinued after a 500-hour exposure due to excessively high corrosion rates. The A212 carbon steel was corroding at a rate of 0.876 mil/year and the X-8003 aluminum at a rate of 175 mils/year.

8. USAEC-AECL COOPERATIVE PROGRAM

Boiling Burnout Studies with 19-Rod Bundles

The electrically heated model of a 0.050-inch spaced 19-rod bundle fuel element, which failed after 18 experimental boiling burnout points were obtained, was removed and repaired. Several of the ceramic beads used to simulate wire wraps were found crushed and the 0.050-inch spacing between rods was not necessarily maintained throughout the tests. In fact, the rod which failed as well as three others were found to have bowed. It is not known when during the course of the tests the bowing occurred. To circumvent this, the test section was repaired using 0.050-inch Inconel tubing filled with compacted MgO for wire wraps. The tubing will conduct electricity and will therefore generate some heat at a heat flux calculated to be about one-third that of the rod surfaces. The repaired test section was further modified to include a thermocouple to measure the water temperature of one of the sub-channels of the bundle just downstream of the heated section.

The repaired test section was installed and testing started. However, a leak developed on one of the rods during the second boiling burnout run, and this was sufficient to warrant removal of the test section for replacement of the leaking rod. When the test section was dismantled, three of the MgO filled lengths of tubing used to simulate wire wraps were found split for lengths of one to four or five inches. This is believed due to very small holes at the ends of the tubing which allowed water to penetrate into the MgO. When the test section was operated, the water was vaporized faster than the pinhole could relieve the steam formed and the resulting pressure split the tube.

The leaking tube and the split tubing were replaced and reassembly of the test section is in progress at the month's end.

9. REACTOR AND NUCLEAR SAFETY STUDIES

Advanced Reactor Concepts Studies

5 Mwe Spacecraft Reactor. Calculations of reactivity and fuel burnup for a plutonium-fueled fast reactor for spacecraft applications performed by Applied Physics Operation indicated that core life or control span are not significantly benefited by varying the isotopic composition of plutonium. Certain mixtures of Pu-239, Pu-240 and Pu-241 did not produce a "phoenix fuel" effect in the

fast spectrum considered here. Studies of the use of a burnable poison are now being performed to ascertain if performance can best be improved in this manner. Fertile material (U-238, Th-232) may be investigated later.

Burnup calculations indicate that a very large control span ($\sim$30% K) is required to compensate for burnup alone for a 20,000 core life unless a reduction can be made through burnable poisons or a more dilute fuel-fertile core. This span can be achieved with reflector control of a suitable type and several control schemes are being compared for feasibility and reliability. However, this great a span of control in the reflector will severely alter power distribution in the core during the core life and will require a two- or three-pass coolant system to compensate for this effect.

Fast Supercritical Pressure Power Reactor. A comparison of the Hanford Supercritical Pressure Power Reactor(1) with a steam-cooled fast breeder reactor conceived by United Nuclear Corporation(2) and a plutonium-fueled fast breeder reactor studied by Atomic Power Development Associates(3) indicated that it should be possible to design a fast breeder reactor basically similar in concept to the Hanford SPPR. The major changes in physical plant would be associated with accommodating a different critical mass of fuel, defining core and blanket fuel regions, eliminating the moderator and substituting an alternate method of cooling for the outer fuel cans and control assemblies, and establishing the control requirements. A conceptual design study of a fast supercritical pressure reactor was therefore initiated to enable evaluation of the economic potential of the reactor.

(1) HW-68420 Rev, "Economic Evaluation of a 300 Mwe Supercritical Pressure Power Reactor," H. Harty et al, June 1961.

(2) NDA-2148-5, "Conceptual Design and Economic Evaluation of a Steam-Cooled Fast Breeder Reactor," G. Sofer et al, November 1961.

(3) APDA-129, "A Plutonium-Fueled Fast Breeder Atomic Power Plant," April 1959.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

Polycrystalline molybdenum tensile specimens with 0.180-inch diameter gage lengths prepared from cold worked rods which were annealed at temperatures of 1050, 1300 and 1500 C have been loaded into capsules for irradiation to three exposure levels, 10^{18}, 10^{19} and 10^{20} nvt.

Thin foils of molybdenum for study by electron microscopy and x-ray diffraction techniques have been and are being irradiated between aluminum spacers in small aluminum capsules. A recent test in such a capsule equipped with a thermocoupled specimen has indicated that the maximum temperature of foil specimens does not exceed 39 C during irradiation in evacuated capsules. If helium gas is introduced in the capsule, the specimen temperature falls to 30 C under normal reactor operation conditions (normal power and flow of water coolant over the capsule).

Capsules containing foil specimens (0.003-inch thick) of Johnson-Matthey high purity stock, and of stock with nominal carbon contents of 10-20, 100-200, and 400-500 ppm, have received a nominal exposure of 1×10^{18} nvt (fast). These capsules contain specimens which had been annealed at 1850 C for one-half hour. Specimens will be processed in the near future.

Foil specimens of Johnson-Matthey molybdenum irradiated in the as-rolled state and in the stress-relieved state have been annealed at 500 C and 600 C, and are currently being examined by transmission microscopy. Some evidence of clustering of defects similar to the "Black Death" reported by British investigators studying radiation-induced damage in f.c.c. metals has been found in the foils annealed at 500 C. Since the density of these defects is low and their size is small ($\sim$25 A diameter), additional results are needed. It is hoped that the foils annealed above 500 C will supply confirmation of this clustering phenomenon.

Molybdenum rod from nominal 0.5-inch diameter recrystallized rod has been subjected to swaging treatments in attempts at introducing various and appreciable levels of "cold-work". Several reductions of cross-sectional area were obtained at temperatures of 200, 300, 400 and 600 C. The "cold-worked" material will be studied in the differential calorimeter to ascertain release of stored energy due to the work introduced in the specimens by mechanical means. This information will be utilized in the interpretation of the release of energy stored in specimens by neutron irradiation only.

An evacuable platinum wound furnace tube capable of operation at temperatures as high as 1600 C has been fabricated, tested, and put into use. Modifcations to the shapes of foil tensile specimens for testing in the electron microscope have been made. Tests currently in progress will establish an optimum shape for the die and punch required for preparing specimens.

A total of five tensile tests, interrupted for x-ray diffraction photographs, have been made on the unirradiated molybdenum single crystal specimens. The specially designed fixture has proven quite satisfactory. Detailed analysis of the deformation modes, as revealed by the x-ray patterns, is being carried out.

Low-carbon single crystals are quite ductile, failing only after being drawn out to a chisel edge. Medium- and high-carbon crystals fail earlier, a flat fracture surface parallel to (100) being formed on the two specimens tested to date. Deformation modes vary with the initial orientation. The first crystal examined, with rod axis initially near $[0\bar{1}1]$, deformed by a simple (211) - $[\bar{1}11]$ mechanism. Crystals of other orientations deform in a more complex manner.

E. CUSTOMER WORK

1. RADIOMETALLURGY EXAMINATIONS

The cause of the failure of the third low exposure element from D-Reactor was determined to be a defective spire that was damaged during fabrication of the fuel element. A low exposure side failure from F-Reactor has been examined. The failure resulted from mechanical damage incurred during the charging operation. A heavy scrape mark adjacent to the rupture area was observed on the male end of the failure from 1372 H. It is the probable cause of the rupture. The male end cap will be examined to determine whether there was a defective weld.

A 3-1/2-inch core sample of sludge from a Redox waste storage tank was removed from a rotary core drill assembly. Photographs of the sample were obtained and samples sent to Redox Technology for analysis.

2. EQUIPMENT DEVELOPMENT

Recent experiments in color photography have resulted in procedures for the use of color film in the conventional and remote metallographs as well as the borescope. The color obtained with this procedure approximates the observed color very closely.

A new vacuum annealing furnace and the remote Rockwell hardness tester have been set up in "E" cell. Both pieces of equipment are in operation on hot samples.

The High Level Utility Machining Cell is essentially complete except for modification of the lathe. Procurement problems continue to plague progress on the Physical and Mechanical Properties Testing Cell with delivery of cell castings delayed until late September 1962. The Instron Tensile Testing unit has been set up in I-cell in preparation for accelerated testing of irradiated structural materials.

3. METALLOGRAPHY LABORATORY

A forged steel connector for NPR process tubing was examined to determine the suitability of a weld which was laid down to increase certain dimensions of the connector. The welding rod contained approximately 0.5 percent Ni with no Cr, while the base metal contained approximately 0.5 percent Cr and no Ni. Otherwise, the compositions were about the same. A diamond pyramid microhardness traverse showed a rise from 250 to 310 DPH (equivalent to a rise of 9 points on the Rockwell C scale) within a distance of about 250 microns in the heat affected area of the base metal. A subsequent stress-relief anneal reduced the hardness gradient sufficiently to expect that all other connectors treated in this manner would be suitable for service in the NPR.

4. N-REACTOR CHARGING MACHINE

Modifications

Fabrication of the new limit switch assemblies which control the vertical movement of the machine while loading or unloading magazines was completed. One assembly was installed on the machine and evaluated. Mechanically, the assembly was inadequate and the assembly was removed. All of the assemblies will be modified.

Fabrication of the modified transfer arm pressure indicators was started.

The new process tube nozzle which was damaged during installation was removed and reconditioned. The process tube which was severely scratched during charging tests will not be used again. A six and one-half foot section of this tube was removed for additional testing and evaluation.

About 80 hours of electrical craft time were expended in additions and modifications to the control circuitry. Installation of the remote panel and connecting wiring was completed and tested. Rewiring of the counters in the Selsyn assembly was completed.

Testing

A working committee has been formed with members from FPD, IPD and HLO to resolve the fuel element shoe to process tube compatibility problem. Present work is directed toward obtaining correlation of the results obtained in 306 Building on short sections of process tube and in 314 Building on a full length process tube. Future work involves the investigation of various parameters such as tube and fuel cleanliness, tube length, fuel column length, and damage mechanism.

The test of autoclaved fuel feet against autoclaved nozzle was completed with no evidence of damage observed.

All flow control, pressure reducing, and pressure relief valves were removed from the charge machine and set or calibrated as required. The valves were re-installed and tested. Set points for all valves were tabulated. This table will be included in the report on Design Test 13.

5. FISSION PRODUCT TRANSIENT SAMPLES

Fabrication of 72 fission product transient samples for Phillips Petroleum Company is continuing. Autoclaving and final assembly of 16 of the coextruded Al - 2 w/o Si clad tubular elements have been completed. Ten, for irradiation, contain 0.2 gm Pu-239 each. The others are calibration standards. Two contain 0.4 gm Pu-239 each, and four have aluminum cores. Preparations for shipping are in progress. Cores for 36 additional elements containing Pu-Li-Al and U-235-Al have been cast and machined. Difficulties in obtaining analytical results with the required precision of one percent have delayed further work with these alloys. Welding of the end fixture on these elements is being done in the glass lathe. A continuous burn down fusion weld is being made without additional filler metal.

J. J. Cadwell

For Manager, Reactor and Fuels Research
and Development

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PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATION

MONTHLY REPORT

JULY 1962

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

Exponential Experiments for NPR

The final analysis of the exponential experiments in the NPR condensed lattice is complete. The method of analysis used was to first determine a suitable extrapolation distance for the front-to-rear, side-to-side and vertical directions by averaging many experiments; then to determine the material buckling from a fit of vertical traverse data.

The values of the extrapolation distances in the NPR condensed lattice were found to be 0.93, 1.03, and -0.47 inches in the side-to-side, front-to-rear, and vertical directions respectively. The negative value for vertical extrapolation is the average of a large number of experiments. One possible reason for a negative value is that the pile is 1.47 inches higher than ten lattice units; thus if the pile were the correct ten units high, the extrapolation length would be 1.0 inches.

For the diffusion length experiment horizontal traverses were not measured and the extrapolation distances were assumed to be one inch. The vertical extrapolation was measured to be 0.67 inches.

The final bucklings are listed in the following table.

Fuel Type	Fuel Wet or Dry?	$B^2(10^{-6}cm^{-2})$	Relaxation Length (cm)	Remarks
None	---	-293 ± 5	36.19	L = 58.46 ± .49 cm.
Nat. U.	Dry	-160 ± 6	39.76	
Nat. U.	Wet	-155 ± 7	39.94	Average of two
.947 U^{235}	Dry	+ 56 ± 6	49.02	
.947 U^{235}	Wet	+250 ± 6	67.06	Average of three

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The errors quoted are due to errors in fitting the exponential plus errors caused by uncertainties of one quarter inch in the horizontal extrapolation lengths and an uncertainty of one-half inch in the vertical extrapolation length.

There were no experiments performed with the condensed lattice exponential in a flooded condition or with control rods inserted.

The ratio of graphite atoms in the NPR mockup to graphite atoms in the condensed lattice is 1.016 using solid graphite densities of 1.695 for the mockup and 1.636 for the condensed lattice.

The streaming factor for the NPR lattice can be determined from the measurements. This ratio is 2.1 for enriched fuel and 1.6 for natural fuel from the ratios of condensed lattice buckling to mockup buckling (reported last month) for each fuel. The wet and dry streaming factors were averaged. The reason for the difference in the buckling ratios for the two fuel enrichments is not understood. A correction for the 1.6% difference in graphite content is needed, but this correction will make the difference still larger.

Optimization of Retubed Lattices

Two vertical and two horizontal flux traverses were measured in the C-pile mockup with C II N fuel. One vertical traverse was performed with one control rod in the center of the pile while the remaining traverses were performed without control rods.

K-Reactor Program

Measurements of the reactivity change with irradiation time for control splines are planned for the PCTR with the K-lattice core. The splines with the lowest irradiation have reached their goal exposure and have been removed from the reactor. The irradiation of the full set will not be complete for several months.

Measurement of the Angular Distribution of Thermal Neutrons at the Surface of Cadmium Rods

The study of neutron transport involves the distribution in position, energy, and direction of the neutrons. Measurements of the space and energy distributions of neutrons have been made for many years. However, the angular distribution has not been studied experimentally in any detail. The present experiment was designed to measure the angular distribution of neutrons at the surface of a strong absorber for comparison with the results of the S_n method for solving the transport equation(1). This is the first experiment ever designed to measure angular fluxes inside of a reactor.

The angular distribution of thermal neutrons was measured at the surfaces of 37-inch long cadmium rods of two diameters, 0.9 inches and 5.3 inches. Each rod was placed in the center of the graphite core of the Physical Constants Test Reactor (PCTR) in a separate experiment. Small holes (63 mils diameter, 140 mils deep) were drilled in the surfaces of the rods at several angles. Small foils (55 mils diameter, 11 mils thick) of a Dy_2O_3-plastic mixture were placed on the ends of aluminum or cadmium pins for insertion into the holes. The activities with the cadmium pins were used to subtract residual activation by epicadmium neutrons. The dimensions of the holes allowed the foils to see about 2% of the 4π solid angle. With this collimation the signal to background ratio (collimated thermal activation to uncollimated epithermal activation) was about 10 for the 0.9-inch rod and 2.5 for the 5.3-inch rod.

The results showed the angular distribution anisotropy to be larger for the larger bar. The experimental data indicated a slightly larger anisotropy than did the calculations with HAPO-Program S-X, although the agreement with theory is remarkably good for such a severe test of the computational model.

The results are useful to indicate the precision of transport codes which are then used for analyzing other systems, as well as to show whether simpler codes can reasonably be used at a reduced cost in manpower and computer operation. Verification of theoretical methods will be useful in reducing the safety factors used in design and hazards analysis, especially in situations which are not directly accessible to experiments, so that decisions must be based upon calculations.

Stability Difficulties with S_n Transport Analysis

Preliminary comparison of S_n transport theory predictions with recent Hanford measurements of neutron angular distribution has provided a baffling contradictory evaluation of the level of reliability of the S_n analysis method.

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(1) B. H. Duane, Neutron and Photon Transport, Plane-Cylinder-Sphere, GE-ANPD Program S, Variational Optimum Formulation, XDD-59-9-118 (1959). See also Albedo Logic for Double-S_n Transport Analysis, pp 4-10 of HW-71747 (1961).

A 52-angle transport calculation of the flux entering the surface of a cadmium cylinder centered in graphite moderator within a fueled annulus shows good quantitative agreement with angular distribution measurements made by placing tiny sensor foils within angular-collimation holes drilled into the cadmium at various angles to its surface. But, in sharp contract with the satisfactory prediction of inward angular distribution, the S_n transport representation of the outward flux shows persistent numerical-instability oscillation in the graphite within a mean free path of the cadmium surface.

This outward-flux oscillation appears to be both spatially damped and tightly constrained in angular-harmonic content, with no discernible oscillation in angularly-integrated flux, angularly-integrated current, or inward angular distribution, despite the fact that the outward angular distribution shows distortion beyond physical interpretation. Consequently, it is felt that this instability may have only localized second-order effects upon the calculation of criticality, power distribution, isotropic sensor activation, neutron moderation heating, photon energy deposition, or other such angularly-integrated information. Nevertheless, until this difficulty is cleared up, all S_n analysis work in curvilinear geometry should be scrutinized critically for evidence of outward-flux oscillation near important material interfaces.

Spatial Resonance Self-Shielding

Work on the program to compute group self-shielding factors, program GROUSS, progressed to the point where an ordered energy lattice combining an equal lethargy mesh and an energy mesh centered on each resonance is properly computed. The treatment of Doppler broadening will be improved by computing the ψ function numerically from the differential form[2] using a central difference approximation.

In order to provide Programming Operation with assistance in fuel element calculations which depend on spatial self-shielding, on a time schedule ahead of program GROUSS, the ANP C-Fine tape generating program, FINDT[3], was successfully used to make a test case tape.

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(2) Amster, Harvey J., "A Differential Equation for Calculating Doppler Broadened Resonances," NSE 11, 343 (1961).

(3) Cooper, J. R. and W. B. Henderson, "Nuclear Data Tape Program with Fine Energy Detail," XDC-60-8-69, August, 1960.

The ANP Doppler broadening code, which computes D-B cross sections from the C-Fine tape by straightforward numerical integration of the product of the cross sections and the relative velocity probabilities, has been modified to punch cards in the format required by FINDT. Checkout is in progress. The FINDT and D-B codes can then be used to generate tapes containing D-B cross sections of one or more isotopes over the desired energy range for self-shielding calculations as described in the monthly report for June, 1962.

Effective Void Diffusion Coefficient

Present multi-group diffusion codes such as HFN incorrectly allow no flux change in curvilinear void regions. Garelis[(4)] in an effort to correct this deficiency has derived an expression for a fictitious "void diffusion coefficient," which preserves P-1 boundary conditions when used in ordinary diffusion theory. Because of memory limitations, the exact expressions cannot be coded into HFN. Adequate, but more compact expressions are now being sought.

HFN

Revised pages for the HFN document, HW-71545, have been issued as informal document HW-71545 Rev. These correct several typographical errors, and describe an additional boundary condition option.

APDAC

Half-life data for dysprosium foils has been added to the built-in APDAC library.

The existence of two bugs in APDAC has been established. When using certain optional features of the code to analyze dysprosium foil activity data, one of the bugs causes an incorrect error analysis. The other bug causes APDAC to "swallow" foil data under some conditions. The cases which originally experienced these troubles were successfully processed by changing the input slightly. The bugs have not been located.

Computational Programming Services

ICEDT, the complete exponential data reduction code, has been written and is now being debugged. Preparation, operation, and revision of various

(4) Garelis, E., Nuc. Sci. and Eng. 12, 547-548 (April, 1962).

versions of the kinetics code, TRIP, have continued. Attempts to enable TRIP004 to satisfy stricter requirements for parameter convergence are still unsuccessful.

Input data preparation for further calculations on three of the four reactor systems to be treated by the S-X code has been turned over to the originators of the problems. Only the problem of the critical mass of dissolving plutonium is still being processed. In addition, a problem involving photon heating in a homogeneous Cs-137 cylinder was begun and returned to the originators.

Instrumentation

The prototype gamma spectrometer was again returned to GE-APED for further modifications to attempt to meet the specifications. The prototype differential alarm module was also returned for changes following extensive testing. The prototype units, when satisfactory, are to be used as models for 24 instruments for use in the slow-scan portion of the NPR Fuel Failure Monitoring System.

Computations were completed and a report issued regarding necessary shielding and absorbers to be used with both the top and bottom shield neutron sensitive ionization chambers for use at NPR. The work was done at the request of Electrical and Instrumentation Design, CE&UO. In addition, the general program of nuclear instrumentation support for NPR continued with review of prints and specifications for the source range monitor, the intermediate range monitor, and associated ion chamber assemblies in the reactor shield. Assistance was rendered in the negotiations with GE-APED regarding fabrication methods for the chambers.

An outline of a proposed set of rupture activity monitors and a rough draft specification for the gross-gamma monitor channel were submitted to Irradiation Testing, IPD, following their request for assistance. The work was for the HAPO-264 project for a fuels development effluent activity monitor for both Irradiation Testing, IPD, and Ceramic Fuels Development, HLO.

Suggestions and advice were rendered to Irradiation Testing and Reactor Physics of IPD regarding the application of fission counters to their poison spline monitoring problem, and in addition, comments were made to Applied Reactor Engineering personnel regarding methods to simplify and improve the fission counters of the subcritical monitors of the existing reactors.

At the request of Design Analysis, IPD, a critical review was made, and comments rendered, of a rough draft report which summarizes a calculational study of anticipated temperature profiles, resulting from abnormal conditions, in the top shield complex of existing reactors. The report concerned shield temperature control by appropriate fringe process tube loading.

An analysis was made and a report was issued regarding the decay and attenuation properties of the radiation emitted by an irradiated sample of titanium as proposed for use as the wire in the NPR Traveling Wire Flux Monitor. Because of the high energy photons from the titanium and from a small manganese impurity in the wire, it was estimated that at least 5.5 inches of lead shielding would be required. If no manganese were present, the lead could be reduced to about 3.5 inches. Since less than two inches of lead can be incorporated in the present design, major design modifications will be required. The analysis was made at the request of Reactor Physics, IPD.

Tests intended to yield information on the flux distribution effects of reactor control rods under equilibrium operating conditions were continued during the month. Simultaneous recordings of neutron flux changes at three locations in the 105-KW reactor, due to four rod position changes made one at a time, were obtained. Measurement methods used in the last test series improved the precision of the measurements. The test method yields useful data with very small rod movements and thus can be carried out during normal operations with negligible effect on the reactor flux distribution. Additional tests are planned for the week ending August 5.

Work was resumed on a reactor control and transfer function study using the EASE and GEDA analog computers. The analog model is the same as the one used for the reactor instrumentation study, except that the scram and vertical safety rod circuits are not needed. The simulation includes a controller for each of the 11 regions in which power and temperature are computed. The controllers hold the regional power levels at a specified value while the entire reactor comes to equilibrium. Initial runs were made with the controllers in and operating; a .1 megawatt step change in power was introduced into each region in turn, and the results in all nodes were observed by recording regional power on an 8-channel strip chart recorder. Type zero and type one control were both used, each with gain constants of both one and ten on the controllers. Additional analog runs are planned to verify our present concept of an equivalent reactor structure.

The 11-node reactor instrumentation analog simulation study has been completed for the present and the results given to IPD. The purpose of the study was to determine, for various instrument trip point settings, how fast the control rods may be withdrawn without causing coolant boiling at the reactor outlet; the problem was studied at several power levels. The instruments providing reactor scram signals were those which monitored neutron flux, reactor outlet temperature, and linear rate of rise of outlet temperature. It is probable that further instrumentation study will be undertaken soon.

A purchase specification for one complete system of Bailey (NPR) control equipment was prepared for NPR Project Operation. A purchase order for the equipment has been placed.

The theory of Dynamic Optimization has been extended to cases of more than one variable, such as large reactors. The theory has been verified, but the applications to practical problems will require the use of a computer, as the mathematics become cumbersome to handle analytically.

SEPARATIONS

Experiments with Plutonium Solutions

Further criticality experiments were conducted with plutonium-nitrate solutions in a 14-inch diameter stainless steel sphere, fully reflected with water.

Criticality was studied as a function of the plutonium concentration and acid molarity of the solutions. The plutonium concentrations were in the range of 33 to 44.7 g Pu/ℓ; the nitric acid molarities of the solutions were approximately two and four. The critical concentration of the full sphere (23.22 liters) is determined from extrapolation of the data (critical volume vs. concentration) as the Pu concentration is adjusted and criticality in the full sphere is approached. Experiments are currently being conducted with Pu solutions at an acid molarity of ~ 6.0. When complete, these data will permit an evaluation of the effect of nitrate on the criticality of the Pu solutions.

Data were also obtained for estimating the correction to the measured critical volumes for the effect of the 0.044-inch stainless steel vessel wall. In one experiment a mockup of the sphere neck support was appended to the vessel to determine its effect on criticality.

The data from the experiments completed during the month are summarized in the following table.

The Pu concentrations which result in criticality with the sphere full

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

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

Exp[redacted]ment [redacted]er	Date	Vessel	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)
114[redacted]6	7-5-62	14" Sphere	Water + .036 SS	39.9	4.12	1.199	857	297	594.9	23.1 +.04 -.05	0.92
1142097	7-9-62	14" Sphere	Water + .072 SS	40.3	4.13	1.199	857	298	589.0	23.3 +.07 -.09	0.94
1142098	7-10-62	14" Sphere	Water	40.4	4.07	1.199	856	295	586.5	22.6 +.04 -.05	0.91
1142099	7-11-62	14" Sphere	Water + Neck Mockup	40.4	4.07	1.199	856	295	586.5	22.6 +.03 -.03	0.91
1142100	7-12-62	14" Sphere	Water	38.4	4.06	1.196	854	292	615.6	23.3 +.04 -.05	0.89
1142101	7-17-62	14" Sphere	Water	44.7	2.14	1.146	923	179	559.6	19.6 +.09 -.15	0.88
1142102	7-18-62	14" Sphere	Water	36.6	2.01	1.126	924	136	683.3	21.4 +.07 -.11	0.78
1142103	7-20-62	14" Sphere	Water	34.0	2.06	1.124	930	163	740.6	22.5 +.03 -.08	0.77
1142104	7-23-62	14" Sphere	Water	34.4	2.08	1.121	926	165	729.1	22.6 +.06 -.07	0.78
1142105	7-24-62	14" Sphere	Water + .036" SS	34.4	2.09	1.121	918	165	722.9	23.0 +.03 -.04	0.79
1142106	7-25-62	14" Sphere	Water + .072" SS	34.4	2.09	1.121	916	165	721.4	23.5 +.05 -.07	0.81
1142107	7-26-62	14" Sphere	Water	33.5	2.07	1.123	920	163	743.8	23.0 +.02 -.02	0.77

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for acid molarities of 2.08 and 4.07 are given below:

Pu Concentration for Criticality in Full Sphere (23.22 liters)	H/Pu Atomic Ratio	Acid Molarity	Total Nitrate	Critical Mass
33.2 g/ℓ	750	2.08	164 g/ℓ	771 g Pu
38.6 g/ℓ	613	4.07	292 g/ℓ	896 g Pu

Of special importance for the execution of the program of criticality studies in the Laboratory is the 234-5 Development Processing Facility for Critical Mass Fuel Preparation. The equipment for concentrating plutonium nitrate solutions is now available. Routine load out of plutonium solutions from the mixing room of the Critical Mass Laboratory was begun during the month; seven PR cans containing a total of 56 liters of dilute Pu solution were sent to the 234-5 Processing Facility for re-concentration. These transfers were accomplished without spread of contamination; the load out procedure has worked very well.

A new spherical critical assembly vessel of 15.3-inch diameter was received during the month. This vessel contains provisions for a re-entrant tube to facilitate flux measurements along the axial diameter of the unit. This is the second sphere of this design to be received. This unit will permit criticality and buckling measurements at high Pu concentrations and acid molarities in the unreflected state, whereas the diameter (11.6 inches) of the previous unit received will permit similar studies with high Pu concentrations in a fully reflected assembly.

Status of Experiments with Plutonium Oxide Plastic Mixtures

Construction is proceeding on the critical assembly (split-half) machine for use in the criticality measurements with the PuO_2-plastic mixtures. Final testing, and the installation of this device into the second hood of the critical assembly room is expected to be completed in December of this year. The electrical wiring of the movable table has now been completed and tested. The operation of one rod drive has also been tested and the construction of the other three rod drives has been started by Technical Shops for delivery in August.

The new neutron source drive for use with the split-half machine has been promised by the vendor about October 1. The plutonium-beryllium neutron source has been received.

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The fabrication of the PuO_2-plastic mixture into small two-inch cubes by the Plutonium Metallurgy Operation for the initial series of experiments is about 75% completed. Since the K-9 incident, no further preparation of the fuels has been made. The problem of coating the individual cubes for contamination control has not been fully resolved.

A supplement is currently being prepared to the Hazards Summary Report for the Laboratory to cover the operation of the split-half machine and the experiments with solid plutonium bearing systems. This report is expected to be ready for submission to the AEC by October 1.

In-Plant Neutron Multiplication Test for Nuclear Safety

On Saturday, July 14, an in-plant neutron multiplication measurement was made for a large array of partially reflected shell castings in hood HC-22SR of the 234-5 Building. The array was reflected on three sides with ten inches of water contained in fixed steel tanks.

Increased storage in hoods HC-22SR and HC-45SR were previously permitted upon the completion of In-Plant Neutron Multiplication Test No. 3.[5,6] The increased storage resulted in increased neutron exposure to personnel; "water walls" were then erected on three sides of the hoods to reduce the neutron exposure. The multiplication measurements were made to examine the effect of these "water walls" on the criticality of the array.

This was a cooperative effort between personnel of Critical Mass Physics, HLO, and CPD. The instrumentation for the experiments was supplied by Critical Mass Physics - together with technical direction during the measurements. The three neutron instruments behaved perfectly during the course of the measurements. Although a total of 592 Kg of Pu were loaded into the hood at 33 positions in three layers, the array was far subcritical; these results show that it will not be necessary to reduce the number of units permitted in the array because of the effect of the "water walls".

This is probably due to the fact that the castings used in these experiments had a smaller stacking density than in previous experiments, i.e., at each position in the storage array, the k_{eff} of the individual stack, or its fraction of critical, was smaller.

(5) HW-71208, In-Plant Neutron Multiplication Test No. 3, R. E. Isaacson and R. L. Stevenson, 10-2-61.
(6) HW-71345, Critical Mass Estimates for Arrays of Castings, R. E. Isaacson and R. L. Stevenson, 10-6-61.

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Measurement of k_∞ in the PCTR for Dilute Solutions

An experiment to study the effect of stainless steel (as used in containment vessels for solutions) on the measured values of k_∞ in the PCTR was completed during the month. The experiment was conducted with dilute uranyl fluoride (UO_2F_2) solution with uranium concentrations in the range 12-20 g U/ℓ; the uranium was highly enriched (93.15% U^{235}).

Data from the experiment is now being analyzed and the results will be used to evaluate a previous experiment on the limiting critical concentration of plutonium in water, which also used stainless steel core tanks.

During the course of the experiments, a somewhat unexpected phenomenon was observed. The reactivity with the core tank filled with He was found to be less than with the core tank filled with air; the results showed helium to act as a poison to the system relative to that of air. A reactivity measurement was then made with the core tank under vacuum. The observed differences in reactivity were dependent on the thicknesses of stainless steel in the core tanks, but in terms of increasing reactivity the order was, He, air, vacuum.

The results show that He cannot be arbitrarily used as a substitution for a vacuum (standard) wherein the PCTR method the k_∞ in the core tank is by definition unity. In previous experiments He has been used to simulate a void ($k_\infty = 1$).

Consulting Services on Nuclear Safety - Criticality Hazards

Nuclear Safety in HLO

The following nuclear safety specifications were approved and issued:

- G-1 (Revised) General Rules for the Storage, Processing, and Transporting of Enriched UO_2 (Ceramic Fuels Development Operation)
- J-1 Rules for Plutonium Handling in Dry Glove Boxes (Plutonium Metallurgy Operation)
- J-2 Special Rules for Handling Plutonium Metal in Glove Box 47B (Casting)
- N-1 General Rules for the Storage, Processing, and Transporting of Plutonium and Plutonium Oxide (Analytical Laboratories)

[redacted]

[redacted]

Nuclear Safety - Training and Education

A course in nuclear safety for the benefit of CPD engineers is being given at the Critical Mass Laboratory. The course consists of a series of ten lectures together with the showing of several films on the subject of criticality and nuclear safety. The lectures are being given by Messrs. Brown and Clayton. The first series of lectures was begun on July 23; the meetings are from 10:30 AM to 12:00 noon on Mondays and Thursdays in the lunch room of the Critical Mass Laboratory. Twenty-three persons from CPD are participating in the first session. It is expected that four sessions running consecutively will be required to accommodate all those interested in participating, and that about 100 supervisors from CPD will ultimately complete the course.

Mass Spectrometry

The heavy-element mass spectrometer for this program has been out of operation this month in order to investigate the possible reasons for the poor resolution of the spectrometer. Measurements of the alignment of the analyzer revealed that its axis is out of alignment by 0.28° from the median plane of the magnet gap and 0.28° from the plane perpendicular to both the median plane and the plane of the magnet pole boundary.

An examination of the inside of the analyzer tube revealed the presence of magnetic filings bridging from one side of the tube to the other in the magnet gap.

It has not been determined as yet that the misalignment and/or magnetic filings could account for the poor resolution. However, a part of the deflection in the direction of the magnetic field which was previously required to bring the beam into the center of the analyzer exit slit could be accounted for.

The linearity and accuracy of the vibrating reed electrometer and recorder system of the mass spectrometer was also calibrated during the month.

Interaction by Matrix Methods

A straightforward extension of the subcritical interaction problem to matrix form using the work of Dowson[7] has been completed. A simple

(7) Dowson, D. C., "Criticality of Interacting Arrays of Fissile Material," AHSB-S-R-28, 1961.

assumption led to a method for predicting k_{eff} of an array of interacting units. Comparison with the available data on such systems (3, 4, 5, cylinders in line; 3 cylinders in a triangle; 7 cylinders in hexagonal array) showed the predicted k for measured critical systems to be in fair agreement. The errors run 1-3% except for very close spacings of the 7 cylinder system, in which case it may be as much as 7%.

A possible method has been found to take account of an external reflector; however, no data is yet available to check out this method.

Buckling of Partially Filled Spheres

A theoretical study to compute (or approximate) the buckling of partially filled, reflected spheres has begun. Knowledge of this buckling will provide much additional information from the experimental data on plutonium solutions.

The first step in the solution of the problem has been taken. The probability function for neutrons leaving the reflector and returning to the fuel solution before re-striking the reflector (assuming constant isotropic flux was integrated analytically for the case of a half-full sphere (hemisphere)). The probability was found to be exactly one-half. Taking account of multiple reflections gives a probability $1/(2-R)$ (R = reflection coefficient) that a neutron leaving the reflector will ever return to the fuel. If the number of neutrons returning to the fuel may be assumed to have a direct relationship to the reflector savings, then the end points of a buckling vs. height curve are known and assumptions can be made about the curve between these points. Insufficient data are available to check the calculations made thus far. An attempt is under way to analytically integrate the probability function for arbitrary height of solution.

Instrumentation and Systems Studies

A radioactive waste disposal heat transfer study was conducted to determine the maximum sphere of influence of the heat produced by a buried cylinder of radioactive waste. Since the radioactivity decays at a known rate, the temperature in the cylinder reaches a maximum at a time determined by the initial heat generation rate and the thermal characteristics of the soil. The magnitude of the maximum temperature and the time of its occurrence were to be determined. Heat flow is described by the diffusion equation, a partial differential equation. To solve this equation on the analog computer, finite difference techniques must be used. Since the equation is second order, two boundary conditions are necessary. However, only one boundary condition was known. This made it necessary to use a large cylindrical radius so that the sphere of influence could be considered as infinite. This restriction made it impractical to design a field with the mesh points spaced closely enough to give the needed results without exceeding the capacity of the computer. Therefore, this study has been postponed until better techniques are developed or more computing equipment is made available.

A study to simulate the Batch Pot Calcination process for treating radioactive waste was started. The purpose of the study is to determine the effect of the variables on the temperature profile in the pot. The problem is highly nonlinear, using 26 diode function generators, 30 multipliers, 35 comparator relays, and some special circuits. A large amount of this equipment has to be designed and fabricated since it is not available on our present computers.

At the Critical Mass Laboratory, some trouble was experienced with the log-N period meter during high level operation because of violent signal fluctuations. The fluctuations can be smoothed by increasing the filter time constant but this increases the time required to actuate the trip circuit. A better solution is to increase the detector efficiency, thus improving the counting statistics and preserving the short trip time. A program to improve detector efficiency was started.

The control rod drive mechanism for use at the Critical Mass Laboratory was operated and tested extensively during the month. Measurements showed a maximum withdrawal speed of 8.25 inches/minute, a positioning error of 1.7% maximum, a repositioning accuracy of ± 0.015 inches, and a controlled fall time of 0.6 seconds which is 2.5 times longer than free-fall. The mechanism was then adjusted to recycle (raise and fall) for six hours. Following this, the gear system was examined and excessive gear backlash was noted. New oversize keys are being installed to alleviate the problem.

NEUTRON CROSS SECTION PROGRAM

Quasi-Elastic Scattering of Neutrons from Water

The analysis of new data on repeat measurements of the quasi-elastic component of the scattering of neutrons from water has been completed. The new measurements which were made with a significantly thinner water sample agreed with the previous measurements within the statistical accuracy of the data. This result agrees with a simplified calculation of multiple scattering for this case which predicts a difference in shape of the angular distribution of intensity of only about two percent.

Further analyses were performed on the results of the quasi-elastic scattering of neutrons from water. These analyses show that the variation which was observed for the width of the quasi-elastic scattering component with scattering angle for neutron energies of 0.1 ev and 0.15 ev is consistent with a proposed model of the diffusion of water. The parameters which describe diffusion that are obtained from the fit of the model to the observed scattering data, however, are not entirely consistent with the expectation values. The results of the scattering which were obtained for 0.25 ev energy neutrons are, however, apparently inconsistent with this model. The reason for this discrepancy has not been determined. However, the data obtained for 0.25 ev energy neutrons are of significantly poorer quality.

Inelastic Scattering of Neutrons from Water

Measurements were completed of the amount of higher-order Bragg diffracted neutrons in the incident neutron beam of the triple-axis spectrometer using a Be(0002) diffraction. Measurements were made for the same neutron energies of 0.15, 0.2, 0.3, and 0.4 ev and in the same spectrometer configuration that was used for the measurements of the inelastic scattering of neutrons from room temperature water. The fraction of higher-order neutrons was found to be the same at all of the measured energies and equal to 0.02 ± 0.002.

In order to obtain direct information on the effect of multiple scattering of neutrons in the scattering sample, measurements were made of the angular distribution of scattered intensity of neutrons of 0.15 ev initial energy and 0.075 ev final energy for water samples of two significantly different thicknesses. These data have not yet been analyzed.

The data which have been obtained on the inelastic scattering of neutrons from room temperature water have now been reduced to an absolute differential cross-section scale and have also been prepared in the Egelstaff Scattering Law presentation. These data are now complete except for corrections for resolution of the measuring spectrometer and multiple-scattering effects in the water sample. These data have been compared with the results of similar measurements obtained from the Chalk River-Harwell program. In the region of overlap, which is for β values ($\Delta E/kT$) of 2 to 6, the results are in good agreement. The Hanford results are of significantly better statistical accuracy and extend to β values of 10.

The Scattering Law data have also been used to derive the generalized frequency distribution function $p(\beta)$ for water. The derived function agrees with the analysis of the Chalk River-Harwell results in the region of overlap. The shape of $p(\beta)$ in the region $6 \leq \beta \leq 10$, which was hitherto unexplored, reveals a strong contribution from the vibrational excitation of water at about 0.2 ev. The derived $p(\beta)$ does not, however, satisfy the

Placzek normalization condition. This inconsistency may be due in part to resolution effects. It may be of no sizeable consequence in the application of the results to thermal-reactor spectra calculations.

The results of the inelastic-scattering measurements are being prepared as a manuscript for presentation at the IAEA Conference at Chalk River.

Rotating-Crystal Spectrometer

Design work on a rotating-crystal time-of-flight spectrometer for inelastic neutron scattering measurements is continuing. Studies are in progress of possible neutron detectors and shielding problems.

Fast Neutron Cross Sections

The computer program for the reduction of fast-neutron total cross-section data, BIGNED, has been essentially completed. Total cross-section data which were obtained last December were re-evaluated using the computer program. The variations which had been observed in the previous hand calculations which were due to improper alignment of time zero appear to have been substantially eliminated in the computer program. The total cross-section data obtained last month are now being processed by the computer program. Additional sub-programs are being prepared to adapt BIGNED to handle different experimental methods of determination of zero time and channel width.

Instrumentation

Work has been completed on the 1024-channel time-of-flight analyzer for use with the neutron spectrometers. The analyzer is presently undergoing long-term test in the lab. It is expected that the analyzer will be put into use within the next few weeks.

Studies of pulse shape in organic scintillators were continued. Data were collected for three-dimensional families of curves of anthracene and Ne 213 liquid. The Ne 213 is used in a 2" cell and a 5" cell for the Fast Time-of-Flight Program. Both cells were measured, and the 2" cell was remeasured following deoxygenation of the liquid. Plots were made of the separation, or dispersion between n and γ peaks as a function of sampling time from the start of the pulse. The electron energy was held constant. These plots can be used to estimate the required integrating time constant to be used in pulse separating circuits. It was found that stilbene and anthracene have very long pulse tails, with photons still coming from the crystal after 10 µsec for stilbene and at least 30 µsec for anthracene. On the other hand, the liquid Ne 213 is an order of magnitude faster than stilbene. This suggests that the gamma suppression circuit used in the Fast Time-of-Flight Program, which was designed for stilbene, can be redesigned with much needed improvements in dead-time.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE

Graphite Lattice Parameters for Low Exposure Pu-Al Fuel

Thirteen-group, average cross section sets have been obtained for the $10\frac{1}{2}$", 8-3/8", and $6\frac{1}{2}$" Lx Pu-Al fueled lattices, both poisoned and unpoisoned. These cross sections were obtained from GAM-1 and SPECTRUM-V. The use of this set in the case of the $10\frac{1}{2}$", poisoned lattice gives a value of $k_\infty = 0.974$ when used in Program S. This case seems to have converged adequately. The calculated cadmium ratio at the cell boundary is 13.9 as compared to the experimental value of 15.2. Analysis of the other two lattice spacings is under way along with further analysis of the $10\frac{1}{2}$" lattice.

The Critical Facility

The rough drafts of the twenty-four PRCF startup tests are complete. Twenty-one of the tests have been distributed for final review and acceptance. Three tests are being revised and prepared for distribution for acceptance.

Assistance was provided on the Hazards evaluation of the PRCF quasi-uniform loading (Pu-Al-UO_2 supercells) and all Pu-Al loadings at 105 inch and 60 inch moderator levels. "Coolant" void coefficients were calculated using the IBM 7090 program SWAP.

The two flux traverses rigs have been completed.

Four different sets of BF_3 counters (each set consisting of a preamp, amplifier, high voltage supply and scalar) have been assembled into two mobile racks for use in the startup of the PRPCF. All of the components have been checked for proper operation and repaired as necessary. These racks are to be taken to the Critical Facility and the equipment sets contained in them are to be used with the particular BF_3 tubes that have been calibrated on them.

Six $\frac{1}{2}$" aluminum jacketed Hanford BF_3 counters have been calibrated on each of two different sets of equipment. The required plots of counts per minute vs. bias voltage have been made for the twelve calibrations.

Twelve $\frac{1}{4}$" aluminum jacketed Hanford BF_3 counters were received during the month which fills the counter requirements for the startup. At present, the calibration of these tubes is under way. Some difficulty has been experienced because the counting equipment in room 1-A picks up signals from the IBM Card Punch machine located in 24-A. These signals have as much as doubled the normal counting rates, depending on the operating speed of the card punch. The point at which these signals enter the system seems to have been located and methods of reducing their effect are being tested. Calibration work will continue as soon as a suitable method is found.

Critical Mass Studies for 1.8 w/o Pu-Al Fuel

The approach to critical with the two zone loading was completed using the low concentration of Pu-240 in the inner zone. The results as tabulated below, have not been analyzed by the least squares method.

Inner Zone			Outer Zone			Extrapolated to Critical	
No. of Rods	Pu-240 Pu Total	Avg. Pu Rod	No. of Rods	Pu-240 Pu Total	Avg. Pu Rod	No. of Rods	Pu
211	5.05%	7.06 gm	240	6.00%	7.19 gm	486	3215 gm

The lattice spacing used for the two zone loading is 0.85".

PRTR Fuel Irradiation Experiment

Gamma-ray activities have been analyzed from foils which were irradiated in the PRTR on unirradiated, Lx Pu-Al fuel elements. The foils were placed vertically on a fuel element and between two rods in the outer ring of the 19-rod cluster. Both bare and cadmium-covered foils were irradiated. The temperature of the moderator was 302°K and that of the coolant was 305°K for this test.

Relative activities of Lu-176m and Lu-177 have been calculated for the cadmium-covered and bare foils. Spectral indices and values of r, the parameter which characterizes the epithermal component of the neutron spectrum, have been calculated from these data as a function of position along the cluster.

The value of the spectral index was constant along the Pu-Al alloy and equal to 327 ± 10°K. For this analysis, it was assumed that the shape of the function which joins the slowing-down distribution has the form of

[redacted]

[redacted]

that measured by Johansson, et al., and reported in the Transactions of the ANS 3, 169, Paper 15-10 (1960). The analysis will be repeated using an energy distribution of neutrons calculated for the fuel region of the outer ring. In addition, data has been taken which will enable the scintillation counters to be recalibrated for the spectral index measurements.

The value of r was constant and equal to 0.064 ± 0.004. The values of r were calculated from the Lu-176m cadmium ratio. Values of 459 barns and 467 barns for the epicadmium (above 0.64 ev) and epithermal resonance integrals of Lu-175 respectively were assumed in the calculations.

Status of PRTR Fuel Irradiation Experiments

Coulometric titrations of 19 unirradiated samples of Pu-Al melts which were used in the six L_x Pu-Al physics test elements and element 5075 have been completed by Analytical Laboratories. The mass spectrographic analyses are 50% complete. The presence of iron in the aluminum, found in about 0.5 w/o quantities, has an appreciable (as much as 3%) effect on the Pu titration. A correction to the burnup results obtained on element 5075 by coulometric titration is required, since neither the unirradiated nor the irradiated samples were corrected for iron. The mass spectrographic analyses for Pu isotopic composition will provide a better basis for estimating burnup than the nominal composition which has been used to date.

Burnup analyses analogous to those performed on L_x Pu-Al element 5075 have begun on UO_2 element 1041, which was exposed to about 2000 MWD/T. Fifteen, one-inch samples have been cut off by Radiometallurgy, and one sample has been dissolved and coulometrically titrated for U, Pu, and Fe. Since the activity and Pu content of the solution was satisfactory for Analytical Laboratories' purposes, dissolution of the remaining samples will proceed.

The effect of the dissolution method, the addition of cesium carrier, and the presence of iron on the burnup analyses of Pu-Al elements will be investigated by taking adjacent samples from one rod of element 5042 being shipped to Radiometallurgy as part of the CPD dissolution process studies.

PRTR Theory-Experiment Correlation

A consistent set of cross sections for use in the analysis of the PRTR is being compiled.

The cross sections of selected isotopes in the RBU Basic Library are being updated. Upon completion of updating the cross sections in the library, a memo will be issued citing the references used and assumptions made in this compilation.

[redacted]

[redacted]

Program BARNS[8] has been altered slightly, so that the output can be obtained on punched cards. These cards become input for a program formulated to obtain the energy dependent absorption and fission cross sections ($\sigma_a \sqrt{E}$ and $\sigma_f \sqrt{E}$), as well as the scattering and transport cross sections ($\xi\, \sigma_s$ and $\left[1 - \bar{\mu}\right] \sigma_s$). This program's output is on punched cards which are in the format used in the TEMPEST[9] updating subroutine.

The GAM-1[10] updating program, yet to be completed, will permit the updating of the cross sections on the GAM-1 nuclear data tape, and the RBU cross sections can be used to update this tape.

With these three nuclear data tapes being updated from the same basic data, a consistent set of multi-group cross sections, for use in either a transport or diffusion analysis with no restrictions on energy structure, will be available.

Heterogeneous H_2O-Pu-Al Reactor Calculations: Statistical Weight of Fuel and Voids

A comparison of the calculated and experimentally critical number of 1.82 w/o Pu-Al rods in H_2O was made in the monthly report for June. The following results concern those lattices which had an effective plutonium isotopic content of 5.63 w/o Pu-240.

Shown in Figure 1 is the reactivity worth of a 1.0% uniform density decrease or void formation in the moderator associated with any rod in the lattice for various lattice spacings. The effect is generally positive in regions which are overmoderated; in the whole core for the larger lattice spacings, and always positive very near the reflector. Not included in this first order perturbation analysis is the effect of an altered neutron temperature which is expected to contribute a nearly constant negative term to the value of the integral of each distribution.

Figure 2 gives the worth of a fuel piece added at a periodic point in any of the critical lattices at any radius. The strong effect of leakage is

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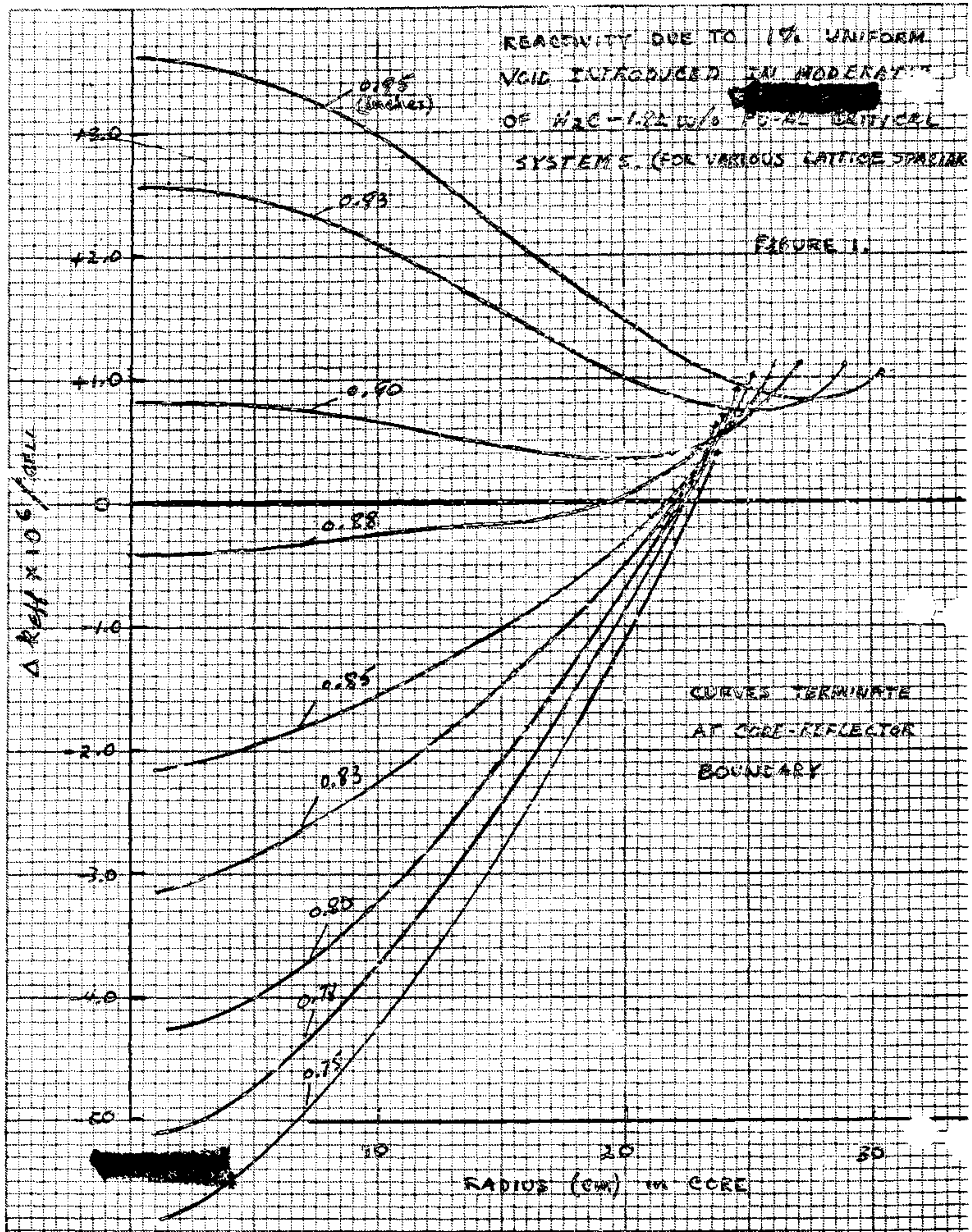
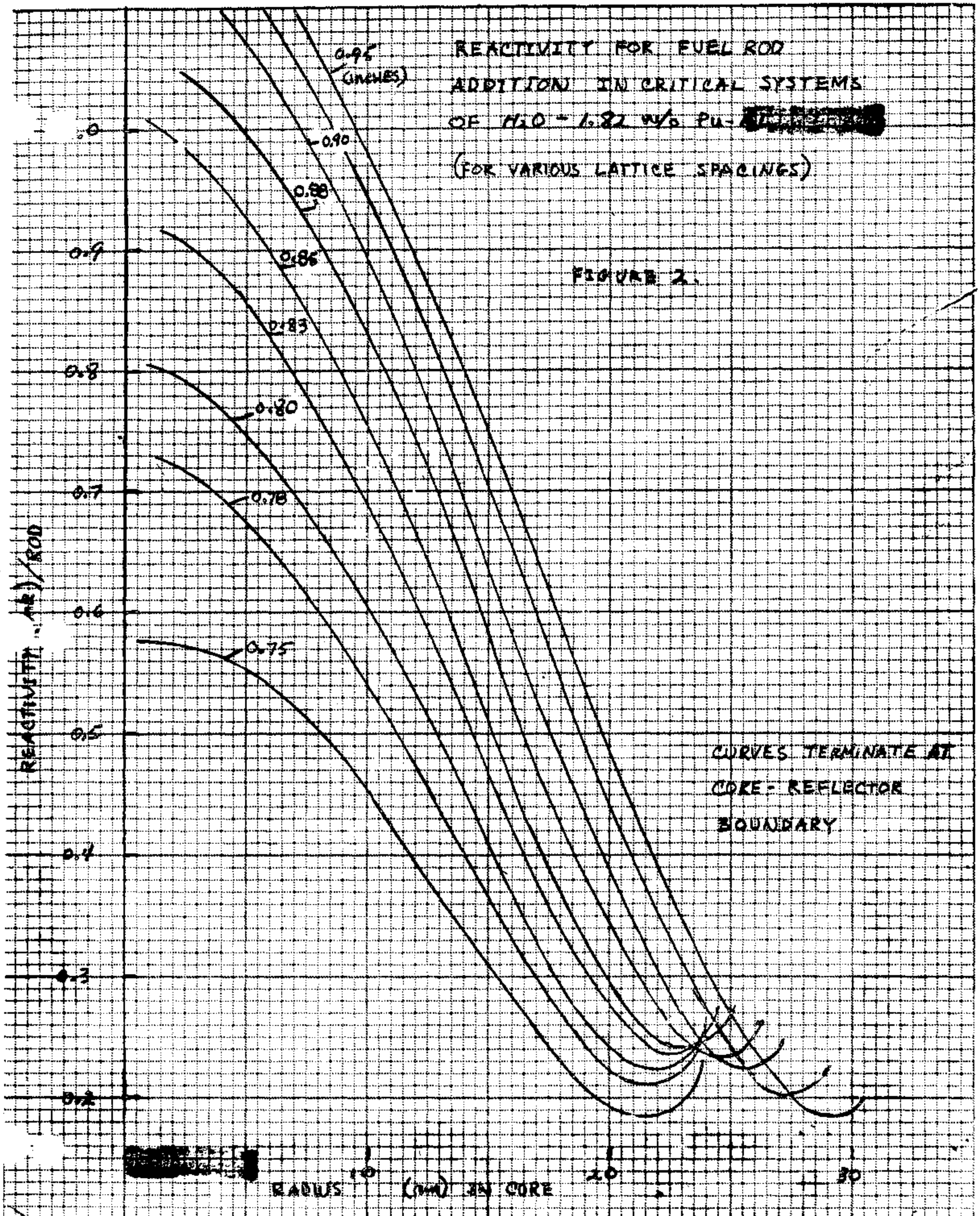
(8) Schlosser, J. E., "BARNS - A Program to Obtain Cross Sections from the RBU Basic Library," HW-72117, December 27, 1961.

(9) Shudde, R. H. and J. Dyer, "TEMPEST - A Neutron Thermalization Code," NAA Program Description, September, 1960.

(10) Joanou, G. D. and J. S. Dudek, "GAM-1 - A Consistent P1 Multi-Group Code for the Calculation of Fast Neutron Spectra and Multi-Group Constants," GA-1850, June 28, 1961.


REACTIVITY DUE TO 1% UNIFORM
VOID INTRODUCED IN MODERATOR [redacted]
OF H_2O-1.82 w/o Pu-Al CRITICAL
SYSTEMS. (FOR VARIOUS LATTICE SPACINGS)
FIGURE 1.
0.95 (inches)
0.93
0.90
0.88
0.85
0.83
0.80
0.78
0.75
CURVES TERMINATE
AT CORE-REFLECTOR
BOUNDARY
+3.0
+2.0
+1.0
0
-1.0
-2.0
-3.0
-4.0
-5.0
$\Delta k_{eff} \times 10^6$/CELL
10
20
30
RADIUS (cm) IN CORE
[redacted]


REACTIVITY FOR FUEL ROD
ADDITION IN CRITICAL SYSTEMS
OF H_2O - 1.82 w/o PU-[redacted]
(FOR VARIOUS LATTICE SPACINGS)
FIGURE 2.
0.95 (INCHES)
0.90
0.88
0.86
0.83
0.80
0.78
0.75
CURVES TERMINATE AT
CORE - REFLECTOR
BOUNDARY
REACTIVITY (Δk)/ROD
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
[redacted]
10
20
30
RADIUS (cm) IN CORE

1230974

dominant in the shape of each curve and does not reflect the competition between thermal utilization and fast leakage for the slowing down neutrons as in Figure 1.

Calculations were performed with the SWAP code in the manner used to calculate PRTR void coefficients[11].

Code Development

CALX

Debugging of portions of the CALX multi-group cell burnup code has begun. The test case, which uses only the main program and the Runge-Kutta numerical integration routine, is a one-group analysis of a reactor containing one fuel isotope and one fission product. Successful completion of this simple case, which can be solved analytically, will furnish a check on the accuracy of the numerical integration. No meaningful results have yet been obtained.

The CALX routines which compute the effects of fuel recycle and handle start-of-life reactivity adjustment are still being changed. The recycle routine, which is apparently complete, contains several features not included in MELEAGER. The code assumes that, on reactor shutdown, the fuel-fission product mix is stored for some time before chemical processing. The recovery efficiency of the chemical processing is isotope-dependent, and may be zero. A new fuel-isotope mix is constructed from the recovered materials and, optionally, a feed material. This mix is then stored for some time before insertion into a reactor. Reactivity initialization options will be able to use the separated isotope mixture as enrichment, as in the use of plutonium recycle enrichment with a natural uranium feed.

The multi-group flux and reactivity cell analysis (program ANNE) gives reactivity values which are in good agreement with comparable HFN results. The fluxes do not agree very well. Several changes, which will give CALX the multi-group theory analogs of some MELEAGER options, have been coded.

RBU

Theoretical work was completed on the double mass parameter approach to the treatment of binding and molecular effects on thermal neutron scattering. Different values of the mass parameters are used to force the RBU

(11) Staff of Hanford Laboratories Operation, "Plutonium Recycle Test Reactor Critical Test Results," HW-61900 BA, December 31, 1961.

[redacted]

[redacted]

Monte Carlo routine to generate different distributions having at any given energy the desired values for the average cosine of the scattering angle, $\overline{\mu}$, and for the average energy transfer, $\overline{\Delta E}$.

The expected values generated by Monte Carlo for $\overline{\mu}$ and $\overline{\Delta E}$ have been derived theoretically as functions of the mass parameters. In calculating $\overline{\mu}$ as generated by the Monte Carlo, integrals for the form, $\int e^{-x^2} E_{rf}(t(x))dx$, were encountered. Although first attempts to evaluate the integrals analytically proved unsuccessful, it was found that suitable transformations of this type of integral will yield the bivariate normal distribution function, which has been tabulated.

Work has started on using kernel codes and experimental measurements to obtain at a given energy the $\overline{\mu}$ and $\overline{\Delta E}$ for the actual physical moderators in question. These values will be compared with the $\overline{\mu}$ and $\overline{\Delta E}$ in the theoretical expressions representing the Monte Carlo results to determine the mass parameters which force the Monte Carlo to generate a $\overline{\mu}$ and $\overline{\Delta E}$ corresponding to those of the actual moderator.

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

Major portions of the work on "Fuel Re-Use" have been redone, using more realistic compositions and more detailed calculational procedures. Some of the implicit assumptions in the burnup calculations have been checked.

In the early fuel re-use calculations the fast reactor blanket loading was assumed to be pure U-238. Subsequent analysis showed that the residual U-235 has an important effect on the fuel economics. In the present calculations, the fast reactor blankets are loaded with depleted natural uranium (.380 w/o U-235).

Also, in the present calculations, the first cycle plutonium compositions have actually been obtained from machine burnup routines, while previously the Pu compositions were estimated by means of simple recipes.

In all fast reactor burnup calculations, the spectrum averaged microscopic cross sections are obtained only for the initial condition. It is assumed that they remain substantially constant during burnup. This assumption has now been checked. In the table below, absorption and fission cross sections of U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242 at various stages of burnup are listed. The variation of these cross sections seems to be small enough to justify a constant cross section burnup calculation. Such a procedure is very much simpler and cheaper than a calculation, wherein cross sections are changed at each time step.

[redacted]

[redacted]

Cross Sections as a Function of Exposure

Nvt $\left(\frac{\text{neuts}}{\text{cm}^2}\right)$		0	1.524×10^{21}	3.810×10^{21}
Mat'l.				
U-235	σ_a	3.111	3.054	2.977
	σ_f	2.375	2.336	2.285
U-238	σ_a	0.405	0.397	0.388
	σ_f	0.019	0.020	0.023
Pu-239	σ_a	2.853	2.818	2.772
	σ_f	2.116	2.100	2.080
Pu-240	σ_a	0.835	0.831	0.826
	σ_f	0.167	0.178	0.194
Pu-241	σ_a	3.839	3.776	3.691
	σ_f	3.348	3.297	3.230
Pu-242	σ_a	0.832	0.828	0.823
	σ_f	0.164	0.175	0.191

Critical Mass Calculations for Potassium-Cooled Cores

Work on physics statics calculations for compact space power units has been continued. The rubidium-cooled cores mentioned in the June monthly report have now been re-examined, using potassium as the coolant. For a very limited range of cores examined, the substitution of K for rubidium does not appear to affect the reactivity to a large extent.

Possible modes of reactor control have been briefly examined. Since the cores are small, very large reflector control margins seem achievable. The possibility of reducing control requirements by introducing Pu-240 and B-10 is also being considered.

Instrumentation and Systems Studies

Technical support continued regarding the fuel rupture monitoring and critical facility work. Since PRTR was inoperative most of the month, the rupture monitor testing and modification work was curtailed.

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The second generation scintillation transistorized effluent monitor for gamma emitters for use at PRTR was completed including the installation of developed adjustable electronic trip (alarm) circuits for both low and high level alarm purposes. The instrument is undergoing complete laboratory testing before installation at PRTR. All tests, to date, have been satisfactory.

The extended PRTR outage during the months of June and July has resulted in a postponement of the scheduled systems tests (PRTR Tests No. 19 and 35). Two memoranda were prepared (SRO 62-23 and 62-24) to describe the test apparatus and to analyze some preliminary test data. A brief recording of moderator level fluctuations was made on a two-channel strip recorder, using both the servomanometer and the "dip-tube" transmitters as signal inputs. The dip-tube transmitter output contained an approximately sinusoidal component with a dominant frequency of approximately 9 to 11 cycles per second. There also appeared to be a lower frequency component present, approximately one cycle per second, although it was masked by the 9 to 11 cycle components. The servomanometer output signal definitely contained a strong one cycle component. Therefore, it appears that there is a system resonance near one cycle per second which is being excited by some local disturbance, possibly the relatively small compressor pressure surges. The source of the 9 to 11 cycle per second signal is not known but it could be a function of the dip tube "bubbling" rate. Tape recordings of these signals will be obtained during the next reactor operating period. Analysis of these recordings are expected to yield accurate information on the relative frequency content of the signals and will aid in determining their source.

An electrical transducer was connected to the PRTR pressurizer level transmitter output signal for use in studying pressurizer level fluctuations. Initial tests of a system for determining the power-density spectrum of reactor noise by means of Fourier transformation have been completed. The final configuration produced chart recordings of the spectrum directly, with power in db as ordinate and frequency along the abscissa on a logarithmically-compressed and indexed scale. Available power range was about 40 db, over frequencies extending from 0.01 to 600 cycles per second. (These ranges can be extended as necessary.) Trace rate was ten minutes per decade of frequency. This method of analysis is perhaps an order of magnitude faster than the initial point-at-a-time technique; in addition, it requires little surveillance on the part of the operator. If properly applied, it is also more accurate. As the system is controlled by a multi-channel tape recorder, it can be programmed to automatically make multiple re-runs in order to provide added accuracy. Other instructions might be programmed as needed. The Sigmatron random signal generator was used to provide the system input, so the above tests served also to check

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the spectrum of this unit, which was purchased recently. The 100-cycle range appeared to be very flat; the 10-cycle range evidenced a four db peak extending from 1.0 to 10 cycles. The 1000-cycle range is beyond the present frequency range of analysis. It is expected that information gained from this set of tests will apply directly to a contemplated cross-spectral analyzer to be used in determining the correlation between moderator-level and power-level fluctuations at PRTR. As such, an analyzer would require a considerable number of multipliers; further studies would require use of the EASE analog computer.

The Critical Facility study is a hazards study designed to determine the nuclear accident potential of the reactor using various types of fuel elements in the core. Since the moderator-coolant of the Critical Facility reactor is in direct contact with the fuel, previous studies were made using a greatly simplified heat transfer model. The current study uses a much more elaborate model including steam and void formation in the moderator and the effect of moderator boiling at the surface of the fuel elements. The design of the simulation has been completed and the study is now in progress on the analog computer.

Coils, for the Mark II probe for measuring the process-to-shroud tube annulus in the PRTR, were wound, balanced, and delivered to Structural Materials Operation; also, some technical advice concerning the assembly and testing of the electrical connections within the probe body was given.

Meetings with PRTR Operations and PRTR Equipment Development personnel were attended to discuss methods of measuring the frequency and amplitude of fuel assembly vibrations, both in the PRTR and in the hot-loop mockup facility in the 314 Building. It is suspected that some movement (vibration) of the fuel element assemblies with respect to the process tubes is reasponsible for at least part of the corrosion marks which have been found on the inside walls of these tubes. A bench mockup of this situation is being assembled, and initial tests have shown that this motion can be measured quite accurately using eddy current techniques. Sensitivities of approximately one millivolt per .001 inch change in process-to-sheath tube spacing were achieved with very little effort. At this time it would appear that the major problems involved in obtaining such a measurement will be encountered in materials selection (to withstand the extreme environmental conditions) and in accessibility requirements. Further investigations will be conducted in an effort to determine the most practical solution to this difficult measurement problem.

NEUTRON FLUX MONITORS

The interim technical study report, titled Feasibility Study of In-Core Neutron Flux Monitoring with Regenerating Detectors, has been typed by Technical Publications, HLO, checked for errors, and is presently being printed.

The neutron flux monitor IBM computer program was revised to utilize more significant figures during the matrix inversion process and several calculational options were added to provide a more useful and versatile program. It became apparent in several cases, however, that the true optimal composition of a regenerating detector for maximum lifetime would require negative quantities of certain isotopes, and since this is impossible to attain physically, it became necessary to modify or revise the computer program.

For testing purposes, a mass spectrometer can be used to determine the spectral parameters of the neutron flux in a test facility in which the regenerating detector elements are to be irradiated. The detector isotopic composition, as a function of exposure, is dependent on the cross section values of the included isotopes.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The new graphical alternating current nulling unit being developed for use with either conventional or multiparameter eddy current nondestructive testing equipment was incorporated in a tubing tester and laboratory tests confirm the anticipated high level of utility of the device. Construction of prototype multiparameter eddy current testing equipment was started, progressing concurrently with further design on the remaining circuits. Exploratory work was started in the use of a low melting temperature alloy, 65°C, for eddy current model tests, in which movable internal probes can be used in the liquid state for studying current and field distributions, and in which fabricated defects can be readily inserted for tests in either liquid or solid states.

The graphical nulling device in its present state of development has a curvilinear geometry due to compromises made in its design in the interest of simplicity in fabrication. A graphical analysis of the nature of the nonlinearities of the device was made, and correlation was observed between test patterns and the results of the graphical analysis. In spite of these nonlinearities, the unit works very well when operating with a tubing tester operating at 500 kc. Plots of test coil output voltage were

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readily obtained for different test specimen conditions, and the detector input of the tester was quickly adjusted to the desired operating point. Several potential problem areas were observed including adjustment of scale factor and phase of the nulling signal, and decrease in test signal sensitivity due to shunting effects all of which will require further attention.

The design work on the multi-frequency eddy current test unit is proceeding. Four L-C type oscillators have been constructed for evaluation. These oscillators produce frequencies of 6 kc, 22 kc, 70 kc, and 200 kc. An ultra-linear cathode follower has been constructed to add these signals together and produce a signal from a low output impedance source. Preliminary checks indicate this signal source is operating properly.

In addition to the above circuits, breadboard circuits of adders, sign changers, and active filter circuits have been evaluated. The characteristics of several commercial operational amplifiers have been determined in the laboratory. Some amplifiers will be ordered for use in the adder and sign changing circuits.

A literature search is being conducted on the diffusion and propagation of eddy currents in conductors.

A simple, direct way to fabricate a variety of discontinuities in metal test specimens is desired to facilitate the demonstration and adjustment of the multiparameter eddy current test equipment and for studying the diffusion and distribution of eddy currents. In the past mercury has been used for eddy current models, but it has the disadvantage of being toxic and having a low freezing temperature, -39°C. Initial tests show that Woods alloy, which freezes at about 70°C, may be useful for the above purposes. Thin pieces of insulating material can be frozen in the metal to produce a variety of simulated defects of known size and shape. Test probes for measuring current density can be used in the liquid metal and possibly in the solid metal, subject to determination of the uniformity of probe contact resistance during cooling.

Zirconium Hydride Detection

Changes in indicated ultrasonic attenuation in Zircaloy-2 samples were observed during stress application, but were believed to be largely due to changes in the glycerine coupling film. A sample holding jig to permit more refined measurements is being designed.

A thin film of glycerine was used as a couplant in the stress dependent attenuation measurements. The 15 Mc quartz crystal used in these measurements was mounted in a jig which was clamped to the sample. Changes in the

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ultrasonic attenuation, observed during application of stress, were not reproducible upon removing the sample from the jig and remounting it. This lack of reproducibility is believed due to the effect of small differences in thickness of the thin glycerine coupling film which changes during elastic straining of the sample surface. A longer coupling path from the crystal to the sample should reduce changes in apparent attenuation which result from small changes in sample surface curvature during loading. Design of a sample holding jig to allow water coupling path lengths up to 16 inches is under way. Tests with a mockup of the jig, using a small laboratory press, show that it will be necessary to apply stress to the samples through a guided push rod. The guides must be mounted rigidly on the sample holding jig to prevent rotation of the sample during loading. Goniometers required to accurately position the transducer crystal with respect to the sample will also be incorporated in the jig.

Heat Transfer Testing

An electronic R-C analog simulator has been fabricated for use in studying the effect of varying heat transfer testing conditions in aluminum clad uranium fuel elements, and a ratio circuit for use in the proposed emissivity independent heat transfer tester has been developed.

The analog simulator allows determination of the effect of variations in bond conductances and other test conditions. Transient fuel element temperatures which would result from a given set of test conditions may easily be recorded as voltages at various points in the simulator. The simulator should aid in optimizing test conditions for detecting differences in overall bond conductance. It should also be useful in determining the minimum bond conductance differences that can be detected with the new dual radiometer emissivity independent heat transfer tester now under development. Minimum surface temperature differences that can be detected with the new dual radiometers must be known before such a determination can be made.

It is necessary to continuously take the ratio of the two, 2 kc carriers from the dual radiometers to obtain the emissivity independent temperature information. A circuit which will accomplish this has been developed. Initial tests showed that the output of the circuit was the ratio of the two carriers within about 2% (at a bandpass of one kc). Inputs to the circuit can be varied by a factor of two. This circuit should also be useful in eddy current, and other nondestructive tests where it is sometimes necessary to find a ratio.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Ultrasonic response measurements were continued on evaluating drilled holes for use as bench-marks for production testing of sheath tubing. Measurements of ultrasonic response as a function of entry angle were made with a 5 Mc, flat reactangular, lithium-sulfate transducer on tubing containing a 5 mil diameter hole through the tube wall plugged with paraffin. The response obtained correlated well with mode propagation at the theoretically predicted angles.

A number of tests were made for the evaluation of production test parameters.

Postulating Lamb-wave propagation for the testing of thin-walled tubing, simpler behavior is realized by the use of the lower modes. To achieve lower modes, for the wall thicknesses involved, will require testing at 5 Mc. Accordingly, tests were made using this frequency, reflecting ultrasound from both notches and holes. In all cases the theoretical behavior was confirmed, indicating lower mode excitation is practical.

However, a disturbing element in the use of the lower modes is the theoretically predicted behavior for response to depth of discontinuities. Calculations show the ultrasonic response to be double valued and preliminary tests appear to confirm this condition. Measurements of ultrasonic response as function of notch depth were made at 5 Mc using a flat rectangular lithium-sulfate transducer. The measurements were made at entry angles at which the second and third Lamb-wave modes would propagate. The preliminary experimental results indicate that the response amplitude increases rapidly, reaches a maximum value, and decreases slightly. In confirmation of the predicted response, amplitude from a 17.5 mil notch was approximately the same as from a 3.5 mil deep notch. Double values would not be troublesome if the ultrasonic test were restricted to detection of only the very fine discontinuities.

Transducer studies were made using a set of holes drilled in Zircaloy tubing, to compare Lamb-wave sound fields in tubing with the fields measured by reflections from ball-bearings. The longitudinal and transverse Lamb-wave test alignments were used to evaluate beam shapes of focused circular transducers, and flat rectangular and flat circular transducers covered with rectangular masks. For the transducers with rectangular beam shapes the two alignments were used to evaluate the fields in the long and short dimensions respectively. The focused transducers were adjusted with water-path distances equal to focal lengths. For the masked transducers, the water-path was made equal to the distance of the last near field peak in the short dimension. The beam irregularities obtained with the drilled holes correlated well with the previous measurements using ball-bearings which indicates that both methods will provide suitable calibration of transducer beams.

Instrumentation evaluation work this month involved transmitted ultrasound frequency and comparison of commercial units. The frequency of ultrasound can vary considerably depending on transducer center frequency, type of pulse, and connecting cable combinations used. Lamb-wave testing is frequency dependent and for valid interpretation of test data transmission frequencies must be controllable. By tuning with inductance in series or parallel with the transducer, transmitted frequencies were respectively lowered or raised to desired values. The proper frequency was determined by observing the signal reflected from a flat surface with a frequency calibrated oscilloscope connected in parallel with the inductance transducer combination. The reflected signal was first maximized in amplitude by mechanically adjusting the transducer with relation to the flat surface. The frequency of the reflected pulse was then observed during inductance tuning adjustments. A flat reflection surface is not essential. However, the signals reflected from curved surfaces have modulation components which make measurement of frequency difficult. Frequency spectrum analysis following this method of tuning confirmed that the reflected signals had the center frequencies observed with the calibrated oscilloscope. Measurements of ultrasonic response as a function of entry angle were made with two ultrasonic instruments representative of commercially available units. The measurements for the two instruments correlated fairly well in indicating mode propagation at theoretically predicted angles.

Fundamental studies to make clear the physical picture of Lamb-wave propagation behavior continued. To investigate the effects of multiple Lamb-wave propagation, a Zircaloy plate was caused to vibrate in the third symmetrical and asymmetrical modes simultaneously. The end leakage observed with the Schlieren system was found to have an unbalanced pattern about the centerline. During multi-mode propagation cancellation and reinforcement can take place through interference of in-phase and out-of-phase wave components. The resulting unbalance of waves about the centerline, causing non-uniform particle vibrations, may give rise to a different detection sensitivity for external surface tubing defects as compared to internal surface defects. A difference in response to external defects compared to internal defects has been experimentally observed frequently with existing pulse-echo test equipment. Simultaneous propagation of more than one Lamb-mode is probable with this equipment since the transmitted pulses have a band of frequencies. Where equal internal and external response is desirable multi-mode propagation should very likely be avoided.

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BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Off-site work in the atmospheric diffusion program at Vandenberg Air Force Base, California, was completed. Twenty-eight field experiments were successfully conducted in this last series, giving a total of 108 experiments at that site. A variety of meteorological conditions were embraced by the studies which, also, included topographic and sea-land interactions. First use of these data will be to provide the basis for computer controlled atmospheric dispersion predictions for fuel handling and advanced missile firings from current meteorological data. Assaying of all field samples, which lagged the experimental program by three weeks, was completed during the month, and editing of the data sheets started.

Field activities in atmospheric dispersion and deposition studies at Hanford increased during the month with return of the on-site force to full strength. On July 22, field forces started on a special series of experiments to be conducted during the night-time hours over a period of two weeks. These experiments were designed to clarify certain features noted in the 1959 Green Glow data relating to emission duration, travel time to a given distance, and averaging time of meteorological parameters in relation to the dispersion parameters during stable atmospheric conditions. By month end, eight experiments had been completed, of which four included sampling at eight miles' distance from the ground level source.

As noted in previous reports, the rate of horizontal plume growth in the distance interval two to eight miles was slower than expected and was partially attributed to the sharp change in elevation at a distance of six miles from the source. Later analyses of the wind velocity spectrum during selected experiments showed that such a plume growth could arise from a dominant one to two-hour meander eddy when the release period was thirty minutes. Therefore, the release period during the current series of experiments has ranged from one-half to three and one-half hours in an effort to determine the meander effect. In addition, drum samplers were operated at two and eight miles to obtain the time of arrival of the tracer material and the time history of collection of the measured dosages.

Dosimetry

Measurement of the radioactivity in Alaskan Eskimos was continued. More than three hundred people were counted at Barrow. The amounts of Cs-137 found were generally less than those found at Kotzebue. This was related to a lower activity density in the meat available at Barrow. The equipment is being moved to Anaktuvuk Pass for further studies.

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A study was started of the use of pulse shape discrimination circuits, such as are used in neutron scintillation counters, for noise suppression in low-energy photon scintillation counters. Initial results were very promising.

The positive ion Van de Graaff developed excessive sparking during the month. Two of the glass sections of the accelerating tube were found to be damaged. The accelerator operated satisfactorily for a few days with these sections shorted out but then failed again. It was found that two of the springs used to make electrical contact with the accelerating tube had worked out of position and made contact with the glass. As a result the glass became very badly chipped which led to the failure. A new accelerating tube was promptly obtained and installed. It is now being broken in. It is not behaving as other tubes have and it may turn out to be unsatisfactory.

The helium ion source that we developed for the Van de Graaff supplies enough current, but too much of it is singly charged helium instead of the doubly charged helium that we need to use. A device was conceived, designed, and is being fabricated that will eliminate most of the singly charged helium as it leaves the ion source.

Arrangements were made with Mound Laboratory and Argonne National Laboratory for the loan of one of our precision long counters so that they can compare theirs with it.

The measurement of near-background neutron fields at different Hanford locations was completed.

Investigations were made to see if a calorimeter calibration error similar to that recently found in our plutonium calorimetry might have existed in any of our earlier work. The results showed an effect of a few tenths of a percent. This is much less than found in the plutonium work. It will not have an important effect on our results.

Radiation Instruments

Tests were conducted on several of six prototype automatic recharging dosimeters, fabricated off-site, for use in the pocket indicating and signalling dose meters. The general fabrication quality was not considered fully acceptable and appropriate comments were conveyed to the manufacturer for incorporation on the remaining 44 units. One dosimeter, tested to failure, worked to an integrated dose of 1200 r. The annoying problem of the fiber sticking to the center rod after extensive cycling is still apparent and tests have shown the effect to be, probably, caused by welding and general surface roughening.

One miniature dose meter prototype was completed in essentially final form in a package which is easily inserted into a shirt pocket. This unit, which performed correctly for tests, incorporates the microminiature binary circuits and indicating lights, and a selectable-level signalling (alarming) trip circuit. Initially, one of the off-site fabricated dosimeters, previously untested, was installed; however, this particular one was determined to have a broken rod. A new "pencil" is being installed and the dose meter will then be ready for demonstration and further testing. At least one more of the indicating and signalling units will be fabricated and two of the indicating-only dose meters will also be assembled.

Experiments were started to develop a relatively insensitive pocket-type signalling dose meter for integrated dose alarm levels of 25 r and greater. With the knowledge gained by extensive work on the sensitive dose meters, the problems involved on such an insensitive scale are comparatively less difficult. The insensitive units will be especially applicable for plant disaster, criticality, and perhaps civil defense needs.

A dual output transistorized pulse generator was developed and tested for use with the just completed "field model" coincident-count alpha continuous air monitor and with the prototype coincident-count alpha air filter counter which was developed to permit rapid assessment alpha contamination on air filters in use throughout HAPO. The developed pulse generator provides a main or initial pulse at one terminal followed by a delayed pulse at the other. The delay time is adjustable from 10 μsec to 400 μsec to permit proper gate adjustment testing of the coincident-count type alpha counters. Tests with the pulse generator were successful and the general operation of both of the experimental alpha counters has been satisfactory for all laboratory testing completed.

A second transistorized pulse generator was developed which requires no battery or 115 VAC line power for operation. The unit derives its necessary operating power from the usual laboratory-type sinusoidal oscillator normally used to vary the pulse rate. The oscillator also supplies the trigger to cause pulse formation. A report was issued.

A relatively simple, yet stable, four to six watts output audio alarming circuit was developed and tested. The circuit uses a variable frequency Unijunction transistor oscillator and two power transistors which drive dual loudspeakers. The alarm unit can be either battery or 115 VAC line powered and can be activated by either a switch or by relay contacts. It was developed for use, in particular, with area-type radiation monitors to provide a suitable "local" alarm, and it is especially valuable for high ambient noise locations.

General design was completed on the Mark II Critical Radiation Monitor for general HLO applications. The design supersedes the Mark I unit, of which some 15 are still in satisfactory use after about 2½ years. The forestated audio alarm circuit will be used to provide low-level type local alarming where the monitor is so employed. For the critical radiation ("criticality") use, the alarm used will be the HAPO Standard which is a cycling Klaxon horn.

Experiments continued with silicon solid state surface barrier alpha particle detectors regarding their application as sensors for continuous alpha air monitoring. Atmospheric inversions, with consequent buildup of radon and thoron on the air filters, resulted in the obtaining of data showing the alpha pulse height distribution of the radon and thoron. The exposed filters were measured with the solid state detectors and the multichannel analyzer. The obtained data were compared with the alpha energy spectrum from Pu-239. With adequate sensor energy resolution and for low dust loading, which causes spectral smearing and shifting, it appears that the application could possibly provide an airborne alpha sensitivity comparable to that obtained with the dual phototube coincident-count alpha air monitor. Several more extensive inversions must be awaited to obtain the necessary complete data.

Several more successful field tests were performed using the experimental continuous operation type zinc sulfide particle detection instrument which was developed for use in air movement and diffusion studies. A method is being developed, using a zinc sulfide screen and an alpha source, to provide calibration of the instrument in the field.

The experimental scintillation transistorized six decade, 1 mr/hr to 1000 r/hr, logarithmic response area radiation monitor was completed and was started in testing. General stability was acceptable; however, the electronic alarm trip circuits showed too much variation in trip point with temperature and the circuits were modified to obtain the necessary stability. Tests, to date, indicate trip point stability of ± 3% of full scale from 70°F to 140°F. Experiments will continue to secure improvement. Operation for two weeks at 250 mr/hr at the probe showed less than ± 2% variation and no general shifting. Fabrication was started on a prototype logarithmic and multi-decade range linear response area radiation monitor which employs several new circuits. A different type of alarm trip circuit will be employed, for comparison purposes, than was used in the forestated logarithmic response-only monitor.

Circuit modification and general rebuilding continued on the experimental beta-gamma hand and shoe monitor which employs gamma background suppression or compensation circuitry. The scintillation transistorized instrument will employ electronic dc trip (alarm) circuits in place of the meter-relays originally used, and in addition, the gamma background detecting probe will be redesigned and located in a different location in the instrument to provide better compensation at higher background levels. An electronic timer circuit was designed to replace the electro-mechanical timer which had been used in the instrument. The change reduced the volume required and the cost. The transistorized timer has an operating range adjustable from one to 60 seconds; the "normal" hand and shoe counting time required is about 15 seconds.

Fabrication continued on the instrumentation for use with the Portable Mast of Atmospheric Physics Operation. All circuitry designed and fabricated to date has performed correctly and several completed portions are undergoing continuous testing while fabrication moves forward on the other sections. Considerable fabrication and assembly work remains to be done; however, a majority of the design effort has been completed.

A special low noise level multiplier phototube was received for use in investigations of low level liquid effluent beta and gamma counting. A transistor emitter follower circuit was designed and fabricated for use with the detector.

A low-noise level, four stage, transistor pulse preamplifier was designed on paper and partly fabricated for testing.

Work was started again on the 400-channel analyzer for the positive ion accelerator. Some redesigning is being done to incorporate new ideas that were obtained by building the 1024-channel time-of-flight analyzer.

An informal report was issued describing the new manual keyboard entry for the Whole Body Counting Facility.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples as received during the month using the mass spectrometer for this program. In addition, two uranium samples were analyzed for Ceramic Fuels Development in order to attain the precision and high sample sensitivity required for these samples.

A series of analyses on uranium standard samples were performed during the month to calibrate the present performance of the spectrometer. The results of these measurements indicated that a bias of -0.6% ± 0.3% exists in the measurement of a U^{235}/U^{238} ratio independent of the value of the ratio. The 95 percent confidence interval on the measured isotopic ratio for a single analysis was determined to be ± 1.4 percent of the ratio.

Further work continued on the use of a solid-state alpha-particle spectrometer system for routine assay of loaded sample filaments. The alpha-particle pulse-height spectrum observed from loaded sample filaments has been found to be very poor. A significant contribution of pulses presumably due to alpha particles degraded in energy is observed. The reason for the apparent energy degradation has not been determined at this time.

TEST REACTOR OPERATIONS

Operation of the PCTR was routine during July although there was one unscheduled shutdown. The unscheduled shutdown resulted from electronic failure.

The experiment to determine the nuclearly safe concentration of uranyl fluoride was completed.

The experiment to measure the angular distribution of thermal neutrons at a graphite cadmium interface was completed.

The experiment to determine design criteria for the proposed HTLTR was started during the month.

The following maintenance item was completed during the month:

The fabrication and installation of the metal fuel storage racks.

The TTR was operated two times during the month to activate foils for calibration purposes.

The first two zone experiment for 1.8% Pu-Al fuel in a 0.85 inch lattice was completed in the critical approach tank.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

Consultation service was rendered on meteorological and climatological aspects of 1) distance criteria for industrial development in the vicinity of 300 Area to RPO, 2) radioactive carbon dioxide releases from reactors' sites to RPO for IPD, and 3) siting of off-site environmental monitoring stations to RPO.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	93	88.6
24-Hour General	62	89.6
Special	166	92.2

Temperatures during July averaged near normal. This average was the result of unseasonably cool temperatures during the first 19 days, followed by one continuous heat wave during the remainder of the month. A low of 42 on the 2nd came within one degree of equaling the all-time record low for the month. Maxima of 100 or above on the last 10 days of July and first day of August equaled the all-time record of 11 consecutive 100-degree days established July 13-23, 1938.

Only precipitation during July was a trace on the 4th.

Instrumentation

Installation of all mechanical components of the Automatic Conveyor Type Alpha-Beta-Gamma Laundry Monitor was completed. Both survey stations are installed and operating. Except for minor repairs for light leaks in several probes, the equipment performed properly. The control console is in service and one complete 24-hour electronics test was made with no false alarms. Following adjustment of the station sequencing circuits and drop stations, the complete system will be ready for a series of continuous operational tests. The instruction and maintenance manual has been partly completed.

Information and advice were rendered to instrument technicians at 100-H Area regarding the conversion of their G.M. tube low-background counting system to use a detector with an organic scintillator. Many HAPO locations have now carried out this conversion with excellent results.

Information was rendered to Radiological Engineering, IPD, regarding the use of nickel-cadmium batteries and a simple battery recharger for direct use in the IPD pencil dosimeter charger-reader.

Five transistorized scintillation beta-gamma air monitors, which use a filter tape transport mechanism, were completed in on-site fabrication for

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use in the Radiometallurgy Laboratory. All five units were inspected, tested, calibrated, and are ready for delivery.

The completed prototype coincident-count alpha air filter counter, for immediate counting of standard HAPO 4 inch x 8 inch filters, operated correctly for the month. More than 100 filters from the 308 Building were test-counted using a preset time of three minutes. All data plotted satisfactorily. In order to eliminate the necessity of hand-plotting, a method is being tried of using the preset count mode on the coincidence scaler and a fixed time interval for system control. If the method proves successful, both very low-level and also high level contaminated filters can be counted without resorting to hand-plotting of data. This counter was developed for Radiation Monitoring, RPO.

Fabrication, on-site, was completed on all electronics and on the detector head assembly for the continuous coincident-count alpha air monitor. Following initial circuit tests, the complete instrument was placed in operation for continuous testing. This unit is the "field model" with the design based on the originally-developed prototype. The scintillation transistorized instrument was designed for Radiation Protection Operation.

Instructions were given to Biology Operation personnel regarding general operation of the transistorized scintillation self-powered radiation monitor and analyzer which was developed for use in general field studies.

A lead shield collimator was designed and fabricated for use with an integral assembly scintillation probe for Biology Operation. In addition, a new transistorized pulse preamplifier was designed and installed in the probe to replace the unsatisfactory commercial vacuum-tube preamplifier.

Discussions were held with representatives of Instron Engineering Corp. to plan the inclusion of shaft-position encoders in the tensile-strength testing machine which they are building for the Fuels Development Operation. The encoders will provide digitized stress-strain data for conversion to punched paper tape readout.

Components are being procured for a paper tape readout system for the new underwater fuel element measuring device. The system is similar to one previously developed for the 306 Building facility.

Calibration of micro-displacement readout systems to be used by Physical Metallurgy Operation for in-reactor creep measurements has continued during July. To date, the basic calibration of the transducer system used in the third generation creep capsules is approximately 65% completed. Since this readout system employs a transistorized dc-dc converter for excitation and demodulation, a series of tests to determine the effects of changing the ambient temperature of the dc-dc converter will be conducted. Another series of tests is planned to determine whether or not the calibration constants vary as a function of the zero control setting.

Zener diode reference voltage packages were installed on each Protecto-Vane temperature limiter at 100-KW in-reactor test facility. It will be recalled that because of extreme line voltage sensitivity in the upscale thermocouple burnout circuits of the Protecto-Vanes, four capsules were dropped out of operation in May. The installation of the reference voltage packages seems to have solved the line voltage sensitivity problems, and for the first time the POV's are indicating the correct temperature. Work continued on the solid state scanner-programmer and it is expected that a prototype micropositioner digital control system will be installed at 100-KW within a week. The Non-Linear Systems digital voltmeter system to be used in the creep capsule data logging system arrived and has been installed at 100-KW. Until other portions of the data logging system are completed, the DVM will be used to manually read the variable permeance x-ducers in the creep capsule.

During optimization of the temperature control systems for the swelling capsule test, it was found that the open-loop transfer function of the system is nonlinear. There are three basic modes of reactor operation, each of which affects the transfer function of the capsule:

1. Reactor down, water pressure down - medium open loop gain.

2. Reactor down, water pressure up - open loop gain high, closer to instability.

3. Reactor up - open loop gain low, easy to control.

Step response tests of the capsules with the reactor down (water pressure up) and with the reactor up gave the following results:

System No.	Open Loop Gain	
	Reactor Up	Reactor Down, WP Up
15	104° C/ma	240° C/ma
16	72° C/ma	140° C/ma

From these data, it can be seen that the open loop gain of both systems doubled between the two reactor modes. This indicates that in order to control at an optimum under all reactor modes, one of the elements of the open loop transfer function must be made nonlinear to compensate for the nonlinearity of the capsule. This could be accomplished by nonlinearizing the static transfer function of the mag-amp, SR unit in such a way as to increase the gain at low power demand and decrease the gain at high power demand. A compromise, as is now the condition at 100-KW, would be to set the control system to an optimum under the "worst" mode (reactor down, water pressure up) and thus control with a slower response time under all other modes.

A review was made of the Minneapolis-Honeywell specifications for the data logging system to be used in the swelling capsule experiments. The vendor appears to have taken several exceptions to our specification HW-73150. Our approval of the M-H system is being withheld pending clarification of these discrepancies.

Optics

A special microscope was assembled, tested, and delivered to the Lawrence Radiation Laboratory. The microscope is adapted to measuring depth of scratches and grooves in objects separated from the user by a hood wall. It is sealed to prevent air flow through the hood wall. Measurements of the width of grooves and of the radius of curvature of the four corners of the groove may also be made. The accuracy and repeatability of the measurements is ± 80 microinches. An informal report, HW-74345, was written to serve as a guide for the use of the instrument.

During the three-week period (July 1-July 22), covered by this report, a total of 256 man-hours shop work was performed. The work included:

1. Completion of the fabrication and assembly of the Groove Depth Microscope.
2. Fabrication and assembly of the electrical readout traverse mechanism.
3. Fabrication of 50 pyrex wheels for FPD.
4. Fabrication of 10 glass bearings for CPD.
5. Repair of two camera shutters.
6. Fabrication of components for a microscope closed circuit TV system for Radiological Development and Calibrations Operation.
7. Repair of a crane periscope head for Redox.

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Physical Testing

The usual nondestructive and supplementary tests were employed in performing customer testing work. A new capability is being developed for vibration testing and measurements. A total of 5,740 tests were made on 3,843 items, representing some 62,695 feet of material. An increased footage was realized this month as work with tubular products increased. Work was done for twenty-two different HAPO components representing most of the operating departments and service organizations, and other AEC contractors. Advice was given on 49 different occasions on general testing theory and applications.

Anticipating the completion of the NPR pressure tube work and to maintain a tube facility on plant capable of handling all sizes and types of tubular components, the Commission agreed to a transfer of NPR project tube testing and treatment equipment to HLO. For the present, arrangements with Kaiser Engineers will continue as in the past, where the equipment and facilities are available for NPR project work on a priority basis. Salvage NPR tube work remains to be done on tubes damaged upon installation.

A limited amount of testing will be done on zirconium tubes being procured for K reactors installation.

Eddy current testing, using an encircling coil, of some 150,000 feet (over 28 miles) of small diameter Inconel instrument lead lines for NPR was almost completed. Preliminary results indicate a rejection of some 4% may be necessary depending upon burst test evaluations currently being made. Burst test data to date have correlated with eddy current indications in that the larger indications burst at lower pressures.

Damage was detected in the vaned flow-straightening portion of the PRTR primary loop by radiography and was confirmed visually. Ultrasonic testing disclosed minor discontinuities in the weld of a replacement section; however, the section was installed with provisions for periodic testing.

Physical Testing Operation was requested to attempt to radiograph a potentially defective area in a section of the primary loop of PRTR containing flow straightening vanes. The section of line involved was 14 inches in diameter with a 1-inch-thick wall filled with heavy water. Using Co^{60}, satisfactory radiographs were obtained indicating that a sizeable section (6" x 15") of one straightening vane was missing.

On the basis of the above information, it was decided to remove the damaged fitting and repair or replace it. Subsequent cut-out confirmed the radio-

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graphic findings. In addition, a fluorescent penetrant test on the radioactively hot section revealed another vane containing an 8-inch-long crack.

Ultrasonic testing of the longitudinal fusion weld of a replacement pipe for PRTR showed small (5-20 mil) discontinuities present. The pipe was cross-sectioned at one end and the weld metallographically examined in an attempt to ascertain the nature of the discontinuities. The weld appeared to be sound and free of discontinuities; unfortunately, only this one surface was available at the extreme end of the pipe and may not have been representative. On the basis of the small size of the discontinuities, and from experience with fatigue tests on the NPR primary loop piping, where discontinuities of this size do not appear to appreciably influence or cause premature failure, installation of the new pipe section is proceeding. Provisions will be made so that the pipe section in question can be periodically monitored in service with ultrasound. Again, NPR experience has shown that any growth of the discontinuities can be ultrasonically detected and observed as propagation occurs.

Equipment has been procured and calibrated for use in measuring vibrations in the PRTR. An attempt will be made to place the accelerometers prior to PRTR startup so that the reactor can be monitored under operating conditions.

Extrusion marks occurring in the inside surface of connector piping for NPR were evaluated. Depth measurements showed the discontinuities to be within dimensional tolerance for the pipe and cross-sectioning demonstrated that the discontinuities did not form a stress-rises notch. Grain structure of the material was normal. A series of tests were made to evaluate the effects of a protective coating used during the fabrication and processing of NPR piping. There was some concern that residual amounts of the coating would be detrimental during heat treatment. The alternative was to sandblast the inside surface of all the pipes. Using several pieces of the pipe in question, the inside surfaces were deliberately contaminated and then subjected to a series of heat treatments representing at one extreme the most severe case. Evaluation by metallographic examination and microhardness traverses demonstrated that in no case was a detrimental condition encountered. On the basis of this test, the sand blasting operation was deleted with considerable cost saving.

Analog Computer Facility Operation

The major problems considered during the month were:

1. Reactor Transfer Function Studies
2. Reactor Instrumentation Study

3. NPR Plant Simulator
4. Radioactive Waste Disposal Heat Transfer
5. Pot Calciner
6. PRP Critical Facility

A number of analog demonstration problems were completed as follows:

1. Equation of a vibrating string and its solution on the analog computer. The wave equation was used to derive the specific equation for the vibrating string. The boundary conditions were specified and the whole equation translated into an analog representation. A memorandum was issued.

2. Analog simulation of coupled differential equation. In this particular problem only one initial condition was employed, which was a first derivative with time (velocity). This was accomplished by charging a capacitor to a specific potential before computing started. Change in velocity and change in direction did not change the basic results of the observed patterns. The equation, the simulation, and the results were published in an SRO memorandum. This setup was demonstrated to EDPO people.

3. Simulation of cylindrical heat transfer with internal heat source on the analog computer. Starting with the most general case of the heat or diffusion equation, the specific equations for the cylindrical coordinate system were developed. The boundary conditions and initial conditions are stated. The equations for the analog computer are given and a 20-node analog representation setup is derived. An SRO memorandum was issued. The analog representation was not put on a computer, due to the fact that all computers are in use at this time.

   An analytical solution of the aforementioned problem was obtained. One equation given was programmed for the 7090 computer. The results will be used to check on the necessity to extend the outer radius to 1000 feet or beyond. In order to do this, the root of the Besselfunction, first kind, zero order ($J_0(x)$) were needed, a routine was written and checked out which does furnish the first 40 roots to an accuracy of $\pm 1 \times 10^{-7}$. Checkout of the main program is in progress.

4. Gain versus stability of a system. A memorandum is being issued on some findings using an analog computer on a system having variable stability with gain. Amplitude limiting and dead zone features are introduced. The influence of the initial condition, in this

case influence of a step function, on the stability is observed. It was found that amplitude limiting gives rise to an oscillation.

Introduction of a dead zone requires the use of describing functions for interpreting the results since it is difficult to assess the over-all amplification factor of the dead zone when excursions occur which go beyond the dead zone.

Instrument Evaluation

Initial acceptance tests were started on 21 of 65 total Model II Scintran instruments fabricated in Seattle and ordered by Radiation Protection Operation. Except for slight signal feedthrough on a few audio circuit boards, which was corrected by transistor replacement, the initial tests were satisfactory.

Two scintillation transistorized alpha hand counters, fabricated on-site, are being thoroughly tested before delivery to the field for general use. A number of wiring errors have had to be corrected and an incorrect layout procedure caused pickup noise to be generated in the circuits for the external clothing-checking probes. A few minor troubles remain to be resolved. The particular field technicians, who will be responsible for the maintenance, will be thoroughly instructed before the instruments are delivered.

Tests continued on a logarithmic and multi-decade linear response scintillation transistorized area monitor slated for use in the 326 Building. As with the foregoing alpha hand counters, a more judicious layout and fabrication procedure would have provided a far more satisfactory instrument. Following correction of errors, recorded operation for the last two weeks has been satisfactory on both the logarithmic and linear scales.

Suggested changes were made on the prints for the standard HAPO "C-P" dose-rate portable instrument since RPO intends to purchase about 150 units to be off-site fabricated. The successful bidder will be required to provide at least one completely satisfactory prototype before fabricating the rest.

Thirty transistorized portable BF_3 tube instruments for neutron monitoring were received from an off-site fabricator. Initial testing has been satisfactory on the instrument. It is planned to employ a small commercial BF_3 tube, which will have about one-thirtieth the sensitivity of the present 10-inch standard HAPO BF_3 tubes. This will provide a far more acceptable range of operation for general field use. The instruments, however, are designed to work correctly with either detector and with

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scintillation neutron detectors if so desired.

One prototype transistorized scintillation portable alpha "poppy" instrument, which successfully completed all tests in June, was used for the preparation of as-built tracings following the completion of a layout satisfactory for easy maintenance.

Field tests have been satisfactory on the prototype transistorized modular "plug-in" gated oscillator loudspeaker circuit for use with most present HAPO portable radiation detection instruments of the count-rate type. The unit, which plugs directly into the earphone jack, eliminates the necessity of using earphones during low-level monitoring. All users, to date, have been quite enthusiastic regarding the unit and three more are being fabricated for general testing and use.

Complete evaluation testing was started on a commercial (Nuclear Measurements Corporation Model GA-2A) logarithmic response area and/or critical radiation monitor at the request of FPD. Temperature, line voltage, and some constant-source level operation tests have been conducted to date on the scintillation vacuum-tubed instrument with the results, in general, reasonably satisfactory. Future tests will include gamma energy dependence, recovery after exposure to high dose-rates and/or integrated doses, recorded running tests at various dose-rate levels, and examination of fail-safe features.

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Manager
PHYSICS AND INSTRUMENT RESEARCH
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CHEMICAL RESEARCH AND DEVELOPMENT OPERATION

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Environmental Studies

Following the recent simultaneous shutdown of all reactors and the subsequent start-up of C reactor, it was possible to measure the flow times to Pasco of the radionuclides by frequent sampling or monitoring as the radionuclide concentration decreased and later increased. The two values obtained were 15 and 20 percent longer than the flow times estimated from float studies. Further studies at the lower flow rates used in our depletion studies are needed to evaluate the effect this change of variable will have on the depletion values calculated using float study flow rates.

Reactor Effluent Water Radioisotope Studies - Silicate Addition

Addition of silicate to a tube in KE Reactor was increased from 10 ppm to 20 ppm (as silicon) on June 24, 1962, and continued for two weeks at a pH of 7. This higher concentration of silicate did not effect the expected improved reduction in the major radioisotopes released in reactor effluent water. In comparison with a control tube without added silicate, the P-32, As-76 and Np-239 were reduced by average factors of 2, 3 and 3.5, respectively, although for shorter periods reductions of 3, 5 and 8 were obtained. Comparable reductions obtained with 10 ppm silicate (as Si) were about 2, 2.5 and 2.5. Addition to the 20 ppm level also reduced the effluent water concentrations of Cu-64 by a factor of 2, Mn-56 and Cr-51 by a factor of 1.5. The Na-24 increased by a factor of 3. The reductions of P-32 and As-76 are considerably less than those obtained in laboratory tests where reductions of the order of a factor of 10 were obtained, but this difference cannot be explained at present. The pH at which these tests are conducted seems to have a marked effect and this will be studied in further tests.

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Water Treatment Facility

The water treatment plant at 1705 KE is now operational and treatment of raw water with this facility has begun. It is expected that water will be supplied to two process tubes by late August. The first studies on this facility will be aimed at optimizing the alum treatment process variables for reactor operating conditions.

Ground Water Temperature Studies

The temperature of the ground water in the region between the 100-D Area and the 100-H Area is several degrees centigrade above normal. The temperature pattern suggests that the source of this thermally hot water is from the 100-D retention basin. In one well, located one-half mile northwest of the 100-D Area, the temperature of the ground water at the water table was 46 C. This new data strongly supports the generalized ground water contour pattern which has been drawn for this region based on a limited number of observation wells.

The maximum temperature of the Yakima River reached 26 C during July. This sharp rise in river temperature, from a recorded 17 C in June, was reflected in several wells located north and east of the river in the vicinity of Horn Rapids. From a study of the ground water temperatures in this region it may be possible to delineate the approximate area of recharge of the Yakima River.

Radionuclide Migration Rates

A test of the applicability of Inoue and Kaufman's equation[1] for predicting time of trace radionuclide migration through soil was made using nitrate ion and Sr-85 breakthrough data from a model crib[2]. Calculated travel times for Sr-85 were 10 to 24 percent less than actual nuclide travel times determined by monitoring the percolating waste water in three test wells.

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1. Kaufman, W.J. "Ground Disposal of Radioactive Wastes," Conference Proceedings, p. 135, Berkeley, California, August 25-27, 1959.

2. Knoll, K.C. and J.L. Nelson. Radioisotopes and Moisture Distribution Beneath a Model Crib, HW-71573, January, 1962.

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Effluent Monitoring

An attempt is being made to decrease the amount of interference encountered in the As-76 monitor by application of a subtraction correction to the arsenic count. The correction is based upon the iodine isotopes present in the energy spectrum of 0.7 to 1.1 Mev. The success of this technique will depend upon the constancy of the ratio of the various iodine isotopes to each other during equilibrium conditions of reactor operation.

An iodine isotope monitor was assembled and is being tested at 146-FR. A continuous carbon tetrachloride extraction of iodine from the 107-F Basin effluent is accomplished in a cell with pretreatment and mechanical mixing. The carbon tetrachloride extractant stream flows into a counting cell where the iodine isotopes are measured by use of a gamma ray spectrometer. Clean separation of iodine with no interference from other isotopes is the main advantage of this system. Difficulty in maintaining constant flows through the cell and consumption of pretreatment chemicals as well as carbon tetrachloride are disadvantages of the system.

Uranium Oxidation and Fission Product Volatilization Studies

A topical report, HW-72321, "Fission Product Release from Uranium - Effect of Irradiation Level," by R.K. Hilliard and D.L. Reid, dated June 20, 1962, was published.

Efficiency of Charcoal in Reactor Confinement Halogen Traps

Iodine-131 adsorption efficiencies of installed reactor confinement charcoal and fresh charcoal were intercompared. Fresh charcoal was about 2 percent more efficient than similar charcoal which had been held in the 105-F ventilation air stream since the reactor confinement installation was completed. This difference is not considered important in the application; it might well reflect variability in charcoal efficiencies rather than indicate deteriorating I-131 removal efficiency with exposure to the air stream.

SEPARATIONS PROCESSES

Purex Process

Recently Purex process equipment was modified to permit alkaline permanganate washing of the second cycle solvent. Almost immediately, improvement in 2D-2E column performance was noted; higher pulse rates without flooding were achievable and waste losses improved. This plant performance substantiated earlier laboratory tests which showed that alkaline permanganate washing of plant 2DX effectively removed or inactivated materials in the solvent which had an adverse effect upon the uranium transfer rate (cf. HW-73514 C and HW-73905 C).

Current studies are directed toward the process water as the source of the detrimental material. That organic and inorganic impurities are present in Purex process water is evident from the residue that collects on the glass wool filters located downstream from the ion exchange beds. Tests on this filter residue to date have indicated that it contains:

1. Solid pieces of predominantly inorganic material containing Al^{+3}, and $SO_4^{=}$. This material, when dissolved in acid, gives a positive test for large organic anions, e.g., detergents, and significantly lowers uranium transfer rate.

2. A much larger amount of brown plastic-like material which is fairly soluble in ether, methanol or hot water and has a limited solubility in chloroform. These extracts also show a positive "detergent" test and affect transfer rate adversely.

3. Amines, as evidenced by their typical odor when the filters are extracted with NaOH solution.

On the assumption that 10^9g H_2O pass through a given filter, the amount of organic recovered represents an initial concentration of a few parts per billion. There is no way of estimating how much was not retained by the filter.

A sample of used "Anthrafilt" from the water treatment system together with its associated alum floc was also obtained and briefly examined. A water extract of this material also produced a retardation of transfer rate and gave a positive "detergent" test.

Further tests are being made to more positively identify functional groups present in these samples. Based on the present evidence, one might tentatively conclude:

1. The alum coagulation is very necessary and that perhaps it should be bolstered somewhat.

2. The ion exchange beds are not doing an effective job, and/or

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3. the ion exchange beds are contributing new impurities, and/or

4. a source of organic and inorganic impurities exists downstream from the beds.

Solid State Electromigration

In preliminary experiments using Fe-59 and Zn-65 traced aluminum, no migration of the tracer elements was observed with current densities of 8300 amp/in^2 (potential gradient of about 0.03 v/inch). Since work at the Albany Station of the Bureau of Mines has shown the existence of a voltage threshold in the migration process, equipment and methods are being developed to permit much higher voltages. Preliminary experiments in which the specimen is cooled by immersion in a molten alkali chloride have permitted current densities of 30,000 amp/in^2 and gradients of 0.25 v/in. This technique should also provide better protection against corrosion and permit a study of electromigration over a much broader range of process parameters.

Electrochemistry of Plutonium in Molten Chloride Salt Solutions

Equilibrium of liquid plutonium metal with 55 m/o $KCl-BaCl_2$ + 55 m/o KCl, which serves as the solvent in the present studies of the electrowinning of plutonium, indicated that the metal was soluble to a slight extent. The addition of $PuCl_3$ (~ 0.3 molal) to the melt caused the solubility of the metal to increase by a factor of about 1.6. These results may explain the anomalous waves observed in the E-I curves of the system $Pu(M)/BaCl_2$-KCl.

Electrowinning runs using a molten plutonium metal pool as the cathode with constant potentials of 2.05 v and of 10 v (vs. Ag/AgCl reference) were unsuccessful in obtaining significant amounts of plutonium. This was likewise the case when tungsten rod cathodes were used in order to obtain high current densities (20 amperes/cm^2).

The melt in the last run, except for a thin layer along the bottom of the cell, appeared to be devoid of $PuCl_3$. Apparently the bottom layer, which was deep blue-green in color, was insoluble in molten $BaCl_2$-KCl, and this precipitate, being unable to contact the cathode, could not be reduced to the metal.

Denitration of Purex 1WW with Formaldehyde

In an attempt to define and eliminate the operating difficulties that are being experienced with the plant denitration unit, an approximately 1/10-scale, glass model of the unit has been built. The glass unit consists of a 9-foot high, 6-inch I.D. reactor pot surmounted by a 9.5-foot high, 4-inch I.D. tower. A 4-foot section of the tower, beginning 2-ft. 8-inches above the reactor pot, is packed with 1/2-inch porcelain Intalox® saddles to provide intimate contact between the gases from the pot and the 1WW feed which enters six inches above the packing. In each run the formaldehyde entered the reactor pot in the vapor space above the liquid.

In general, the unit operated smoothly and the reaction was easily controlled. Waste (1WW) flow rates equivalent to Purex capacity requirements and at 1WW to HAF flow ratios of 0.1 were explored with formaldehyde to 1WW flow ratios ranging from 0.01 to 0.11. At all rates investigated the pressure drop across the tower was less than one inch of water and the foam layer on the liquid in the reactor pot was about 1/2-inch thick. The nominal free acid and total nitrate concentrations in the simulated 1WW feed were 5.1 M and 7.35 M, respectively. Filtration of the 1WW feed did not appear to decrease foaming or affect the operation of the unit.

WASTE TREATMENT

Waste Characterization

Sludge samples were obtained in the 108-SX tank by rotary core drilling and in the 101-C tank by push-tube, in a continuing program of waste characterization. In the 108-SX tank, the 14-inch thick sludge layer (remaining after reducing a 30-inch layer by water dissolution) consisted of relatively hard, but thin, top and bottom layers with a softer center layer. Apparently only the top layer (2 - 3 inches) was sampled, with the harder material blocking entry of the softer material into the tube. A new attempt will be made to obtain a complete sample.

The coating waste sample from the 101-C tank consisted of about one liter of a highly viscous dark brown "mud" obtained from the tank bottom. Chemical analyses of both samples are in progress.

Product Forms

Evaluation of synthetic alumino-silicate zeolite materials continued with a tumbling attrition test of Linde 4A and 4AXH, 4 x 8 mesh beads.

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No significant difference was observed between 4AXH (0.8 percent fines) and 4A (0.7 percent fines) despite previous information that 4AXH would be more attrition resistant. In general, bead form materials are three- to five-fold more resistant to attrition than pelletized materials.

Glass-like solids are produced when the Linde Company high-capacity synthetic Zeolite 13X is fused with a mixture of SiO_2, B_2O_3 and LiF. A mixture containing 50 percent zeolite - 12.5 percent SiO_2 - 12.5 percent B_2O_3 - 25 percent LiF by weight fused at a temperature near 800 C. After solidification, the solid volume was about one-third that of the zeolite only in pellet form. Similar glasses prepared from zeolite loaded with cesium traced with Cs-137 are being prepared for leaching studies.

Iodine-131 in Airborne Effluents

A charcoal bed through which had passed Redox stack gas was examined for I-131 throughout the depth of bed. Two separate I-131 components were identified. One component, adsorbed with an efficiency of 92 percent on an 1-1/2 inch bed, made up 98 percent of the entering I-131. A second component represented 2 percent of the entering I-131 and was only 7 percent adsorbed. This result supports and partially explains the early conclusion that I-131 appearing in the stack exhaust is not as efficiently adsorbed on charcoal as is laboratory-generated I-131.

Cesium Removal from Formaldehyde-Treated Waste

A column of Clinoptilolite has been recycled eight times as per the process steps reported earlier for recovery of cesium from Purex FTW. The volume to cesium breakthrough declined about 2 percent per cycle for a total of 16 percent decrease over the period of the experiment. This appears to be due to a slow breakdown of the mineral in the 0.5 M acid since some aluminum is released to the effluent and some silica to the alkaline eluate. During the course of the experiment the clinoptilolite was in contact with the acid for more than 60 days. The cesium capacity loss thus amounts to less than 0.3 percent per day. Occasional bed make-up is expected to take care of this loss.

Cesium losses in 0.5 M HNO_3 used in sodium removal operations were determined for various degrees of cesium breakthrough. At 6, 18 and 53 percent cesium breakthrough levels, respectively, the cesium losses were 0.23, 0.56, and 2.5 percent for eight column volumes of 0.5 M HNO_3.

Behavior of Fission Products During Cesium Removal from Formaldehyde-Treated Wastes

More quantitative data are now available on the behavior of major fission products during the loading of cesium on clinoptilolite from a simulated FTW solution. (CR & D May Monthly Report, HW-73905 C). Decontamination factors for strontium, cerium, ruthenium, and niobium were 2.1×10^3, 9×10^4, 1.4×10^5, and 7, respectively.

Gassing of Ammonium Carbonate Solutions

Several experiments were performed to measure the effect of temperature and ammonia concentration on the gassing of ammonium carbonate solutions used in eluting cesium from a column of clinoptilolite. The results showed that gas evolution can be minimized by maintaining the temperature at or below 55 C and by adding ammonia to the ammonium carbonate solution.

Strontium Packaging

The first pilot plant experiment on the packaging of strontium on an inorganic zeolite was completed. A feed rate of 0.2 column volumes/min (1.5 gpm/ft^3) was used for the experiment with 1/16-inch pellets of Linde 4A in a 2-inch diameter by 36-inch long column. Fifty percent breakthrough occurred after 10 column volumes had been processed. (Laboratory experiments based on bed material of a smaller particle size reach this breakthrough after 31 column volumes.) About 8 to 10 column volumes of rinse water were required to reduce the residual nitrate in the rinse effluent by a factor of 1000.

Cesium Packaging

Cursory tap water leaching tests were continued on fused Cs-clinoptilolite obtained from volatilization studies. Leaching rates of the glass at room temperature approached a constant value of approximately 9×10^{-9} g/cm^2/day after 72 days. The continuous reduction from the initial apparent leach rate of 2.3×10^{-7} g/cm^2/day depicts the decreasing influence of surface effects on leaching rates. These dissolution rates are approximately equal to those reported for the most insoluble glass and are better than most by an order of magnitude.

TRANSURANIC ELEMENT AND FISSION PRODUCT RECOVERY

Cesium Solvent Extraction

The minimum in the distribution ratios of cesium between dipicrylamine in nitrobenzene and nitric acid solutions has been shown to be due to extraction of cesium by nitrobenzene. The distribution coefficient for the nitrobenzene system has a positive dependence on nitric acid concentration whereas that for the dipicrylamine-nitrobenzene system has an inverse dependence. Thus, at an aqueous nitric acid concentration of 1.25 - 1.5 molar, extraction by dipicrylamine is essentially suppressed and extraction by nitrobenzene becomes the predominant process.

Scouting experiments were performed to investigate cesium extraction into nitrobenzene from nitrate salts. The results from $Ca(NO_3)_2$, $Al(NO_3)_3$ and $LiNO_3$ indicated lower distribution ratios than from nitric acid. Experiments are in progress to determine the extractability of sodium, rubidium and cesium by dipicrylamine in nitrobenzene from lithium hydroxide solutions.

Assistance to Purex Plant Head-End Fission Product Recovery

Tracer level investigation of the complexing of the rare earths by radiolytic and peroxide decomposition products of tartaric acid during the precipitation of strontium and lead sulfates was continued. The addition of either dihydroxymaleic acid or tartronic acid after pH adjustment gave an increased solubilization of cerium but no effect on strontium. The addition of 10 grams per liter of either reagent increased the cerium content of the supernate from 3 to 20 percent. The use of 75 grams of tartaric acid plus 100 ml of 30 percent hydrogen peroxide per liter of feed caused 70 percent of the cerium to go into solution. This indicates that much of the tartrate is probably converted to the "active" complexant in the tartrate-peroxide system, and the addition of additional amounts of the specific complexant is not economically feasible. Some additional work will be directed toward the use of the "pure" complexants, but the major effort will be centered on the tartrate-peroxide system.

Tracer level experiments were performed to determine the effect of the addition of various amounts of tartrate on the amount of precipitate obtained during the sulfate precipitation from 1WW or FTW at pH 1.5 and 2.0. At pH 1.5, the amount of precipitate obtained in the absence of tartrate or with small additions of tartrate was small in every case, but the addition of 0.25 to 0.50 moles of tartrate per liter of feed (1WW containing 0.5 $\underline{M}$ Fe or FTW containing 0.67 $\underline{M}$ Fe) resulted in gross quantities of precipitate. FTW containing 0.5 $\underline{M}$ Fe did not yield excessive precipitates under any of these conditions. At pH 2, the volume of precipitate varied from 25 volume percent in the absence of tartrate, through little or no precipitate at 0.1 to 0.2 moles of tartrate per liter, to a more voluminous precipitate at 0.3 to 0.8 moles of tartrate per liter. The volume of precipitate then decreased to zero when over 1 mole of tartrate was used. At present, Purex uses about 0.8 moles of tartrate per liter of feed. A slight reduction in this amount would be expected to yield a gross precipitate, while a reduction by a factor of about 4 should yield only the normal sulfate precipitate. This will be checked in B Cell in the near future. The present interpretation of the precipitate is that iron sulfate or hydroxide precipitates in the absence of tartrate, and tartrate complexing of the iron prevents this precipitation. However, certain iron to tartrate ratios result in an iron tartrate precipitate which redissolves in an excess of tartrate.

Gamma Dose Rate from Pm-147 and Pm-148

Shielding requirements for a promethium source were made and checked experimentally using a 3.8 ml nitrate solution containing 11.8 curies of the Pm-147 purified late last year in A Cell. Promethium was compared with other fission products as heat sources. The basis for comparison was a generator of about 250 watts (thermal) with an allowable radiation level of 200 mr/hr at one foot from the source. The results are summarized in the table below (all values are approximate):

A Comparison of Fission Product Heat Sources

Nuclide	Kilocuries	Grams	Pb Shielding Thickness (inches)
Sr-Y 90	40	480	4.5
Pm-147 (1Y)	660	660	4
Pm-147 (2Y)	660	660	1.25
Pm-147 (3Y)	660	660	0.2
Cs-Ba 137	52	1500	4.5
Ce-Pr 144	31	80	8

On the basis of shielding requirements, strontium and cesium are equivalent to each other and to one year old promethium. When promethium is aged 3 years or more, Pm-148 does not control and only a very small amount of shielding is required. It is interesting that for Pm-147 the best shield is a mixture of lead and tungsten because an appreciable amount of radiation passes through lead behind the K absorption edge while tungsten still has a high absorption coefficient in this region.

Acid Destruction in 1WW

Work has been initiated to determine the rate of hydrogen ion destruction by the addition of organic acids or salts to nitric acid in a radiation field. Preliminary results have been obtained for the destruction of nitric acid in solutions of sucrose, acetic, citric and oxalic acids. These results indicate that the nitric acid in a synthetic 1WW solution can be destroyed at ambient temperatures over a period of a few weeks by the addition of sucrose without radiation. An increase in temperature is beneficial. The benefits of sucrose as opposed to formaldehyde are being examined.

Recovery of Np and Pu from Purex 1WW and FTW Solutions

Batch contact extractions of Purex Plant 1WW solution with D2EHPA-Soltrol solvent were made to test flowsheets for the recovery of plutonium and neptunium from the waste. About eighty percent of the plutonium present was extracted when the 1WW, as received, was contacted with solvent under conditions wherein plutonium extraction was greater than 95 percent from simulated 1WW spiked with Pu(IV). This suggests the possibility that not all of the plutonium in plant 1WW is present as Pu(IV). To reduce neptunium present to the extractable Np(IV) state the plant 1WW was made 0.2 M in hydrazine and allowed to stand 30 minutes at 25 C before contact with solvent. Eighty-three percent of the neptunium present was extracted as compared to 97 percent when similar procedures were applied to synthetic 1WW spiked with Np(V). Apparently more severe reducing conditions will be required to assure reduction of neptunium in plant 1WW. Other experiments with the plant 1WW showed that Zr-Nb decontamination in the extraction step is increased from 8 to about 70 if the waste is made 0.01 M in EDTA prior to extraction; plutonium extraction is not impaired.

Experiments to develop a flowsheet for co-extraction of neptunium and plutonium from Purex Plant FTW (1965 composition) were promising. In these, synthetic FTW spiked with Np(V) and Pu(IV) was contacted with 0.1 volume of D2EHPA-Soltrol solvent. While still in contact with the solvent, the aqueous phase was treated with hydrazine and ferrous sulfamate and the two phases were mixed again. When the solvent contained 0.1 M D2EHPA and the aqueous phase was made 0.05 M in hydrazine and ferrous sulfamate between the two contactings, the final organic contained 98 percent of the plutonium and 97 percent of the neptunium initially present in the aqueous.

Pilot Plant Extractions with D2EHPA

Initial pilot plant pulse column runs demonstrating cerium and strontium extraction concepts for the proposed Hanford Waste Management Program were successful in extracting greater than 99 percent of the cerium in a citric acid-complexed feed. The extraction was carried out in a 9-foot high extraction column using 0.2 M D2EHPA (di-2-ethylhexyl phosphoric acid) in Soltrol-170®. The fear that slow cerium extraction kinetics might seriously limit the degree of cerium extraction appears to be unfounded.

Subsequent attempts to strip the cerium out of the organic phase were only partly successful, apparently because of the presence of some Ce(IV) in the Ce-144 tracer used to facilitate cerium analyses. Only 70 percent of the Ce-144 was removed on the first pass with nitric acid 1CX, and several succeeding passes with nitric-oxalic acid mixtures succeeded in removing only an additional 20 percent.

Solvent Extraction of Fission Products from Purex FTW

Miniature-mixer-settler runs tested partitioning of strontium from rare earths and subsequent stripping of rare earths from D2EHPA-TBP-Soltrol extractant. Feed for these runs was solvent produced in previous runs testing the extraction column flowsheet. For the partitioning, seven mini-stages were used. With 0.03 M HNO_3 as the partitioning agent and with an organic to aqueous flow ratio of four, strontium loss to the solvent was 0.3 to 1.3 percent and decontamination factors from cerium, europium and iron were 500, 7600 and 40, respectively. Similarly, with 0.2 M citric acid as partitioning agent, strontium loss was 0.5 percent and cerium and iron decontamination factors were 130 and 20, respectively.

Rare earths remaining in the organic after partitioning were readily stripped with 1-3 M HNO_3 as shown by batch contacts and a mini-mixer-settler run. In the latter, seven stages were used. The stripping agent was 2.0 M HNO_3 at an organic to aqueous flow ratio of about eight. Cerium loss to the organic was less than one percent. No reason attributable to the organic has been found for the poor cerium removal in the pilot plant pulse column runs.

Batch contact studies of the effect of temperature on extraction of various constituents from citric acid complexed FTW feed solutions by D2EHPA-TBP-Soltrol extractants were made. Extraction of cerium and iron

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was higher at 60 C than at 25 C. Extraction of strontium and chromium decreased with increasing temperature - probably due to increased loading of the solvent with iron. Further temperature effect studies will include extraction rate determinations to evaluate desirability of operating the extraction column at elevated temperature.

The Centrifugation of Cesium 12-Tungstophosphate

The precipitation and centrifugation of cesium 12-tungstophosphate from dilute and four-fold concentrated, simulated FTW were studied in two pilot plant runs. Essentially quantitative precipitation of the cesium occurred with a 50 percent excess of 12-tungstophosphoric acid over that needed to form the dicesium salt and with one hour agitation at 25 C.

Centrifuge efficiencies in the continuous, 26-inch, baffled, solid bowl centrifuge were nearly independent of holdup time in the range 3.2 to 10.6 minutes. At 340 G's the centrifuge efficiencies were 73 and 81 percent for the dilute and concentrated FTW, respectively, for these feeds. Although the cake was readily slurried from the centrifuge, a film of precipitate tended to adhere to the centrifuge internals.

Recovery of Cesium from Purex Waste by Ion Exchange

The ability of sulfonated phenol type resin Duolite C-3 to remove cesium from a simulated Purex 103A tank supernate containing 5×10^{-4} M cesium was confirmed. Five and 50 percent breakthroughs of cesium occurred at 8.3 and 21.7 bed volumes, respectively, in a 4-inch diameter resin bed 119 inches long. The run was made at room temperature with an average flow rate of 90.8 gal/hr/sq.ft. The performance compares favorably with a previously reported run made with a 33-inch bed (HW-74153 C).

A 2.5 M ammonium sulfate solution effectively removed cesium at room temperature from a 33-inch bed which had been loaded to 100 percent breakthrough. More than 90 percent of the cesium was removed in six bed volumes at an average flow rate of 49.7 gal/hr/sq.ft.

Solvent Extraction Recovery of Cesium

Miniature-pulse-column studies tested stripping of cesium from 0.01 M DPA - 50 percent nitrobenzene - 50 percent Tetralin solution. This organic was produced in earlier runs testing extraction of

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cesium from simulated Purex stored waste supernatant liquid. The column was operated at an ℓ/v of one with 0.5 M HNO_3 as the stripping agent. Cesium recovery was greater than 99.9 percent. Hydraulic performance of the column appeared satisfactory although the murky-brown color of the organic when in contact with acid interfered with observation of column performance.

Radiolysis of Strontium Citrate and Tartrate Solutions

In continued studies of radiolysis of strontium-containing citrate and tartrate solutions, total exposures up to 2.4×10^9 R have been accumulated. Solutions simulating Hot Semiworks strontium product solution were subjected to CO-60 gamma radiation. Strontium citrate-citric acid solutions initially 6 M in nitric acid were stable toward strontium precipitation during exposure to 8×10^8 R but over 99 percent of the strontium was precipitated at 1.6×10^9 R. A similar solution (6 M HNO_3) containing tartrate instead of citrate was stable toward strontium precipitation at doses up to 2.4×10^9 R. In the absence of nitric acid, tartrate solutions convert almost completely to a jelly-like mass at 2.4×10^9 R.

EQUIPMENT AND MATERIALS

Induction Furnace Design

A stainless steel induction heating coil was demonstrated for application to the heating of a hollow, cylindrical, stainless steel charge. Matching was perfect within ± 5 percent. Important dimensions and electrical characteristics were:

Coil: 1/2-inch, schedule 80, 304 stainless steel pipe, 12-inch inside diameter, 29 inches long, 12 turns.

Charge: 10-inch, schedule 20, 304 stainless steel pipe.

Frequency: 10 kilocycles

Shielding Window Design

HW-73998, "Survey of Effects of Radiation on Shielding Windows," was issued. This document reviewed the dielectric discharge phenomena and recommends that windows be operated at mildly elevated temperatures (100 F) and be designed so that thick glass panes are placed in the low dose rate zone of the window. Thin panes are less subject to dielectric charging and can be used on the radioactive side of the window. Further information is being obtained.

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Remote Welding

HW-73933, "Remote Welding Closure for Waste and Fission Product Containers - Interim Report," was issued.

Stress Cracking in Mild Steel

Eight large (3 ft. x 3 ft. x 3/8 in.) weldments were prepared from 1020 mild steel plate. Exposure of part of these to 50 percent sodium nitrate and part to simulated, neutralized Purex FTW was started. After two months of exposure, the FTW will be butted to about 6 M nitrate. The primary variable in the fabrication of the weldments was in the weld metal used. These included 6010, 6016 and Inconel 182. In one set of weldments, a 95 Cd-5 Ag rod was welded to the plates parallel to the weld seams to provide cathodic protection.

Corrosion Samples from Purex 101-A Tank

Ultrasonic cleaning and identification of all samples from a set of corrosion samples recently removed from the Purex 101-A waste storage tank has been completed. Preliminary calculations based on total weight loss indicate an average corrosion rate of about 0.1 mil/yr. Pit depth measurements and microscopic examination should be completed within the next month.

Failed Redox Demineralized Water Heat Exchanger

Huey tests coupled with visual and microscopic examination indicate that the corrosion failure of a heat exchanger in a Redox demineralized water unit was due to the presence of sulfuric acid in the water inlet stream. Sulfuric acid could be in this stream after regeneration if incomplete flushing was given to the ion exchange bed feeding the heat exchanger.

Non-Metallic Materials

A new adhesive designed for plastic cementing was tested for use in bonding polyvinyl fluoride film to polymethyl methacrylate panels. This adhesive appears very promising for sealing the transparent, highly resistant film to the inside of hood panels.

PROCESS CONTROL DEVELOPMENT

Purex Canyon Electrical Connectors

A special instrument-electrical connector jumper has been devised to overcome the problem of limited canyon connectors. The jumper, containing two coaxial connectors and two power contacts, is designed for installation in Purex E cell. The initial application will provide a coaxial connector for a neutron monitor in the E-6 tank. A prototype coaxial cable connector has been completed and tested mechanically and electrically. No difficulty has been incurred in simulated remote installation. Electrically, the connector has a satisfactorily high insulation resistance; an ion chamber current of 1×10^{-12} amperes was passed through the connector and read out on a standard instrument.

C-Column Studies

An extensive series of calibration tests were made on the relocated experimental C-Column. Calibration data were obtained on column temperature, pulse frequency, and aqueous stream pH instrumentation. The shorter pulse leg in the new installation results in less variation of pulse amplitude with changes in pulse frequency, a situation which should produce better experimental data for use in refining the mathematical model. Several runs were made to develop additional information on fluctuations in column variables under nominal steady state conditions. Also, the variations in uranium concentration at a particular point in the column were studied as a function of the pulse frequency.

Using tape punched by the new scanner-punch system, the C-Column Data Reduction Code was checked out with a new Paper Tape Input conversion code. A minor bug in the latter code has been eliminated. A procedure is being set up to process punched tapes on a "production" basis so that experimental data taken one day will be returned in final processed form the following morning.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Electrolytic Decontamination of Molten Halides - Purification of contaminated alkali chloride melts will reduce waste volume in the Salt Cycle Process. Two electrolytic purification techniques were investigated. Impurities added were europium and iron. Residual amounts of plutonium chloride and uranyl chloride were also removed since uranyl reduction precedes that of iron and reduction of plutonium ions to metal precedes that of europium.

1. A molybdenum cathode and an open graphite anode were used in the first, and least successful, method, with no attempt to exclude anodic chlorine from the cell. Electrolysis was performed at -2.0 volts relative to an Ag/AgCl reference electrode. Current efficiency for the removal of all impurity ions was less than 5 percent. Percentages of the ions removed were: uranium, 53; iron, 36; plutonium, 23; and europium only about 0.01. Furthermore, the deposit would not adhere to the molybdenum cathode.

2. The second technique achieved high removal of impurities even with a chlorine atmosphere in the cell. This method made use of a MgO cell-liner, a molten lead cathode, and an open graphite anode at a cell potential of 4.5 - 5.0 volts. Percentages of the ions removed were: uranium, 99.99; iron, 97.5; plutonium, 98; and europium, 60.

Electrolytic Preparation of Large Single UO_2 Crystals - Ceramic Fuels Development Operation has been attempting to obtain large single crystals of uranium dioxide containing enriched uranium without success. Approximately 30 g of enriched uranium dioxide crystals in the 1-gram size range were produced in two molten salt electrolysis runs. Crystal size was limited in each case due to premature shutdowns caused by instrument trouble. Previous runs with normal uranium have produced crystals in the 3 to 4 gram size by the same techniques, which utilizes the $PbCl_2$-2.5 KCl system as solvent. Aluminum impurity in the electrodeposited UO_2 was reduced tenfold from its concentration in the starting material.

Pressurized Wall Containers for Molten Salts - Impermeable graphite would be a suitable container for molten chloride salts because of its corrosion resistance. To prevent salt permeation of porous graphite, application of external gas pressure to induction-heated, 3-liter graphite crucibles with 1/2-inch thick walls was tested with LiCl-KCl molten Salt at 500 C. During a 16-hour period, 25 percent of the salt leached through the wall of an unpressurized graphite crucible. An identical crucible was operated for 71 hours at an external pressure of 2 psig with salt penetration of less than 0.10 inch into the crucible wall. This pressure was about three times that exerted by the salt at the bottom of the crucible and appreciable gas bubbled through the vessel wall. The gas consumption was 1.7 liters per minute but could be reduced by decreasing the external pressure, although chlorine gas might be used in this way to dissolve materials in the salt bath.

RADIOACTIVE RESIDUE FIXATION

Synthetic Zeolites

The determination of equilibrium constants (K_c) for the synthetic zeolites Linde 4AXW, Linde 13X, Linde AW-300, Linde AW-400, Linde AW-500, Zeolon and clinoptilolite was continued. Systems for which the equilibrium relationships were determined with the above zeolites included strontium-calcium, strontium-barium, strontium-magnesium, lithium-cesium and ammonium-cesium; the K_c for the hydrogen-cesium system was determined for clinoptilolite only.

Several additional checks were made on the accuracy of the K_c curves of the above zeolites using column data. Column 50 percent breakthrough capacities of 0.037 and 0.011 meq Cs/g for Linde AW-500 and Linde 4AXW, respectively, were predicted from cesium-ammonium curves for an influent containing 0.001 $\underline{N}$ Cs^+ + Cs-134 + 1.00 $\underline{N}$ NH_4^+. Actual capacities obtained were 0.039 meq Cs/g of Linde AW-500 and 0.011 meq Cs/g of Linde 4AXW. Similarly favorable comparisons resulted in other experiments.

Condensate Treatment

During MPP Run 22 the steam-stripped Purex Tank Farm condensate feed acidified to pH 4.2 was passed through an activated carbon bed prior to being passed through a bed of Amberlite IR-120 in the sodium form. Both beds removed ruthenium so that the final effluent concentration was less than the isotope's MPC_w (greater than 95 percent removal). MPP Run 29, which was a repeat of this test but without the activated carbon bed, resulted in much less ruthenium removal (ca. 22 percent).

Clinoptilolite Beneficiation Studies

Preliminary crushing and grinding tests on selected Hector, California, clinoptilolite were completed by Dr. J.A. Pask of the University of California. These tests were to determine if means exist for improving the yield from mine-run ore and minimizing the creation of fines in clinoptilolite mineral columns. Rolling mill and rod mill tests disclosed that a maximum of about 50 percent could be obtained in the 10 to 20 mesh size, about 25 percent in the 20 to 50 mesh size, and that about 25 percent fines (less than 50 mesh size) would be produced by the best crushing methods. Some degradation of clinoptilolite particles was noted in the wet screening tests, suggesting the presence of water soluble minerals in the ore. Tests are under way at Hanford to determine the physical and chemical properties of the crushed and screened size fractions relative to earlier-procured material.

Clinoptilolite, powdered to < 100 mesh, was pelletized with silicic acid. These pellets were then crushed and screened to 0.4 - 0.7 mm size. Performance characteristics of columns of this material and columns of crushed clinoptilolite of the same size were compared for cesium loadings. Flow rates were adjusted so that the residence time was the same for each column, and the results were compared on the basis of column volumes of actual clinoptilolite present. There was very little difference in performance, but the crushed pellets showed slightly better kinetics.

In-Cell Calciner Feed

Approximately 200 gallons of 1WW were obtained and set aside to allow the iodine-131 to decay prior to calcination. A small portion of the 1WW was transferred to B Cell for inspection and sampling. Although the 1WW was centrifuged at Purex, it was observed to contain a large amount of solids. A 1500 ml sample in a 2-liter graduate was found to contain 500 ml of solids after settling for 30 minutes, and 200 ml of solids after settling overnight. The solids were readily suspended in the supernate by air sparging, and the slurry passed easily through a screen with openings about the same size as the opening in the calciner spray nozzle. Therefore, operation with the slurry might not cause excessive problems. However, for the initial calcination experiments, a clear solution is much more desirable than a slurry.

Characterization of both the supernate and the solids has been initialed but conclusive results are not yet available.

In-Cell Analysis of Off-Gas from Waste Calciner

The in-cell gamma analyzer has been modified to allow counting the activity collected on the off-gas filters from the waste calciner. Standard samples prepared by absorbing known amounts of cesium-137 on the filters show the minimum cesium activity detectable to be about 10^7 γ/min. Gamma energy analysis can be made with about 5×10^7 γ/min. These results show that any samples which are too "hot" to handle without shielding (out of the cell) can be analyzed in-cell.

A-Cell Installation

The calciner equipment has been moved from the mock-up location into A-Cell. The transfer went smoothly and the pipefitting will be essentially completed by the end of July. There were no problems in placement of equipment for remote manipulator operation.

The electrical hookup is not yet complete but should be completed during the first week of August. The in-cell and exterior bus work, installation of transformers, switchgear, etc., for 4000 ampere power to the calciner is complete. Remaining work on the calciner power is limited to control instrumentation, painting, installation of cooling blowers and the external bus shields. The water-cooled coaxial cables for conducting radio frequency power to the pot calciner have been fabricated and installed. The Tocco unit has been placed and modified for this installation. The electrical and cooling water circuits in-cell are not as yet complete.

Tests of the electrostatic bubble scrubber were made with magnesium oxide smoke. The de-entrainment factors obtained for the sub-micron smoke were of the order of those obtained previously with calciner off-gas. DF's of up to 1000 were obtained. The magnesium pickup of the downstream filter varied almost directly as the flow rate and as a negative exponential function of the voltage. The results of the scrubber development and experimental data were summarized in a report, "The Electrostatic Bubble Scrubber," (HW-74210) by R.T. Allemann and U.L. Upson. The mechanically "beefed-up" unit has been installed in A-Cell. The high voltage leads into A Cell and line power to the HV supply location are yet to be installed.

Radiation Resistant Elastomer

A larger series of tests than previously reported were made on the 65 durometer (B-3 batch) ethylene propylene rubber. Seven unirradiated samples were compared with four samples irradiated to 2×10^8 R. The irradiated samples lost 3 percent hardness, 22 percent elongation, and 47 percent tensile strength. Samples of Estone rubber, adiprene, nitrile, styrene-butyl, and urethane rubbers were all destroyed by 24-hour immersion in 6 M HNO_3 at 70 C. However, ethylene propylene rubber was still strong and flexible under the same treatment. The rubber may have several hot cell applications.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

Initial runs were completed with the Fortran program for calculating time for three-dimensional flow between a crib and the river. Included in the run was a solution for flow between the NPR crib and the river, which was treated previously in two dimensions through use of electrical conductance paper analogs.

The first arrival times at the river for flow along the shortest streamlines in the two methods was in close agreement (less than 1 percent difference). However, as expected for longer streamlines, the earlier two-dimensional analog gave conservative times in comparison to those found with the three-dimensional results. For example, the time for the longest streamline is over 50 times that of the analog.

Several samples of volcanic ash were obtained from the John Day Formation in central Oregon and analyzed by the X-ray diffraction technique to determine if the mineral clinoptilolite was present. The results show that samples from one locality, located six miles west of Kimberly, Oregon, on Highway 19, had a relatively high percentage of this mineral present. There was no evidence that this mineral was present in the other samples. (A laboratory determination of the cesium distribution coefficient, K_d, showed the ash to have about one-half the exchange capacity of the clinoptilolite material obtained from California and Nevada.) In the large areal extent of this particular clinoptilolite-bearing ash deposit in Oregon it is possible that in some regions the mineral content is as high as that found in the material from California and Nevada.

Drilling began on Project CAH-963. About 10 percent of the estimated footage was completed. The first well at N 10,000 - E 12,000 lies in the area toward which tritium from Separations Areas wastes is moving through ground water. The second well, one mile south of the 300 Area, will confirm the southern limit of movement of uranium-contaminated ground water from the 300 Area.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Fallout Studies

The nuclides Zr-95 - Nb-95 and Ru-106 have been present in surface air all through this year at steady levels of about 0.1 d/m per cubic foot. Since the Zr-95 - Nb-95 concentration has remained nearly constant during the last six month period, it must be fed to the surface air at a rate about equal to its decay. Air samples were taken on July 24, 1962, at altitudes of 3000, 6000, 9000 and 12,000 feet, which showed an interesting increase in Zr-95 - Nb-95 concentration at the 12,000 foot level about 100 times the surface concentration. I-131 has been present in near surface air at a concentration of about 10^{-3} d/m per cubic foot since June 13, 1962. The I-131 is collected in about equal amounts on a membrane filter to remove particulates and a back-up charcoal filter to remove the gaseous material.

Radiation Chemistry

Previous determinations of "protection indices" of compounds in aqueous solution have all been made on solutions containing less than 0.1 percent solute by weight. As more concentrated solutions are examined, one can expect to deal with systems of more complicated structure and hence more complex radiation-induced kinetics, until finally the "direct effect" of irradiation on the solute is predominant. This "in-between region" of concentrated solutions has perplexed many investigators, the general conclusion being that the protection offered by chemicals in irradiated solutions does not increase proportionately as their concentration is increased. Several phenomena postulated to explain this "saturation" effect have been tested. In order to obtain a new perspective for this problem, experiments were begun to extend the protection index determinations. In terms of protection indices, the "saturation" effect should appear as a decreasing protection index as the concentration of protector is increased. Urea was the first compound examined at high concentrations since it has a very low protection index even at low concentrations. Complications were observed immediately since the usual linear plot of log [Dye] versus absorbed dose was curved more and more as the concentration of urea was increased. Utilizing the initial straight-line portion of these curves, the following protection indices of urea were evaluated:

[Urea]	P.I. 25°	% Urea by Weight
2.00×10^{-3} $\underline{M}$	3.0×10^{-4}	<0.1
5.00×10^{-3} $\underline{M}$	3.5×10^{-4}	<0.1
2.00×10^{-2} $\underline{M}$	8.8×10^{-5}	0.1
5.00×10^{-2} $\underline{M}$	1.8×10^{-4}	0.3
2.00×10^{-1} $\underline{M}$	3.4×10^{-5}	1.1
5.00×10^{-1} $\underline{M}$	2.2×10^{-5}	2.8
2.00 $\underline{M}$	3.0×10^{-5}	11
4.50 $\underline{M}$	2.8×10^{-5}	25

The protective ability of urea as measured by the protection indices is only decreased by a factor of 10 even when the concentration is increased by a factor of 2000. The smallness of this decrease is surprising since, at the higher concentrations, a substantial amount of the radiation must be adsorbed directly by the urea (urea and water each have approximately the same electron density, 3.2×10^{23} electrons/gram and 3.3×10^{23} electron/gram, respectively, so that the fraction of high-energy radiation each absorb is equal to their weight percent in the solution).

Thin layer chromatography techniques are being used to examine the purity of separated dye fractions for the studies of chemical protection from radiation. Glass plates coated with an absorbing medium such as kieselguhr form the stationary phase while development is accomplished by ascending chromatography using various solvents. Using this technique it was found that Fraction IV (Food and Drug Administration nomenclature) of erioglaucine could be resolved into at least two components for which a practical separation procedure is being sought.

In 0.05 $\underline{M}$ sodium hydroxide the blue form of the dye, erioglaucine, is converted to one of its colorless forms through the addition of a hydroxyl ion. In addition to the colorless form, a purple derivative is observed to form. To the extent that this purple species is formed, the protolysis and hydrolysis constants measured for the interconversion of the various species of erioglaucine will be in error. The rate of formation of the purple species increases as the pH increases. It is readily formed in the absence of oxygen indicating that its formation is not a base catalyzed oxidation of erioglaucine. In basic solution it slowly converts itself to a colorless species, while, in strongly acid solution an orange species is slowly formed.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Computer programs were written and used to calculate particle and fluid flow parameters for all current deposition data. The much more rapid handling of the data permitted a re-examination of the correlation previously developed. The independent review showed the theoretical basis upon which the correlation of variables was established to be valid and supported the observation that a conduit diameter term was needed to yield better correlation of the data.

A computer program is being written to calculate particle deposition for a distribution of particle sizes passing through a conduit.

W. H. Reas
Manager
Chemical Research and Development

WH Reas:cf

BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

Dr. Roy E. Nakatani was appointed Manager, Aquatic Biology, effective July 1, 1962.

GENERAL

On July 5 and 6, 1962, the AEC's Inhalation Toxicity Group met at Hanford. Eighteen scientists from AEC-supported laboratories attended. Dr. W. J. Bair was in charge of arrangements.

Dr. R. F. Foster was guest of the Internal Emitter Committee on July 11. Results of environmental monitoring programs and P^{32} uptake studies were discussed. Dr. R. C. Thompson also reviewed the recent meeting of the NCRP, held in Salt Lake City.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

The swimming performances of young Chinook salmon (73-74 mm) reared in 0, 3, and 5 per cent reactor effluent were compared in a hydraulic test flume with a 1.6 fps water velocity. Eight experimental groups with 11 fish per group were tested after they were reared in each of the three solutions. No statistical differences due to the effluent were found. The mean performance times for the second trial were 370, 312, and 425 seconds for the 0, 3, and 5 per cent effluent groups, respectively. As expected, the more important sources of variation in the observed swimming time were trials or number of experiences with the test flume and size of test fish. The mean performance time for each test group (73-74 mm) was treated as the independent normal random variable for the preliminary analysis of variance summarized as follows:

Source of Variation	Degrees of Freedom	Mean Square	Variance Ratio
Effluent effect, E	2	65,080	1.30
Trial effect, T	1	106,597	2.13
Interaction (E x T)	2	58,925	1.18
Error	42	50,017	

Monitoring of 100-KE reactor effluent with young rainbow trout was temporarily postponed because of adverse water temperature problems.

Columnaris

Columnaris was noted for the first time this season in 146-FR troughs and 100-KE troughs. River fish from 100-H slough and Hanford area were not infected.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Phosphorus

Cichlids on chronic feeding of P^{32} are approaching sexual maturity. Spawning pairs will be isolated soon to study the effect of body burden of P^{32} on reproduction. No apparent difference in mortalities have been observed after seven months of P^{32} feeding.

Zinc

Preliminary results of a single intraperitoneal Zn^{65} injection given to rats fed 25, 50, or 75 ppm cadmium indicated that as the dietary cadmium level increases, Zn^{65} uptake by certain tissues, notably liver and kidney, increases. An exception was the small intestine, less the duodenum, in which Zn^{65} accumulations decreased as the cadmium levels increased.

Some preliminary work to study the distribution of Zn^{65} in rainbow trout after a single oral administration was initiated with the assistance of Mr. William P. Miller, an AEC Health Physics Fellow from University of Washington. The distribution of Zn^{65} in various tissues (blood, eyes, kidney, liver, muscle) at different times is to be studied.

Iodine-131

Eleven female Palouse swine, including five controls and six that had received 5 μc I^{131}/day for four and one-half years, were killed and autopsied. The control animals were each given 50 μc of I^{131} at 18 to 48 hours prior to posting. Iodine-131 concentrations per gram of tissue in thyroids of the control group ranged from 0.8 to 2.1 per cent of the single 50 μc administered dose and the concentration in the 5 μc/day group of animals was 18 per cent of the daily administered dose. The thyroids from the 5 μc/day group appeared more fibrotic and weighed one-half, or less, than thyroids from the control animals.

TSH determinations on serum from three ewes (control, 5 μc I^{131}/day level, and one that received a single oral 3 mc dose in 1958) indicated that the 3 mc animal had a TSH level in the serum of 4.9 mμ USP per ml, while the other two ewes had levels not detectable ($<$0.45 mμ USP per ml) by this method.

A ewe given a single oral 3 mc dose of I^{131} in 1958, when six months old, expired showing symptoms of hypothyroidism. Histological examination of the severely damaged thyroid remnant revealed an adenoma. This is the first time a thyroid adenoma has been observed in a sheep following a single dose of I^{131}.

Thyroid uptake of I^{131} in four weaned male lambs about 3 months of age was determined with the whole-body monitor. Each lamb was administered 5 µc of I^{131} by a different route, i.e., intravenously, subcutaneously, orally, and topically. Peak I^{131} concentration in the thyroid was about 15 per cent of the administered dose in all but the animal given the topical dose, in which peak concentration appeared to be less than two per cent. The study is being repeated with higher tracer doses and varying the routes of administration.

Plutonium

The liver function of two male sheep intravenously injected with 180 µc of Pu^{239} was determined using I^{131}-labeled rose bengal. There appeared to be no acute effect from this dose of Pu^{239} on liver function.

To study the comparative toxicity of Pu^{238} and Pu^{239}, rats were injected intravenously with 0.9 µc of one or the other of these isotopes and distribution determined after 1, 3, 8, and 21 days. Data available for the 1- and 3-day sacrifices indicate no difference in the distribution pattern of the two isotopes. This experiment will be extended to study distribution of a higher dose, 20 µc, at which level differences in toxicity of the two isotopes have been noted.

Neptunium

Studies were continued to evaluate the chemical toxicity of Np^{237}. Rats were intravenously injected with 6, 12, or 24 mg of citrated neptunium per kg of body weight, and sacrificed after 24, 48, and 72 hours. Animals receiving 12 and 24 mg did not survive past 48 hours. Some of the 6 mg/kg rats died within 48 hours but others survived to the 72-hour sacrifice. Gross examination revealed fatty livers in the Np^{237} injected animals, even at the 6 mg/kg level. This was apparent as early as 24 hours after injection. A parallel experiment wherein rats were injected with lead nitrate resulted in distinctly different symptoms. None of these animals died but all showed grossly enlarged spleens and livers, apparently due mainly to edema. Histological results on these animals are not yet available.

In cooperation with W. E. Keder, Chemical Research and Development Operation, spectrophotometric studies were made of the valence state of neptunium in various forms suitable for injection or feeding to rats. Techniques are available for the administration of neptunium in either the 4, 5, or 6 valence states. The presence of citrate seems to stabilize the 4 valence state and earlier results obtained using neptunium-citrate solutions in which other valence states were assumed to be present, now appear questionable.

Radioactive Particles

A further experiment to determine the effectiveness of I^{127} as a diluent of I^{131} aerosols was completed. Using I^{127} crystals as a source of free I^{127} only slightly increased the aerosol concentration (2×10^{-3} μg I^{127}/ml of air) obtained in previous experiments. The maximum I^{131} level in thyroids of the rats exposed to mixed I^{131}-I^{127} aerosol was about 2.5 per cent of the total I^{131} deposited during the 42 hours following exposure. This is about two-thirds of the lowest levels seen in previous experiments, Table 1. These results encourage further efforts to increase the levels of the I^{127} exposure.

In order to determine whether the adrenal cortex is involved in the lymphopenia caused by exposure to $Pu^{239}O_2$ aerosols, 20 rats were adrenalectomized and 10 were sham operated. Ten of the adrenalectomized rats are being maintained on hydrocortisone. The lymphocyte levels of all rats will be studied following exposure to $Pu^{239}O_2$ aerosols.

Biochemical studies on animals exposed to radioactive aerosols were started. $Ce^{144}O_2$ and $Pu^{239}O_2$ deposition had no effect on transaminase and lactic dehydrogenase levels in the blood of dogs over a period of four months. Methods for determination of 17 hydroxy corticosteroids and for extracting lung mucopolysaccharides were developed.

In dogs, aerosol and intraperitoneal injection administrations of DTPA were equally effective in clearing the lungs and body of Ce^{144}-Pr^{144} following exposure to $Ce^{144}O_2$. Treatment beginning immediately after exposure was more effective than when started after 5 or 10 days, Table 2.

Table 2. Effectiveness of DTPA in Removing Inhaled $Ce^{144}O_2$

	Retention of Ce^{144}-Pr^{144} one month after exposure	
Dogs	Per cent of total deposited	Per cent of body burden five days after exposure*
Controls	75, 74, 63, 66	86, 88, 82, 87
Treated immediately**	5.6	32
Treated after 5 days	25	29
Treated after 10 days	27	42

* These values are given because occasionally 50 per cent or more of the total radioactive aerosol is cleared and excreted by the 5th day after exposure. It did not occur in any of the controls, but there remains a slight possibility that it could have occurred in the treated dog had it not been treated. If so, the effectiveness of DTPA would be over-emphasized by the 5.6 per cent value.

**A second dog, given DTPA by intraperitoneal injection, showed a similar response. However, the dog died about one week after exposure (cause unknown).

Table 1

Effectiveness of I^{127} as a Diluent of I^{131} Aerosol

Time after exposure	Thyroid uptake (Per cent of total I^{131} deposited)			
	A	B	C	D
	6 mc I^{131}*	6 mc I^{131} + 100 mg I^{127}* I^{131}/I^{127} = 250**	5 mc I^{131} + 250 mg I^{127}* I^{131}/I^{127} = 50**	4 mc I^{131} + ? I^{127}* 8**
Immediately	0.92 1.4 1.4	0.39 0.54 0.49	0.23 0.61 0.32	0.53 0.8 0.7
18 hours	12 14 7	3.8 3.9 4.1	4.1 2.1 2.3	2.4 1.4 2.8
21 hours	7.4 11 3.5	4.5 4.7 4.2	1.5 2.7 3.	2.8 2.5 1.7
24 hours	9.5 3.6 4.1 10	3.1 4.3 3.5 4.7	4.5 1.6 3.7 2.6	1.6 2.3 2.2 2.3
42 hours	27 18 14	19 3.9 5.2	5.3 2.6 3.3	2.9 1.5 2.1

* I^{131} and I^{127} released, ** ratio of I^{131} (μc) to I^{127} (mg) in chamber.

The clearance of Ce^{144} removed by DTPA is via the urine. The biological half-life for whole-body retention of inhaled Ce^{144}-Pr^{144} was 400 days in two dogs over a 100-day period after exposure. The effective half-life was about 170 days.

A second dog died about two and one-half years after exposure to $Pu^{239}O_2$ aerosols. The lungs showed gross effects. Three other dogs are beginning to show respiratory involvement but no gross changes are detectable radiographically.

Radiation Protective Agents

Rats which received 800 r whole-body X ray were pre-treated with 1.5 mM/kg EDTA as the zinc, manganese, calcium, or nickel salt. Five out of ten untreated control rats died within 27 days. Seven out of ten of the nickel and calcium EDTA animals survived. Eight out of ten zinc EDTA animals, and all of the manganese EDTA animals survived. Although the numbers of animals employed were small, the results suggest that the protective effects of treatments employed may be dependent upon the cation rather than on the EDTA moiety. The superiority of the manganese salt may possibly be explained by the fact that it is present in the divalent state and has more available reducing power than the other cations employed.

Preliminary toxicity studies have been made with the polystyrene sulfonate obtained from Kroll. This material is highly water soluble but is expected to be very slightly absorbed and may be the useful chelating agent for wound decontamination.

Bone Marrow Studies

Studies on modification of secondary disease continue with no definitive results to report this time.

Cellular Studies

After X-irradiation, yeast cells lost K to washing solutions. The capacity to retain K was restored after being placed in glucose solution, thus indicating damage to K absorptive mechanism was reparable. Cellular phosphate, unlike K, continued to decrease.

Uptake of fully deuterated glucose by yeast cells was less than normal glucose, regardless of media, either D_2O or H_2O.

Plants

Uptake of Cs^{137} by beans from contaminated field plots was slightly increased by potassium. Uptake of Cs from soils which had been surface contaminated was less than tilled soils.

All plants grown in nutrient solutions during the last six weeks were abnormal and not satisfactory for experimental purposes. The toxic agent has not been identified but is believed to be in the water supply.

Plant Ecology

Uptake of Se^{75} by cheatgrass from field soil placements was greatest at 6-inch depths with diminishing amounts from 12, 18, and 24-inch placements. Uptake from the maximum placement depth, 30 inches, was barely detectable.

Annual above-ground harvest yield of cheatgrass averaged 86 g dry weight per square meter in 1962 as compared to 100 g per square meter in 1961. The reduction in harvest yield is attributable to less fall and winter precipitation during the 1961-1962 season.

The uptake of Cs^{134} by duckweed (Lemna sp.) was studied under varying ecological conditions. Results obtained recently suggest that Cs^{134} assimilation is inversely related to light intensity.

Columbia River Limnology

A study of the accumulation and transfer of radionuclides from the Hanford reactors by Columbia River plankton was initiated. The gamma emitters found in plankton, in order of decreasing abundance were Mn^{56}, Cu^{64}, Na^{24}, Cr^{51}, La^{140}, Zn^{65}, Zn^{69}, Sc^{56}, Sb^{122}, Mn^{54}, and Co^{60}. The levels of Mn^{56}, Cr^{51}, and Zn^{65} were 3.6×10^4, 1.1×10^4, and 2.3×10^3 pc/g wet weight.

A significant decrease in the chloride concentration of the water has been evident since May. Nitrate and phosphate concentrations continue to decline. This is assumed to result from utilization by phytoplankton populations.

Radiation Effects on Insects

Ephestia larvae were given 0, 10, 15, and 20 kr X radiation. The controls pupated and then developed into adults, but all radiated groups remained as larvae. Larvae that three weeks previously had received 22 kr X radiation and controls were analyzed electrophoretically to determine protein changes in hemolymph. Four protein fractions were demonstrated. Two fractions from the X-rayed larvae were lower and one was higher than in the controls. The other fraction appeared similar in both groups.

Project Chariot

The total body burden of Cs^{137} in 150 permanent resident Eskimos from Kotzebue, Alaska, averaged about 135-145 nc, with a maximum of about 500 nc. Eskimo students who had recently returned from attending schools elsewhere in Alaska for a nine-month period averaged about 45 nc. White residents of Kotzebue generally contained lower Cs^{137} levels than the Eskimos.

From Point Barrow, Alaska, 256 permanent Eskimo residents contained an average of about 35-50 nc of Cs^{137}, with a maximum of about 170 nc. Students who had recently returned to Point Barrow averaged about 15 nc.

The highest Cs^{137} body burdens generally occurred among the Anaktuvuk Eskimos. Fifty-three permanent residents of the Anaktuvuk Pass region averaged about 420 nc with a maximum of about 790 nc. These higher levels are apparently due to greater usage of native foods from the terrestrial environment.

The diet of the Kotzebue and Point Barrow people includes much marine fish and mammals--seal, whale, and walrus. The Anaktuvuk natives, on the other hand, rely upon caribou as a primary food source.

Three men from the Radioecology Operation reported to the Project Chariot site to initiate field collections for radioecological studies of the environs.

SPECIAL

On July 29, at approximately 4:15 p.m., a fire occurred in the climatizer room of the 108-F building. The fire was attributed to a defective ballast serving the light system. The fire was restricted to one room, and total damage to room and equipment was estimated to be about $20,000. Radioactive isotopes (tracer quantities) were contained and no spread of contamination was detected. There was no reported injury associated with the incident.

J. J. Davis

Acting Manager
BIOLOGY OPERATION

JJ Davis:es

C. Lectures

a. Papers Presented at Society Meetings and Symposiums

None

b. Off-Site and Local Seminars

Inhalation Toxicity Meeting, Richland, Washington, July 5 and 6, 1962:

Brown, M.G. Hanford's dog colony.

Park, J. F., D. H. Willard, S. Marks, J. E. West, G. S. Vogt, and W. J. Bair. Acute and chronic toxicity of inhaled plutonium in dogs.

Tombropoulos, E. G., and W.J. Bair. Treatment for removal of inhaled $Ce^{144}O_2$.

Bair, W. J., J. P. Herring, and L. A. George, Jr. Retention and excretion of inhaled PuO_2.

Rhyneer, G. S. Proposed study of Pu^{239} induced lymphopenia.

Tombropoulos, E. G. Proposed lung biochemistry studies.

Casey, H. W., W. J. Bair, J. F. Park, E. G. Tombropoulos, and W. J. Clarke. Planned experiments on acute effects of $Ce^{144}O_2$ inhalation in the dog.

Summer Institute, University of Utah, Salt Lake City, Utah:

Dockum, N. L. July 3, 1962. Autoradiography of tissue.

Uhler, R. L. July 19, 1962. Factors affecting the accumulation of cesium and strontium by plants.

Bair, W. J. July 26, 1962. Inhalation studies of internal emitters.

Summer Institute, University of Washington, Seattle:

Mahlum, D. D. July 2, 1962. Effects of radiation on cells.

Mahlum, D. D. July 3, 1962. Biochemical effects of radiation.

Uyeki, E. M. July 12, 1962. Modification of radiation injury.

Health Physics Fellowship Program, 300 Area:

Thompson, R. C. July 2, 1962. Internal emitters.

Uyeki, E. M. July 30, 1962. Cellular radiobiology.

c. Seminars (Biology)

Dr. Vernon H. Cheldelin, Dean of Science, Oregon State University, Corvallis, Oregon, "Biosynthesis of glutamic acid in an organism lacking a Krebs cycle" - July 23, 1962.

Dr. Marylou Ingram, University of Rochester, Rochester, New York, "Some newer hematological responses to radiation" - July 30, 1962.

d. Miscellaneous

None

D. Publications

a. Open Literature

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.

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.

McClellan, R. O., J. R. McKenney, and L. K. Bustad. 1962. Dosimetry of caesium-137 in sheep. Nature 194: 1145-46.

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

b. Documents (HW)

None

OPERATIONS RESEARCH AND SYNTHESIS OPERATION
MONTHLY REPORT - JULY, 1962

ORGANIZATION AND PERSONNEL

Effective July 1, 1962, D. P. Granquist transferred to the Operations Research and Synthesis Operation from the Programming Operation, and R. J. Brouns transferred from the Operation to the Chemical Research and Development Operation.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

Consulting assistance continues to be provided in connection with interpreting output from the MERCY program concerned with evaluating measurement errors at various stations within FPD.

Ingot chemistry data from the feed sites are being incorporated in the HAPO data system. For some elements with lower detection limits exceeding impurity levels for a fair proportion of ingots, some technique has to be used which gives a realistic summarization of the data. It was found that utilization of censored distribution techniques, on either the raw or transformed data, provides a solution. Appropriate calculation formulas were supplied for incorporation into the data processing system.

Additional data are being analyzed from the multiple failure lot, HZ-065. Several types of measurements are being made on the uncharged fuel elements from this lot and on "control" lots to determine if its rupture proneness could have been anticipated on the basis of preirradiation measurements.

In connection with re-measurement of fuel elements in 100 percent inspection, appropriate formulas were derived to use in estimating tester efficiency and product quality when the items are measured three times. Previously, formulas were developed for the situation when items were measured twice.

Wettability data from two pilot plant experiments were analyzed as requested.

Silicon concentration data from the duplex and canning baths over a one month period were analyzed to determine the ability of the process to remain within specifications.

[redacted]

Data from an experiment to determine the effect of silicon content, lead temperature, and lead preheat time on such dependent yield variables as internal and external total bond count, percent UAL_3, primary and total bond layer thicknesses, stud pull strength and number of cracks have been analyzed. A response surface model has been fitted to each of the yield variables and the results have been graphically depicted. The combination of independent variable levels which gives the best results in general still needs to be determined.

Data from an experiment to determine the effect of different core and sleeve combinations on the ellipticity of the canned fuel element have been analyzed. Several different core and sleeve combinations were investigated. The results confirmed a previous analysis.

A seven variable experiment ($2^2\ 3^5$) was designed for use in evaluating effects of certain control variables in the ultrasonic welding operation.

Rail height data for KIVN fuel elements were analyzed to determine if a given type of sizing can produce fuel elements meeting minimum specifications on bumper height and maximum specifications on projected diameters.

Comments were provided on the rupture testing of dingot material produced after the moratorium.

The behavior of wall thickness along an extrusion was characterized for NOT extrusions.

Extensive preparations are being made to analyze NIT data back to the first extrusions. The primary purpose of the analysis is to determine process capabilities with respect to control of warp, wall, and clad thickness.

<u>Irradiation Processing Department</u>

During the month the time distribution records for the period 8/27/61 through 5/27/62 were matched to the active work order master file for 7/1/62. From this operation two sets of work order listings were obtained, active and closed. Some time was expended to determine whether these orders could be translated back to expense account codes which describe specific job activities. These work order sets are now in the hands of IPD Financial personnel where each is to be tagged with the appropriate expense account code, or deleted. The analytical program to accumulate these data by reactor, by job, by craft, by assigned or borrowed labor, by straight or overtime hours is in the final stage of debugging. A program has yet to be written to substitute the above mentioned expense account codes for the various work order numbers in the source data. By the time the master listings are created and returned, the analytical and substitution program should be operabl[redacted]

[redacted]

An extensive analysis is being made of the warp and hot spot relationship, one purpose being to compare reactors. Preliminary results show that there are differences between reactors with respect to the frequency of hot spots for a given amount of warp. In view of the differences in annuli between reactors, this result is not surprising. Attempts will be made to see if differences between reactors disappear after accounting for differences in annuli.

The high incidence of 80 and 90 profiles in the PT-310 data has spurred on an investigation into the measurement capabilities of the C-basin Profilometer, particularly with regard to measuring bumper fuels. Changes in techniques and processing of the data will probably be recommended as a result of this study.

Measurement precisions and biases for various types of inner diameter gages were estimated based on several measurement runs by different operators on different gages.

Work continued on the problem of estimating defect frequency and size distributions in connection with welded primary piping for the NPR project.

Chemical Processing Department

The demonstration of the specified fabricated part plutonium and Pu^{240} content was readily made for the second quarter of this calendar year.

A review was made of the shipper-receiver differences for total U^{235}. As a result of the significant difference noted in the mass spectrographic measurements by the two laboratories, a modification of the routine monthly measurement procedure was adopted to aid in the elimination of bias and/or identify unexplained variance.

The available data obtained in the demonstration of dimensional stability of fabricated parts are being examined to better define stability characteristics. While an over-all tendency for shrinkage is evident, significant interactions exist, indicative of differences in types and amounts of shrinkage from one part to another.

A study of the patterns of statistical regularity among railroad accidents is being continued for use in assessing the risks associated with shipping radioactive materials. Data, which have been received from the ICC, give substantial information on each railroad accident which has occurred in the years 1958 through 1960. These data have been put on punch cards and use is being made of several data processing programs which produce univariate and

[redacted]

bivariate frequency distributions. The ultimate objective is to obtain a function giving the probability that a freight car, x cars behind the engine will sustain a given impact.

Work was begun on a spare parts and general inventory control analysis.

Relations Operation

Work continues in planning for the forthcoming HAPO-wide attitude survey.

D. STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HLO

2000 Program

Pulse Column Facility

A joint paper was written which describes the data logging system for and calibration of the gamma absorptiometer used to assay the uranium concentration of pulse column aqueous and organic phases.

Work continued on the power spectrum analysis of the stability experiments for the estimation of the pulse column flooding curve as a function of capacity and pulsing characteristics.

Fuel Element Swelling Model

The NELLY nonlinear least squares program was adapted to the estimation of the parameters in a fuel element swelling model derived by Fuels Development. The program was tested successfully with experimental swelling data and was turned over to the customer as a research tool.

General

The mathematical analysis and EDPM program which determines the steady-state nonviscous flow pattern of a fluid in a cylindrical tank has been completed and placed into service for the customer. Studies can now be made on the effectiveness of various axially-located circulating devices on concentrating suspended waste products.

Modifications and additions have been completed on an EDPM program for the rapid and efficient evaluation of the longevity and sensitivity of a proposed neutron flux monitor as a function of its original isotopic constituents.

Density and homogeneity measurements on additional ceramic fuel elements whose particle sizes and size distributions had been determined by theoretical

[redacted]

[redacted]

considerations continue to confirm their exceptional quality. Calculations are being made for other fuel element geometries.

Detailed mathematical analysis is under way on the theory of the propogation of Lamb waves in metal plates. This study was initiated in an effort to more effectively interpret data obtained from specimens in the nondestructive testing program.

A nomograph was made which enables one to calculate the weight of material of a given percent enrichment which may be put in combination with other materials of various amounts and enrichments and still maintain a safe criticality level.

3000 Program

Machining Development

The pulse motors, magnetic tape reader, and associated equipment have been assembled on the experimental Gorton lathe. A magnetic tape was generated to mill a prototype model of an actual manufactured part, and several experimental runs have been made. In general, the lathe appeared to perform as intended, but a more critical evaluation awaits the results of dimensional and surface finish measurements.

Agreement was reached on several experimental designs for metal blanks which are to be shear-spun on a Floturn machine into preselected shapes. An EDPM program is being written which will specify the precise geometry of the blank, lathe coordinates to machine such a blank, and lathe coordinates to machine mold surfaces for casting blanks.

4000 Program

Plutonium Fuels Research

An experimental design was devised for the estimation of the radial void distribution in tubular fuel elements using micrographs of cross sections of the element. The design is based on the systematic point count method of estimating void fraction from a two-dimensional random section. Data on a single micrographed cross section were collected by two observers and analyzed on the IBM 7090 with appropriate analysis of variance routines. The results of the analysis, including the radial void distribution, the observer effect, and accompanying precision estimates were supplied to the customer.

Swelling Studies

Modification of the existing program for detailed analysis of micrograph pore size distributions was completed, and the data reevaluated.

5000 Program

Actinide Element Research

Work on indexing crystals is being concentrated on the hexagonal case. Work on the orthorhombic case has been temporarily suspended until the major difficulties of the hexagonal case have been solved.

A report (HW-74198) has been issued describing the function and use of the "90° extrapolation" program to determine lattice parameters. A generalized version of this program has been requested. Two meetings have been held, the exact problem defined, and the desired results decided upon.

The final draft of the report on the search pattern and detection grid problem has been completed.

Division of Research Programs

Several improvements in the GEM FORTRAN language program were made during the month. A subroutine was added which would check the inversion of the coefficient matrix and, if necessary, compute the inverse using a slower but more accurate method. Calculation of the inverse was attempted using eigenvectors, double precision, and an iterative (Crout method) routine with the last method giving the greatest improvement. Several other modifications have been made to GEM to improve the numerical accuracy of the program.

Several cases have been run on the Monte Carlo program which will test program GEM using data generated in a controlled manner.

6000 Program

Environmental Studies

The statistical analysis of data to investigate the relationship of certain factors to the uptake of P^{32} and Zn^{65} in Whitefish was continued. The factors under consideration are temperature of the river water, concentration of isotopes in river water, and the rate of flow of the river water.

Biology

Work continued on applying a multicompartment model to a study of Pu retention i[redacted] Better approximations of rate parameters are being sought by using an iterative technique.

[redacted]

Work was renewed on data from an Alaskan fallout study. Presence or absence of isotopes is to be detected by analyzing data from a multi-channel analyzer.

A model has been fitted to a problem associated with liver damage in sheep. Further analysis is being done to provide tolerance bands for the curves being plotted.

Other

Instrumentation

The mathematical details of the GRA program modification providing for the use of monthly updated background estimates and the print out of individual nuclide precision estimates were worked out. Copies of the mathematics and some suggestions for inclusion in the FORTRAN language program were supplied to the customer and to EDPO.

Personnel Monitoring

The statistical analysis of data to evaluate the present pencil program was continued. The relationship between dose level and exposure level was determined and calibration curves were constructed.

Statistical analysis of pencil data for the first four months of 1962 was completed. The purpose of the analysis was to determine if the readings of pencils worn together were significantly different and to make a tolerance statement regarding the magnitude of the differences of the readings of two pencils worn together.

Work is continuing on developing a technique where the individual urinalysis samples are tested for Pu content only after a composite sample fails to meet acceptance criteria. The detection of Pu is sensitive enough and the frequency of samples above the permissible level is small enough to enable a procedure of this sort to work.

General

Some time was spent defining and organizing a study to delineate the operational processes in the Radio-Met laboratory to permit establishment of an information system which will provide information to use in justifying additional manpower, equipment and/or facilities.

Carl C Bennett

Manager
Operations Research and
Synthesis Operation

CA Bennett:dgl

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Code Development

1. MELEAGER

The calibration of MELEAGER carried on over a period of several months is now completed for the present. The calibration included the following changes:

a. The spectral index r calibration indicates that MELEAGER code can now be applied to evaluate reactors having far more plutonium enrichment than before. The change involves adoption of a formulation derived empirically to give agreement with SPECTRUM V code which uses 75 energy groups to develop cross sections.

b. Americium-241, formed by decay of Pu-241, has a pronounced effect on exposure attainable in plutonium enriched reactors; thus, economics are also affected since credit is not given for the valuable by-products from Am-241. SPECTRUM V with 75 energy groups, was used to calibrate cross sections in MELEAGER. Values were chosen for the cross section constants which leave MELEAGER slightly pessimistic. Other isotopes were also checked with SPECTRUM.

c. A neutron temperature correction term was added to MELEAGER to properly increase the effective neutron temperature as the neutron spectrum is hardened. The correction is based upon earlier work by Applied Physics Operation which shows that a neutron temperature correction could be based on the cell averaged absorption cross section and the cell averaged moderating power.

d. An improved over-all formulation for the reactivity was adopted that agrees better with theory and experiment for reactors having large amounts of nonfuel absorbers than the previous formulation.

e. The fast effect ϵ was not precisely reflected in the power summation (exposure MWD/Ton) and this was corrected as studies with significant fast effects are now contemplated.

f. An additional array was added to the Physics library tape to supply the specific fission energies for the various fissile species. Fission energies vary as much as seven percent between fissile species such as U-233 and Pu-241. Formerly, a median value had been used for all fissions regardless of species. This becomes important in the Combined Cycles studies where different fuels are to be compared.

g. A new method of estimating the neutron flux for each burn-up step was adopted. It was observed that considerable error could exist from large time steps due to the fact that the flux for a time step is based on the initial fissile concentration. Little or no error existed if either the time steps were made small or the rate of grow-in of fissile material approximately equaled the rate of burnout. It was shown that relatively large time steps can be taken with negligible error if the flux for a time step is based on a predicted (for the time step) average fissile population, rather than on the fissile population at the beginning or end of the step. The flux is now based on a sophisticated prediction of the population density of the fissile species, and comparisons of the energy released versus decreased mass of the fuel give excellent agreement.

h. The linkage of MELEAGER (known as MELEAGER D) and a two-group diffusion code has been debugged. A new generating code has been written to handle the extra data needed for diffusion.

Three factors will require future attention, either in MELEAGER or in other improved burn-up codes that are being written.

a. The calibrations were limited to a maximum energy of 4.2 ev, the limit of SPECTRUM V code as utilized for these experiments.

b. The calibrations were for the homogeneous case. No geometrical shielding is considered by SPECTRUM V. MELEAGER does have some geometrical shielding capabilities but they are not calibrated with the care given to other factors because a suitable convenient standard is not available. Geometrical shielding assumes greater importance with plutonium fuels than with U-235. As currently formulated, MELEAGER does not fully account for geometrical shielding and, as a consequence, tends to underestimate plutonium value.

c. Several variations exist of the basic MELEAGER deck. These special options are to be combined into a single master deck in the near future. These variations include: (1) a burnable poison option, and (2) a zoned spectrum option that allows for adjustment of lattice parameters during the burn-up cycle.

2. Special CHAIN

The bookkeeping of the various codes tapes became so involved that it was decided to chain the codes for general burn-up analysis as was done for recycle analysis. At the same time that the codes were chain-linked they were modified so that future chain linkage would become very simple. Each code is so written that it can operate equally well as a separate code or as a link in a chain sequence. In addition, JASON code has been so modified that it prepares an input tape for MELEAGER-CASE GENERATOR. Also, the CASE GENERATOR has been modified to receive changes from the JASON output tape and to prepare input cases for MELEAGER code.

Due to the outstanding cooperation given by the authors of the various codes, it was possible to complete a working chain quickly containing JASON-CASE GENERATOR-MELEAGER-PROTEUS-QUICK-PLOTTER links. DIFFUSION and CALYX codes are to be added. QUICK and PLOTTER will be modified so they can call subsequent links if desired (at present they are terminal codes -- when they complete, they stop the entire computer run).

Combined Cycles Studies

An experiment was carried out with the latest version of MELEAGER to compare fuel costs with those calculated previously. It was concluded that the previously calculated library of fuel costs has considerable value for obtaining preliminary answers to numerous questions; however, they are to be checked with the later code before publication.

JASON code is utilized to make more valid comparisons among the various fueling schemes. It calculates the input parameters for MELEAGER code at each enrichment level for the actual fuel element geometry under study. The nonfuel neutron absorptions are dependent on the amount of moderator (which can absorb neutrons), the quantity of fuel that affects the flux depression in a fuel element, and the neutron spectrum that affects the effective fuel cross section. In previous combined cycle studies, the neutron spectrum was assumed to be independently variable because it was not practical to compute the proper SNF* for each combination of moderator and enrichment.

Three generalized reactor types have been selected for a pilot study of the various fueling schemes. These include reactors moderated with D_2O, graphite, and water. Preliminary results have been obtained with the ECONOMICS CHAIN sequence during debug operation.

During the original design phase of a reactor, it is possible to choose a lattice spacing that gives optimum results for a specific fuel. This might not be the optimum for another fuel. It is not expected that the lattice spacing ideal for U-235 will also be the ideal for plutonium. This becomes a very important factor of the Combined Cycle Study in which comparisons should be made between each of the ten fueling schemes at minimum fuel cost at optimum spacing. This minimum fuel cost is determined with both lattice spacing and fuel enrichment as prime variables.

* SNF = Sigma Nonfuel; i.e., the flux weighted nonfuel absorption cross section.

The following are preliminary results and are presented only for illustration of the principle.

Description

Reactor Type	- Water moderated.
Fuel	- UO_2 95 percent theoretical density U-235 enriched 0.520-inch diameter fuel in hexagonal array 0.010-inch cladding stainless steel and zirconium (alternates).
Lattice Spacing	- Varied, reflected by SDPV* term.
Moderator Temperature	- 260 C.
Specific Power	- 15 MW/Ton of uranium.
Size	- 3000 MW thermal.
Fueling Method	- Batch.

TABLE I

THE EFFECT OF LATTICE PITCH (Reflected in SDPV) ON MINIMIZED TOTAL FUEL COSTS (MTFC)

Lattice Pitch (in.)	0.642	0.673	0.710	0.745	0.81	0.928	1.127
SDPV	0.75	0.95	1.2	1.4	1.9	2.8	4.7
MTFC, mills/kwh	1.79	1.48	1.35	1.31	1.33	1.44	1.63

Thus, by means of the CHAIN-LINKED code sequence, it will be possible to make more nearly valid comparisons rapidly among the various fueling schemes because it is now practical to optimize each fueling scheme for each of three basic reactor types.

* SDPV is the moderator index and is the slowing down power of the moderator normalized to unit fuel volume.

Calculation of Fuel Element Fabricating and Jacketing Increments ($\triangle$FEFJ)

The new fuel fabrication cost code has been completely tested. Cases were set up for a model fuel element fabrication plant (19-rod PRTR type) as described in HW-74304 that is soon to be published by H. Hanthorn. Both the uranium and plutonium fabrication lines were used.

The "delta" values, plutonium fabrication costs minus uranium fabrication costs, were calculated varying the recycle and reject rates for each fabrication step. The totals and corresponding "deltas" for various multiples of the base recycle and reject rates are shown in Table II.

TABLE II

CALCULATED FUEL FABRICATION COSTS FOR DIFFERENT RECYCLE AND REJECT RATES

Multiples of Reject and Recycle	Calculated FEFJ, $/pound			
	1	2	4	8
Pu enriched fuel	$30.30	$31.46	$34.09	$41.24
U enriched fuel	28.30	29.20	31.26	36.93
"Delta"	2.00	2.26	2.83	4.31

Another set of calculations was made in which the added costs for labor and equipment in the plutonium steps were increased to represent further difficulties that shielding may present as higher exposure plutonium is processed. The "deltas" resulting from these calculations are shown in Table III.

TABLE III

CALCULATED ENRICHMENT FEFJ, "DELTAS," FOR VARIOUS OPERATING EFFICIENCY AND SHIELDING COMPLEXITIES

Multiples of Recycle and Reject	1	2	4	8
Multiples of Labor and Equipment	"Delta," $/pound of Uranium			
1	2.00	2.26	2.83	4.31
2	3.73	4.03	4.76	6.45
4	6.04	6.39	7.18	9.20
8	10.75	11.21	12.27	14.85

At the extreme, this shows that if the calculation of labor, equipment, reject, and recycle were in error by a factor of 8, the "delta" would be only $14.85 per pound.

Plutonium and U-233 Values Computed with MELEAGER CHAIN

Calculations of plutonium value for the five reactors, with most of them having six fueling systems, are nearing completion with 35 percent in their final form and the rest in various stages of iteration. This work is a gross extension of the analysis reported in HW-72217, "Fuel Cycle Analysis for Successive Plutonium Recycle - Part I." Aside from adoption of logical improvements in the codes and re-evaluating the cases reported in HW-72217 over a wider parameter range, two additional cycles are considered. The first is named Plutonium Enriched Successive Recycle, which presumes an unlimited stockpile of plutonium identical in composition to the plutonium discharged from the previous cycle. This plutonium is mixed with natural or tails uranium and does not reach an equilibrium composition, as does successive recycle in which the available plutonium limited to the quantity discharged from the previous cycle. The second cycle considered involves thorium enriched with U-233 from the previous cycle and U-235 from the cascade as necessary. The status of these computations is summarized in Table IV.

TABLE IV

PROGRESS SHEET CHAIN MELEAGER COMPUTATIONS*

Reactor Simulation	Fueling System: U-235 Enriched Pu Recycle, Batch	U-235 Enriched Pu Recycle, Graded	Pu Enriched Pu Recycle, Batch	Pu Enriched Pu Recycle, Graded	U-235 Enriched U-233 Recycle, Batch	U-235 Enriched U-233 Recycle, Graded
APWR	F	1	F	X	F	X
BWR**	2	2	2	1	F	F
HWR	2	2	2	2	F	2
GCR	2	2	F	2	F	2
OMR	2	1	F	X	1	X

* Symbols used in the Table have the following meanings:

1 = no iteration
2 = iterated
F = no further iteration required
X = not to be investigated

** There will be many variations made to the BWR case including reactor parameters as well as economics.

Premature Discharge Price Calculation

An investigation was begun to determine the market price of bred plutonium sufficient to justify interruption of a planned irradiation and premature sale of the plutonium and spent uranium, rather than to await planned discharge dates. If the premature discharge price should become the firm market price for plutonium, it would be more economical to reduce the enrichment of the reactor in subsequent fuel loadings and appropriately maximize production for this market price. The premature discharge price represents the worth of the increased exposure contributed by the bred plutonium. A similar price could be allocated to uranium or special fission products. The premature discharge price for plutonium is a function of the nuclear performance of the plutonium in a given reactor and assumed economic environment. It bears little relation to the recycle value (PUVE solution); in that, it does not represent optimal reactor operation at equilibrium but, rather is a guidepost for dynamic adjustments if the plutonium market price is subject to large unforeseen swings.

The term in situ value, has sometimes been applied to the premature discharge plutonium price. However, this price does not represent in situ value for the usual case in which the fuel is irradiated to a reactivity limit and this is reflected in the premature discharge price formulation when evaluated for vanishingly small increments near the reactivity limit. In situ values are academic in the sense that plutonium to a potential customer is valued in marketable form such as plutonium nitrate. There are exceptions to this such as with fuel reuse between thermal and fast reactors, and as with the fuel in variable spectrum machines. However, each of the exceptions does represent changing the environment, which does not quite evaluate the type of in situ value associated with essentially burned fuel in a fixed lattice situation. An informative in situ formulation can be derived for the hypothetical situation of extending a reactivity limited fuel exposure by adding a minute amount of plutonium corresponding to the composition and location of the plutonium in the otherwise spent fuel.

Stated formally, the premature discharge price of the plutonium in reactor after an exposure E is the unit price assigned to the plutonium that will yield the same fuel cost whether: (1) the burn-up is terminated and the plutonium sold at the assigned price, or (2) the burn-up is terminated after an exposure $E + \triangle E$ and the plutonium sold for the assigned price. Equating the expressions that represent these alternatives will give an expression for the premature discharge price. For example, if the fuel cost can be considered as a function only of the uranium burn-up costs, the plutonium credit and the jacketing costs, then:

$$F = \frac{D_o - D_e - XP_e + C}{YE}$$

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where:

F = fuel cost, mills/kwh_e

D = uranium price $\$/cm^3$

P = plutonium concentration, $gram/cm^3$

C = jacketing costs, $\$/cm^3$

X = plutonium price, \$/gm

E = exposure, MWD/Ton

K = constant

o = initial conditions

e = condition at exposure E

Equating this to a similar expression for the fuel cost $E + \triangle E$ and considering the limit as $\triangle E \longrightarrow 0$ gives an expression that, for a given burn-up situation and economic environment, is a function only of the exposure E. This is:

$$X = \frac{D_o + C - D_e + D \frac{dD_e}{dE}}{P_e - E \frac{dP_e}{dE}}$$

Prices have been calculated with this formula by constructing polynomial expressions for the isotopic concentrations during burn-up. (This technique permits X to be calculated as a continuous function of exposure.) Comparisons of the prices obtained at the reactivity limited exposure with the recycle values (i.e., PUVE 2 step values) show that the premature discharge prices are always substantially higher. Since this particular formulation is for vanishingly small $\triangle E$'s, data are also being calculated for consideration of premature discharge substantially before the planned date.

Salt Cycle Economics

Several additional debug runs were made using the Salt Cycle Economics code to check out various options in the code. On the basis of spot-check hand calculations, all features of the code appear to be functioning properly.

A detailed outline of the changes required in the previously prepared Conventional Reprocessing code for compatibility with the Salt Cycle code has been worked out, and the process of translating these changes into Fortran language is well along. With continued progress as represented by the recent results, this revised code should be ready for use by the end of next month.

MISCELLANEOUS

Radioisotope Studies

In reviewing the radioisotopic heat sources which may most likely be alternates for Pu-238, Pm-147 appears to be the only candidate with reasonable expectations for recovery in the near future. Although its half-life is much less (2.5 years), it is believed that it will be adequate for many applications now considered limited to Pu-238. Although the gamma emitting contaminant Pm-148 is a problem, substantial sources of Pm-147 aged sufficiently to essentially eliminate this objectionable isotope are at hand at Hanford.

Hanford Science Colloquium

Professor Francis Birch, Harvard University geophysicist, will address the Hanford Science Colloquium on August 29 on the subject, "The Internal Constitution of the Earth."

Professor Harold C. Urey will address the Hanford Science Colloquium on October 2 on the subject, "The Problem of the Abundances of the Elements."

W. K. Woods

Manager,
Programming

WK Woods:jm

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF JULY, 1962

A. ORGANIZATION AND PERSONNEL

Changes in the work force included the transfer of Robert O. Budd from Missile and Space Vehicle Department into External Dosimetry, and assignment of Rose Marie V. Allen (new-hire) to Internal Dosimetry. Shirley B. Bridge resigned from the Company.

B. ACTIVITIES

Occupational Exposure Experience

One new case of plutonium deposition was confirmed by bioassay analyses during July. The total number of plutonium deposition cases that have occurred at Hanford is 292, of which 210 are currently employed.

The new deposition case resulted from a plutonium oxide contaminated injury received by a CPD process operator in the 234-5 Building. The injury occurred when a plutonium casting was unintentionally flung from a lathe and sliced through the hood glove and a surgical glove on the man's hand causing the injury. After two excisions, the plutonium in the wound was reduced to about one-tenth of the maximum permissible body burden. A total of about 1.9 μc Pu was removed from the injury by both excisions (the maximum permissible body burden for Pu^{239} is .04 μc). It is not yet possible to provide a reliable estimate of the magnitude of the total plutonium deposition in the body because of the effect of the administered treatment with DTPA on excretion rates of plutonium.

Two additional CPD employees experienced plutonium contaminated wounds. Examination with the wound counter revealed 0.026 μc and 1×10^{-3} μc at the wound sites. In the first case, excision of tissue was successful in removing the contamination from the wound. Local skin decontamination was effective in the second case.

There were 12 incidents at the 234-5 Building and 4 incidents in HLO facilities which required special plutonium bioassay sampling for 25 potentially involved employees.

Two employees were exposed momentarily to high gamma dose rates on the front face work platform at the 105-B reactor during the charging of a poison column control facility tube with the reactor operating. The charge machine broke free from the ball valve assembly and allowed reactor cooling water to flush several poison and aluminum pieces from the tube. Although the men left the work platform immediately, evaluation of the dosimeters which they were wearing indicated gamma doses of 0.78 r and 0.19 r. The exposure was believed to have resulted from three aluminum pieces that had been irradiated about three minutes before the backflow occurred.

Autoradiographic examination of a radioactive particle detected on the foot of a CE&UO plant laundry employee indicated the average dose rate to one square centimeter of skin area was about 18 rads/hour. Based on the five hour maximum estimated time of exposure, the localized dose to the skin of a small area on the ankle could have been as high as 90 rads.

Radiation beams to 1.5 r/hour and extremity doses to 3.2 rads/hour were encountered during the extended outage at PRTR. Use of lead-impregnated gloves was effective in reducing the extremity exposures 90 percent.

Activated sodium silicate was accidentally spread over major floor areas in the load-out facility, storage basin area, and change room at PRTR resulting in floor contamination to 750 mrads/hour. During the incident, two cases of nasal and facial contamination (maximum 30,000 c/m) and seven cases of shoe contamination (maximum 1000 c/m) occurred. Examination in the Whole Body Counter indicated no internal deposition occurred.

Environmental Experience

Levels of I^{131} in local milk returned to normal values of 3 to 4 $\mu\mu c/l$ in early July, following the sharp rise noted last month (maximum of 68 $\mu\mu c/l$ on about June 20, 1962). Concentrations of fallout materials in air filter samples for the Pacific Northwest decreased steadily in July. The monthly average was 3 $\mu\mu c/m^3$ noted in June.

A total of 112 fish was taken from Columbia River sampling stations at Priest Rapids, Hanford, Ringold, Richland, Burbank and McNary Dam.

The following produce samples were collected: 55 sets of beef thyroids; 6 pounds of Willapa Bay oysters; 16 samples of pasture grass; 120 gallons of milk; and 30 samples of 18 varieties of vegetables and fruits.

One aerial monitoring flight was made as part of the background radiation study.

Studies and Improvements

All of the 30,100 new personnel dosimeters were shipped by the vendor. Transportation trays for these dosimeters were also fabricated and readied for routine use. The first 8000 dosimeters were assembled and audited in preparation for the August 10, 1962, issue date.

The ad hoc committee that is preparing the final dosimetry report of the Recuplex incident issued a second draft of the document. This report will be used as an appendix of the main investigating report on the incident.

The 0.075" silicon diode neutron dosimeters were irradiated with 0.4 and 4.0 Mev neutrons to compare the constant voltage and constant current readout systems. Three constant currents and five constant voltages were used in the study. At constant current, the sensitivity of the diodes varies between 0.1 and 0.3 percent change in current per rad. At constant voltage, the sensitivity varied from 0.2 to 0.75 percent change in voltage per rad. The observed variations were dependent upon the particular diode and the voltage or current used. Based on this study, it would be possible to detect a dose of 0.2 rads of fast neutrons with these particular p-type silicon diodes.

Equipment and methods for conducting gamma energy studies at field locations were assembled and tested during the month. A review of gamma radiation exposure experience at Hanford indicates that gamma energy studies at the production reactors, Purex, Redox, and Buildings 234-5, 231-Z, 308, 327 and 326 should be included in the study. A large fraction of the total Hanford gamma radiation exposure is obtained in these buildings.

Several special Columbia River measurements were performed last month during and subsequent to the simultaneous outage of all the production reactors. No change was detected in the Columbia River temperature at 300 Area or Pasco over the temperatures noted while the reactors were operating. Forecasts of river travel time were confirmed to be reasonably correct by checking against the actual flow time of reactor effluents.

An instrument for measuring the Columbia River elevation was installed at the PRTR intake to measure fluctuations in the river level. This device coupled with Priest Rapids river flow data should give us additional information on river flow and the times during which the McNary Pool extends to the 300 Area.

A final review of the CPD's hazard analysis was completed and the document is now being issued.

The Fuels Recycle Pilot Plant (FRPP) design guides for stack emission rates were reviewed and revised. Included were the upper gaseous emission rates for Natural Uranium, Pu^{239}, unknown beta-gamma mixtures, Sr^{89}, Sr^{90} and I^{131}.

A nuclear safety audit of the Calibrations Operation, Radiation Protection Operation, was performed on July 31, 1962. The report of a nuclear safety audit of the Experimental Reactors Operation, Physics and Instrument Research and Development Operation was issued.

Three HV-70 air sample filters and one membrane air sample filter were autoradiographed for a week to provide an estimate of the number of radioactive particles contained. The three HV-70 air sample filters were used in the 231 Building during a plutonium carbide air contamination incident. The membrane filter was used in collecting air from a plutonium carbide powder loading hood. Counting in an alpha-beta-gamma air sample counter prior to the autoradiographing revealed no detectable beta-gamma activity. The radiograph of the filters appeared to indicate a significant difference in size between the particles collected in the room during the contamination incident and the particles collected in the hood. The majority of the particles sampled from the hood were two to five times as large as the ones sampled in the room. In addition, the room filter showed an overall haze on the radiograph which would appear that the contamination was very finely divided material.

C. VISITS AND VISITORS

Visitors consulting with members of the Radiation Protection Operation staff during the month included:

D. Wagstaff- - - Oregon State Board of Health, Salem, Oregon
G. Barr- - - - - U. S. Public Health Service, Portland, Oregon
M. Mattys- - - - Euratom
B. G. Lindberg - U. S. Atomic Energy Commission, Washington, D. C.
M. Lammering - - U. S. Public Health Service, Cincinnati, Ohio
W. F. Marlow - - U. S. Atomic Energy Commission, Washington, D. C.

Visitors who toured the Whole Body Counter and the Bioassay Laboratory during the month included:

6 AEC Management trainees

Members of the Radiation Protection Operation visiting off-site during the month included:

G. E. Backman - Attended 1962 AEC Emergency Radiation Monitoring Training Course at Sandia Base, Albuquerque, New Mexico
E. C. Watson - Consulted with J. R. Horan and staff at the Idaho Operations Office, U.S.A.E.C., Idaho Falls, Idaho

D. RELATIONS

Six suggestions were submitted by personnel of the Radiation Protection Operation during July. Two suggestions were adopted; five were rejected. Two suggestions are pending evaluation.

Six non-employees were examined in the Whole Body Counter during July. Three were housewives, two were children, and one was an employee of the Atomic Energy Commission.

Radiation orientations were presented to 26 laboratory employees. A lecture covering the use, function and interpretation of emergency monitoring equipment was presented to 300 Area firemen. A one-hour talk was presented to members of the Region 8 AEC Off-site Radiological Monitoring Assistance Team. Fifty-seven persons attended the Disaster Level Monitoring course bringing the total to 179 General Electric and AEC employees that have attended the course. Approximately 95 percent of the 300 Area radiation monitoring personnel have attended the Emergency Monitoring classes.

E. SIGNIFICANT REPORTS

HW-72691-6 - "Summary of Radiological Data for the Month of June, 1962", by R. F. Foster.

HW-58312REV- "Columbia River Flow Time Calculation", by J. K. Soldat.

HW-74518 - - "Monthly Report for July 1962, Radiation Monitoring Operation", by A. J. Stevens.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

	July	1962 to Date
External Exposure Above Permissible Limits		
Whole Body Penetrating	0	3
Whole Body Skin	0	3
Extremity	0	2
Hanford Pocket Dosimeters		
Dosimeters Processed	2,262	21,500
Paired Results - 100-280 mr	4	62
Paired Results - Over 280 mr	0	8
Lost Results	0	0
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	10,293	67,439
Results - 100-300 mrads	315	2,239
Results - 300-500 mrads	28	208
Results - Over 500 mrads	10	81
Lost Results	29	170
Average Dose Per Film Packet - mrad (ow)	12.07	12.73
- mr (s)	21.92	27.18
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	734	9,716
Results - 50-100 mrem	0	9
Results - 100-300 mrem	0	29
Results - Over 300 mrem	0	2
Lost Results	17	35
Fast Neutron		
Film Processed	214	2,485
Results - 50-100 mrem	19	342
Results - 100-300 mrem	73	419
Results - Over 300 mrem	0	11
Lost Results	15	15
Hand Checks		
Checks Taken - Alpha	35,677	219,542
- Beta-Gamma	52,391	368,206
Skin Contamination		
Plutonium	36	158
Fission Products	52	318
Uranium	1	12
Tritium	0	0

Whole Body Counter	Male	Female	July	1962 to Date
GE Employees				
Routine	33	4	37	114
Special	17	0	17	153
Terminal	19	1	20	73
Non-Routine	15	2	17	180
Non-Employees	1	5	6	19
Pre-Employment	1	4	5	8
	86	16	102	547

Bioassay	July	1962 to Date
Confirmed Plutonium Deposition Cases	1	9*
Plutonium - Samples Assayed	99	2,627
- Results Above 2.2×10^{-8} μc/sample	13	121
Fission Product - Samples Assayed	61	3,275
- Results Above 3.1×10^{-5} μc/sample	0	15
Uranium - Samples Assayed	0	1,070
Biological - Samples Assayed	11	244
Strontium - Samples Assayed	0	299

Uranium Analyses

	Following Exposure Units of 10^{-9} μc U/cc			Following Period of No Exposure Units of 10^{-9} μc U/cc		
Sample Description	Maximum	Average	Number Samples	Maximum	Average	Number Samples
Fuels Preparation						
Fuels Preparation**						
Hanford Laboratories						
Hanford Laboratories**						
Chemical Processing						
Chemical Processing**						
Special Incidents						
Random						

URANIUM INPUT DID NOT PROCESS. RE-SCHEDULED FOR NEXT DATA PROCESSING.

Tritium Samples	Maximum	Count	Total
Urine Samples			
> 5.0 μc/l	71	211	
< 1.0 μc/l		14	
Samples Assayed			260
D_2O Samples			
Moderator	720.6 μc/ml	10	
Primary Coolant	249.6 μc/ml	4	
Reflector	798.5 μc/ml	10	
			24
Other Water Samples			
No. 3018-37-08 299 W 22-14	0.1887 μc/ml		142
			426

* The total number of plutonium deposition cases which have occurred at Hanford is now 292, of which 210 are currently employed.

**Samples taken prior to and after a specific job during work week.

Calibrations	Number of Units Calibrated July	Number of Units Calibrated 1962 to Date
Portable Instruments		
CP Meter	972	7,032
Juno	253	1,910
GM	544	3,892
Other	137	1,336
Audits	102	728
	2,008	14,898
Personnel Meters		
Badge Film	2,484	11,896
Pencils	-	12,670
Other	260	2,817
	2,744	27,383
Miscellaneous Special Services	1,411	7,279
Total Number of Calibrations	6,163	49,560

ARKeene
Manager
RADIATION PROTECTION

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FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

The FY 1963 control budget used in July reporting was the interim financial plan received from HOO-AEC the first of the month. Details of the financial plan were made available to Hanford Laboratories' management, program levels reviewed, and comments transmitted to Contract Accounting. The tentative operating cost budget for FY 1963 totals $28,049,000. Tentative control budgets for capital equipment by program are as follows:

02 Program	$1 600 000
03 Program	100 000
04 Program	1 200 000
05 Program	115 000
06 Program	262 000
Total	$3 277 000

A proposed allocation of the financial plan by the General Manager-HAPO to Hanford Laboratories follows:

1. All funds for 04, 05, and 06 Programs were allocated to Hanford Laboratories.

2. 02 Program research and development funds allocated to Hanford Laboratories are detailed below:

	Thousands of Dollars
Plutonium Nuclear Safety	$ 375
Columbia River Studies	109
Separations Development	350
Metallurgical Development	135
Reactor Technology Development	100
Total	$1 069

3. The 03 Program research and development allocation to Hanford Laboratories is $175,000.

As an interim authorization, $80,000 of the $250,000 total authorization for FY 1963 research and development support for Project Whitney has been received from UCLRL. No accompanying authorization for capital equipment purchased has been received. An authorization of $100,000 was received to cover Project Whitney fabrication work during the six-month period ending December 31, 1962.

Funds allocated by the HAPO General Manager to the Laboratories for attendance at professional and trade societies and at off-site courses and seminars in FY 1963 amount to $53,500 and $4,400, respectively. The professional and trade society funds were in turn allocated to Hanford Laboratories' sections.

Special requests established during the month are listed below:

Accounting Code	Activity
.3D	Development of dome-type pressure tube closure for USAEC/AECL cooperative program. Total estimated cost is $20,000 and the program should be completed in FY 1963.
.3E	Boiling burnout without a wire wrap for the USAEC/AECL cooperative program. FY 1963 funds authorized amount to $75,000.
.3F	Fog cooling of fuel elements for the USAEC/AECL cooperative program. FY 1963 authorized amount is $100,000.
.6B	R. J. Sloat - Participation at APED in an evaluation of the engineering aspects of the development of processing UF_6 to UO_2.
.6C	Packaging and handling charges for shipment of an excess tensile testing machine to University of California, Lawrence Radiation Laboratory at Livermore, California. Estimated cost - $350.
.6D	S. H. Bush - Participation in a continuing review of the technical audit of superheat problems at APED.
.2A	Argonne National Laboratory - Pyrophoricity Studies. Additional authorization - $20,000.

The following organizational code changes became effective July 1, 1962:

New Codes

7124 - Visual Displays
7626 - Waste Calcination Demonstration

Cancelled Codes

7123 - Advanced Engineering Courses
7523 - Shielding Functions

End function program code changes during the month were:

Code	Title	Remarks
.83	Mechanism of Graphite Damage	Program cancelled.
.72	Columbia River Studies	New research and development program sponsored by the Division of Production.
.73	Plutonium Nuclear Safety	Program sponsored by Division of Production; in FY 1962 was a portion of CPD sponsored funds.
.74	Reactor Technology Development	Title changed from Pile Technology Development.
.19	Plutonium Ceramics	Title changed from Plutonium Utilization.

Outstanding routine work orders were reviewed with Laboratories' field personnel. Required cancellations, code revisions, and reissuances have been completed.

Preparation of a budget booklet for use by Hanford Laboratories' management is expected to be completed in the near future.

General Accounting

Following is a summary of the status of letters or agreements covering specific actions requiring AEC concurrence:

AT-244	Participation in Wallowa County Educational Day Camp	Returned unapproved July 13, 1962
AT-246	Special Science Seminars	Approved June 8, 1962 (Not May 23 as reported last month)
AT-247	Participation in Standardizing Activities	Approved May 29, 1962

AT-256	Participation in Standardizing Activities	In Process
AT-252	Miniature Swine for Colorado State University	To AEC August 6, 1962

Agreement No. AT-6 requests from the Commission follow:

Training of Euratom Health Physicist at Hanford

Fabrication of Fuel Element Plates for University of Washington

Development of Dome-Type Pressure Tube Closure Seal for USAEC/AECL Cooperative Program

With the cooperation of representatives from each section, information on committees and group activities of Hanford Laboratories' personnel was compiled and forwarded to Counsel's office for review. This is part of a study, still in progress, to determine the status of these activities in relation to President Kennedy's Executive Order 11007 on Government Advisory Committees.

Total investment in Hanford Laboratories' Plant and Equipment in Service at July 1, 1962 amounted to $69.6 million compared to $67.3 million at July 1, 1961. The net increase of $2.3 million includes equipment valued at $2.4 million transferred from Work In Progress Accounts, $.7 million for equipment transferred from University of California - Lawrence Radiation Laboratory, $3.7 million representing major projects such as Structural Material Irradiation Testing Equipment - ETR, Modifications and Additions to High Pressure Heat Transfer Apparatus - 189-D Building, additional billings to PRTR and PFPP and other projects of lesser magnitude. The major offsetting entry during FY 1962 was the transfer of Hot Semi-Works ($4.5 million) to CPD.

The following is a summary of Hanford Laboratories' plant investment at July 1, 1962.

(Amounts in Thousands)

Building and Structures	$27 146
Equipment	42 329
Utilities	22
Improvement to Land	88
	$69 585

At June 30, 1962, Hanford Laboratories had 17,828 individually controlled items on record, including both movable and fixed plant and equipment.

The Laboratory Equipment Pool has been requested to provide storage space for approximately 25 tons of structural materials. This material will be used by the Physical Metallurgy Operation on the studies of irradiation effects on reactor structural materials sponsored by the Division of Reactor Development. Nine research laboratories are participating in this program with Hanford Laboratories acting as the coordinator. Hanford Laboratories' responsibilities include the procurement, storage, and disbursement of structural materials of general program interest to each participating site. Approximately ten alloys in various shapes and sizes are expected on-site within the next two months. The Pool presently contains approximately 35 tons of Hastalloy also to be used in this program. The Laboratory Pool facility is expected to provide suitable accountability procedures including records which will show weight, heat number and piece number.

The physical inventory of movable catalogued equipment in the custody of Reactor and Fuels Research and Development Operation is progressing on schedule. The inventory count started on July 5, 1962 and will continue through October 1962.

Hanford Laboratories investment in materials at July 1, 1962 totaled $25.2 million as shown below:

(Amounts in Thousands)

SS Material	$24 148
Reactor and Other Special Materials	744
Spare Parts	324 -1)
	$25 216

(1- Includes a reserve established at July 1, 1962 of $81,062.

Total HAPO Nuclear Materials Consumed in Research for the fiscal year ended June 30, 1962, is $4.2 million of which $3.7 million is applicable to Hanford Laboratories and $.5 million to Fuels Preparation Department. The increase of $1.7 million applicable to Hanford Laboratories resulted primarily from devaluation of research and development material assigned to the PRTR. A detail of the Hanford Laboratories' portion by program shows:

(Amounts in Millions)

2000 Program	$1.1
3000 Program	.2
4000 Program	2.4
	$3.7

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The transfer of certain plant facilities from Fuels Preparation Department to Hanford Laboratories included transfer of 550 movable capital equipment items valued at approximately $230,000 in July. Remaining equipment will be transferred during August.

Based on an analysis made by the Savannah River Operations Office of heavy water shipped there during May 1962, an adjustment charge of $6,180 was made to operating cost during the month of July. Our records show 20,908 pounds shipped, whereas SROO analysis indicates 20,567 pounds received - a variance of 342 pounds.

A summary of Laboratory Equipment and Material Pool activity for the month ending July 31, 1962 follows:

Equipment	Quantity	Value
Items received	60	$ 28 572
Withdrawals by custodians	15	6 339
Equipment reassigned (purchased eliminated)	25	14 701
Total equipment items on hand at July 31, 1962	1 112	574 110 -1)

(1- Includes 140 items on loan valued at $62,999.

The inventory of materials in the Laboratory Pool at month end was comprised of the following:

Material	Quantity	Value
Beryllium	1 035 grams	$ 592
Gold	2 182 grams	2 924
Palladium	2 224 grams	2 535
Platinum	1 896 grams	5 537
Clean scrap	10 grams	29
Contaminated scrap	6 703 grams	19 573
Silver	6 633 grams	463
Hafnium	2 939 grams	1 499
Zirconium	5 443 lbs.	107 575
		140 727
All other material held for the convenience of others		148 838
Total value of material held		$289 565

During the month 18 Zircaloy-2 ingots (3,886 pounds) were shipped to Wolverine Tube Company for use on IPD P.O. H2K-33584. The dollar value ($11,658) was backcharged to create an accounts receivable item for billing to Wolverine Tube Company on their P.O. 9085-D. In addition 10 excess Zircaloy-3 ingots (5,862 pounds) and 14,249 pounds of zirconium scrap accumulated at the Laboratory Pool were forwarded to Central Stores for disposition.

During FY 1962 Reactor and Other Special Materials received at the Laboratory Pool totaled $361,325 while $269,238 worth of such materials was disbursed during the same period.

Action as indicated occurred on the following projects during the month:

New Money Authorized Hanford Laboratories

CAH-916	Fuels Recycle Pilot Plant	$40 000
CAH-958	Plutonium Fuels Testing and Evaluation Laboratory, 308 Building	2 000
CAH-962	Low Level Radiochemistry Building	31 000

The following contracts were processed during the month:

SA-227	Bowen Engineering Incorporated
CA-339	E. C. Lingafelter
DDR-157	Battelle Memorial Institute
DDR-156	Armour Research Foundation
CA-343	The Sheffield Corporation
CA-342	American Machine & Foundry Co., AMF Atomics Division
DDR-152	Oregon Metallurgical Corporation
CA-344	H. A. Laitinen
DDR-138	Battelle Memorial Institute Supp. No. 1
CA-345	J. K. Minasian

OPGs issued in July are listed below:

OPG No.	Title
3.4.11	Reduction of Force--Weekly Salaried Employees
7.10	Weapon Data
11.2	Numerical Index
22.1.5	Biology Operation

Functional OPGs for all level 3 managers and specialists with the exception of Physics and Instrument Research and Development were forwarded to the Manager - Hanford Laboratories for approval.

As of the first of fiscal year 1963 there were 130 off-site contracts, including consulting agreements, in force at Hanford. The distribution by sponsoring component is tabulated below.

Component	Number of Contracts	% of Total
HLO	52	40
CE&UO	30	23
IPD	23	18
R&OHO	9	7
FPD	7	5
C&AO	5	4
CPD	4	3
Total	130	100

Personnel Accounting

Number of Hanford Laboratories Employees

Changes During Month	Total	Exempt	Nonexempt
Employees on payroll at beginning of month	1 512	699	813
Additions and transfers in	24	14	10
Removals and transfers out	20	10	10
Employees on payroll at end of month	1 516	703	813

Overtime Payments During Month	July	June
Exempt	$ 6 695	$ 7 239
Nonexempt	22 392	33 932
Total	$29 087	$41 171

Gross Payroll Paid During Month		
Exempt	$ 667 441	$ 640 067
Nonexempt	436 251	560 622
Total	$1 103 692	$1 200 689

Participation in Employee Benefit Plans at Month End	July		June	
	Number	Percent	Number	Percent
Pension	1 332	99.1	1 323	99.1
Insurance Plan - Personal	376		378	
- Dependent	1 099	99.9	1 093	99.7
U. S. Savings Bonds				
Stock Bonus Plan	88	38.1	90	38.6
Savings and Security Plan	1 133	88.3	1 112	89.0
Savings Plan	72	4.7	74	4.9
Good Neighbor Fund	989	65.2	991	65.5

Insurance Claims				
Employee Benefits	Number	Amount	Number	Amount
Life Insurance	1	$23 129	0	$ 0
Weekly Sickness & Accident	7	820	6	501
Comprehensive Medical	32	2 987	29	2 139
Dependent Benefits				
Comprehensive Medical	90	7 487	81	6 763
Total	130	$34 423	116	$9 403

TECHNICAL ADMINISTRATION

Employee Relations

Sixteen nonexempt employment requisitions were filled during June with thirteen remaining to be filled.

Two PBM classes and one BOCE class will be conducted this fall. Class members have been named and all arrangements completed except selection of instructors.

Professional Placement

Advanced Degree - Four Ph.D. applicants visited HAPO for employment interviews. Four offers were extended; four acceptances and nine rejections were received. Current open offers total five.

BS/MS - During the month eight direct placement offers were extended; seven acceptances and two rejections were received. Five program offers were made. At month's end, four direct placement and eight program offers were open.

Technical Graduate Program - Six Technical Graduates were placed on permanent assignment; eight members were added to the roll and two terminated. Current program members total 56.

Technical Information

The first quarterly report for the Division of Reactor Development covering January-March 1962 was completed.

ECONOMIC EVALUATIONS

A comprehensive analysis was made of Part IV, the complete cost structure of H. E. Hanthorn's fuel fabrication plant study which is scheduled to be issued as "Calculated Cost of Fabrication of Plutonium Enriched Fuel Elements." Based on 10-12 statistical guidelines derived from other manufacturing businesses, especially composite Metals Fabrication Industry, the hypothetical fabrication plants and costs appear reasonable, provided the suppositional yield ratios and consequent direct labor man power requirements are realistic.

There was one significant omission in the calculations which when corrected increased the amount of return on investment. Other criticisms of the cost data are of a minor and compensating nature.

A complete cost recast from functional to fixed and variable classifications was completed for the 19-rod U and Pu enriched plants as a base case for Programming Operation's fuel element fabrication computer code. This code is designed to analyze the effect on unit production costs of significant variations in yield and recycle rates and in labor and equipment costs.

The differences in cost of working capital between leased and owned nuclear material during the irradiation period were investigated with R. W. McKee in order to resolve logical decisions for his comparative economics computer program. Examples of the cash flow and consequent average inventory investment were developed in order to complete equations for calculating this important cost.

Additional progress was made on the middle portion of a major report on "Nuclear Cost Estimating and Electric Utility Economics."

PROCEDURES

Hanford Laboratories' secretaries were informed by letter that a two-week sampling of 300 Area duplicating orders indicated that little thought is being given to the cost and work saving benefits of reproducing on both sides of each sheet of copy paper. Their consideration of this and other office procedure improvements was encouraged.

FACILITIES ENGINEERING

Projects

At month end Facilities Engineering Operation was responsible for 13 active projects having total authorized funds in the amount of $2,744,600. The total estimated cost of these projects is $9,117,000. Expenditures on these projects through June 30, 1962 were $1,484,000.

The following summarizes project activity in July:

Number of authorized projects at month's end ---------------- 13

Number of new projects authorized -------------------------- 0

Projects completed -- 0

New projects submitted to AEC ------------------------------ 1
CAH-977, Facility for Radioactive Particle Inhalation Studies

New Projects awaiting AEC authorization -------------------- 1
CAH-977, Facility for Radioactive Particle Inhalation Studies

Project proposals complete or nearing completion ----------- 5
Addition to Radionuclide Facilities
307 Basin Expansion
Additions to the 222-U Building
Neutron Calibration Facility - 3745-A Building
Graphite Machine Shop

Pages appended to this report provide detailed project status information.

Services

Satisfactory progress was made in the engineering services provided on the following jobs:

Proposed 300 Area analog simulation facility
Split-half machine
Controlled environment facility - 108-F Building

Pressure system assistance was provided on:

Operating procedures for autoclaves in C-25 Building
A review of the Laboratories Pressure System Manual
The Tube Burst Test Facility vessel

Technical Shops code compliance problems
The testing of the visible fuel rupture autoclave
The design of a high temperature uranium test furnace
Third Party inspection of five PRTR vessels

Plant Engineering effort was expended on:

308 Ventilation system instrumentation modification
325 Ceramic Fuels Area air conditioner
309 M & M Building office addition
325 Second floor office addition survey
3760 Northwest corner conference room noise diffusers
325 Analytical laboratory glove box fabrication
325 Laboratory vacuum pump replacement
325 Decontamination facility installation
321-A Electrical load survey
308 Emergency power tie-in
327 Fifteen ton crane operability
327 Critical incident alarm planning
108-F Transformer addition
100-F Animal farm expansion and relocation of electrical lines and substations
231-Z Fire, criticality and evacuation alarm standardization

Maintenance and Operation

Costs for June were $151,526 bringing FY 1962 costs to a total of $1,751,925 or 98% of budget. The construction strike together with a significant refund from Fuels Preparation Department for previous overliquidations accounted for most of the underrun. Despite the strike, improvement maintenance costs were $20,456.

The following tabulation summarizes waste disposal operations:

	June	May
Concrete Barrels	4	3
Loadluggers	1	4
Crib Waste (gallons)	300,000	240,000

Drafting

The equivalent of 158 drawings were produced during the month for an average of 21.6 man-hours per drawing.

Major jobs in progress are: 280 T extrusion press, PRTR "as-builts," new PRTR seven-rod fuel element, PRTR shim-rod control, electrical resistivity sample holder, scintillation scanner housing, cladding cutter assembly for PRTR, remote welding chamber, Mark 1-H fuel element end bracket, and PRTR borescope.

Construction

There were 97 existing J. A. Jones Company orders at the beginning of the month with a total unexpended balance of $175,719. One hundred and twenty-seven new orders, five supplements and adjustments for underruns amounted to $72,047. Expenditures during the month were $65,304. Total J. A. Jones backlog at month's end was $182,462.

	Hanford Laboratories Unexpended Balance
Orders outstanding beginning of month	$175 719
Issued during the month (inc. sup. & adj.)	72 047
J. A. Jones Expenditures during month (inc. C.O. costs)	65 304
Balance at month's end	182 462
Orders closed during month	148 896

Maintenance W.O. total 7 - Face Value $23,000.

Construction and maintenance activities completed during July included:

141-C/141-F Install desert coolers
309 Electrical and mechanical modifications to rupture loop
309 Modify PRTR storage basin crane
325 Reroute waste line; install pump and hold-up tank
325 Install electrical service and base work for calciner
305-B Install air conditioner

W. Sale
Manager
Finance and Administration

W Sale:whm

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAH-822	Pressurized Gas Cooled Loop Facility	4141 Operating

AUTHORIZED FUNDS			COST & COMM. TO 7-15-62	ESTIMATED TOTAL COST
$ 1,170,000	DESIGN $ 43,000 / CONST. $ 1,127,000	AEC $ 15,000 / GE $ 1,155,000	$ 1,131,131	$ 1,170,000

STARTING DATES	DATE AUTHORIZED	DIR. COMP. DATE	EST'D. COMPL. DATES
DESIGN 8-19-59	2-2-62*	6-30-62	DESIGN 4-29-60
CONST. 10-17-60			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	100	91
PF	1.4	100	0
CPFF	22.1	100	99
FP	6.6	100	100
Govt.Eq.	69.9	100	90

SCOPE, PURPOSE, STATUS & PROGRESS

Struthers Wells' estimate for heater completion by August 3, 1962, appears firm.

Project Proposal CAH-822, Revision V, has been forwarded to HOO-AEC for approval. This revision requests extension of completion date to 12-31-62 with no increase in project funds.

First gas bearing blower scheduled for test on July 16, 1962. Earliest completion of both units is still 8-15-62.

*Initial authorization date was December 18, 1958.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS			COST & COMM. TO	ESTIMATED TOTAL COST
$	DESIGN $ / CONST. $	AEC $ / GE $	$	$

STARTING DATES	DATE AUTHORIZED	DIR. COMP. DATE	EST'D. COMPL. DATES
DESIGN			DESIGN
CONST.			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

1231071

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CGH-857	Physical & Mechanical Properties Testing Cell - 327 Building	0290

AUTHORIZED FUNDS			COST & COMM TO 7-15-62	$ 296,066
$ 460,000	DESIGN $ 45,000	AEC $ --	ESTIMATED TOTAL COST	$ 460,000
	CONST. $ 415,000	GE $ 460,000		

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 11-2-59	DATE AUTHORIZED 9-22-61*		DESIGN 3-15-61	
CONST. 2-12-62	DIR. COMP. DATE 12-15-62		CONST. 12-15-62	

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		34
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		833
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	2#	2
PF			
CPFF	18	9#	9
FP			
Equip.	82	0	0

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 - $55,000. Detailed design started 4-1-60. Procurement and construction authorized 9-22-61.

Basement floor and foundation concrete work is completed. Construction has stopped until cell assembly is delivered.

Number of purchase orders required	19 Value (Est.)	$253,000**
Number of purchase orders placed	19 Value	203,789

Construction is planned to resume 7-27-62 based on agreement with the U. S. Steel Company that delivery of the cell assembly will be made 9-1-62, f.o.b. Los Angeles, except for the south panel, which will come 9-21-62.

* Original authorization for design was October 1, 1959.
**Includes delivery charges, inspection and contingency.

\# A revision to the schedule of construction is being prepared, reflecting the late cell delivery.

1231072

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAH-866	Shielded Analytical Laboratory - 325-B Building	61-a-1

AUTHORIZED FUNDS $ 700,000	DESIGN $ 60,000	AEC $ 546,500
	CONST. $ 640,000	GE $ 153,500
COST & COMM TO 7-15-62	$ 135,839 (GE	
ESTIMATED TOTAL COST	$ 655,000	

STARTING DATES	DESIGN 9-5-59	DATE AUTHORIZED 5-31-60*	EST'D. COMPL. DATES 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	10	1994
COST PLUS FIXED FEE		
PLANT FORCES	2	5
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	1	

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	86
PF	3	1	1
CPFF	2	0	0
FP	95	100	87

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.

Directive No. HW-495, Mod. No. 4, dated July 10, 1962 changed the project completion date to November 15, 1962. A Work Authority is expected increasing General Electric's authorization on Field Engineering by $7,000.

Ceilings and walls were plastered where required.

Cell partitions and window installations were completed.

Ventilation tie-ins were made and electrical building tie-in is schedule for July 28, 1962.

* Original authorization for preliminary design was August 12, 1959.

1231073

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAH-867	Fuel Element Rupture Test Loop	58-e-15

AUTHORIZED FUNDS			COST & COMM. TO 7-15-62	$ 556,396 (GE)
$ 1,500,000	DESIGN $ 130,000	AEC $ 820,000	ESTIMATED TOTAL COST	$ 1,500,000
	CONST. $ 1,370,000	GE $ 680,000		

STARTING DATES		DATE AUTHORIZED 6-24-60*	EST'D. COMPL. DATES	
DESIGN	8-1-60		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	5	2455
COST PLUS FIXED FEE		
PLANT FORCES	10	2290
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	9	100	100
CONST.	100	100	99
PF	2	100	50
CPFF	57	100	98
FP (1)	10	100	100
(2)	31	100	99

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 produce release.

A revised project proposal has been approved which extends the project completion date to October 31, 1962.

Loop design tests and filter plant acceptance tests are in progress.

* Initial authorization was on 10-1-59.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS			COST & COMM. TO	$
$	DESIGN $	AEC $	ESTIMATED TOTAL COST	$
	CONST. $	GE $		

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

1231074

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAH-888	Biology Laboratory Improvements	60-h-1

AUTHORIZED FUNDS			COST & COMM. TO 7-15-62	$ 60,356 (GE)
$ 420,000	DESIGN $ 44,000	AEC $ 359,500	ESTIMATED TOTAL COST	$ 420,000
	CONST. $ 376,000	GE $ 60,500		

STARTING DATES			EST'D. COMPL. DATES	
DESIGN	8-8-60	DATE AUTHORIZED 4-18-61*	DESIGN	3-31-61
CONST.	7-10-61	DIR. COMP. DATE 3-31-62	CONST.	8-1-62

ENGINEER

FEO - JP Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		2660
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	99
PF	1	100	100
CPFF	10	NS	90
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.

The 100 Area shutdowns caused work stoppage on installation of the Radiation Handling Facilities. Checking out and tieing in the console-instrument wiring is now nearing completion. The storage shield is in place and the control rod assembly has been hung from the ceiling.

The Lump Sum Contractor is in process of ordering vinyl floor covering for conference room and secretary's office. Minor Construction is to furnish an estimate of cost for replacing the exhaust fan motor sheave for reducing fan speed and also rebalancing first and third floor. This is to reduce the noise level.

The paint manufacturer has been requested to recommend and specify sealant and paint for pumice block walls to prevent moisture absorption. Completion of other exceptions are underway by CPFF Construction Service Contractor.

* Original authorization for design was May 3, 1960.

1231075

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAH-916	Fuels Recycle Pilot Plant	4-62-d-3

AUTHORIZED FUNDS			COST & COMM. TO 7-15-62	$ 424,800
$ 465,000	DESIGN $ 465,000	AEC $	ESTIMATED TOTAL COST	$ 5,450,000***
	CONST. $ -0-	GE $ 465,000		

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	3-15-61	6-29-62**	DESIGN	10-15-62
CONST.	9-15-62*	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		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	98	96
TITLE I	11	100	100
GE-TIT. II	89	98	95
AE-TIT. II	0		
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.

Design effort on the project has been reduced in the last month pending firming up of the scope criteria on the waste calcination proposal and to make changes to design to reduce the estimate to $5,000,000.

Of a total of 325 drawings, 322 have been issued for comment including 190 for approval. The specifications are 70% complete. These totals and the above per ent complete do not include the waste calcination work.

* 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.

1231076

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes	62-k

AUTHORIZED FUNDS			COST & COMM. TO 7-15-62	
$ 29,600	DESIGN $ 29,600	AEC $		$ 29,600
	CONST. $	GE $ 29,600	ESTIMATED TOTAL COST	$ 289,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN 11-7-61	CONST. 10-1-62	10-23-61	DESIGN 5-31-62	CONST. 10-1-63
		DIR. COMP. DATE - - -		

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER - Bovay Engineers		260
DESIGN ENGINEERING OPERATION		200
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 for construction funds has been forwarded to HOO-AEC.

A meeting is being requested with the Commission to resolve questions developed at the AEC Review Board meeting of 7-19-62.

PROJ. NO.	TITLE	FUNDING
CAH-927	Additions to the 271-CR Building Waste Treatment Demonstration Facility	61-J

AUTHORIZED FUNDS			COST & COMM. TO 7-15-62	
$ 92,000	DESIGN $ 11,000	AEC $ 76,300		$ 14,888 (GE)
	CONST. $ 81,000	GE $ 15,700	ESTIMATED TOTAL COST	$ 92,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN 6-15-61	CONST. 2-15-62	5-15-61	DESIGN 6-5-62	CONST. 10-1-62
		DIR. COMP. DATE 7-31-62		

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		80
COST PLUS FIXED FEE		110
PLANT FORCES		
ARCHITECT - ENGINEER		150
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II	100	100	100
CONST.	100	100	50
PF			
CPFF	33	100	90
FP	67	100	30

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides facilities for pilot plant development of decontamination processes for intermediate level chemical processing plant waste for safe discharge to the plant environs. Design was accomplished by the Bovay Engineers.

A revision No. 3 to the project proposal has been submitted, requesting extension of the completion date to 10-31-62 because of the strikes which started 5-16-62 and ended 7-23-62.

1231077

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-74522

DATE 7-31-62

PROJ. NO.	CAH-936
TITLE	Coolant Systems Development Laboratory 1706-KE Building Addition
FUNDING	62-k
AUTHORIZED FUNDS	$ 130,000
DESIGN $	9,000
CONST. $	121,000
AEC $	115,000
GE $	15,000
COST & COMM. TO 7-15-62	$ 14,809 (GE)
ESTIMATED TOTAL COST	$ 130,000
STARTING DATES — DESIGN	9-8-61
STARTING DATES — CONST.	5-1-62
DATE AUTHORIZED	4-5-62*
DIR. COMP. DATE	10-31-62
EST'D. COMPL. DATES — DESIGN	1-1-62
EST'D. COMPL. DATES — CONST.	11-30-62
ENGINEER	FEO - KA Clark

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	43	23
PF			
CPFF			
FP	100	43	23

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	3	230
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		166
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	.4	10

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides facilities for the conduct of corrosion and decontamination studies for nuclear reactor coolant systems, by the addition of 2,700 sq. ft. laboratory facility on the west side of the 1706-KE Building. Design was accomplished by the Bovay Engineers. Current estimate of Title I and II costs - $11,000.

An extension to the FP contract, of 33 days, has been granted because of the work stoppage due to the carpenters strike.

* Original authorization for design 8-9-61.

PROJ. NO.	CGH-951
TITLE	A-C Column Facility - 321 Building
FUNDING	0290
AUTHORIZED FUNDS	$ 55,000
DESIGN $	5,000
CONST. $	50,000
AEC $	-0-
GE $	55,000
COST & COMM. TO 7-15-62	$ 19,939
ESTIMATED TOTAL COST	$ 55,000
STARTING DATES — DESIGN	1-30-62
STARTING DATES — CONST.	3-25-62
DATE AUTHORIZED	1-12-62
DIR. COMP. DATE	10-31-62
EST'D. COMPL. DATES — DESIGN	4-1-62
EST'D. COMPL. DATES — CONST.	10-31-62
ENGINEER	FEO - OM Lyso

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II	0		
CONST.	100	55	55
PF	100	55	55
CPFF			
FP			

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

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 in progress.

1231078

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CGH-955	Reactivation of the H-1 Loop - 105-H Building	0490

AUTHORIZED FUNDS			COST & COMM TO 7-1-62	$ 2,614
$ 10,000	DESIGN $ 10,000	AEC $	ESTIMATED TOTAL COST	$ 105,000
	CONST. $	GE $ 10,000		

STARTING DATES		DATE AUTHORIZED	3-29-62	EST'D. COMPL. DATES	
DESIGN	4-15-62			DESIGN	8-30-62
CONST.		DIR. COMP. DATE		CONST.	

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		11
TITLE I			11
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide 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.

Design work has been stopped. A project proposal, requesting cancellation of this project, is being prepared.

PROJ. NO.	TITLE	FUNDING
CGH-957	Small Particle Technology Laboratory - 325 Building	62-k

AUTHORIZED FUNDS			COST & COMM. TO 7-15-62	$ 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			DESIGN	5-31-62
CONST.	7-6-62	DIR. COMP. DATE	11-1-62	CONST.	11-1-62

ENGINEER

FEO - DS Jackson

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
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	5*	3
PF	100	5*	3
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.

Construction was started by J. A. Jones Construction Company forces on July 6, 1962.

The floor was cut, drain lines roughed in and the floor patched. Electricians installed service conduits to the electrical panel location and started installation of the emergency power line.

* Project Planning Schedule

SEMI – MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAH-958	Plutonium Fuels Testing and Evaluation Laboratory - 308 Bldg.	62-k

AUTHORIZED FUNDS			COST & COMM. TO 7-15-62	$ -0-
$ 150,000	DESIGN $ 15,500	AEC $ 148,000	ESTIMATED TOTAL COST	$ 150,000
	CONST. $ 134,500	GE $ 2,000		

STARTING DATES				EST'D. COMPL. DATES	
DESIGN	8-1-62	DATE AUTHORIZED	6-22-62	DESIGN	11-1-62
CONST.	11-1-62	DIR. COMP. DATE	5-15-63	CONST.	3-15-63

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.

The project was authorized by HOO-AEC June 22, 1962.

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.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS			COST & COMM. TO	$
$	DESIGN $	AEC $	ESTIMATED TOTAL COST	$
	CONST. $	GE $		

STARTING DATES				EST'D. COMPL. DATES	
DESIGN		DATE AUTHORIZED		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

1231080

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAH-962	Low Level Radiochemistry Building	05-1-63-H-001-23

AUTHORIZED FUNDS	DESIGN $ 113,000	AEC $ 82,000	COST & COMM. TO 7-15-62	$ -0-
$ 113,000	CONST. $	GE $ 31,000	ESTIMATED TOTAL COST	$ 1,200,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN 7-23-62		6-28-62	DESIGN 5-15-63	
CONST. 8-1-63		DIR. COMP. DATE --	CONST. 8-1-64	

ENGINEER

FEO - KA Clark

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	0	-0-
TITLE I			
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.

The project proposal requesting $113,000 total design funds was submitted to the AEC, for authorization on April 30, 1962.

Directive No. AEC-207, dated June 28, 1962 authorizing HOO-AEC total design funds in the amount of $113,000 has been issued.

Work Authority CAH-962(1) dated July 5, 1962 authorized the General Electric Company $31,000 for preparation of the design criteria and minor detailed equipment design.

A design scope planning meeting was held with CE&UO and HLO representatives on 7-25-62.

A schedule for the design criteria progress is being prepared.

1231081

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-74522

DATE 7-31-62

PROJ. NO.	TITLE	FUNDING
CAE-963	Geological & Hydrological Wells - FY-1962	62-k

AUTHORIZED FUNDS			COST & COMM. TO	
$ 80,000	DESIGN $ 1,400	AEC $ 68,500	7-15-62	$ 10,318 (GE)
	CONST. $ 78,600	GE $ 11,500	ESTIMATED TOTAL COST	$ 80,000

STARTING DATES			EST'D. COMPL. DATES
DESIGN 5-18-62	DATE AUTHORIZED 5-9-62		DESIGN 6-1-62
CONST. 7-9-62	DIR. COMP. DATE 4-1-63		CONST. 4-1-63

ENGINEER: FEO - HE Ralph

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	4	72
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	10	9
PF			
CPFF	2	1	1
FP	98	10	9

SCOPE, PURPOSE, STATUS & PROGRESS

This project involves the continued drilling of special hydrological research, test and monitoring wells.

Approximately 400 feet of holehave been drilled to date. Boulders and heavy gravel have slowed drilling speed.

Contractor has 3rd drilling rig on well site 67-86.

Contractor unloaded a carload of well casing at the Batch Plant during the week of July 16, 1962.

PROJ. NO.	TITLE	FUNDING
CAH-977	Facilities for Radioactive Inhalation Studies	62-k

AUTHORIZED FUNDS			COST & COMM. TO	
$	DESIGN $	AEC $		$
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 140,000

STARTING DATES		EST'D. COMPL. DATES
DESIGN 11-15-62*	DATE AUTHORIZED --	DESIGN 1-15-63*
CONST. 3-15-63*	DIR. COMP. DATE --	CONST. 1-15-64*

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.

Approval by AEC Review Board on 7-19-62 was deferred for study of detailed costs and construction

* Based upon AEC approval by August 15, 1962.

1231082

TEST REACTOR AND AUXILIARIES OPERATION

JULY 1962

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

The reactor was shut down the entire month to effect repairs to the damaged flow straightening vanes noted late in June. Complete reactor discharge and draining of the primary system was accomplished without incident. During various stages of the outage, advantage was taken of system conditions to inspect and clean the primary system of all foreign objects, conduct a total power failure test, containment tests, and special process evaluation tests.

The primary system was refilled and the reactor was charged with startup activities underway at month end.

D_2O inventories indicated a net gain of 663 pounds which is a reflection on the June inventory. Net loss for June and July was 3,031 pounds. Helium losses were 28,800 scf.

Equipment Experience

One of the flow straightening vanes which consisted of 10 gage sheets, 2 feet long with unsupported widths of approximately 6 inches was found to have come loose. Failure was attributed to fatigue. Five major pieces of the vane were found and determined to be all of the vane. One short section was still intact in the vane assembly, one large section was lying in the vane assembly, two trianglar (~4" x 6") were located in the lower ring header, and one small (~1" x 2") piece was located in a process tube jumper. The replacement assembly was made up of a bundle of 2" stainless steel tubing, 2 feet long.

Other foreign particles were found in two inlet header angle valves, and another angle valve was repaired. Fourteen additional angle valves were repacked.

Two pressurizer pressure relief valves were overhauled. The spring from one of the valves had a surface crack.

The 14" motor operated discharge valve was completely disassembled for inspection and repair.

About 50% of a planned wiring improvement program in the control room was completed during the month. The improvement program involves installing a better quality wire marker on about 1500 wires (old gum type labels were becoming illegible or falling off), removing about 70 unused wires and rerouting others for easier and quicker trouble-shooting.

The 2400 volt cable to primary pump motor #3 (single speed) shorted during startup activities destroying a section of the cable. The short occurred where the molded rubber plug is vulcanized to the cable insulation. The cable had been in use only a short time. A spare new cable was installed and all three motors and the wiring between the motors and breakers were "megged" to ground as well as phase to phase. No defects were found. No damage occurred to the motor, its connection box or its breaker.

Programmed maintenance required 633 man hours or 12% of the total available man hours.

Improvement Work Status (significant items)

Work Completed:

Modification of primary pump discharge check valves*
Modification of the pump by-pass check valve*
Installation of improved primary system drain provisions*
Removal of abandoned primary system injection check valve and installation of 4" header for future use*
Shim rod readout modification
Rupture monitor sample line changes
Modification to storage basin crane

*These items were performed to take advantage of the drained primary system condition

Work Partially Completed:

Safety circuit ground and low voltage detector
Outlet nozzle cap modification (now 90%)
Fueling vehicle hoist modification
Primary oxygen analyzer installation
Core blanket system piping modifications
Flanges for safety relief valves in helium system - 85%
Chain barricade for rotating shield
Fuel transfer pit hoist drain

Design Work Completed:

Enlarge chemical feed system
Decontamination facility
Primary pump recording ammeters
High pressure helium compressor inter-after cooler relief
Outlet nozzle bracing
Modification to helium valve stems
Interlock between charge - discharge machine, shroud seat and discharge hoist
Control room ventilation scope
Third exhaust air activity channel
Fuel transfer system modifications

Design Work Partially Complete:

Additional fuel storage and examination
Oil storage building
Boiler feed pump seals
Compressed air supply revisions

Process Engineering and Reactor Physics

After reactor discharge special flow tests were conducted to determine the flow required to backseat the primary pump discharge check valves and the primary pump bypass check valve to ascertain that light water injection flows would provide adequate cooling under certain postulated emergency conditions. It was found that over 700 gpm was required and that this was not satisfactory. The valves were modified during the month such that the bypass valve would be positively closed by an external actuator, and the other valves would be more nearly closed under no-flow conditions. Subsequent tests resulted in 30 - 50 gpm flows closing the check valves.

A total power failure test was conducted. Emergency lighting, D.C. power and compressed air were found to be satisfactory. Instrumentation was also observed to function as needed. Air supply was shown to be adequate for approximately two hours.

Selected pipes, nozzles and orifices were subjected to nondestructive and destructive examination to determine the condition of the primary coolant system. All pieces that were examined showed no defects and were within specifications. Radioactive decontamination of these pieces was difficult.

Twelve UO_2, eleven Pu-Al and three moxtyl fuel elements were subjected to visual examination and go:no-go gauge inspection of the end brackets in the PRTR basin. The data are to be related to the previously reported process tube fretting corrosion damage.

Procedures

Revised Operating Procedures issued			13
Revised Operating Standards issued			1
Temporary Deviations to Operating Standards issued			3
Revised Process Specifications accepted for use			2
Maintenance Manuals issued			2
Maintenance Procedures issued			0
Drawing As-built status	June	Total	
Approved for as-built	83	739	
Ready for approval		15	
In drafting		15	
Voided		51	
No change required		81	
		901 (corrected)	
Scheduled for review		.290	
		1 191 (corrected)	

Personnel Training

Qualification subjects	370 manhours
Specifications, Standards, Procedures	35
Fueling Vehicle	10
	415 manhours

Status of Qualified Personnel at Month End

Qualified Reactor Engineers	10
Provisionally Qualified Reactor Engineers	1
Qualified Technicians	6
Qualified Technologists	19
Provisionally Qualified Technologists	2

Plutonium Recycle Critical Facility

Design Test Status:

Electrical	100%
Instrumentation	96%
Fuel Handling	35%
Moderator System	100%
Thimble Coolant	75%

Each safety rod was cycled 50 times and inspected. All performed satisfactorily. Testing and development work on the irradiated fuel transfer thimble system was conducted. Improvements were made to the flux monitoring circuitry to eliminate difficulties developed in testing. Cell cover blocks were installed and total confinement features of the PRCF were observed in conjunction with the PRTR total containment test. Satisfactory performance was observed. Initial drafts of startup physics tests were issued. Sixty-eight PRCFO and PRTRO manhours were devoted to training.

Fuel Element Rupture Test Facility

Project Status (Project CAH-862)

The revised project proposal extending the completion date was approved and revised schedules are being prepared. The project is 98% complete. The filter plant is 99% complete. Acceptance tests for the filter plant were started in July. The electrical design tests for the loop were 80% complete and instrument testing and calibration were 50% complete. Revised designs were issued for approval to provide needed refinements and equipment orders were placed. Initial testing of the discharge equipment was started.

Visual aids for classroom training were completed. A total of 55 manhours was spent on training.

GAS COOLED POWER REACTOR PROGRAM

Project Status (Project CAH-822)

The revised project proposal, requesting extension of the completion date, was approved and revised schedules are being prepared. The replacement heater fabricator indicated early August shipment. The first blower was re-built and run successfully to 12,500 rpm outside of its case. Final testing was scheduled for August 3. The second unit was scheduled one week later. Construction forces worked on punch list items. Modified inlet jumpers were installed on PRTR to facilitate the upcoming test section installation.

All but two operating procedures have been completed. Fifteen manhours were spent on training.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 21,572 hours. This includes 16,104 hours performed in the Technical Shops, 3,975 hours assigned to Minor Construction, 1,303 hours assigned to off-site vendors, and 190 hours to other project shops. Total shop backlog is 21,929 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 4.8% (1,137) of the total available hours.

Distribution of time was as follows:

	Manhours	% of Total
Fuels Preparation Department	4,063	18.83%
Irradiation Processing Department	2,886	13.38%
Chemical Processing Department	644	2.99%
Hanford Laboratories Operation	13,971	64.76%
Construction Engineering and Utilities	8	.04%

Requests for emergency service declined from the previous month, reaching a level which is considered normal for this operation.

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
M. Lewis	Proposed Use of Magnets for a Magnetic Ski Binding. July 20, 1962
D. C. Brandt	Continuous Apparatus to Produce Aluminum Sulfate by the Process Described in HWIR-990
R. H. Moore	A Pyrochemical Method for Separation of Uranium and Thorium Using Immiscible Molten Salts (HW-74425)

H M Parker
Manager, Hanford Laboratories

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